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Determination of the Neutron Fluxes Energy Spectrum of the WWR-SM Reactor of the INP AS RU

The article presents the results of analyzing the energy spectrum and spatial distribution of neutrons in the core of the WWR-SM (Water-Water Reactor, Serially Modernized) research reactor after switching to low-enriched fuel (19.75 % ^{235}U). The increase in the number of fuel assemblies (FAs) from 18 to 24 altered the neutron characteristics of the reactor. A combination of computational methods (IRT-2D and WIMS codes) and experimental data obtained from neutron activation analysis enabled a detailed study of flux distribution. Fast neutrons dominate in the central part of the core, while the proportion of thermal neutrons increases significantly in the beryllium reflectors. Measurements showed that in vertical channels, the thermal neutron flux density is 2.3 times higher than that of fast neutrons. In horizontal experimental channels, values up to $1.8 \cdot 10^{12}$ neutrons/(cm^2·s) with a cadmium ratio of 28 were recorded, confirming their suitability for research. Analysis of the thermal power of FAs revealed its maximum values in the center of the core with a gradual decrease toward the periphery, correlating with the ^{235}U burnup distribution. The obtained results have practical significance for optimizing fuel loading, planning refueling campaigns, and testing prospective fuel compositions (UO_2+Al, U_3Si_2+Al). The study emphasizes the need for further verification of computational models and more detailed investigation of neutron spectra under various core configurations. The presented data contribute to enhancing the efficiency and safety of WWR-SM reactor operation while expanding its research potential in nuclear physics and materials science.

Keywords: nuclear fuel, energy spectrum, fast neutrons, thermal neutrons, reactor core, neutron activation analysis, WWR-SM reactor, fuel burnup

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Introduction

In the field of research related to radiation: processes of interaction of radiation with matter [1–3], processes of material production using radiation of different spectral composition [4–7], and others are widely applied and studied. In this area, the processes related to radiation in nuclear reactors are of particular interest. In the core of a research reactor, the energy spectrum of radiation (neutrons) is a key characteristic that determines the efficiency and safety of nuclear experiments.

The energy spectrum of neutrons in the core of a research reactor is a key characteristic that determines the effectiveness and safety of nuclear experiments.

Knowledge of the spectrum is necessary for accurate planning of neutron physics studies, assessment of fuel burnout, dose loads on materials, and accuracy of activation analysis [8–10].

Classification of neutron spectra.

In research reactors operating primarily with thermal neutrons, the spectrum is conventionally divided into three regions [10, 11]:

Thermal range (E < 0.5 eV);

Epithermal region (0.5 eV < E < 100 keV);

Fast range (E > 100 keV).

Methods for determining the neutron spectrum.

Experimental methods:

Activation methods using standard foils (Au, Mn, Co, etc.) [12];

The method of responses and convolution [13];

The TOF (time-of-flight) method [14].

Calculation methods:

Montecarlo programs: MCNP, SERPENT [15];

Deterministic codes: WIMS, DRAGON, IRT-2D [16–18];

The use of ENDF/B-VII, JENDL, JEFF, and others nuclear data libraries [19].

The features of the spectrum in research reactors are determined by the moderator, reflector and configuration of the core. In the VVR-SM reactor, for example, there is a pronounced predominance of thermal neutrons in the central core zone and an increase in the proportion of fast neutrons in the periphery [20]. The spectra significantly depend on the degree of fuel burnout and loading of experimental devices [21].

Application of spectral analysis data:

Calculation of reactivity and fuel burnout [22];

Planning of radiation testing of materials [23];

Preparation of macroscopic constants [24];

Activation analysis and isotope production [25].

The WWR-SM research reactor is a key facility for fundamental and applied research in nuclear physics. Since 2009, the reactor has been using low-enriched IRT-4M fuel (19.75 % uranium-235) with a uranium concentration of 2.8 g/cm^3 [26]. In the near future, it is planned to test new types of fuel: UO_2+Al (3.3 g/cm^3) and U_3Si_2+Al (3.6 g/cm^3) [27], which requires a detailed study of the neutron-physical characteristics of the core. Previously, studies of neutron fluxes in WWR-type reactors were carried out in works [28–30], where methods for calculating and measuring neutron spectra were developed. However, for the WWR-SM reactor with its unique core configuration and beryllium reflectors, such studies require updating. This work uses modern calculation methods, including the IRT-2D code and the ASTRA program [31], as well as neutron activation analysis to verify the results [32].

The purpose of the work is to determine the energy spectrum of neutron fluxes, the distribution of neutron density in the core and channels of the reactor, and to analyze the thermal power of the FA. The data obtained will help optimize the operation of the reactor and prepare for the use of new types of fuel.

The analysis of the neutron energy spectrum is a fundamental part of neutron physics calculations. Combining experimental data with numerical simulation results provides the most accurate representation of neutron fields in the core and contributes to improving the reliability and efficiency of nuclear installations.

Calculations of neutron fluxes in vertical channels

The IRT-2D code was used to calculate the neutron flux density distribution in the WWR-SM reactor core. Two-group macroscopic cross-sections for each FA obtained using the WIMS code [33] were used.

The reactor core is loaded with 24 FAs with different degrees of uranium-235 burnup. Fresh FAs with minimal burnups are located in the central part of the core, and fuel assemblies with high burnups are located on the periphery. Beryllium reflectors are placed around the 24 FAs.

Figure 1 shows the results of calculating the distribution of thermal and fast neutron fluxes in the core of the WWR-SM research reactor.

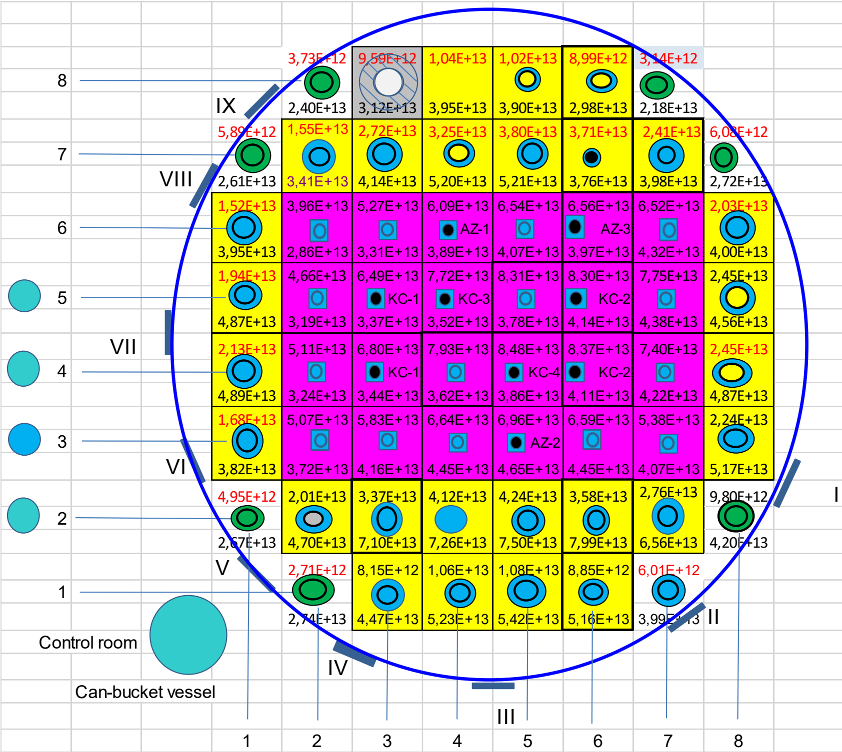

Figure 1. Distribution of thermal and fast neutron fluxes in the WWR-SM research reactor's core.
The upper row in red shows the results of calculations of fast neutron fluxes with energy $E > 0.1$ MeV,
and the lower row shows the results of calculations of thermal neutron fluxes with energy $E < 0.625$ MeV

Figures 2 and 3 show the distribution of fast and thermal neutron fluxes in the WWR-SM reactor core. As can be seen from the figures, in the center of the core, where the FAs are located, there is an increase in the number of fast neutrons, which is 2 times greater than the number of thermal neutrons. On the periphery of the core, where the beryllium reflectors are located, the number of thermal neutrons exceeds the number of fast neutrons. It is also clear that the thermal neutron flux density is greater where 6FAs are located in series (column index), compared to the section where 4 FAs are located in a row (row index).

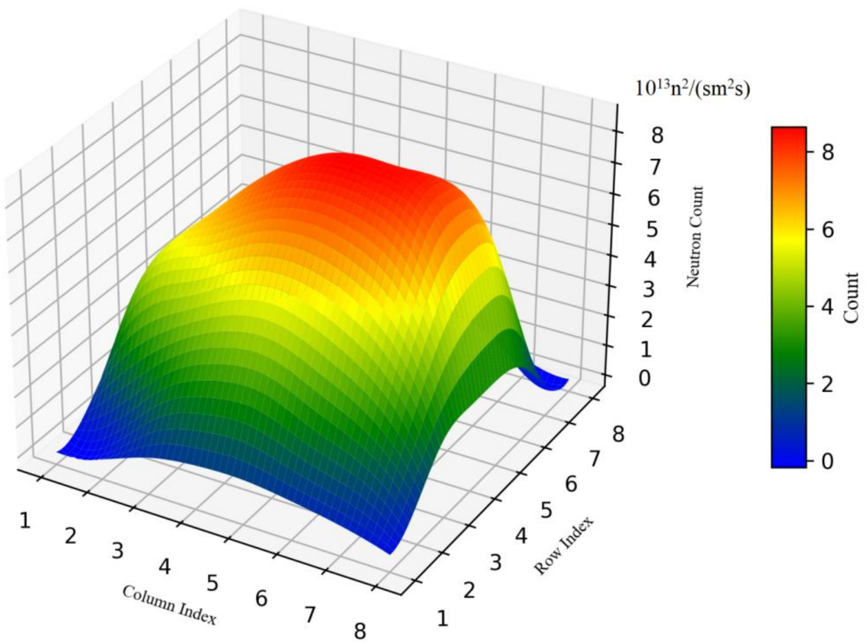

Figure 2. Distribution of fast neutron flux in the WWR-SM reactor's core

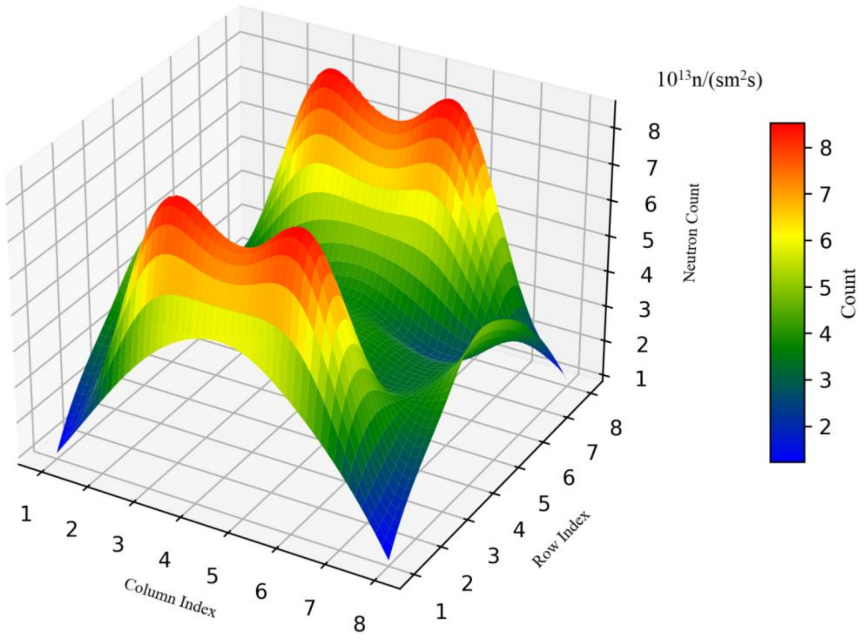

Figure 3. Distribution of thermal neutron flux in the WWR-SM reactor's core

Measurement of neutron fluxes in the core and vertical channels

The measurement of the neutron flux density in the core and in the beryllium reflector of the WWR-SM reactor was carried out using the well-known method of neutron activation analysis of samples (foils) introduced into the neutron flux. The activity of the irradiated sample was determined by the formula

$$A_0 = 1.628 * 10^{-15}\,\Phi\sigma мp\left(1-\exp\left(\frac{-0.693t}{T}\right)\right) * \exp\left(\frac{-0.693t_1}{T}\right)A^{-1}, \tag{1}$$

where Φ — neutron flux, neutron/(cm^2*s); σ — activation cross section, mbarn; *м* — weight of activated sample, mg; *p* — isotope abundance, %; *A* — atomic weight of the irradiated isotope; *T* — half-life of the product isotope, seconds; t, t_1 — irradiation time and cooling time (in the same units as T), seconds.

And was measured using a GC1020 germanium detector with a diameter of 46 mm, a length of 29 mm and an efficiency of 10 % with a GENIE 2000 spectrometric system from CANBERRA. The spectrometer resolution for the ^{60}Co γ-radiation line $E\gamma$ = 1332 keV was 1.8 keV, and the calibration accuracy in the 0–1500 keVγ-radiation energy range was ±2 keV.

In measurements of the thermal neutron flux in the channels of the WWR-SM reactor, an aluminum-cobalt alloy containing 0.1 % ^{59}Co with a mass of 2 mg was used as a sample. A comparison of the results of measurements and calculations of the thermal neutron flux density for different vertical channels is presented in Table 1.

Table 1

Thermal neutron flux densities in the WWR-SM reactor core

Channel number according to Fig. 2	Monitor weight, mg	Result of measurements of thermal neutron flux density, (10^{13} n/cm2 s)	Calculation result of thermal neutron flux density, (10^{13} n/cm2 s)
2-4	2.1	7.41	7.26
2-5	2.3	7.62	7.50
3-1	2.8	3.9	3.82
3-8	3.3	5.28	5.17
4-1	2.4	5.07	4.89
5-1	2.2	4.92	4.87
6-8	3.0	4.19	4.00
7-1	2.5	2.69	2.61

Energy spectrum of neutrons in the WWR-SM reactor core

The energy spectrum of neutrons was calculated depending on their energy in the WWR-SM reactor core. Using a program written in PYTHON, the energy spectrum of neutrons was constructed on a logarithmic scale.

Figure 4 shows the energy spectrum of neutrons in the vertical channel 4-1 of the WWR-SM reactor. As can be seen from the figure, in the vertical channel 4-1, where the beryllium reflector is located, there are 2.3 times more thermal neutrons than fast neutrons. This is due to the fact that most neutrons are slowed down in distilled water, which is located in the gaps of the beryllium reflector.

As can be seen from Figure 5, the vertical channel 4-2 is located inside the fuel assembly in the reactor core, so the fast neutron flux density is 1.6 times greater than the thermal neutron flux density. This is due to the fact that during the fission of uranium-235, neutrons with an average energy of 2 MeV are born, and they do not have time to slow down inside the nuclear fuel.

The obtained neutron spectra are in good agreement with the results obtained at the VVR-K reactor [34].

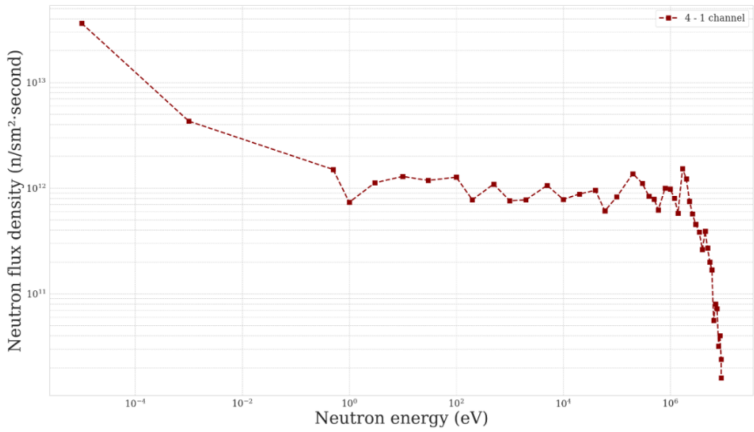

Figure 4. Neutron flux density depending on neutron energy in vertical channel 4-1 of the WWR-SM reactor

Figure 5 shows the neutron spectrum in channel 4-2.

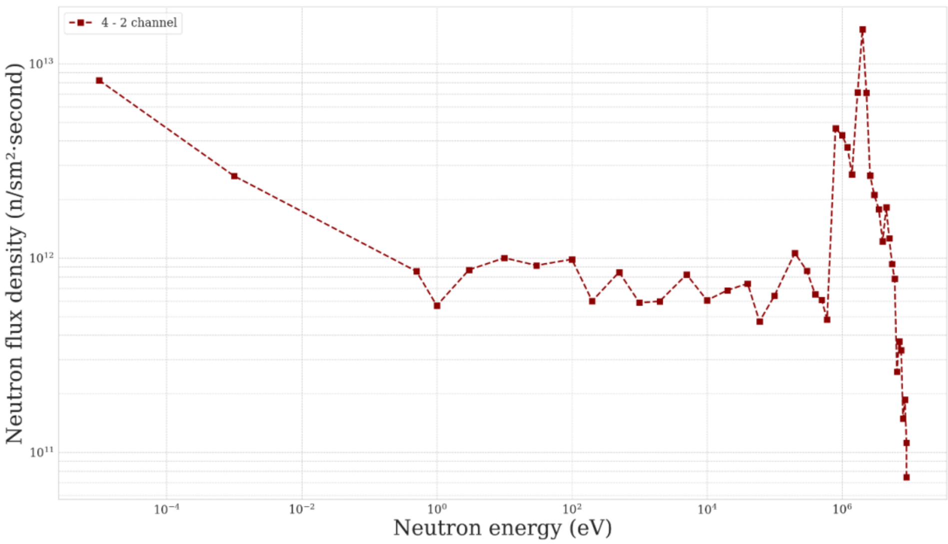

Figure 5. Neutron flux density depending on neutron energy in vertical channel 4-2 of the WWR-SM reactor

Measurement of neutron fluxes and cadmium ratio in the horizontal channel and thermal column

To determine the neutron flux values in the horizontal channel of the reactor, a mass spectrometric measurement technique was used. For this purpose, gold foils 10–20 μm thick and weighing 1–5 mg were placed in pairs in the mass spectrometer input arm near the bottom of the channel, in the middle, and at the output in the mass spectrometer. One of the gold foils in each pair was placed in a 0.5 mm thick cadmium case. Irradiation was carried out for 10 minutes, and then the samples were kept for 24 hours in the biological shield of the mass spectrometer. Before measuring the activity of the sample, the background NΦ of the set-up was measured. The exposure time was selected based on the condition of collecting sufficient statistics.

The measured number of N decays was determined by the formula

$$N = (N - N\Phi)\cdot n/(W\cdot S\cdot K\cdot Q\cdot \varepsilon), \quad (2)$$

where n — correction for the resolution of the installation; W — solid angle correction; S — attenuation adjustment; K — self-shielding correction; Q — backscatter correction; ε — efficiency of the measuring setup.

The resolving power R of the installation was determined by the formula:

$$R = \frac{1}{1 - 10^{-6} N}\,. \quad (3)$$

Here the following values were used: n = 1.005, W = 0.95, ε =1, and due to the small thickness of the gold foils, corrections for backscattering, attenuation and self-shielding were neglected.

The measurements and calculations carried out showed that the thermal neutron flux inside the 6th horizontal channel is:

$$\Phi_T = 1{,}8\cdot 10^{12} n/(\mathrm{cm}^2\ \mathrm{s}) \pm 12\ \%. \quad (4)$$

Cadmium ratio:

$$R_{\mathrm{Cd}} = 28 \pm 3. \quad (5)$$

At the exit of the channel:

$$\Phi_T = 1{,}1\cdot 10^{10} n/(\mathrm{cm}^2\ \mathrm{s}) \pm 12\ \%. \quad (6)$$

At the outlet of the thermal column:

$$\Phi_T = 1{,}3\cdot 10^{10} n/(\mathrm{cm}^2\ \mathrm{s}) \pm 12\ \%. \quad (7)$$

Calculation of the thermal power distribution of the IRT-4M type FA in the WWR-SM reactor core

There are several types of application programs for calculating certain reactor parameters. One of them is the IRT-2D program, developed by the staff of the Kurchatov Institute of Atomic Energy of the Russian Academy of Sciences. Calculations according to this program are carried out in a 2-group approximation on a plane in two-dimensional geometry.

The IRT-2D program was used to calculate the thermal power for 24 IRT-4M fuel assemblies with different uranium-235 burnup values. The results of the IRT-2D program are in good agreement with the experimental data [9].

The results of calculations of power distribution in the active zone with 24 fuel assemblies (FA) are given in Table 2.

Table 2

Results of calculations of thermal power for 24 cells of the WWR-SM reactor core

3,44 55,16	4,22 47,25	3,73 AZ-1 59,80	4,24 42,56	2,66 AZ-3 62,74	3,08 48,44
4,49 34,68	4,88 KC-1 30,08	5,39 KC-3 19,28	5,34 14,12	4,45 KC-2 26,35	3,25 54,30
3,49 61,74	4,95 KC-1 27,09	5,62 14,63	5,35 KC-4 19,67	4,61 KC-2 30,78	4,13 34,99
3,79 49,27	3,44 67,42	4,79 43,18	3,60 AZ-2 64,65	4,01 48,74	3,06 58,25

It is evident from the data presented in Table 2 that the maximum power is observed in the central section of the reactor core, and the minimum is at the edges of the core.

The center of the table shows the vertical channels inside the fuel assemblies (FA). The FAs with black circles in the center represent the emergency protection rods (AP-1, AP-2, AP-3) and the compensating rods (KS-1, KS-2, KS-3, KS-4).

The top row of cells in the table shows the fuel assembly power in percent, and the bottom row of cells shows the uranium-235 burnup in percent.

Conclusion

A comprehensive study of neutron fluxes in the WWR-SM research reactor core of the INP AS RU is carried out in this work. The main attention is paid to determining the energy spectrum of neutrons, their flux density distribution in vertical and horizontal channels, as well as the analysis of the influence of the core configuration on the neutron-physical characteristics of the reactor.

As a result of the analysis of the neutron flux distribution and its energy spectrum in the reactor core, the following was established:

The calculated and experimental data showed good agreement, which confirms the correctness of using the IRT-2D and WIMS codes for modeling neutron fluxes and thermal modes of the reactor.

It was found that the neutron fluxes distribution in the core depends on the location of the FAs and the fuel burnup degree: fast neutrons predominate in the central part, while the proportion of thermal neutrons increases in the beryllium reflectors.

Measurements in the vertical channels demonstrated that the thermal neutron flux density in the reflector zone exceeds that for fast neutrons by 2.3 times, which is due to the effective moderation of neutrons in water.

In the horizontal channels and the thermal column, thermal neutron flux values of up to $1.8 \cdot 10^{12}$ $n/(\mathrm{cm}^2\ \mathrm{s})$ with a cadmium ratio of $R_{\mathrm{Cd}} = 28$ were recorded, which confirms the suitability of these channels for experiments.

The thermal power of the FA is maximum in the center of the core and decreases towards the periphery, which is consistent with the distribution of uranium-235 burnup.

The results obtained are of great practical importance for:

Optimization of fuel loading and planning of refuelling campaigns.

Efficient use of experimental reactor channels for applied and fundamental research.

Preparation for testing new types of fuel (UO_2+Al, U_3Si_2+Al) with increased uranium concentration.

Further research prospects may include:

In-depth study of neutron spectra for various core configurations.

Verification of calculation models for new types of fuel.

Development of methods for improving the accuracy of neutron flux measurements in reactor channels.

Thus, the work carried out confirms the uneven distribution of neutron flux and power in the WWR-SM reactor core, which must be taken into account when operating reactor installations to ensure their efficient and safe operation.

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Нейтрон ағындарының энергетикалық спектрін анықтау ӨзР ҒА ЯФИ ССР-СМ реакторлары

Бұл зерттеуде (Өзбекстан Ядролық физика институты) ССР-СМ реакторының белсенді аймағында нейтрон ағындарының спектрі мен таралуы қарастырылды (19,75 % U-235 аз байытылған отынға көшкеннен кейін, отындық жинақтар саны 18-ден 24-ке дейін өсті). IRT-2D және WIMS программалары мен нейтрондық-активациялық талдау деректері пайдаланылып, нейтрон ағынының тығыздығы және энергетикалық спектрі зерттелді. Нәтижелер реактордың орталық бөлігінде жылдам нейтрондар басым екенін, ал бериллий рефлекторларында жылу нейтрондарының үлесі артатындығын көрсетті. Тік каналдардағы өлшемдер рефлектордағы жылу нейтрондарының ағыны жылдам нейтрондарға қарағанда 2,3 есе жоғары екенін анықтады. Көлденең каналдарда кадмий коэффициенті 28 болатын $1{,}8 \cdot 10^{12}$ н/(см²·с) дейінгі жылу нейтрон ағындары тіркелді, бұл оларды тәжірибелер үшін қолайлы етеді. Отындық жинақтардың жылу қуаты орталықта максималды болып, периферияға қарай азаяды, бұл U-235-тің жануына сәйкес келеді. Алынған нәтижелер отынды тиімді пайдалану, оны ауыстыруды жоспарлау және жаңа отын түрлерін (UO_2+Al, U_3Si_2+Al) сынау үшін маңызды.

Кілт сөздер: ядролық отын, энергетикалық спектр, жылдам нейтрондар, жылу нейтрондары, реактордың белсенді аймағы, нейтрондық-активациялық талдау, ССР-СМ реакторы, отынның жануы

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Определение энергетического спектра нейтронных потоков реактора ВВР-СМ ИЯФ АН РУ

В данном исследовании изучены спектры и распределение нейтронных потоков в активной зоне реактора ВВР-СМ (Институт ядерной физики, Узбекистан) после перехода на низкообогащенное топливо (19,75 % U-235) с увеличением числа тепловыделяющих сборок (ТВС) с 18 до 24. С применением кодов IRT-2D и WIMS, а также данных нейтронно-активационного анализа исследованы плотность потока нейтронов и их энергетический спектр. Результаты показали, что в центральной части активной зоны преобладают быстрые нейтроны, тогда как в бериллиевых отражателях возрастает доля тепловых нейтронов. Измерения в вертикальных каналах выявили, что поток тепловых нейтронов в отражателе в 2,3 раза выше, чем поток быстрых нейтронов. В горизонтальных каналах зафиксированы тепловые потоки до $1{,}8 \cdot 10^{12}$ н/(см²·с) с кадмиевым отношением 28, что делает их пригодными для экспериментов. Тепловая мощность ТВС достигает максимума в центре и снижается к периферии, что коррелирует с выгоранием U-235. Полученные данные важны для оптимизации загрузки топлива, планирования перегрузок и испытания новых видов топлива (UO_2+Al, U_3Si_2+Al).

Ключевые слова: ядерное топливо, энергетический спектр, быстрые нейтроны, тепловые нейтроны, активная зона реактора, нейтронно-активационный анализ, реактор ВВР-СМ, выгорание топлива

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