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On April 12, 1943, Vice President of the USSR Academy of Science Academician A.A. Baikov signed an order on setting up Laboratory No. 2 of the USSR AoS, which later on grew to become the Kurchatov Institute. I.V. Kurchatov was appointed as Head of the Laboratory, the main purpose of which was to create nuclear weapons.

"I am happy that I was born in Russia and devoted my life to the atomic science of the great country of Soviets. It is my profound belief and firm knowledge that our people, our government will use the achievements of this science for no other purpose but the benefit of mankind".

Institute building

The Russian Scientific Center "Kurchatov Institute", the status of which has been recently changed to the National Research Center "Kurchatov Institute" (NRC KI), has unique experimental capabilities, including: 6 nuclear research reactors in service; a complex of hot cells; 14 critical facilities for studies of reactor neutronics; thermal, thermoelectric and electrochemical facilities; a facility for investigation of deep neutron and gamma-quanta penetration into material, and other experimental equipment.

Nuclear research facilities are largely found at the Institute of Reactor Materials and Technologies and at the Institute of Nuclear Reactors.

Development of the NRC KI experimental capabilities

On September 28, 1942, Chairman of the USSR State Committee of Defense I.V. Stalin signed an order to start work on an atomic bomb. To do so, it was decided to set up a scientific atomic weapons center in Moscow, and on April 12, 1943, Laboratory No. 2 led by I.V. Kurchatov was instituted by the resolution of the Presidium of the USSR Academy of Science. Design of an atomic pile (as nuclear reactor was referred to at that time) and an atomic bomb necessitated data on the nuclear, physical and chemical properties (constants) of uranium, plutonium and other elements. To this end, a number of experimental facilities had to be built.

The ciphered designations of the laboratory buildings have long been abandoned. Today, the buildings are numbered. Former building "K" ("Montazhka" – "Assembly Nook"), for example, constructed to P.I. Sidorov's design in 1946 (with its height increased in 1949), is now Building 30, and it is here that the first Soviet atomic reactor F-1 was installed. In fact, it was the first building in the country that was designed especially for accommodation of a new device – an atomic uranium pile with its potential danger of a nuclear explosion and radiation release. That is why the building had an underground 10 m-deep canyon where graphite blocks with the total weight of 400 t and uranium rods (50 t) were put together to make the pile.

On December 25, 1946, the F-1, first nuclear reactor in Europe and Asia, reached criticality and was brought by I.V. Kurchatov to the power level of several hundred watts. The reactor was used for vitally important experiments which underpinned the design of the large (100 000 kW) uranium-graphite reactor "A" – plutonium generator brought into operation in 1948. The "grandfather" of Russian reactors, F-1, certified as a standard neutron source and having the status of a monument to Russian science and engineering, is still in service.

To study the stability of nuclear processes in water-cooled, water-moderated reactors of ship-borne power systems and to optimize the biological shield design for the first Soviet nuclear submarine, a 300 kW tank reactor (VVR-2) was built on the territory of the Gas Plant in July 1954; it operated till 1983. In 1960, the 300 kW organic-cooled OR reactor was installed in the same building to the design of Yu.N. Aleksenko. It served as a basis for development of the

Nuclear research facilities of NRC KI

Type	Name	Thermal power, kW	First criticality year	Status	Operation time, years*
RR	F-1	24.00	1946	In operation	66
RR	Gidra	10.00	1971	In operation	41
RR	Argus	20.00	1981	In operation	31
RR	Gamma	125.00	1981	In operation	31
RR	IR-8	8 000.00	1981	In operation	31
RR	OR	300.00	1988	In operation	24
CF	UG	...	1965	Safe storage	23
CF	SF-1	0.10	1972	In operation	40
CF	Efir-2m	0.10	1973	In operation	39
CF	SF-7	...	1975	In operation	37
CF	Grog	0.10	1980	In operation	32
CF	Astra	0.10	1981	In operation	31
CF	RBMK	0.02	1981	In operation	31
CF	Narciss-M2	0.01	1983	In operation	29
CF	Delta	0.10	1985	In operation	27
CF	V-1000	0.20	1986	In operation	26
CF	P	0.20	1987	In operation	27
CF	Kvant	1.00	1990	In operation	22
CF	SK-fiz	0.60	1997	In operation	15
CF	Aksamit	...	2002	In operation	10

** As of 2012*

750 kW ARBUS reactor at NIIAR – a prototype of the energy source for the station in Antarctica. At a later point, the reactor was upgraded, with its coolant replaced with water, and returned to service in 1989.

In 1973, a critical assembly with the name of Efir was brought into operation to support the design of the production reactor Ruslan.

The water-water pool-type reactor IRT-1000 was built in 1957 to meet the growing needs of scientists at the Institute of Atomic Energy and some other Moscow research institutes for neutronic, materials and biological research, as well as for probation of specialists from Soviet republics and foreign countries. Its first startup at low power was performed by High Commissioner for Atomic Energy of France and well-known physicist F. Perrin. After upgrades in 1965, the reactor power was raised to 4 MW, and it received the name of IRT-M. In 1981, it was retrofitted and brought into operation at 8 MW under the name of IR-8.

The simple design, reliability and good experimental capabilities of the IRT and VVR facilities paved the way for about 20 reactors of these types built by the mid-1960s in the USSR and abroad.

During the 1960-1980s, the experimental facilities of the Institute were extended by provision of critical facilities for modeling the reactor cores of nuclear-powered ships; critical facilities for studying the cores of power reactors (of the VVER type); the RNM zero-power reactor; several thermal engineering facilities for studying heat transfer in fuel channels of power reactor facilities (including the large KS, TPZ and KSB facilities); a large uranium-graphite facility (UG) for neutronic studies of production and power reactors (RBMK); a water-water reactor facility (Gamma) with thermoelectric conversion of heat to electricity (thermal power rated at 200 kW and electric power of 5 kW); critical facilities Delta and RBMK.

Seeking to develop the line of nuclear power systems for aircraft and space vehicles, the Institute built a "hot" neutronic facility (FR-100) in 1955 to study the characteristics of a uranium-beryllium reactor for aircraft; in 1961–1963, an annex was added to Building 116 (facility R) for a high-temperature thermoelectric conversion power reactor called Romashka ("Cammomile") – a prototype of a space nuclear power system. When Romashka was finally dismantled, its place was taken by a facility for ground tests of a space power system with a thermionic converter Yenisei (Topaz-2). In 1970, critical facility Narciss was commissioned, which was a neutronic model of the Yenisei reactor with single-element thermionic channels. For the purpose of neutronic studies on a space NPS with multi-element thermionic channels, a critical assembly (RP-50) was built into the Aksamit facility found in the same building. Here, in 1980, another critical facility – Astra – was installed to model and study the cores of high-temperature reactors under development.

Development of high-energy engineering was supported by construction of the pulsed solution-type reactors Gidra and Argus, which are in operation to this day. As science and technology progressed along these lines, new experimental facilities (Grog, Iskra) appeared and hot cells were provided for tests.

Large physical facilities (V-1000, P, SK-fiz and others) were built for model-based studies of the cores for nuclear power systems under development.

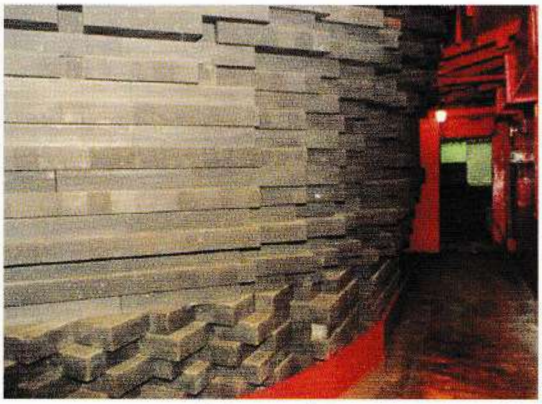
URANIUM-GRAPHITE REACTOR WITHOUT FORCED COOLING – F-1

F-1 is a uranium-graphite research reactor requiring no forced cooling. It is used as a standard thermal neutron source for metrological certification of detectors and neutron flux monitoring hardware for NPPs and other nuclear facilities as well as a tool for research in nuclear physics and engineering of nuclear facilities.

First criticality and first power operations took place on December 25, 1946.

In 1952, the F-1 was retrofitted: a control board and an operator console were manufactured, prefabricated I&C components were installed, servo drives were replaced and moved upstairs, to the CPS room.

The F-1 reactor building



F-1 reactor hall

The spherical core 6 m in diameter is made of graphite blocks measuring 100×100×600 mm, in which slugs of natural uranium are installed with a pitch of 200 mm. Among them, slugs of uranium metal 35 mm in diameter and 100 mm in length make a total of 36 t. Twelve tons of uranium oxide are distributed along the core periphery as compacts of various shapes (sphere, parallelepiped, etc.). The core center contains 81 blocks (50 kg) with fuel enriched to 2 % in ^{235}U. All the loaded uranium is unclad.

The F-1 reactor is controlled by four cadmium rods. Two emergency protection rods and one manual control rod are moved by servo drives in the vertical plane and one shim rod is driven by a hand winch in the horizontal plane.

The maximum reactivity margin of the F-1 core is 0.3 β. Heat is removed by natural circulation of air.

Main performance of F-1

Power, max. .... 24 kW

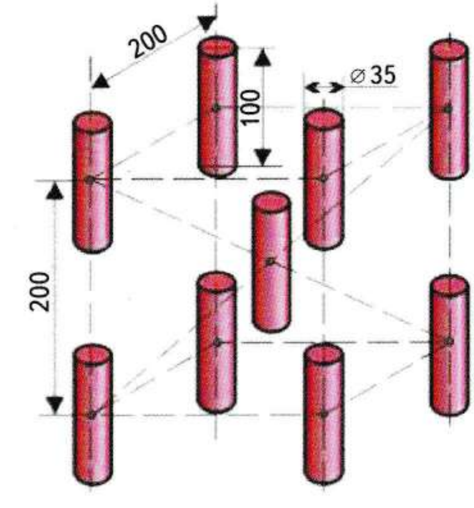
Moderator .... Graphite

Coolant .... Air

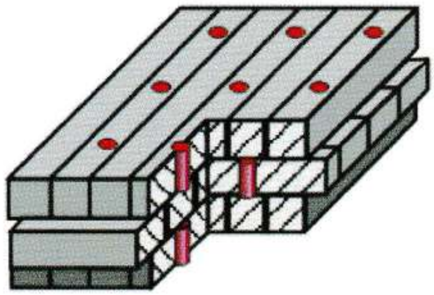
Reflector .... Graphite

Thermal and fast neutron flux, max. .....$6.13 \cdot 10^9$ $cm^{-2} \cdot s^{-1}$

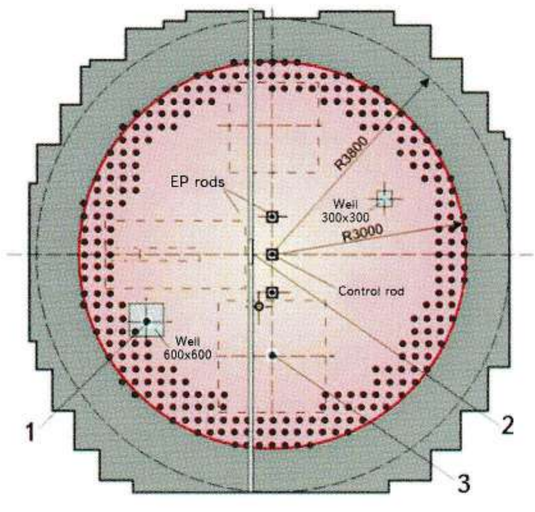
Intermediate neutron share .... 0.066+0.003



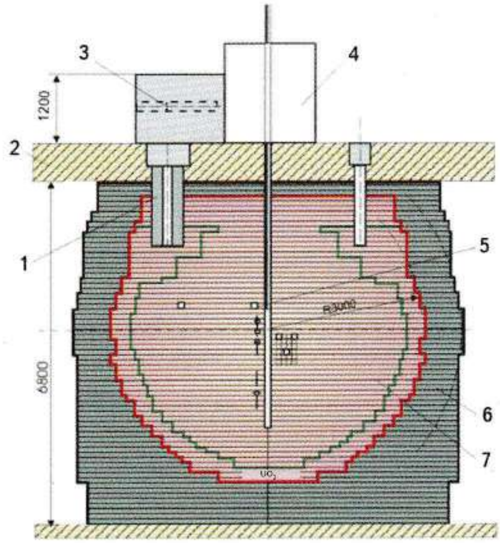
Core lattice configuration



A fragment of the core stack



Horizontal section of F-1:
1 – irradiation cavity in the calibration well, Ø 89 mm; 2 – irradiation cavity in the center of the horizontal channel; 3 – irradiation cavity in the cold assembly, Ø 89 mm



Vertical section of F-1:
1 – calibrating well; 2 – biological shield; 3 – irradiation cavity in the thermal column; 4 – cold assembly; 5 – irradiation cavity in the center of the vertical channel, measuring 100×100 mm; 6 – reflector; 7 – core

Experimental capabilities of F-1

Research tool	Neutron flux, $cm^{-2} \cdot s^{-1}$
Horizontal channel with a cavity measuring 100×100×600 mm	$6.13 \cdot 10^9$
Vertical channel	$0.97 \cdot 10^9$
Thermal column	$1.63 \cdot 10^7$
Cold assembly	$0.98 \cdot 10^5$

Main research areas

Certification and calibration of components and sensors designed for neutron flux measurement and monitoring at various nuclear facilities.

Main activities

Documents are prepared for further life extension.

According to theoretical calculations, the loaded uranium will last for 300 years of the F-1 reactor use as a standard neutron flux source.

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SELF-QUENCHING PULSED LIGHT-WATER REACTOR – GIDRA

Gidra is a pulsed homogeneous reactor whose fuel is water solution of uranyl sulfate. The reactor is designed for testing fuel elements of space nuclear power and propulsion systems and other types of nuclear reactors under conditions representing those of severe accidents, as well as for production of short-lived radioisotopes.

The Gidra reactor reached first criticality on May 5, 1971, and was brought to first power on November 24, 1972.

At present, the Gidra reactor is approaching the stage of planned retrofitting: preparations for its tank replacement are under way.

The reactor tank is a cylinder with the inner diameter of 392 mm. The tank is made of high-strength steel.

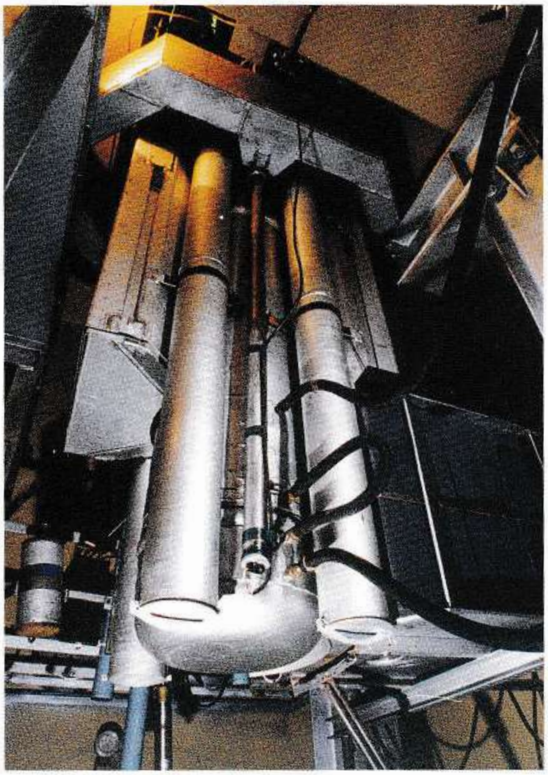
Five vertical tubes run through the tank lid. The central tube has the outer diameter of 127 mm. The other tubes are arranged symmetrically in a circle (with the radius of 95 mm), their diameter is 48 mm.

The central channel is a hollow cylinder containing a startup rod (a tube made of boron carbide), which provides the starting reactivity. A duralumin tube is inserted into the hollow of the startup rod. This tube forms the central research channel. Control rods of boron carbide move inside 4 peripheral channels.

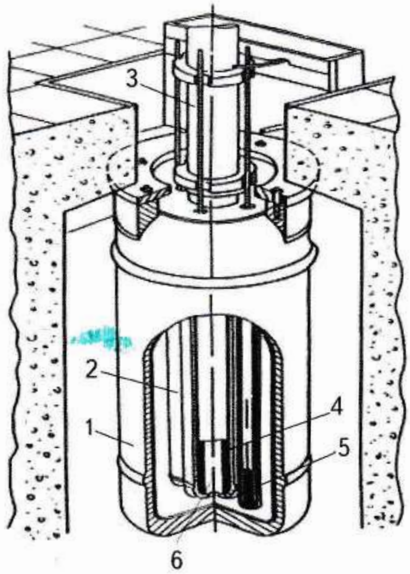
All tubes form thimbles where control rods travel.

A special holder located above the solution provides for free expansion of the tubes.

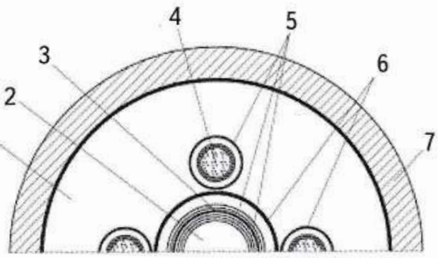
Four holes in the plug are meant for connecting the tank with the detonating gas burning system as well as for insertion of temperature and pressure sensors into the reactor. The detonating gas burns owing to heating of a platinum wire in the chamber connected with the tank space by a pipeline.



Central hall of the Gidra reactor



Gidra reactor in section:
1 – tank; 2 – control rod channels; 3 – pneumatic drive; 4 – startup rod; 5 – control rods; 6 – central research channel



Gidra reactor core in section:
1 – nuclear fuel; 2 – central research channel; 3 – startup rod; 4 – control rod; 5 – shrouds; 6 – coating; 7 – tank

Main performance of the Gidra reactor

Thermal power:

steady-state operation .... 10 kW

pulsed operation.. $2 \cdot 10^4$ MW

Nuclear fuel .... Water solution of UO_2SO_4

Enrichment in ^{235}U .... 90 %

^{235}U load: ....

critical .... 2.4 kg

working .... 3.2 kg

Reactivity margin.. 6 β

Pulse duration, minimum .... $2 \cdot 10^{-3}$ s

Neutron flux, maximum .... $8 \cdot 10^{17}$ $cm^{-2} \cdot s^{-1}$

Maximum flux during steady-state operation:

thermal neutrons .... $1 \cdot 10^{11}$ $cm^{-2} \cdot s^{-1}$

fast neutrons.. $1 \cdot 10^{12}$ $cm^{-2} \cdot s^{-1}$

Maximum flux during pulsed operation:

thermal neutrons .... $5 \cdot 10^{16}$ $cm^{-2} \cdot s^{-1}$

fast neutrons.. $5 \cdot 10^{17}$ $cm^{-2} \cdot s^{-1}$

Main areas of studies

– Studies on radiation resistance of components exposed to high fast neutron fluxes.

– Application of high-sensitivity activation analysis to short-lived radionuclides.

– Production of short-lived radionuclides.

– Studies and tests of nuclear fuel elements in transient and off-normal conditions.

– Studies of fast processes by the method of dynamic neutronography.

The Gidra reactor may be used for education and training purposes (it is 100 % safe in the event of human errors).

HOMOGENEOUS, SOLUTION REACTOR WITH A GRAPHITE REFLECTOR – ARGUS

Argus is the world's only stationary homogeneous reactor with fuel in solution, which serves to try out innovative processes of medical isotope production. The reactor reached first criticality on November 2, 1981, and was brought to first power on November 22, 1981.

The Argus control and protection system was upgraded in 1998.

The reactor facility occupies the following rooms:

– reactor room with walls (60 m^2) of concrete 2.2 t/m^3 in density;

– process room (10 m^2);

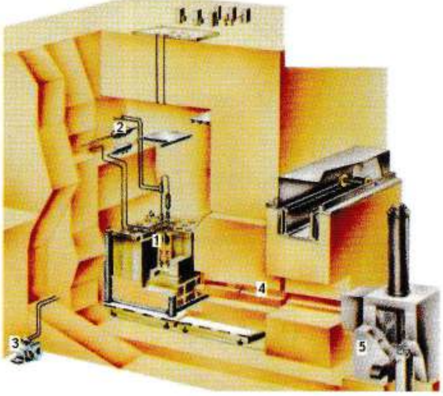
– control room (30 m^2).

The reactor components exposed to the fuel solution for a long time (tank, channels, coiled pipes), are made of stainless steel. A special study of this steel, and welded joints, showed its high corrosion resistance and non-susceptibility to intergranular corrosion.

The reactor tank – a welded cylinder with a hemispherical bottom – has the inner diameter of 305 mm, wall thickness of 5 mm, and cylinder length of 500 mm. Vertical tubes (with the inner diameter of 44 mm and wall thickness of 2 mm) are welded into the reactor tank lid. They form thimbles – one in the center and two at the periphery, the latter placed symmetrically in a circle of diameter 150 mm. The thimbles are immersed in liquid fuel to a maximum depth. Inside the reactor tank there is a cylindrical cooling coiled pipe.

On its sides and at the bottom, the reactor tank is surrounded by a graphite reflector no less than 450 mm in thickness. The facility is shaped as a parallelepiped with the base measuring 1500×1500 mm and the height of 1200 mm. The inside of the reflector can be made of beryllium. Such a combination improves the reactor neutronics. The outer side surfaces of the reflector are covered with a layer of borated polyethylene no less than 50 mm in thickness.

Argus core compartment



Argus reactor in section:
1 – core; 2 – gas exhaust device; 3 – cooling system pumps; 4 – horizontal neutron guide; 5 – neutronography system

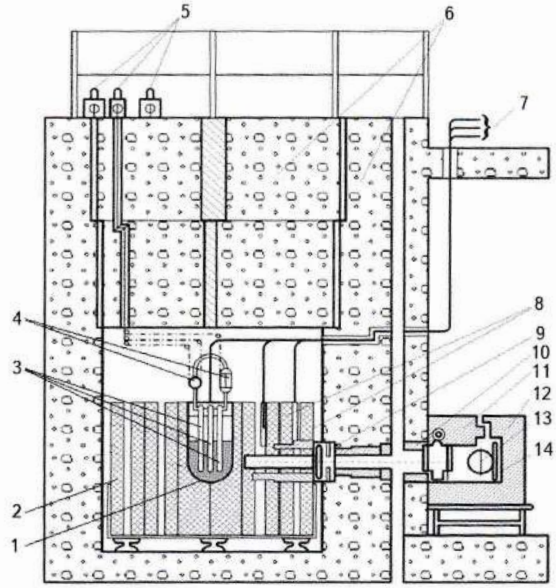
Main performance of the Argus reactor

Nominal power	20 kW
Fuel	Water solution of UO_2SO_4
Enrichment in ^{235}U	90 %
Concentration of ^{235}U	73.2 g/l
Solution volume	22 l
Thermal neutron flux:	
in the central channel	$5\cdot10^{11}$ cm$^{-2}\cdot$s^{-1}
in the reflector	$(1.0\ldots2.8)\cdot10^{11}$ cm$^{-2}\cdot$s^{-1}

Gaseous products of water radiolysis in the fuel solution are continuously regenerated in a special catalytic recombiner, which makes one sealed system with the reactor tank. This approach prevents release of fission products to the environment.

Two emergency protection (EP) rods have the form of boron carbide cylinders in steel sheaths. A tube of aluminum alloy is inserted into the inner space of an EP rod, forming a channel of diameter 27 mm for movement of a shim rod made of boron carbide (with the density of 1.3 g/cm^3) and having a 24 mm diameter. Such a combination of EP and shim rods is used in the peripheral channels of the reactor tank. Control rods (manual and automatic) travel in the space between the reactor tank and the reflector.

The solution reactor safety depends largely on the self-regulation properties provided by physical phenomena in the fuel solution, such as solution density decrease due to heating (temperature effect) and generation of radiolytic gas by fission products (void effect). The working uranium concentration in the solution provides a minimum critical mass and automatic shutdown of the reactor in the event of inadvertent increase or decrease of uranium concentration.



Argus reactor in section:
1 – tank with channels and cooling coiled pipe; 2 – graphite reflector; 3 – control members; 4 – catalytic recombiner; 5 – drives; 6 – biological shield; 7 – pneumatic channels (rabbit routes); 8 – research channels; 9 – beam-shaping device; 10 – collimator; 11 – gate; 12 – neutronography chamber; 13 – specimen; 14 – detector

Experimental capabilities

The reactor has the following research devices:

- horizontal channel;
- vertical channel;
- in-core channel;
- 7 channels in the reflector.

Thermal neutron flux is:

- $5 \cdot 10^{11}$ cm$^{-2} \cdot$s^{-1} in the central channel and
- $(1.0...2.8) \cdot 10^{11}$ cm$^{-2} \cdot$s^{-1} in the reflector.

Several tangential channels in the reflector shape the diverging neutron beams of various collimation degrees (10 to 60 angular minutes) for use in neutronographic studies. Through special penetrations in the biological shield, the beams arrive at the radiography chamber, which is equipped with remotely controlled mechanisms for positioning and moving the tested components and the detector unit. The chamber allows handling radioactive components and has an entry window measuring 200×400 mm.

Main areas of studies

The reactor is used for unique studies of nuclear and radiation safety of various nuclear power systems as well as for development of innovative methods for medical radionuclide production.

Together with the series of its shielded chambers, the Argus reactor can produce ^{99}Mo, ^{89}Sr, ^{131}I and other radionuclides in an environmentally safe and economically efficient way.

The reactor is used for:

- activation analysis of elements;
- radiographic analyses;
- non-destructive testing of components;
- production of nuclear filters;
- production of radionuclides;
- education and training.

WATER-WATER THERMAL-NEUTRON TANK REACTOR – GAMMA

Gamma is a tank water-cooled and water-moderated reactor with naturally circulating coolant, which is equipped with a thermoelectric converter and is a prototype of a small NPP.

The facility design is based on the three key principles, which later dictated the configuration of unattended self-regulated nuclear thermoelectric plants. These principles are:

– use of a water-water reactor with self-regulation of power as a heat source;

– heat removal by a cooling system with natural coolant circulation in the primary and intermediate circuits;

– thermoelectric conversion of heat to electricity.

The Gamma reactor reached first criticality on December 30, 1981 and was brought to first power on March 1, 1982.

By 1990, the reactor was converted to operation at lower parameters (power of 150 kW, pressure of 2 MPa and coolant temperature of 190 °C) for studies with the existing experimental devices.

This thermionic facility consists of a tank reactor, with the external pressurizer and the thermoelectric generator integrated into one unit.

The facility sits in a water pool, which serves the purposes of heat removal from the thermoelectric generator and radiation protection.

The intrinsic safety feature of the reactor relies on the negative temperature coefficient of reactivity over the whole operation range. The maximum reactivity introduced by experimental devices, specimens and components, is no greater than $\pm 0.3\ \beta_{eff}$.

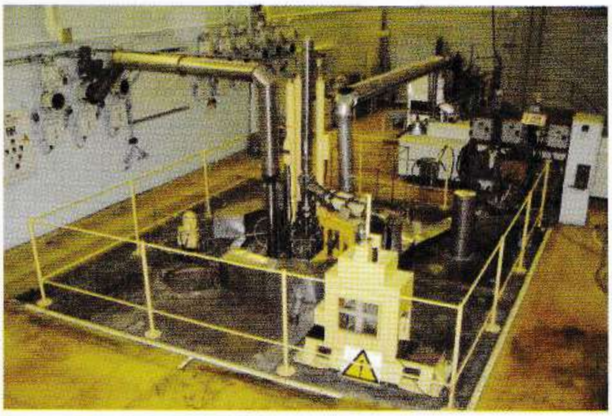
The Gamma reactor core measuring 0.55 m in diameter and 0.5 m in height accommodates 69 fuel assemblies comprising 1311 fuel elements of the VMK-12 type. Water is used as moderator, reflector and coolant.

Four collectors of the facility have 24 thermoelectric modules connected to them, which act together as a thermoelectric generator.

The electric energy produced by the thermoelectric generator is taken up by a resistor load system, and can be also transferred to the reactive load via an inverter.

Heat is removed by a pump-free system with natural coolant circulation in the primary and intermediate circuits.

The alkaline water chemistry of the Gamma facility was tried out in the primary circuit to suppress corrosion processes and water radiolysis, requiring no circuit cleaning or periodic adjustment. The secondary circuit has a neutral water chemistry which needs no correction.

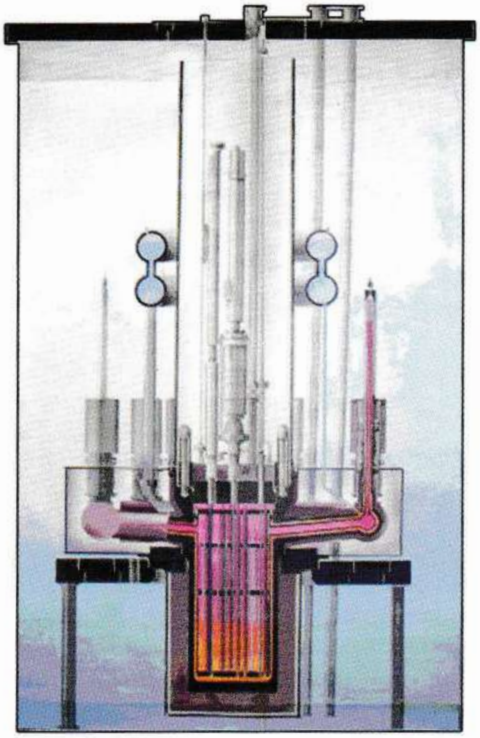


Central hall of the Gamma reactor

Main performance* of the Gamma reactor

Thermal power	250/150 kW
Electric power	5/2 kW
Fuel type	Uranium alloy
Enrichment	20 %
Coolant	Water
flow rate	10.8 m^3/h
pressure	17.5/2.0 MPa
temperature	330/190 °C
Moderator	Water
Reflector	Water
Cycle duration (with a nominal fuel load)	2 083 days

* *Design / current*



Gamma reactor in section

Top view of the Gamma reactor

Experimental capabilities

The reactor core houses the following process and experimental devices:

– "dry" shells for thermoelectric converters (thermocouples) which monitor primary coolant temperature at the core inlet and outlet;

– "dry" shells for in-core irradiation of gamma-neutron detectors;

– thermal channel – fuel channel modified to carry thermocouples for fuel and coolant temperature monitoring and power density sensors for online reactor power measurement to fulfill research objectives;

– probe with thermocouples distributed in the space between channels through the core height;

– corrosion test channel to study the resistance of various materials and to deal with other experimental jobs.

Main areas of studies

• Experimental testing of the concepts of unattended self-regulated nuclear thermoelectric plants under development.

• Try out of high-sensitive methods of automated computerized diagnosis allowing to detect a failure more early than it appears in determined routine signals.

• Widely applied noise analysis to detect variations of parameters as a tool for dynamic studies and early diagnosis of the facility condition.

• Experimental demonstration of the method with short-circuiting modules to maintain constant generator voltage under variable off-site loads.

Contact person

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POOL REACTOR IR-8

The IR-8 pool-type reactor is the result of upgrades at the IRT reactor which had been commissioned in 1957. IR-8 reached first criticality on August 12, 1981, and was brought to first power on October 30, 1981. IR-8 is used for studies in the fields of solid-state physics, nanotechnologies and nanomaterials, radiation materials science, nuclear physics, radiation chemistry, radiobiology, for tests of fuel composition specimens and production of various medical radioisotopes.

The configuration of the reactor core ensures high neutron flux parameters both in the core and in the reflector.

The IR-8 core is immersed in a water pool to provide biological shielding, to ensure safety of maintenance and handling operations as well as of the reactor facility itself, and to reduce accidental radiation release.

The reactor core contains 6-tube fuel assemblies of the IRT-3M type (with the option of using 8- and 4-tube assemblies of the same type).

Main performance of IR-8

Reactor power .... 8 MW

Number of fuel assemblies in the core 16

Core volume, ... 47.4 l

^{235}U mass in the core with fresh fuel .... 4.35 kg

Neutron flux, max.:

thermal:

in the core .... $1.5 \cdot 10^{14}$ cm$^{-2} \cdot$s^{-1}

in a replaceable beryllium block .... $2.3 \cdot 10^{14}$ cm$^{-2} \cdot$s^{-1}

at the end of the horizontal channel .... $1.1 \cdot 10^{14}$ cm$^{-2} \cdot$s^{-1}

at the outlet of the horizontal channel .... $1.8 \cdot 10^{10}$ cm$^{-2} \cdot$s^{-1}

fast (E > 3 MeV):

in the core .... $5.7 \cdot 10^{13}$ cm$^{-2} \cdot$s^{-1}

in a replaceable beryllium block .... $1.8 \cdot 10^{13}$ cm$^{-2} \cdot$s^{-1}

Annual output .... 15 000 MW·h

Utilization factor, average .... 33 %

Burnup of discharged fuel .... 52 %

Main building of the IR-8 reactor

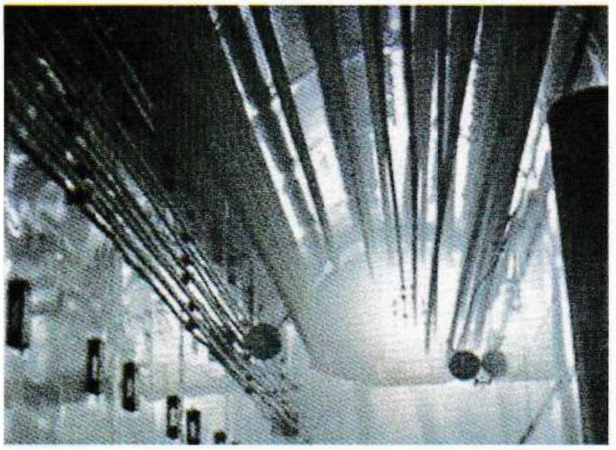
Central hall of IR-8

An IRT-3M assembly consists of tubes (fuel elements) of square cross-section. Fuel enrichment in ^{235}U is 90%.

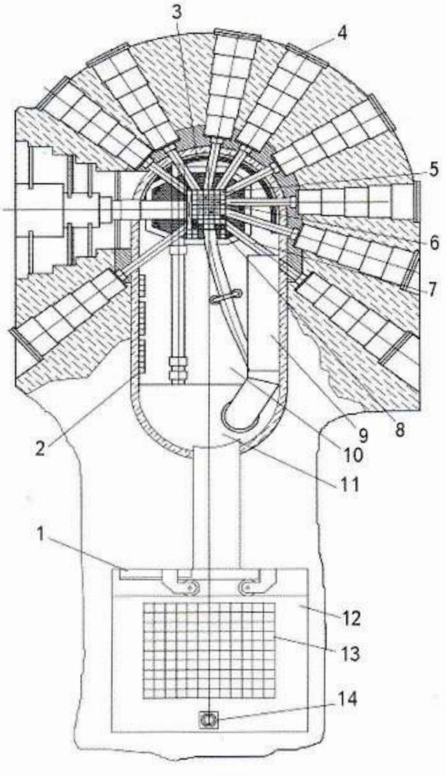
Fuel assemblies and replaceable beryllium blocks are mounted on a support grid 90 mm in thickness. At the top, fuel assemblies and beryllium blocks are aligned by means of lugs on their top end-piece.

The middle fuel layer is uranium dioxide in an aluminum matrix with the thickness of 0.4 mm and the height of 580 mm.

The CPS members are 13 rods: 2 for emergency protection (scram), 10 for shimming, and 1 for automatic control. The absorber of the CPS rods appears as boron carbide pellets in a sheath of stainless steel. The height of the absorber part of the rods is 600 mm.



IR-8 core in a water pool

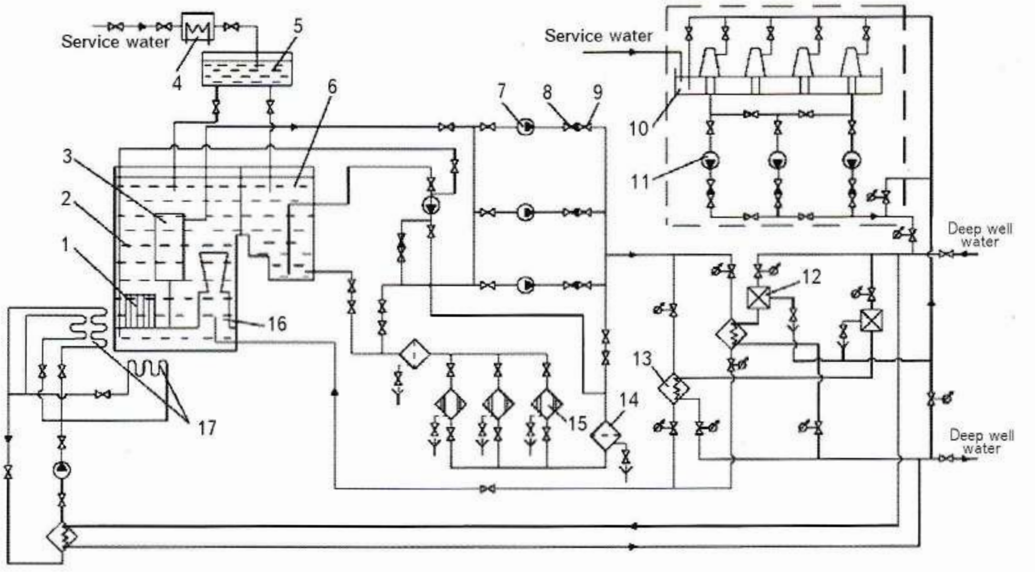


Cross-section of IR-8:
1 – lock gate; 2 – interim SNF storage; 3 – steel shield; 4 – shutter; 5 – replaceable beryllium block; 6 – fuel assembly; 7 – HEC; 8 – stationary beryllium block; 9 – retention tank; 10 – reactor pool; 11 – retention tank; 12 – storage pool; 13 – SNF storage; 14 – container receptacle for unloading spent fuel from the pool

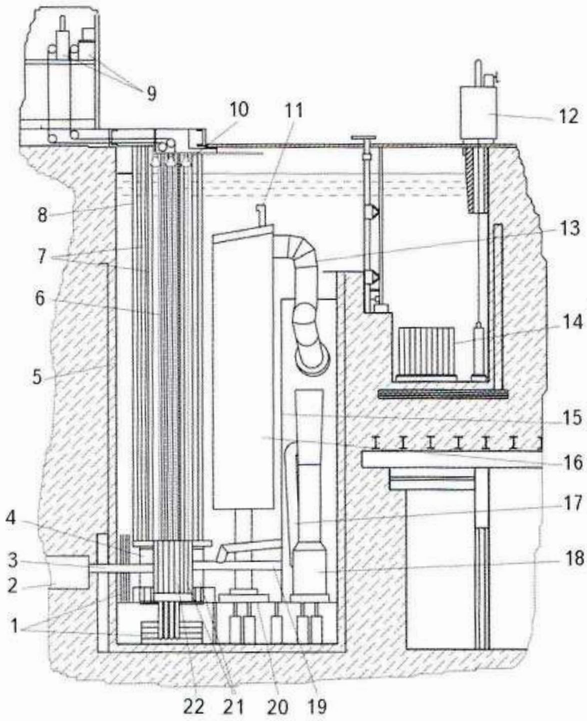
The reactor's heat transfer system has a two-circuit arrangement. The primary circuit consisting of reactor coolant pumps, heat exchangers, ion exchange filters, valves and pipelines between these components, removes heat from the reactor core and carries it to the recirculating water circuit. The primary water runs through fuel assemblies from the top down.

Experimental capabilities

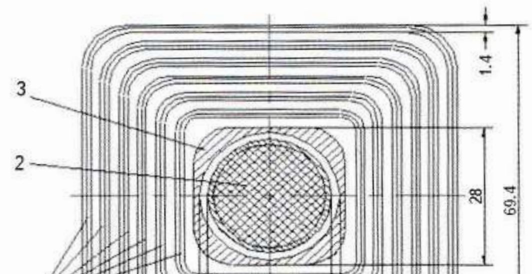
The reactor has 12 horizontal beam channels with equipment installed in them for neutron study of condensed media, and vertical channels (up to 42) where radioactive isotopes can be



IR-8 cooling circuit diagram:
1 – reactor core; 2 – reactor pool; 3 – retention tank; 4 – electric distiller; 5 – makeup tank; 6 – storage pool; 7 – primary pumps; 8 – check valve; 9 – valve gate; 10 – cooling tower; 11 – secondary pumps; 12 – secondary water filter; 13 – heat exchanger; 14 – mechanical filter; 15 – ion exchange filter; 16 – ejector; 17 – biological shield cooling coils



Longitudinal section of IR-8:
1 – steel shields; 2 – HEC shutter; 3 – HEC; 4 – reactor vessel; 5 – reactor tank; 6 – CPS rod channels; 7 – VEC; 8 – CPS channels; 9 – CPS rod drives; 10 – sprinkler; 11 – air vent; 12 – transport cask; 13 – suction pipeline; 14 – SNF storage; 15 – vertical partition; 16 – retention tank; 17 – pressure pipeline; 18 – ejector; 19 – ultracold channel; 20 – dividing bottom; 21 – beryllium reflector; 22 – fuel assembly

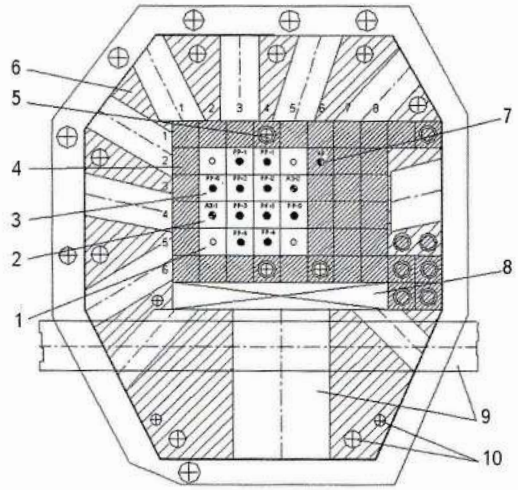


Cross-section of a 6-tube IRT-3M fuel assembly:
1 – fuel element; 2 – channel; 3 – CPS rod

produced and fuel and structural materials can be irradiated with online measurements.

Other facilities of IR-8 include:

– a complex of neutronic hardware for studies of the structure, phase transitions, heterogeneity of and flaws in materials, including various research devices mounted on horizontal



Core map:
1 – fuel assembly with a displacer; 2 – fuel assembly with an EP rod; 3 – fuel assembly with a shim rod; 4 – replaceable beryllium reflector block; 5 – replaceable beryllium block with a vertical experimental channel; 6 – stationary beryllium reflector; 7 – automatic control rod; 8 – lead shield; 9 – horizontal experimental channels; 10 – vertical experimental channels

channels of the IR-8 reactor with the average thermal neutron flux at channel ends equal to $\sim 5 \cdot 10^9$ $cm^{-2} \cdot s^{-1}$, such as:

- five-circle neutron diffractometer MOND;
- triaxial crystal spectrometer ATOS;
- polycrystal multidetector circular diffractometer DISK;
- triaxial perfect crystal-based spectrometer STOIK;

– a complex of hardware for neutron and gamma introscopy of high-tech materials and products (turbine blades, welded joints, fuel assemblies, etc.), including additional devices installed in monochromatic neutron beam traps of the above facilities. Gamma-ray chambers and Imaging Plates are used as neutron and gamma-ray detectors.

Main areas of studies

– Studies with polarized and ultracold neutrons.

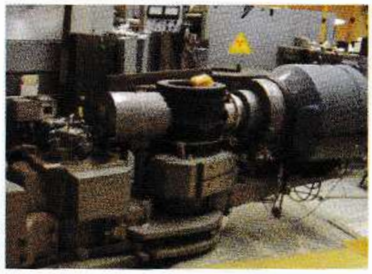
– Studies of gamma-ray and neutron interaction with nuclei.

– Conversion electron spectroscopy.

– Element analysis and medical biological research.

– Studies of inelastic neutron scattering by material.

MOND diffractometer



ATOS spectrometer

DISK diffractometer

– Studies of elastic neutron scattering by crystals.

– Input inspection of IRT-3M fuel assemblies by neutron and gamma introscopy.

– Development of methods and study of new materials in a wide range of pressures and temperatures.

– Studies of the structure, properties and flaws of non-irradiated and slightly activated reactor materials.

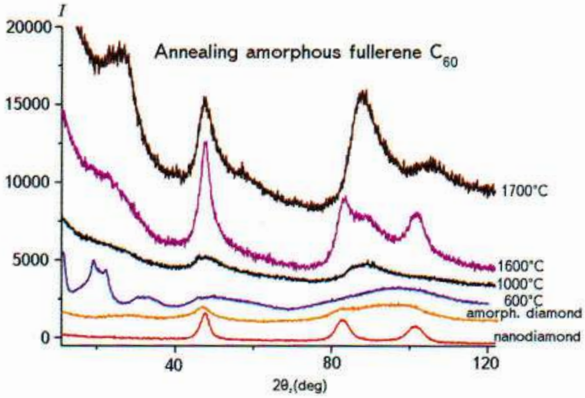
International cooperation

The IR-8 reactor is used in research activities in cooperation with Belarus, Latvia, Armenia and other countries.

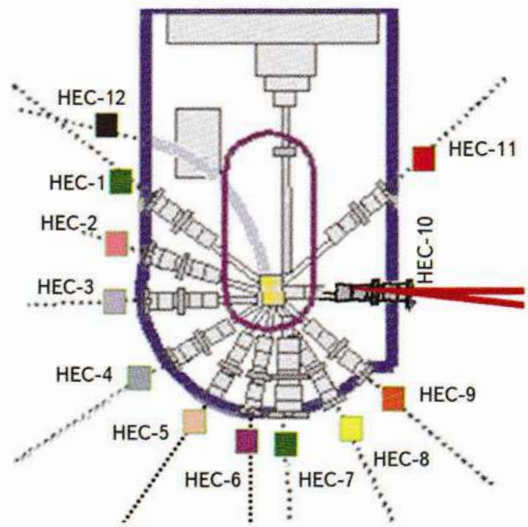
Main activities

Phase transitions were discovered in nanosize fullerites C_{60}, C_{70} and their mixtures at high temperatures.

Experiments were carried out to study the structure of and manufacturing flaws in mono- and polycrystalline materials of turbine blades, welded joints, micro- and macro-fuel compositions, composite and monocrystalline superconducting materials.



Polymorphic transition in amorphous fullerenes



Experimental hall layout:
HEC-1 – neutron microscopy; HEC-2 – nuclear spectroscopy; HEC-3 – fission physics; HEC-4 – single crystals (MOND); HEC-5 – excitation spectra (ATOS); HEC-6 – high pressures (DISK); HEC-7 – capillary optics; HEC-8 – neutron radiography; HEC-9 – small-angle scattering (STOIK); HEC-10 – inelastic scattering; HEC-11, HEC-12 – ultracold neutrons

In 2011, the IR-8 reactor approached the end of its specified 30-year service life, some of its systems grew old, and for its operation to continue, it became necessary to repair or replace:

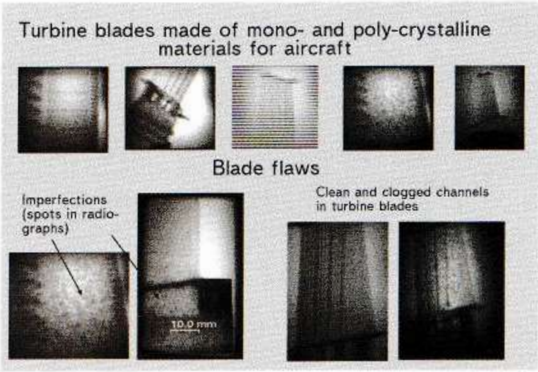
– heat exchangers, primary coolant pumps and cooling tower;

– control and protection system and instrumentation;

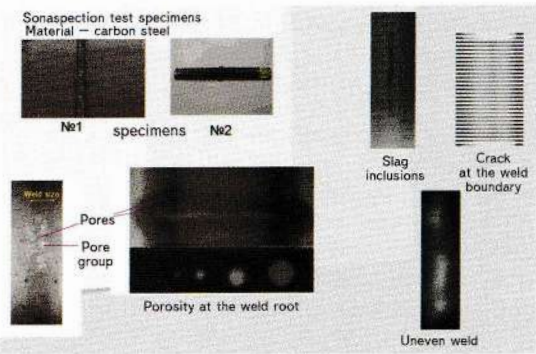
– radiation monitoring system;

– power supply system.

Work is under way at present to replace the heat exchangers, pumps and instrumentation in the primary circuit.



Flaws in aerospace products



Neutron inspection of flaws in welds

Personalities

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WATER-WATER HETEROGENEOUS POOL REACTOR – OR

Research reactor OR is part of the OR-M Complex which was put into operation in 1954 and had upgrades in 1964, 1981, 1989, and 2007. Its last rise to criticality took place on May 20, 1988 and to power, on October 6, 1989.

The special-purpose small (0.5 m long) nuclear reactor OR with short operating time and low residual gamma background is used as a source of neutrons and photons. It operates without core refueling in the unsteady mode of repeated relatively short startups and rises to a preset power level (from 0.01 to 300 kW) with an arbitrary on-off time ratio and radiation exposure of several hours.

The average annual power of the reactor approaches 3 kW, which results in a low level of fission-fragment activity accumulated in the core, low burnup and a refueling interval of at least 10–15 years.

The OR-M Complex with water-water reactor OR is intended for basic macroscopic research in transport of neutrons and photons with the reactor-range energy in shielding, core and structural materials produced with the use of state-of-the-art technologies; for studies, experimental development and tests of radiation shields for advanced space and other reactors as well as of components exposable to penetrating nuclear radiation.

Experimental capabilities

The OR-M Complex has an irradiation volume of 100 m^3 in the form of a niche in the reactor's radiation shield with a horizontal tunnel extending from it. The expanding four-stepped niche abuts on the side graphite reflector of the reactor core; its total length is 2 m and the cross-section varies from ~1.3×1.3 m to

~1.6×1.6 m. The tunnel 2.1×2.1 m in section and over 20 m long, with the concrete wall thickness up to 1.7 m has an installation opening in its middle part, shut off by a sliding shield.

The tunnel accommodates:

– a system of beam-collimating diaphragms remotely controlled to move on platforms along a rail track all the way through the irradiation space and to shut off its section;

– a set of neutron and photon filters made of various materials are placed along the beam travel path; with the beam passing through the filters, it is possible to vary widely the energy and composition of neutrons and gamma radiation at the point of specimen location.

The experimental hall measuring 150 m^2 in area and 13 m in height and communicating with the tunnel is used for preliminary assembly of the mockups to be studied as well as for reference measurements and experiments with the help of isotope radiation sources.

A bridge crane with the load-lifting capacity up to 20 t is installed in the hall for handling specimen pieces up to 2.5 m long in the tunnel and larger pieces in the hall.

There are two storage places for irradiated specimens 40 m^3 in capacity each; their separable shielding covers are components of the hall floor with the loading limit of 50 t.

A radiometry set, including sensors of spatial, energy and angular flux distributions of both radiation types, logs neutrons in the energy range from thermal to 15 MeV and gamma radiation – in the range from 0.1 to 10 MeV.


Central hall of OR-M

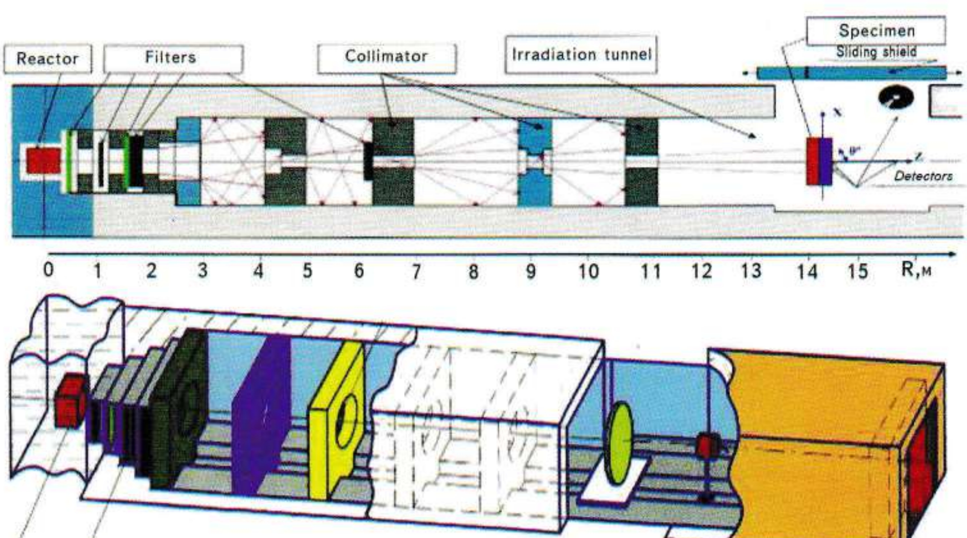
A special system monitors the radiation intensity of a beam, when upon leaving the reactor it falls on the shield under investigation.

Measurement control and logging are provided by a local area network of personal computers with the dedicated software.

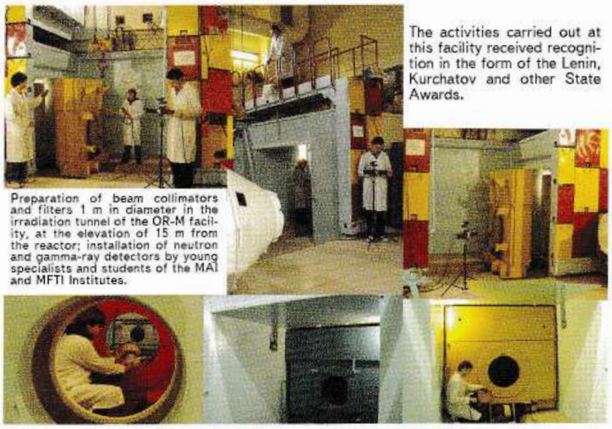
An up-to-date radiation monitoring, security and physical protection system makes it possible to handle various specimens and components, while keeping them safe and inaccessible to unauthorized persons.

Main areas of studies

Experimental development and testing of radiation shields for various nuclear reactors as



Neutron beam shaping diagram



Achievements of the OR-M personnel

The Institute leaders and scientists at the OR-M facility

well as systems for protection of components to investigate:

– the attenuating potential of shadow shields for reactors of space nuclear power systems and the spatial and energy distribution of radiation in the shaded area up to the instrumentation module;

– the neutron and gamma ray distribution in instrumentation modules and the attenuating potential of local shields for individual units of the spaceship hardware;

– the reactor radiation scattering by antenna-feeder systems and other components;

– further attenuation of reactor radiation by components of a nuclear power system;

– the radiometric characteristics of standard precombustion chambers of space reactors and of a spaceborne dosimetry system;

– the scope for modeling real-life irradiation conditions and testing the shielding properties of radiation-proof components;

– the attenuating ability of mockup reactor-enclosing shields in various directions of radiation transfer.

Basic macroscopic experiments to study transport of neutrons and photons of a reactor energy range in various materials and complicated multidimensional compositions:

– studies on deep penetration of radiation into material;

– studies of radiation transfer with the use of a wide-angle unidirectional neutron or photon beam of a reactor spectrum in the direct (propagation) and inverse (reflection) geometry of an experiment;

– determination of energy, angular and space characteristics of primary radiation transfer in various materials;

– study of differential characteristics of secondary gamma radiation release from metals, normalized to unit flux of generating neutrons.

The OR-M Complex is also used for personnel training in radiation physics, physics of radiation transfer and radiation shielding.

Personalities

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CRITICAL FACILITY UG

The UG full-scale uranium-graphite critical facility is intended for carrying out critical experiments with uranium-graphite lattices.

The UG critical facility achieved first criticality on 18 December 1965.

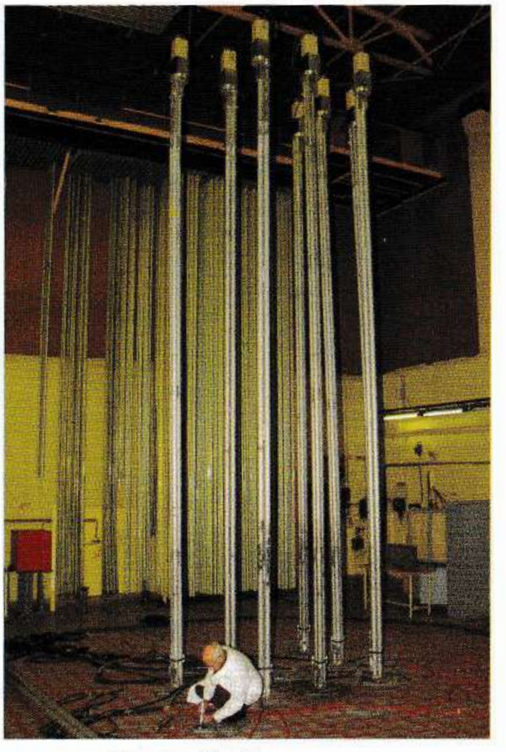
The UG facility is a large-scale critical facility, which enables experimentation for studying the neutronic characteristics and processes in the production reactor cores.

In 1988 the UG facility was shut down for upgrading. In 1996–1998 the facility was re-equipped by means of replacement of morally and physically obsolete components of the control system and the emergency protection, and by creation of a modern system for automation of experiments for the purpose of improving the reliability and nuclear safety during the facility operation.

Main areas of studies

A broad program of research has been undertaken at the facility in the following fields:

- experimental validation of the conversion of production uranium-graphite reactors to the power mode without plutonium generation;
- experimental validation of the detailed design of the MKER-800 increased-safety pressure-tube reactor.



UG critical facility compartment

The technical capabilities of the retrofitted UG facility offer the environment for carrying out research at the level of international standards.

At present time, the UG facility is operated in the long-term shutdown mode (safe storage).

CRITICAL FACILITY SF-1

Hot critical facility SF-1 – a uranium-water reactor prototype – is used in experimental studies of the neutronic characteristics of various propulsion reactors with water moderator in hot states. The facility came to first criticality on May 19, 1972.

In 1996, the facility was upgraded, with the control board moved to a separate room, the control and protection system radically modified and the data acquisition and display system brought up to date.

The operating period of the SF-1 facility was not specified.

SF-1 compartment

Main performance of SF-1

Thermal power, max. .... 0.1 kW

Moderator .... Water

Coolant: .... Water

pressure .... Up to 20 MPa

temperature .... Up to 300 °C

Thermal neutron flux, max .... $5 \cdot 10^9$ $cm^{-2} \cdot s^{-1}$

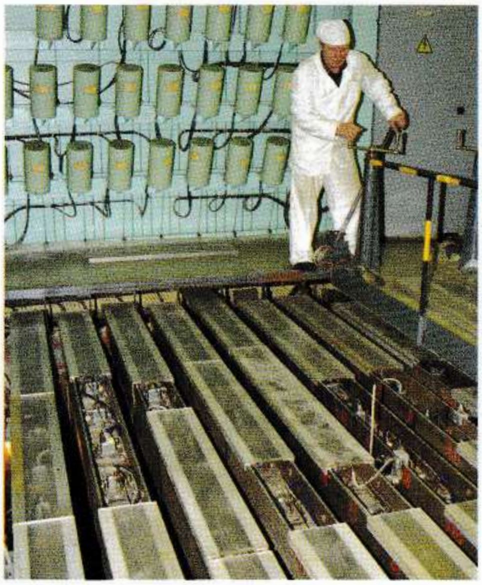
Main areas of studies

Tests of fuel elements and fuel assemblies for propulsion reactors of a new generation.

CRITICAL FACILITY EFIR-2M

The Efir-2M critical facility is a heterogeneous zero-power reactor with a light-water moderator and a complicted spatially separated structure of the neutron spectrum. The Efir-2M critical facility achieved first criticality on 1 January 1973.

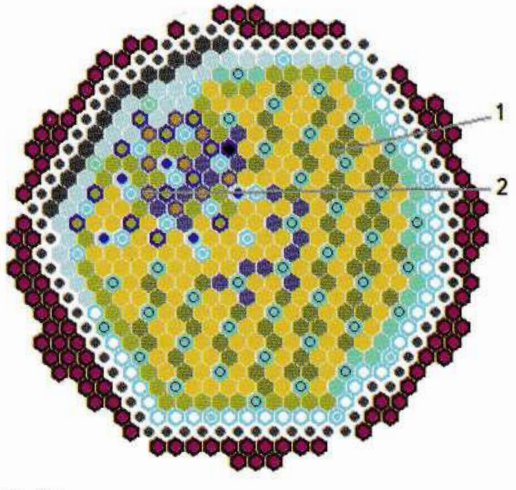
In 2004–2005 the critical facility's control and protection system was partially upgraded, which has led to improved safety and efficiency of research at the facility.



Efir-2M compartment

Experimental capabilities

The Efir-2M nuclear critical facility was built to validate the neutronic characteristics of reactors. The unique design of the critical facility and the systems that ensure nuclear safety makes it possible to create large-size fragments (sectors) for simulation of practically any cores in terms of composition and structure, and study both the neutronic characteristics of separate fragments by optimizing their load structure, and the parameters that characterize the fission and absorption reaction rates in separate fuel elements and in absorber elements.



Efir-2M core map:
1 – regular facility core; 2 – sector with a core model for the investigated reactor operating mode

Main areas of studies

– Experimental research to validate the composition and structure of the core for a production reactor facility, as well as new core items for generation of isotopic products.

– Experimental research into the characteristics of upgraded neutronic parameter monitoring systems, and development of functional algorithms.

– Experimental research to validate the verification of codes.

Main performance of the Efir-2M critical facility

Thermal power, max	0.1 kW
Moderator	Water
Working temperature	20 °C
Pressure	0.1 MPa
Fuel	UO_2
Fuel enrichment in ^{235}U	90 %
Reactivity margin	Up to 1 β_{eff}
Subcriticality in shutdown condition with full moderator level, not less than	3 β_{eff}
Temperature reactivity effect	$-2.5 \cdot 10^{-2}$ β_{eff}/°C

Main activities

The recent years have seen:

– the development and pilot commissioning of a measuring probe required to study the characteristics of the upgraded subcore monitoring system at the Efir-2M facility;

– the assembly and preliminary testing of the probe, which have proved that its use does not worsen the safety conditions of the Efir-2M facility operation and does not change its standard operating conditions.

The scheduled activities for the examination and timely replacement of overaged components aimed at extending the facility life make it possible to use the Efir-2M facility for research purposes for at least 10 more years.

It is planned to:

– upgrade the CPS system and some of the critical facility's process components;

– introduce the computer-aided measurement system for processing the experimental results, specifically when carrying out activities using the commissioned measuring probe.

Personalities

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CRITICAL FACILITY SF-7

Hot critical facility SF-7 – a uranium-water reactor prototype – is intended for experimental studies of the neutronics of water-moderated cores. The facility reached first criticality on March 12, 1975.

In 1996, the facility was upgraded, with the control board moved to a separate room, the control and protection system radically modified and the data acquisition and display system brought up to date.

The operating period of SF-7 is not specified.

The SF-7 facility allows building cores from fuel assemblies or from separate fuel elements.

The facility is a unique multipurpose installation meant for experimental studies in physics of propulsion reactor cores and for assessing the performance of various devices and detectors.

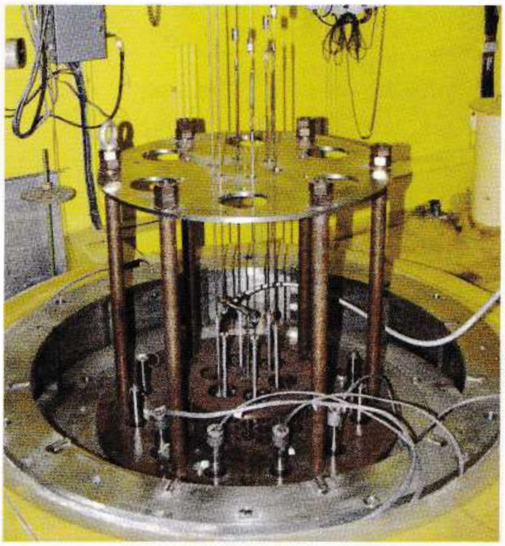
Main areas of studies

– Tests of fuel elements and fuel assemblies for propulsion reactors of a new generation.

– Tests and adjustment of instrumentation channels and chambers designed for propulsion reactors.

Main performance of SF-7

Thermal power, max.	0.1 kW
Moderator	Water
Coolant:	Water
pressure	Atmospheric
temperature	Ambient
Thermal neutron flux, max.	$5 \cdot 10^9$ $cm^{-2} \cdot s^{-1}$



SF-7 compartment

CRITICAL FACILITY GROG

The Grog multipurpose critical facility is intended for neutronic research on different reactor types, including HTGR, GT-MHR, pressure-tube graphite reactors, compact small-power reactors (local NPP dual-purpose reactors), and reactors for space nuclear power and propulsion systems of a megawatt class. The Grog critical facility achieved first criticality on 17 December 1980.

The Grog critical facility with a uranium-graphite critical assembly enables experiments on critical assemblies comprising two cores: the investigated one composed of life-size elements, and a driver core composed of mock-up elements and ensuring the criticality conditions. Different neutronic characteristics of the core can be obtained for expanding the experimental capabilities.

The possibility of varying broadly the neutronic parameters using simulation experiments and the similarity of such assemblies to the investigated cores offer solutions to a great deal of problems in physics and nuclear safety cases using a small number of full-scale elements, which is important at the initial development phases of a local installation.

Variations in neutronic parameters are achieved using the modular simulation principle. The required modules can be composed of various combinations of simple and easy-to-operate elements, which include:

Grog critical facility compartment

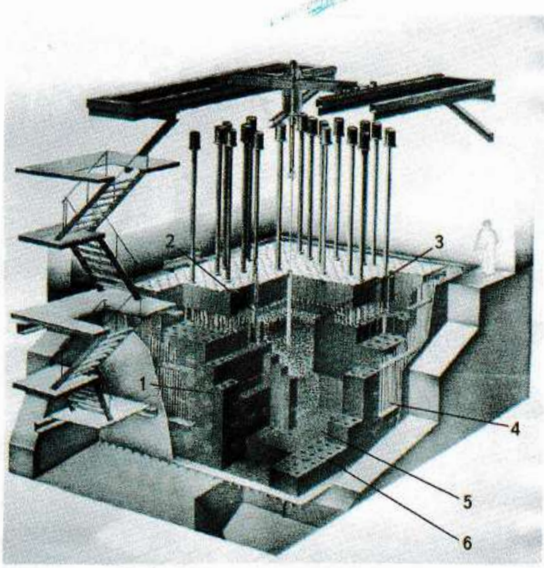
– a fuel element, which is a homogeneous mixture of a carbon-containing moderator and uranium dioxide with an enrichment in uranium-235 ranging from the natural level to 10%;

– a graphite element (in the form of inserts and blocks);

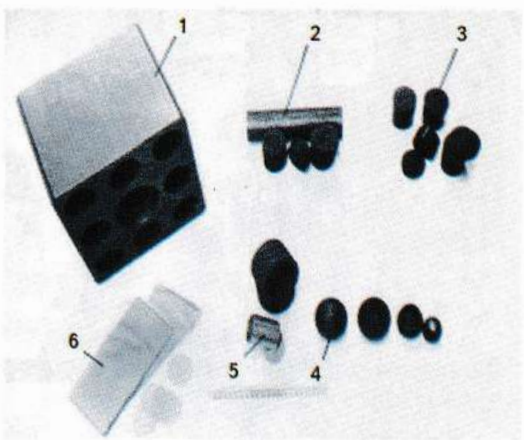
– a boron-containing element.

Burn-up products are simulated using ^{235}U (with the aid of an enriched nuclear fuel), ^{238}U (with the aid of a fuel of a natural composition) and boron.

The required combination of fuel and boron-containing elements is selected based on the condition of the similarity of life-size and simulated fission integral, absorption and microscopic absorption cross-section systems.



Grog critical facility with a uranium-graphite critical assembly: 1 – spherical fuel elements of the investigated core; 2 – driver core graphite blocks; 3 – graphite reflector blocks; 4 – experimental channel; 5 – CPS rod; 6 – oscillator; 7 – process channel storage



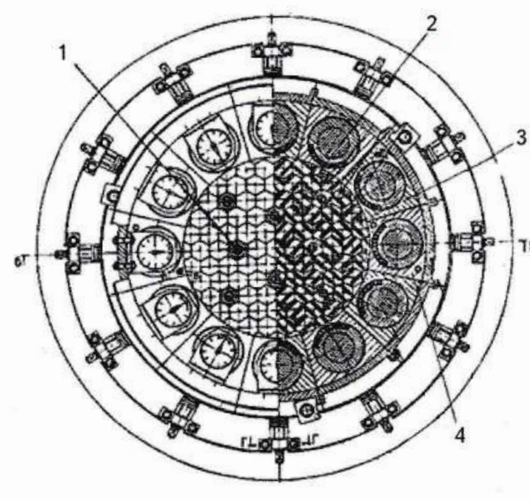
Components of the Grog critical facility uranium-graphite assemblies: 1 – graphite block; 2 – graphite elements; 3 – fuel elements; 4 – physical mockup of a spherical fuel element and its components; 5 – sleeves; 6 – boron-containing paper

Main performance of Grog CF

Fuel	U, enrichment 10 %
Moderator	Graphite
Reflector	Graphite
Core geometry	Cube 3×3×3 m
Graphite stack geometry	Cube 4.5×4.5×4.5 m
Fuel element mock-up geometry	Cylinder Ø5×5 cm

Main performance of the Chayka modular critical assembly under design

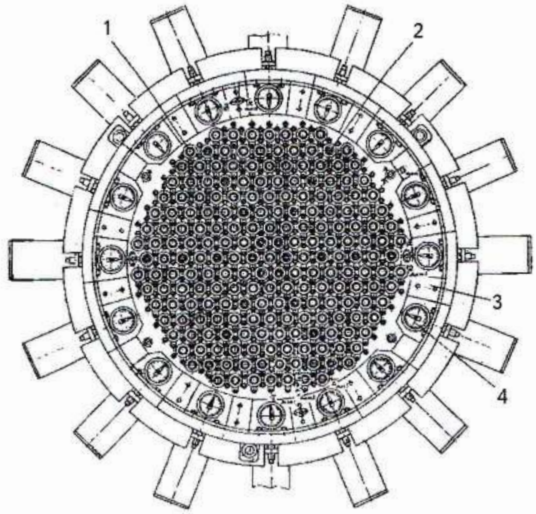
Power, max	0.1 kW
Moderator	None
Core height	100…600 mm
Core diameter	360…440 mm
Number of fuel elements	90…121
Fuel	Uranium-nitride spherical fuel elements Ø 7.36 mm with 90% enrichment in ^{235}U
Number of channels with absorber rods	Up to 8
Side neutron reflector	Beryllium, thickness 110 mm, height up to 600 mm
Number of rotary drums in side reflector	12



Cross-section of the Chayka critical assembly: 1 – absorber rod; 2 – control drum; 3 – fuel module; 4 – side reflector

Main performance of the Filin channel critical assembly under design

Power, max	0.1 kW
Moderator	Highly purified distilled water
Core diameter	500…655 mm
Core height	100…1100 mm
Fuel	Rods of Ø 2 mm (solid solutions of uranium carbide in refractory alloys) with 90% enrichment in ^{235}U
Spacing of hexagonal channel grid	40.5…46.3 mm
Number of FA	Up to 210
FA diameter	32 mm
Number of absorber rod channels	Up to 24
Side neutron reflector	Beryllium, thickness up to 78 mm, height up to 1100 mm
Number of rotary drums in side reflector	16



Filin critical assembly cross-section:
1 – FA channel; 2 – absorber rod channel; 3 – side reflector; 4 – control drum

Availability of the investigated core and the driver core makes it possible to conduct experiments with fragments of cores from different reactors including few life-size elements, which is important at the initial development phases.

Experimental capabilities

After the control system completion, experiments have been planned with reactors for space and nuclear power and propulsion systems of a megawatt class based on the Chayka and Filin critical assemblies that simulate reactor cores without moderator and with a hydrogen-containing moderator.

Therefore, the Grog multipurpose critical facility makes it possible to obtain experimental information for verifying physical analysis codes and carry out investigations for validation of parameters and optimization of the components and designs of developed reactors for various applications.

Personalities

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CRITICAL FACILITY ASTRA

The critical facility Astra is intended to study the neutronic characteristics of high-temperature gas-cooled reactors (HTGR) for the reactor safety and intrinsic safety cases. The critical facility Astra achieved first criticality on 5 August 1981.

The Astra critical facility is the world's only operating facility of such a kind. The simplicity and flexibility of the facility's design enable simulation of the configurations which are typical of different innovative designs. The results of long-term experimental research at the facility have made it possible to validate the most advanced Russian HTGR designs (VGR-50, VG-400, VGM) and have formed the basis for licensing the South African PBMR project. The program of experimental research based on the Astra facility is an integral component of the efforts for the demonstration of critical technologies as part of the GT-MHR international project.

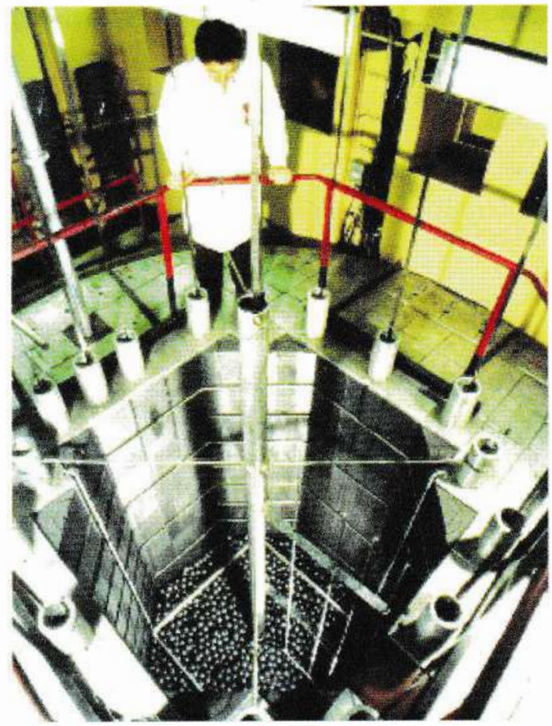
The critical assembly has a cylindrical steel vessel with a bottom. The vessel accommodates the lower and side reflector graphite blocks, forming so a core cavity of varied geometry, which is filled with spherical fuel elements. The vessel has the outer diameter of 3820 mm, the height of 4630 mm, the wall thickness of 10 mm.

It is possible to form the inner reflector and the upper end reflector of both graphite blocks and graphite spheres.

In section, the side reflector block is a square. Along its centerline, the block has a channel, into which a cylindrical graphite plug can be installed. The lower reflector blocks have the same cross dimensions and the height of 40 cm.

Critical experiments on the Astra facility use a dioxide-uranium fuel of an intermediate enrichment in ^{235}U (about 21%), which has the form of coated fuel particles distributed in the graphite matrix of spherical fuel elements.

The external diameter of each spherical fuel element is 6.0 cm. Each fuel element core contains 2.440 g of uranium. The coated fuel particles inside the fuel kernel have a spherical form and consist of a spherical uranium-dioxide kernel with a coating applied to it in four layers.



Astra facility's critical assembly

Main performance of the Astra critical facility

Neutron power	0.1 kW
Moderator	Graphite
Reflector	Graphite
Neutron flux	10^6 cm$^{-2}\cdot$s^{-1}
Fuel	UO_2
Enrichment in ^{235}U	21%
Heat removal	Natural air convection
Core temperature	Ambient
Number of control rods	24

The diameter of the uranium-dioxide kernel is 511 µm. The layers of the fuel particle coatings have the following characteristics (the layers are numbered in the order from the inner layer to the outer one):

- layer 1: pyrocarbon (PyC), the buffer layer adjoining the kernel, density 1.01 g/cm^3, thickness 94 µm;

• layer 2: pyrocarbon (PyC), density 1.95 g/cm^3, thickness 72 μm;

• layer 3: silicon carbide (SiC), density 3.26 g/cm^3, thickness 51 μm;

• layer 4: pyrocarbon (PyC), density 1.89 g/cm^3, thickness 57 μm.

The Astra critical assembly core may be composed of not only fuel elements: the existing set of spherical elements also includes absorber elements and graphite spherical elements of the external diameter 6.0 cm. And, similarly to the fuel elements, the absorber elements have

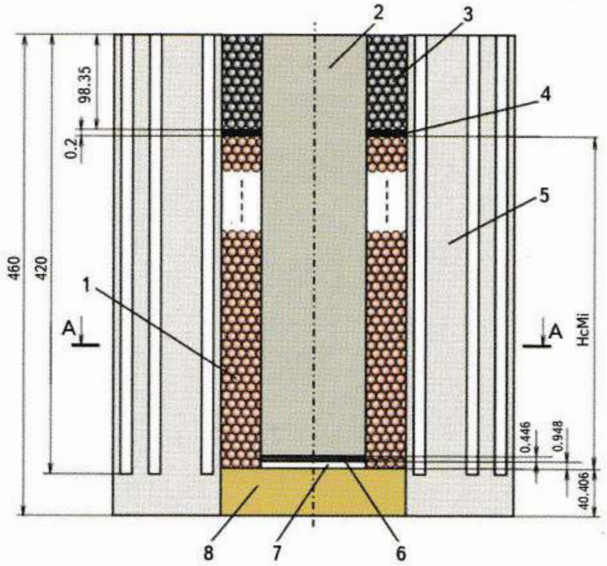
Graphite block

an inner core of the diameter 4.0 cm, which is a graphite matrix with uniformly distributed particles of natural boron carbide. The average diameter of the particles is 60 μm, and the total weight of the boron carbide in one spherical element is 0.1 g. The spherical graphite elements are solid and made of reactor graphite.

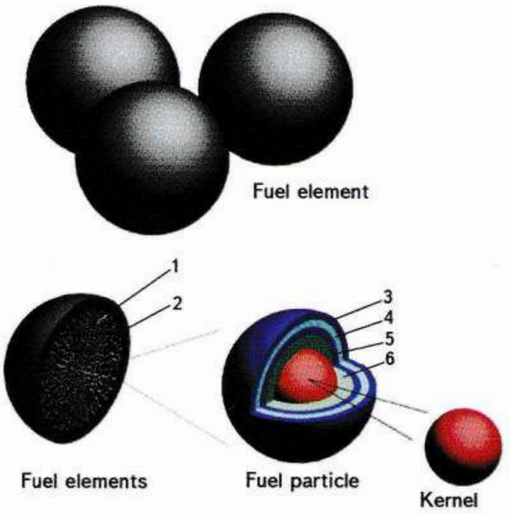
Therefore, the core can be filled with a mixture of fuel, absorber and graphite spherical elements, which provides a greater flexibility for simulation of different core designs for high-temperature reactors.

Control rods are used to regulate the Astra critical assembly and compensate for the reactivity margin. Functionally, the control rods are divided into shim rods, emergency protection (EP) rods and manual control (MC) rods. The shim and EP rods are of an identical design. A rod consists of two hinged interconnected parts positioned vertically one above the other.

The height of each rod is 387.5 cm. The shim and EP rods are capable of moving in the central channels of the side (and/or inner) reflector blocks.



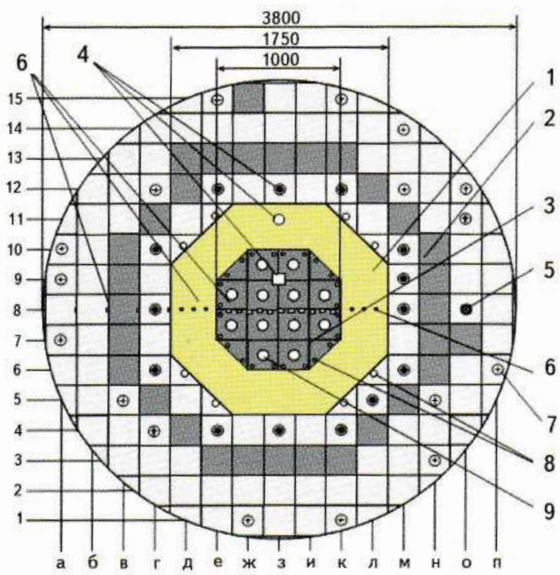
A sectional view of the critical assembly with a circular core (dimensions are given in cm):
1 – circular core with spherical fuel elements; 2 – inner reflector; 3 – upper end reflector; 4 – aluminum-alloy division plate; 5 – side reflector; 6, 7 – inner reflector support structures; 8 – lower end reflector



An Astra CF sphere:
1 – fuel core with coated fuel particles; 2 – graphite coating of a spherical fuel element; 3, 4, 5, 6 – coatings

A manual control (MC) rod is used for "fine" regulation. This is a double tube of an aluminum alloy with only air inside. The MC rod height is 388.5 cm.

The heat is removed by natural convection of air in the facility room. Forced heat removal is planned in the event of an upgrade to enable electrical heat-up.



Cross-section of the critical assembly with a circular core: 1 – circular core, 2 – side reflector; 3 – inner reflector, 4 – CPS rod channel, 5 – neutron source channel, 6 – experimental channels for sensors, 7 – neutron counter and ionization chamber channels, 8 – channels for power shaping absorber elements, 9 – unplugged inner reflector block channels

Experimental capabilities

The Astra critical facility is used for the following experiments on studying the peculiarities of the HTGR physics:

– determination of critical parameters for different configurations of critical assemblies;

– efficiency determination for single rods, systems of control rods and their interference;

– efficiency determination for different materials and samples;

– studies into the specific features of radial and axial neutron flux distribution (rates of various nuclear reactions) in the fuel assemblies;

– studies into the options of power density bulk distribution shaping over the core volume;

– studies into the effects of the synchronous insertion of an absorber rod group on the in-core axial and radial power density distribution;

– measurement of spectral (energy) neutron characteristics;

– measurement of the absolute power and kinetic parameters for the assemblies.

The facility does not have irradiation channels.

Main areas of studies

The results of experimental research are used to demonstrate the feasibility of conceptual designs for selection of innovative configuration options for high-temperature reactors and validate analytical codes.

Apart from scientific experiments, the Astra critical facility can be used for the following training activities:

– training of scientific and operating staff for next-generation reactors;

– scientific research as part of university projects;

– advanced training of nuclear industry specialists;

– training of students and specialists for operation of a full-scale small-power training reactor.

International cooperation

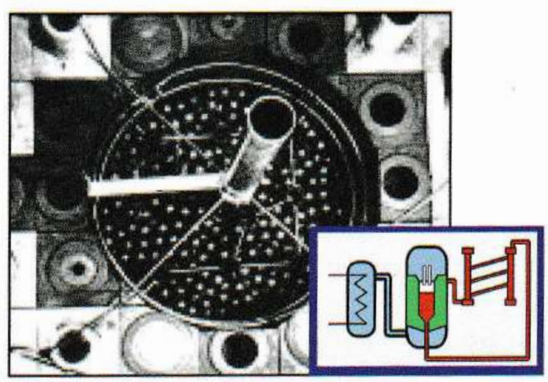
Since 1994 the facility has been hosting experiments on critical assemblies that simulate physical peculiarities of high-temperature reactors with a circular core, specifically, South Africa's PBMR and the Russian-US GT-MHR.

Preparations are under way as part of the GT-MHR international project for upgrading the facility to enable unique experimentation in studying the neutronic characteristics of circular-core HTGR during heat-up.

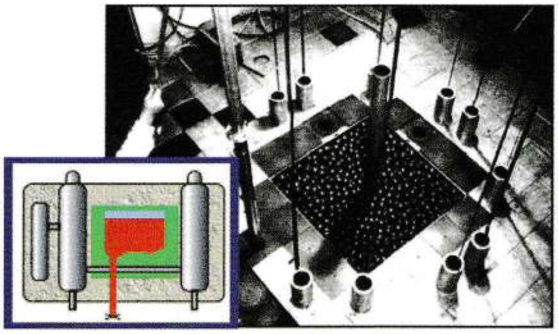
Main activities

The critical assemblies investigated at the Astra facility:

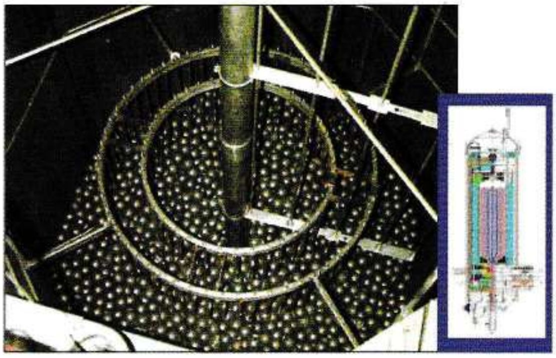
VGR – 50



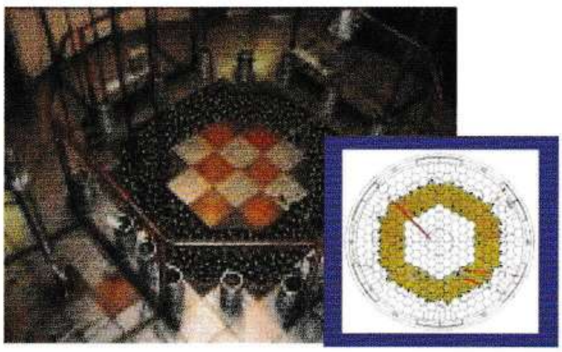
VG-400



PBMR



GT-MHR



The results of the experiments have been published in international handbooks.

The Astra-related activities have been made part of the new Russian-US GT-MHR international project agreement for 2011–2014.

Data have been generalized on experimental (ambient temperature) simulation of the physical peculiarities of the GT-MHR core's circular configuration with different in-core absorber element arrangements.

There have been demonstrated the expediency and feasibility of experiments at the Astra facility for studying the GT-MHR reactor temperature reactivity effects during heat-up of the critical assembly, which form the temperature reactivity effect as well as the dependence of the absorber rod efficiency on temperature.

Under the five-year work plan, the Astra facility's 30-year design service life will be extended and an upgrade of the facility will be undertaken to enable experiments with heat-up.

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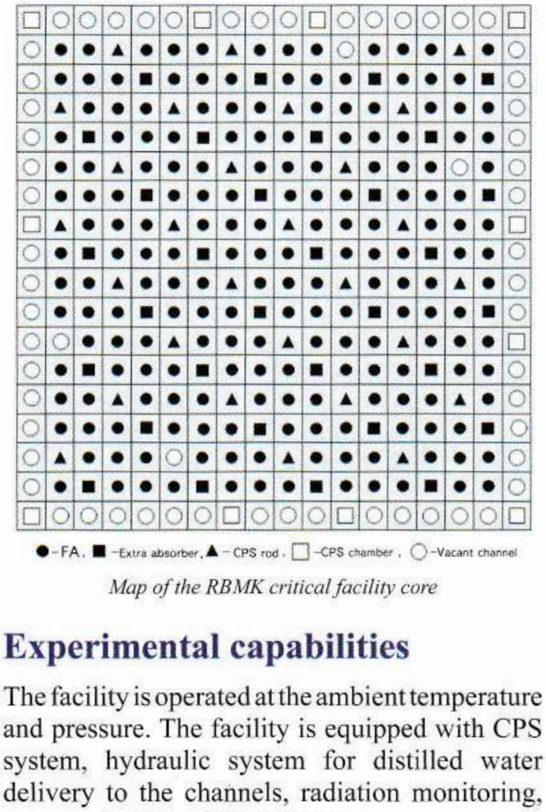
RBMK CRITICAL FACILITY

The RBMK critical facility is the only facility of the kind and is intended for experimental research in physics of uranium-graphite reactor cores. It achieved first criticality on 6 January 1982.

The major structural elements of the core (FAs, CPS rods, extra absorbers, graphite blocks, process channels) are standard elements of the RBMK reactor. RBMK FAs with uranium dioxide (from 0.4 to 3.6% enrichment in ^{235}U) and uranium metal (enrichment from 0.4% to 1.6%) are used for the process channel loading.



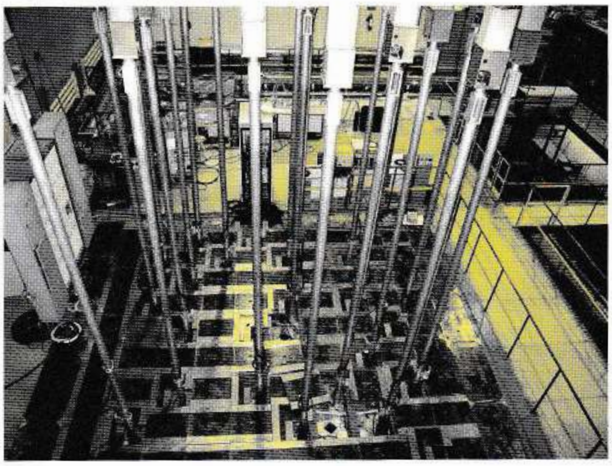
Map of the RBMK critical facility core

Experimental capabilities

The facility is operated at the ambient temperature and pressure. The facility is equipped with CPS system, hydraulic system for distilled water delivery to the channels, radiation monitoring, communication, TV surveillance, ventilation and gas treatment systems, and FA and graphite block storages.

The facility enables simulation of both small critical masses in the form of single assemblies and polylattices of different compositions, and fragments of the full-scale RBMK load of 256 channels for the purpose of:

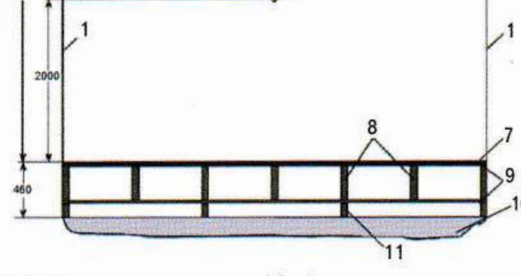
– examining new elements of the RBMK core load (new CPS rod designs, new FA types and so on);



RBMK critical facility compartment

Main performance of the RBMK critical facility

Rated power	25 W
Thermal neutron flux, max	$5 \cdot 10^6$ cm$^{-2} \cdot$s^{-1}
Graphite stack dimensions	4.5×4.5×4.1 m
Channel grid spacing	250 mm
Core height	3460 mm
Number of process channels	324



Side view of the BRMK critical facility:
1 – cadmium sheets; 2 – aluminum decking; 3 – borated-polyethylene plates; 4 – steel beams; 5 – graphite stack; 6 – neutron source channel; 7 – steel plates; 8 – foundation's steel structure; 9 – steel foundation mats; 10 – concrete floor; 11 – steel columns

– obtaining reference data to verify the employed and developed codes;

– improving the procedures and devices for measuring the RBMK reactor and SNF storage reactivity (subcriticality), and testing the NPP neutron flux sensors;

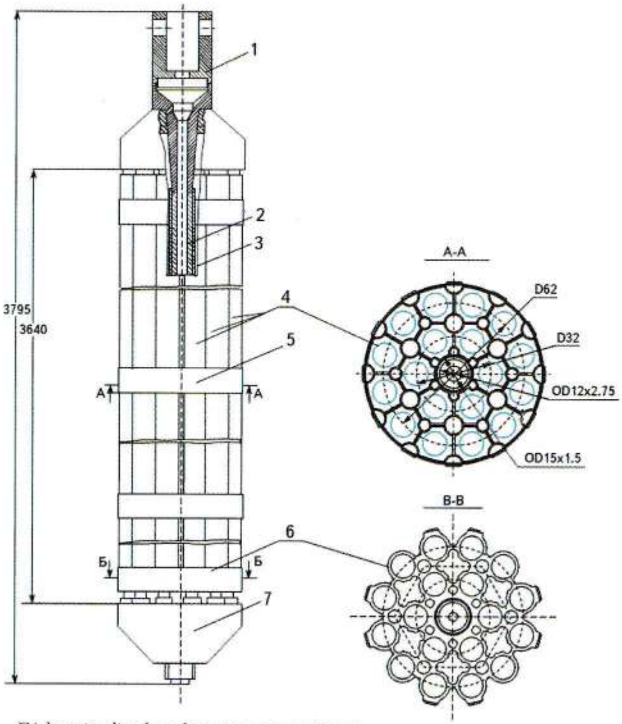
– validating the initial load of the Kursk NPP's Unit 5 with an upgraded graphite stack.

Basic experimental results

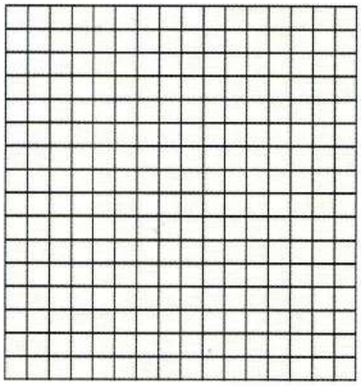
The prime objective of the first RBMK experiments was to investigate the assemblies of the Ignalina NPP's RBMK-1500 reactor FAs, which differed from the standard RBMK-1000 FAs in that there were steel heat exchange intensifiers in the FA upper half. In 1982-1983 investigations were conducted for the minimum critical masses from the RBMK-1500 FAs and the set of critical assemblies resulting from a consecutive changeover from the assembly with rows of unloaded channels of 23 FAs to a fragment of a full-scale reactor load of 256 channels, as well as studies into the neutron flux excursions at the ends of the fuel elements in a partially submerged FA.

In 1989 the facility's graphite stack of standard square-section graphite blocks was replaced for an upgraded stack of octagonal blocks. The upgraded stack (to be used at the Kursk NPP Unit 5) ensures a reduced reactivity void effect. It is also possible to fill the square-shaped cavities in the cell angles with graphite inserts to get back to the standard graphite stack of the type RBMK-1000 reactors. A great deal of experimental data has been obtained at the upgraded graphite stack on the neutronic characteristics of RBMK lattices with a 2% enriched fuel, which has helped verify the codes, chose the composition and structure of the initial reactor loads and get Gosatomnadzor's license for the construction of the Kursk NPP Unit 5.

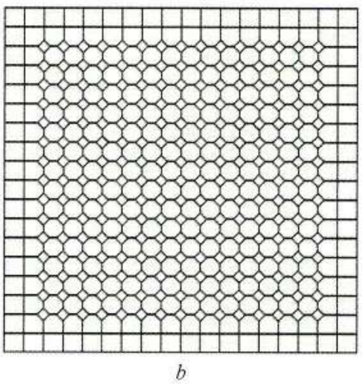
The experiments on the RBMK facility involved uranium-erbium fuel, belt and cluster CPS rods, fast-acting shutdown system (FASS) rods, steel and cluster extra absorbers, thorium absorbers, a FASS based on helium-3, and additional emergency protections based on boron carbide and gadolinium nitrate solutions. No load element has been introduced at operating reactors without being tested at the RBMK facility.



FA longitudinal and transverse sections:
1 – upper connecting element; 2 – central tube; 3 – framework tube; 4 – fuel element; 5 – spacer grid; 6 – end grid; 7 – lower connecting element



a



b

Facility's standard (a) and upgraded (b) graphite stack

International cooperation

As part of OECD/NEA International Criticality Safety Benchmark Evaluation Project, all components of the RBMK facility core have been certified in detail in terms of structure, dimensions, tolerances, compositions of materials and potential impurities. The fuel and graphite isotopic compositions and impurities therein have been studied. In 2004 a description of 28 RBMK facility's critical assemblies was included in the International Handbook of Evaluated Critical Safety Benchmark Experiments.

Main activities

In 2005–2007 the KENTAVR-KS measurement and computation system was introduced at the RBMK facility for determining and monitoring neutronic and process characteristics. This enables:

– measurement of the neutron flux throughout the core in static and dynamic modes;

– computation of reactivity (subcriticality) using different techniques;

– determination of subcriticality on assemblies with a negative reactivity margin;

– monitoring and recording of neutronic and process parameters of the facility, and data archiving.

The introduction of the KENTAVR-KS system has led to:

– a much higher nuclear and radiation safety;

– an increased amount and higher quality of experimental data obtained with a much shorter time needed for measurement and processing;

– much greater procedural capabilities of experiments (subcriticality measurements on assemblies with a negative reactivity margin).

Activities have been undertaken using the KENTAVR-KS system for improving the safety and efficiency of the RBMK reactors:

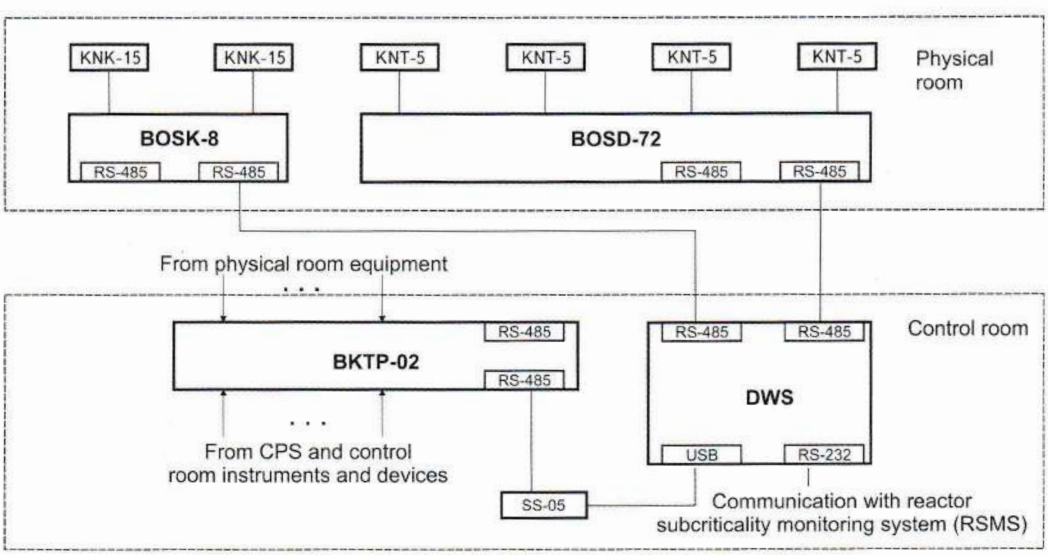
– investigation of the cluster EP rod prototypes (KRO-AZ);

– demonstration and trials of a new technique for determination of subcriticality without reaching the critical condition at an RBMK reactor;

– examination of a reactor load fragment with shaped FAs.

The RBMK facility is expected to be in operation until 2035. The Kurchatov Institute has a Gostekhnadzor license for operation of the RBMK nuclear critical facility until 10 July 2016.

Further successful operation of the RBMK nuclear critical facility requires upgrading of the CPS system.



Block diagram of the KENTAVR-KS system:
BOSK-8 – KNK-15 chamber signal processing unit; BOSD-72 – KNT-5 sensor signal processing unit; BKTP-02 – process parameter monitoring unit; DWS – display workstation

By now:

– technical assignments have been developed for the manufacturing of the starting and working monitoring and protection channels based on the MIRAZh MB units;

– amendments to the design documentation have been elaborated;

– the KENTAVR-KS system software has been upgraded;

– prototypes of the starting and working monitoring and protection channels based on MIRAZh MB units have been tested.

It is planned that the upgrading will be carried out for three years in a phased manner.

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CRITICAL FACILITY NARCISS-M2

Critical facility Narciss-M2 with its small critical assembly moderated by zirconium hydride is intended for modeling reactors with thermionic conversion of heat to electricity designed for space nuclear power systems. Narciss-M2 was brought to first criticality on August 28, 1983.

Critical facility Narciss-M2 is a neutronic model of the Yenisei reactor with single-element thermionic channels (TIC). Its critical assembly is a heterogeneous system with solid hydrogen-containing moderator (zirconium hydride), beryllium metal reflectors, and heat-producing fuel rods as mockup TIC.

The moderator appears as 5 zirconium hydride blocks. The blocks have 37 channels to accommodate guide tubes and mockup TIC with fuel.

One fuel rod contains about 720 g of ^{235}U.

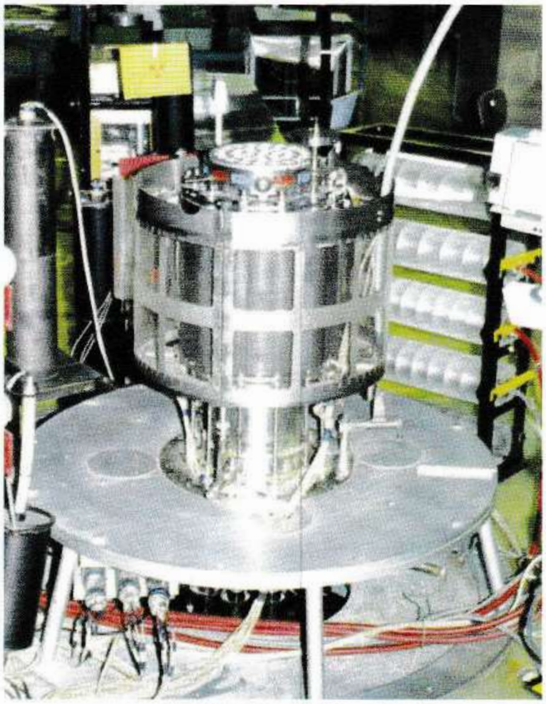
The critical assembly has a split radial beryllium reflector. Placed between the reflector inserts are 12 rotary beryllium drums (68 mm in diameter) with neutron-absorber segments. The top and bottom reflectors are also made of beryllium.

Main performance of the Narciss-M2 facility

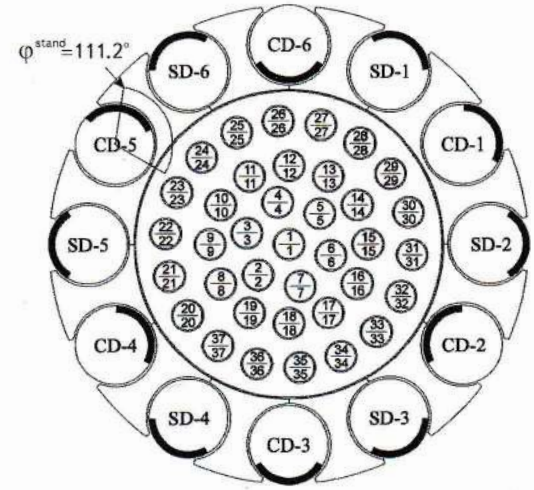
Power	0.01 kW
Fuel	UO_2
Enrichment in ^{235}U	96 %
Moderator	Zirconium hydride
Reflector	Beryllium

Experiments at the critical facility are supported by:

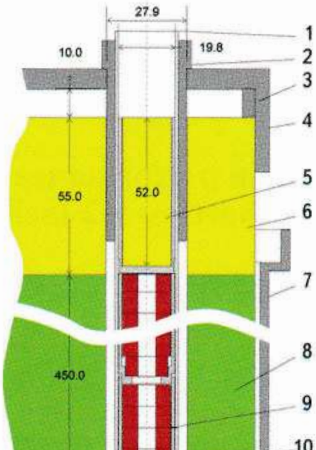
- control and protection system (CPS) of critical facilities;
- power supply system;
- experiment automation system;
- system for remote neutron source delivery;
- communications and video surveillance systems;
- canyon with a biological shield;



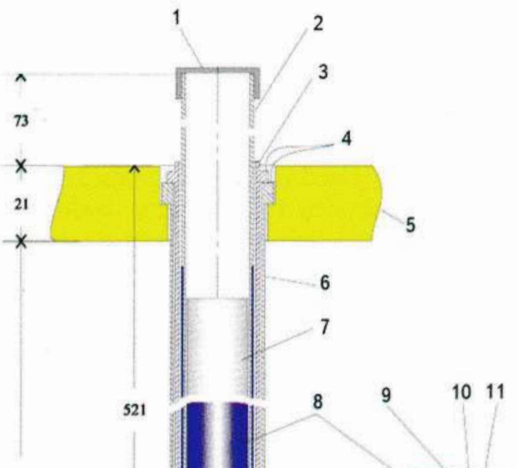
Narciss-M2 compartment



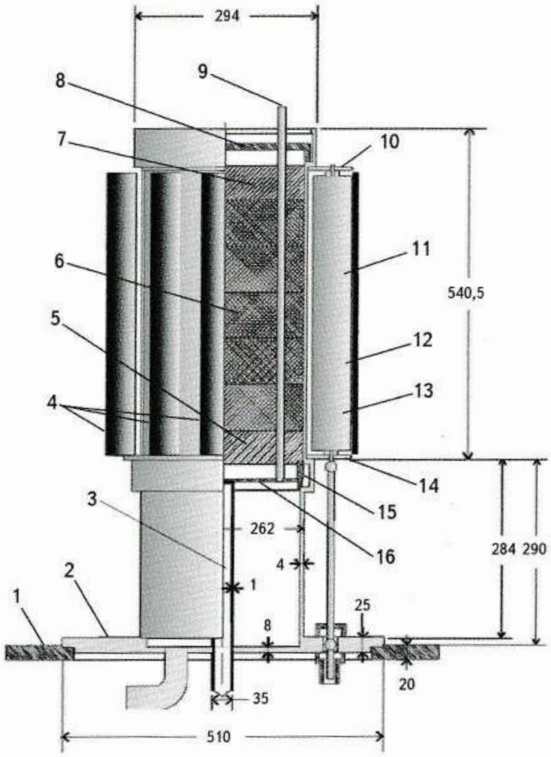
Narciss-M2 in section



Single-element TIC of the Narciss-M2 facility:
1 – collector tube; 2 – split sleeve; 3 – spacer ring; 4 – top tube plate; 5 – BeO reflector; 6 – top beryllium reflector; 7 – upper part of the vessel; 8 – moderator blocks (6); 9 – mockup emitter sections; 10 – fuel; 11 – bottom beryllium reflector; 12 – BeO reflector; 13 – support ring; 14 – bottom tube plate



Multi-element TIC of the Narciss-M2 facility:
1 – cap; 2 – mockup emitter; 3 – mockup collector; 4 – shoulders; 5 – top tube plate; 6 – guide tube; 7 – BeO; 8 – fuel; 9 – emitter; 10 – collector; 11 – guide tube; 12 – coating; 13 – aluminum foil; 14 – bottom tube plate; 15 – manufacturing clearance; 16 – BeO



Sectional view of Narciss-M – neutronic prototype of the Yenisei reactor facility:
1 – support plate; 2 – steel plate of the vessel; 3 – neutron source tube; 4 – rotary drums; 5 – bottom end reflector; 6 – moderator; 7 – top end reflector; 8 – top tube plate; 9 – mockup TIP with fuel (37); 10 – top mounting ring; 11 – assembly vessel; 12 – drum casings; 13 – rotary drum; 14 – bottom mounting ring; 15 – additional ring; 16 – bottom tube plate

Neutronic prototype of the Yenisei reactor

- dosimetry system;
- ventilation system;
- system for physical protection of nuclear materials;
- nuclear materials control and accounting system.

Experimental capabilities

The Narciss-M2 facility allows modeling small cores with thermionic converters for space nuclear power systems.

Methodological support

A critical state is determined by incremental buildup of the core accompanied by control of multiplication of the neutron source with a good accuracy (the error in estimating a critical state in reactivity terms does not exceed $2.5 \cdot 10^{-5}$ ΔK/K).

Reactivity measurements rely on a number of methods. These include: the method of a "steady-state period" in the event of a positive reactivity ramp; the method of a "shooting source", with a quick removal of the neutron source from the assembly acting as a perturbation; the method of a pulsed neutron source; statistical methods based on analysis of fission chains, with the assembly operating at a preset power level; the method of reverse kinetics based on numerical analysis of time-dependent neutron flux behavior in the assembly.

Statistical methods are used for measuring absolute values of neutron power and kinetic parameters of the assembly.

To determine the worth of controls, use is made of all reactivity measurement methods, including rod release.

Power density and reaction rates distribution as well as spectral indices are measured not only by regular instrumentation but also by activation of detectors with subsequent analysis of their radioactivity at special facilities.

Various reactor cores are modeled at the Narciss-M2 and Aksamit facilities with the use of a unique assortment of nuclear materials, moderators and absorbers, which also permits simulating a certain range of emergencies with nuclear safety implications.

Hardware

The above methods rely on the following specially developed experimental hardware interfaced with personal computers (PC):

– a multichannel computer-based system for measuring the current of ionization chambers, which allows application of dynamic reactivity measurement techniques;

– a multichannel system of pulse lines, which is controlled from personal computers and is intended for statistical experiments;

– a timing device, which involves IBM PC in its operation with pulse lines; it is employed in pulsed-mode and statistical experiments;

– ODIMOR system designed to perform gamma-scanning of fuel elements, automatic polling of a large number of sensors and to provide for measurements of power density and reaction rate distributions.

All the instrumentation channels are computer-based with installed software for processing experimental data and storing the results in a database.

Research objectives:

– to confirm the neutronic characteristics of reactors;

– to prove the nuclear safety of reactors;

– to confirm the neutronic characteristics of various reactor design options, to qualify software systems and their applications.

International cooperation

Some of the work at the critical facility was carried out together with US specialists from the Los Alamos, Oak Ridge, Sandia and Idaho laboratories.

The critical facility also served as a training and probation ground for Chinese and Belorussian personnel.

Main activities

Recent and current activities at the Narciss-M2 critical facility:

– studies of neutronic characteristics of space reactors, including the effects of reactivity, the worth of control rods and the reactivity balance in normal and emergency conditions;

– studies of nuclear safety characteristics of space high-temperature nuclear power systems;

– studies of power distribution among core components;

– studies of the applicability of neutronic methods (method of critical studies and reactivity measurements) for high-accuracy determination of fissile material content in specimens for the purposes of material accounting and control as well as for its security;

– training and probation of personnel from China and Belarus.

The Narciss-M2 facility was used in benchmark experiments.

The facility made it possible to carry out essential neutronic studies during development of a thermionic conversion reactor for the space nuclear power system Yenisei. Plans have been made to use this facility for developing a similar system for China.

Personalities

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CRITICAL FACILITY DELTA

Critical facility Delta – a uranium-water reactor prototype – is intended for experimental studies of the neutronics of water-moderated reactors and their cores. Its first criticality was reached on April 24, 1985.

In 2002, the facility was upgraded: the control board was retrofitted, the data acquisition and display system was improved, and the process circuit was modified to have moderator electrically heated to 90 °C. Preparations are under way for an upgrade. The operating period of Delta is not specified.

Main performance of the Delta facility

Thermal power, max.	0.1 kW
Moderator	Water
Coolant:	Water
pressure	Atmospheric
temperature	Up to 90 °C
Thermal neutron flux, max.	$5 \cdot 10^9$ $cm^{-2} \cdot s^{-1}$

Delta facility compartment

The Delta facility is a multipurpose installation intended for experimental studies on the physics of propulsion reactor cores, for investigations into the neutronics of standard reactor cores and measurement of power density distribution among the core components. In terms of physics, Delta is a thermal water-water facility with options for core construction from fuel assemblies or separate fuel elements.

Experimental capabilities

The facility is designed for testing the cores of various propulsion reactor systems; it allows testing standard cores and measuring power density distribution therein.

Main areas of studies:

– tests of fuel elements and fuel assemblies for propulsion reactors of a new generation;

– tests and adjustment of instrumentation channels and chambers designed for propulsion reactors;

– measurement of power density distribution in a reactor core.

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CRITICAL FACILITY V-1000

The V-1000 critical facility is intended for investigations on reactor cores of the VVER-1000 or VVER-440 fuel assembly types. The V-1000 critical facility reached first criticality on 3 September 1986.

The facility's critical assembly is a uranium-water core with a water reflector, a prototype of the VVER power reactor. It includes a FA load with UO_2 fuel enriched up to 4.4% in ^{235}U.

The critical facility tank capacity is 70 m^3. The number of the CPS members in the VVER-1000 assembly is up to 109, and that in the VVER-440 assembly is up to 73. The maximum number of the VVER-1000 assemblies is 163. The maximum number of the VVER-440 working assemblies is 312. The maximum number of the VVER-440 shim rod assemblies is 37.

There is no forced heat removal.

Main areas of studies

This is a multipurpose facility intended to study the neutronic characteristics of the VVER-1000 or VVER-440 power reactor cores, perfect their fuel cycles, improve technical and economic performance and safety of the operating NPPs, and demonstrate the reliability of neutronic calculations.

International cooperation

Applied activities to validate the neutronic characteristics of foreign PWR projects.

Personalities

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SERGEY GRADSKOV
Director of the Experimental Facilities Department

V-1000 critical assembly room

V-1000 core

Main performance of the V-1000 critical facility

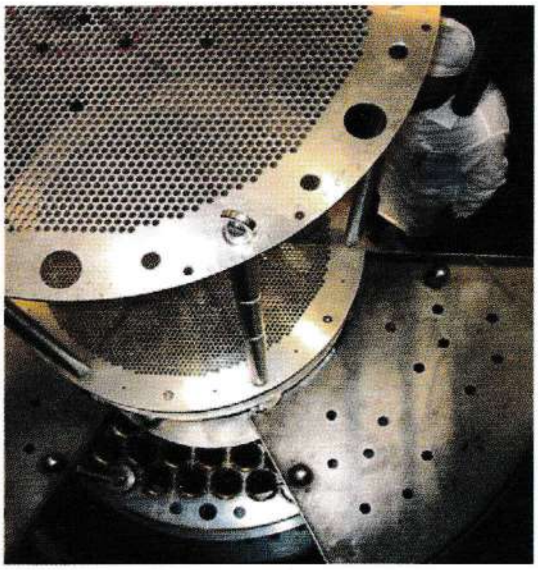
Rated power	0.2 kW
Fuel	UO_2
Enrichment in ^{235}U	Up to 4.4%
Moderator	Distilled water or solution of boric acid
Reflector	Distilled water or solution of boric acid
Thermal neutron flux, max.	$5 \cdot 10^7$ cm$^{-2} \cdot$s^{-1}

V-1000 control panel

CRITICAL FACILITY "P"

Critical facility "P" is a water-water reactor prototype built for experimental studies of various uranium-water multiplication systems consisting of fuel bundles or fuel elements. Its first criticality was attained on July 17, 1987.

This facility is a stainless steel tank which is filled with water for experiments. Its height is 5613 mm, the inner diameter being 2300 mm, and volume 16 m^3. The facility is equipped to maintain constant moderator temperature in the range from 16 to 95 °C. It has a versatile configuration, and its core can be built of individual fuel elements or bundles of types VVER-1000, VVER-440, etc. The maximum number of bundles is 91 for VVER-440 and 37 for VVER-1000, while the maximum number of fuel elements in one barrel is 10330.



Critical facility "P"

The fuel element pitch depends on the choice of the support plate with the required spacer grid. Thus, for a triangular lattice it can be 11.0, 12.7, 16.0 mm. The fuel lattice configuration is not limited to the triangular or square options. All a new core requires is to install the internals: support plate, spacer grid and the CPS platform.

For experiments with a core built of fuel elements, a bottom support plate, will be installed in the tank; mounted upon it will be fuel elements and the upper spacer grid. Criticality is attained by raising the moderator level or changing the position of CPS rods.

Main performance of facility "P"

Power, nominal	0.2 kW
Moderator	Distilled water or boric acid solution
Fuel	UO_2
Enrichment in ^{235}U	Up to 6.5 %
Thermal neutron flux, max	$5\cdot10^7$ $cm^{-2}\cdot s^{-1}$
Number of CPS rods	19

The main structural material of components and systems is stainless steel.

There is no forced heat removal.

Main areas of studies

– Determination of critical moderator levels.

– Study of induced activity of irradiated fuel or indicators.

– Determination of the worth of boric acid dissolved in moderator.

– Study of the worth of absorbers.

– Determination of power density distribution.

The experiments performed to study the worth of various absorbers involved boron (as boron carbide), hafnium (in the form of rods or tubes), dysprosium (as dysprosium titanate).

International cooperation

Applied activities are in progress to validate the neutronic characteristics of foreign PWR designs.

Personalities

YURI SEMCHENKOV
Director, VVER Department

CRITICAL FACILITY KVANT

Critical facility Kvant – a uranium-water reactor prototype – is intended for experimental studies of the neutronic characteristics of water-moderated reactors and their cores as well as the radiation transport. The facility reached first criticality on May 20, 1990.

In 2006, the facility was upgraded, with the control board retrofitted and the data acquisition and display system modified. Preparations for its further upgrades are in progress. The operating period of the facility is not specified.

Critical facility Kvant is designed for tests of cores for various propulsion reactor facilities. It is fit for testing and adjusting standard instrumentation channels with the use of an instrumentation channel certified in terms of neutron flux.

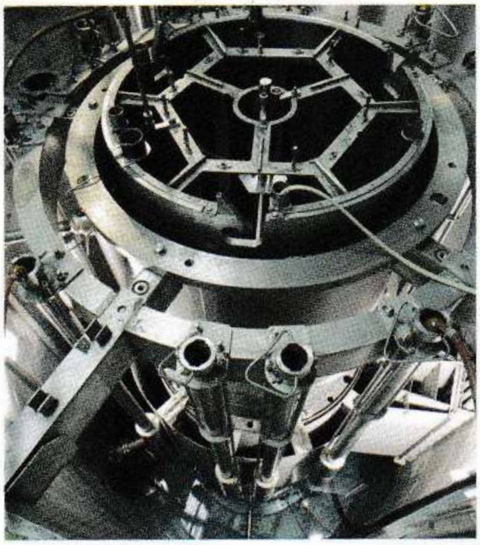
Main areas of studies

– Tests of fuel elements and fuel assemblies for propulsion reactors of a new generation;

– tests and adjustment of instrumentation channels and chambers designed for propulsion reactors;

– tests of radiation shields for propulsion reactors;

– studies of the impact made by neutron and gamma-radiation on electronic components.



A view of the Kvant facility

Main performance of the Kvant facility

Thermal power, max. .... 1 kW

Moderator.. Water

Coolant:.. Water

pressure .... Atmospheric

temperature .... Ambient

Thermal neutron flux, max........................ $5 \cdot 10^{10}$ $cm^{-2} \cdot s^{-1}$

CRITICAL FACILITY SK-FIZ

Critical facility SK-fiz – a water-water power reactor prototype – is designed for experiments to study the physical characteristics of inserts with advanced fuel of water-water reactors and to try out experimental procedures. Its first criticality was attained on June 25, 1997.

The facility is a stainless steel tank which is filled for experiments with distilled water or boric acid solution with boron concentration up to 15 g/kg. The empty tank of the critical assembly has the volume of 8.5 m^3.

Main performance of the SK-fiz facility

Power, nominal .... 0.6 kW

Moderator............. Distilled water or boric acid solution

Fuel .... UO_2

Enrichment in ^{235}U .... Up to 4.4 %

Thermal neutron flux, max........................ $1 \cdot 10^{11}$ $cm^{-2} \cdot s^{-1}$

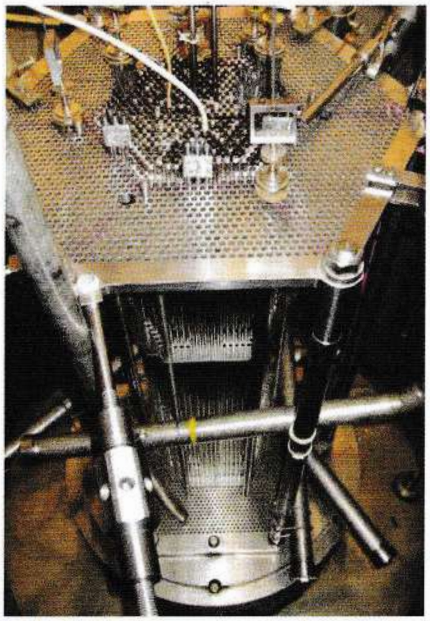
Number of CPS members 27

Experimental capabilities

This facility is a versatile installation intended for experiments to study various uranium-water multiplication systems. Its design allows studying cores made of fuel elements only or as mixed fuel element-bundle configurations. While studying a small RNM (zero power reactor) core, it is possible to try out methods for interpreting indications of the in-core instrumentation as well as experimental and computational methods for determining fuel assembly power, for measuring power density and neutron fluxes by means of miniature sensors – as applied to VVER reactor cores.

International cooperation

Applied activities are in progress to validate the neutronic characteristics of foreign PWR designs.



A view of the SK-fiz facility

CRITICAL FACILITY AKSAMIT

The Aksamit critical facility is one of the installations in the «R» testing complex at the Research Center's Institute of Nuclear Reactors. The Aksamit critical facility achieved first criticality on 25 February 2002.

The facility is intended for research into the neutronic and critical parameters and nuclear safety characteristics of advanced installations.

Work has been under way since 2002 to build a space nuclear power system with a thermionic converter reactor of the electric power 50…100 kW with a zirconium-hydride moderator and unified multielement thermionic channels (TIC).

RP-50, a critical reactor facility, is being built for neutronic studies of the reactor for such facility. This is a neutronic simulator of the RP-50 reactor and the experimental power reactor for ground tests of the power reactor unit (PRU), which is intended to try out TIC in near-full-scale conditions. It is planned that the cycle of research cycle will be undertaken at the Aksamit facility, a part of which the RP-50 critical facility will be after the upgrading.

Aksamit compartment

Main areas of studies:

– experimental research into the neutronic characteristics of the RP-50 critical assembly to confirm the selected parameters of the space nuclear power system and the PRU experimental reactor;

– reactor nuclear safety analysis;

– validation of the radiation situation and determination of the radiation protection parameters;

– certification of software systems and their applied uses in the given realm of knowledge.

A key task is experimental research on a neutronic prototype at the critical facility. These activities include:

• an integrated research into the neutronic characteristics of the critical assembly with a zirconium-hydride moderator;

• development of physical weighing procedures for the acceptance of the major PRU reactor components (CPS control drums and safety rods);

• acceptance of the PRU reactor's major components for nuclear power tests;

• simulation of the neutronic characteristics of the PRU with TIC simulators;

• research into the neutronic characteristics of the simulator assembly, and experimental demonstration of the PRU representativeness with respect to the standard RP-50 item;

• facility-based supervision of the PRU using the critical assembly.