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ABOUT SOME CHARACTERISTICS OF WWR-SM REACTOR AT WORK WITH THE LOW ENRICHED NUCLEAR FUEL

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In the present work results of neutron flux density determination in the reactor active zone for vertical and horizontal channels after the complete replacement of IRT-3M type fuel assemblies with 36% enriched ^{235}U are presented. Calculations for the optimal configuration finding of the FA loading in the reactor active zone with the aim of obtaining the uranium fuel minimal expense and the maximal power at the minimal power peaking factor were carried out. Agreement between experimental data and calculated results would allow the choice of optimum configurations that provide the best utilization and safety.

Keywords: *Reactor,* fuel assemblies (FA), *neutron flux, nuclear fuel.*

INTRODUCTION

In the Institute of Nuclear Physics in September 1959 nuclear reactor of the type WWR-SM with power of 2 MW was put into operation. In a reactor in quality of decelerating and heat – carrier the water with the expense in the first contour 1250 m^3/hour and maximal temperature in an active zone 50^o C is used. There are 9 horizontal channels and thermal column used for investigations on the physics of heavy nucleus division, neutron physics, solid state physics and study of materials structure. 27 vertical channels are used for radioisotope production and other applications. After modernization in 1979 the power of nuclear reactor was increased up to 10 MW. This allowed to carry out scientific and technical programs in the field of metallurgy, medicine, and agriculture, to organize the production of new isotopes.

Under now a days conditions, along with science problems, a special attention is given to the problems of nuclear safety and organization of more optimum regimes of reactor utilization. For these purposes in 1998 conversion of WWR-SM reactor to the use of fuel assemblies (FA) of IRT-3M type with 36% enrichment in ^{235}U begun. Now the process of replacement of FA from fuel of 90% enrichment to FA with has 36% enrichment is completed [1].

The choice of optimum regimes of reactor operation provides for fuel economy and creates favorable conditions for realization of scientific research and isotope production. More detailed knowledge of physical and thermal-hydraulic parameters of the active core of the reactor is necessary for rational use of vertical and horizontal channels of WWR-SM reactor.

The purpose of the present work is to report the results of measurements and calculations of thermal and neutron parameters in the active core and channels of WWR-SM reactor to use FA such as IRT-3M with fuel of lowered enrichment in ^{235}U.

NEUTRON FLUX MEASUREMENTS AT THE ACTIVE CORE OF WWR - SM REACTOR AND VERTICAL CHANNELS

Measurements of neutron fluxes were carried out at the active zone and at the beryllium reflector of the WWR-SM reactor using well-known method of neutron activation of samples, placed in the core [2]. In these measurements samples of 2 mg weight - out of an alloy of Al and 0.1 % ^{59}Co were used. The activity of samples was measured with the help of Ge-detector with efficiency 10 % with spectrometer system GENIE 2000 of firm CANBERRA.

The resolution of the spectrometer system for γ line of ^{60}Co with $E_\gamma = 1332$ keV was 1.8 keV and calibration accuracy was ± 2 keV in the energy range of γ - radiation 0-1500keV .

Results of neutron flux measurements for different channels of the active core are presented in Table 1.

Densities of neutron fluxes in the vertical channels of the WWR-SM reactor were measured by thermal neutron sensor (TNS). TNS is the thermo-pair, the principle of operation of which is based on the occurrence of temperature difference between sensitive thermal element and environment while placing of the sensor in neutrons field. Sensitive element of the TNS is uranium-nickel alloy. The range of flux measurements of thermal neutrons is in limits from 5×10^{12} up to 5×10^{14} neutrons/cm^{2}s in water medium. Before measurements the sensor was calibrated with the help of an Al Co alloy sample irradiated during 10 minutes at 2 MW power of WWR-SM reactor with the subsequent definition of thermal neutrons flux based on ^{60}Co line.

The measurements began with the mark located on 10 cm below the top edge of FA (IRT-3M with 36% enrichment on ^{235}U), with the subsequent moving of the measuring bar in steps of 5 cm up to distance 60 cm deep in the channel.

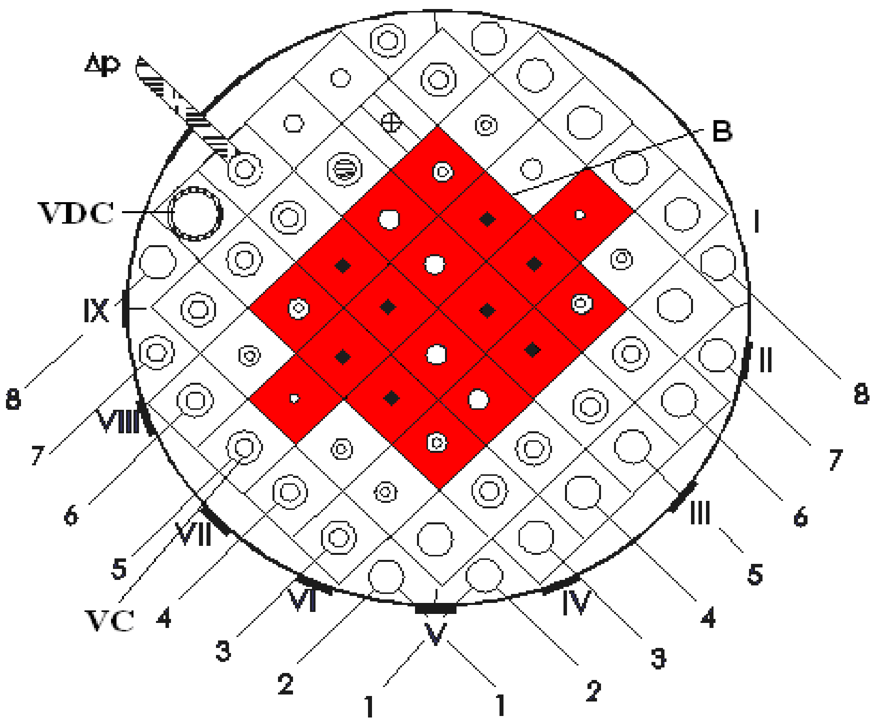

Figure 1. Active core cartogram of WWR-SM reactor

VDC - Vertical dry channel,
B - ^{235}U fuel.
VC– Vertical channel,
KCI - 6 - Shim rods
I, II… IX – Numbers of horizontal channels at the reactor.
Δp – Channel for measuring water pressure difference at the reactor.
AR – Automatic regulator,
A_3 - Disaster protection rods.

Results of measuring reactor neutron flux density are presented at the Table 2. Accuracy of measuring is better than 10 %.

Table 1. Thermal neutron flux at the active core of WWR-SM reactor

Channels number according to Figure 1.	Monitor Mass, mg	Thermal neutron flux, neutrons/cм^{2}s
2-4	2.1	1.5×10^{14}
4-4	27	1.2×10^{14}
5-5	30	1.08×10^{14}
6-5	30	0.89×10^{14}
3-4	33	1.08×10^{14}
5-7	22	1.2×10^{14}
4-2	24	1.15×10^{14}
2-5	23	1.22×10^{14}
7-4	25	1.19×10^{14}
3-1	28	1.07×10^{14}
7-1	27	0.79×10^{14}

Table 2. Thermal neutron flux in vertical channels, ($\times10^{13}$ neutrons/cm$^2\cdot$s).

Distance from the top edge of FA, cm	Fluxes in channel # 6-2	Fluxes in channel # 4-4	Fluxes in channel # 4-2	Fluxes in channel # 3-2	Fluxes in channel # 7-1	Fluxes in channel # 3-7
10	4.4	6.1	5.7	5.5	3.3	5.5
15	5.6	6.1	7.8	6.1	4.4	6.9
20	7.5	7.8	10.2	7.6	5.7	9.0
25	8.9	9.4	12.0	9.2	6.7	10.9
30	10.0	10.9	13.9	10.6	7.4	11.6
35	11.0	12.0	14.9	11.3	7.8	13.0
40	11.1	12.1	15.7	11.7	7.8	13.8
45	11.1	12.1	15.4	11.6	7.8	14.3
50	10.2	10.2	14.2	11.0	7.2	12.7
55	8.8	8.8	12.7	10.5	6.4	11.0
60	8.5	8.2	12.1	9.3	6.3	10.9

The reported measured results of thermal neutrons fluxes in the vertical channels were compared with calculations, which have been carried out with the help of the MCNP4C program [3]. Results of comparison between measurements and calculations for the vertical channel number 3-2 is shown in Figure 2. flux occurred. It is clear that for the maximal flux of irradiated samples is observed the maximal flux at the level 25-55 cm from the top edge of FA.

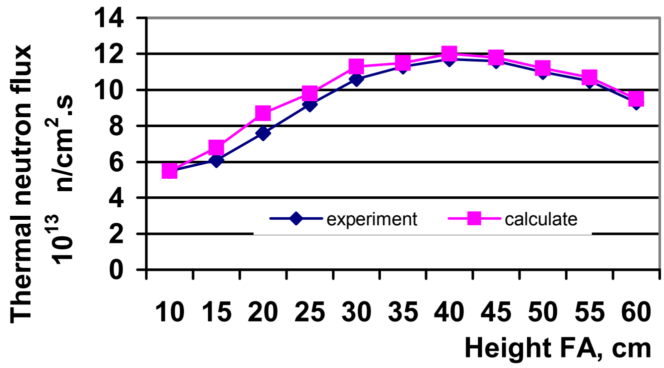

Figure 2. Comparison of experimental and calculated values of neutron flux distribution in vertical channel as a function of FA height.

MEASUREMENT OF NEUTRON FLUX AND CADMIUM RATIO IN THERMAL COLUMN AND IN HORIZONTAL CHANNEL

For the determination of values of neutron fluxes in the horizontal reactor channels the technique of thin foils activation was also used. Measurements were carried out in the channel number VI of the reactor. Foils from gold with thickness 10-20 µm and weight 1-5 mg were used. They were placed by pairs on the entrance of the channel and on its exit. One gold foil in each pair was placed in cadmium cover with thickness 0.5 mm. Irradiations carried out within 10 minutes. Before measurement of activity the sample was kept for a day in biological protection.

From the results of the measurements the flux of thermal neutrons on the entrance in the channel was equal $F_{6en} = 1.8\times10^{12}$ neutrons /$cm^2\cdot s$ and on the exit from the channel equal $F_{6ex}= 1.1\times10^{10}$ neutrons /$cm^2 s$. On the exit of the thermal column the density of flux of thermal neutrons is equal $F_{col} = 1.3\times10^{10}$ neutrons/ $cm^2 s$ at the cadmium relation $R_{Cd} = 28.3$.

CALCULATIONS OF THE THERMAL LOADING OF FA

To find an optimum configuration of the FA loading in the active core of WWR-SM reactor, allowing at the minimal expense of uranium fuel to receive the maximal power and minimal power peaking factor, the program MCNP4C [3] was used. The program allows changes of fuel content and loading of FA elements in the active core of the reactor at various positions of compensating rods. Using this program modeling of reactor active core reactor for 18 FA is executed. Model of the active core is shown in Figure 3.

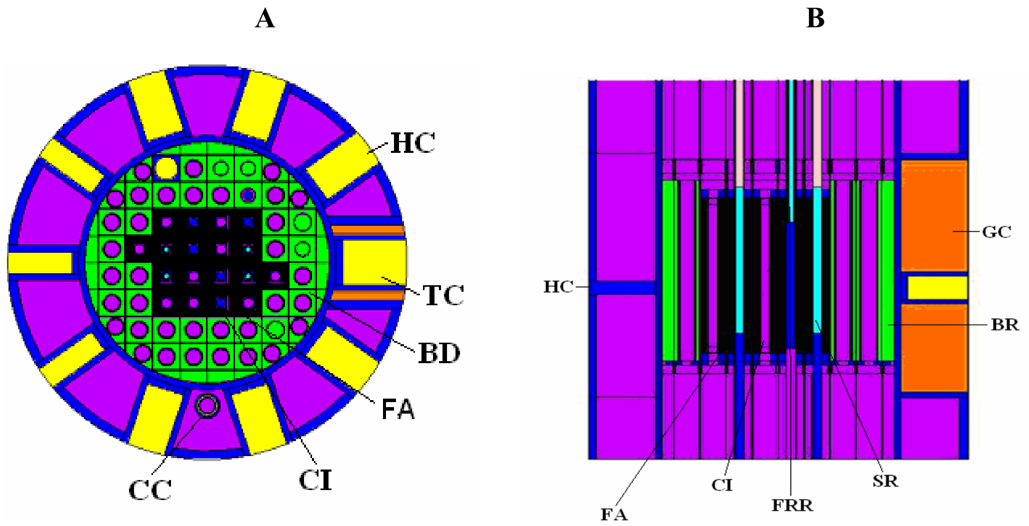

Figure 3. Active core model of WWR-SM reactor "**A**" is horizontal slice and "**B**" is vertical slice.

HC – horizontal channel; TC – thermal column;
BD – beryllium displacer; CI – irradiation channel,
CC – cadmium channel; FA – fuel assembly,
GC – graphite column; BR – beryllium reflector
SR - shim rods; FRR - feedback regulator rods.

For such a configuration of the reactor active core, calculations of thermal loading for FA such as IRT-3M were executed. In the calculations as a limit of safe performance of the reactor, the condition of non-boiling of water on the FA surface was adpted. It is known, that the calculate d value of temperature of the nucleate boiling on the FA surface is a function of local pressure of water in the gaps between FA, the saturated water temp and the local density of the thermal flux. For the choice of safe thermal regime of the reactor (allowable level of power) the status of the most heat-stressed of FA in the active core was analyzed. For this purpose the most heat-stressed FA was determined using the code ASTRA [4]. In the calculation the power got out such, that factor of the stock prior to the beginning of water boiling on the surface of FA was not less than 1.45. The results of calculations are submitted in the Table 3.

Table 3. Results of calculations of thermal power for 18 cells of FA

Number of FA cells according to Figure 1.	Thermal Power of FA, kW	Power peaking factor.	Burning up of ^{235}U, %
6-3	315.04	1.77	54.85
6-4	445.48	1.90	21.14
6-5	396.11	1.87	35.22
6-6	392.00	2.15	25.45
5-2	295.11	1.77	59.39
5-3	395.28	1.65	31.40
5-4	495.15	1.37	8.59
5-5	514.40	1.38	2.04
5-6	392.21	1.87	44.73
4-3	423.83	1.81	43.24
4-4	534.19	1.34	2.11
4-5	510.92	1.33	8.73
4-6	416.71	1.64	31.55
3-7	334.15	1.85	57.63
3-3	508.96	1.83	26.51
3-4	475.43	1.91	33.68
3-5	475.30	2.01	21.15
3-6	340.45	1.55	52.23

MCNP4C program was used to find the optimal configuration of FA's loading at the reactor active core with the purpose of minimization of expense of uranium fuel and obtaining the maximum power at minimization of the power peaking factor of thermal power. For this in MCNP4C program the power's calculation was carried out for 18 FA, dividing each of them into 10 equal parts of volume 7.15x7.15 x 5.8 см^3 = 296.51 cm^3. In the calculations the reactor power was supposed to be equal to 10 MW. As example, in Table 4. the results of calculation are presented for 9 FA.

Table 4. Power distribution in 9 FA depending on their height, kW.

Height from the top point of FA, cm	Number of FA cells				
	5-2	3-3	4-3	5-3	6-3
58.0-52.2	34.89	37.80	21.83	28.07	37.16
52.2-46.4	44.45	46.92	26.50	31.51	45.70
46.4-40.6	54.64	62.25	34.82	39.19	54.56
40.6-34.8	65.09	70.86	40.84	42.89	66.83
34.8-29.0	70.31	74.02	41.51	49.98	71.68
29.0-23.2	71.47	78.17	45.12	50.99	72.88
23.2-17.4	70.22	72.92	43.99	46.44	70.00
17.4-11.6	62.64	67.09	41.93	46.94	63.28
11.6-5.8	54.44	57.78	39.31	45.28	54.02
5.8 - 0.0	42.59	49.60	35.86	42.56	43.14
Full-power, kW	570.76	617.43	371.72	423.86	579.26

As is clear from Table 4. the maximum power observed on the middle cell of FA, and minimum on the edges of FA. Similar situation is observed also for all FA.

In summary we note the good agreement between experimental and calculated data, that allows using the results of the given work for the choice of best regimes of exploitation of WWR-SM reactor for its safe performance and for research purposes.

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حول بعض الخصائص لمفاعل الأبحاث WWR-SM الذي يعمل بوقود نووي منخفض الإثراء

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تم في هذا البحث تعيين كثافة فيض النيوترونات في المنطقة النشطة للقنوات الأفقية والرأسية بعد الاستبدال الكامل للوقود من نوع IRT-3M بنسبة إثراء ^{235}U قدرة ٣٦% ومن الحسابات أمكن تحديد الوضع الأمثل لتحميل الوقود في المنطقة النشطة بهدف الحصول علي أقل تكلفة للوقود عند أقصي قدرة ولأقل معامل تحميل. والتوافق بين النتائج المعملية والحسابات قد يسمح باختيار التوزيع الأمثل للوقود الذي يحقق أفضل استخدام وأمان.