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Determination of the Neutron Flux in the Reactor Zones with the Strong Neutron Absorption and Leakage

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Abstract: The procedures for the numerical and experimental determination of the neutron flux in the zones with the strong neutron absorption and leakage are described in this paper. Numerical procedure is based on the application of the SCALE-4.4a code system where the Dancoff factors are determined by the VEGA2DAN code. Two main parts of the experimental methodology are measurement of the activity of irradiated foils and determination of the averaged neutron absorption cross-section in the foils by the SCALE-4.4a calculation procedure. The proposed procedures have been applied for the determination of the neutron flux in the internal neutron converter used with the RB reactor core configuration number 114.

Keywords: Nuclear reactor, Neutron flux, Cross-section, Resonant absorption.

1 Introduction

Built in early 1958, as a zero power heavy water natural uranium critical assembly, the RB reactor [1] in VINČA Institute of Nuclear Sciences was later modified to operate with 2% enriched (^{235}U) metal uranium (1962) and 80% enriched (^{235}U) UO_2 fuel (1976-2002). In more than forty years of operation, the reactor has survived a series of modifications trying to follow modern safety operation demands and to overcome the ageing problems. Main changes of the RB reactor in the last twenty years have been related to the realisation of the fast neutron fields.

Experimental needs for the irradiation of samples and for the calibration of integral personal dosimeters in the fast neutron fields at the RB reactor initiated design and realisation of the coupled fast-thermal system HERBE [2] in late eighties of the last century. Characteristics of the HERBE system as well as the application examples are given in the references [2,3]. Further requirements related to the calibration of the absorbed neutron dose-rate-meters of large volume in the fast neutron fields have led to the realization of the RB reactor configuration with the inner neutron converter of cylindrical shell geometry and

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the inner diameter of 28 cm [4]. Experimental requirements for the irradiation of samples and for the calibration of large integral personal dosimeters (85 cm^3 volume) in the fast neutron fields resulted in the realization of the new boron neutron filter with the inner cylindrical hole [5]. Application of this boron neutron filter for the analyses of the fast neutron influence on the DNA molecules is described in the reference [6]. All the mentioned RB reactor core configurations used fuel elements containing high-enriched UO_2 (80 % of ^{235}U) dispersed in aluminum matrix.

In august 2002 all the high-enriched fuel elements from both the VINČA reactors have been sent back to Russia, as a Serbia and Montenegro contribution to the RERTR Program (Reduced Enrichment for Research and Test Reactors). Therefore, the recent need for the calibration of RHP (Radio-Chemical Prototype) neutron dosimeters of large volume (about 1700 cm^3) in the fast neutron fields at the RB reactor up to 10 Gy of cumulative neutron dose required design of new core configuration using the available metal fuel elements of 2% and 0.714% of ^{235}U enrichment. Complex analyses done showed that different RB reactor core configurations with 2% and 0.714% fuel elements combined with the existing neutron converters and filters, suitable for the RB reactor core configurations with 80% enriched fuel elements, could not provide the criticality of the system against the delayed neutrons even at the maximal moderator height in the reactor tank (about 170 cm, that corresponds to the total amount of the heavy water available). Solution has been found by the realization of a new inner neutron converter (INC) loaded by the 2% enriched metal uranium fuel elements with thin cadmium layer. The thermal core is composed of the 0.714% enriched fuel rods and 2% enriched metal uranium fuel elements arranged in the 12 cm square lattice, moderated and reflected by heavy water.

2 Description of the RB#114 Core Configuration

Basic design requirement was to ensure maximal absorbed dose in the fast neutron field in the centre of the INC using only available types and quantities of the nuclear fuel (708 of 2% enriched fuel elements and 177 fuel elements of natural enrichment in ^{235}U) and heavy water (about 5000 kg) at the RB reactor. Additional requirements were related to the safety of the RB reactor operation during the potential accidental sinking of the INC by the heavy water. Proposed RB#114 core configuration (cross-sections shown on Fig. 1 and Fig. 2, INC picture at Fig. 3) optimally satisfies the opposite requirements for the maximal dose and possibility for safe operation and shutdown of the reactor.

Number of analyses performed has shown that the experimental requirements can be fully satisfied only by the INC designed as an double rectangular aluminum box (dimensions of outer box: basis 34.2 x 33.2 cm^2, height 50 cm, aluminum thickness 0.3 cm) loaded with the 2% enriched fuel elements, which is partially dipped into the inner central reflector of the thermal core. The thermal core of this configuration consists of 44 fuel channels, each of them with 13 fuel elements of 2% enrichment in ^{235}U (total of 572 fuel elements) and 100 fuel rods of natural enriched metal uranium. Fuel channels are arranged in a 12 cm square lattice, moderated and reflected by heavy water. INC takes the central part of the thermal core (4 x 4 fuel cells of 12 cm lattice pitch, covering 48 x 48 cm^2 area). An additional cylindrical aluminum vessel (20.4 cm of outer radius) is welded at the top of this box.

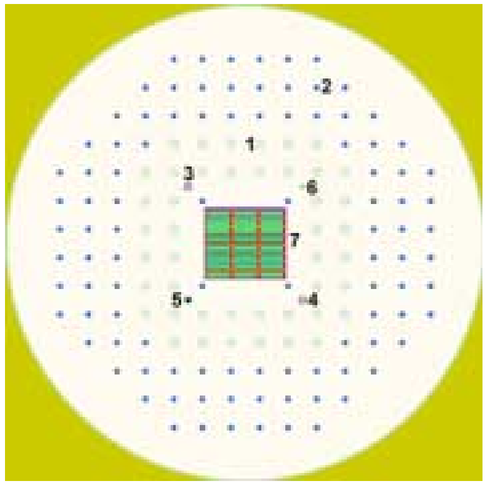

Fig. 1 - *Horizontal cross section of the RB#114 core configuration at a half height of the first fuel layer in the INC:*
1 - fuel channel with 13 fuel elements (2% enrichment);
2 - fuel rod with metal uranium of natural enrichment (0.714%);
3 and 4 - safety rods SR1 and SR2;
5 - heavy water level meter; 6 - safety rod SR3; and 7 - inner neutron converter.

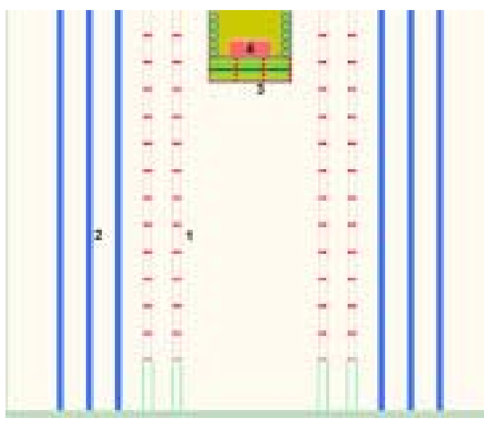

Fig. 2 - *Vertical cross section of the RB#114 core configuration (plane 6 cm north from the core centre):*
1 - fuel channel with 13 fuel slugs (2% enrichment);
2 - fuel rod with metal uranium of natural enrichment (0.714%);
3 - INC with 2% enriched fuel elements; and
4 - neutron dosimeter for calibration in the fast neutron field.

Fig. 3 - *Aluminum vessel of the INC at the position inside the RB reactor tank.*

There are 119 fuel elements of 2% enrichment placed in three layers into the INC. A rectangular cadmium box with dimensions 26.1 x 25.1 cm^2, height 25.4 cm, cadmium thickness 0.1 cm is placed on the third fuel layer holding the inner aluminum vessel of 0.1 cm thickness. All the lateral sides of the INC are also loaded by the 2% enriched fuel elements. Cadmium strongly absorbs the thermal neutrons providing the fast neutron field in the central INC hole, which serves as a space for the irradiation of large volume samples.

3 Calculation Methodology

Application of the standard procedures for the neutron flux calculation, based on the multigroup transport methods in one-dimensional (1D) and two-dimensional (2D) models of the fuel cells, which prepare averaged macroscopic neutron cross-sections, for the further few-groups diffusion methods in three-dimensional (3D) core and reflector models is not suitable due to limitations of the diffusion approximation to take into account processes in the zones with the strong neutron absorption and leakage, as it is in the INC of the RB#114 core configuration. This problem can be solved by the application of the Monte Carlo method. Nowadays two referent Monte Carlo codes MCNP-4C [7] and KENO-V.a [8] are widely used. MCNP-4C code enables high accuracy of the calculation due to continuous energy cross-section representation in the data libraries, but the manner how the energy distributions of the reaction rates and neutron fluxes are obtained makes it time consuming. Multigroup KENO-V.a code of the SCALE-4.4a code system [9] is faster than MCNP-4C (for the factor over 10), but the accuracy of the calculation results is dependent of the accuracy of the Dancoff factor [10] determination. In this paper KENO-V.a with the 238 energy group neutron cross-section library [11] based on the ENDF/B-V evaluation is used for the determination of the neutron flux and the absorbed neutron dose in the INC of the RB#114 configuration.

For the treatment of the resonant neutron processes in the thermal nuclear reactor cores SCALE - 4.4a code system uses simplified model of one isolated fuel element (usually a fuel rod in an infinite moderator is considered) where the influence of the surrounding ones is included by the Dancoff factor. There are several well-known analytical and numerical methods for the Dancoff factor determination implemented in the CSAS control module [12] and widely used in the calculations of the regular lattice nuclear reactors. In all these procedures, as well as in the recent ones based on the Monte Carlo techniques [13], determination of the Dancoff factor is based on its definition derived from the theory of the first neutron collision probability. For the most widely used arrangements, such as regular square or hexagonal lattices, the SCALE-4.4a code system provides Dancoff factors. However, in the more complex geometries, the Dancoff factors are to be supplied as input by the user.

In order to provide a solution for any geometry for which a flux calculation by collision probability method is possible, and to include resonance interference and overlapping effects in the SCALE-4.4a code system, a procedure based on the equivalence principle is proposed for effective Dancoff factor calculation. VEGA2DAN code [14], developed in VINČA Institute, determines the Dancoff factor by a procedure ensuring the equality of the total neutron absorption $\overline{\Sigma}_{ai}$ in the considered resonant zone and in the defined energy range determined by - 1) numerical solution of the neutron slowing-down equation in the real geometry of the heterogeneous system and 2), simplified equivalent model of one fuel channel (slab, sphere, cylinder or cylindrical shell geometry) where the surrounding is included through the Dancoff factor to be determined. The goal of this procedure is to ensure the conservation of the collision probabilities in the fuel, as well as the conservation of the effects of the spatial resonant self-shielding and the interference of the resonances of different nuclides.

Proposed procedure is based on the assumption that for all the energy groups, which contain resolved resonances, a Dancoff factor value can be found for given resonant zone, which provides conservation of the total neutron absorption, although neutron absorption could not be conserved for each energy group. It is chosen that this equivalence be established in the energy range from $E_l = 1$ to $E_u = 300$ eV for the thermal reactor applications. There are two reasons for such a selection. All the resolved resonances of the nuclides ^{235}U and ^{239}Pu in the ENDF/B-V evaluation of the neutron cross-sections are present in this range. In thermal reactors there are more than 90% of the ^{238}U resonant absorption in the selected energy range also.

For the numerical solution of the neutron slowing-down equation, VEGA2DAN code introduces multigroup procedure with the narrow energy groups of equal letargy width. The energy groups are chosen so that there are more than 50 groups in the energy interval corresponding to the maximal change of the neutron letargy for the heaviest resonant nuclide. Procedure gives the collision probabilities in the 2D *x-y* and *r-θ* geometries.

The proposed procedure is based on the Nordheim integral theory [15], which is valid only for the heterogeneous systems with one resonant zone (i). This theory supposes that the neutron flux in the moderator and fuel cladding follows $1/E$ distribution and that at the zonal boundaries there is an isotropic flux distribution over the angle. In an iterative procedure, starting from the initial Dancoff factor value c_i, solution of the Nordheim neutron slowing-down equation is found, from which the new Dancoff factor value is determined so that the total neutron cross-section in the zone i, according to the Nordheim theory, is equal to the exact value $\overline{\Sigma}_{ai}$. This iterative procedure proceeds until the convergence of the effective Dancoff factor value.

VEGA2DAN code calculates the Dancoff factors for the complex systems, where the resonant neutron absorption can be determined in the 2D *x-y* and *r-θ* models with sufficient accuracy. Unfortunately, mentioned 2D models are not capable to ensure the Dancoff factors for the 2% enriched fuel in the INC with the high accuracy, due to specific arrangement of the fuel elements. In order to overcome this problem, an additional feature is incorporated in the VEGA2DAN code providing the possibility to enter exact value of the macroscopic neutron absorption cross-section $\overline{\Sigma}_{ai}$ as an input parameter, continuing determination of the Dancoff factor as described above. In this paper MCNP-4C Monte Carlo code has been used for the calculation of the neutron cross-section $\overline{\Sigma}_{ai}$ in the fuel elements of 2% enrichment in the INC, due to its capability to enable sufficient accuracy for the reliable modelling of the neutron resonant absorption in these fuel elements.

At the same time, there was an interest to determine Dancoff factors by the simple 1D models using only existing features of the CSAS control module. Opposite to the case of fuel rods with the natural enriched metal uranium, for which the CSAS module gives exact values of the Dancoff factor, its calculation for the fuel channels with the 2% enriched metal uranium fuel is not simple. In these elements there are inner and outer moderators, while the nuclear fuel is one of nine cylindrical shell layers (Fig. 4). The dimensions of the Wigner-Seitz's cell with the 2% enriched uranium fuel are given in Table 1. In the reference [16] the nine-zonal cylindrical 1D model is described in more details. This model used with the combined model described in the reference [17] can give high accuracy of the Dancoff factors and effective neutron multiplication factor. Dancoff factor for the cylindrical shell of inner radius r_I and outer radius r_O in this model is determined according to the equation

$$c = c_I + c^* \frac{r_O^2}{r_O^2 + r_I^2}, \tag{1}$$

where the factors c_I and c^* include the contribution of the inner and outer moderator and cladding, respectively. According to the definition, contribution c_I is determined as the Dancoff factor for the 1D multizonal cell with the vacuum boundary condition at the outer cylindrical surface, excluding is such a way the contribution of the outer fuel cladding and moderator. Described nine-zonal 1D model with the vacuum boundary condition is sufficiently reliable for the determination of the factor c_I for the RB reactor cells with the 2% enriched fuel in the thermal core. As the high accuracy of the Dancoff factors in the CSAS module is ensured only for the three-zonal cells in the square or hexagonal lattice with the reflective boundary condition, it is optimal to calculate the c^* factor using three-zonal model where the first zone (fuel zone) has radius equal to the outer radius of the fuel cylindrical shell (r_O). In the case of the RB reactor fuel cells all the claddings and all moderator zones around the fuel shell have to be homogenized in one effective cladding and one effective moderator zone.

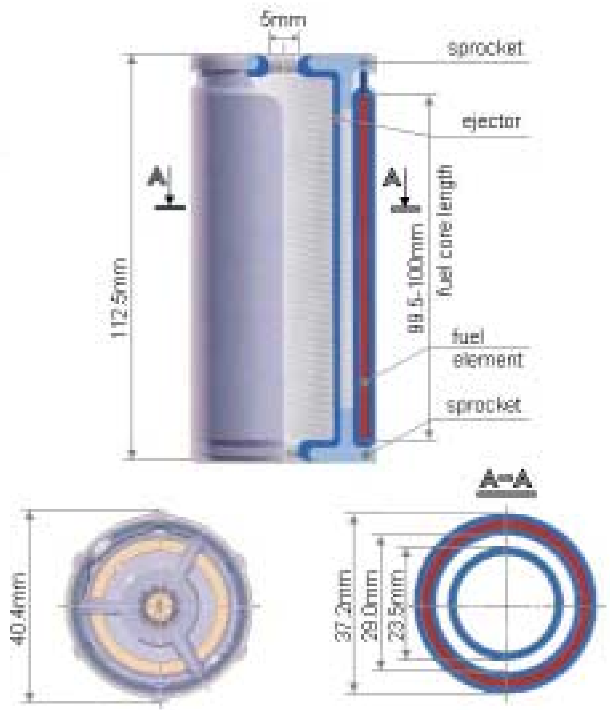

Fig. 4 - *TVR-S fuel element with 2% enriched metal uranium.*

For the determination of the Dancoff factor for the fuel elements of the INC, mentioned combined model is improved keeping the described way of the factor c_I determination, while the factor c^* is obtained as mean value of two parameters c_L^* and c_R^*. Here the c_L^* represents contribution of the outer effective fuel cladding and outer effective moderator in the hexagonal cell, while c_R^* describes the contribution of the outer materials for the large moderator volume that practically means $c_R^* = 0$.

Table 1
Dimensions of the Wigner-Seitz's cell with the 2 % (^{235}U) enriched uranium fuel

Zone	Material	Radius [cm]	
		2D and 3D-Model	1D-Model
1	Heavy water	1.05000	1.00870
2	Cladding (Al-SAV)	1.02000	1.20000
3	Heavy water	1.45000	1.42153
4	Cladding (Al-SAV)	1.55000	1.55000
5	Metal fuel (2 % ^{235}U)	1.75000	1.75000
6	Cladding (Al-SAV)	1.85000	1.87109
7	Heavy water	2.05000	2.05000
8	Fuel channel (Al-YU)	2.15000	2.16217
9	Heavy water	12 cm square cell	7.18096

Values of the Dancoff factors in the thermal core cells of the RB#114 configuration, for both the fuel types (0.714% and 2% enrichment) are given in Table 2. Dancoff factors obtained for the INC fuel and presented in the Table 3 demonstrate good agreement between the simplified 1D model and the detailed model used in the VEGA2DAN/MCNP-4C calculation methodology.

4 Results

Although the Dancoff factors for the fuel elements with the natural and 2% enrichment in the thermal core and INC of the RB#114 configuration can satisfactory be determined by the CSAS control module using mentioned improved combined 1D model, in all the calculation of the neutron flux as well as the absorbed and equivalent neutron dose rate presented in this paper the Dancoff factors obtained by the VEGA2DAN/MCNP-4C methodology are used, due to the higher accuracy.

Experimental determination of the neutron flux has been performed by the measurement of the activity ratio for the two gold foils, one of them irradiated at the top of the horizontally placed detector in the INC of the RB#114 core configuration, second irradiated at the position with the well known neutron flux in the VINČA standard graphite column with the Ra-α-Be neutron source [18]. Unperturbed thermal and total neutron fluxes as well as the fluxes perturbed by the 30 mg gold foil at three positions in the graphite column are shown in Tables 4 and 5. The averaged neutron absorption cross-sections for the ^{197}Au in the 30 mg gold foil for these positions are given in Table 6.

Table 2

Dancoff factor in the cells of the thermal core of RB#114 configuration.

Model	Cell with 2% enriched fuel, 12 cm lattice			Cell with 0.714% enriched fuel, 12 cm lattice		
	c_I	c^*	c	c_I	c^*	c
Nine-zonal 2D *x-y* model with reflective boundary condition (VEGA2DAN code)	-	-	0.23159	0	0.03338	0.03980
Detailed 3D *x-y-z* model with reflective boundary condition (VEGA2DAN/MCNP-4C)	-	-	0.23005	0	0.03980	0.03338
Nine-zonal 1D (cylind.) combined model with reflective boundary condition (CSAS)	0.20894	0.00340	0.21085	0	0.00206	0.00206

Table 3

Dancoff factor in the INC fuel elements of the RB#114 core configuration.

Model	Fuel of 2% enrichment (^{235}U) in the dry INC			Fuel of 2% enrichment (^{235}U) in the INC dipped in heavy water		
	c_I	c_L*	c	c_I	c_L*	c
Detailed 3D model of the RB#114 configuration (VEGA2DAN/MCNP-4C)	-	-	0.65200	-	-	0.30350
Nine-zonal 1D combined model (CSAS)	0.43059	0.99812	0.71030	0.20894	0.54794	0.36247

Table 4

Neutron fluxes in the standard graphite column (unperturbed).

Distance from the neutron source [cm]	Neutron flux [cm^{-2} s^{-1}] / Uncertainty [%]				
	Thermal			Total	
	Measurement (Ref. [18])	Measurement (Ref. [19])	Calculation (MCNP-4B, Ref. [20])	Measurement (Ref. [19])	Calculation (MCNP-4B, Ref. [20])
45	7.721 10^3 (±10)	7.6850 10^3 (±7.5)	7.5608 10^3 (±1.9)	9.7621 10^3 (±8.3)	9.7385 10^3 (±2.1)
55	5.585 10^3 (±10)	5.7119 10^3 (±8.1)	5.7451 10^3 (±1.7)	6.6514 10^3 (±8.6)	6.6922 10^3 (±1.6)
65	4.097 10^3 (±10)	4.0310 10^3 (±8.7)	4.0319 10^3 (±1.8)	4.4143 10^3 (±8.8)	4.4042 10^3 (±1.8)

Table 5

Neutron fluxes in the standard graphite column perturbed by the 30 mg gold foil.

Distance from the neutron source [cm]	Neutron flux [cm^{-2} s^{-1}] / Uncertainty [%]			
	Measurement (Ref. [19])		Calculation (MCNP-4B, Ref. [20])	
	Thermal	Total	Thermal	Total
45	7.4373 10^3 (±7.5)	9.4999 10^3 (±8.3)	7.3171 10^3 (±1.6)	9.4769 10^3 (±2.0)
55	5.6102 10^3 (±8.1)	6.5418 10^3 (±8.6)	5.6428 10^3 (±1.7)	6.5819 10^3 (±1.6)
65	3.9536 10^3 (±8.7)	4.3353 10^3 (±8.8)	3.9545 10^3 (±1.8)	4.3254 10^3 (±1.8)

Table 6

Averaged neutron absorption cross sections for the ^{197}Au in the 30 mg gold foil in the standard graphite column.

Distance from the neutron source [cm]	Averaged neutron absorption cross sections (energy range 10^{-5} eV - 15 10^6 eV) for the ^{197}Au in the 30 mg gold foil [10^{-28} m^2]	
	Calculation VEGA2*	Calculation MCNP-4C
45	75.445	75.321 (±3.0%)
55	78.244	78.700 (±2.0%)
65	79.734	80.648 (±2.3%)

*VEGA2DAN is one module of the VEGA2 assembly spectrum code [21]

The influence of the neutron energy spectrum at the position of the foil irradiation on the averaged neutron absorption cross-section (and thus on the foil activity) is illustrated by the results given in the Table 7. The results of the total neutron flux measurements, obtained by the foil activity measurements applying calculated averaged ^{197}Au neutron absorption cross-sections, as well as the results of the KENO-V.a Monte Carlo calculations with 238-group neutron cross-section data library for the detailed 3D model of the INC partially dipped into the heavy water are given in the Table 7 also. Neutron spectrum and the averaged neutron absorption cross-section at the position of the foil irradiation in the INC are very sensitive to the material composition of the RHP dosimeter. Data used in the calculation model, provided by the RHP dosimeter manufacturer, are shown in Table 8. Absorbed and equivalent neutron dose rate calculated from the determined neutron fluxes according to the NCRP-38 standard [22] at the position of neutron dosimeter for 1 W fission power of the RB reactor are presented in Table 9. Nuclear safety aspects of the RB#114 core configuration operation are analysed for different accidental conditions by the KENO-V.a code. Results of the nuclear safety analyses are given in Table 10. It has been demonstrated that the existing safety systems at the RB reactor ensure safe operation and control of the proposed configuration. The ejection of the cadmium box or the whole INC during the operation, that could cause super criticality of the system, is disabled by the mechanical fixation of the INC from both the top and bottom sides (Fig. 3).

Table 7
Averaged values of the neutron absorption cross-section for the nuclide ^{197}Au and the total neutron flux.

Position of the gold foil	Averaged value of the ^{197}Au neutron absorption cross-section [10^{-28} m^2]	Total neutron flux [cm^{-2} s^{-1}]	
		Measurement (reactor power 24.7 W)	Calculation (KENO-V.a) for 1 W of RB reactor fission power
Top of the RHP dosimeter in the partially dipped INC	13.7 ± 1.3	$2.490 \cdot 10^7$ (±11%)	$1.008 \cdot 10^6$ (±3%)
At the half height of the vertical channel 31 cm from the centre of the RB#114 core configuration	63.8 ± 2.4	$2.630 \cdot 10^8$ (±7%)	$1.065 \cdot 10^7$ (±1%)

Table 8
Material composition of the homogenised RHP dosimeter used in the MCNP-4C calculation model.

Nuclide	Nuclide concentration [10^{24} cm^{-3}]
Al	$5.35 \cdot 10^{-4}$
Si	$6.42 \cdot 10^{-4}$
Fe	$6.46 \cdot 10^{-4}$
Cu	$2.84 \cdot 10^{-4}$
Sn	$1.82 \cdot 10^{-4}$
Au	$5.50 \cdot 10^{-7}$
Pb	$6.96 \cdot 10^{-5}$

Table 9
Absorbed and equivalent neutron dose rate (according to NCRP-38 standard) at the position of neutron dosimeter for 1 W fission power of the RB reactor.

Energy group	Lower energy [eV]	Upper energy [eV]	Neutron flux [cm^{-2} s^{-1}]	Absorbed neutron dose rate [Gy h^{-1}]	Equivalent neutron dose rate [Sv h^{-1}]
1	$8.0000 \cdot 10^5$	$1.0500 \cdot 10^7$	$3.7394 \cdot 10^5$	$5.1834 \cdot 10^{-2}$	$4.9160 \cdot 10^{-1}$
2	$4.6500 \cdot 10^3$	$8.0000 \cdot 10^5$	$6.1014 \cdot 10^5$	$2.9313 \cdot 10^{-2}$	$2.7058 \cdot 10^{-1}$
3	$4.6500 \cdot 10^{-1}$	$4.6500 \cdot 10^3$	$3.6240 \cdot 10^5$	$7.3520 \cdot 10^{-3}$	$1.4926 \cdot 10^{-2}$
4	$1.0000 \cdot 10^{-5}$	$4.6500 \cdot 10^{-1}$	$1.7415 \cdot 10^3$	$3.6005 \cdot 10^{-5}$	$7.2011 \cdot 10^{-5}$
Total	$1.0000 \cdot 10^{-5}$	$1.0500 \cdot 10^7$	$1.3482 \cdot 10^6$	$8.8535 \cdot 10^{-2}$	$7.7717 \cdot 10^{-1}$

Neutron flux energy spectrum in the INC of the RB#114 core configuration presented in Fig. 5 shows that the RB#114 core configuration with the designed INC can be successfully used for the needs of calibration and irradiation of large volume samples in the fast neutron fields. However, obtained energy spectrum of the neutron flux, compared with the neutron spectrum of the fast power reactors, is shifted towards the lower neutron energies.

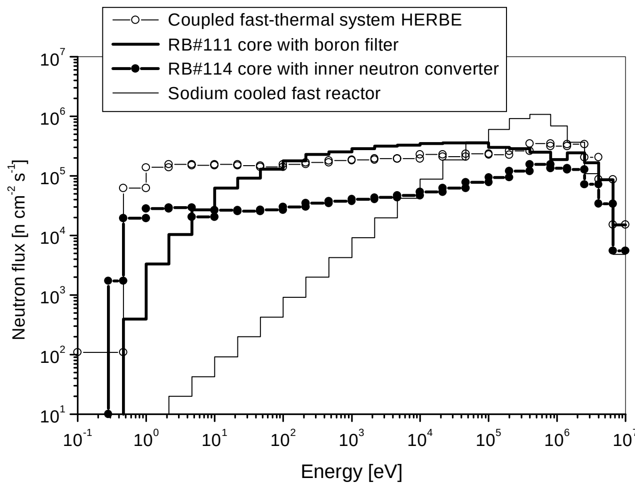

Fig. 5 - *Energy spectrum of the neutron flux at the position of neutron dosimeter for 1 W fission power of the RB reactor.*

Table 10

Results of the nuclear safety analyses of the RB#114 core configuration (SR - safety rod, HWLM - heavy water level meter).

RB#114 core configuration condition	Effective neutron multiplication factor $k_{eff} \pm 1\sigma$
Critical level of heavy water (161.50 cm) - experiment	1.0000 ± 0.0015
Criticality with INC, H_m = 161.3 cm	1.0010 ± 0.0003
Super criticality with INC, H_m = 165.0 cm	1.0040 ± 0.0003
RB#114 without INC, H_m = 161.3 cm	0.9953 ± 0.0003
Basic design accident: sunken INC, H_m = 161.3 cm	1.0150 ± 0.0003
Sunken INC, Cd box ejected, H_m = 161.3 cm	1.0210 ± 0.0003
With INC, all safety rods inserted, H_m = 161.3 cm	0.9655 ± 0.0003
With INC, SR1 or SR2 inserted, H_m = 161.3 cm	0.9977 ± 0.0003
With INC, HWLM inserted, H_m = 161.3 cm	0.9874 ± 0.0003
With INC, SR3 inserted, H_m = 161.3 cm	0.9923 ± 0.0003
Sunken INC (at H_m/2), H_m = 161.3 cm	1.0013 ± 0.0003
Sunken INC (at H_m/2), Cd box ejected, H_m = 161.3 cm	1.0413 ± 0.0003
Sunken INC (at the bottom), H_m = 161.3 cm	1.0051 ± 0.0003
Sunken INC (at the bottom), Cd box ejected, H_m = 161.3 cm	1.0137 ± 0.0003
INC out of the reactor, sunken by ordinary water	0.4100 ± 0.0005

5 Conclusion

Proposed calculation and experimental procedures, based on the SCALE-4.4a code system application and measurement of the activity of irradiated foils, enable reliable determination of the neutron flux in the zones characterized by the dominant fraction of the fast neutrons, strong neutron absorption and leakage. In the paper it is shown that the full heterogeneous core of complex geometry, as the RB#114 core configuration, can be calculated by the equivalence approach implemented in the VEGA2DAN code. The possibility for the irradiation of large samples in the field of fast neutrons at the RB reactor with the designed INC is also presented.

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