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Neutron flux density measurements in the active core, vertical and horizontal channels and thermal column of the WWR-SM research reactor

Sh.A. Alikulov,[1] S.A. Baytelesov,[1] F.R. Kungurov,[1] D.P. Tadjibaev,[1] and D.D. Tojiboev[1]

[1]*Institute of Nuclear Physics of the Academy of Sciences of the Republic of Uzbekistan*

Paper presents results of neutron flux density measurements in the active core, after changing the configuration of the reactor core, vertical and horizontal channels and thermal column of the WWR-SM research reactor of the Institute of Nuclear Physics of the Uzbekistan Academy of Sciences after full conversion to the IRT-4M low enriched fuel (LEU, 19, 75% enrichment on ^{235}U). Good concordance of experimental and calculation results allows to use the results of this work to choose reactor operation optimal modes.

Keywords: IRT-4M fuel assemblies, neutron flux, vertical channels, horizontal channel, thermal column.

I. NEUTRON FLUX MEASUREMENTS IN THE ACTIVE CORE

After full conversion of the active core of the WWR-SM research reactor from using of the IRT-3M (36% % enrichment on ^{235}U) to IRT-4M LEU fuel it was necessary to determine the neutron flux density in the active core of the reactor and its channels. For these purposes, corresponding measurements using different methods were performed and compared with the results of calculations.

The neutron flux density measurement in the active core and in the beryllium reflector of the WWR-SM reactor was carried out by the well-known method of neutron activation analysis of samples (foils) introduced into the neutron flux [1]. An aluminum-cobalt alloy containing 0,1% ^{59}Co and a mass of 2 mg was used as a sample in these measurements. The induced activity of the sample was measured using a high purity germanium detector with an efficiency of 10% with a CANBERRA GENIE 2000 spectrometric system. The resolution of the spectrometer on the γ-radiation line of the ^{60}Co element E_γ=1332 keV was 1,8 keV, and the calibration accuracy in the γ-radiation energy range of 0-1500 keV was $\pm$2 keV.

The measurement results are presented in Table 1, where the measured values of the neutron flux densities are given for different points in the center of the reactor active core.

Channel number according to picture 1	Mass of the monitor, mg	Thermal neutron flux densities, neutron/(cm$^2\times$s)
2-4	2,1	$7{,}64\times10^{13}$
4-4	2,7	$3{,}81\times10^{13}$
5-5	3,0	$3{,}98\times10^{13}$
6-5	3,0	$4{,}29\times10^{13}$
3-4	3,3	$4{,}68\times10^{13}$
5-7	2,2	$4{,}61\times10^{13}$
4-1	2,4	$5{,}15\times10^{13}$
2-5	2,3	$7{,}91\times10^{13}$
7-4	2,5	$5{,}47\times10^{13}$
3-1	2,8	$4{,}02\times10^{13}$

TABLE I. Thermal neutron flux densities in the active core of the WWR-SM research reactor.

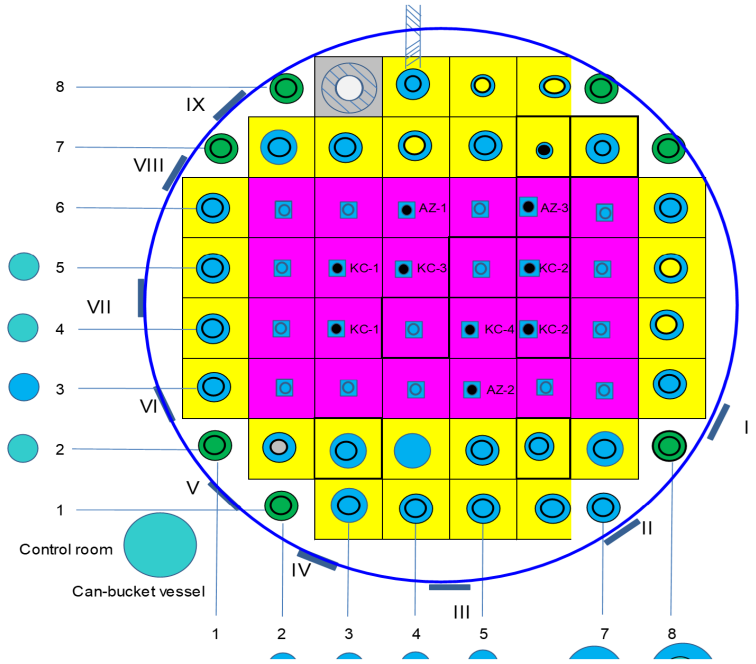


FIG. 1. WWR-SM research reactor active core

II. NEUTRON FLUX DENSITY MEASUREMENTS IN THE VERTICAL CHANNELS

The thermal neutron flux density in the vertical channels of the WWR-SM research reactor was measured by the thermo-neutron sensor (TNS) method. The TNS itself was made in the form of a differential thermocouple with a material sensitive to thermal neutrons at the "hot" junction. Its operation principle is based on the occurrence of a temperature difference between sensitive thermal elements and the environment when the sensor is placed in the neutron field. The sensitive element of the TNS is a uranium-nickel alloy. The range of measured thermal neutrons fluxes densities is in the range from 5×10^{12} to 5×10^{14} neutron/cm$^2\times$s in an aqueous medium.

Figure 1 shows a cartogram of the reactor core with 24 fuel assemblies of the IRT-4M type (red color), 19.75% enrichment in uranium 235. Beryllium reflectors (yellow color) are located around the fuel assemblies, some of them are loaded with vertical radiation channels, and others are loaded with emergency protection rods that compensate rods and a rod for automatic power control of the WWR-SM reactor. Around the core of the WWR-SM reactor, there are 8 cups for temporary storage and a bucket for reloading operations [2,3]. The measurements were carried out starting from a mark located 10 cm below the upper edge of the IRT-4M fuel assembly (FA), followed by moving the measuring rod through 5 cm deep into the channel up to 60 cm. The sensor was preliminary calibrated at a reactor power of 2 MW by 10-minute irradiation of a sample made of an aluminum-cobalt alloy, followed by determination of the thermal neutron flux at ^{60}Co.

The results of measurements of the reactor neutrons flux density in a number of vertical channels are presented in Table 2. The measurement accuracy is no worse than 12%.

Presented above measurements results of the thermal neutron flux density in vertical channels were compared with the calculations carried out using the MCNP4C code. Figure 2 shows the correspondence of the measurements results and calculations for vertical channel No. 2-4. It can be seen that to irradiate the tested samples at the maximum dose, they must be placed at a level of 20-40 cm from the upper edge of the FA.

III. NEUTRON FLUX DENSITY MEASUREMENTS AND CADMIUM RATIO IN THE HORIZONTAL CHANNEL AND THERMAL COLUMN

To determine the neutron fluxes densities values in the horizontal channel of the reactor, the method of activation of thin foils was used. The measurements were carried out in the 6^{th} horizontal channel of the reactor. We used a foil made of gold with a thickness of 10-20 microns and a weight of 1-5 mg in pairs, placed at the entrance to the channel and at the exit. One of the gold foils in each pair was placed in a 0,5 mm thick cadmium cover. The irradiation was carried out for 10 minutes. Before measuring the activity, the sample was kept for 24 hours in biological protection.

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Distance from the top edge of FA, cm	Φ_{2-4}	Φ_{4-4}	Φ_{3-1}	Φ_{3-4}	Φ_{7-4}
10	3,46	2,24	1,70	2,47	2,51
15	4,41	2,24	2,33	2,74	3,15
20	5,91	2,87	3,04	3,41	4,11
25	7,01	3,46	3,58	4,13	4,98
30	7,87	4,01	4,15	4,75	5,30
35	8,66	4,41	4,45	5,07	5,94
40	8,74	4,45	4,69	5,25	6,30
45	8,74	4,45	4,60	5,20	6,53
50	8,03	3,75	4,24	4,93	5,80
55	6,93	3,24	3,79	4,71	5,02
60	6,69	3,01	3,61	4,17	4,98

TABLE II. Density of thermal neutron fluxes in vertical channels 2-4, 4-4, 3-1, 3-4 and 7-4, ($\times 10^{12}$ neutron/cm$^2\times$s)

Height from the top point of the FA, cm	FA cells numbers								
	5-2	3-3	4-3	5-3	6-3	3-4	4-4	5-4	6-4
58,0-52,2	34,89	37,80	21,83	28,07	37,16	39,57	35,47	34,92	39,61
52,2-46,4	44,45	46,92	26,50	31,51	45,70	48,84	46,75	42,56	48,68
46,4-40,6	54,64	62,25	34,82	39,19	54,56	62,49	57,99	53,08	58,03
40,6-34,8	65,09	70,86	40,84	42,89	66,83	70,31	66,21	62,35	68,71
34,8-29,0	70,31	74,02	41,51	49,98	71,68	81,57	74,02	68,52	72,12
29,0-23,2	71,47	78,17	45,12	50,99	72,88	82,55	73,66	70,64	72,13
23,2-17,4	70,22	72,92	43,99	46,44	70,00	78,42	71,85	67,06	74,76
17,4-11,6	62,64	67,09	41,93	46,94	63,28	76,08	73,52	60,48	64,82
11,6-5,8	54,44	57,78	39,31	45,28	54,02	65,47	63,43	48,93	54,41
5,8-0,0	42,59	49,60	35,86	42,56	43,14	52,40	53,20	43,54	43,11
Total power, kW	570,76	617,43	371,72	423,86	579,26	657,70	616,11	552,09	596,37

TABLE III. Power distribution in 9 fuel assemblies depending on their height, kW

According to the measurements results, the thermal neutrons flux density inside the horizontal channel with 12% accuracy is equal to $\Phi_{6hch} = 1.8\times10^{12}$ neutron/cm$^2\times$s, and at the exit from the channel - $\Phi_{6hch} = 1.1\times10^{10}$ neutron/cm$^2\times$s. At the exit from the thermal column, the thermal neutron flux density Φ_{thc}=1.3$\times10^{10}$ neutron/cm$^2\times$s $\pm$12%, and the cadmium ratio R_{Cd} is 28$\pm$3.

To find the optimal configuration of the arrangement of fuel assemblies in the core of the WWR-SM reactor, which allows, at minimum uranium fuel consumption, obtaining the maximum power and the minimum coefficient of volumetric nonconformity, the MCNP4C code was used, which allows to vary as fuel contents as fuel assemblies locations in the reactor active core at various positions of the compensating rods. With this in mind, using the MCNP4C code, the power was calculated for fuel assemblies, dividing each of them into 10 equal parts by volume: 7.15$\times$7.15$\times$5.8 cm^3=296.51 cm^3. The reactor power was assumed to be 10 MW. As an example, Table 3 shows the calculation results for 9 fuel assemblies.

From the data presented in Table 3, it can be seen that the maximum power is observed in the middle section of the fuel assembly, and the minimum is at the edges of the fuel assembly. A similar picture is observed for the remaining 9 fuel assemblies.

IV. CONCLUSION

The results of determining the neutron flux densities in the active core, vertical and horizontal channels, and the thermal column after the complete conversion of the WWR-SM research reactor of the Institute of Nuclear Physics of the Uzbekistan Academy of Sciences to IRT-4M LEU fuel are presented. Good concordance between the experimental and calculation results allows to use the results of this work to choose reactor operation optimal modes.

Obtained results give possibility to determine the time and place of irradiation of samples in the reactor for carrying out research work.

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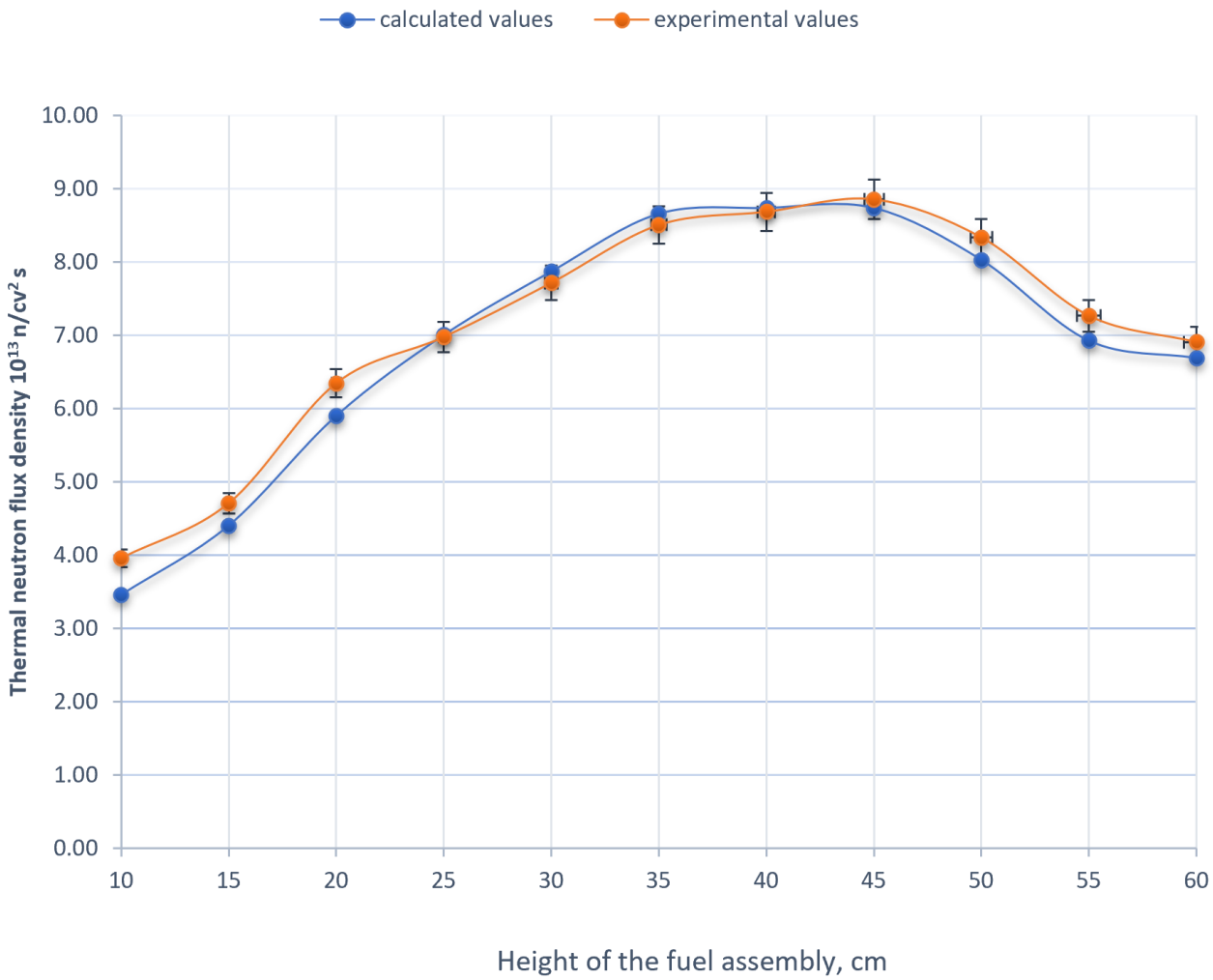


FIG. 2. Comparison of experimental and calculated values of the neutron flux densities distribution along the height of the fuel assembly in the vertical channel No. 2-4.

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