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*National Research Nuclear University «MEPhI», Moscow, Russia***INFLUENCE OF CORE MODEL PARAMETERS ON THE CHARACTERISTICS OF NEUTRON BEAMS OF THE RESEARCH REACTOR**

IRT MEPhI reactor is equipped with a number of facilities at horizontal experimental channels (HEC). Knowing of parameters influencing spatio-angular distribution of irradiation fields is essential for each application area. The research for neutron capture therapy (NCT) facility at HEC of the reactor was made. Calculation methods have been used to estimate how the reactor core parameters influence neutron beam characteristics at the HEC output. The impact of neutron source model in Monte Carlo calculations by MCNP code on the parameters of neutron and secondary photon field at the output of irradiation beam tubes of research reactor is estimated. The study shows that specifying neutron source with fission reaction rate distribution in SDEF option gives almost the same results as criticality calculation considered the most accurate. Our calculations show that changes of the core operational parameters have insignificant influence on characteristics of neutron beams at HEC output.

Keywords: neutron-capture therapy, MCNP, Monte Carlo method, dosimetry.

Introduction

Every research reactor is the unique nuclear facility designed for carrying out nuclear physics experiments in both reactor physics and many applied problems. In the last case there are horizontal and vertical channels where output neutron beams are formed with specified spectral and angular radiation distribution. Regarding the core as an initial neutron source we can form neutron spectrum at the reactor channel output with either changes in the core and the reflector or selecting peripheral equipment impacting the way of the neutrons pass through the core to the channel output. As usual the first method is extremely rare for functioning reactor and the problem is solved by modifying peripheral equipment. This becomes exceptionally important if the experimental task for the reactor changes. Recently many studies on NCT-facilities constructions are held at research reactor (for example at Kyiv Research Reactor (10 MW) [1] and at Research Reactor in Tashkent (10 MW) [2]). How changes in construction and operational parameters of the reactor core influence neutron beam characteristics is essential for the estimations of experimental results errors as physicist has only beam monitoring, and there is no information on how its characteristics are influenced by the changes in the core.

Monte Carlo full-sized three dimensional modeling of neutron spectrums formation during outputs of the research reactor channels requires estimation of errors, caused by inaccuracies in specifications of source parameters. These errors help to reduce calculation times without losing accuracy results in multivariate tasks.

In the presented computational research issues of sensibility of neutron beam characteristics at horizontal reactor channel to the core modifications are considered for NCT-facilities at IRT MEPhI. All calculations are made with MCNP-4c2 [3].

Description of experimental devices for NCT at the reactor

IRT MEPhI Research Reactor 2.5 MW pool-type currently uses IRT-3M fuel assemblies (FA) with high-enriched (90 %) fuel. The reactor first reached criticality in May 1967. Reactor has 10 horizontal experimental channels (two in graphite thermal column and eight in the pool) and vertical irradiation channels in the reflector. General view of reactor IRT MEPhI and the core is shown in Fig. 1. The core consists of 16 FA: 6 standard FA and 10 FA with control rod (CR).

Experimental irradiation facility for pre-clinical NCT-studies has already been created at the IRT MEPhI. Radiobiology channel HEC-4 started to operate for NCT [4]. It was decided to create a new facility at HEC-1 for more opportunities of the reactor using for NCT. Project of clinical channel HEC-1 at the thermal column of IRT MEPhI reactor was developed for NCT with thermal and epithermal neutrons.

Calculational MCNP model for NCT studies describing in details real three-dimensional geometry of irradiation facilities at the HEC-1 and HEC-4 was developed. Design of the facilities at the HEC-4 [4] and HEC-1 [5] was based on various calculations of neutron and contamination photon fields characteristics at these channels and satisfying NCT requirements were suggested and partially realized.

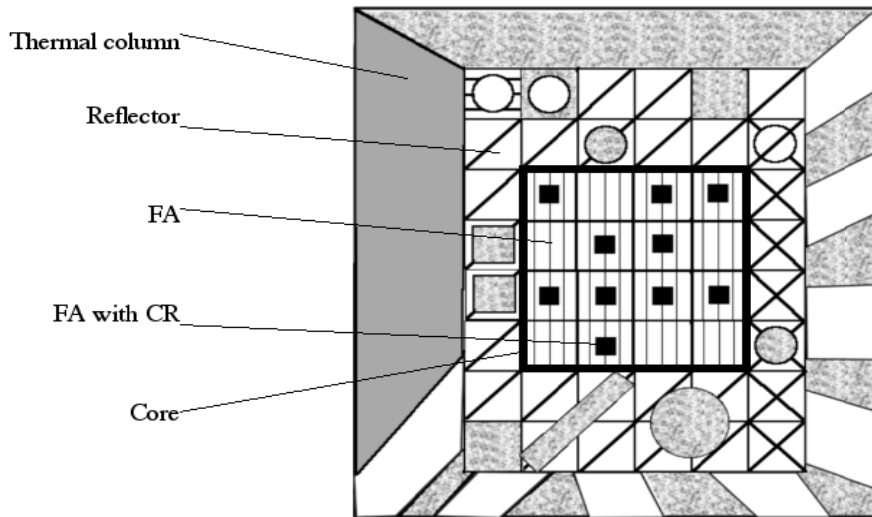


Fig. 1. General view of reactor IRT MEPHI and the core.

Problem statement

Characteristics of the neutron beam used for experimental studies depend on both parameters of facility (geometry and material composition) at the HEC and parameters of the reactor core being as neutron source. Facility parameters at the channel are studied and selected at the design stage. Parameters of the reactor core usually come from general reactor design aspects and are not specialized for the experimental needs.

Influence of both the core parameters and their ways of specifying within the calculation model on the results of horizontal beam characteristic calculations was investigated. The parameters are the following: number of fuel assemblies in the core, material composition of fuel, material composition of non-fuel components of the core, geometry of the fuel assemblies, reflector geometry. The research was made for NCT-facility at HEC-1 of the reactor IRT MEPHI.

The geometry of NCT-facility at HEC-1 is shown in Fig. 2.

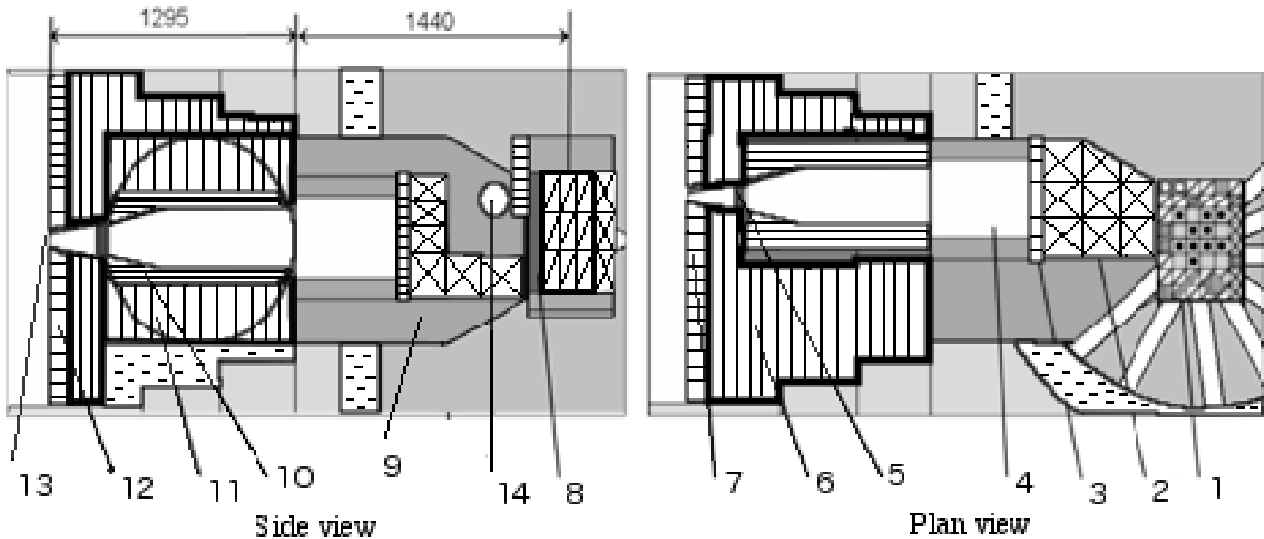


Fig. 2. Design of NCT-facility at the HEC-1. Designation: 1 – core; 2 – aluminum; 3 – lead shielding; 4 – cavity; 5 – Al + Bi filter; 6 – heavy concrete; 7 – second lead collimator; 8 – beryllium reflector; 9 – thermal column graphite; 10 – first lead collimator; 11 – heavy concrete shutter; 12 – lead wall; 13 – HEC-1 output; 14 – HEC-4.

Local estimation was used to calculate different energy neutrons fluxes: thermal ($E < 0.5$ eV), epithermal (0.5 eV $< E < 10$ keV), fast ($E > 10$ keV) and air kerma of secondary photons (tally F5) at the beam output. The kerma was calculated with DE and DF cards in MCNP program with formulae

$$K = \int_E \varphi(r, E) \delta_K^{air}(E) dE,$$

where $\varphi(r, E)$ is energy fluence, $\delta_K^{air}(E)$ is fluence-to-kerma conversion coefficients that were taken from the Ref. [6].

Calculations methods to define the core as a neutron source

Two ways of core modeling as neutron source in MCNP calculations can be used:

1. To carry out the criticality calculation (KCODE) and obtain spatio-energy distribution of the neutron source during this calculation.

2. To specify spatial distribution of the neutron source with the SDEF option. Fission rate distribution resulting from preliminary neutronic calculations using diffusion or Monte Carlo code can be used as the spatial distribution of the source.

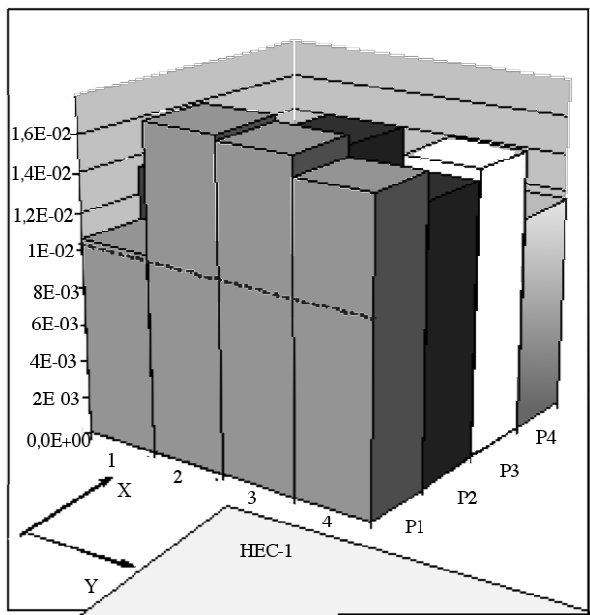
The first option was considered as the most accurate. Variant with criticality calculation requires accurate and detailed setting of the parameters of the core and the nearest to it reflector. In this option the geometric model for the parameters calculation of neutron horizontal beams was combined with detailed geometric core model that was previously made for calculations of the operational parameters of the reactor such as excess reactivity and core

power distribution. In the reference criticality calculation variant the core parameters are highly close to reality. Real geometry of FA, burnup distribution and CR location that were actual for summer of 2009 were considered.

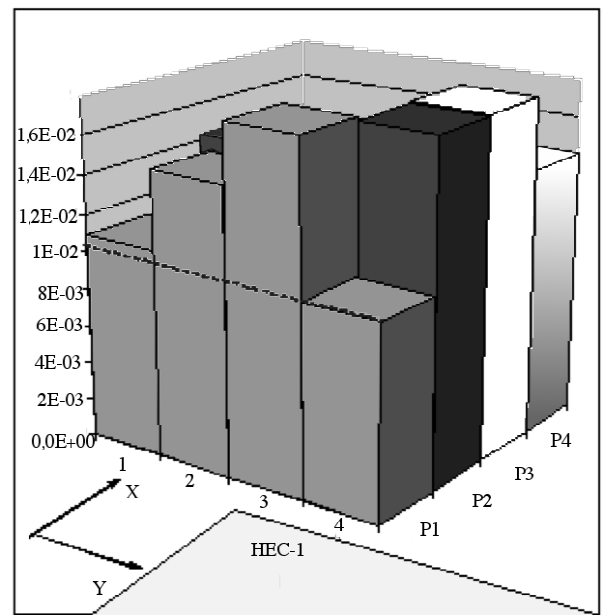
Using this combined model is not always useful. In this study we use this model as the reference but it is possible to make faster and less detailed model, using SDEF option, while accuracy remains the same.

In variants calculated with SDEF option FA are represented as homogeneous mixture of fuel, aluminum claddings of fuel elements (FE) and water. The central channel that can contain either water or an absorbing rod (B_4C) is considered as a special zone.

Source distribution is given in proportion to the fission reaction rate in nine levels of different height and was calculated by diffusion program. Distributions for two real loads of IRT MEFH reactor core are shown in Fig. 3.



Load 1 of the core



Load 2 of the core

Fig. 3. Fission reaction rate distribution in FA of the core at the axial level opposite to HEC-1. P1 is the nearest to the thermal column. Dash line is uniform fission rate distribution.

Results of calculation of beam parameters at HEC-1 with different spatial source distribution are given in Table 1. They are the following: mentioned above variant with the distribution proportional to the fission rate distribution («real» distribution for the load No. 1 of the reactor core – SDEF-1); the variant with the uniform distribution (SDEF-2) and the variant with cosine distribution (SDEF-3). As shown in Fig. 3 the «real» distribution in high level in front of HEC-1 is 1.5 times higher than the uniform one.

Presented results show that thermal and epithermal neutron flux in the variant with the uniform source distribution (SDEF-2) is fewer by 12 - 15 % than in SDEF-1 variant. The difference reaches 25 % in SDEF-3 variant. This difference is caused by the fact that neutron source in FA adjacent to thermal column is lower than «real» for the uniform and the cosine distributions. Contribution of these FA in neutron flux in HEC-1 is higher than others'.

Table 1. Characteristics of neutron beam at HEC-1 with different spatial source distribution

Characteristic	SDEF-1	SDEF-2	SDEF-3	KCODE
Neutron flux, $s^{-1}cm^{-2}$				
Thermal	$2.05 \cdot 10^9 \pm 0.012$	$1.81 \cdot 10^9 \pm 0.02$	$1.77 \cdot 10^9 \pm 0.03$	$2.06 \cdot 10^9 \pm 0.017$
Epithermal	$1.19 \cdot 10^9 \pm 0.023$	$1.02 \cdot 10^9 \pm 0.04$	$8.75 \cdot 10^8 \pm 0.06$	$1.12 \cdot 10^9 \pm 0.032$
Fast	$2.6 \cdot 10^8 \pm 0.05$	$2.2 \cdot 10^8 \pm 0.09$	$2.35 \cdot 10^8 \pm 0.13$	$2.3 \cdot 10^8 \pm 0.046$
Air kerma of photons, Gy/s	$2.57 \cdot 10^{-3} \pm 0.029$	$1.57 \cdot 10^{-3} \pm 0.11$	$1.81 \cdot 10^{-3} \pm 0.20$	$2.76 \cdot 10^{-3} \pm 0.041$

Presented results also show that variant with homogeneous core and the specified source distribution (SDEF-1) gives almost the same results as the variant with detailed core model and the criticality calculation (KCODE).

We considered variants of calculation where source distribution was specified in the same way (SDEF-1) but material composition of the core was given with different levels of details. In the first variant FA are represented as homogeneous mixture of fuel, aluminum claddings of FE and water, absorbing rods are placed at the real depth. Isotopic fuel composition is the same for all FA and corresponds to

the 30 % of burnup (SDEF-1, see Table 1).

The second variant differs from the first only in the isotopic fuel composition, corresponding to the 0 % of burnup (fresh core). In the third variant FA are also represented as homogeneous mixture, absorbing rods are replaced by water, isotopic fuel composition is the same for all FA and corresponds to the 0 % of burnup. The fourth variant differs from the first one in the hydrogen concentrations of FA materials which are increased by 1.2 times.

Calculation results for these variants are presented in Table 2.

Table 2. The HEC-1 parameters for different variants of material composition of the core

Parameter	Variant							
	1		2		3		4	
Neutron flux, $10^9 s^{-1}cm^{-2}$								
Thermal	2.05	(± 0.012)	2.03	(± 0.012)	2.07	(± 0.017)	2.00	(± 0.017)
Epithermal	1.19	(± 0.023)	1.15	(± 0.02)	1.19	(± 0.03)	1.10	(± 0.03)
Fast	0.26	(± 0.05)	0.21	(± 0.04)	0.22	(± 0.06)	0.21	(± 0.06)

Detailization of material components definition influences only the results of fast neutron flux calculation.

Presented results show that if the spatial source distribution is fixed then specifying parameters such as a heterogeneous structure of FA, fuel burnup, control rods location are not needed in details. This is caused by the fact that only those neutrons reach HEC-1 output that have not slowed down in the core. Absolute number and spatial distribution of produced fast neutrons are fixed in SDEF option. Some of fast neutrons slow down in the core and at the nearest reflector; others become a neutron source of the horizontal channel. Above-mentioned parameters have little effect on fast neutrons moderating in the core, they basically influence on absorbing of the thermal neutrons. Since a fission process is not considered in SDEF option, the inaccuracy of rate calculation of thermal neutron absorbing in the core has no effect on fast neutron source of horizontal channel.

Specifying of spatial source distribution known from preliminary reactor calculation allows making significantly easier core model for horizontal beams calculation. It should be noticed that for criticality calculation such parameters as FA heterogeneous structure, fuel burnup, CR location are needed to

specify in details due to the source forms on basis of real thermal and fast neutron balance in the core. It should be noted also that criticality calculation time is higher than time of calculation with SDEF option by 2 times. Thus, it is important to specify spatial source distribution for calculation with SDEF option, as close to real, as it possible.

Influence of core operation parameters

One of the main reactor parameters influencing on neutron source distribution is a number of FA in the core. Calculation results for one of the facility constructions at HEC-1 with the reactor core contains 16 or 12 FA are presented in Table 3.

The spatial source distribution in SDEF option is different for the core loads with 16 and 12 FA and is specified in accordance with fission reaction rate distribution of these loads.

Within the same reactor power neutron flux per FA is higher for the load with a smaller amount of FA. Thermal and epithermal neutron fluxes at the HEC-1 output are also higher by 25 - 30 % in this case. It should be noticed that FA number in the core usually remains the same, except changing fuel type in the reactor, such as the conversion from high-enriched fuel to low-enriched.

Table 3. HEC-1 parameters for IRT MEPHI Reactor core of 16 and 12 FA

Parameter	16 FA	12 FA
Neutron flux, $10^9 \text{ s}^{-1} \text{ cm}^{-2}$		
Thermal	0.60(± 0.016)	0.76(± 0.021)
Epithermal	0.90(± 0.02)	1.2(± 0.025)
Fast	0.34 (± 0.033)	0.44 (± 0.038)
Air kerma of photons, 10^{-3} Gy/s	1.4	1.6

Influence of fuel burnup distribution in the core and CR location on neutron beam characteristics at the HEC-1 was investigated. Two real loads of IRT MEPHI reactor containing 16 FA at different levels of burnup were considered.

Fission reaction rate distributions in FA of the core at the axial level opposite to HEC-1 for loads No. 1 and 2 are shown in Fig. 3. Insertion depth of

absorbing rods for load No. 1: AR = 250 mm, CR-1, 2 = 0, CR-3 = 344 mm; for load No. 2: AR = 200 mm, CR-1 = 0, CR-2 = 219, CR-3 = 580 mm (580 mm – rod is fully inserted). AR rod is located in cell 6-2, CR-2 rods are located in cells 4-2 and 4-5, CR-3 rods – in cells 3-3 and 4-3 (cell numbering is shown in Fig. 1, the first number is pattern column, the second number is pattern row).

Table 4. HEC-1 parameters for two different loads of the core containing 16 FA

Parameter	Load No. 1		Load No. 2	
Neutron flux, $10^9 \text{ s}^{-1} \text{ cm}^{-2}$				
Thermal	2.05	(± 0.012)	1.92	(± 0.012)
Epithermal	1.19	(± 0.023)	1.06	(± 0.023)
Fast	0.26	(± 0.05)	0.199	(± 0.046)

Obtained results show that thermal, epithermal and fast neutron fluxes for load No. 2 are less than for load No. 1 by ~5, ~10 and ~20 % respectively. So, mentioned above HEC parameter changes can be observed in transition from one load of the core to another during routine reactor operation.

Both changing of loading pattern and changing of control rods insertion depth influence parameters of HEC. Two calculations of load No. 1 were performed to estimate possible changes of these

parameters within the entire burnup cycle: at the beginning of the cycle when CR insertion depth was AR = 250 mm, CR-1, 2 = 0 mm, CR-3 = 344 mm; and at the end of the cycle when CR insertion depth was AR = 250 mm, CR-1, 2, 3 = 0 mm. The first variant is KCODE variant from Table 1. The second variant is also obtained from criticality calculation of load No. 1 but with another rods location, as shown in Table 5.

Table 5. HEC-1 parameters for load No. 1 of IRT MEPHI core with different locations of absorbing rods

Parameter	AR = 250 mm, CR-3 = 344 mm		AR = 250 mm, CR-3 = 0 mm	
Neutron flux, $10^9 \text{ s}^{-1} \text{ cm}^{-2}$				
Thermal	2.06	(± 0.017)	2.00	(± 0.017)
Epithermal	1.12	(± 0.032)	1.09	(± 0.031)
Fast	0.23	(± 0.046)	0.24	(± 0.06)

Changing of the absorbing rods location during the campaign has insignificant effect on HEC-1 parameters. It should be noticed that this conclusion is applicable only for mutual location of absorbing rods and the horizontal channel at IRT MEPHI.

Conclusions

Studies shows that it is not always useful to use highly detailed model of the core and horizontal beams especially with criticality calculation for the investigation of neutron beams parameters. It is more useful for operational neutronic calculation to have special detailed model of the core and the

nearest reflector. Within this model a spatial distribution of fission reaction rate is determined by criticality calculation. For calculations of horizontal neutron beams parameters it is suggested to use a model with simplified presentation of the core and with setting of neutron source distribution in SDEF option according to mentioned above fission rate distribution. Fission rate distribution can be taken from either diffusion model or Monte Carlo model calculations. We assume that the obtained results are applicable for any HEC with output located at more than 1.5 meter from the core.

Model with a homogeneous core and the specified source distribution (SDEF) gives almost

the same results as the variant with detailed core model and the criticality calculation (KCODE). The difference in SDEF-1 and KCODE calculations is not more than statistical error, while SDEF calculation takes about 2 times less than KCODE. Changes in the core (burnup, fuel reloads, CR

position) with the same FA number can cause changes of thermal and epithermal neutron fluxes by ~10 %. The change in fast neutron flux is ~20 %. If FA number changes dramatically, change of neutron flux reaches 30 %.

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ВПЛИВ ПАРАМЕТРІВ АКТИВНОЇ ЗОНИ НА ХАРАКТЕРИСТИКИ НЕЙТРОННИХ ПУЧКІВ ДОСЛІДНИЦЬКОГО РЕАКТОРА

ДРТ МІФІ – дослідницький реактор басейного типу, оснащений горизонтальними експериментальними каналами (ГЕК). Відомості про параметри, що впливають на просторовий розподіл полів випромінювання, є критичним для будь-якого експерименту. Дослідження були проведені для установки нейтрон-захватної терапії, розташованої на одному з ГЕК. Методами математичного моделювання було зроблено оцінку впливу параметрів активної зони на характеристики пучка на виході з ГЕК. Оцінено вплив способу моделювання джерела нейтронів у розрахунках методом Монте-Карло в програмі MCNP на отримувані параметри нейтронного й вторинного фотонного поля на виході каналу. Проведені дослідження показують, що завдання розподілу джерела нейтронів з розподілом швидкості реакції поділу в опції SDEF дає практично такі ж результати, як і розрахунок на критичність. Розрахунки показують, що зміни експлуатаційних параметрів активної зони неістотно впливають на характеристики нейтронного пучка на виході ГЕК.

Ключові слова: нейтрон-захватна терапія, MCNP, метод Монте-Карло, дозиметрія.

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ВЛИЯНИЕ ПАРАМЕТРОВ АКТИВНОЙ ЗОНЫ НА ХАРАКТЕРИСТИКИ НЕЙТРОННЫХ ПУЧКОВ ИССЛЕДОВАТЕЛЬСКОГО РЕАКТОРА

ИРТ МИФИ – исследовательский реактор бассейнового типа, оснащенный горизонтальными экспериментальными каналами (ГЭК). Сведения о параметрах, влияющих на пространственное распределение полей излучения, является критичным для любого эксперимента. Исследования были проведены для установки нейтрон-захватной терапии, расположенной на одном из ГЭК. Методами математического моделирования была произведена оценка влияния параметров активной зоны на характеристики пучка на выходе из ГЭК. Оценено влияние способа моделирования источника нейтронов в расчетах методом Монте-Карло в программе MCNP на получаемые параметры нейтронного и вторичного фотонного поля на выходе канала. Проведенные исследования показывают, что задание распределения источника нейтронов с распределением скорости реакции деления в опции SDEF дает практически такие же результаты, как и расчет на критичность. Расчеты показывают, что изменения эксплуатационных параметров активной зоны несут незначительно влияют на характеристики нейтронного пучка на выходе ГЭК.

Ключевые слова: нейтрон-захватная терапия, MCNP, метод Монте-Карло, дозиметрия.

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