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Research Article

Conducting in-reactor studies of objects the length of which exceeds the height of the IVV-2M reactor core

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Abstract

When solving several research problems, the need arises for reactor testing of non-standard objects. The IVV-2M reactor is a pool, heterogeneous, light-water reactor. The 500 mm-high reactor core comprises six sections of six fuel assemblies (FA) with central water traps and a water trap in the center of the core. The personnel of INM JSC were tasked with testing an object whose length exceeded the height of the IVV-2M reactor core. A non-standard core layout was developed with six sections of seven fuel assemblies to accomplish this task.. The test object is located in the center of the core, surrounded by special aluminum blocks. Several fuel assemblies are installed on special supports, which allow the fuel assemblies to be installed 200 mm higher. Shim rods have been moved to the center of the sections. The safety of the developed arrangement was confirmed by neutron-physical calculations using the certified MCU-PTR software tool. Calculations have shown that the maximum maximum power in the fuel assembly is 236 kW, which is 50% less than the permissible value. To confirm the safety of the layout and ensure the required test conditions, tests of a mock-up of the object were carried out. The efficiency of the control members was experimentally determined. The overall efficiency of the control members was 4.85% Δk/k, the physical weight of the object was 1.46% Δk/k. Based on the requirements of the Nuclear Safety Rules for Research Reactors, the test duration is no more than three days. The distribution of neutron flux density over the height of the core was determined using indium neutron activation detectors. The developed design solutions made it possible to form a unique IVV-2M reactor core layout for testing objects exceeding its height.. Neutron-physical calculations and tests of the irradiated device mock-up confirmed the safety and performance of this core arrangement.

Keywords

research reactor, core layout, fuel assembly, water trap, neutron flux density, neutron activation detector

Introduction

Currently, most nuclear research reactors (RR) worldwide are multifunctional and designed to solve a wide range of scientific and industrial problems. The scientific direction includes research in the field of materials science, biology, nuclear physics, radiation chemistry, radiation, nuclear and reactor technologies; nuclear materials and full-scale products with fissile substances; studies of the influence of reactor radiation on devices, sensors, semiconductor materials, and various equipment (Androsov et al. 2015; Russkikh et al. 2015; Tashlykov et al. 2023; Van Thuong et al. 2023). The industrial use of RRs is mainly associated with producing isotope products for medical and industrial purposes (Egorov et al. 2012; Litvinov et al. 2020; Tashlykov et al. 2017; Zlokazov et al. 2017).

In order to accomplish the tasks listed above, it is necessary to develop experimental or irradiation devices that include the test object of irradiation. In this case, it is necessary to take into account the features and capabilities of the reactor core, where the device being developed is supposed to be located, depending on the specified test parameters. In several cases, it is not possible to design a device for a specific reactor core, which is primarily due to the device's geometric characteristics, such as diameter, length of the working section of the device (test object), etc.

The task of providing conditions for conducting in-reactor studies of objects with a length of more than 500 mm, exceeding the height of the IVV-2M reactor core, was solved at INM JSC by creating a special layout for the IVV-2M reactor core. At the same time, the main requirement of the tests is ensured - creating a symmetrical neutron field relative to the central horizontal plane of the test object with a given coefficient of uneven distribution along the object's height.

Description of IVV-2M research reactor

The IVV-2M reactor is a pool-type light-water research reactor with a rated power of 15 MW. The reactor pool is located in a concrete mass, covered from the inside with a sealed stainless steel shell (Figs 1, 2). The pool in horizontal section is an ellipse with axial dimensions of 1800×4300 mm and a height of 8000 mm (Russkikh 2017).

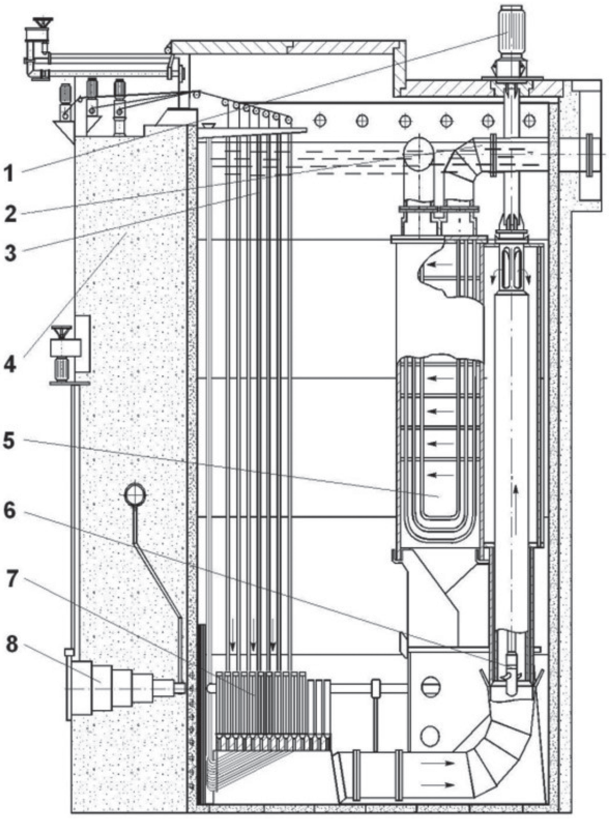

Figure 1. Design of the IVV-2M reactor (vertical section). 1 – pump electric motor; 2 – pipeline of the second cooling circuit; 3 – control rod drives; 4 – concrete mass; 5 – heat exchanger; 6 – pump; 7 – active zone; 8 – horizontal experimental channel.

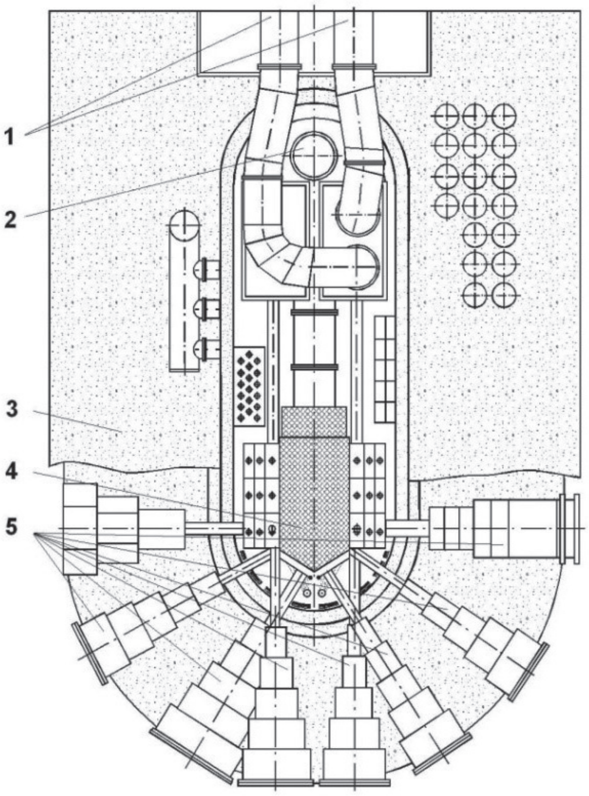

Figure 2. Design of the IVV-2M reactor (horizontal section). 1 – pipelines of the second cooling circuit; 2 – heat exchanger; 3 – concrete mass; 4 – active zone; 5 – horizontal experimental channels.

The heat exchanger with a built-in circulation pump is located in the reactor basin. Water is sucked in by a circulation pump from under the base plate of the core and, passing through the inter-tube space of the heat exchanger, returns to the pool space above the core (top-down coolant flow pattern). Water from the second cooling circuit circulates through the heat exchanger pipes.

The reactor is equipped with ten horizontal (six radial and four tangential) and two vertical experimental channels.

At the bottom of the reactor pool there is an aluminum alloy base plate, which has 186 mounting holes arranged in a triangular pattern with a pitch of 64 mm. The core is formed by installing hexagonal fuel assemblies (from 36 to 42 pieces), hexagonal beryllium reflector blocks, experimental and irradiation devices into the base plate.

The fuel assemblies consist of five tubular hexagonal fuel elements mounted coaxially between hexagonal shroud tubes. The structural material of the fuel cell assemblies and shells is aluminum alloy. The fuel layer is uranium dioxide dispersed in an aluminum matrix (Aden et al. 1986).

The fuel assemblies installed in the base plate form six sections. A cavity is formed in the center of each section, intended to install experimental and irradiation devices with an outer diameter of up to 60 mm. It is possible to form cavities with a diameter of 105 mm, 123 mm and

130 mm from shaped beryllium blocks, both in the center of the active zone and in the reflector (Fig. 3a). In addition, it is possible to form a cavity with a diameter of up to 400 mm in the active zone reflector by removing part of the beryllium blocks (Fig. 3b).

Each fuel assembly also has a cavity formed by an internal hexagonal boot pipe with an inscribed diameter of 29.1 mm. Various irradiation devices are installed in these cavities. In the cavities of some fuel assemblies, liners of the control members (CM) are placed. The core cells for placement of the CM are strictly defined for standard configurations and are presented in Fig. 4 and Table 1.

The total efficiency of shim rods is on average 7.5%Δk/k, emergency shutdown rods - 1.7%ΔK/K, automatic control rod - 0.4%Δk/k.

The block structure of the core and reflector, the uniformity and interchangeability of their elements provide flexibility in design solutions and physical parameters when conducting scientific experiments and carrying out production activities. Up to 60 experimental and irradiation devices are simultaneously irradiated in the reactor core and reflector.

The high neutron multiplication factor and uranium-water ratio of the core, as well as the presence of a beryllium reflector, provide a high neutron flux density and a relatively hard neutron energy spectrum. The main technical and physical characteristics of the IVV-2M reactor are given in Table 2.

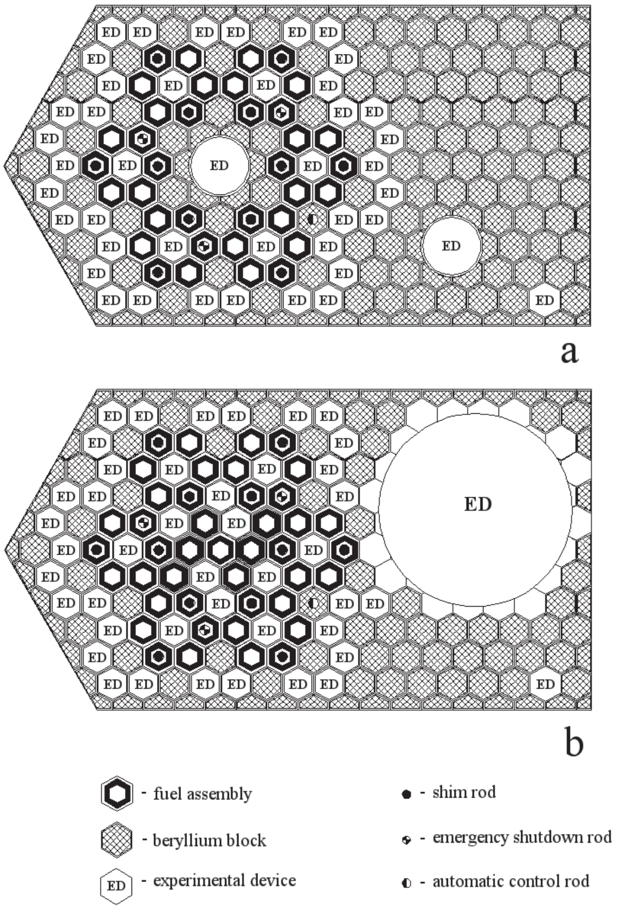

Figure 3. IVV-2M reactor core layout.

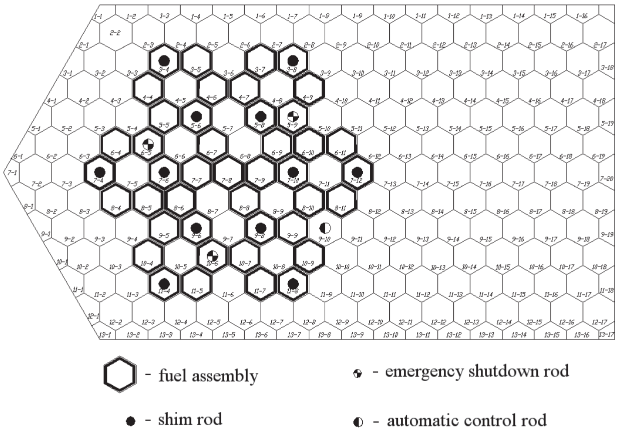

Figure 4. Cartogram of the installation of the CM with standard configuration.

Table 1. Locations of the CM in the core of the IVV-2M RR with standard configurations

Control member (CM)		CM installation cell
Control and protection system (CPS) shim rods (SR)		
Group No. 1	SR1-1	3-4
	SR1-2	5-6
Group No. 2	SR2-1	3-8
	SR2-2	5-8
Group No. 3	SR3-1	9-8
	SR3-2	11-8
Group No. 4	SR4-1	9-6
	SR4-2	11-4
Group No. 5	SR5-1	7-10
	SR5-2	7-12
Group No. 6	SR6-1	7-4
	SR6-2	7-6
Emergency shutdown rods (ES)		
ES-1		10-6
ES-2		5-9
ES-3		6-5
Automatic control rod (AC)		
AC		9-10

Table 2. Main characteristics of the IVV-2M reactor

Parameter	Meaning
Reactor thermal power, MW	15
1st circuit coolant	Chemically demineralised water
Coolant flow in circuit 1, m^3/h	1200
Coolant flow through fuel assemblies, m^3/h	19
Heat removal surface of one fuel assembly, m^2:	0.796
Coolant pressure drop across the core, MPa	up to 0.055
Coolant pressure at the entrance to the core, MPa	0.16
Coolant temperature at the entrance to the core, °C	45
Heating the coolant in the core, °C	up to 28
Core height, mm	500
Maximum power of fuel assembly, MW	0.47
Maximum fuel element temperature, °C	94
Maximum neutron flux density, $1/(cm^2 \cdot s)$:	
– thermal	$5 \cdot 10^{14}$
– fast (E > 0.1 MeV)	$3 \cdot 10^{14}$
Reactor utilization time factor, %	80-90

The new proposal design

To solve the problem of ensuring research of test objects with a length of more than 500 mm at the IVV-2M reactor, a core layout is proposed, a cartogram of which is shown in Fig. 6. The core consists of 42 IVV-2M fuel assemblies. The fuel assemblies are combined into six sections of 7 fuel assemblies. The seventh fuel assembly of each section is installed in the central cell of the section, used in the standard core layout as a water trap for producing radioactive isotopes and conducting various studies. A device with a test object more than 500 mm long is placed in a central water trap with a diameter of up to 192 mm, formed using special aluminum blocks.

The external rods of the groups of shim rods (SR) remain installed in their standard places, the internal ones (closest to the central cell of the core) are rearranged into the fuel assemblies located in the central cells of the corresponding sections. The working elements of emergency shutdown and the working element of the automatic control (AC) are installed in standard cells.

Part of the core fuel assemblies (15 pieces) adjacent to the device with a test object more than 500 mm long are mounted on special aluminum supports developed and manufactured at INM JSC. The diagram of the stands is shown in Fig. 5.

The stand is an aluminum hexagon with a wrench size of 63 mm, in which there is a hole for installing fuel assemblies into it. A standard shank of core elements (fuel assemblies, beryllium blocks, plugs, etc.) is attached to the stand for installation in the reactor support grid. This design allows the fuel assemblies to be placed 200 mm higher than with standard installation in the support grid, which in turn provides test conditions for the height distribution of the neutron field.

The most important requirement when conducting research is ensuring nuclear safety. The main conditions for compliance with nuclear safety requirements when installing fuel assemblies on supports are reliable fixation of the fuel assemblies in the support and the support itself in the support grid to prevent spontaneous movement of the fuel assemblies; the necessary efficiency of the control members; reliable cooling of fuel assemblies with primary coolant; not exceeding the permissible values of energy release in the fuel assembly.

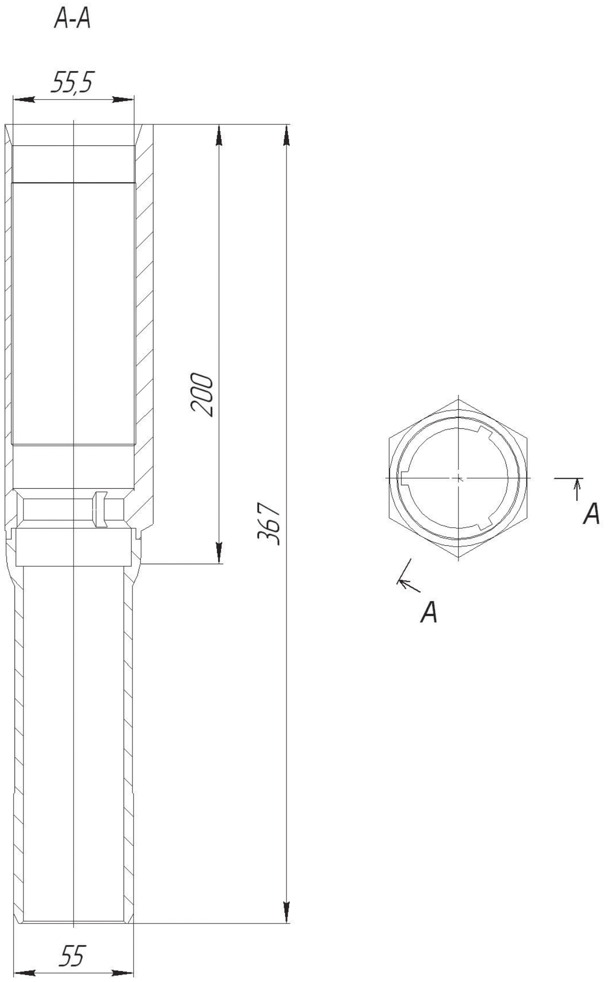

Figure 5. Aluminum stand for raising the fuel assembly by 200 mm, relative to its standard position in the core of the IVV-2M reactor.

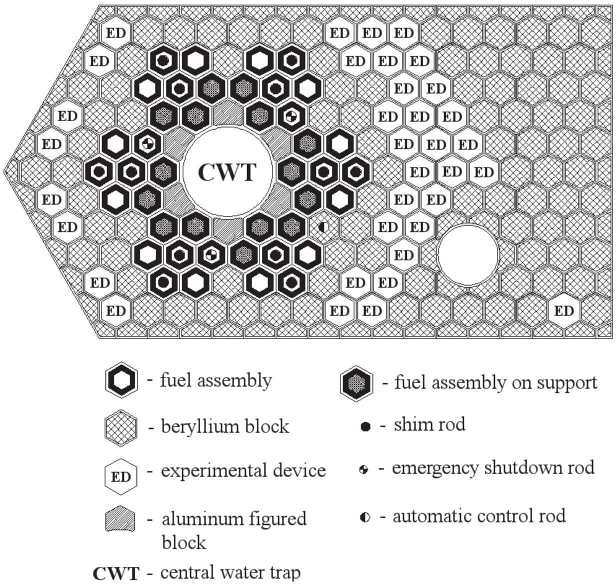

Figure 6. Cartogram of the IVV-2M research reactor core during research of a device with a test object more than 500 mm long.

MCU-PTR software description

MCU-PTR program (Alekseev et al. 2010, 2011) is designed to simulate the processes of transfer of neutrons, photons, electrons, and positrons using the Monte Carlo method based on assessed nuclear data in systems with three-dimensional geometry, taking into account changes in the nuclide composition of materials during interaction with neutrons. For all of the above particles, the inhomogeneous particle transport equation is solved, and for neutrons, the program also allows you to solve the homogeneous equation. Mathematically, this means that for the system under consideration, a kinetic equation with given boundary conditions is solved, describing the distribution of the particle flux in it. At the same time, the effective neutron multiplication factor, energy release distributions across fuel assemblies and individual fuel elements, the effective fraction of delayed neutrons, particle fluxes, and other functionalities are recorded.

The MCU-PTR program has a modular structure and is composed of the following modules:

- control module performing monitor functions;
- transport module, which models the trajectories of particles in the system;
- a composite physical module designed to simulate the interaction of particles with matter based on the MDBPT50 library data;
- a geometric module that models straight sections of trajectories between collisions;
- source module, which models the phase coordinates of source particles (or zero-generation neutrons when solving criticality problems);
- registration module, which allows calculating a wide range of neutron flux functionals;
- registration module for burn-up, which provides calculation of neutron flux functionals for the burn-up module;
- a hardware module that includes programs that may depend on the type of computer and operating system;
- burn-up module, designed to calculate changes in the isotopic composition of reactor materials during its operation.

The constant support of the MCU-PTR program consists of the MDBPT50 data bank. The neutron interaction-related sections of the MDBPT50 bank are listed below:

- ACE/MCU - a library of cross-sections and other characteristics of the interaction of neutrons with nuclei in the epithermal region of neutron energy. Here, the information from the estimated neutron data files is recorded in a point-by-point representation. ACE was generated using the NJOY program, mainly from ENDF/BVII.0 files;
- BNAB/MCU - an extended and modified version of the 26-group constant system. BNAB-94. The resonance characteristics of cross-sections are described using subgroups;
- MULTIC - 301-group library containing, among other things, data on the temperature dependence of subgroup parameters of nuclides in the region of unresolved resonances;
- LIPAR - a library of resonance parameters of nuclides in the region of allowed resonances. Allows you to obtain cross-sections at a given energy, taking into account the temperature of the material;
- KORT – a library of neutron cross-sections in the low energy region (upper limit - 5 eV) in point-by-point representation;
- TEPCON - 40-group cross-sections of neutrons in the thermalization region . The upper energy limit is 1 eV. Used in modeling neutron history in the group transport approximation;
- VESTA – a library of scattering laws of slow neutrons S(α,β) of various moderators. Used to model the neutron history in the thermalization region, taking into account the correlation between the energy change and the scattering angle. The data are presented in the form of probability tables in the energy region up to 4.65 eV;
- BOFS – a library of generalized phonon spectra of moderators. Used in calculating the law of scattering of slow neutrons;
- ABBNL – a library of 63 group cross-sections, necessary to obtain cross-sections of the "total isotope";
- BURN5 - a library for solving the burn-up problem with: half-lives of nuclei, yields of fission products, chains of radioactive transformations, etc .;
- PHOTONS – a library of multigroup cross-sections for photon generation during the interaction of neutrons with matter;
- PHOTONT – a library of multigroup cross-sections for the interaction of photons with matter. The physical module allows one to take into account the effects of continuous changes in particle energy during collisions, as well as both continuous and stepwise dependence of cross-sections on energy. Some features of modeling only neutron interactions are described below;
- DOSIM – a library of activation cross-sections in point-by-point representation.

The geometric model of the IVV-2M reactor core was built in accordance with the design documentation. The horizontal experimental channels, graphite reflector, concrete mass, support grid and stainless steel lining of the reactor pool have a simplified representation, which does not affect the criticality of the system and the distribution of neutron flux density in the core. Drawing the remaining equipment of the reactor tank is impractical.

The effective neutron multiplication factor was K_{eff} = 1.0012.

The difference between K_{eff} and 1 is due to the inaccuracy in determining the concentration of poison nuclei in the beryllium blocks of the core.

In the process of modeling the history of a neutron or photon, the registration module on each straight segment of the trajectory records its length, the coordinates of the particle collision, the nucleus on which the collision occurred, the type and result of the reaction. Next, averaging is carried out over all histories and the spatial-energy distribution of the neutron and photon flux is determined. For neutrons, linear and bilinear flux functionals are also calculated - the effective multiplication factor K_{eff}, energy release distribution, reaction rates and group cross-sections on individual nuclei, the lifetime of fission neutrons, the effective number of delayed neutrons, the probability of leakage, etc. Together with the average values of the quantities, their statistical errors (variances) are also calculated. The functionals are calculated using standard estimates used in the Monte Carlo method. Estimates based on collision points, absorption points, and travel lengths are used.

The heat dissipation in the MCU - PTR is calculated using the burn-up module, the results are given in the N. REZ file for each given material in kW/cm^3

The uncertainty of the simulation results for the Monte Carlo method is determined by the number of histories, and as a consequence, the calculation time. In this case, the statistical error in calculating heat release is 2%, $\delta K_{eff} = 0.0001$.

The experimental measurement uncertainty is 10%

The maximum relative difference in the efficiency of the control members between calculation and experiment is about 5%.

Discussion of results

In order to confirm the neutron-physical conditions for the altitude distribution of the neutron field, a mock-up of the device was made at the test object, which is an aluminum cylinder in which two through holes-channels are made parallel to the axis for measuring the neutron flux density. The mock-up was placed in the central water trap of the IVV-2M reactor core (Fig. 6). Activation detectors were used to measure the neutron flux density. The fast neutron flux density was determined by the threshold reaction $^{115}In(n,nr)^{115m}In$. The effective energy threshold for the reaction is $E \geq 1.15$ MeV. The measurements were performed at a reactor power of 1 MW.

The relative distribution of fast neutron flux density is presented in Fig. 7. The maximum neutron flux density is observed at a height of ~32 cm from the bottom of the core.

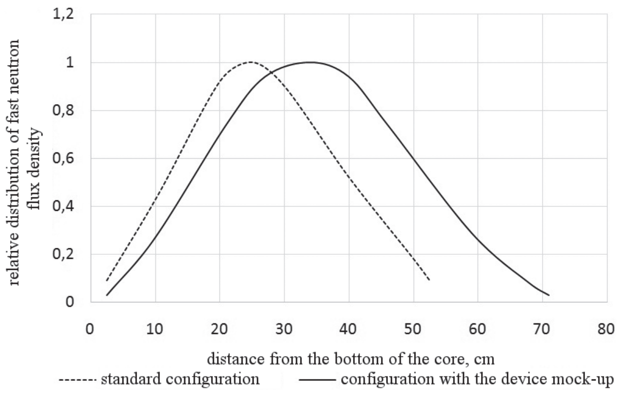

Figure 7. Relative distribution of fast neutron flux density in the central water trap along the height of the IVV-2M research reactor core.

In order to determine the effectiveness of the CM during the first configuration of the core for studying a device with a test object more than 500 mm long, the SR and AC rods were calibrated. When calibrating the SR and AC rods, the prototype of the device was loaded into the central water trap of the reactor core. Before loading the prototype and after loading it, the reactor was brought into a critical state with the position of the shim rods and the automatic control rod fixed. The positions of the control members are presented in Table 3. After calibrating the rods, the reactivity introduced by the device when placed in the reactor core was determined.

Table 3. Position of the CM before loading and after loading the device mock-up, cm

	SR1	SR2	SR3	SR4	SR5	SR6	AC
Before loading the device mock-up	0	0	0	0	0	0	28
After loading the device mock-up	15	15	15	15	15	15	27

At the time of calibration, there was no loss of reactivity for xenon-135 poisoning; according to calculated data, for samarium-149 poisoning it was ~1.5%Δk/k. The efficiency of the SR and AC rods was determined by the acceleration and compensation method. The essence of the method is to create a supercritical state of the reactor by moving one group of SR, determining the introduced reactivity determining the introduced reactivity throughout doubling the reactor power, and compensating and compensating for the introduced reactivity by moving another group of SR to the critical state of the reactor. At the end of the calibration, differential and integral characteristics of the shim rods and the automatic control rod are compiled.

The total efficiency of the groups of shim rods and the automatic control rod measured for the presented core layout is given in Table 4. According to the data obtained during calibration, the reactivity introduced by the device layout is +1.46%Δk/k.

Table 4. Efficiency of control members (CM)

CM	SR 1	SR 2	SR 3	SR 4	SR 5	SR 6	AC	Sum
eff ., %ΔK/K	0.57	0.73	0.74	0.81	0.80	0.80	0.40	4.85
eff ., %ΔK/K (calculation)	0.55	0.72	0.75	0.8	0.82	0.84	0.41	4.89

SR – Shim rod; AC – Automatic control rod.

The efficiency of the control members obtained by calculation and experiment differs within 5%.

The total compensating capacity of the CPS (without taking into account the efficiency of the ES rods) for this core layout was 4.85% Δk/k, or 6 β_{eff}. It follows that, taking into account the requirements of the Nuclear Safety Rules for Research Reactors (NP-009-17 2017) to ensure the subcriticality of the reactor in the operating mode of at least 1%Δk/k, the maximum reactivity margin was 3.85%Δk/k. Accordingly, the operational reactivity margin when the reactor is operating at a power of 8 MW, taking into account the loss of reactivity due to xenon-135 poisoning (~3.3%Δk/k), will be ~0.55%Δk/k. In this case, the loss of reactivity due to fuel consumption is ~0.05%Δk/k per day. The duration of research on a device with a test object longer than 500 mm will be ~ 3 days.

Reliable fixation of the fuel assembly in the stand is ensured by the design of the stand. The hole in the stand for installing the fuel assembly is made similar to the hole in the base plate, and the standard shank of the core elements for installation in the base plate is attached to the stand itself. In addition, the movement of the coolant of the primary circuit of the IVV-2M reactor is implemented according to a top-down scheme. The combination of this implementation of coolant movement and the design of the stand ensures reliable installation of the fuel assembly with the stand into the base plate, excluding the possibility of arbitrary movement and ascent of the fuel assembly.

Installing a fuel assembly on a support should not impair the heat removal mode from the fuel assembly. For these purposes, at the stage of development of the stand, its design was agreed upon with the general designer of the IVV-2M reactor, NIKIET JSC. The corresponding hydraulic calculations were carried out to justify the safe operation of the reactor core when installing the fuel assembly on a support, and it was also experimentally confirmed that the hydraulic regime of the coolant when removing heat from the fuel assembly under such conditions is not disturbed.

At the stage of calculating the neutron-physical characteristics of the reactor core during testing of the device prototype, the distribution of heat release throughout the fuel assemblies in the core was determined. The calculation of neutron-physical characteristics was carried out using a certified precision software tool MCU - PTR with the MDBPT 50 data bank (certification passport No. 498 dated December 14, 2020). The MCU-PTR program (Certificate of the program 2020), assembled from modules of the MCU5 package, is designed for precision modeling of neutron and photon transfer processes using analog and weight (non-tax) Monte Carlo methods based on assessed nuclear data in nuclear reactors, taking into account changes in the isotopic composition of reactor materials during the campaign (Alexeyev et al. 2010).

Fig. 8 shows the absolute values of heat release (above in bold in kW) and the relative distribution of heat release (below in italics) in the fuel assemblies at the reactor thermal power of 8 MW at the locations where the fuel assemblies are installed. When justifying the safety of operation of a research reactor, it was established that for one fuel assembly the maximum calculated heat release, taking into account the axial and azimuthal unevenness of distribution throughout the fuel assembly, should not exceed 470 kW.

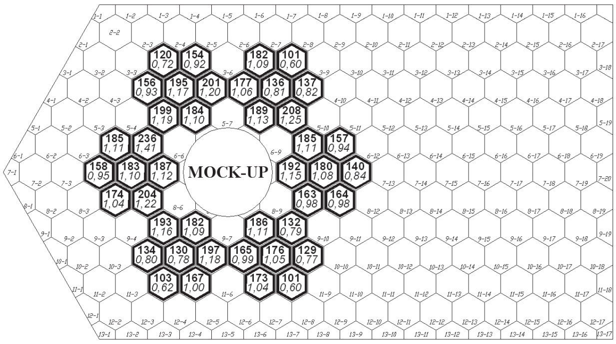

Figure 8. Distribution of heat release across fuel assemblies during the layout of the IVV-2M reactor core for testing a device prototype at a reactor thermal power of 8 MW.

From the data presented in Fig. 8 it can be seen that the maximum heat release in the fuel assembly for this core configuration is 50% less than the permissible one.

Conclusion

To conduct in-reactor studies of irradiation devices with a working section length exceeding 500 mm (the height of the IVV-2M fuel assembly), a unique layout of the IVV-2M reactor core has been developed. The carried out neutron-physical calculations and experimental studies showed the possibility of ensuring the required distribution of neutron flux density along the length of the object under study. The requirements of regulatory documents to ensure nuclear safety during testing, both from the point of view of nuclear and general research reactor safety, have been confirmed. The studies of real objects confirmed the correctness of the preliminary assessments of irradiation conditions.

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