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Research complex, which is now called OJSC "Institute of Reactor Materials" (hereinafter referred to as IRM), was established in early 1960s for testing reactor materials, studying reactor physics and producing isotopes. IVV-2M reactor is the main research facility at IRM.

Complex site is located within the Beloyarsk NPP controlled area.

A.A. DYAKOV,
Director of OJSC IRM

General view of research complex

Company's test facilities include research nuclear reactor, hot cells building and test equipment workshop. They determine specific nature of company's activity, namely: in-pile tests, pre- and post-irradiation examination of various samples.

IVV-2M reactor is used in a wide range of basic and applied researches in solid state physics, magnetic structure of materials, nature of magnetic interaction and neutron irradiation effect on various crystal structures.

IRM performs the following activities:

– R&D to support design, construction, safe operation and decommissioning of various-purpose reactors including propulsion and space nuclear reactors;

– experimental research to introduce new technologies and materials as well as to enhance performances of introduced objects which use nuclear energy;

– basic researches, experimental and theoretical works in nuclear energy fields;

– experimental and analytical justification for nuclear, radiation and environmental safety and reliability of nuclear facilities;

– facility life management, monitoring of equipment, structures, materials of reactor cores at nuclear facilities and nuclear power plants, researches in extending life of nuclear power facilities and objects;

– engineering and production of units for propulsion nuclear power facilities as well as their pilot test benches and components;

– production of isotopes for medical and technical application.

E.M. SULIMOV,
Chief Engineer

Production and research work areas

IVV-2M reactor includes the following test benches and facilities for testing fuel and fuel element mockups to be used in various reactors and for studying fission product release under irradiation:

- channels for studying structural material behavior;
- horizontal experimental channels equipped with neutron difractometers and spectrometers for studying polycrystalline and monocrystalline materials.

Experiments are conducted within a wide range of temperatures, pressures and magnetic fields.

IVV-2M WATER-COOLED RESEARCH REACTOR

IVV-2M site

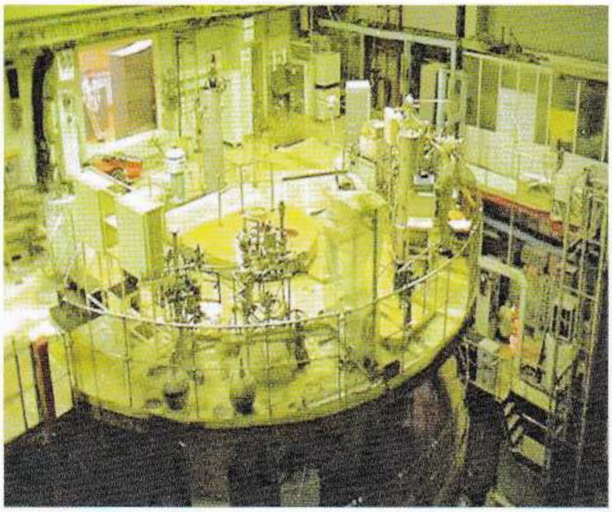
IVV-2M multi-purpose research reactor is water-cooled pool-type heterogeneous research reactor which was built in 1966 to the NIKIET design. First criticality was in April 23, and first power was in October 18. Rated power of the reactor was 10 MW.

IVV-2M reactor is used to tackle the following tasks:

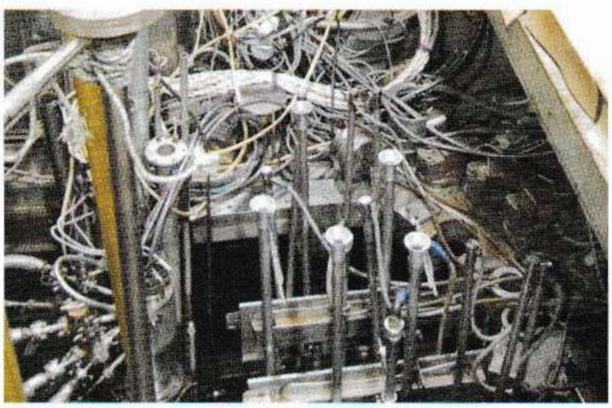
- justification for operability of components of new types of reactors;
- justification for operability of special-purpose facility components;
- justification for designs of power facilities and space propulsion systems;
- justification for lifetime and emergencies of gas-cooled reactor FAs;
- justification for the design and lifetime of direct converters;
- production of radionuclides.

IVV-2M main performance

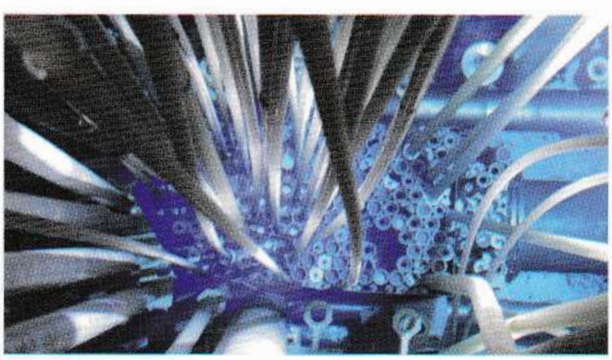
Thermal reactor power	15 MW
Maximum neutron flux:	
thermal	$5\cdot10^{14}$ $cm^{-2}\cdot s^{-1}$
fast (E>0.1 MeV)	$2\cdot10^{14}$ $cm^{-2}\cdot s^{-1}$
Core height	500 mm
Effective diameter of the core	500 mm
Primary coolant	Water
Secondary coolant	Water
Primary coolant flow rate	Up to 1200 m^3/h
Secondary coolant flow rate	Up to 1200 m^3/h
Core inlet temperature	Not more than 40°C
Core outlet temperature	Up to 65°C
Number of stationary experimental channels:	
horizontal	10
vertical	2
Experimental devices (ampoules, loops) in the core:	
down to ø 60 mm	Up to 40
ø 120 mm	2
ø 130 mm	1
ø 190 mm	1
ø 400 mm	1
Fuel cycle	From 300 h to 500 h



IVV-2M main hall



a



b

View under retractable shielding plates (a) and view of IVV-2M core (b)

The reactor is also used for implementing basic researches in solid state physics, magnetic structure of materials and nature of magnetic interactions.

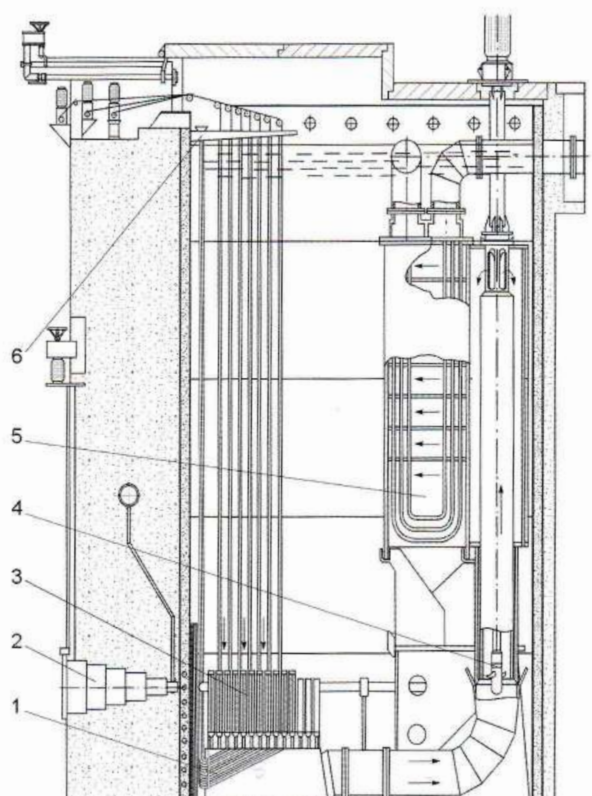
In 1975–1988 the reactor had been retrofitted. A tank made of stainless steel, steel-zirconium adaptors for horizontal experimental channel tubes and additional upper biological shielding were installed. Core and reflector support grid, reactor tank shell and core suction pipe were replaced. Fuel failure detection system was installed, conversion to tubular FAs of IVV-2M type with cermet fuel was performed; reactor heat exchanger, primary pump and CPS were replaced.

Reactor retrofitting allowed core neutronics to be enhanced, rated power of the reactor to be increased up to 15 MW and reactor life to be extended till 2025.

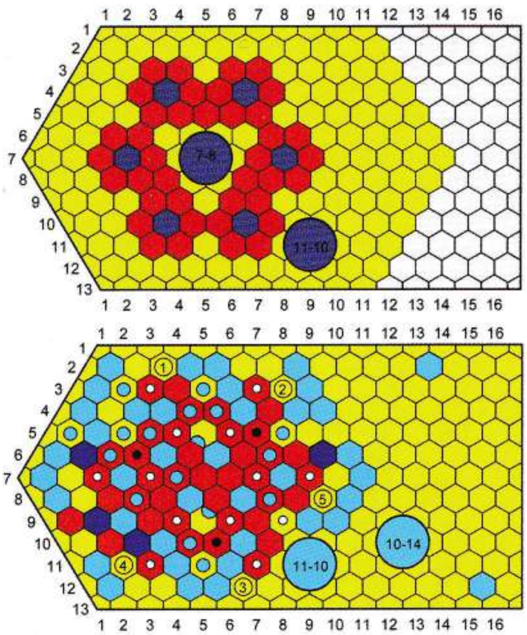
IVV-2M FAs are used in the reactor core. As per the specific task for the fuel campaign the core arrangement is selected on the basis of standard or newly developed core maps.

Every fuel element is a 3-layer hexagonal tube which consists of cermet (UO_2+Al) fuel meat, inner and outer claddings, top and bottom plugs.

Fuel enrichment in terms of uranium-235 is 90%.

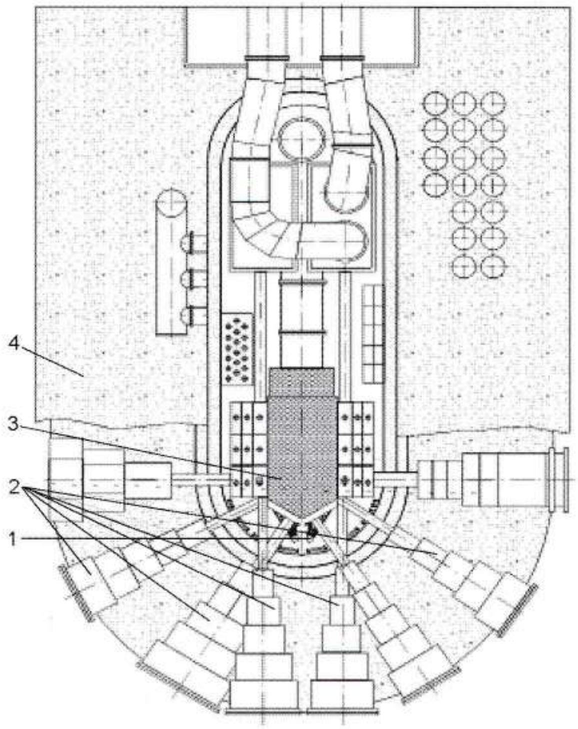


Sectional drawing of IVV-2M reactor:
1 – fuel failure detection system, 2 – horizontal experimental channel; 3 – reactor core; 4 – reactor coolant pump; 5 – heat exchanger; 6 – CPS floor

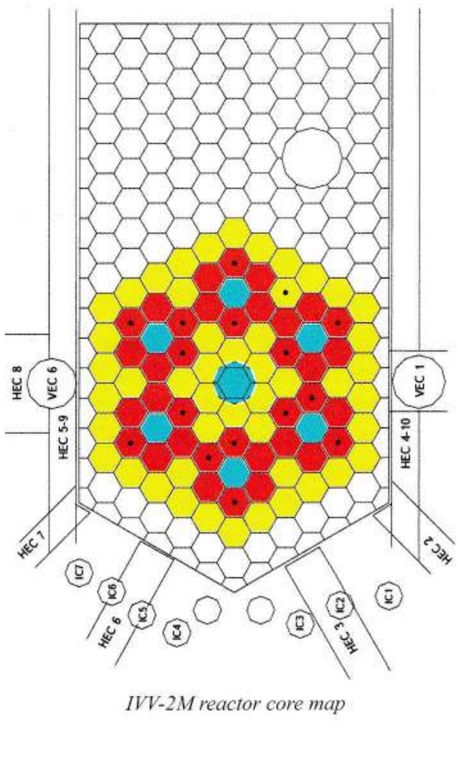


Options for IVV-2M core maps

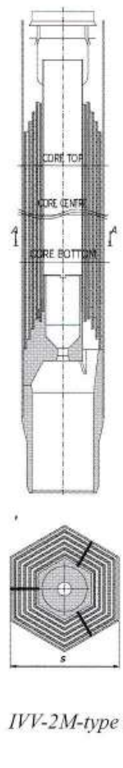
- *Fuel assembly (FA)*
- *FA + shim rod (ShR)*
- *FA + scram rod (ScR)*
- *Water cavities (traps) with experimental (irradiation) devices*
- *Beryllium block*
- *Beryllium block + AR (additional absorber rod)*
- *Beryllium block + in-core ionization chamber KTV*



IVV-2M cross-section:
1 – vertical experimental channel; 2 – horizontal experimental channel; 3 – reactor core; 4 – biological shielding



IVV-2M reactor core map



FA element	Width across flats, mm	Element thickness, mm
Internal casing tube	30.90	0.90
Fuel element 1	35.80	1.35
Fuel element 2	41.55	1.35
Fuel element 3	47.30	1.35
Fuel element 4	53.05	1.35
Fuel element 5	58.80	1.35
External casing tube	63.00	0.80

IVV-2M-type FA

Number of FAs in the standard loading of the core can vary from 30 to 42. It is achieved due to the combination of additional loading of FAs and partial reloading of FAs with ultimate fuel burn-up.

Primary coolant circulates through FA downwards by means of a circulation pump which is submerged into the reactor pool. Heat is removed from primary coolant through the heat exchanger which is installed into the reactor tank and cooled with service water.

Experimental capabilities

IVV-2M reactor is equipped with the following:

– test benches and facilities for gas testing fuel and fuel element mockups to be used in various facilities and for studying fission product release;

– channels for studying structural material behavior in gas media within a temperature range from 40 to 1200 °C;

– 8 horizontal experimental channels equipped with neutron diffractometers and spectrometers for studying polycrystalline and monocrystalline materials.

Experiments can be conducted within a wide range of temperatures, pressures and magnetic fields. The reactor is used to produce radionuclides, i.e. ^{14}C, ^{32}P, ^{33}P, ^{192}Ir, etc.

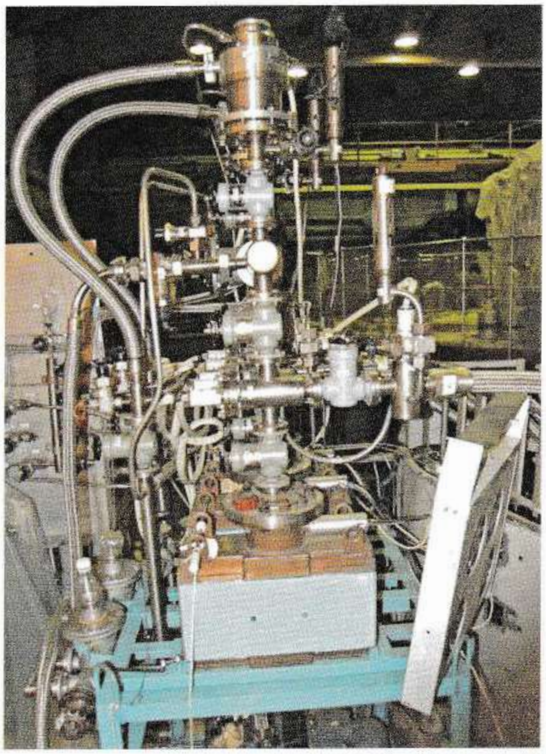
Power generating channel tests. SKAT-6-type devices with thermionic power generating channels for two-mode space nuclear facilities are tested at IVV-2M reactor.

Tests are conducted at the reactor test benches which are collectively called PURS. These test benches are equipped with state-of-art gas-vacuum devices, load-diagnostic device, measurement and gamma-ray spectrometry systems to record fission product release during the tests.

PURS test benches are a complex of several sophisticated test benches ensuring continuity of the unique tests at specified parameters throughout a year. More than 80 various parameters can be maintained and recorded once a second.

Load-diagnostic device allows electrical parameters of power generating channels to be maintained within a wide range of power.

Gamma-ray spectrometric system makes it possible to obtain on-line information on activity and composition of inert radioactive gases which release from fuel in power generating channels in various test modes.



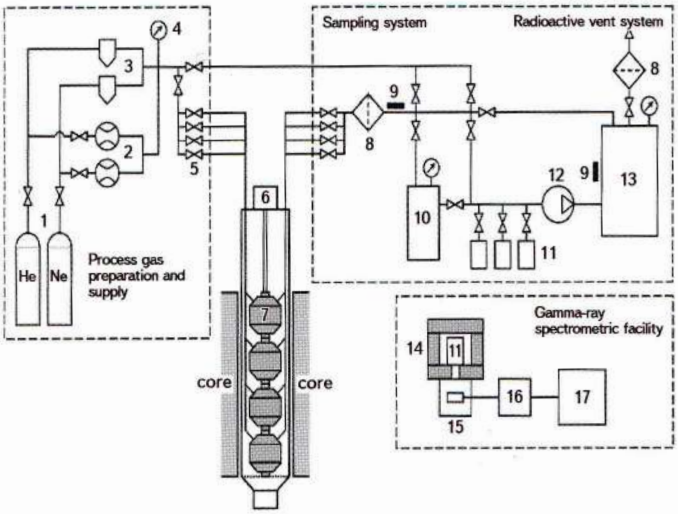
PURS test bench

When testing SKAT-6-type channels the neutronics are measured and calculated, and the data on thermal power of power generating channels as well as on neutron and gamma fields are obtained.

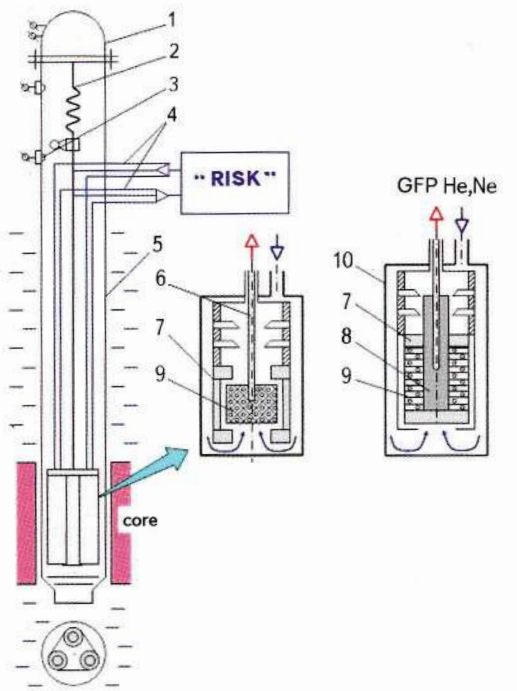
After in-pile tests the power generating channels are transferred to hot cells where post-irradiation examinations of structural materials, fuel and power generating channel elements are conducted as per test programs.

Fuel and fuel composition tests. RISK test bench was developed to conduct in-pile tests of fuel pellets, fuel element mockups, spherical fuel elements and coated particles. This facility is used to test the behavior of aforesaid objects under close-to-reactor conditions (i.e. thermal load on fuel elements, fuel cladding temperature, fuel temperature, burnup rate and fluence rate are close to real conditions).

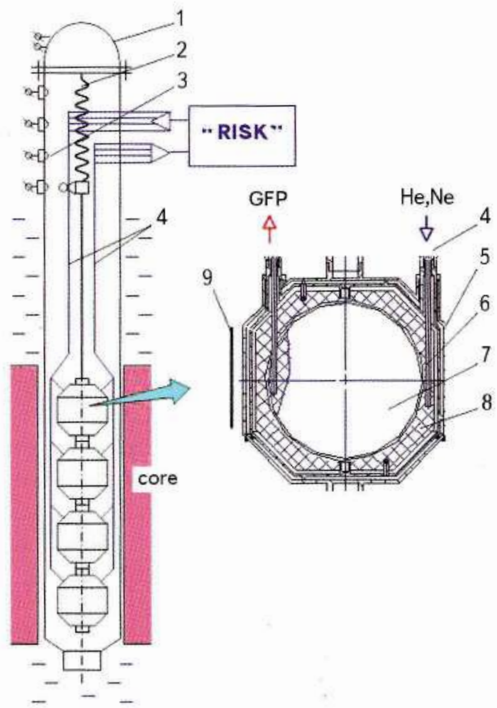
To test fuel compositions the ASU-8-, ASU-18- and Vostok-type facilities are used. They are instrumented ampoule-type test facilities



RISK block scheme:
1 – cylinders with process gases (helium, neon); 2 – flow meters; 3 – gas supply dosers; 4 – pressure sensor; 5 – "gas rake" set; 6 – device for capsule displacement along the core axis; 7 – capsule with samples under tests; 8 – iodine filter; 9 – dosimeter; 10 – 20-liter accumulation tank; 11 – measurement samplers with capacity of 0.27 l each; 12 – backing pump; 13 – cooling tank with capacity of 200 l; 14 – lead shielding (with collimating lens, if required); 15 – GC2018 gamma-ray detector produced by U.S. Canberra Inc.; 16 – PU-G-1K2 preamplifier; 17 – PC with input and output peripherals



ASU-8 instrumented ampoule-type test facility:
1 – electric drive; 2 – screw-type gear; 3 – terminal switch; 4 – gas ducts; 5 – channel case; 6 – thermocouple; 7 – thermal insulation; 8 – pin; 9 – graphite disk with coated particles; 10 – capsule casing



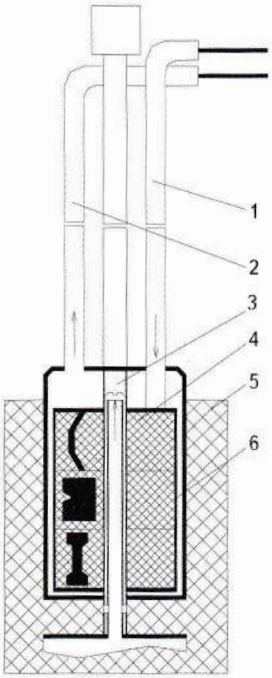
Vostok instrumented ampoule-type test facility:
1 – electric drive; 2 – screw-type gear; 3 – terminal switch; 4 – gas ducts; 5 – capsule; 6 – thermocouple; 7 – spherical fuel element; 8 – graphite thermal insulation; 9 – thermal neutron detector

and make it possible to monitor gaseous fission product release from the samples, the latters dimensions, mechanical stress, temperature and neutron flux.

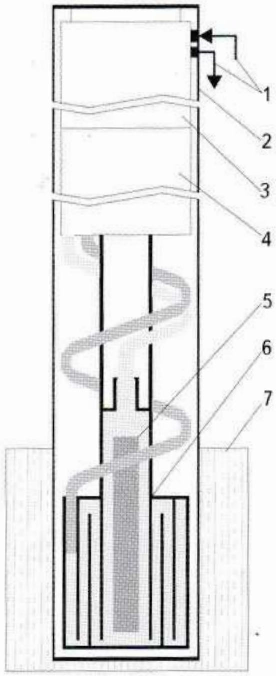
RISK test bench provides: 1) high-precision dosed supply of inert gases (i.e. helium, neon, krypton, argon) to the test facility and 2) high-accuracy measurements of activity of fission products released using state-of-art gamma-ray spectrometric equipment.

IVV-2M reactor was used to implement the programs which made it possible to justify advantages of uranium-erbium fuel for RBMK reactors, license fuel for such high-temperature gas-cooled reactors as HTR-10 (China) and PBMR (the SAR).

Facilities for testing HTGR fuel and fuel elements as well as in-pile tests together with pre-irradiation and post-irradiation examinations are the best in the world practice.



IVV-2M irradiation device (one of the options):
1 – medium supply pipe; 2 – process gas pipe; 3 – central cooler; 4 – mounting device with samples; 5 – core; 6 – ampoule



IVV-2M corrosion test device:
1 – steam inlet and outlet; 2 – vessel; 3 – biological shielding; 4 – transfer mechanism; 5 – mounting device with samples; 6 –steam reheater; 7 – core

Structural material tests

To test structural materials under ionizing radiation the reactor is equipped with a set of test facilities and devices which make it possible to conduct complex in-pile tests of structural material samples and product elements at undisturbed neutron flux of up to $2\cdot10^{14}$ $cm^{-2}\cdot s^{-1}$ (E>0.1 MeV) and duration of test run of up to 7000 h a year. Irradiation facilities and devices for in-pile tests can be placed in FA cavity which is 30 mm in diameter, core periphery cavity with diameter of 60 mm or central cavity with diameter of 120 mm.

The reactor is used to:

– irradiate samples in liquid nitrogen;

– irradiate samples in inert gas media and in vacuum in the range of temperatures from 60 to 1500 °C;

– conduct corrosion tests in liquid and gas media in the range of temperatures from 30 to 1000 °C;

– conduct mechanical tests of structural material samples.

Irradiation devices

To irradiate samples in inert gas media and in vacuum the devices which can house samples of various (i.e. flat, spherical, figured and tubular) shape are used. Conditions required for reaching specified neutron fluence are provided by positioning these devices in various points of the reactor core.

Test temperature is achieved due to radiation heating-up of the samples in ionizing radiation flux or use of electrical heaters. Irradiation device arrangement and size can vary. Temperature, duration of test and medium composition (rarefaction) in the working volume are monitored during these tests. Thermal behavior of the samples can be controlled and maintained within the specified range by the composition of the medium (inert gas volume-to-volume ratio).

Corrosion test device for testing materials in superheated steam flow

Corrosion test device for testing materials in superheated steam flow makes it possible to simulate abnormal operation of structural elements of the reactor at temperature of up to 1000 °C and steam flow velocity of up to 100 m/s.

Humidity chamber is housed in the reactor core. Water is supplied to deaerator, then to steam generator and to working chamber. Steam is reheated by radiation heater. Temperature, medium composition and flow rate are monitored during the tests.

Mechanical test facility

IVV-2M reactor has a complex of facilities which make it possible to conduct short- and long-term mechanical tests of materials when samples are subject to uniaxial tensile and compressive stresses, and tubular samples are subject to internal gas pressure. In case of uniaxial stresses the deformation data are continuously recorded, and it is possible to test up to 6 samples at a time. When tubular samples are subject to internal gas pressure the deformation data are periodically recorded by comparing the length of the sample to a gage.

OR facility

OR facility is designed for conducting short- and long-term uniaxial tensile tests. It is housed in the vertical channel, the lower part of which is located in the reactor core.

The facility has module design and consists of transfer mechanism, loading mechanism and irradiation device with samples and linear movement detectors.

Design of OR facility makes it possible to conduct long-term tests of samples at constant speed of operating arm movement or at stresses maintained as per the specified program. In case of short-time tests the sample is loaded into the reactor, irradiated to the specified fluence and then strained at constant speed till it is destructed.

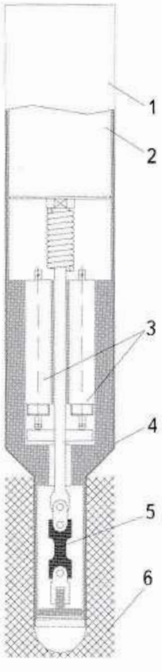
The facility allows samples to be tested at temperature of up to 1500 °C, stress of up to 2000 N and neutron flux of up to $2\cdot10^{14}$ $cm^{-2}\cdot s^{-1}$ (E>0.1 MeV). Temperature, stress and deformation are monitored during the in-pile tests.

SIGNAL facility

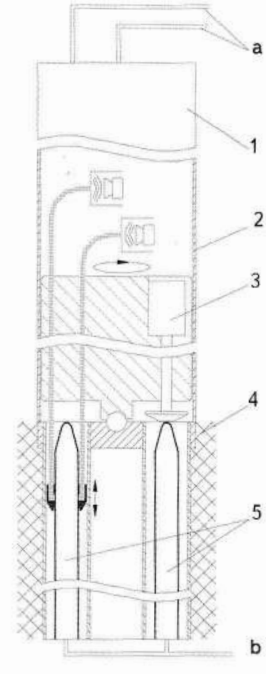
SIGNAL facility for long-term test of samples stressed by internal gas pressure is designed to study deformation of fuel claddings due to intermittent stresses. The basis for measurement is a method of comparison between length and diameter of the reference sample and those of the studied sample during the irradiation. Five fuel cladding samples with external diameter of 9.13…9.15 mm and active length of 400 mm are used in tests. The channel is 60 mm in diameter and filled with helium. Neutron flux reaches $1.5\cdot10^{14}$ $cm^{-2}\cdot s^{-1}$ in the central portion of sample height and $0.4\cdot10^{14}$ $cm^{-2}\cdot s^{-1}$ on the periphery of the samples.

Temperature increases smoothly from 50°C on the periphery to 320°C in the center throughout the length of the samples.

Temperature, acoustic signal response, actual position of fuel cladding sample and gas pressure are monitored during the tests.



OR experimental facility:
1 – transfer mechanism; 2 – loading mechanism; 3 – linear movement detectors; 4 – irradiation device; 5 – sample; 6 – core



SIGNAL facility:
1 – mechanical arm; 2 – channel tube assembly; 3 – acoustic cantilever; 4 – core; 5 – samples to be irradiated; a, b – loading system

Circulation facilities

LP loop

Low pressure loop is designed for testing structural and fuel materials in specified water composition at coolant temperature from 60 to 100°C and water flow rate of 4 m^3/h at most. The facility is used to test LEU fuel composition for research reactors. Coolant flow rate, temperature and medium composition are monitored during the tests.

HP loop

High-temperature high-pressure water loop has been developed to test fuel claddings at coolant temperature of 300 °C and coolant pressure of up to 20 MPa. The facility allows corrosion tests of VVER and RBMK fuel claddings to be conducted at the specified water chemistry parameters. It is possible to monitor temperature, pressure, flow rate and water chemistry.

Prospects for developing IVV-2M test facilities

To justify heavy liquid metal cooled reactor design approaches, an in-pile loop has been developed. It is liquid metal-cooled (C00 lead) non-isothermal circulation loop equipped with a few test areas where samples can be placed for corrosion tests as well as with gas loop and systems which maintain specified loop operation parameters.

The loop simulates conditions of operation of heavy liquid metal cooled reactor. It makes it possible to study coolant activation, transfer of radionuclides along the primary circuit, their release into gas cavity and deposition on structural elements of the reactor core.

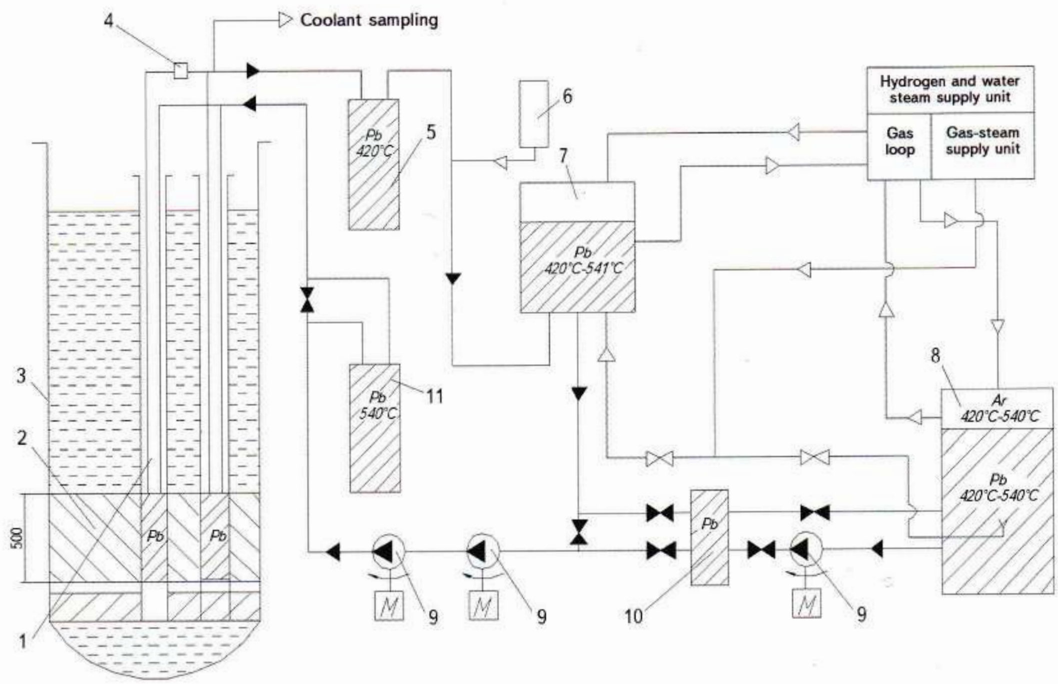
Structural material corrosion tests are conducted under simultaneous exposure to neutron flux and liquid metal coolant flow during loop operation.

Main areas of studies

Reactor material tests:

– tests of materials and structural elements of various-purpose power reactor cores in steady and dynamic modes;

– in-pile irradiation and post-irradiation examination of candidate materials to be used in structural elements of ITER blanket module and simulation of its operation conditions.



Principal diagram of lead-cooled loop:
1 – irradiation device; 2 – core; 3 – IVV-2M reactor vessel; 4 – radiation detector; 5 – area 2 with working temperature from 450 to 540 °C; 6 – oxygen supply unit (mass-transfer device); 7 – pressurizer; 8 – bubbler column; 9 – circulation pump; 10 – mixer; 11– area 1 with working temperature from 420 to 450 °C

Fuel tests:

– in-pile tests of fuel pellets and fuel element mockups to be used in VVER and RBMK light water reactors;

– in-pile tests and post-irradiation examination of coated particles, spherical fuel elements and coated particles to be used in high-temperature gas-cooled reactors;

– in-pile tests of U-Mo dispersion fuel as per program of conversion of research reactors to LEU (in terms of ^{235}U).

Justification for operability of space nuclear power facilities:

– in-pile tests of thermionic power generating channels for two-mode space nuclear facilities, where thermionic power generating channels are housed in SKAT loop channel. Post-irradiation examination of power generating channel elements.

Radionuclide production:

– production of iridium-192 and carbon-14.

Reactor safety:

– in-pile tests and post-irradiation examination of physical protection materials (concrete-based compositions);

– in-pile tests of optical fiber detector mockups for measuring temperature and deformation of RBMK graphite column.

Advanced researches:

– assessment of radionuclides release from BREST reactor coolant and its structural material tests;

– development of tritium release modes for experimental module of ITER blanket;

– tests to justify operability of prospective fuels and graphite materials to be used in high-temperature gas-cooled reactors with increased parameters:

- fuel burnup > 13 %;
- fast neutron fluence (E>0.1 MeV) - up to $5 \cdot 10^{25}$ m^{-2};
- rated fuel temperature > 1300 °C;
- emergency fuel temperature > 1600 °C;

– in-pile tests and post-irradiation examination of fuel compositions and pilot fuel elements for investigating the possibility of thermal neutron reactors conversion to Th-U fuel cycle.

Main activities

Reactor utilization factor came to 0.828 in 2011.

Operability of elements of space nuclear power facilities was studied.

Production of carbon-14, iridium-192, phosphor-34 and phosphor-35 was carried out.

Reactor CPS modernization project is planned to be developed and implemented.

Reactor radioactive vent system retrofitting project is planned to be developed and implemented.

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