COMPARING FAST NEUTRON TRANSFER CALCULATIONS WITHIN CODE PACKAGE KATRIN-2.0 ACROSS VARIOUS OPTIONS FOR DESCRIBING THE CORE OF VVER-440

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1. Introduction

Calculation of radiation exposure for VVER reactor vessel is a key task for the following determination of their burn-up life. This problem is especially topical for VVER-440 reactors of the first generation which now require substantiation report of safe operation over designed service life.

The estimation of radiation exposure of the reactor vessel involves calculation of fast neutron fluence (E≥0.5 MeV) on welded joints and basis metal of the reactor vessel.

The result of neutron fluence calculation and the associated estimation error are affected by the precision of fission neutron source definition in the reactor core. This study examines two approaches to defining fission neutron source in the fuel assembly: specifically, assembly-wise distribution, as well as rod-wise distribution approaches.

The main objective of this work is to characterize the difference in calculation results of neutron fluence (E≥0.5 MэB) on the VVER-440 reactor vessel at rod-wise and assembly-wise definition of fission neutron source in peripheral fuel assemblies, and to estimate the accuracy of fluence calculation for each source definition method, by comparing them with some experimental data. The present study examines results of two different experiments carried out on the block of No.1 of Kola Nuclear Power Plant (V-230): each involving activity measurement for templates cut out on the internal reactor vessel surface and activity measurement of neutron-activation detectors of Niobium on the external reactor vessel surface.

2. Calculation methods

2.1. Description of calculation method

Calculation of neutron transfer at set source was carried out by KATRIN software. KATRIN (Three-dimensional neutral and charged particle transport) is intended for solving the neutron, photon and charged radiation transfer multigroup equation using a method of discrete ordinates in three-dimensional r,ϑ,z geometry.

We calculated 47 power groups of neutrons in the range from 10^{-9} to 17.3 MeV using problem-focused library of constants BGL440 [2]. The geometry input to the problem was prepared with preprocessor (mesh generator) as a 'mixmap' format file; the file also contained information on shares of materials for spatial cells containing several basic materials.

2.2. Description of geometrical model

The geometry of reactor unit VVER-440 set using combinatorial geometry methods in NCGSIM language of MCU [1] program is presented in Fig. 1, 2.

3D-calculation using KATRIN software in r,ϑ,z geometry was carried out for a sector 30° of rotational symmetry on spatial r,ϑ,z mesh of 218×120×189 =4 944 240 cells in P_3S_8 approximation with the accuracy of internal iterations convergence at 10^{-3}. Mirror boundary condition was used on the borders of rotational symmetry sector 30° on ϑ variable.

Radial tracing method implemented in ConDat [3] and ConSource [4] converters was used for converting combinatorial setting of geometry and source on a problem mesh.

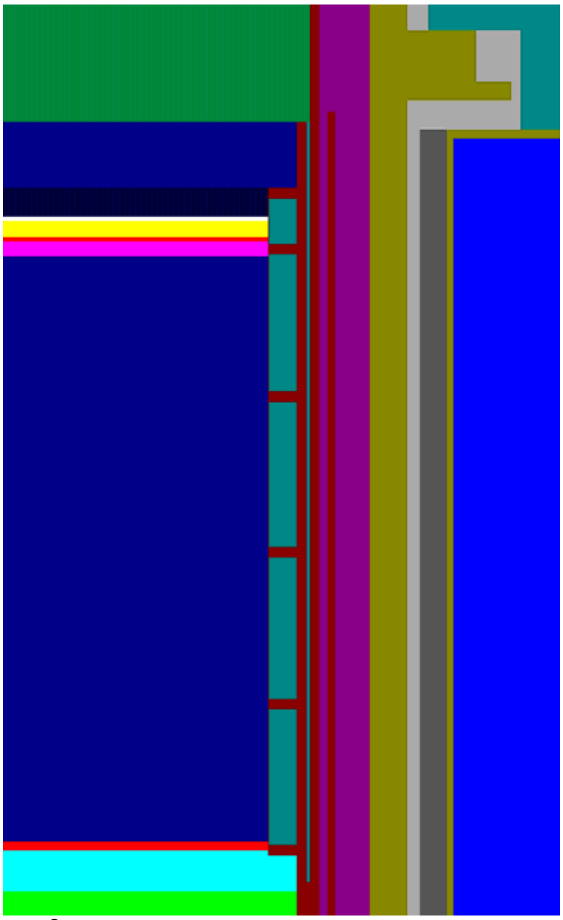

Fig.1. Axial section 3D $r,\ \vartheta,\ z$ of radiation protection model of VVER-440(V-230) reactor unit.

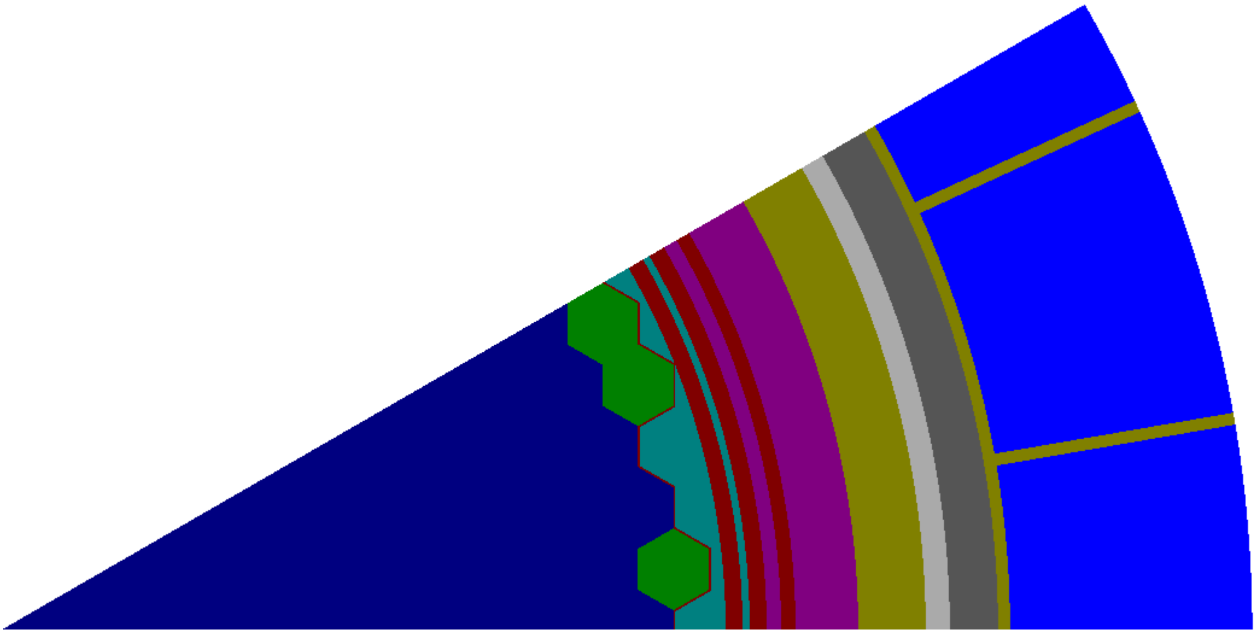

Fig.2. Lateral section r, ϑ, z for radiation protection model of VVER-440(V-230) reactor unit at z=100cm from the bottom of the active zone

2.3. Determining fission neutron source and fission neutron spectrum in the active zone

The source of fission neutrons *Q* is defined as a number of fission neutrons emitted per time unit within a volume unit of reactor core. When determining the average source volume capacity of one operating period, it is assumed that the average number of fissions in one fuel assembly within one operating period is proportional to burn-up increment $\Delta\rho$ within operating period in this fuel assembly. Thus, if ν is a number of secondary neutrons formed in single fission, then the source of neutrons is proportional to product $\Delta\rho \cdot \nu$:

$$Q \sim \Delta\rho \cdot \nu \qquad (1)$$

Assembly-wise and rod-wise burn-up estimates are produced with neutron and physical calculations, using certified codes BIPR-7A and PERMAK [5].

The burn-up in peripheral fuel assemblies is uneven across rods, and has a gradient from the center of the reactor core to periphery. Assembly-wise definition of fission neutron source does not account for non-uniformity of fuel assembly rods' burn-up. The dependence of relative source on position of fuel rod in the assembly is presented in Fig. 3.

The peripheral row rods of fuel assembly have the greatest influence on fast neutron flux on the reactor vessel. As Figure 3 shows additional conservatism in comparison with rod-wise calculation is exercised when using assembly-wise definition of the source in peripheral rods of peripheral fuel assemblies.

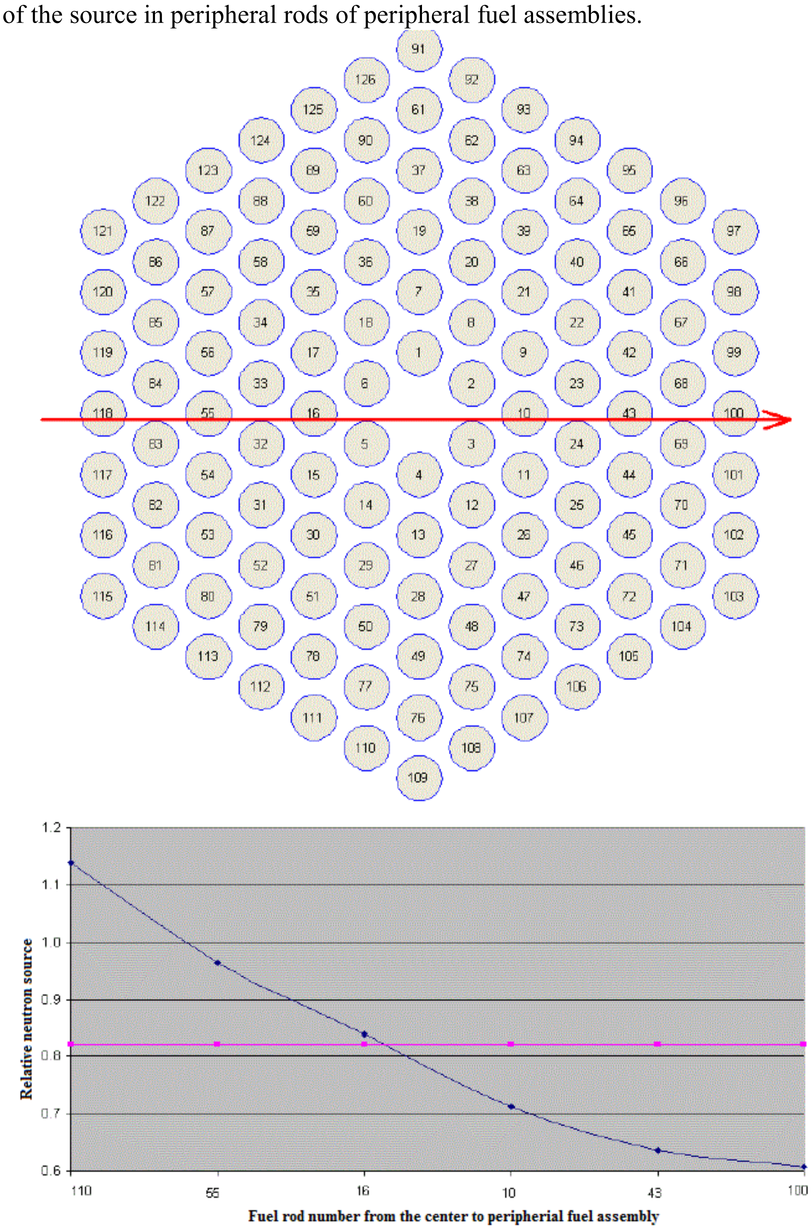

Fig.3. Distribution of relative fission neutron source by radius of peripheral fuel assemblies (from the center to periphery) at assembly-wise and rod-wise source settings.

3. Comparison of calculation results with experiment

The method recommended by the Russian (RB-018-01, RB-007-99) and the foreign (for example, RG1.19) normative documents based on the comparison with activation experimental data on threshold reactions was used for estimating fast neutron fluence calculation accuracy. Such experimental data can be obtained by measuring activity of templates cut out from the internal surface of a first generation VVER-440 reactor vessel, or activity of threshold detectors placed on the external surface of a VVER-440 reactor vessel.

When using templates data, the comparison is carried out based on activity of ^{54}Mn which accumulates in reactor vessel metal as a result of ^{54}Fe (n, p) ^{54}Mn threshold activation reaction. The Russian and the foreign specifications recommend using quite a wide set of threshold detectors, where activation occurs on reactions (n, p), (n, n'), (n, f), (n, α) for comparing activity of neutron and activation detectors. Detectors activated on reactions ^{93}Nb (n, n') ^{93m}Nb, ^{58}Ni (n, p) ^{58}Co and ^{54}Fe (n, p) ^{54}Mn are used most widely for fast neutrons. Effective threshold energy of these detectors' activation is about 1 MeV, 2.5 MeV and 3 MeV correspondingly.

The data on templates' activity of the joint weld No. 4 material and the basis metal [6] which were cut out from the internal surface of the reactor vessel of power units No.1 and No.2 of Kola Nuclear Power Plant in 2001 and in 1999 were used.

Experimental data on templates activity were provided by National Research Centre "Kurchatov Institute".

The data on neutron and activation detectors *^{93}Nb (n, n'),* irradiated on the external surface of the power unit No. 1 reactor vessel of Kola Nuclear Power Plant during the 26th operating period (2002-2003) were used.

Experimental data on activity of detectors were provided by National Research Centre "Kurchatov Institute". The experiment is described in [7] in detail.

Assembly-wise and rod-wise burn-up for determination of fission neutron source were calculated using software BIPR-7A and PERMAK-A respectively.

A summary comparison of all results is presented in the following table and in Figures 4-6.

In the table we present the average deviation of calculated vs. experimental (R/E-1) activity values for assembly as well as rod-wise source setting computed across all experimental points.

Comparison of rated (R) and experimental (E) activities for welded joint No. 4 of VVER-440

Welded joint No.4		Basis metal	
Rod-wise source	Assembly-wise source	Rod-wise source	Assembly-wise source
Templates of block No.1 of Kola NPP			
2.9%	20.4%	1.0%	16.6%
Detectors of block No.1 of Kola NPP			
^{93m}Nb(n,n')^{93}Nb			
-11.7%	-4.2%	0.4%	5.9%

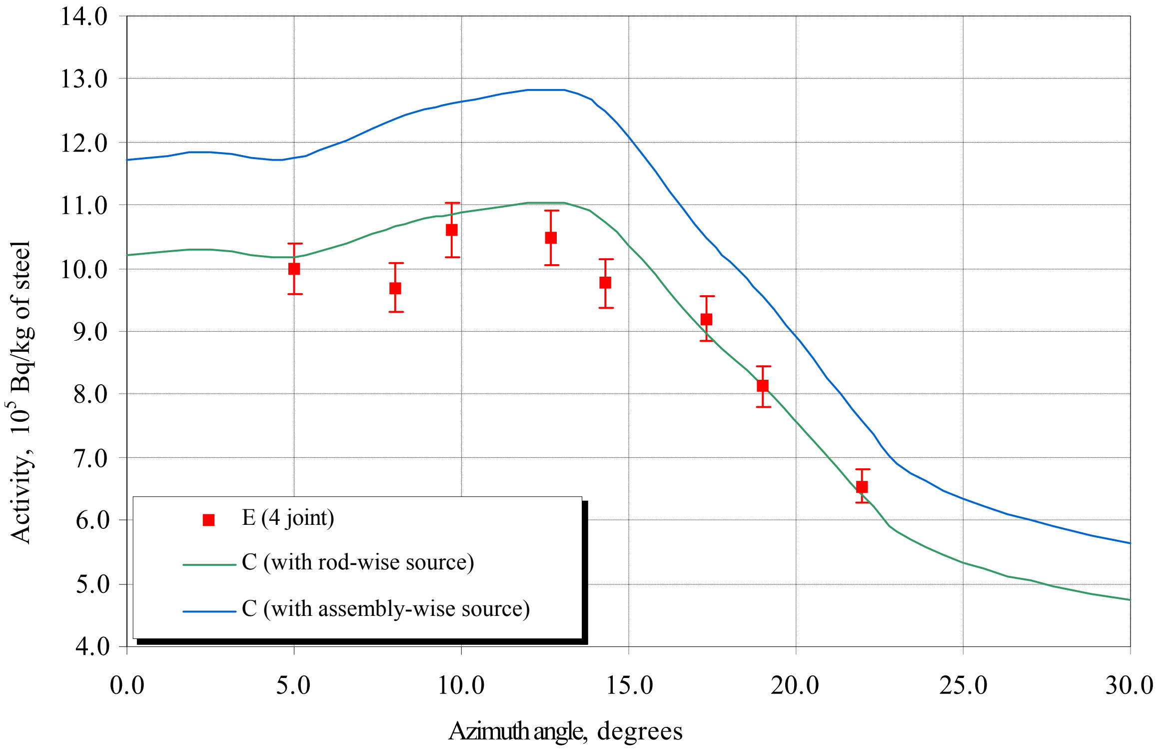

Fig.4. Rated (R) and experimental (E) activity distributions for ^{54}Mn in templates of welded joint No. 4 (z=30cm) of reactor vessel of Kola Nuclear Power Plant power unit No.1. Experimental data obtained from National Research Centre “Kurchatov Institute”. The measurement error of templates activity is $2\sigma_{exp} = \pm 4\%$.

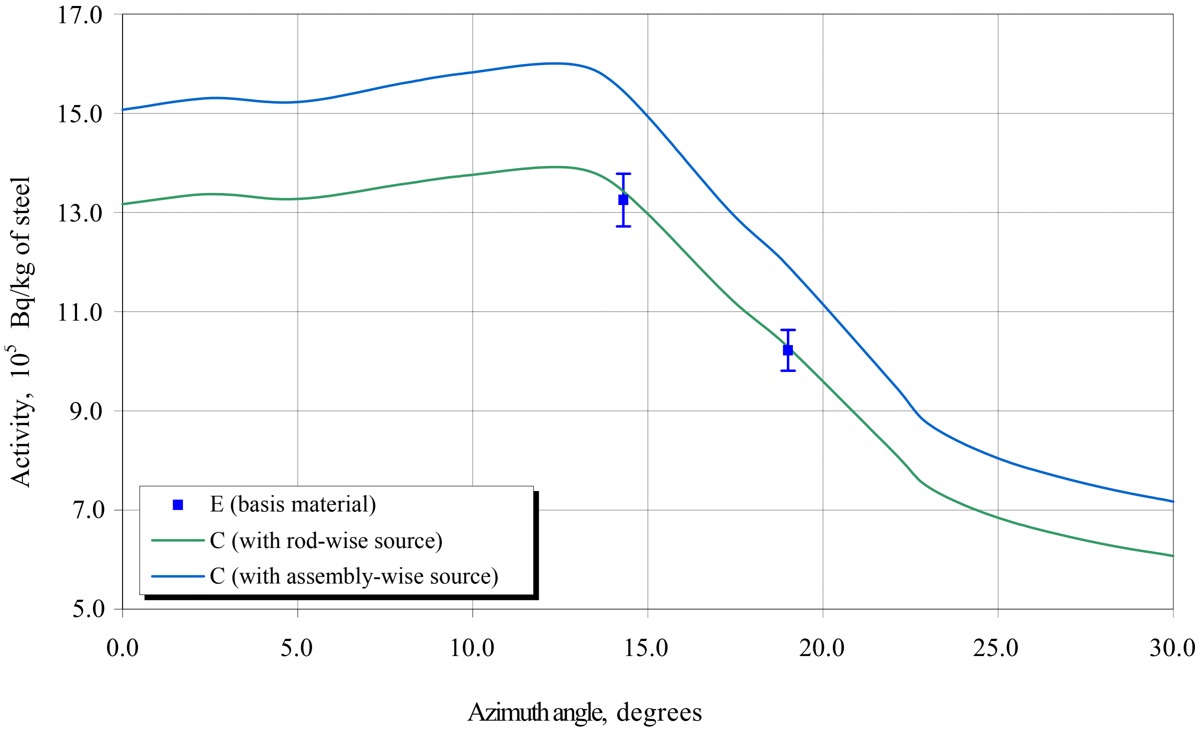

Fig.5. Rated (R) and experimental (E) activity distributions for ^{54}Mn in templates of basic metal (z=115 cm) of reactor vessel of Kola Nuclear Power Plant power unit No.1. Experimental data obtained from National Research Centre “Kurchatov Institute”. Measurement error of templates activity is $2\sigma_{exp} = \pm 4\%$.

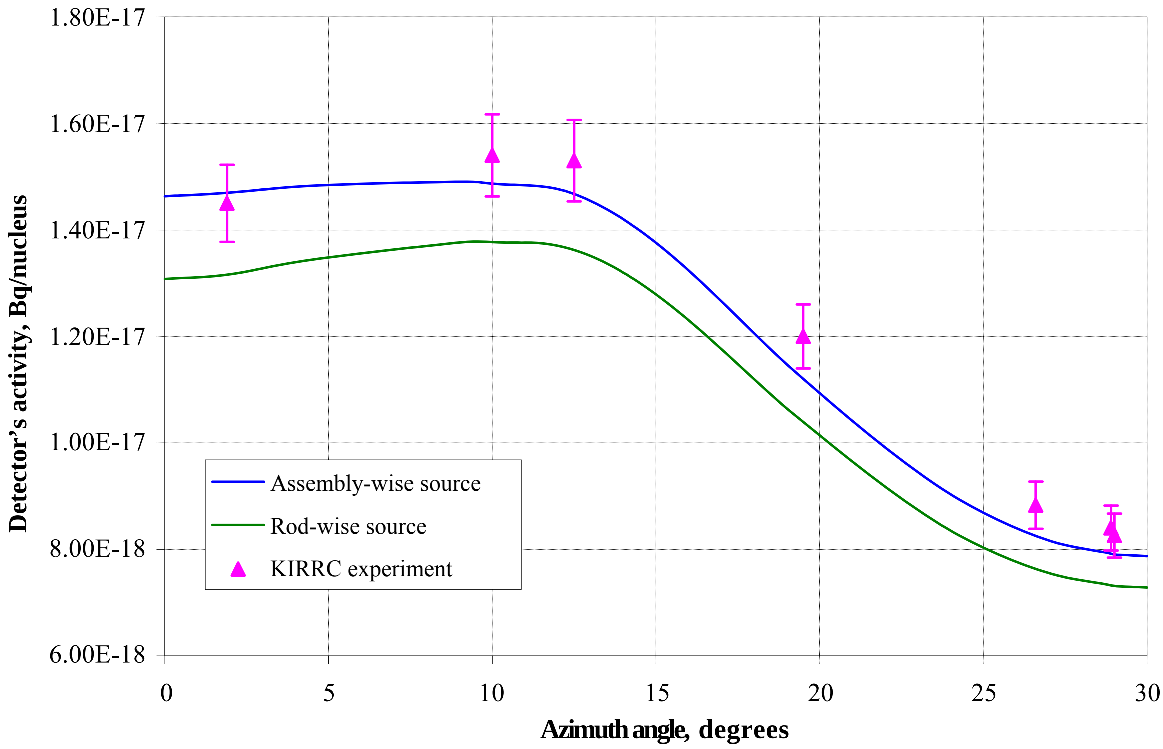

Fig.6. Rated (R) and experimental (E) axial activity distributions for ^{93}Nb for z=30cm on the external reactor vessel surface of Kola Nuclear Power Plant power unit No.1. Experimental data obtained from National Research Centre "Kurchatov Institute". Measurement error of activity is ^{93}Nb $2\sigma_{exp} = \pm 4.5\%$.

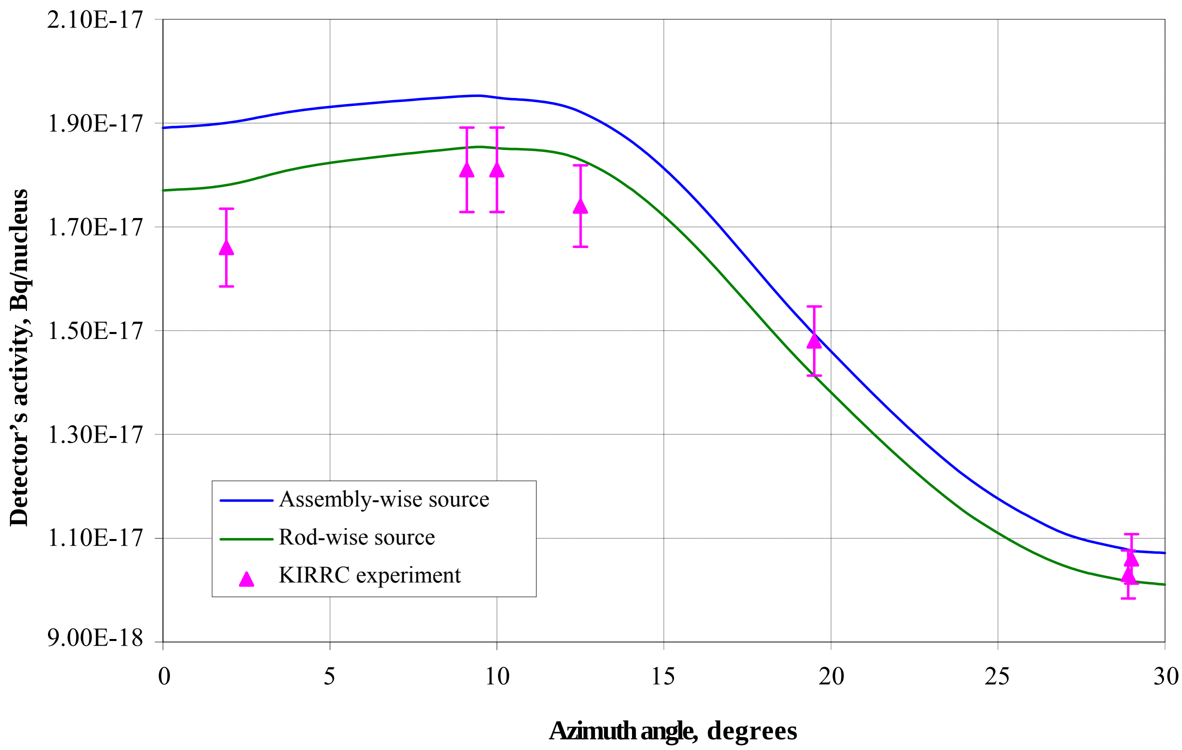

Fig.7. Rated (R) and experimental (E) axial activity distributions for ^{93}Nb for z=115cm the external reactor vessel surface of Kola Nuclear Power Plant power unit No.1. Experimental data were received from National Research Centre "Kurchatov Institute". Measurement error of activity makes up ^{93}Nb $2\sigma_{exp} = \pm 4.5\%$.

4. Conclusions

Comparing the results of calculations using different ways of source setting and experimental data allowed make the following conclusions:

- the difference between the values of fast neutron flux on the internal surface of the vessel with assembly-wise vs. rod-wise source is about 15 %;
- the deviation of calculated from observed experimental values for rod-wise source distributions is +2.9 % on average for joint No. №4, and +1 % on average for basis templates metal; thus, the corresponding deviations for assembly-wise distribution are +20.4 % and +16.6 % on average;
- for out-of-vessel detectors of Niobium, the deviation of calculated values from experiment for rod-wise source distribution makes up -11.7 % on the average for joint No. 4 and +1 % on the average for the basis metal; thus, these deviations are -4.2 % and +5.9 % on average for assembly-wise source;

Thus, we can draw a preliminary conclusion based on the calculated vs. experimental templates' activity data comparison that the use of assembly-wise distribution of source on the reactor core periphery provides a conservative estimate for the value of fast neutron flux on the internal surface of the reactor vessel of up to 16 % on the joint No. 4 and up to 12% on the basis metal. The difference between the calculation and the experiment on the internal surface when using rod-wise source method is within the range of experiment error at 2 σ (±4 %). Therefore safety factor of at least 4 % should be used on the internal surface of reactor vessel, in order to observe a principle of conservatism (according to RB-007-99) on rated values of flux.

On the external reactor surface the comparison of calculated vs. observed experimental data leads to a different conclusion. Safety coefficients additional to those produced by KATRIN software should be applied for the external surface of reactor vessel for both rod-wise and assembly-wise source definition: for assembly-wise distribution we recommend adding at least 9 % for the joint No.4; for rod-wise distribution we recommend adding at least 16 % for the joint weld No.№4 and at least 4 % for the basis metal.

Further comparisons with experimental data are recommended to challenge, and provide support to the conclusions made here.

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