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The institute was formed in 1956 on the initiative of Academician I.V. Kurchatov for carrying out engineering and scientific research in the field of nuclear power.

In 2008 the institute was transformed into Open Joint-Stock Company "State Scientific Center – Research Institute of Atomic Reactors" (RIAR).

At present time, RIAR is Russia's largest experimental research center in civilian nuclear power.

*RIAR office building*

**V.M. TROYANOV,**
*Director of RIAR*

**M.N. SVYATKIN,**
*First Deputy Director and Chief Engineer of RIAR*

The Institute includes 6 operating research reactors (SM-3, RBT-6, MIR.M1, RBT-10, BOR-60, VK-50), 2 critical facilities (SM, MIR), Europe's largest facility for post-irradiation examination of production reactor core components, a nuclear fuel cycle research facility, a radiochemical facility and a radioactive waste handling facility.

RIAR also includes the "Irradiation – Materials Technology – Research" multiuser center intended to give scientific, guidance and instrumentation support to research and technological activities, providing outside organizations with access to advanced knowledge-intensive technologies in the field of radiation materials technology, and to search for new and modify the existing materials for general industrial and nuclear power needs.

RIAR's unique multidiscipline experimental framework offers research and production capabilities in activities on the most topical scientific issues of nuclear power science, including:

– development and demonstration of innovative nuclear technologies in pilot production;

– rendering of knowledge-intensive engineering services;

– transfer of nuclear technologies to other industries, including nuclear medicine and manufacturing industries, and for solving environmental problems.

The Institute offers services in irradiation and post-irradiation examination of nuclear technology materials and items, innovative nuclear fuel manufacturing and processing technologies, and radioactive waste disposal.

RIAR develops and produces a variety of radionuclides and ionizing radiation sources for scientific, industrial and medical applications.

The Institute includes its own personnel skills improvement framework and collaborates actively with regional higher education establishments in training staff both for the Institute and other organizations in the given region.

The Institute carries out conservational activities and research for the purpose of studying the conditions for safe isolation of low-level waste in deep-earth geological formations and above-surface storage of spent nuclear fuel.

The Institute's production facilities include in-house power generation systems for heat, hot water and cold water production, auxiliary facilities for manufacturing and repair of equipment, and a vehicle fleet for transportation services, including in the field of nuclear material and special-purpose cargo carriage.

RIAR's site possesses a high potential of evolution.

Thus, it is intended to house a new research reactor MBIR in the near future.

## RIAR's research reactors and critical facilities

| Type | Designation | Thermal power, kW | First criticality year | Status | Operating time, years |
|---|---|---|---|---|---|
| RR | SM-3 | 100 000.00 | 1961 | In operation | 51 |
| RR | RBT-6 | 6 000.00 | 1975 | In operation | 37 |
| RR | MIR.M1 | 100 000.00 | 1966 | In operation | 46 |
| RR | RBT-10/2 | 10 000.00 | 1983 | In operation | 29 |
| RR | BOR-60 | 60 000.00 | 1969 | In operation | 43 |
| RR | VK-50 | 200 000.00 | 1964 | In operation | 48 |
| RR | MBIR | 150 000.00 | 2019 | Planned | – |
| CF | SM | 0.02 | 1970 | In operation | 42 |
| CF | MIR | 0.005 | 1966 | In operation | 46 |

** As of 2012*

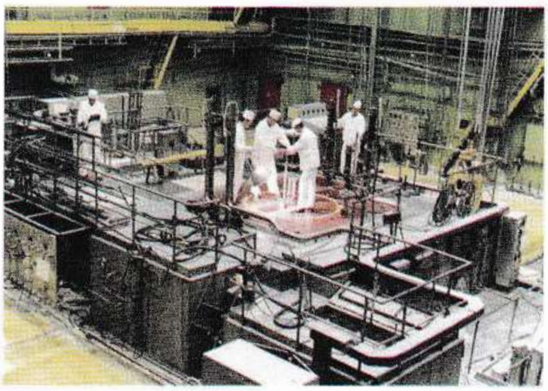
# SM-3 PRESSURIZED WATER-COOLED WATER-MODERATED RESEARCH REACTOR

The SM research reactor is a NIKIET-developed vessel-type high-flux nuclear reactor with an intermediate neutron spectrum and a pressurized water cooled core. The reactor achieved first criticality on 23.05.1961 and its energy startup took place on 15.10.1961.

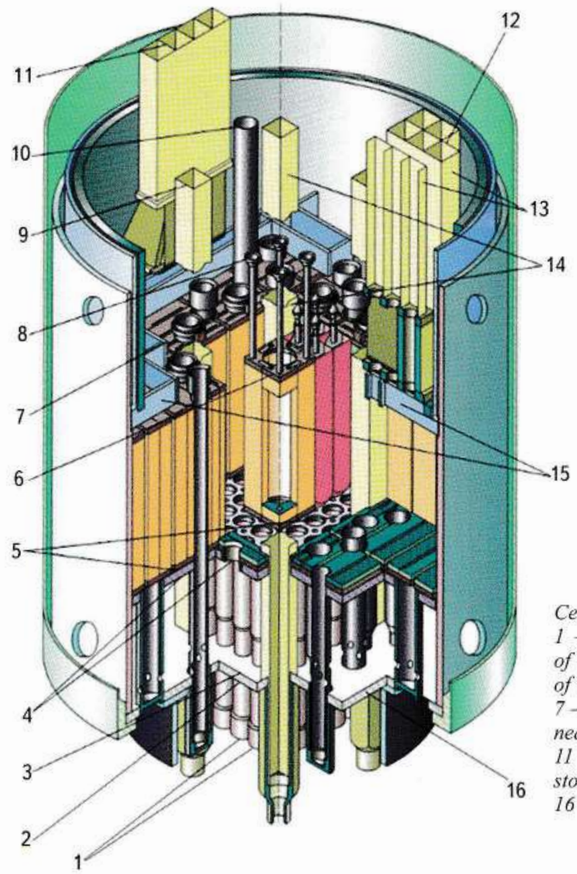

*Main building of the SM-3 reactor*

The SM reactor design was the first one to embody the concept of a high thermal neutron flux with a hard spectrum in a moderator trap in the core center.

The reactor is intended for experimental research in irradiation of reactor material samples in the given conditions, investigation of regularities in radiation-induced changes in the properties of materials, and generation of transplutonium elements and radioactive nuclides of lighter elements.


*Central hall of the SM-3 reactor*

Over the years of its operation, the reactor was retrofitted more than once to expand the experimentation capabilities and improve the safety of operation. The core, the reflector and the reactor main process systems were redesigned extensively, including, specifically, the introduction of X-shaped fuel elements with




an increased active part height, the adoption of beryllium metal as the reflector material and absorbers based on europium dioxide, and an increase in the reactor power.

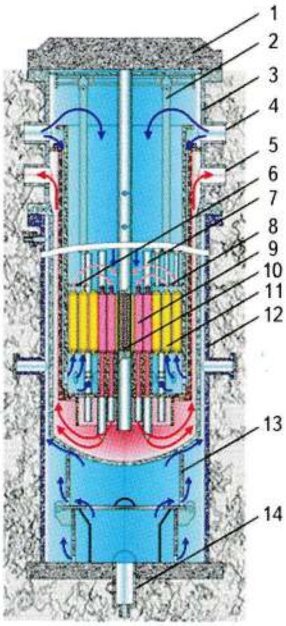
Following the retrofit of 1991–1992 the reactor was reindexed SM-3. It reached first criticality in December 1992, and saw its energy startup in April 1993. The retrofit gave the reactor the adequacy in terms of modern safety requirements and greater experimental capabilities.

Five FA types can be used as working FAs in the SM-3 reactor core: two FA types (fuel elements with a load of 5 g of $^{235}U$) that contain 0.940 and 0.8 kg of $^{235}U$, and three FA types (fuel elements with a load of 6 g of $^{235}U$) that contain 1.128, 0.960 and 0.948 kg of $^{235}U$.

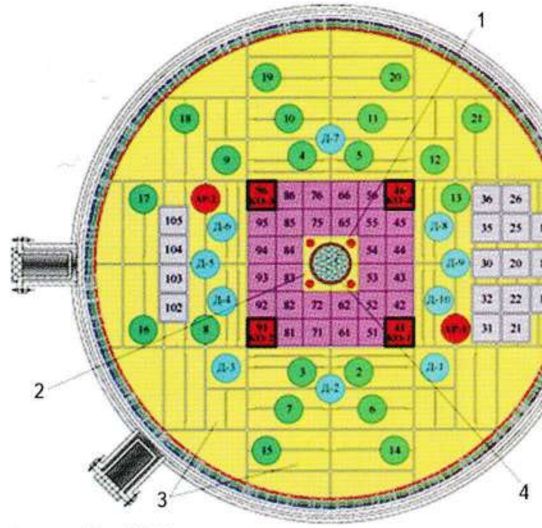
*Central portion of the SM-3 reactor:*
*1 – bottom of diffusers; 2 – bottom of lower support plate; 3 – top of lower support plate; 4 – bottom of upper support plate; 5 – top of upper support plate; 6 – top of Be-insert in CBTT with EP-1; 7 – top of reflector pressure grid near SM-2; 8 – top of EP-2 guide tube near Be-insert; 9 – top of SM-2 guide tube; 10 – top of AC-2 guide tube; 11 – top of storage facility 2; 12 – top of storage facility 1; 13 – FA storage facility 1; 14 – guide tube of SM-3; 15 – reflector pressure grid; 16 – lower support plate*

## Main performance of the SM-3 reactor

| | |
|---|---|
| Power | 100 MW |
| Undisturbed thermal flux in central trap, max. | $5\cdot10^{15}$ см$^{-2}\cdot$с$^{-1}$ |
| Moderator | Water |
| Coolant | Water |
| Reflector | Beryllium |
| Fuel | Uranium dioxide, enrichment 90% |
| Core configuration in plan | Square, with a central trap |
| Outer core dimensions | 420×420 mm |
| FA grid spacing | 70×70 mm |
| Number of FA cells | 32 (including 4 cells for shim rods with fuel suspensions) |
| Number of cells occupied by central trap | 4 |
| Core height | 350 mm |
| Core geometrical volume: | 61.7 l |
| including trap volume | 6.8 l |
| Power generating volume | 48…54.9 l |
| Fuel element | X-shaped, SM-type |
| FA fuel grid | Triangular, spacing 5.23 mm |
| Thermal load, average in power generating volume | 1.82…2.08 MW/l |




*Sectional view of the SM-3 reactor:*
*1 – lid;*
*2 – refueling mechanism;*
*3 – SM-3 vessel;*
*4 – inlet nozzle;*
*5 – outlet nozzle;*
*6 – smaller unloading platform;*
*7 – shim rod tube;*
*8 – larger unloading platform;*
*9 – FA;*
*10 – central high-flux cavity;*
*11 – beryllium reflector;*
*12 – SM-2 vessel;*
*13 – SM-2 heat shielding;*
*14 – annulus cooling system inlet nozzle*




*A map of the SM-3 reactor core:*
*1 – central beryllium block; 2 – beryllium inserts; 3 – reflector beryllium blocks; 4 – central shim member*

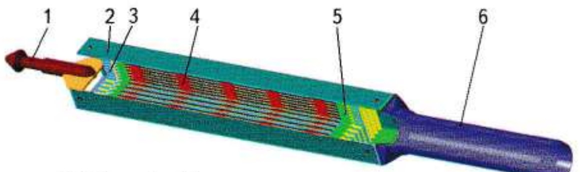
5 Д-1 *Channel and its number*

*Shim member*

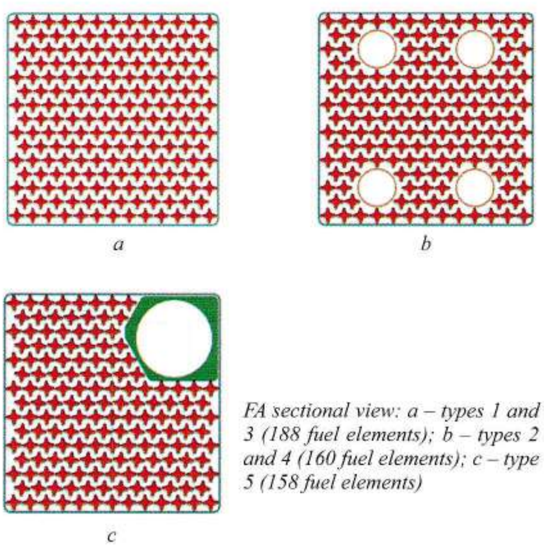
*AC working member*

61 *Core cell with FA*

*EP working member with Be reflector*




*SM-3 reactor FA:*
*1 – head; 2 – jacket; 3 – spacer grid; 4 – fuel elements; 5 – support grid; 6 – tail*

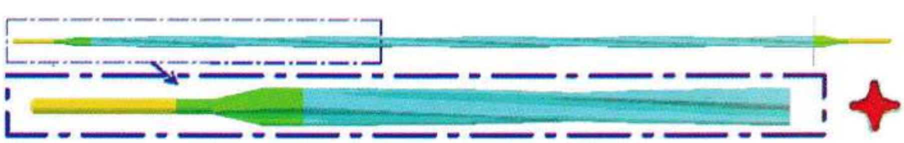



*FA sectional view: a – types 1 and 3 (188 fuel elements); b – types 2 and 4 (160 fuel elements); c – type 5 (158 fuel elements)*

All FA types have standard overall and connecting dimensions.

A FA is an integral structure consisting of a square-shaped shroud of stainless steel (for FAs of types 1 and 2) or zirconium (for FAs of types 3, 4 and 5) with the outer dimensions of 69×69 mm, a cylindrical tail and an upper rack having a head for transportation. The FA overall length is 910 mm. The fuel elements are retained at their tail portions using plate-type spacer grids.

The fuel elements have an X-shaped cross-section. The fuel cladding thickness is 0.15 mm, and the fuel composition is uranium dioxide dispersed in a matrix of a copper and beryllium bronze mixture. The fuel element active height is 350 mm.



*SM-3 reactor fuel element*

## Experimental capabilities of the SM-3 reactor

The SM-3 reactor is fitted with a broad range of experimental devices, which can be accommodated in the central trap, in the reflector cells and in dedicated FAs. The total number of irradiation positions is up to 37, of which:

– one is in the trap (a block type: the central block has 27 cells for targets; a channel type: the central channel has 18 cells for targets);

– up to 6 (FAs of type 2 or 4, each with four Ø12.5 mm cells for targets) or up to 4 (FAs of type 5, each with one target cells) are in the reactor core;

– 30 are in the reflector (including 20 locations for the installation of channels with information channels or coolant channels led out through the reactor lid).

## Design data of the SM-3 reactor fuel element

Fuel meat length.......... 346±4 mm

Fuel element width across corners.......... 5.15 mm

Cladding thickness.......... 0.15 mm

Rib winding spacing.......... 300 mm

Fuel composition:

$UO_2$.......... 5.6 g

PMS-A copper.......... 13.7 g

BrB2 bronze.......... 2.0 g

Fuel composition density.......... 9.24 g/cm$^3$

Enrichment in $^{235}U$.......... Up to 90 %

Weight of $^{235}U$.......... 5.0±0,1 g
6.0±0,1 g

Fuel element weight.......... 32 g

Factor of fuel distribution irregularity along meat length.......... Not more than 1.1

## Neutron flux in irradiation devices, $m^{-2} \cdot s^{-1}$

| Device | Neutron energy | | |
|---|---|---|---|
| | ≤0.67 eV | 0.67…100 eV | ≥0.1 MeV |
| Separator structure | $1.9 \cdot 10^{19}$ | $1.1 \cdot 10^{18}$ | $1.0 \cdot 10^{19}$ |
| Central block of transuranium targets (CBTT) | $1.3 \cdot 10^{19}$ | $1.7 \cdot 10^{18}$ | $1.4 \cdot 10^{19}$ |
| FAs of types 1, 4 and 5 | $1.3 \cdot 10^{18}$ | $9.3 \cdot 10^{17}$ | $2.0 \cdot 10^{19}$ |
| Reflector cells | $1.35 \cdot 10^{19} \ldots 9 \cdot 10^{17}$ | - | $3.3 \cdot 10^{18} \ldots 2.5 \cdot 10^{16}$ |

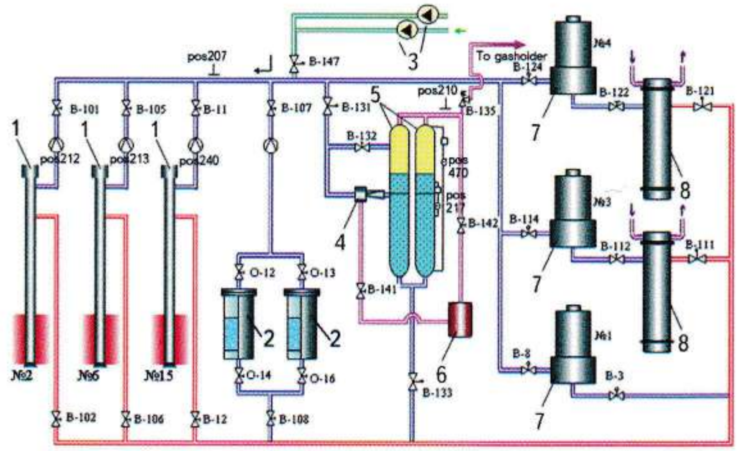
The SM-3 reactor includes the operating VP-1 low-temperature loop facility (VP-1 LF) (put into operation in 1963 and retrofitted in 2003) and VP-3 high-temperature loop facility (VP-3 LF) (put into operation in 1977 and retrofitted in 1994).

The VP-1 LF is intended for tests of fuel element prototype models, irradiation of structural and absorber material samples, and production of isotopes.

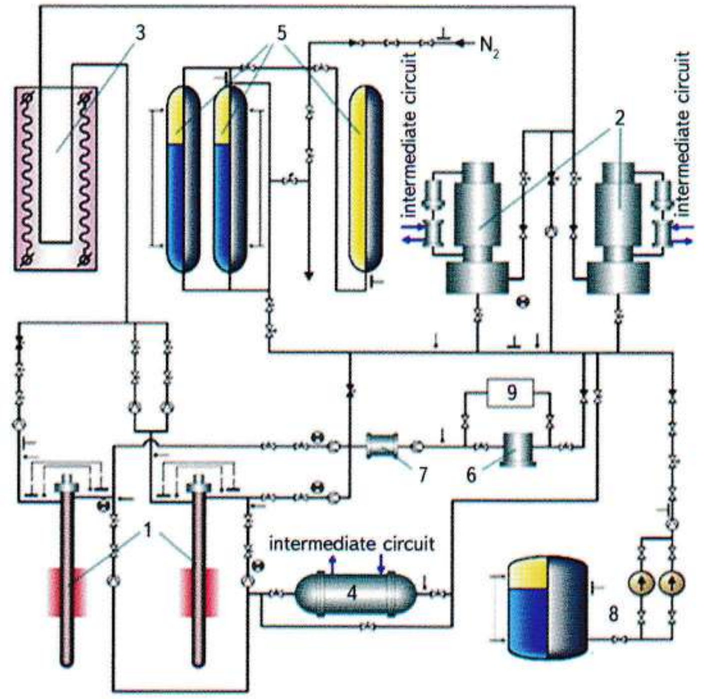
The VP-3 LF is used to investigate the functionality of fuel elements for reactors of different types, study the release of fission products from failed fuel elements and the methods to remove these from the primary circuit, and examine structural and absorber materials.

## Main performance of the VP-1 and VP-3 loop facilities

| Characteristic | VP-1 LF | VP-3 LF |
|---|---|---|
| Coolant | Distilled water | Distilled water |
| Maximum working pressure, MPa | 4.9 | 18.5 |
| Coolant temperature, °C | 90 | 300 |
| Coolant flow rate, m$^3$/h | 30 | 5…8 |
| Power of one channel, kW | 500 | 50 |
| Number of irradiation channels | 3 | 3 |




*Flow sheet of the VP-1 loop facility: 1 – loop channel; 2 – ion-exchange filter; 3 – makeup pumps; 4 – ejector; 5 – degasifier; 6 – contact apparatus; 7 – circulation pump; 8 – heat exchanger*




*Flow sheet of the VP-3 loop facility: 1 – loop channels; 2 – circulation pumps; 3 – electric furnace; 4 – heat exchanger, 5 – pressurizer system; 6 – low-temperature filter, 7 – chiller, 8 – makeup system, 9 – coolant chemistry control system*

## Main areas of studies

***Material science*** – studies into the mechanisms of radiation damage to existing and future fissionable, structural and absorber materials.

***Isotope program***

The unique capabilities of the SM-3 reactor have made it Europe's lead facility in production of transuranium elements and accumulation of radionuclides of a high specific activity.

The following features contribute to this:

– there is a large number of irradiation positions in the reflector with a thermal neutron flux range from $10^{14}$ to $1.5 \cdot 10^{15}$ cm$^{-2}\cdot$s$^{-1}$;

– targets can be irradiated inside the core fuel bulk, where the share of epithermal and fast neutrons is great;

– there is a neutron trap with an undisturbed thermal neutron flux of $5 \cdot 10^{15}$ cm$^{-2}\cdot$s$^{-1}$ in the core center.

Multistage generation of heavy curium and californium isotopes has been achieved in the neutron trap cells and in the two reflector cells that are the nearest to the core. Besides, radionuclides of a high specific activity are accumulated in a broad range (nickel-63; tin-113, -119m; tungsten-188; iron-55, -59; chromium-51 and others). Also blanks of irradiation sources based on selenium-75 and iridium-192 are activated.

The experimental channels in the core are used to irradiate phosphorus-32 and -33, gadolinium-153, tin-117m and other radionuclides in a hard neutron spectrum.

Large-scale accumulation of radionuclides, such as cobalt-60 and iridium-192, takes place in the reflector cells that are the nearest to the core. The annual output is 200…300 kCi of cobalt-60 and 400 kCi of iridium-192.

Precision activation of medical irradiation source blanks based on cobalt-60 is conducted in a special instrumented device which ensures continuous monitoring of the neutron flux during irradiation. Annually, some 1000 sources are produced.

## International cooperation

Research is conducted on the SM-3 research reactor in collaboration with the IAEA, China, the USA, the EU, the CIS and other countries.

## Main activities

The reactor utilization factor was 0.70 in 2011.

Service life tests of the SM-3 fuel elements with a low harmful absorption of neutrons have been completed.

Thermal and radiation heat tests of electrical engineering materials have been completed.

In-pile tests of the ITER diverter material samples at temperature up to 200 °C have been completed.

A great deal of research in the field of radiation material studies was performed at the SM-3 reactor in 2011, namely:

– pilot in-pile tests of zirconium samples were prepared and conducted at temperatures of 350…380 °C in a cell in the nearest SM-3 reactor reflector row as part of long-term testing up to the damaging dose of 25…30 dpa;

– in-pile tests of a set of beryllium samples were completed to study the properties of the irradiated material;

– in-pile tests were prepared and initiated as part of the program to build a propulsion module based on a megawatt-class nuclear propulsion system:

- tests of fuel element mockups with a carbide-nitride fuel;
- tests of fuel refractory cladding samples in a channel installed inside the SM-3 core;

– in-pile tests of samples have been prepared and begun under the GT-MHR program as part of the G1-2 irradiation device with matrix and block graphite samples, and with coated particle fuel and fuel compact simulators at temperatures from 800 to 1250 °C;

– in-pile tests of SAV-1 alloy rods are conducted as part of the activities to extend the life of the VVR-M at St. Petersburg Institute of Nuclear Physics;

– in-pile tests of a set of simulators clad in a chromium-nickel alloy are conducted for the material creep investigation.

Production of radionuclides with a high specific activity of both transuranium elements ($^{244,248}Cm$, $^{242}Pu$, $^{243}Am$, $^{248}Cf$, $^{249}Cf$, $^{252}Cf$) and "light" elements ($^{33}P$, $^{153}Gd$, $^{192}Ir$, $^{60}Co$, $^{188}W$, $^{63}Ni$, $^{55}Fe$, $^{59}Fe$, $^{113}Sn$, $^{119m}Sn$, $^{89}Sr$ and others).

Pilot batches of the radionuclides $^{125}I$ and $^{177}Lu$ have been produced.

In the nearest time it is planned to:

– install two loop channels of the diameter up to 68 mm in the fuel part of the core;

– install ampoule-type channels of the diameter 24.5 mm in dedicated FAs, while preserving the existing 24 ampoule-type channels of the diameter 12 mm;

– development of fuel elements with a low harmful absorption.

## History

The reactor was retrofitted more than once during its operating time.

***Minor retrofit of 1964***

*Major activities:*

– plate-type fuel elements of the fuel assemblies were replaced for X-shaped ones with an increase in the fuel active length from 250 to 350 mm;

– the CPS shim rods had the beryllium displacers replaced for additional fuel batches, and an extra shim rod was installed into the vertical experimental channel.

*Results:*

– a higher fuel burn-up was achieved and the time of operation between refuelings was extended to 15 days;

– it became possible to adapt some of the core cells to irradiation of start materials for the purpose of accumulating transuranium elements.

***Retrofit of 1965 (including central zone replacement)***

*Major activities:*

- beryllium metal was adopted for the reflector;
- a fuel cladding failure detection system was put into operation with individual sampling from FA in the core;
- the primary circuit pumps were replaced for more advanced and reliable ones;
- two water loops were united into one low-temperature loop with its full isolation from the reactor primary circuit;
- a pump was installed to ensure reliable core cooldown during refueling operations inside the reactor with a depressurization.

*Results:*

- the reactor was given safety and reliability operation at 75 MW, with the neutron flux increased to $3.3 \cdot 10^{15}$ $cm^{-2} \cdot s^{-1}$.

***Retrofit of 1974***

*Major activities:*

- the heat exchangers were replaced for new ones (of stainless steel);
- the emergency core cooling system was upgraded.

*Results:*

- the reactor power was increased to 100 MW, with the neutron flux having reached $5 \cdot 10^{15}$ $cm^{-2} \cdot s^{-1}$.

***Retrofit of 1977–1978***

*Major activities:*

- a new central zone was installed;
- hydraulic shaping of the core cells with the aid of throttling inserts was introduced;
- the electrical components of the main circulation pumps were upgraded.

*Results:*

- the coolant flow rate was increased from 2000 to 2400 $m^3/h$;
- the dryout margin was increased to 1.7.

***Major retrofit of 1991–1992***

*Major activities:*

- a new reactor vessel was manufactured and installed, with the old vessel converted to the function of a safeguard vessel that withstands the primary circuit pressure;
- the primary circuit piping was rerouted with the coolant delivery and discharge arranged in the vessel upper portion;
- the core was reconfigured;
- the emergency core cooling system was upgraded using active and passive features;
- a computer-based measurement and computation system for all process parameters was set up;
- an additional power source was introduced (a diesel power station).

*Results:*

- the facility safety was increased;
- modern safety requirements were fulfilled;
- experimental capabilities were expanded.

There was a great deal of experience in improving different systems and components and increasing the operating safety gained during the reactor operation.

Work is continued to give the reactor greater experimental capabilities. A concept was developed and is implemented to upgrade the reactor core, which is aimed primarily at enabling long-term irradiations of NPP material samples, including in large-diameter instrumented devices, in a hard neutron spectrum, the damaging speed being up to 15 dpa per year.

Orderly and task-oriented work to improve operations and extend the service life of the reactor facility will make it possible to use the SM-3 reactor for research applications for at least 25 years more.

## Personalities

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# RBT-6 POOL-TYPE REACTOR

The RBT-6 research reactor is a satellite of the SM-3 high-flux reactor and uses the latter's spent fuel. This is a pool-type water-cooled water-moderated thermal-neutron reactor. The RBT-6 reactor achieved first criticality on 24.09.1975 and its energy startup took place on 29.12.1976.

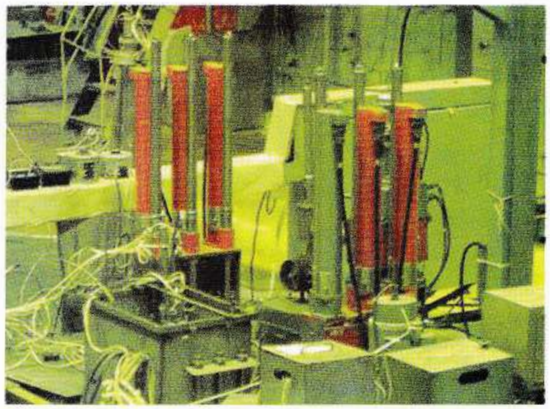
The RBT-6 reactor is intended for experiments on studying the properties of materials during long-term irradiation with invariable parameters and irradiation modes, as well as for production of radionuclides.

The detailed design of the RBT-6 research reactor is specific in that it was constructed in the building, which accommodated the effective SM-3 reactor facility.

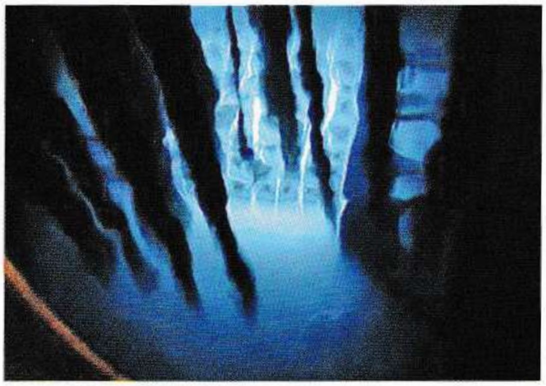
The RBT-6 and SM-3 reactor facilities have the same central hall and share many reactor systems.

The RBT-6 reactor has not been given major upgrades, still work is undertaken continuously to give it increased reliability and improve some of its parts and components for the purpose of raising the reactor's operating safety.

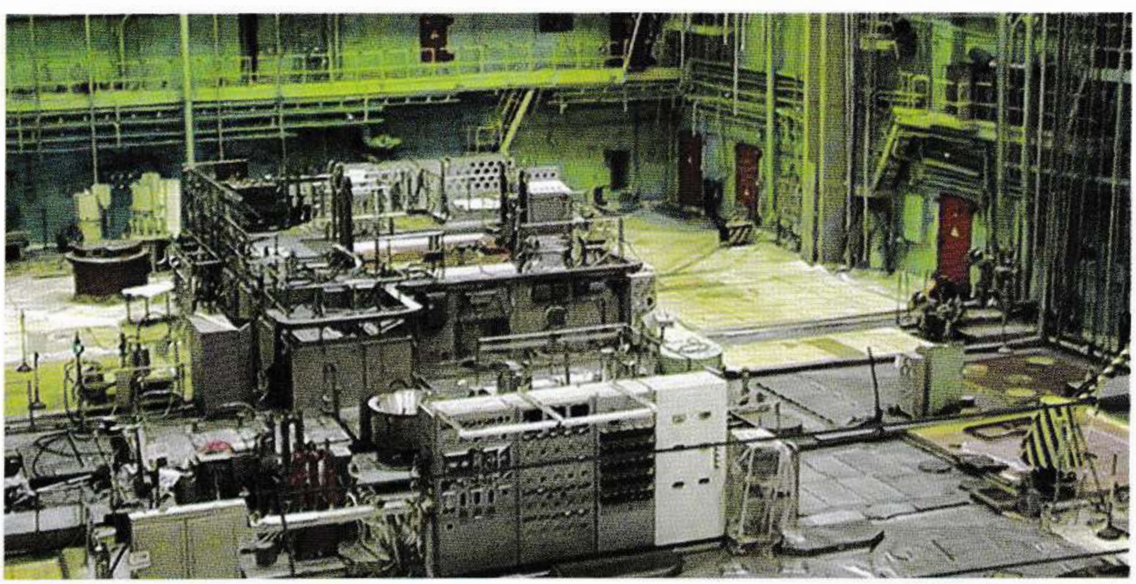
The reactor pool consists of two identical tanks interconnected via a connecting pipe of Ø1000 mm.



*Motor drives of the CPS members*



*A view of the RBT-6 reactor core*



*SM-3 and RBT-6 RR central hall*

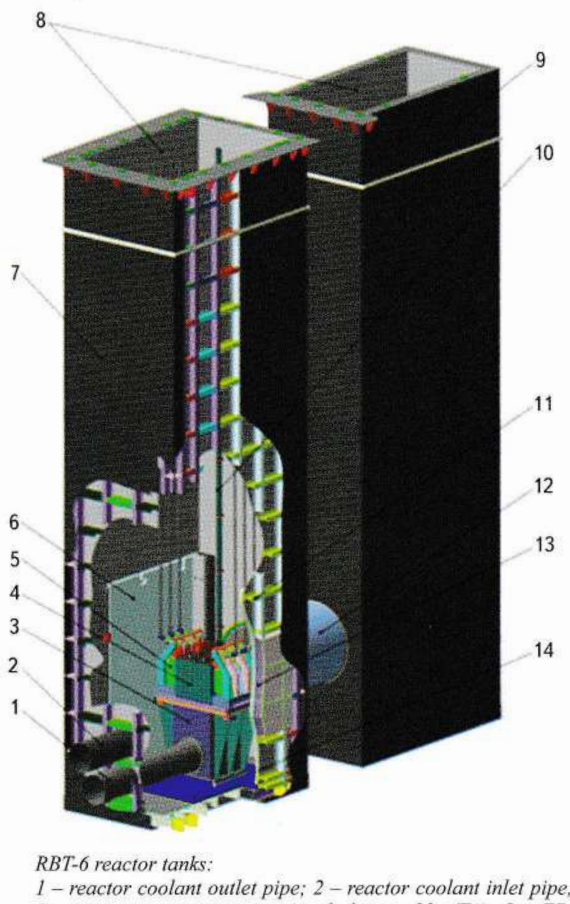
The reactor core is accommodated in tank 1, which is a rectangular double-wall vessel filled with distilled water. The internal wall of the tank is of stainless steel and the external wall is of carbon steel.

The core is made up of FAs installed in a support grid, which represents a welded structure based on two horizontal plates, in which there are 64 holes forming a square lattice with meshes of (8×8) spaced at 78 mm.

The gaps between the FAs are used to insert the EP-SM controls. The core consists of 56 fuel assemblies, 6 shim members (SM) and emergency protection (EP) members combined therewith, one automatic power control (AC) rod and 8 experimental channels with displacers.

The core is made up of spent SM-3 FAs burnt up to 10…30 % (≤ 47 %), and fresh FAs may be also used.

The enrichment in $^{235}U$ is about ~90 %, and the weight of $^{235}U$ in the fuel element is 5.0±0.1 g.

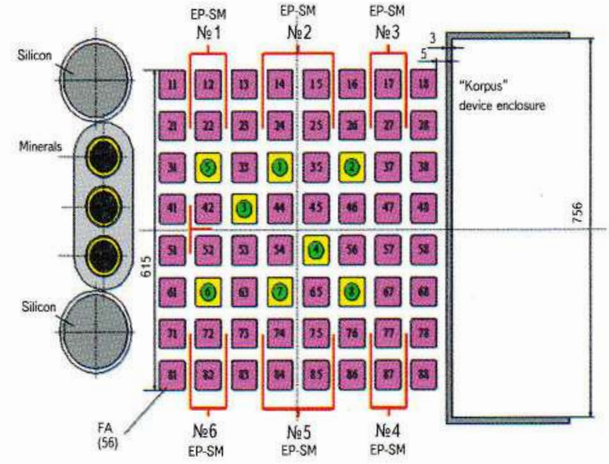



*RBT-6 reactor tanks:*
*1 – reactor coolant outlet pipe; 2 – reactor coolant inlet pipe; 3 – core support structure; 4 – fuel assembly (FA); 5 – EP-SM; 6 – main side shielding; 7 – reactor tank 1; 8 – tank inner surface; 9 – reactor tank 2; 10 – overflow tube; 11 – I-beam; 12 – connection; 13 – extra side shielding; 14 – lower heat shielding*

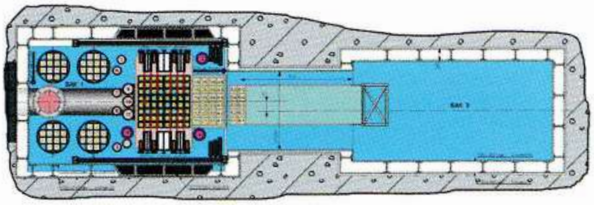
## Main performance of the RBT-6 reactor

| | |
|---|---|
| Design thermal power, max. | 6 MW |
| Core shape and dimensions | A rectangular parallelepiped with a square foundation of 620×620 mm, height 350 mm |
| Core volume | 132 l |
| Number of FA cells | 56 |
| Nuclear fuel* | $UO_2$, dispersed in a matrix of copper and beryllium bronze |
| Enrichment in $^{235}U$ | Up to 90 % |
| Coolant | Water |
| Moderator | Water |
| Reflector | Water |
| Number of CPS members: | |
| emergency protection and shim (EP-SM) | 6 |
| automatic control (AC) | 1 |
| Absorber | Cadmium |
| Primary coolant flow rate | 600 $m^3/h$ |
| Coolant speed in the core | 0.9 m/s |
| Reactor inlet temperature | ≤ 60 °C |
| Reactor outlet temperature | ≤ 70 °C |
| Core power peaking factor, not more than: | |
| axial | 1.17 |
| radial | 1.94 |
| across FA | 1.75 |
| bulk | 4.00 |

** Fresh and SM-3 spent fuel assemblies are used*



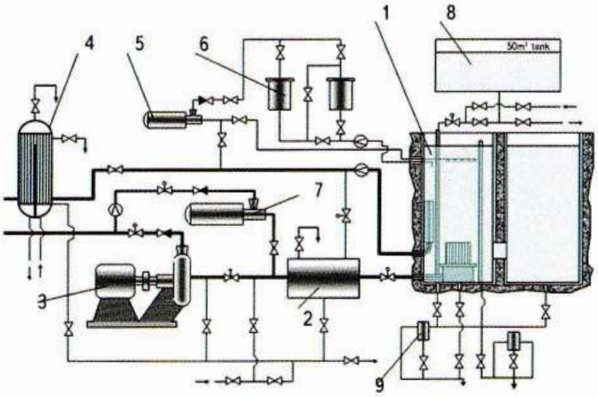

*RBT-6 core map*



*Sectional view of the RBT-6 reactor*

*RBT-6 control room*

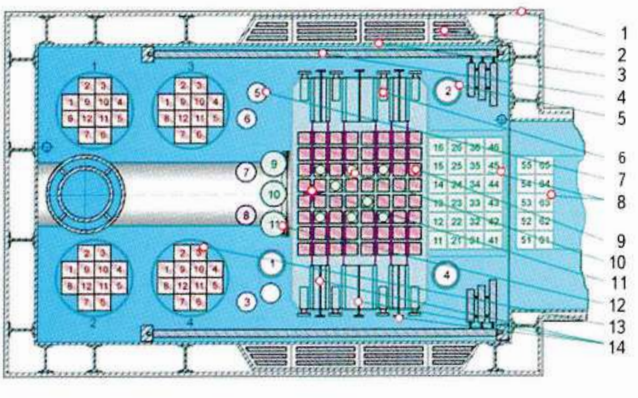
There are FAs of two types used in the RBT-6 reactor. The assembly of type 1 comprises 188 fuel elements. The FA design of type 2 enables in-core irradiation of test targets, which are placed in 4 dedicated channels inside tubes of the diameter 12.5×0.3 mm installed instead of 28 withdrawn fuel elements. Therefore, this FA type comprises 160 fuel elements.



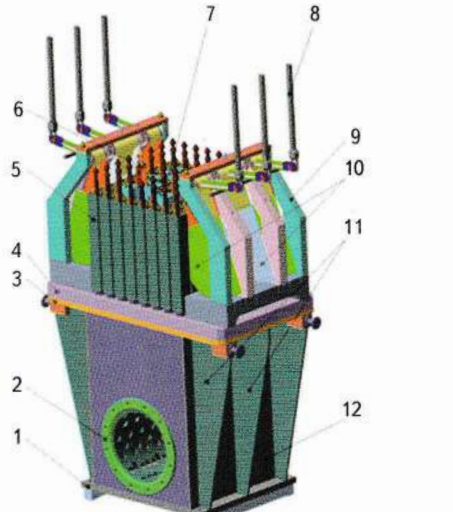

*Flow sheet of the RBT-6 primary circuit:*
*1 – reactor pool; 2 – oxygen activity damper; 3 – main pump; 4 – heat exchanger; 5 – decontamination treatment pump; 6 – decontamination treatment columns; 7 – emergency pump; 8 – makeup tank; 9 – alarm tank*

## Experimental capabilities of the RBT-6 reactor

There are eight vertical channels in the core neutron traps. A change in the composition of the fluid inside the channels (gas, water) or in the gaps between them and the fuel assemblies (installation of displacers) makes it possible to change the neutron spectrum hardness depending on the experiment. The displacers may be made of any material (aluminum, lead, beryllium) and of any size. A channel may be installed in the core also without a displacer as well.




*A map of the RBT-6 reactor facility tank 1:*
*1 – tank outer surface; 2 – extra side shielding; 3 – tank inner surface; 4 – main side shielding; 5 – ionization chamber safety screen; 6 – EP-SM rack; 7 – unscreened ionization chamber; 8 – "Korpus" irradiation device cells; 9 – fuel assembly; 10 – (in-core) irradiation channel; 11 – AC member; 12 – (peripheral) irradiation channel; 13 – spent FA cartridge; 14 – EP-SM sections*




*RBT-6 core support structure:*
*1 – lower plate; 2 – flange; 3 – vessel plate; 4 – removable plate; 5 – working FA; 6 – AC rod; 7 – displacer; 8 – EP-SM linkage; 9 – EP-SM rack; 10 – EP-SM sections; 11 – rigidity rib; 12 – vessel wall*

Three vertical channels of the diameter 158 mm may be installed in the reactor reflector for silicon doping.

An irradiation device was developed in 2007 for radiation coloring of minerals. This may be placed in the reactor reflector instead of one of the three large-diameter channels. The arrangement of the two remaining large-diameter silicon doping channels is left unchanged.

The "Korpus" irradiation device is intended to test vessel steels for the VVER and PWR reactors in environments that simulate, in a broad range, their operating conditions in terms of neutron flux and energy spectrum, irradiation temperature, gradients of these parameters and parameter behaviors during operation.

The device consists of two parts, namely: a fixed part contained in the tank next to the core and a movable part moved closer to the fixed part from tank 2 through a connection of the diameter 1000 mm between the tanks. The movable part of the device is based on a dedicated trolley that travels horizontally on rail guides.

## Main areas of studies

The reactor is primarily used for research into the radiation damage mechanisms of the existing and advanced structural and absorber materials.

## Neutron flux (Φ) in the RBT-6 core irradiation channels

| Channel No. | Channel filling fluid | $\Phi$, $10^{13}$ $cm^{-2} \cdot s^{-1}$ | |
|---|---|---|---|
| | | < 0.67 eV | > 0.1 MeV |
| 1 | Gas | 6.1 | 5.6 |
| | Water | 22 | 4.2 |
| 2 | Gas | 5.2 | 5.0 |
| | Water | 19 | 3.6 |
| 3 | Gas | 5.8 | 5.5 |
| | Water | 20 | 4.1 |
| 4 | Gas | 5.5 | 5.7 |
| | Water | 21 | 4.2 |
| 5 | Gas | 4.7 | 4.2 |
| | Water | 17.5 | 3.0 |
| 6 | Gas | 3.2 | 2.8 |
| | Water | 11.0 | 2.0 |
| 7 | Gas | 4.6 | 3.7 |
| | Water | 16.2 | 2.4 |
| 8 | Gas | 4.1 | 3.3 |
| | Water | 15.0 | 2.2 |

## Neutron flux (Φ) in the RBT-6 reflector irradiation channels

| Channels | $\Phi$, $10^{12}$ $cm^{-2} \cdot s^{-1}$ | | |
|---|---|---|---|
| | $E < 0.5$ eV | $0.5$ eV $< E < 0.5$ MeV | $E > 0.5$ MeV |
| Row 1 of the "Korpus" irradiation device | 0.53 | 13.0 | 0.51 |
| Row 2 of the "Korpus" irradiation device | 0.14 | 3.0 | 0.046 |
| Si doping channel | 22.0 | 5.2 | 0.75 |

Recently:

– a great deal of post-irradiation research has been performed to investigate the creep of cladding tubes of the E110 and E635 alloys during longitudinal tension and pressure loading;

– work has been undertaken to test the copper alloys used in the ITER project;

– vessel steels for French power reactors have been tested under a contract with EdF;

– test production of advanced isotopes for medical applications ($^{59}Fe$) has been under way;

– test irradiation of minerals have been conducted as well as pilot and experimental work has been under way to identify the irradiation modes for radiation coloring of minerals.

## International cooperation

Research is conducted at the RBT-6 reactor as part of collaboration with the IAEA, the USA, the EU, the CIS and other countries.

## Main activities

The reactor utilization factor was 0.52 in 2011 and the experimental channel utilization factor was 0.15.

In 2011 the RBT-6 reactor was heavily involved in a series of experiments in both the field of reactor materials technology studies and test production of radioisotopes:

– experimental work was continued to investigate the radiation stability of copper alloys for the ITER international fusion reactor;

– test irradiations of RIAR-manufactured targets were conducted for production of $^{99}Mo$ in flow-type reactor channels;

– experimental calculation and development work was undertaken as part of the program to improve the efficiency of $^{99}Mo$ generation using square targets manufactured by the Novosibirsk Plant of Chemical Concentrates;

– uranium dioxide samples with a regulated microstructure were irradiated for 60 days running in a irradiation test as part of the effort on in-pile studies into the radiation creep of uranium-dioxide fuel with a regulated microstructure (the experiment was conducted to demonstrate the functionality of advanced nuclear fuel for the VVER-type reactors);

– long-term in-pile service-life tests of gas-filled samples were conducted at temperatures up to 750 °C;

– in-pile relaxation tests were performed under the Kvadrat FA structural material testing program on samples of the spring material out of a high-nickel alloy, and preparations were continued for in-pile tests using the Neytron loading machines;

– minerals were irradiated as part of a pilot experimental work on identification of the irradiation modes in irradiation devices for radiation coloring of minerals.

The near-term plans include:

– in-pile tests of materials in ampoule-type devices at different temperatures and in different environments with in-pile studies on creep and radiation-induced growth;

– neutron activation analyses and neutron radiography;

– tests of structural materials for the ITER international fusion reactor;

– tests of structural materials in the "Korpus" irradiation device;

– an increase in the range and volume of radionuclide production ($^{131}Cs$, $^{131}Ba$, $^{14}C$, $^{60}Co$ and others);

– arrangement of channels for nuclear doping of silicon ingots of the diameter up to 200 mm;

– creation of radiation technologies for transmutation and change of the physical and chemical properties of materials for industrial applications;

– on-the-job training and training of students and post-graduate students majoring in nuclear physics and in nuclear power plants and facilities;

– in-pile tests of fuels (coated particles, kernels and fuel compacts) for the developed gas-turbine modular helium reactor (GT-MHR);

– safety analysis for using spent SM-3 reactor FAs with the initial $^{235}U$ content of 6 g per a fuel element.

## History

The development of the reactor design began after the resolution to build the reactor was approved on 06.06.1969 by A.I. Petrosyants, the Chairman of the State Committee for Peaceful Uses of Atomic Energy. The design was completed in 1973.

The work to increase the reliability and improve some of the RBT-6 reactor components and systems has been undertaken continuously:

– August 1980 – a '2 out of 3' logic circuit was introduced for emergency protection for the rate of the power level increase;

– July 1982 – to avoid exceeding a rated power value after the automatic level is achieved, a power setter was introduced for use for the further power increase;

– June 1987 – technologically obsolete instrumentation in the emergency protection circuit was replaced for three power-level protection devices based on a '2 out of 3' actuation logic;

– July 1987 – the reactor site was fitted with light and sound alarms to warn personnel of more than two EP-SM being inserted into the core;

– August 1989 – steel valves were installed instead of cast-iron ones as required by PNAE G-7-008-89 rules for the secondary circuit valves;

– April 2009 – the EP-SM drive control circuit was updated to enable the control member motor power supply circuit disconnection from the reactor control room.

In 1991–1992 work was also undertaken during the SM reactor upgrading to improve the RBT-6 reactor safety:

– instruments were replaced for more advanced instrumentation, and monitoring lines for the most important process parameters were backed up;

– the CPS was improved;

– the ECCS equipment had its power supply circuit altered and was additionally connected to the uninterruptible power supply system;

– two diesel power stations were built with a function of automatic actuation in the event of offsite power loss;

– a gate valve was installed in the primary circuit line between the heat exchanger and the reactor pool to cut off the pool from the pressure pipeline in the event of the latter's break;

– the pool's overflow drain pipe was fitted with an electrically operated valve to isolate the faulty length and avoid spontaneous pool emptying in the event of its break within the pool.

## Contact person

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# MIR.M1 MATERIAL TEST REACTOR

The MIR-M1 material test reactor is a unique multirole facility used to test structural, fuel and absorber materials applied or considered for use in nuclear industry.

The MIR.M1 research reactor reached first criticality on 24.12.1966 and its energy startup took place on 11.08.1967. It was in 1968 that experimental research activities were launched at the reactor.

Physically, the MIR.M1 reactor is a heterogeneous thermal-neutron reactor with a moderator and a reflector of beryllium metal. Structurally, it is a channel-type reactor and is accommodated in a pool with water. It was such design that allowed combining the best advantages of the pool-type and channel-type reactors.

As part of the retrofit in 1975 the reactor had six working FAs and four or five control rods installed around each of its loop channels. This made it possible to make all loop cells in the core practically equivalent and achieve better conditions for individual regulation of test modes in each of the channels.

*MIR.M1 reactor main building*

Orderly and task-oriented work on improving the operations of the reactor facility and extend its service life will make it possible to use the MIR.M1 reactor for research applications until approximately 2025.

Each FA of the MIR.M1 reactor is made of four tubular fuel elements retained in the upper and lower racks. The fuel elements have the thickness of 2 mm and three azimuthally oriented ribs on their outside throughout the length. The fuel is uranium dioxide ($UO_2$) with a 90% enrichment in the $^{235}U$ isotope.

The fuel layer thickness is 0.56 mm. The fuel cladding made of the SAV-6 alloy has the thickness of 0.72 mm. Each FA contains from 345 to 350 g of $^{235}U$. There is a displacer of the diameter 38 mm made of the SAV-6 alloy inside the central fuel element.

Heat is removed from the reactor in two circuits and is discharged to the atmosphere through a cooling tower.

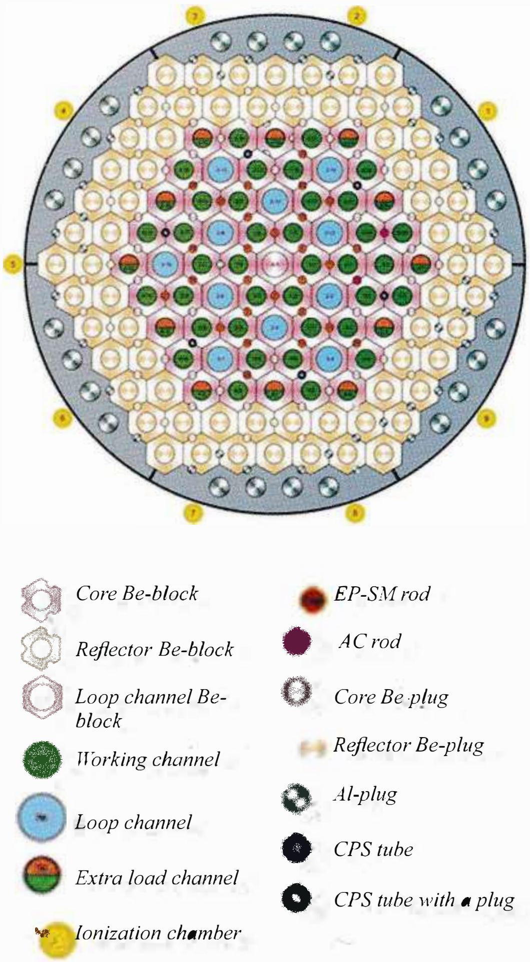
The coolant circulation through the FA is downward.

The primary circuit comprises reactor coolant pumps, heat exchangers, pressurizers, valves and pipelines. It is intended to remove heat from the core and transfer it to the recirculating water supply system, and to retain the active fluid.

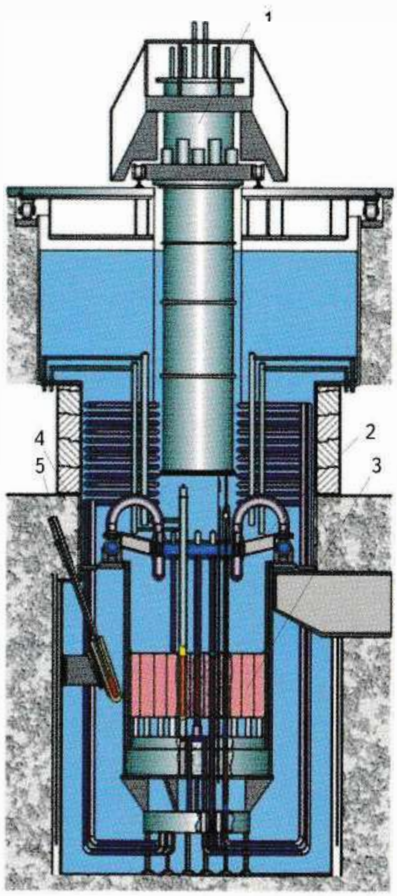
For the purpose of biological shielding, ensuring the safety of underwater maintenance

## Main performance of the MIR.M1 reactor

| | |
|---|---|
| Thermal power, max. | 100 MW |
| Loop channel diameter, max. | 120 mm |
| Number of loop channels, max. | 11 |
| Thermal neutron flux, max | $5 \cdot 10^{14}$ cm$^{-2} \cdot$s$^{-1}$ |
| Average core bulk power density | 0.85 MW/l |
| Coolant: | Water |
| core inlet pressure | 1.25 MPa |
| reactor inlet temperature | 30…70 °C |
| reactor outlet temperature | Up to 98 °C |
| Cycle length | Up to 40 days |




A map of the MIR.M1 reactor core

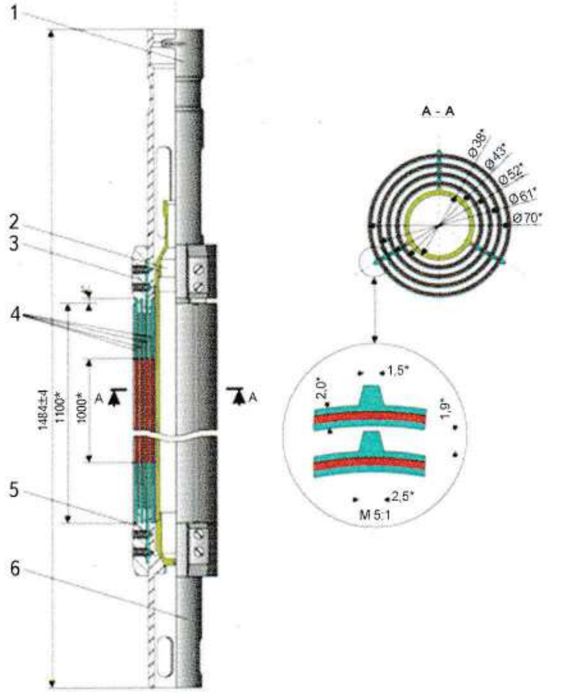



Sectional view of the MIR.M1 reactor:
1 – CPS drive area; 2 – delivery header system; 3 – reactor core; 4 – discharge piping system; 5 – ionization chamber

and refueling, improving the reactor facility safety and confining radioactive emissions in emergencies, the MIR.M1 reactor core is deployed in a water-filled pool, which is a component of the reactor pool cooling circuit (PCC).

The PCC is intended to remove radiation heat from beryllium blocks, actuators and structural components of the reactor.

The PCC coolant flows downward between the beryllium blocks of the core stack and the CPS tubes, and in process gaps in the reactor structures.



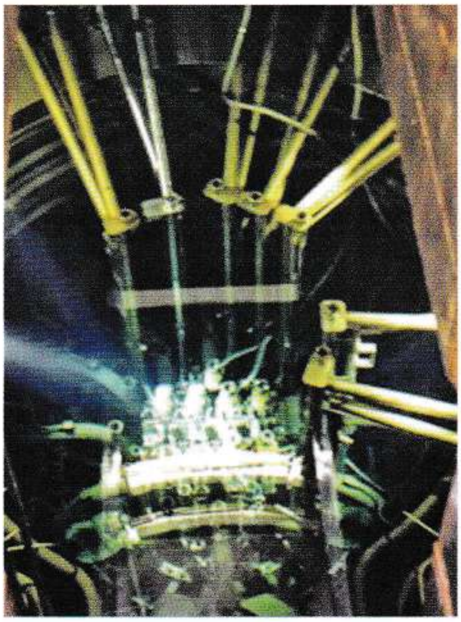

*MIR.M1 FA:*
*1 – head; 2 – displacer; 3 – upper rack; 4 – fuel elements; 5 – lower rack; 6 – leg*

## Experimental capabilities of the MIR.M1 reactor

The MIR.M1 reactor is designed for testing prototype fuel elements, fuel assemblies and structural materials of nuclear facilities for different applications operating at different loads in different environments (gas, water, liquid metals, organic compounds).

The reactor's major features include 11 in-core experimental loop channels, which are connected to independent loop facilities with different coolant types, for tests in different thermal and hydraulic conditions.

At present time, there are 7 loop facilities in operation at the MIR.M1 reactor, with one or two loop channels connected to each of these.



*Central hall. The reactor lid as seen from above*

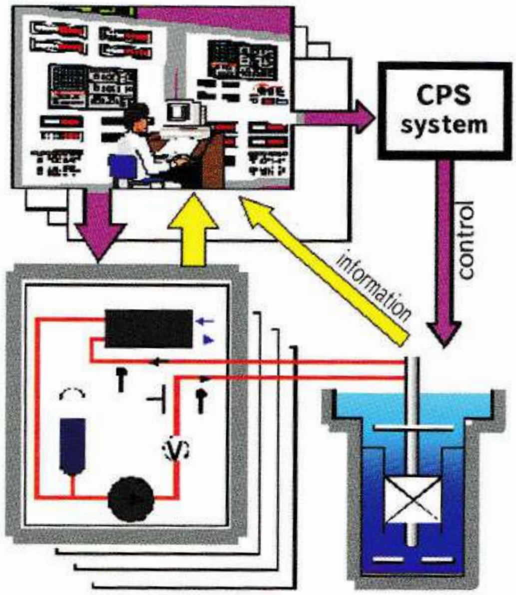
*The core inside the pool.*
*Loop channel communication lines are seen*

## PV-1 water loop facility

Intended for in-pile tests of fuel elements, fuel assemblies and structural materials for water-cooled nuclear reactors. One or two experimental in-pile loop channels are connected to the facility.

Maximum parameters of the PV-1 facility:

- thermal power of the facility – 2000 kW;
- thermal power of one channel – 1000 kW;
- primary circuit pressure – 17 MPa;
- coolant temperature:
  - channel inlet – 300 °C;
  - channel outlet – 350 °C;
- coolant flow rate – up to 16 t/h.



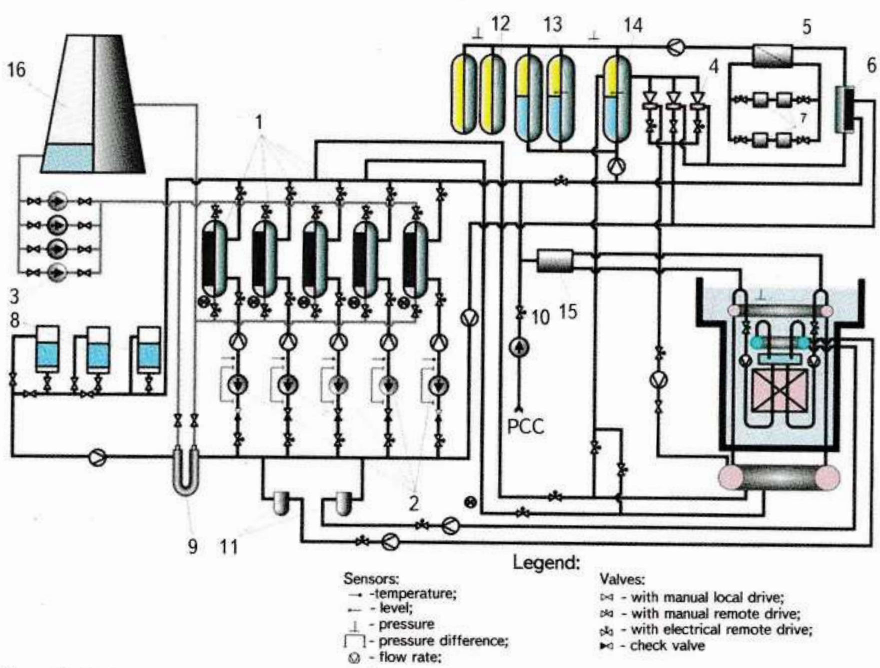

*Experimentation flow sheet for loop facilities*

## PV-2 water loop facility

Intended for in-pile tests of fuel elements, fuel assemblies and structural materials for water-cooled nuclear reactors. One or two experimental in-pile loop channels are connected to the facility.

Maximum parameters of the PV-2 facility:

- thermal power of the facility – 2500 kW;
- thermal power of one channel – 1500 kW;
- primary circuit pressure – 18 MPa;
- coolant temperature:
  - channel inlet – 310 °C;
  - channel outlet – 350 °C;
- coolant flow rate – up to 16 t/h.




*Flow sheet of the MIR.M1 reactor core cooling:*
*1 – primary circuit heat exchangers; 2 – primary circuit pumps; 3 – recirculating water supply system pumps; 4 – ejectors; 5 – regenerator; 6 – heat exchanger of the oxyhydrogen gas combustion system; 7 – contact apparatuses; 8 – ion-exchange filters of decontamination treatment system; 9 – heat exchanger of decontamination treatment system; 10 – emergency cooldown pumps; 11 – mechanical filters; 12 – gas cylinders; 13 – pressurizers; 14 – degasifier; 15 – fuel cladding failure detection system; 16 – cooling tower*

## PVK-1 steam-water loop facility

Intended for in-pile tests of fuel elements, fuel assemblies and structural materials for water-cooled nuclear reactors in both modes with and without coolant boiling. One or two experimental loop channels are connected to the facility.

Maximum parameters of the PVK-1 facility:

- thermal power of the facility – 2000 kW;
- thermal power of one channel – 1000 kW;
- primary circuit pressure – 17 MPa;
- coolant temperature:
  - channel inlet – 300 °C;
  - channel outlet – 350 °C;
- coolant flow rate – up to 16 t/h;
- in-channel steam quality – up to 40 %.

## PVK-2 steam-water loop facility

Intended for in-pile tests of fuel elements, fuel assemblies and structural materials for water-cooled nuclear reactors in modes both with and without coolant boiling. One or two experimental in-pile loop channels are connected to the facility.

Maximum parameters of the PVK-2 facility:

- thermal power of the facility – 2500 kW;
- thermal power of one channel – 1500 kW;
- primary circuit pressure – 18 MPa;
- coolant temperature:
  - channel inlet – 350 °C;
  - channel outlet – 355 °C;
- coolant flow rate– up to 16 t/h;
- in-channel steam quality – up to 40 %.

## PVP-1 steam loop facility

Intended for service life tests of prototype fuel elements, fuel assemblies and structural materials in the loop channel of the reactor core. In the primary coolant circulation pipeline, slightly superheated steam is generated in a heat exchanger and used to cool the fuel assembly of the loop channel and to transfer heat to the condenser. One experimental in-pile loop channel is connected to the facility.

Maximum parameters of the PVP-2 facility:

- FA power – 100 kW;
- FA inlet steam temperature – 300 °C;
- FA outlet steam temperature – 510 °C;
- fuel cladding temperature – 655 °C;
- FA outlet coolant pressure – 6.5 MPa.

## PVP-2 steam-water loop facility

Intended to carry out serviceability and feasibility studies for the core components of advanced power reactors in a broad temperature and pressure ranges, including emergency modes of the fuel element operation. The facility has three circuits.

Maximum parameters of the PVP-2 facility's primary circuit:

- channel outlet coolant temperature – 550 °C;
- coolant flow rate – up to 10 t/h;
- pressure – 20 MPa.

## PG-1 loop facility

Intended for comprehensive studies into the serviceability of fuel elements and fuel assemblies for advanced gas-cooled power reactors of the HTGR type. Heat is removed by the primary gas coolant and transferred to the fluid in the heat exchangers of two parallel cooling system circuits. From the cooling system, heat is transferred to the production circuit and to the recirculating water supply system of the MIR.M1 reactor facility.

Maximum parameters of the PG-1 facility's primary circuit:

- channel outlet coolant temperature – 600 °C;
- coolant flow rate – 1.3 kg/s;
- pressure – 20 MPa.

## Main areas of studies

Tests of fuel elements and fuel assemblies for the cores of improved-safety new-generation reactors in design modes.

Tests of irradiated and unirradiated fuel elements in conditions that simulate different emergencies.

Studies into the behavior of failed irradiated fuel elements in steady-state and transient modes.

## International cooperation

The MIR.M1 research reactor is used for research programs as part of collaboration with the IAEA and a number of countries, including China and the Republic of Korea.

## Main activities

The reactor utilization factor in 2011 was 0.66.

The MIR.M1 emergency control room was set up.

The emergency power supply system was improved.

Neutron flux intellectual instrumentation channels produced by ASPECT were assembled for the fuel cladding failure detection system.

The system was upgraded for monitoring the parameters responsible for the safety of the PV-1 and PVK-1 loop facilities.

The FA emergency cooldown system was assembled and put into operation.

Radiation tests were completed on miniature fuel elements with monolithic uranium-molybdenum fuel and fuel elements based on cermet fuel in a zirconium cladding.

The near-term plans include:

- introduction of a new radiation monitoring system;
- improvement of the FCFD system;
- upgrading of the PV-1 LF's measuring systems;
- upgrading of the PVK-1 LF's measuring systems.

## History

The idea of building a high-power loop-type test reactor for the progression of experimental work in nuclear power (for tests of prototype fuel elements for power reactors and structural materials) was conceived by I.V. Kurchatov. It was on his proposal that the development of such reactor called MIR was begun at the Institute of Atomic Energy (IAE) in 1956. In 1958–1959 experiments were conducted at the IAE's MIR critical facility, as the result of which the physical scheme was chosen for the reactor. In 1958 the reactor design began to be developed at NIKIET, while in 1966 the design was implemented at RIAR's site and the MIR reactor was put into operation.

Yu.M. Bulkin was immediately in charge of the work at the Head Designer organization (NIKIET). The IAE was responsible for the effort's scientific supervision at the design and construction stage (V.V. Goncharov, Yu.G. Nikolaev), and RIAR took this over for the commissioning preparation stage (V.A. Tsykanov). VNIPIET was appointed the General Architect Engineer (M.L. Barskiy). The concepts implemented first in the RFT reactor (1952) and then in the MR reactor (1964), were embodied, as fully as they could be, in the MIR reactor design. This included conceptual features that determined the achievements of the country's reactor building:

- channel-type design of the reactor;
- core deployment in a water-filled pool;
- use of independent loop channels in the core;
- mounting of the CPS drives on a retractable trolley.

Unlike the MR reactor, which was built as the replacement of the RFT in a big city, the MIR reactor was constructed rather far from large residential areas. Among other things, this allowed bringing the reactor power to 100 MW. A fundamentally new approach was that the pipelines of the loop channels were installed not inside the reactor but in dry chambers beneath the pool water layer. This made it possible to decrease the dimensions of the loop channels, facilitated refueling operations and increased safety during operation and maintenance.

The reactor design was given the capabilities to carry out research in in-pile independent loop facilities with different types of coolants in different thermal and hydraulic modes.

Initially, the reactor operated with four loop channels in the core, which were connected in pairs to the two loops available at the time: the PV-1 water loop and the PVK-1 steam-water loop of 2 MW each. A bit later two lead-bismuth-cooled loops were put into operation. These had the power of 500 kW and one experimental channel each. There was also a sodium-cooled loop of 2 MW with two loop channels put into operation. After the water and the steam-water facilities had one more channel connected to each, the total number of loop channels in the core reached 10.

The core was found to have a drawback at the initial reactor operation stage. This consisted in that the power density level in the loop channels of the stack block row 4 turned out to be too low, and the power peaking in the loop assemblies of those channels was found to be excessively high. The upgrade given to the reactor in 1975 made it possible to make all loop cells practically equivalent and achieve improved conditions of the test mode regulation in separate channels, and so harmonize the set of diverse modes. After the upgrade, the reactor was fitted with the PV-2 and PVK-2 water loops. Following the launch of the BOR-60 reactor, the sodium-cooled loop was transformed into a loop with a high-temperature organic coolant.

A loop with a steam-water coolant, indexed PVP-1, was made operational in 1983. This had parameters close to those of the RBMK reactor coolant. Two loop facilities, the gas-cooled PG-1 facility and the PVP-2 facility with a steam-water coolant, were put into operation in 1989-1990. These facilities have additional safety barriers in the form of leak-tight containments. This makes the facilities fit for conducting specific experiments on simulating and studying emergencies with fuel elements in the loop channel.

Orderly work was carried out to refurbish the major reactor components and systems:

- 1992 – replacement of the CPS rods;
- 1995–2002 – phased replacement of the MIR.M1 reactor beryllium blocks;
- 2001 – application of fire safety paste to the KRU-6kW cables;
- 2002–2006 – partial replacement of the channels for the working FAs;
- 2003:
  - retrofit of the monitoring system for primary circuit parameters, reactor and working FA thermal power based on digital hardware;
  - replacement of worn out and outdated instrumentation for monitoring the coolant flow through the supply pipelines, working channels and extra-load channels of the MIR.M1 reactor facility, and introduction into the reactor emergency protection system of a '2 out of 4' majority logic to secure against a flow rate reduction in supply pipelines;
  - upgrade of the primary circuit heat exchangers;
- 2005 – replacement of the CPS electronics for the ASUZ-09P hardware set;
- 2006:
  - upgrade of the PV-1 and PVK-1 LF process monitoring and automated radiation monitoring systems;
  - upgrade of the reactor coolant pump independent circuits at the MIR.M1 reactor facility;
  - construction of a new nuclear material storage facility at the MIR.M1 reactor;
- 2007:
  - upgrade of the automatic chemical control system for LF primary coolant;
  - upgrade of the reactor working FA emergency cooldown system;
  - reinforcement of the physical protection system.

A review into the state and results of operation of the major process components and pipelines demonstrates the feasibility of their further operation. A program has been developed for the reactor improvement and service life extension to 2020. The near-term plans include further replacement of the core and CPS components and creation of additional safety systems.

## Personalities

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# RBT-10/2 POOL-TYPE REACTOR

The RBT-10/2 research reactor is a pool-type water-cooled water-moderated thermal-neutron reactor deployed in a pool, which has the same design as that for the RBT-10/1 reactor.

The reactor was designed as the source of neutrons for long-term ampoule tests of various materials to study their properties in neutron fluxes up to $(2\ldots7)\cdot10^{13}$ cm$^{-2}\cdot$s$^{-1}$ in steady-state modes.

The first criticality was achieved at the RBT-10/2 reactor in two steps:

- on 24.11.1982 (with the RBT-10/1 fuel);
- on 26.11.1984 (with spent SM reactor FAs).

The energy startup of the RBT-10/2 reactor took place on 24.12.1984.

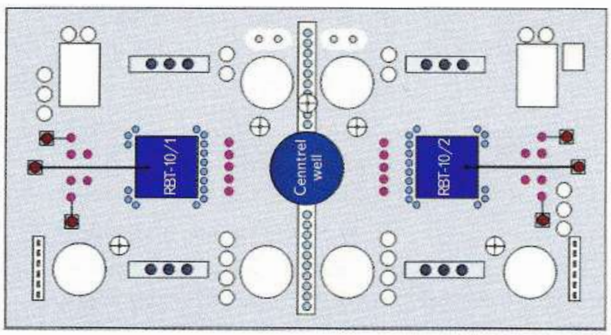
Work is continuously under way at the RBT-10/2 reactor to raise the reliability and improve the reactor parts and components for the purpose of increasing the reactor operating safety.

The reactor core is deployed in a pool, which is a component of the system for heat removal from the core and discharge of heat into the atmosphere via a cooling tower using a two-circuit system.

## Main performance of the RBT-10/2 reactor

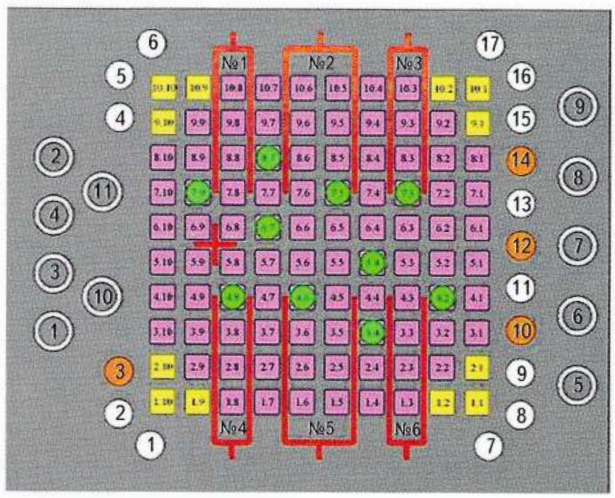
| | |
|---|---|
| Thermal power: | |
| design | 10 MW |
| permitted | 7 MW |
| Number of working FAs in the core | 78 |
| Number of ampoule-type channels in the core | 10 |
| Cycle length | Up to 120 days |
| Number of ampoule-type channels in the reflector | 17 |
| Starting burn-up in loaded FAs, average | 32 % |
| Core coolant flow rate: | |
| at N=7 MW | 650…700 m³/h |
| at N=10 MW | 820…1000 m³/h |
| Core inlet coolant temperature | up to 60 °C |
| Core outlet coolant temperature | up to 70 °C |

Spent FAs of the SM-3 reactor are used as the RBT-10/2 fuel assemblies. The core is made up of largely FAs with the burn-up of 10 to 30 % (but no more than 50 %). Fresh FAs may be also used.



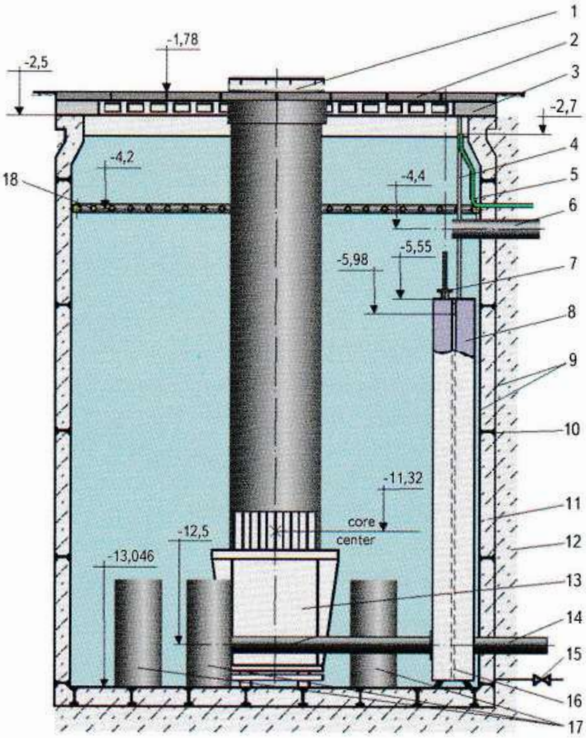

*Arrangement of the RBT-10/1, /2 reactor facilities*

*RBT-10 control room*




*A map of the RBT-10/2 reactor core:*

- *working FAs;*
- *beryllium blocks;*
- *cells for ampoule-type channels in the core;*
- *peripheral ampoule-type channels;*
- *ionization chambers;*
- *peripheral ampoule-type channels for nuclear silicon doping*




*RBT-10/2 reactor pool:*
*1 – central well; 2 – servicing platform; 3 – ventilation duct; 4 – oxygen activity damper (OAD) surge line; 5 – overflow tube; 6 – pressure pipeline; 7 – K-2 valve; 8 – OAD; 9 – pool lining; 10 – I-beam; 11 – sand-and-gravel solution; 12 – shielding concrete; 13 – support structure; 14 – suction pipeline; 15 – Ø56 test tube; 16 – OAD partition; 17 – 12 FA baskets; 18 – decontamination treatment header*

## Experimental capabilities of the RBT-10/2

### Production of $^{131}I$ preparation

As stipulated by the RIAR plans for opening production of $^{131}I$-based radionuclide products, the RBT-10/2 reactor facility ampoule channels were upgraded. In 2000 test irradiation of two tellurium-dioxide target batches was conducted in an irradiation device in a flowing ampoule-type channel.

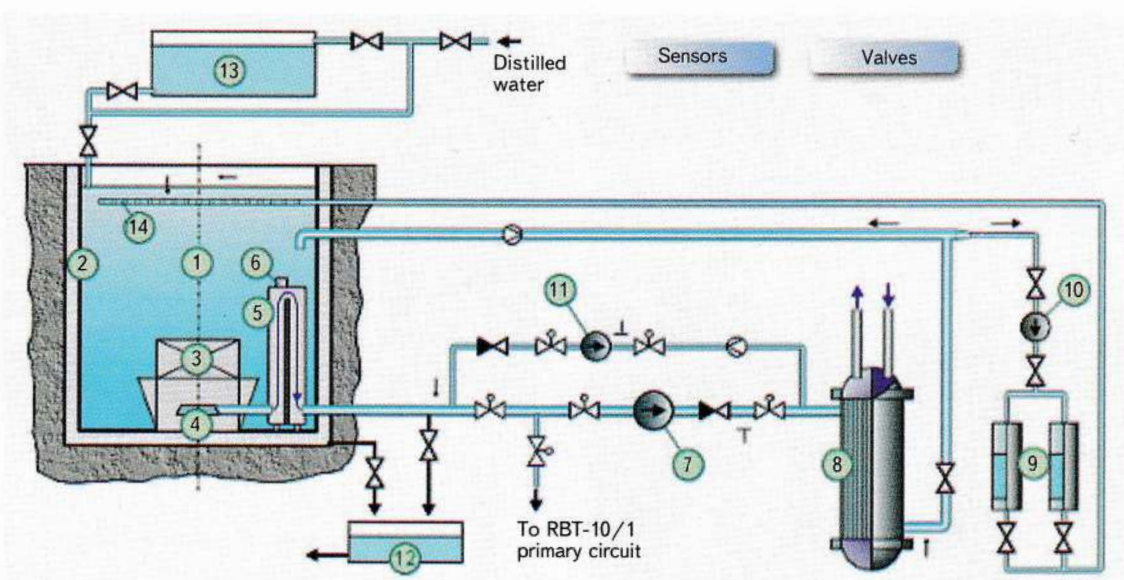
### Nuclear doping of silicon

Ingots of monocrystalline silicon are irradiated in the reactor's peripheral channels in a dedicated experimental device for silicon irradiation. The device is fitted with an electric motor that ensures reciprocating and rotational movements of the container with samples. These movements, together with the sample rotation by 180° after irradiation during a half of the design time, make it possible to dope samples along the length with an irregularity of not more than 3...5 %.

### Radiation coloring of minerals

A technology for radiation coloring of minerals was developed and patented for the RBT-10/2 reactor facility. At present time, commercial coloring of minerals is carried out.

Minerals are irradiated in the "Mineral" experimental device for 2…8 days depending on the reactor power, the fluence preset and the size of stones.




*Flowchart of the RBT-10/2 primary circuit:*
*1 – reactor pool; 2 – pool liner; 3 – reactor core; 4 – suction header; 5 – oxygen activity damper; 6 – valve; 7 – circulation pump; 8 – heat exchanger; 9 – ion-exchange columns; 10, 11 – pumps; 12 – neutralization tank; 13 – emergency cooling tank; 14 – decontamination treatment pipeline*

## Main areas of studies

At present time, the RBT-10/2 reactor facility is used for:

- accumulation of industrial and medical $^{131}I$;
- nuclear doping of silicon;
- radiation coloring of minerals;
- work to prepare for the accumulation of $^{99}Mo$.

The long-term program for the RBT-10/2 utilization includes 4 topics:

- research into the properties of materials for components of NPP cores in in-pile conditions;
- irradiation of samples and targets for commercial production of radionuclides ($^{32}P$; $^{33}P$; $^{60}Co$; $^{99}Mo$; $^{131}I$; $^{133}Xe$; $^{192}Ir$ and others);
- irradiation of samples and targets for obtaining materials with changed electrophysical, thermal, electrical and optical properties (nuclear doping of silicon, radiation coloring of minerals, irradiation of polymers and others);
- feasibility study of processes for production of various radioactive nuclides.

## International cooperation

Work has been carried out as part of international cooperation with foreign partners (Germany, Poland, China, Switzerland, Israel) in:

- accumulation of industrial and medical $^{131}I$;
- radiation coloring of minerals.

## Main activities

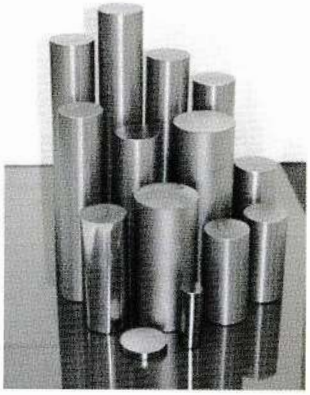
The reactor utilization factor was 0.69 in 2011.

The pilot rise of the RBT-10/2 reactor to rated power (N=10 MW) was undertaken, which demonstrated the feasibility of the reactor operation with the upgraded primary circuit and the improved system for monitoring of thermal parameters at the rated power. Physical measurements have proved it to be possible to improve the efficiency of the reactor's experimental channels.

An information and measurement system has been put into operation.

Experimental work was undertaken to validate the RBT-10/2 neutronic characteristics during $^{99}Mo$ production.

Work was conducted on nuclear doping of silicon.



*Monocrystalline silicon ingots*

*Samples of minerals*

The near-term plans include:

- the reactor rise to the power level of N=10 MW;
- introduction of projects to perfect handling equipment;
- introduction of the project "Radiation Monitoring System for the RBT-10 – MIR.M1 Reactor Complex";
- generation of $^{99}Mo$;
- a safety analysis for the use of spent SM-3 reactor FAs with the initial $^{235}U$ content of 6 g per fuel element.

## Contact person

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# BOR-60 FAST EXPERIMENTAL REACTOR

The BOR-60 reactor facility is a fast-neutron sodium-cooled reactor facility. This is a prototype NPP with the power generation of 12 MW. The reactor reached its first criticality on 30.12.1968 and its energy startup took place on 28.12.1969. The BOR-60 reactor facility systems were connected to the grid on 28.12.1970.

The BOR-60 fast-neutron experimental reactor is a unique multipurpose facility used for tests of structural, fuel and absorber materials employed or to be employed in different types of nuclear reactors, including fusion reactors. Additionally, tests of separate parts for fast reactor circuit components are conducted.

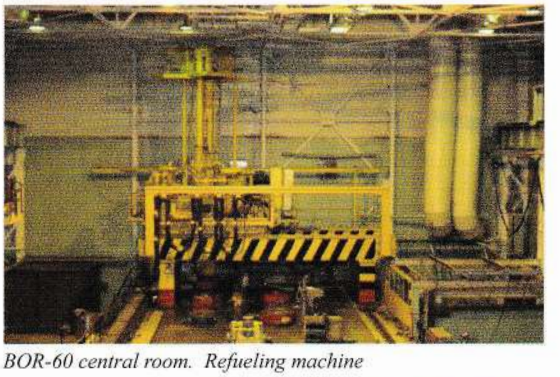
Reactor material test programs extend across the spectrum of reactor types in existence or under development from fast reactors (BN-800, BN-1200, BREST, SVBR) and thermal reactors (AES-2006, VVER-1500, GT-MHR, HTGR) to fusion reactors (ITER) and special-purpose reactors.

Scheduled and purposeful activities undertaken for improving components, systems and processes under the reactor facility service life extension program make it possible to operate the BOR-60 for research purposes until 2015.

## Main performance of the BOR-60 reactor

| | |
|---|---|
| Thermal power, max. | 60 MW |
| Electric power | 12 MW |
| Fast neutron flux, max. | $3.7 \cdot 10^{15}$ cm$^{-2}\cdot$s$^{-1}$ |
| Average neutron energy | 380 keV |
| Bulk in-core thermal power density | 1100 kW/l |
| Linear thermal rating on the fuel surface, max. | 450 W/cm |
| Damaging dose accumulation rate | Up to 25 dpa/year |
| Coolant: | Sodium |
| flow rate through the reactor | 1100 m$^3$/h |
| reactor inlet temperature | Up to 360 °C |
| reactor outlet temperature | Up to 530 °C |
| Reactor coolant pump discharge pressure | 0.55 MPa |
| Microcycle length | 90...120 days |
| Interval between microcycles | 45 days |

*BOR-60 building*



*BOR-60 central room. Refueling machine*

*BOR-60 control room*

The BOR-60 reactor facility has a two-loop three-circuit system for heat removal from the reactor. Sodium is used as the primary and the secondary coolants. The tertiary circuit uses a steam-water coolant and comprises steam generators, a turbine generator and a district heating plant.

The primary circuit is intended for removal of heat from the reactor and its transfer to the secondary circuit in the intermediate heat exchangers. The circuit has two symmetrical loops connected to the reactor vessel. Each loop removes from the reactor 50% of its power.

In the event of a defect in one loop, the reactor is scrammed, and the faulted loop is isolated with

## BOR-60 nuclear fuel performance

| | |
|---|---|
| FA width across flats | 44 mm |
| FA length | 1575 mm |
| Shroud wall thickness | 1.0 mm |
| FA spacing | 45 mm |
| Fuel element dimensions (diameter×cladding thickness) | 6.0×0.3 mm |
| Fuel element length | 1083 mm |
| Number of fuel elements per FA | 37 |
| Core fuel: | |
| granulated material composition | $PuO_2+UO_2$ or $UO_2$ |
| $PuO_2$ share | 20…29 w. % |
| gettering agent – depleted U metal | 5…10 w. % |
| Material of end screens | $UO_2$ (depleted) |
| In-core fuel density, eff. | $8.8…9.2\cdot10^3$ kg/m$^3$ |
| Weight of fuel in FA (active part) | 3440 g |
| Core fuel burn-up | |
| maximum | 20 % h.a. |
| average | 7…9 % h.a. |

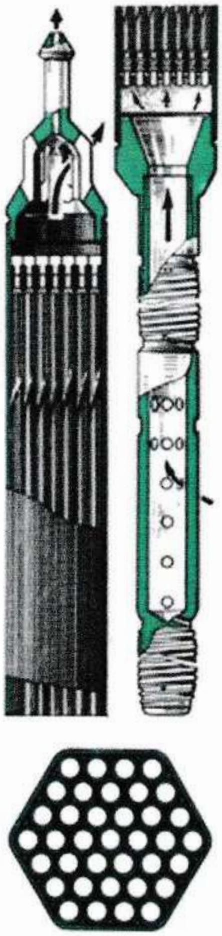
the aid of valves, while the reactor is cooled down by the operating loop.

Each primary circuit loop comprises a circulation sodium pump, an intermediate heat exchanger, pipelines with valves and auxiliary systems.

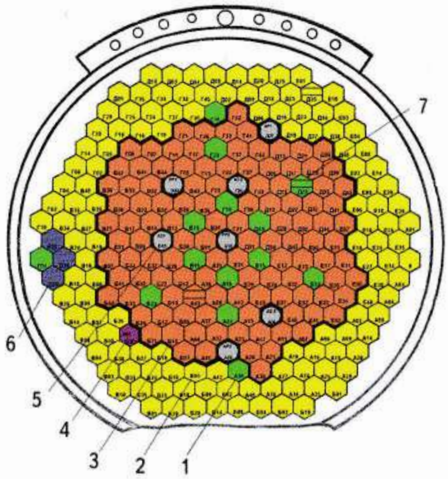
The secondary system is intended to prevent contamination of the tertiary, steam-water, circuit with the primary circuit radionuclides when an intermediate heat exchanger tube bundle is not leaktight, that is to improve the radiation safety of the reactor facility. This is achieved through the elevated arrangement of the secondary circuit relative to the primary circuit and a more high pressure (~ 6.0 atm gage) against the primary circuit pressure (0.5 atm gage). Besides, the secondary circuit is used to remove heat from the primary circuit in the intermediate heat exchangers and transfer it to the tertiary circuit in the steam generators.

Likewise the primary circuit, the secondary circuit has two symmetrical loops of the same power (30 MW), which makes it possible to use similar thermal and mechanical components.

Each secondary circuit loop comprises an intermediate heat exchanger, a sodium circulation pump, a steam generator, an air heat exchanger (AHX) (4 AHX sections are connected to a hot header and a cold header, which integrate the two secondary circuit loops), main circulation pipelines with valves, and auxiliary systems.



*BOR-60 fuel assembly*




*BOR-60 core map:*
*1 – material test assembly; 2 – side screen assembly; 3 – FA; 4 – neutron source; 5 – CPS rod; 6 – zirconium hydride; 7 – instrumented cell*

## Experimental capabilities

For experiments the reactor offers:

– an experimental cell, where instrumented fuel and material test assemblies or independent loops are installed, enabling data acquisition during irradiation;

– a capability for the simultaneous insertion of up to 12 irradiation devices with structural materials in the reactor core. Number of experimental FAs with advanced fuel compositions and irradiation devices with structural materials in the side screen is practically not regulated;

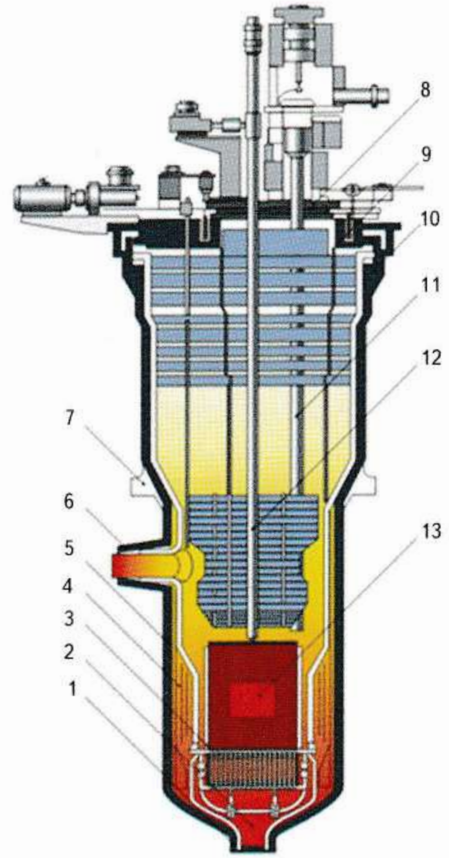
– 9 "dry" vertical channels behind the reactor vessel of the diameter 90 and 200 mm and of the height 7 m, with the neutron flux of $10^{13}$ $cm^{-2} \cdot s^{-1}$;

– 2 horizontal channels of the diameter 300 mm (1 tangential channel and 1 central channel) behind the reactor vessel with the neutron flux of $9 \cdot 10^{10}$ $cm^{-2} \cdot s^{-1}$;

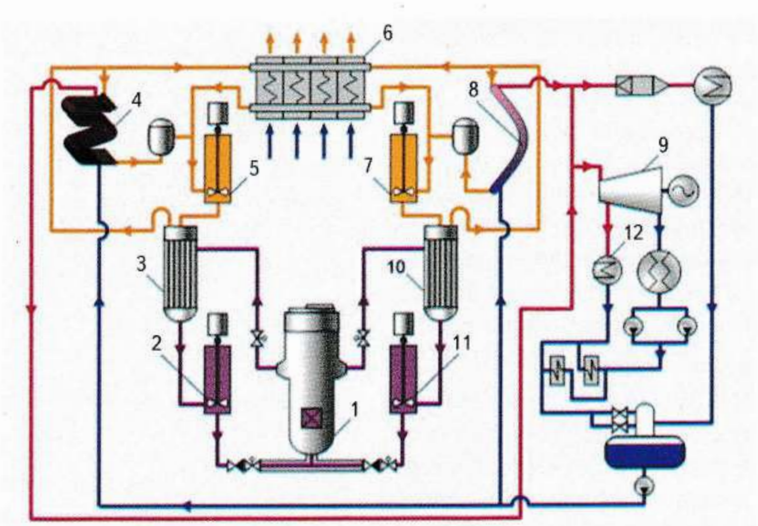
– a capability for testing prototypes of sodium-cooled circuit components and the diagnostic and protection systems in the facility's primary and secondary circuits.

The BOR-60 reactor facility operates in an integration with the material test laboratory and a pilot facility for fabrication and processing of nuclear fuel.

The research results obtained have been used to verify and certify the JAR-FR and MCU-RFFI/A neutronic codes and the DINBOR code for




*Sectional view of the BOR-60 reactor:*
*1 – inlet nozzle; 2 – HP chamber; 3 – barrel; 4 – thermal and neutron protection of the reactor vessel; 5 – safeguard vessel; 6 – outlet nozzle; 7 – support flange; 8 – smaller rotary plug; 9 – larger rotary plug; 10 – support flange; 11 – refueling channel; 12 – CPS drive; 13 – core and side screen assemblies*




*BOR-60 thermal diagram:*
*1 – reactor; 2, 5, 7, 11 – primary and secondary circuit pumps; 3, 10 – intermediate heat exchangers; 4, 8 – steam generators; 6 – air heat exchanger; 9 – turbine; 12 – heating unit*

thermal-hydraulic calculations and emergency mode analysis.

Procedures have been developed to safely monitor the modes and parameters of material irradiation both in instrumented and non-instrumented reactor cells.

## Main areas of studies

**Reactor material studies:**

– austenitic stainless steels used for the internals of water-cooled, water-moderated reactors (accelerated tests);

– Be and alloys based on V, Nb and Mo;

– absorber compositions;

– samples and materials for more accurate life determination and extension for the major fast reactor components;

– dependence of the change in shape, long-term strength and crack resistance at temperatures from 330 to 1000 °C for the dose of 200 dpa.

**Fuel studies:**

– tests of experimental FAs for statistical demonstration of the fuel element serviceability with a vibrocompacted fuel;

– tests of fuel elements and FAs up to the burn-up of over 30% h.a. in steady-state, transient and emergency conditions;

– demonstration of the possibility for burning weapons-grade plutonium;

– research into the technological aspects of fuel fabrication and processing for the minor actinide burner reactor and others;

– demonstration of the feasibility of closing the fuel cycle based on dry fuel processing techniques;

– in-pile tests of fuel compositions and pilot fuel elements for thorium breeding cores;

– study of the fuel element serviceability in ultimate operating modes (achievement of superhigh burn-up and superhigh damaging doses);

– in-pile validation of advanced fuel types and fabrication technologies (U-Pu-Zr alloys, uranium and plutonium dioxide mixtures, nitrides, carbides, carbonitrides).

**Radionuclide production:**

– generation of $^{89}Sr$ and $^{153}Gd$.

**Safety analysis for different types of reactors in operation or under design:**

– in-pile tests of the BREST-OD-300 and SVBR reactor fuel elements;

– tests of boron-containing materials to validate the service life of CPS rods made of refabricated boron carbide for fast-neutron reactors;

– treatment of the reactor primary coolant for removal of radioactive impurities using compact adsorbers installed into the core instead of FAs or instead of the side shield assembly;

– tests of advanced large-scale models and parts of steam generators and other components;

– development and testing of procedures, instruments and reactor core and component inspection and diagnostic systems.

## Performance of irradiation devices

| Type | Temperature, °C | Coolant flow rate, kg/h |
|---|---|---|
| Non-pressurized, flow-through | | |
| without heater | 300…350 | 2000…3000 |
| with metallic heater (tungsten) | 450…500 | 100…200 |
| with fueled heater (90 % $UO_2$) | 500…700 | 1000…2000 |
| based on standard FA (heater: 90 % $UO_2$) | Up to 700 | 2500…3500 |
| with convection (heater: tungsten) | 450…600 | 100…200 |
| Pressurized, with liquid metal or inert gas | | |
| with regulated heat resistance | 400…500 | |
| with conic capsule | Up to 600 | |
| with heat pipe (thermosyphon) | 650…900 | |
| Independent ampoule-type loops | 300…1000 | Up to 2500 |

## International cooperation

As part of international cooperation with the USA, France, Japan, Germany, China, Czechia and Poland, work has been under way to test different components, structural materials and fuel compositions, and numerous bilateral meetings are conducted for intensive exchange of experience in fast reactor operation.


*Geography of cooperation*

The BOR-60 reactor was used for training and on-the-job training of French, Indian and Chinese experts.

In 2006, the BOR-60 experts began to render consulting services to Chinese specialists on the sodium technology and adjustment of the CEFR experimental fast-neutron reactor components, and, in July and August 2010, took an active part in the program to achieve CEFR first criticality.

## Main activities

The reactor utilization factor was 0.65 in 2011.

Irradiation of structural materials at temperatures of 320…450 °C.

Production of target radionuclides: $^{63}$Ni, $^{89}$Sr, $^{153}$Gd.

Experimental research on long-term strength and creep of cladding steel under the action of internal pressure in conditions of irradiation at temperatures from 600 to 650 °C.

In-pile tests of hafnium hydride at temperatures of 500…600 °C.

In-pile tests of the SVBR-100 reactor fuel element mockups.

Irradiation of assembly to validate the serviceability of fuel elements with vibrocompacted and pelletized MOX-fuel and a getter thermal insulator.

In March 2010 a scientific and technical seminar, "The Role of the BOR-60 Reactor in the Innovative Evolution of Nuclear Industry", was held, which was dedicated to the 40th anniversary of the BOR-60 reactor.

The near-term plans include:

– implementation of the measures developed to extend the reactor life until 2015 and demonstration of the feasibility for its further operation after 2015;

– safe and reliable operation of the reactor, and arrangement of experiments for testing of structural materials and nuclear fuel for the existing and future nuclear reactor under international and Russian programs;

– validation of the SVBR-100 and BREST-OD-300 reactor core characteristics;

– reproduction, in the experimental environment of the BOR-60 reactor, of the most near-service conditions for advanced nuclear power plant core components, with a combined action of damaging factors and tests on the closed nuclear fuel cycle;

– generalization of a more than 40-year reactor facility operation experience, and transfer of the knowledge and skills in carrying out world-class technological and experimental work to personnel trained for operation of NPPs with fast reactors.

## History

The idea of building a fast-neutron nuclear fuel breeding reactor was conceived by Academician A.I. Leypunskiy, the scientific supervisor at IPPE. Together with O.D. Kazachkovskiy, the head of laboratory at IPPE, he proposed that an experimental reactor, BOR-60, to be built to validate the major process and design approaches for fast power reactors.

In 1964 a decision was made by the USSR Council of Ministers to build the BOR-60 reactor, and the reactor design effort was launched at OKB "Gidropress".

The RIAR site near Melekess, now Dimitrovgrad, was allocated for the reactor facility construction by the Ministry for the Medium Machinebuilding.

The construction of the reactor building was begun in the summer of 1965.

The first criticality was achieved at the reactor facility on 30 December 1968 without sodium loaded into it.

28 December 1969 saw the facility energy startup with heat removal to a sodium-air heat exchanger.

*A.I. Leypunskiy*

The integrated BOR-60 reactor facility was accepted for operation by a Government Commission on 28 December 1970. The reactor facility has been operated in full scale with power output to the grid.

## Personalities

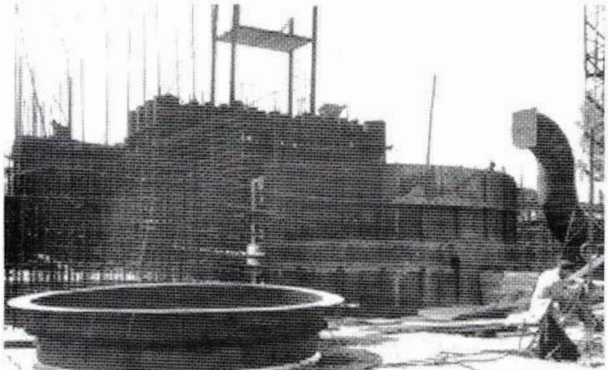
The scientific and construction work supervisor for the BOR-60 reactor was O.D. Kazachkovskiy, the director of RIAR.

The reactor design was developed by OKB "Gidropress" (Chief Designer V.V. Stekolnikov, Deputy Chief Designer S.M. Blagovolin, Head of Bureau B.I. Lukasevich).

The BOR-60 General Architect Engineer was VNIPIET in Leningrad. The project's Chief Engineer was M.L. Barskiy.

N.V. Krasnoyarov, Scientific Director of RIAR, and A.V. Smirnov, the first head of the facility, were in charge of organizing the activities for achieving first criticality and carrying out its energy startup.

*Launch of the BOR-60 site work (from left: I.V. Dmitriev, A.M. Smirnov, L.M. Levin, a construction worker, B.N. Nechaev, T.A. Yemelyanenko)*



*BOR-60 construction site*

The operation of the facility was carried out under the leadership of A.S. Korolkov (1987–2008), V.N. Marashev (1991–2008) and Yu.M. Krasheninnikov (BOR-60 Chief Engineer since 2008).

*O.D. Kazachkovskiy*

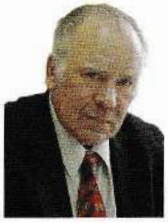
*V.V. Stekolnikov*

*S.M. Blagovolin*

*B.I. Lukasevich*

*N.V. Krasnoyarov*

*A.M. Smirnov*



*A.S. Korolkov*

*V.N. Marashev*

## Contact person

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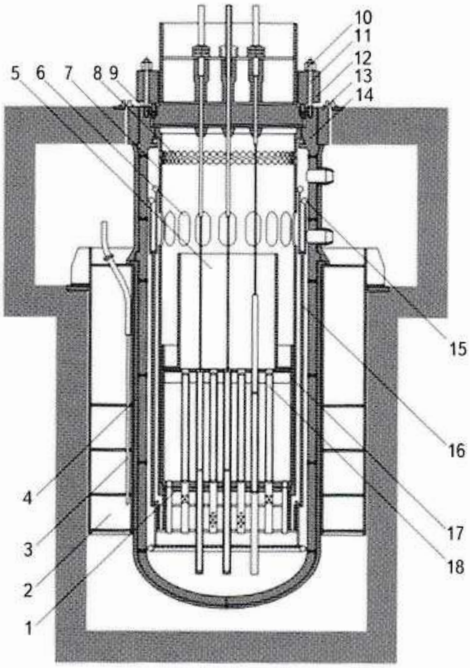
# VK-50 BOILING WATER-COOLED WATER-MODERATED REACTOR

The VK-50 reactor facility is a nuclear system with a pressure vessel boiling water-cooled water-moderated reactor and natural coolant circulation, in which steam is supplied immediately from the reactor to the turbine. VK-50 is a prototype of an NPP (the electricity generation capacity is up to 50 MW). Its first criticality was achieved on 15.12.1964 and the energy startup of the reactor took place on 20.10.1965.

The VK-50 reactor facility generates and outputs electricity to the offsite grid and heat for heating RIAR's site.

Work is continuously under way to increase the reliability and improve components and parts of the VK-50 reactor for raising its operating safety.

The VK-50 reactor facility has an integral arrangement of the reactor components, in which the core and the water circulation circuit are contained in one vessel.




*Sectional view of the VK-50 reactor:*
*1 – lower plate; 2 – biological shielding tank; 3 – IC channel; 4 – thermal insulation; 5 – chimney; 6 – overflow windows; 7 – cavity; 8 – steam windows; 9 – sealing assembly; 10 – nut; 11 – stud; 12 – pressure ring; 13 – retainer ring; 14 – reactor vessel; 15 – feedwater headers; 16 – separation shell; 17 – barrel; 18 – FA*

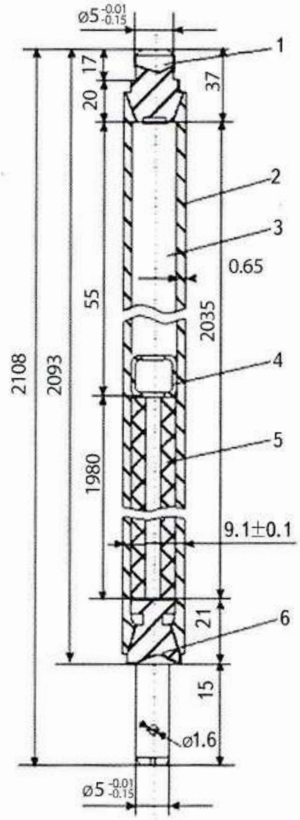
*VK-50 building*

## Main performance of the VK-50 reactor

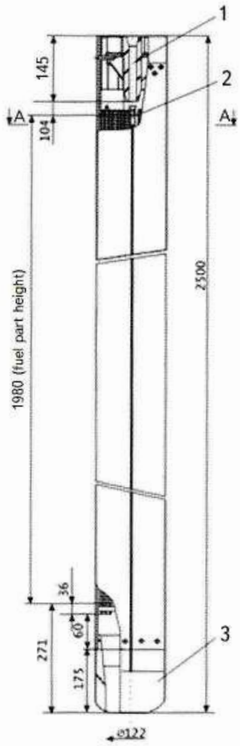
| | |
|---|---|
| Thermal power | Up to 200 MW |
| Electric power | Up to 50 MW |
| Heating station power | Up to 37 Gcal/h |
| Working pressure | Up to 6.0 MPa |
| Steam output | 300 t/h |
| Reactor outlet steam quality | Up to 15% |
| Specific core power | Up to 40 W/l |
| Heat flux, average | 0.29 MW/m$^2$ |

## Performance of the VK-50 reactor fuel

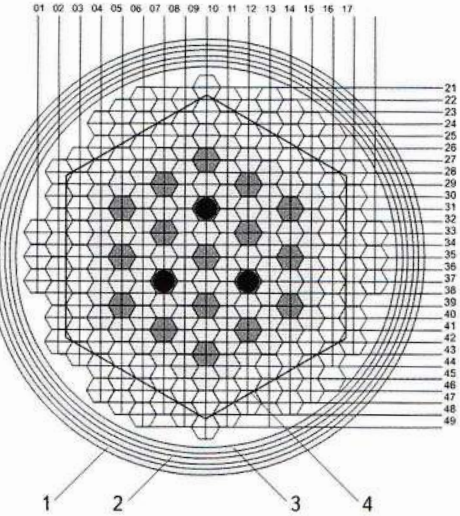
| | |
|---|---|
| FA width across flats | 176.5 mm |
| FA length | 2500 mm |
| FA shroud wall thickness | 1.5 mm |
| FA spacing | 185 mm |
| Fuel element dimensions (diameter×cladding thickness) | 7.58×0.65 mm |
| Fuel element length | 1980 mm |
| Number of fuel elements per FA | 138 |
| Core fuel | $UO_2$ |
| End screen material | Water |
| In-core fuel density, eff. | 40…45 kg/m$^3$ |
| Fuel weight in FA (active part) | 3000 g |
| Fuel burn-up in discharged FAs | 24…30 MW·day/kg |
| Core cycle length | 1 to 2 years |




*VK-50 FA fuel element:*
*1 – upper tip; 2 – cladding; 3 – gas gap; 4 – retainer; 5 – fuel pellet; 6 – lower tip*




*VK-50 FA:*
*1 – head; 2 – fuel element bundle; 3 – tail*



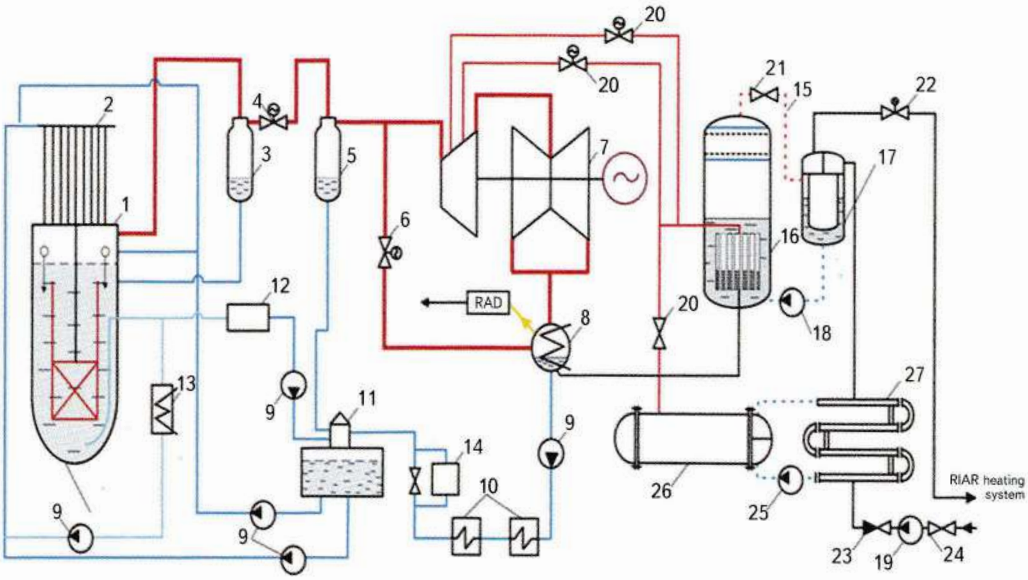

*VK-50 core map:*
*1 – reactor vessel; 2 – separation shell; 3 – well; 4 – upcomer section enclosure*

*– fuel assembly;*

*– shim rod assembly;*

*– emergency protection rod assembly;*

*– plugged vacant cells*




*VK-50 flowchart:*
*1 – reactor; 2 – CPS mechanisms; 3 – HP separator; 4 – control valve; 5 – LP separator; 6 – FAPCD; 7 – turbine-electric generator; 8 – condenser; 9 – pump; 10 – LP heaters; 11 – deaerator; 12 – water treatment system; 13 – heat exchanger; 14 – condensate polishing system; 15 – heating station intermediate circuit; 16 – evaporator; 17 – network heater; 18 – intermediate circuit pump; 19 – main-line pump; 20 – control valve; 21 – reheater steam line gate valve; 22 – main-line water outlet gate valve; 23 – check valve; 24 – main-line water inlet gate valve; 25 – intermediate water circuit pump; 26 – intermediate water circuit heat exchanger; 27 – main-line water reheater for intermediate water and steam circuits*

*VK-50 reactor (central hall)*

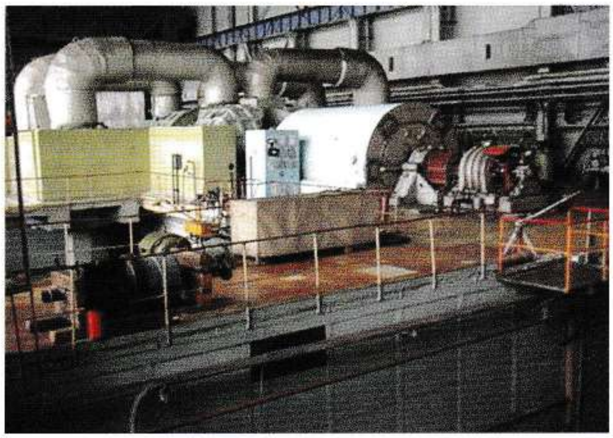
The VK-50 reactor vessel is a welded cylindrical vessel of the diameter 3550 mm and the height 11230 mm with an elliptical bottom. At the top, the vessel has a flat cover, on which there are the CPS drive and ionization chamber housings forming a multipurpose upper block. The vessel wall thickness in the nozzle area for discharge and supply of water and steam is 165 mm and that in the other portions is 105 mm.

There are 72 working fuel assemblies installed in the core. Core dimensions: efficient diameter 1.8 m, height 2.0 m.

The reactor core is cooled due to natural coolant circulation, i.e. based on a passive principle of action. The saturated steam generated by the reactor core comes sequentially into high-pressure separators, the throttling control valves and the low-pressure separators. The dried steam comes into the turbine. The waste steam is further dumped into the condenser and is pumped, now as condensate, into the deaerators, from where it is delivered to the reactor using feedwater pumps. Some of the feedwater goes to the reactor via the CPS actuator drives.

## Experimental capabilities

The specific features of a boiling reactor (lower values of the working pressure, heat flux and neutron fluence onto the vessel than in VVER-type reactors) make it possible to ensure high



*Turbine generator (turbine hall)*

engineering level of safety for the research conducted at the VK-50 RF through simple approaches.

The flowchart of the experimental nuclear facility enables problems of emergency conditions, water chemistry and gas modes to be addressed, as well as the serviceability and characteristics of different NPP components to be checked.

## Main areas of studies

Work is under way at VK-50 to support the nuclear cogeneration plant project including:

– FA service life tests to validate the design burn-up of up to 42 GW·day/t;

– tests of the level meter and reactor water level alarm prototypes;

– tests of sensors for the in-core power density distribution monitoring system and of pulse-current fission chambers for the reactor subcriticality monitoring system;

– service life tests of the CPS drives and absorbers;

– study of the in-vessel natural circulation and separation characteristics of the VK-50 reactor facility;

– acquisition of data for verifying the codes adopted for the VK-300 calculation with required experimentation.

Research programs are under way to:

– extend the reactor life to 60 years (until 2025);

– study the natural coolant circulation circuit with an expanded core and a retrofitted upcomer section;

– study the influence of feedwater temperature increase on the steam entrapment into the downcomer section and the resonant stability of the reactor.

## Main activities

The reactor utilization factor was 0.66 in 2011.

An expert review of the thermal-hydraulic analyses for the reactor upcomer section upgrading project has been completed.

The reactor core was analyzed, which has proved it possible to expand the core, reduce its resonant instability and increase the power output.

The extension of the reactor facility operation till 2025 has been validated.

Materials have been prepared for the NCGP construction in Dimitrovgrad, the Ulyanovsk Region.

The near-term plans include:

– conversion to six-row ("expanded") fueling of the VK-50 reactor core;

– renewal of the VK-50 operation license;

– integrated activities to extend the facility life to 2025, which will make it possible to operate installations in the field of atomic energy uses for up to 60 years.

## History of the VK-50 reactor facility

VK-50 is the first domestically built direct cycle pressure-vessel reactor, in which boiling water is used both as the coolant and as the moderator.

The design and construction of the VK-50 reactor were begun in 1956. The head designer of the reactor was OKB "Gidropress", the scientific supervisor for the project was RSC "Kurchatov Institute" and the general architect engineer was VNIPIET, St. Petersburg.

The design stipulated that the reactor facility would operate at the thermal power of up to 400 MW using a combined scheme with the direct supply of saturated steam immediately from the reactor to the turbine in the amount of up to 450 t/h with a pressure in the reactor vessel of 100 kg/cm$^2$ and the additional generation of 180 t/h of steam in three peripheral steam generators at a pressure of up to 30 kg/cm$^2$. Such power could be achieved thanks to the loading of the "larger core", which comprised 187 fuel assemblies.

The construction and erection work was completed during the period from October 1957 to September 1965. The energy startup of the reactor with its "smaller core", consisting of 65 FAs, took place on 20.10.1965. The reactor was officially put into operation on 15.12.1965.

In 1965 integrated commissioning activities were undertaken, as the result of which the peripheral steam generators (a test showed that the design thermal power of the reactor was achieved without these being in operation and with the facility operated in a direct cycle with the steam supply to the turbine set) and two isolation condensers (instead of these, a circuit for the reactor steam discharge to the turbine condenser via a pressure-relief and cooling device was introduced) were disconnected from the reactor. Some of the auxiliary systems were dismantled, and a more efficient radioactivity damper (RAD) was introduced instead of the surge gas volume holding gasholders.

In 1969-1974, the reactor was upgraded to reduce the in-core power peaking, intensify natural circulation and raise the efficiency of steam separation.

In 1974-1986, the reactor operated at the thermal power of 210 MW. In 1992, after the integrated activities were completed to improve the reactor safety, the working pressure was set at 6 MPa, the operating pressure parameters at 5 MPa and the power parameters at 200 MW. In 2007, the operating pressure parameters were set at 5.5 MPa.

Based on modern safety requirements, integrated activities were undertaken at the reactor to improve the reliability of its operation. All existing systems were brought into conformity with the effective regulatory documents of Gosatomnadzor.

On 11.04.1996 the VK-50 reactor facility was given the status of a research reactor.

## Personalities

In different periods of the VK-50 construction and operation, the work on the facility was led by Vitaliy Morozov, Vladimir Shchepetilnikov, Yury Solovyev, Yevgeny Kozin, Viktor Yeshcherkin, Nikolay Turtaev, and Aleksandr Kurskiy.


A.S. Kurskiy


A.V. Morozov


V.A. Shchepetilnikov


Yu.A. Solovyev


Ye.V. Kozin


V.M. Yeshcherkin


N.P. Turtaev

At present time, the reactor facility is directed by Vladimir Shirokov, the head of the reactor facility, and Dmitry Protopopov, chief engineer of the VK-50 reactor facility.

## Contact person

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# SM CRITICAL FACILITY, A PHYSICAL MODEL OF THE SM-3 RESEARCH REACTOR

First criticality was achieved at the SM critical facility on 16.01.1970. The facility's critical assembly is a physical model of the SM high-flux research reactor and offers capabilities for simulation of all its modifications (SM-2, SM-3). Within the core and the reflector, the geometrical dimensions and the material composition of the critical assembly are the same as in the SM reactor.

The critical facility comprises a control and protection system, a moderator filling system, a power supply system, a radiation monitoring system, a ventilation system, a fire alarm system, a distillate filling and makeup system, a water supply system, a heating system and a contaminated water discharge system.

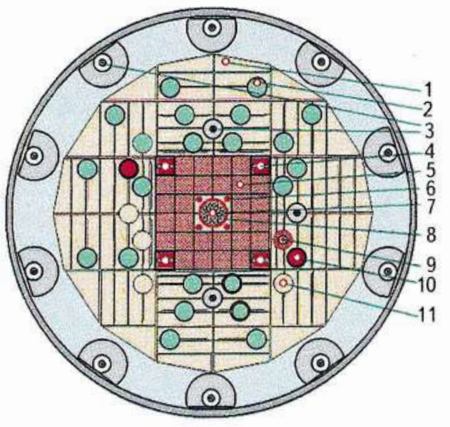
The SM core is a square of 420×420 mm (6×6 cells spaced at 70 mm). The four central cells are used as the central moderator space (CMS) (a neutron trap), and the four corner cells are used


*KS SM core reloading*

to accommodate shim members. Altogether, the core can accommodate up to 28 FAs (the total number of FAs reaches 32 when the shim members are introduced into the fuel suspensions area during the experiment). The four emergency protection (EP) members are at the CMS

## Main performance of KS SM

| | |
|---|---|
| Ultimate power level | Up to 20 W |
| Number of FAs in core | Up to 28 |
| Fuel | $^{235}U$ (cermet with $UO_2$ in a copper matrix) |
| Enrichment | 90 % |
| Moderator | Water |
| Reflector | Beryllium |




*SM CF core map:*
*1 – beryllium reflector blocks; 2 – experimental channel mockups; 3 – ionization chambers in dry channel; 4 – shim members; 5 – working FAs; 6 – EP rod in beryllium insert; 7 – CBTT mockup; 8 – central SM mockup; 9 – neutron source; 10 – AC rod simulators; 11 – beryllium plugs*

angles. The core height is 350 mm. The FAs have a square shape and the jacket diameter of 69×69 mm. Inside the jacket there are X-shaped fuel rods, which are helically twisted along the longitudinal axis. The FA design makes it possible to disassemble FAs into separate fuel elements.

Fuel elements with the $^{235}U$ content from 100 to 40% of the rated value are used for fuel burnup simulation.

The processes of the fuel poisoning with fission products are simulated by introducing absorber elements among the fuel elements. Some of the fuel elements may be removed to arrange experimental channels. Different experimental devices can be inserted into the core cells instead of FAs.

The central moderator space of the KS SM core consists of four beryllium inserts and a device for the accommodation of irradiated targets.

*Overall view of SM CF*

The assembly of four inserts forms a cylindrical cavity of the diameter 104 mm in the core center. Each insert has a longitudinal through hole, in which there is an emergency protection rod.

The CPS members have the same design as the SM reactor CPS. An EP rod is an assembly of absorber elements of the diameter 20 mm and consists of an absorber and a displacer. An EP member can be used with an aluminum or beryllium displacer. A shim member consists of two parts. The upper part is a square-shaped assembly made up of absorber elements based on europium oxide and the lower part is a dismountable FA.

## Experimental capabilities

Having the critical facility's core components fully accessible offers unique research capabilities.

The SM critical assembly can be used to simulate all required arrangement options for the SM reactor's main experimental device (the neutron trap in its central moderator cavity) in the environment, which depend on the tasks of experiments. A separator structure out of 27 zirconium tubes of the diameter 14×0.5 mm can be used as the trap's structural element to position targets with water in gaps or a beryllium cylindrical block of the diameter 93 mm and the height 500 mm with 27 longitudinal holes arranged on three radiuses.

There is a reflector around the SM critical assembly core, which is composed of 48 beryllium blocks with the dimensions of 210×100×500 mm. The blocks have vertical cylindrical holes for the insertion of 30 experimental channel mockups and two reactor automatic control rod simulators. The coordinates of the holes in the reflector blocks match the reactor coordinates.

The experiments on the critical facility define the reactivity effects during refueling and of experimental devices and structural elements, the efficiency of CPS rods, core power density distributions, and distributions and spectral characteristics of neutron fluxes in the core and in experimental devices.

## Main areas of studies

The SM critical assembly is used for:

– nuclear safety justification of operation;

– neutronic characterization of the experimental channels and devices;

– selection of means to shape irradiation modes and harmonize the preset modes for testing experimental devices irradiated simultaneously in the reactor;

– studies to substantiate the reactor core upgrade concepts and the adopted designs;

– experimentation to try out procedures for calculating the reactor neutronic characteristics;

– personnel training.

In a longer term it is planned to:

– experimentally validate phase 2 of the reactor core retrofit for conversion to fuel elements with a low harmful neutron absorption;

– validate the neutron trap rearrangements;

– validate the replacement of the neutron absorber in CPS members.

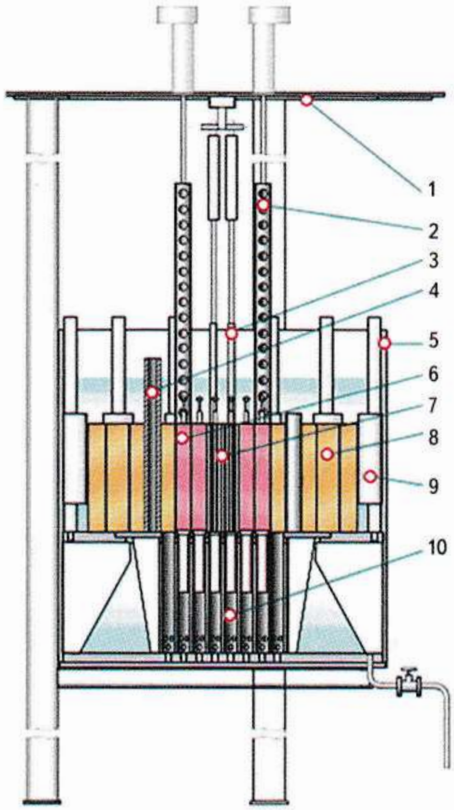
## Main activities

The following activities were conducted at the SM critical assembly:

– the feasibility of forming new high-flux irradiation volumes in the reactor core was experimentally demonstrated;

– redesign of the EP and AC members was experimentally validated;

– test experiments were conducted to adjust calculation procedures for the reactor neutronic characteristics.




*SM CF design:*
*1 – CPS drives platform; 2 – shim member guide tube; 3 – EP rod; 4 – experimental channel mockup; 5 – experimental tank; 6 – working FA; 7 – central moderator cavity; 8 – beryllium reflector block; 9 – ionization chamber channel; 10 – support grid assembly*

The accomplishment achieved so far are as follows:

• the key measurement point in the nuclear material accounting and control system was equipped with up-to-date measuring devices;

• the method for laying out the core, including the formation of additional irradiation volumes in the core, was justified based on experimental results and practically introduced during the reactor operation;

• the new design of the EP and AC members was selected and implemented based on the investigation results;

• a TV process surveillance system was introduced.

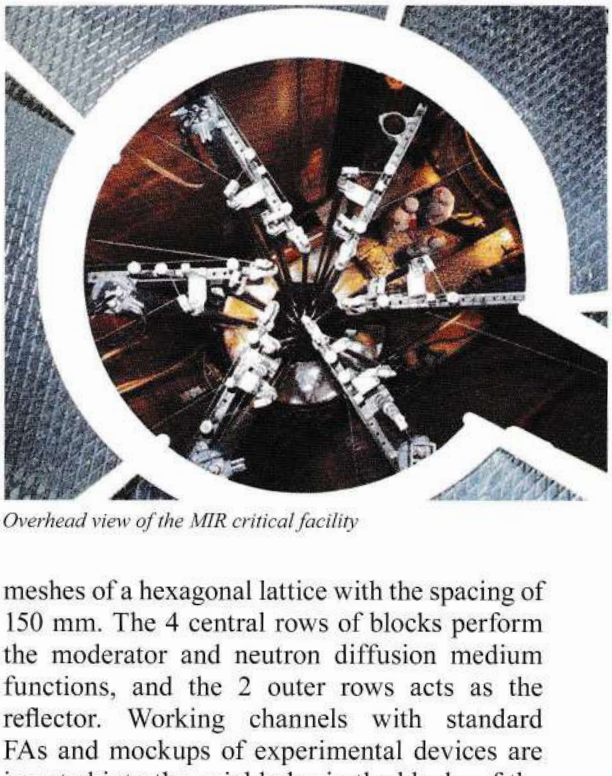
# MIR FULL-SCALE PHYSICAL MODEL OF THE MIR.M1

First criticality was achieved at the MIR critical facility on 07.10.1966. The critical facility is a physical model of the MIR.M1 material test reactor. Within the core and reflector, the geometrical dimensions and the materials of the critical facility are the same as in the MIR.M1 reactor.

The MIR critical facility includes a control and protection system, a moderator filling system, a power supply system, a radiation monitoring system, a ventilation system, a fire alarm system, a distillate filling and makeup system, a water supply system, a heating system and a contaminated water discharge system.

The facility's critical assembly and systems have been redesigned extensively during the operation both to bring the design into conformity with the changing layout of the MIR reactor core, and to meet the regulatory safety requirements. The specified service life of the critical facility is until 2015.

The critical assembly's core and reflector are accommodated in an experimental tank filled with distilled water. The core and the reflector are made up of hexagonal beryllium blocks of the width across flats 148.5 mm and the height 1100 mm. The blocks are installed in the

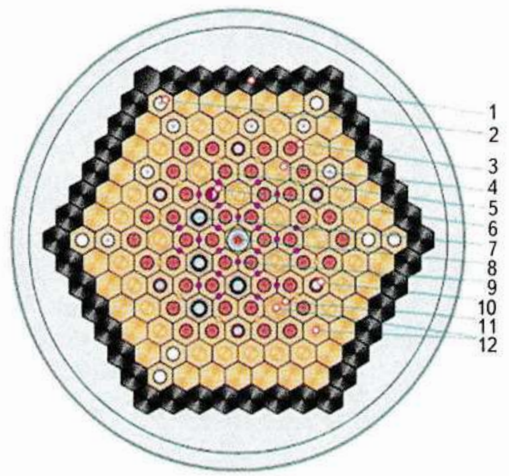


*Overhead view of the MIR critical facility*

meshes of a hexagonal lattice with the spacing of 150 mm. The 4 central rows of blocks perform the moderator and neutron diffusion medium functions, and the 2 outer rows acts as the reflector. Working channels with standard FAs and mockups of experimental devices are inserted into the axial holes in the blocks of the first four rows.

The inner reflector layer material is beryllium. The outer reflector layer consists of a row of continuous aluminum-clad graphite blocks.

As provided by the structure of the reactor core, there are 12 cells in the core to insert mockups of experimental devices in. These are arranged in the stack's rows 2 and 3 such that each is surrounded by 6 cells with working channels.




*Cross-section of the MIR critical assembly core:*
*1 – graphite blocks; 2 – ionization chamber suspension in dry channel; 3 – beryllium blocks for working channel; 4 – beryllium blocks for loop channel; 5 – working channels with FAs; 6 – extra-load channel; 7 – loop channels; 8 – neutron source; 9 – CPS rods in guide tube; 10 – immovable absorbers; 11 – beryllium sleeves; 12 – beryllium plugs*

## Main performance of the MIR critical facility

| | |
|---|---|
| Maximum thermal power | 5 W |
| Number of working FAs in the core | Up to 42 |
| Fuel | $^{235}U$ (cermet with $UO_2$ in an aluminum matrix) |
| Fuel enrichment | 90 % |
| Neutron moderator | Water and beryllium |
| Reflector | Beryllium and graphite |

Working FAs of 6 and 4 fuel elements are used in experiments at the critical assembly. An FA of coaxially arranged circular fuel elements is a fuel assembly of the old design used earlier in the reactor. Where required, standard MIR reactor FAs of a modern design, with 4 circular fuel elements each, are used in experiments. The dimensions of the 4 outer fuel elements for both assembly types are identical. Each fuel element is a three-layer tube (a fuel layer has a cladding of an aluminum alloy on both sides).

The fuel burn-up in the working FAs is simulated with a reduced content of $^{235}U$.

The CPS members have the same design and arrangement as those in the reactor. The working CPS members (6 EP and 18 shim members) move inside guide tubes inserted in the holes at the core stack block joints. The EP and shim members have an identical design and have the form of rods (diameter 24 mm) comprising four stacked sections. The length of one section is 550 mm with hinged joints between the sections. The two upper sections form the absorber (30 % $B_4C$ and 70 % Al, the cladding material is steel), and the two lower sections form a beryllium-metal displacer clad in stainless steel.

The design also includes up to 3 extra-load compensators installed in the core.

## Experimental capabilities

Mockups of the MIR.M1 reactor loop channels are accommodated in the critical assembly core, into which experimental FAs (or simulators thereof) are inserted for investigations prior to the experimental FA loading into the reactor. The loop channel mockups simulate the physical parameters of the reactor loop channel materials radially and at the end reflectors. In terms of design, the loop channel mockups form a set of coaxially arranged metal tubes attached to each other. The size of the tubes, the materials thereof and the gap between the tubes depend on the in-pile investigation tasks.

The experiments at the critical assembly are used to determine the reactivity effects during refueling, and of experimental devices and structural elements; the reactivity effects during the moderator withdrawal from or insertion into different cavities of experimental devices; the efficiency of the CPS members; the power density distribution in the core; the neutron flux distribution and spectral characteristics in the core and in experimental devices.

## Main areas of studies

The MIR critical facility is used for the following research activities:

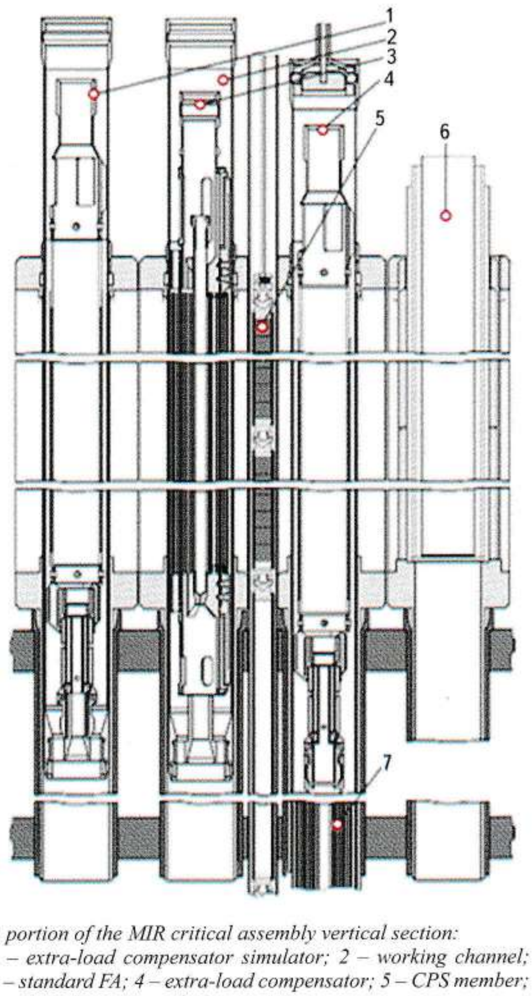
– nuclear safety analysis for the operation of the reactor;

– neutronic characterization of the experimental channels;

– selection of the means to generate irradiation modes and harmonize the preset modes for testing the experimental devices simultaneously irradiated in the reactor;

– experimentation to try out the reactor neutronic characterization techniques;

– personnel training.




*A portion of the MIR critical assembly vertical section:*
*1 – extra-load compensator simulator; 2 – working channel; 3 – standard FA; 4 – extra-load compensator; 5 – CPS member; 6 – loop channel; 7 – fuel suspension*

The research undertaken at the critical facility has helped with finding the physical conditions, which make it possible to achieve the preset parameters of irradiation in experimental devices and meet the nuclear safety requirements when carrying out the following activities, earlier not typical of the reactor earlier:

– experiments to simulate, for the tested fuel elements, the conditions which are typical of emergencies with power excursions;

– experiments to simulate the conditions, which are typical of loss-of-coolant accidents;

– testing of fuel elements in cyclic power variation modes;

– batch production of radionuclides, primarily of $^{192}Ir$.

## Main activities

The MIR critical facility was used for:

– research on the nuclear safety substantiation for the reactor and its operating modes;

– selection of means to generate and harmonize the preset modes for testing experimental devices in the reactor;

– test experiments to adjust the reactor neutronic characterization techniques.

To support the experimentation program, scheduled activities are under way for keeping the components, systems and parts of the critical facility in a serviceable condition, as well as activities for technical certification and service life extension.

During the recent years of operation, most of the improvements to the critical facility's technical systems have been aimed at giving extra perfection to the MIR physical protection and the nuclear material accounting and control systems. The nuclear material storage facility has been retrofitted as part of a Russian-US cooperation program. The most important activities include the following:

– the key measurement point in the nuclear material accounting and control system has been equipped with up-to-date measuring devices;

– a TV process surveillance system has been introduced;

– the shim and EP members to be installed in the reactor as part of the scheduled replacement of all worn out CPS members have undergone incoming inspection and efficiency determination.

The experiments on the MIR critical facility have helped choosing the conditions for safe performance of new classes of experiments.

In recent years, the major problem in operation of the critical facilities has been the shortage of skilled personnel.

**Near-term plans**

Maintenance of the MIR critical facility in a serviceable condition while extending timely the service life of its major components.

Introduction of new nondestructive measurement methods in the nuclear material accounting and control system.

Experimental research into the characteristics of the MIR.M1 experimental devices.

Further use of the critical facility is expected to place emphasis on testing of simulation codes and more detailed neutronic characterization of the reactor.

## Personalities

Both the SM facility and the MIR facility are attended by the same personnel. The scientific supervisor for the experiments on the SM and MIR critical facilities is Andrey Malkov, the head of the nuclear safety department of RIAR's reactor research complex.

*A.F. Yevseev*

Another person deserving special note is Aleksandr Yevseev, the SM and MIR critical facility mechanic, who has a service record of over 45 years.

## Contact person

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