ISSN 1063-7788, Physics of Atomic Nuclei, 2015, Vol. 78, No. 11, pp. 1238–1245. © Pleiades Publishing, Ltd., 2015.
Original Russian Text © A.V. Surkov, V.N. Kochkin, Yu.E. Pesnya, V.A. Nasonov, V.I. Vihrov, D.Yu. Erak, 2014, published in Voprosy Atomnoi Nauki i Tekhniki. Seriya: Fizika Yadernykh Reaktorov, 2014, No. 4, pp. 34–42.

Experimental Investigation of Neutronic Characteristics of the IR-8 Reactor to Confirm the Results of Calculations by MCU-PTR Code

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Received January 31, 2014

Abstract—A comparison of measured and calculated neutronic characteristics (fast neutron flux and fission rate of ^{235}U) in the core and reflector of the IR-8 reactor is presented. The irradiation devices equipped with neutron activation detectors were prepared. The determination of fast neutron flux was performed using the ^{54}Fe (n, p) and ^{58}Ni (n, p) reactions. The ^{235}U fission rate was measured using uranium dioxide with 10% enrichment in ^{235}U. The determination of specific activities of detectors was carried out by measuring the intensity of characteristic gamma peaks using the ORTEC gamma spectrometer. Neutron fields in the core and reflector of the IR-8 reactor were calculated using the MCU-PTR code.

Keywords: IR-8 reactor, neutron flux, neutron activation measurement, MCU-PTR code.

DOI: 10.1134/S1063778815110162

INTRODUCTION

The IR pool-type research reactor is designed to perform a wide range of research in the fields of reactor materials science, nuclear physics, solid-state physics, and nuclear medicine. To plan irradiation experiments and form loadings in the IR-8 core which comply with security requirements for the reactor use, reliable determination of the operating parameters of the neutron field in the reactor core and prediction of their changes in time are required. A specialized precision MCU-PTR code [1] is employed for this purpose, which implements the Monte Carlo method for calculation of neutron transport. Using the MCU-PTR code, computational and experimental studies on determination of neutron fluxes in the IR-8 reactor were conducted. The main results are given in [2]. However, experimental determinations of characteristics of thermal neutron fields in the core and reflector of the reactor were not performed within these investigations. The verification of computational data on thermal neutron fields in the core of the IR-8 reactor was done on the basis of experiments carried out during power start-up of the reactor in 1981 in loading the core with "fresh" U-Al alloy [3] FAs. The results of verification of the MCU-PTR code were obtained using those from experiments upon the power start-up of the IR-8 and are shown in [4]. Because the available experimental evidence on the characteristics of thermal neutron fields in the core and reflector of the reactor do not correspond to the real conditions under which current reactor operates, obtaining new experimental data is of great importance. This article reports on calculated and experimental data relevant to currently used reactor loadings, precisely, the fast neutron flux density and ^{235}U fission rate.

1. METHOD AND PROCEDURE OF EXPERIMENT

The IR-8 reactor has 12 horizontal experimental channels for extracting neutron beams. To carry out materials science investigations and produce radioisotopes, four vertical channels in six or four tubular FAs, eight channels in changeable beryllium blocks, 11 channels in a stationary reflector, and six channels located beyond the confines of the aluminum reactor pressure vessel can be used. The nominal power of the reactor is 8 MW. The core consists of 16 six-tube FAs of the IRT-3M type with tubular fuel elements with a square cross section. The kernel of fuel elements is made from uranium dioxide in an aluminum matrix. The ^{235}U enrichment is 90%. Metallic beryllium is used as a reflector.

In the verification experiment, ampoule devices (ADs) equipped with neutron activation detectors (NADs) were prepared. The fast neutron flux density was measured with the ^{54}Fe (n, p) and ^{58}Ni (n, p) reactions. The ^{235}U fission rate was estimated using uranium dioxide with 10% enrichment of the ^{235}U isotope. Detectors were placed in aluminum capsules, which subsequently were packed in an aluminum tube 10 mm in diameter serving as a casing for the experimental AD (Fig. 1).

Important parameters that make aluminum attractive as a constructional material are its low activation and rather short half-lives of the main radionuclides produced by irradiation. One AD contains three capsules with NAD sets for determination of the fast neutron flux and two ampoules containing uranium dioxide, one of which is wrapped in 0.76-mm-thick cadmium foil. Three reactor cells (Fig. 2)—two in the core (cells 2-2 and 5-5) and one in the reflector (cell 4-6)—were chosen for irradiation of detectors. This choice was made because of the necessity to perform measurements both in the core and in the reflector, the convenience of performing measurements, and the use of these cells in in-core studies of constructional materials and fuel compositions. Irradiation was carried out in experimental vertical channels 22 mm in diameter positioned in the center of FAs in cells 2-2 and 5-5 of the core as well as in the channel 42 mm in diameter located in the center of cell 4-6 of the beryllium reflector (Fig. 2). Prior to the experiment, a preliminary computational analysis was conducted with a view to determine the optimal position of devices along the height of experimental channels. The computational evidence allowed determination of neutron flux distribution along the height (Fig. 1) and positioning of the AD in such a way as to provide similar irradiation conditions for ampoules with uranium dioxide and obtain three measuring positions at the maximum of the fast neutron flux density.

The AD irradiation was conducted under steady-state reactor conditions with loading no. 2013-05, 342 h after the reactor began operation in the operating mode for 77 min at thermal power of 3.25 MW. This was enough to produce a sufficient number of ^{235}U fission products for subsequent measurement of their activities. The measurement of specific activities for nuclides in detectors was carried out with regard to defined intensities of characteristic gamma lines using the ORTEC gamma spectrometer. The gamma spectrometer consists of a GEM-35 type high-purity germanium detector and a DS-pecJunior 2.0 digital multichannel analyzer. The processing of the obtained spectra was performed using Gamma Vision software. The error of specific activity measurements of the Fe and Ni detectors was no more than 5% (2σ), whereas that of the ^{235}U (n, f) reaction products for the UO_2 detectors did not exceed 7% (2σ). The specific activities of the following ^{235}U (n, f) reaction products were measured for the UO_2 detectors: ^{95}Zr, ^{103}Ru, ^{137}Cs, and ^{140}Ba. The characteristics of detectors are given in Table 1. The results of the specific activity measurements for the Fe and Ni detectors and their comparison with computations are presented in Table 2. The mean experimental activity of the Ni detectors was 19.4×10^{-16} Bq/nucleus for cell 2-2, 16.9×10^{-16} Bq/nucleus for cell 5-5, and 5.8×10^{-16} Bq/nucleus for cell 4-6. The mean experimental activity of the Fe detectors was 3.3×10^{-16} Bq/nucleus for cell 2-2, 2.8×10^{-16} Bq/nucleus for cell 5-5, and 0.961×10^{-16} Bq/nucleus for cell 4-6.

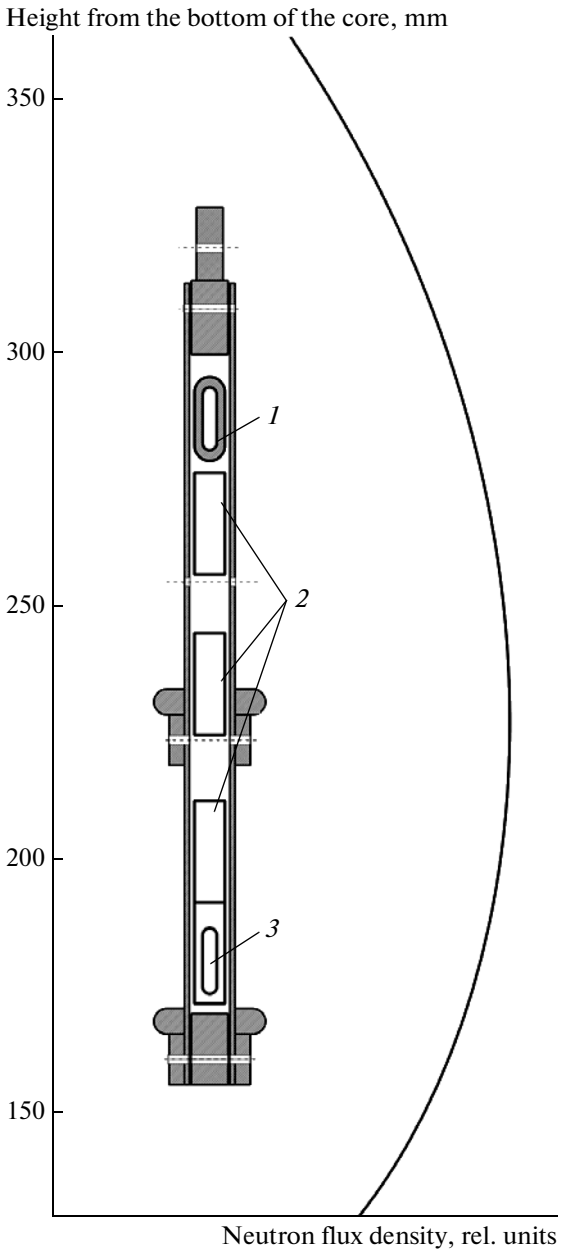

Fig. 1. Schematic view of AD and neutron flux distribution along the height of the core: (*1*) glass ampoule containing uranium dioxide in cadmium foil; (*2*) aluminum capsule with NAD; (*3*) glass ampoule with uranium dioxide;

2. CALCULATION OF NEUTRON FIELD CHARACTERISTICS

The calculations were performed using the MCU-PTR program. The program is designed for simulating neutron, photon, electron, and positron transport with analog and nonanalog (weighted) Monte Carlo methods. These methods are based on evaluated nuclear data in three-dimensional systems with allowance for changes in nuclide composition of

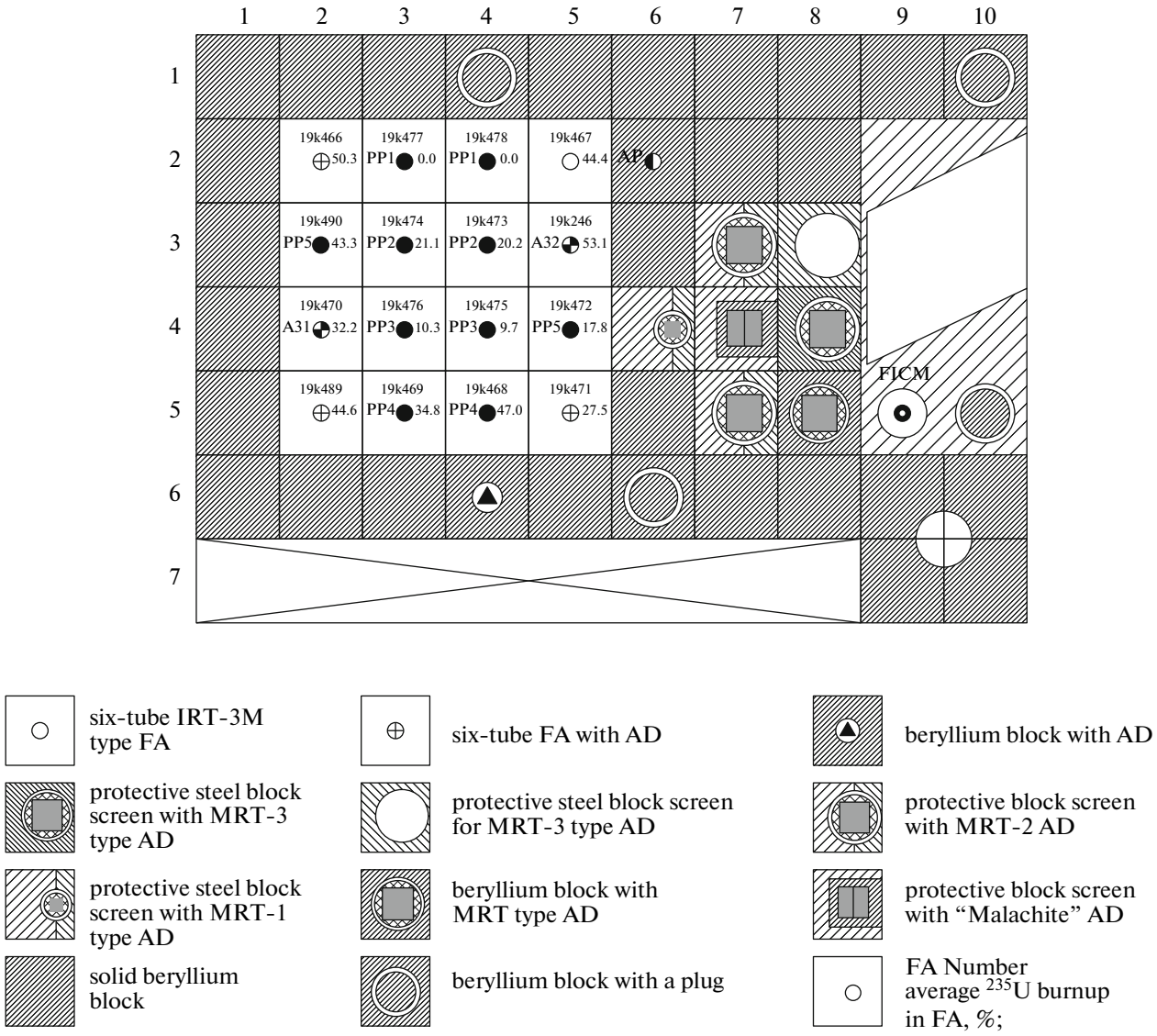

Fig. 2. Cartogram of loading no. 2013-05 of the IR-8 reactor core and reflector.

materials when interacting with neutrons. The neutron composition is measured with the BURNUP program (as part of the MCU-PTR program).

A nonhomogeneous equation of transport is solved for all the above-listed particles. In the case of neutron transport, the program can also solve a homogeneous equation (calculation of the criticality of neutron-multiplying systems).

Mathematically, this means that a kinetic equation with postulated boundary conditions is solved for the system under consideration, describing the distribution of particle fluxes in the system. So the effective neutron multiplication factor, energy release in FAs and in separate fuel elements, effective fraction of delayed neutrons, particle flux density, and other functions are registered. A detailed computational model was used in computer modeling of the conducted experiment. The model was supplemented with AD models and almost completely corresponded with real experimental conditions (isotopic composition of the core and reflector, fuel burnup in FAs, position of control rods in the reactor control and protection system (RC and PS), power level, and irradiation time). The geometric forms of objects were simplified as little as possible in computer modeling of AD.

The scheme of the computational model of ADs and their location in the core and first row of the reflector are given in Figs. 3 and 4. Owing to a small number of registration areas with activation detectors, nonanalog modeling, in particular, splitting and roulette method [5], together with traditional modeling

Table 1. The characteristics of detectors

Detector	Reaction	Measured nuclide	Eff. threshold energy, MeV; yield of reaction products, %	Half-lives	Irradiation energy E_γ, keV	Type and geometrical dimensions of detector	Detector material
^{54}Fe	(n, p)	^{54}Mn	3.0 MeV	312.8 days	834.8	Foil, Ø4 mm	^{54}Fe isotope, 99.6% enriched
^{58}Ni	(n, p)	^{58}Co	2.7 MeV	71.3 days	810.75	Foil, Ø4 mm	^{58}Ni isotope, 67.76% enriched
UO_2	(n, f)	^{95}Zr	6.5027	64 days	724.2	Chips, Ø1 mm	UO_2 with 10% enrichment of U^{235}
					756.7		
		^{103}Ru	3.0309	39.2 days	497.1		
					610.3		
		^{137}Cs	6.1883	30.05 years	661.6		
		^{140}Ba	6.314	12.7 days	304.8		
					423.7		
					437.6		
					537.6		

Table 2. Specific activities for Fe and Ni detectors

Cell no.	Ampoule no.	Detector	Specific activity, 10^{-16} Bq/nucleus		$(C - E)/E$, %
			Experiment (E)	Computation (C)	
2-2	6	^{54}Fe	3.33	3.34	0.56
		^{58}Ni	19.77	19.87	0.47
	4	^{54}Fe	3.41	3.29	−3.44
		^{58}Ni	19.73	19.55	−0.90
	2	^{54}Fe	3.15	3.22	2.20
		^{58}Ni	18.66	19.17	2.73
5-5	7	^{54}Fe	2.74	2.74	0.11
		^{58}Ni	16.64	16.28	−2.15
	5	^{54}Fe	2.87	2.71	−5.64
		^{58}Ni	17.03	16.08	−5.56
	3	^{54}Fe	2.77	2.61	−5.64
		^{58}Ni	17.13	15.52	−9.42
4-6	2.1	^{54}Fe	0.96	1.04	8.61
		^{58}Ni	6.24	6.21	−0.43
	2.7	^{54}Fe	0.99	1.02	2.99
		^{58}Ni	5.88	6.11	4.01
	2.3	^{54}Fe	0.93	0.99	6.26
		^{58}Ni	5.42	5.91	9.06

(method of generations), was employed to improve calculation accuracy. As part of the computational analysis of the conducted experiment, neutron flux density and dosimetric reactions rates were measured for all ADs with NADs. The estimated values of poison accumulation during irradiation and the fission reaction rate were measured for ADs containing uranium samples. The overall error of computational data did not exceed 5% (2σ).

3. COMPARISON OF RESULTS

Table 2 gives experimental and calculated specific activities for Fe and Ni. The comparison of results

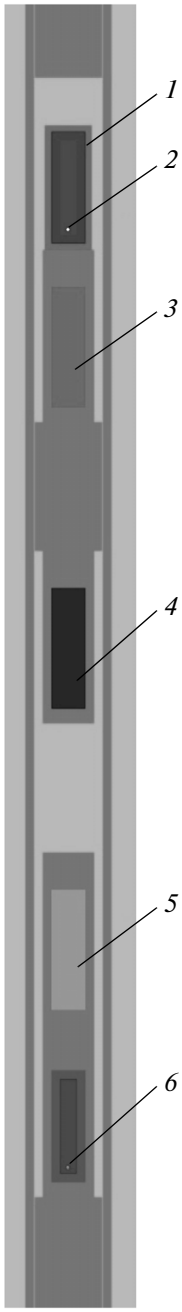

Fig. 3. Scheme of the computational model of AD: (*1*) cadmium foil; (*2*, *6*) uranium dioxide sample in a glass ampoule; (*3*–*5*) models of Al capsules with NAD sets.

indicates that the mean deviation of calculations from experiment was 2% for Fe detectors and 1.4% for Ni detectors in cell 2-2, 3.8% for Fe detectors and 5.7% for Ni detectors in cell 5-5, and 6% for Fe detectors and 4.5% for Ni detectors in cell 4-6. The observed discrepancies do not exceed 10%, which is fairly in line with the overall error of the calculations and experiment. Therefore, the calculated and experimentally estimated specific activities of Fi and Ni detectors are in satisfactory agreement. Figure 5 depicts the computed fast neutron flux density with energy above 3 MeV along the height of the channels as well as the experimental fast neutron flux density based on the specific activities of the iron detectors and calculated by the formula

$$\varphi_{>E_{\text{thresh}}} = A_0/[N_0\sigma_{\text{ef}}(1-e^{-\lambda\Delta t})], \quad (1)$$

where A_0 is the activity at the end of irradiation, Bq; N_0 is the concentration of target nuclei per one gram of detector material; Δt is the duration of irradiation, s; λ is the decay constant of a nuclide, 1/s; and E_{thresh} is threshold energy for the fission reaction.

As can be seen from the results, the calculations describe fairly well the distribution of the fast neutron flux density in cells 2-2, 5-5, and 4-6, which is in good agreement with the results of previous studies [4] and confirms the accuracy of calculations of the fast neutron distribution in the whole region of the reactor core and reflector.

The fast neutron flux density in cell 4-6 of the reflector is approximately one third as high as that observed in peripheral cells of the core. The maximum experimentally estimated fast neutron flux density with E >3.0 MeV is registered at the position of ~230 mm from the bottom of the core.

The comparison of the measured and calculated specific activities of the ^{235}U (n,f) reaction products is presented in Table 3. The calculated activity of the fission products was measured in terms of the calculated concentrations of nuclides formed at the end of irradiation using the formula

$$A_0 = N_0\lambda. \quad (2)$$

The mean deviation of calculations from the experiment in cell 2-2 reached 2.7% for the detector without cadmium and 10% for the detector in cadmium; in cell 5-5, the value was 9.4% for the detector without cadmium and 9.8% for the detector in cadmium; and in cell 4-6, the mean deviation was 25.5% and 8.1% for the detector without cadmium and the one in cadmium, respectively. On the basis of the measured values of activities and using the data on the reaction products yield (Table 1), the average rate of the ^{235}U (n,f) reaction was obtained for each UO_2 detector. The error of the ^{235}U (n,f) reaction rate estimated in such a manner reached 13% (σ) for the UO_2 detectors covered with cadmium foil and 2% (σ) for those without cadmium foil. The comparison of measured and computed values is given in Table 4.

The obtained ^{235}U fission rates allow evaluation of the ratio of thermal neutron flux to epicadmium neutrons, i.e., the cadmium ratio (Table 5). As can be seen from the data, the number of thermal neutrons in cells 2-2 and 5-5 exceeds that of the epithermal neutrons by approximately a factor of 22 and 17, respectively; and in cell 4-6, the number of thermal neutrons exceeds that of the epithermal neutrons by a factor of 31 and 43 according to the experimental evidence and calculated data, respectively.

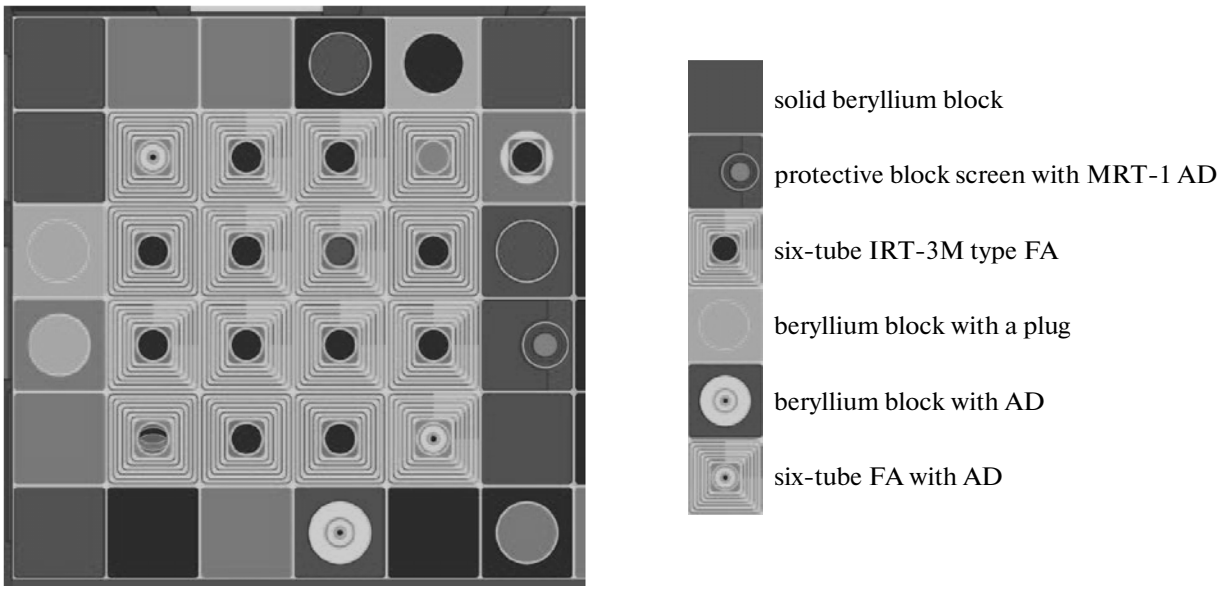

Fig. 4. Scheme of the computational model of the IR-8 core and the first row of the reflector with AD.

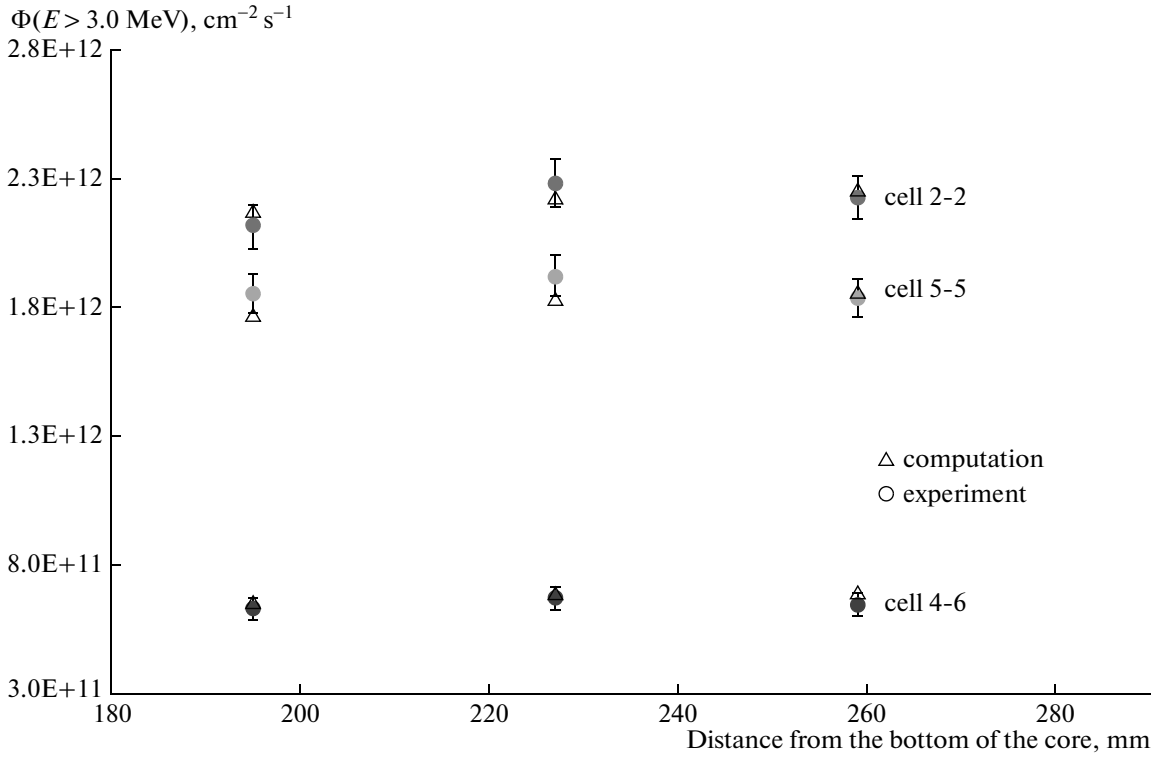

Fig. 5. Comparison of computed and experimental neutron flux density with $E > 3.0$ MeV along the height of the channels in cells 2-2, 5-5, and 4-6.

In the general case, the data confirm the accuracy of calculations on fuel burnup within the IR-8 reactor core. Additional studies are required to clarify the causes of significant discrepancy between experimental and computed values in cell 4-6 of the changeable beryllium reflector. Toward this end, a duplicate experiment is planned on the basis of the technique employed in this work.

Table 3. Specific activity for ^{235}U (n, f) reaction products

Cell no.	Ampoule no.	Detector	Specific activity, 10^6 Bq/g		$(C-E)/E$, %
			Experiment (E)	Computation (C)	
2-2	2.52	^{95}Zr	271.28	260.35	−4.03
		^{103}Ru	207.32	199.70	−3.67
		^{137}Cs	1.47	1.45	−1.20
		^{140}Ba	1273.04	1249.76	−1.83
	4.7 (in cadmium)	^{95}Zr	12.24	11.45	−6.39
		^{103}Ru	10.79	9.61	−10.90
		^{137}Cs	0.05	0.06	19.77
		^{140}Ba	57.30	55.59	−2.97
5-5	2.36	^{95}Zr	172.65	153.99	−10.81
		^{103}Ru	129.99	118.32	−8.98
		^{137}Cs	0.94	0.86	−8.25
		^{140}Ba	818.90	739.34	−9.72
	4.23 (in cadmium)	^{95}Zr	8.75	9.14	4.46
		^{103}Ru	7.62	7.64	0.31
		^{137}Cs	0.07	0.05	−22.68
		^{140}Ba	39.72	44.34	11.63
4-6	2.4	^{95}Zr	214.40	264.08	23.17
		^{103}Ru	162.35	202.06	24.46
		^{137}Cs	1.16	1.47	27.25
		^{140}Ba	997.87	1267.28	27.00
	2.5 (in cadmium)	^{95}Zr	6.75	6.11	−9.36
		^{103}Ru	5.70	4.97	−12.81
		^{137}Cs	0.04	0.03	−2.48
		^{140}Ba	32.07	29.56	−7.83

Table 4. ^{235}U (n, f) reaction rates

Cell	Ampoule no.	Fission rate, 10^{-12} s^{-1} MW^{-1}		$(C-E)/E$, %
		Experiment (E)	Computation (C)	
2-2	2.52	23.2	23.4	0.97
	4.7 (in cadmium)	1.04	1.04	0.20
5-5	2.36	14.7	13.8	−6.35
	4.23 (in cadmium)	0.844	0.832	−1.38
4-6	2.4	18.2	23.7	30.09
	2.5 (in cadmium)	0.589	0.554	−5.90

Table 5. Cadmium ratio

Cell	Cadmium ratio		$(C-E)/E$, %
	Experiment (E)	Computation (C)	
2-2	22.3	22.5	0.9
5-5	17.4	16.6	−4.6
4-6	30.9	42.8	38.5

CONCLUSIONS

1. The fast neutron flux density with $E > 3.0$ has been experimentally determined in two cells of the IR-8 core and one cell of the reflector.

The fast neutron flux density in cell 2-2 was shown to be 15% higher than that in cell 5-5, and in cell 4-6, it is approximately one-third as high as that in angular cells of the core.

2. On the basis of the experimental evidence, the distribution of fast neutron flux with $E > 3.0$ MeV along the height of the core has a maximum at the position of about 230 mm from the bottom of the core.

3. Use of a 0.76-mm-thick cadmium foil results in a decrease in the ^{235}U fission rate by approximately a factor of 22 in cell 2-2, by a factor of 17 in cell 5-5, and by a factor of 31 and 43 in cell 4-6 according to experimental evidence and calculated data, respectively.

4. With consideration of the errors, the experimentally found ^{235}U fission rate within the reactor core is in good agreement with the calculated data obtained with the use of the MCU-PTR program.

Additional studies are required to clarify the causes of the discrepancy between the calculated and experimental data in cell 4-6 of the changeable beryllium reflector. Towards this end, a duplicate experiment is planned on the basis of the technique used in this work. The experiment will allow clarification of results obtained in the cell of the reactor reflector as well as expansion of the experimental data by adding new measuring points in the IR-8 core and reflector.

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Translated by S. Savinykh