Calculation and Experimental Data Analysis of Neutron Spatial-Energy Distribution in the BOR-60 Blanket

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Abstract. At present, a wide range of tests is performed in the BOR-60 reactor in support of reactors under operation, construction and design in Russia and worldwide. Most of the tests are performed in the reactor core regions of the peak dose accumulation rates. However, there is a high demand for irradiation testing to be performed in the BOR-60 blanket.

An important feature of any nuclear facility is neutron spatial-energy distribution in the reactor. An experimental data analysis of neutron spectra takes much effort and time. The effective volume of an irradiation rig is rather limited, which makes it difficult to install dozens of neutron activation detectors at the expense of tested samples. Therefore, irradiation parameters are confirmed experimentally using several detectors; and spatial-energy distribution of the neutron field is obtained in calculation.

RIAR's experience in thorough calculations and experiments in support of BOR-60 operation shows good agreement between the calculated and experimental core parameters. The deviation of the calculated values from the experimental ones in the blanket is higher, and there are much less experimental data. Therefore, verification of the applied calculation codes, models and methods is a crucial relevant issue.

Key Words: reactor BOR-60, blanket, neutron flux & spectrum.

Introduction

The spatial and energy distribution of neutrons in the reactor is of critical importance at any nuclear facility. A wide range of irradiation tests of different materials and fuel compositions is carried out in the BOR-60 research reactor [1]. The experimentally obtained data related to the neutron spectra and flux in the research reactor is necessary:

- to verify and specify the applied calculation codes, procedures, models and constants.
- to enhance the validity of irradiation tests with the use of different irradiation rigs.

However, to determine experimentally the neutron spectra is an expensive long-term complicated process that includes in-pile testing itself and further measurement of dozens of activation detectors. Based on the measurement data, the neutron spectrum and flux in the irradiation position are reproduced in calculation and experiment. The volume inside the irradiation rig is rather limited, and it is quite difficult to install the whole set of activation detectors. Therefore, to verify experimentally the irradiation parameters, several detectors are installed to obtain some reference data. As for the spatial and energy distribution of neutrons, it is calculated.

At present, BOR-60 is intensively utilized to perform in-pile tests using different irradiation rigs for a wide range of research to show the feasibility of both foreign and Russian reactors under operation, construction and design. Most of these tests are carried out in the reactor core regions with the maximum damage dose accumulation and heat rates. More recently, the interest grows to conduct irradiation in the BOR-60 blanket. This can be explained by the fact that the core is overloaded with different experimental programs, and there are a relatively large number of irradiation positions in the blanket the characteristics of which are sufficient to achieve the necessary irradiation parameters. However, the calculation in support of irradiation testing conditions is not entirely satisfactory.

Long-standing experience in calculation and experiments in support of BOR-60 operation as well as experimental research conducted in this reactor show good agreement between the calculated and experimental parameters for the rigs irradiated in the core. The number of the tests performed in the BOR-60 blanket is not quite big, and during the recent 10-15 years there have been no such tests performed in the reactor blanket. During these years the codes, nuclear data files, calculation models and procedures have been changed. Therefore, it becomes relevant to verify the used calculation means.

Due to the above reasons, we will use the earlier experience to verify the procedures, codes, models and nuclear constants applied in calculation in support of BOR-60 operation and experimental research carried out in this reactor [2, 3, 4].

1. Applied software, constants and models

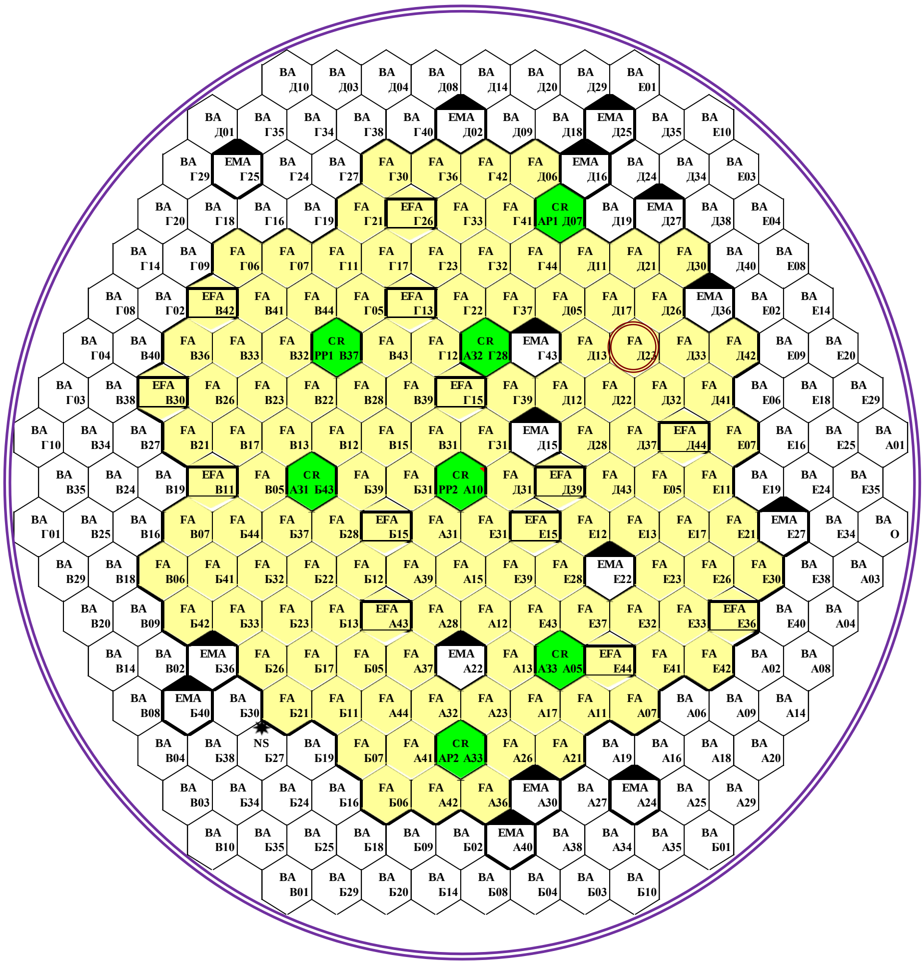
Since 1990s the BOR-60 neutronic parameters have been calculated using software TRIGEX [5] and KAR [6]. Over the long-term reactor operation different TRIGEX versions and constant data files have been applied, and modifications have been introduced into KAR and BOR-60 calculation models.

KAR is automated software to calculate the BOR-60 neutronics. It is used to create BOR-60 calculation models taking into account the actual arrangement of the assemblies in the reactor, composition of the nuclear fuel, absorber and structural materials of all assemblies and control rods. KAR enables analysis and processing the BOR-60 neutronics, detailed study of the neutronic parameters, and simulation of different irradiation modes for specific assemblies and fuel elements.

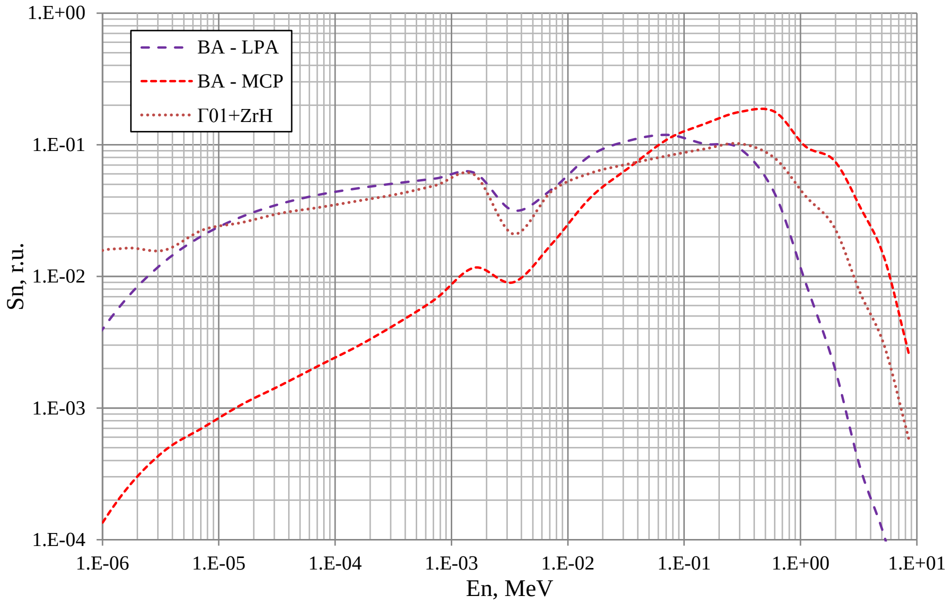
TRIGEX has been long used for calculation in support of BOR-60 operation and experimental research. TRIGEX is intended to calculate fast reactor neutronics in three-dimensional hexagonal geometry in multigroup (26 groups) diffusion approximation based on the BNAB-93 (nuclear data files) and CONSYST-2 (system of constants). The TRIGEX software was verified and adapted in the BOR-60 reactor. It was attested for calculation in the BN-600 and BN-800 reactors. The comparison between the TRIGEX-based calculation data and experimental and calculation data obtained by applying other codes (MCU, MCNP, JARFR, NF-6) showed their good agreement in the core (within the uncertainty range) and somewhat worse agreement in the blanket, which can be also explained by a rather unsatisfactory description of the blanket in the calculation model.

The model used in calculation in support of BOR-60 operation evolves as calculation tools and software develop (TRIGEX). Usually in the model of the core and breeding regions of FA used axial partitions 5 cm, to take into consideration the axial profile of changes in the nuclear fuel composition, i.e. the physical zones (prisms) were set with unique material compositions. Other components of the FAs, experimental rigs and blanket assemblies were set in a rather approximate way: the axial size of the hexagonal prisms could be considerably larger than the one used in the FA operating (“active”) part. It was admissible to set the assembly components by the averaged physical zones with no reference to the irradiation conditions.

Despite the stable parameters of BOR-60, the given approximations lead to some errors in obtaining few-group constants. Considering the minor differences in the core and blanket arrangement, these approximations do not lead to considerable deviations. However, the modern reactor core and blanket are represented by greatly heterogeneous regions. *FIG.1* shows BOR-60 core arrangement. The reactor is loaded with assemblies containing different fuels and absorber, moderator and structural materials. *FIG.2* shows neutron spectra in different parts of the blanket assemblies. The given data show that the neutron spectra differ considerably. Thus, the percentage of neutrons (E >0.1 MeV) for the given spectra ranges 30 %–70 %, and the average neutron energy ranges from 1 keV to 125 keV.



CR – control rod; FA – fuel assembly; EFA – experimental fuel assembly; EMA – experimental materials assembly; BA – steel blanket assembly; NS – neutron source
FIG. 1. BOR-60 core arrangement.



BA - LPA – the bottom part of the ninth-row assembly; BA-MPC – the core mid-plane level, the sixth-row blanket assembly; G01 (Russian: Г01)+ZrH – moderator-surrounded cell
FIG. 2. Neutron spectra in gas plenums of different FAs.

The comparison of the calculation data obtained by using the TRIGEX and MCU and experimental data for the rigs irradiated at the core-blanket boundary shows that the used simplification in the blanket also leads to some deviations.

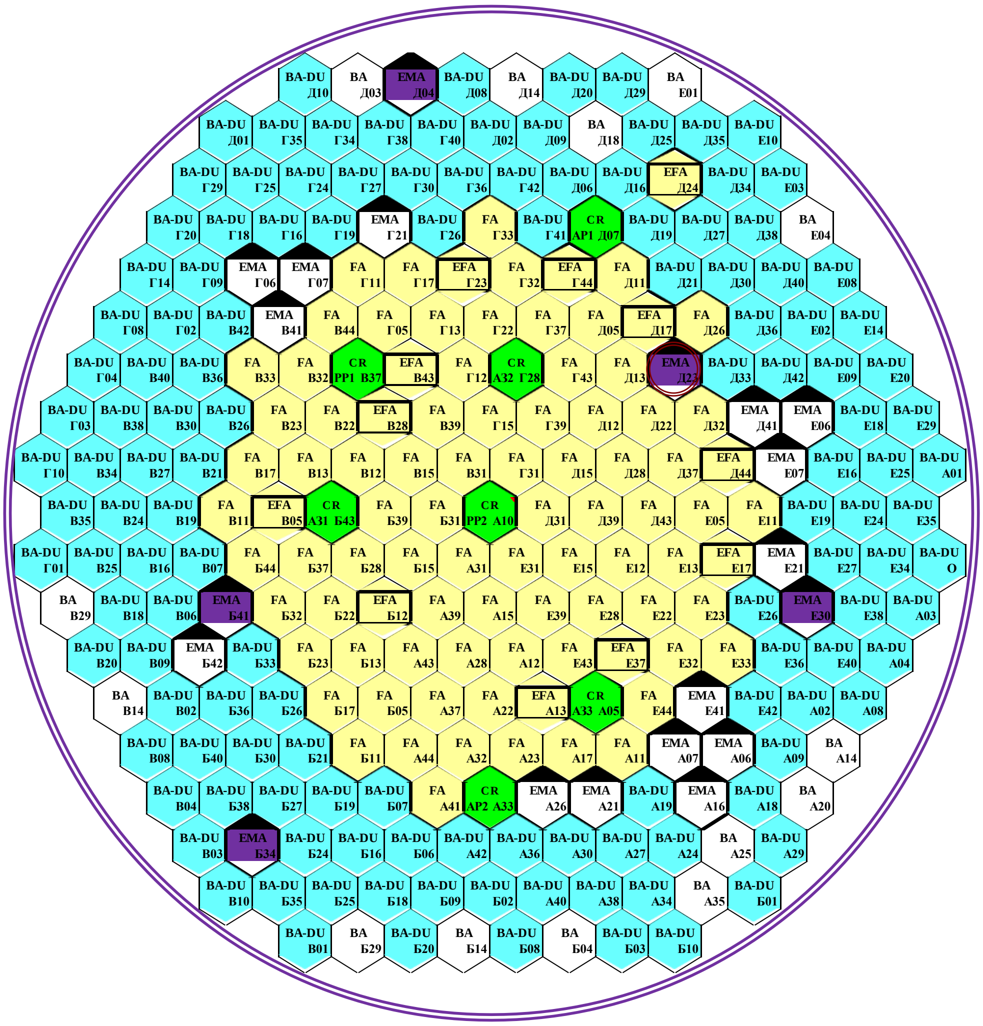
Thus, to improve calculation and experiments in support of BOR-60 operation, some changes have been introduced into the calculation model providing for a more correct and full use of the TRIGEX features, and more detailed elaboration in describing BOR-60. In this paper, are used the new calculation model of BOR-60 with a more detailed axial description of the fuel assemblies, control rods, blanket assemblies and experimental rigs.

As a result of the model update, the axial dimensions of the hexagonal prisms (with unique material compositions and temperature) in the examined region make up 5 cm, which is comparable to the reactor cell size (4.5 cm).

To verify the calculation model, the neutron spectra from the experiment to determine neutron fields in the BOR-60 blanket cells were compared with the calculation data.

2. Experiment

The experiment in BOR-60 was carried out in February 1991 prior to the micro-run start. During the experiment, the reactor core comprised 79 standard FAs and 11 EFAs, and the blanket comprised 137 assemblies containing depleted uranium and 14 steel assemblies. *FIG.3* presents the BOR-60 core arrangement with the specified cells where the detectors were irradiated (Table 1).



CR – control rod; FA – fuel assembly; EFA – experimental FA; EMA – experimental materials assembly; BA – steel blanket assembly; BA-DU – blanket assembly with depleted uranium.
FIG. 3. BOR-60 core arrangement during the experiment.

TABLE I – PARAMETERS OF THE CELLS SELECTED FOR THE EXPERIMENT

Cell	Row	R, cm *
D23 (Д23)	5	19.6
B41 (Б41)	6	23.8
E30 (E30)	7	28.1
B34 (Б34)	8	34.0
D04 (Д04)	9	35.7
* R – distance from the core center to the cell center		

The in-pile test lasted for 2.5 hours, during which BOR-60 was brought to a thermal power of ~ 1.4 MW.

3. Compared calculation and experimental data

FIG.4 presents the data related to calculation of the neutron flux density in the examined cells of BOR-60 (*the data are normalized for 60 MW*).

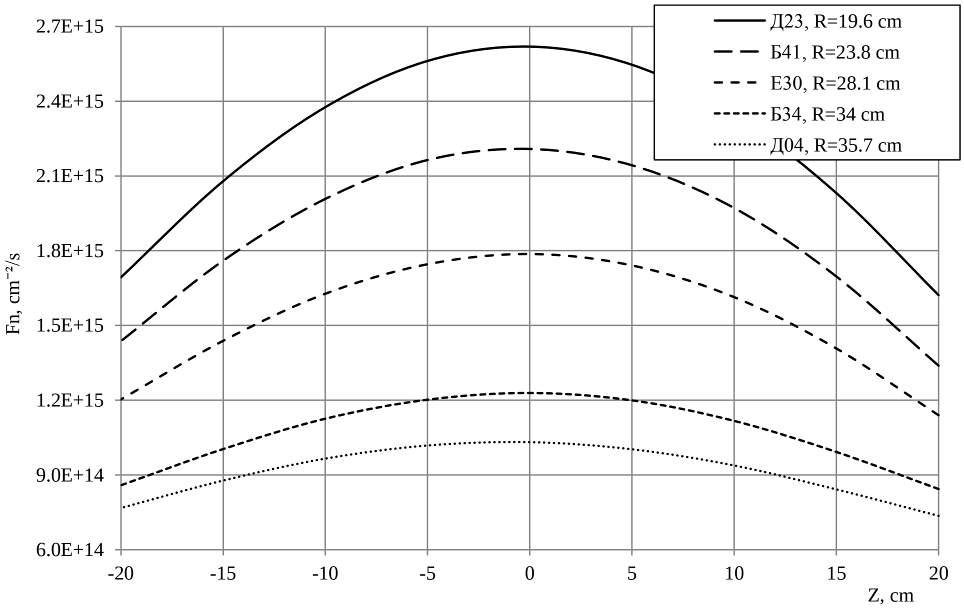
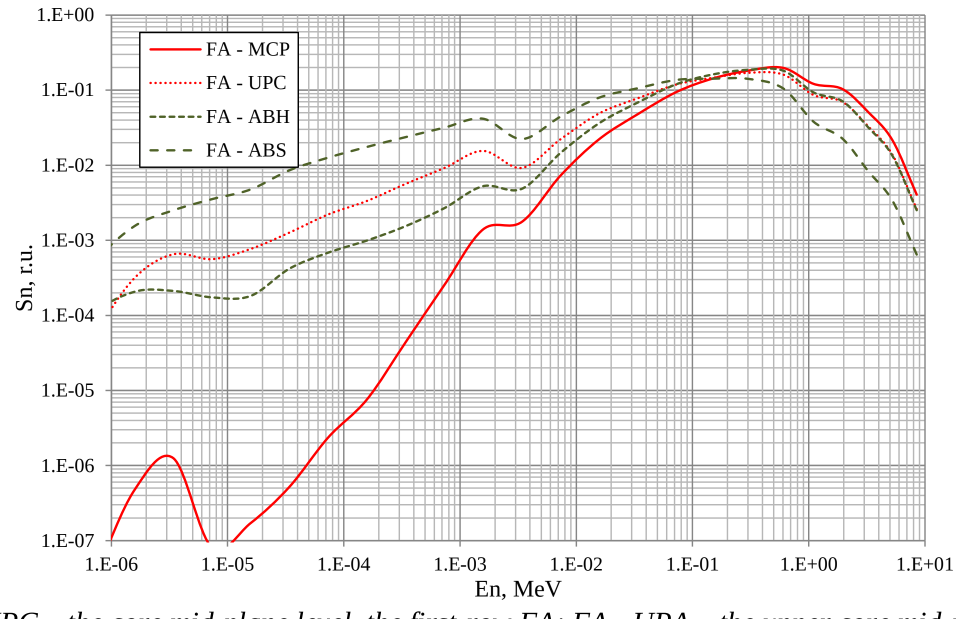


FIG. 4. Axial distribution of the neutron flux density in the examined cells of BOR-60.

FIG.5 presents calculated neutron spectra in different parts of the core FAs. As shown in the Figure, there is a rather “hard” neutron spectrum in the FA active part. As for the percentage of neutrons (E>0.1 MeV), it ranges 0.7– 0.8. The average neutron energy ranges 110 keV–360 keV.



FA - MPC – the core mid-plane level, the first-row FA; FA - UPA – the upper core mid-plane, the seventh-row FA; FA – ABH – the breeding region of the first-row FA; FA – ABS – the breeding region of the seventh-row FA.

FIG. 5. Calculated neutron spectrum of different FAs.

FIG.6-10 show the results of comparison between the calculation and experimental data. The data show that the differences between the calculated and experimental neutron spectra are observed at energies lower than 100 keV. First and foremost, this can be explained by the fact that when processing the experimental data, the fine structure of the sections of certain nuclides (Na, Mn, Fe) was not taken into account, and there is no corresponding strain in the experimental spectra. In the high-energy spectrum region containing over 70% neutrons, there is a very good agreement between the calculation and experimental data. Therefore, the calculation prediction of parameters, such as fast neutron fluence and damage dose in steel, will almost coincide with the experimental data. The average neutron flux energy calculated in the given spectra differs by no more than 5%.

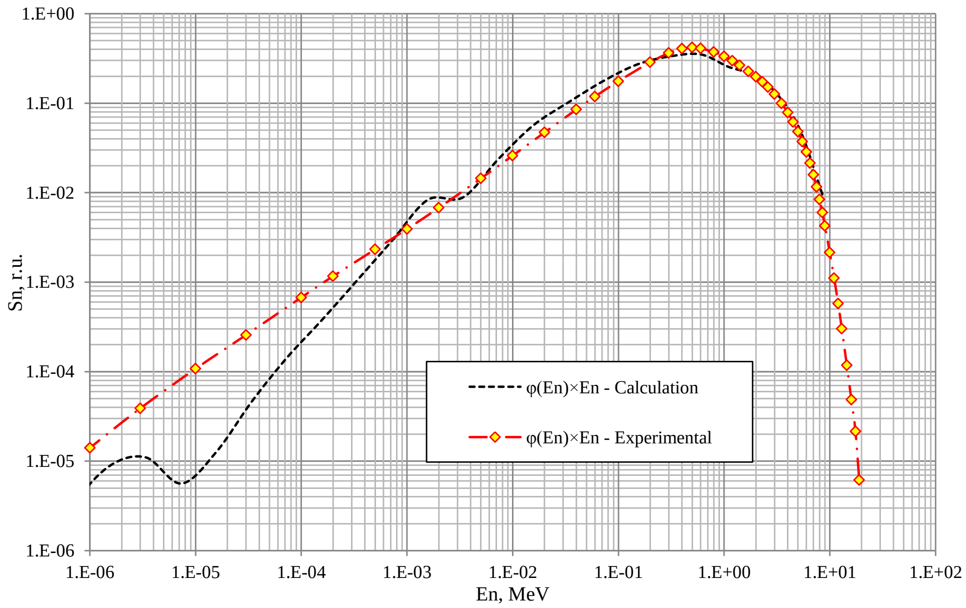


FIG. 6. Neutron spectrum in cell D23 (Д23).

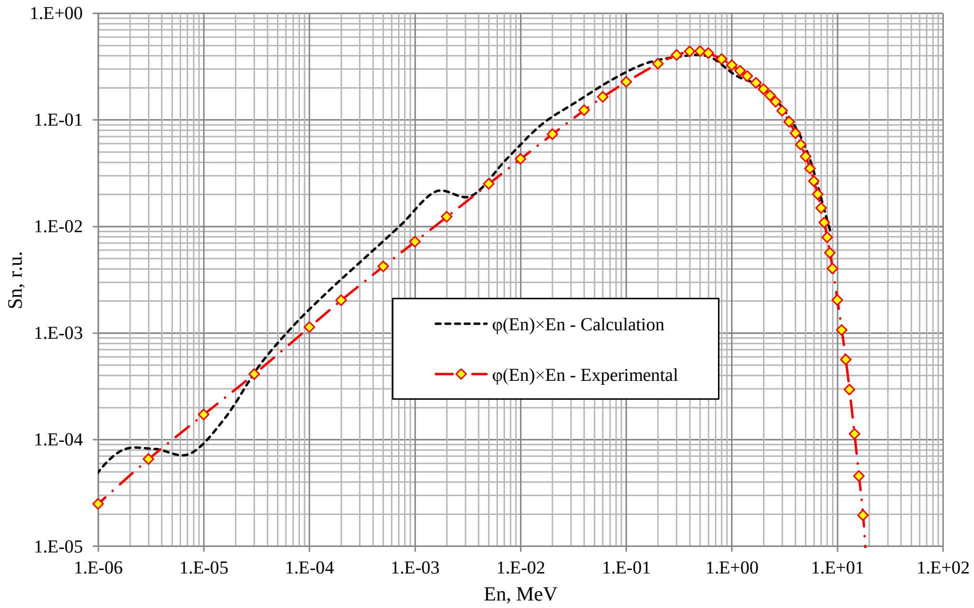


FIG. 7. Neutron spectrum in cell B41 (Б41).

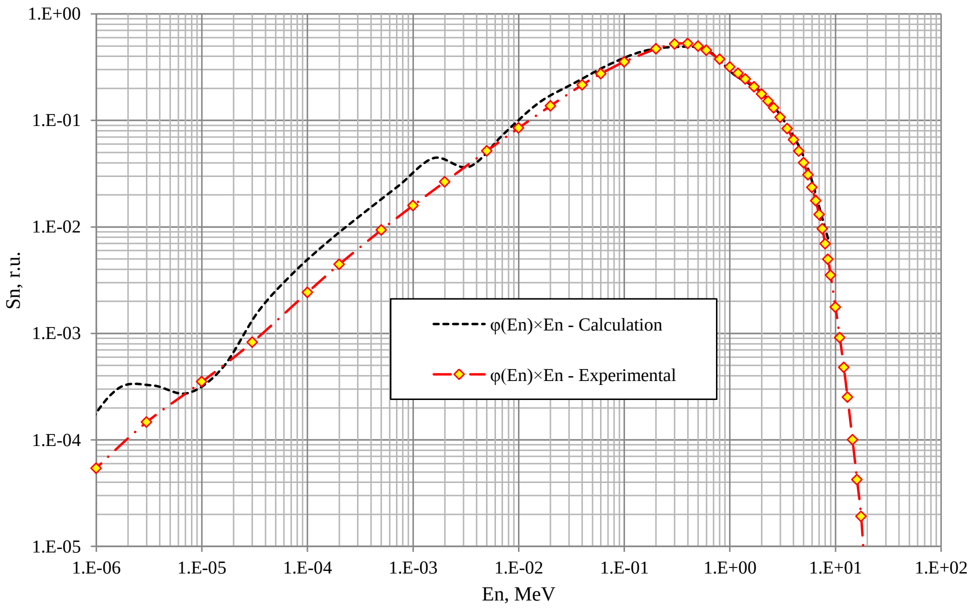


FIG. 8. Neutron spectrum in cell E30.

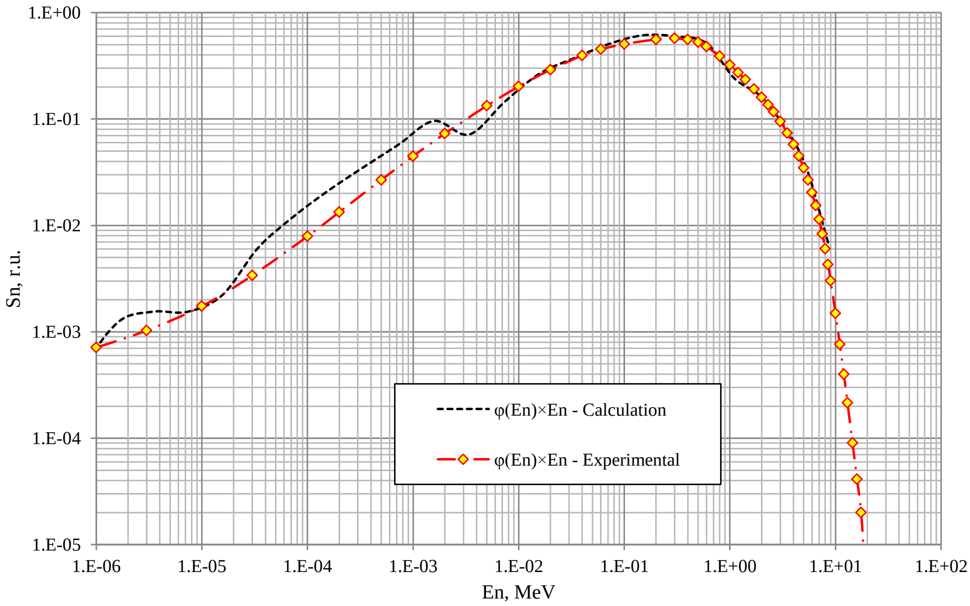


FIG. 9. Neutron spectrum in cell B34 (Б34).

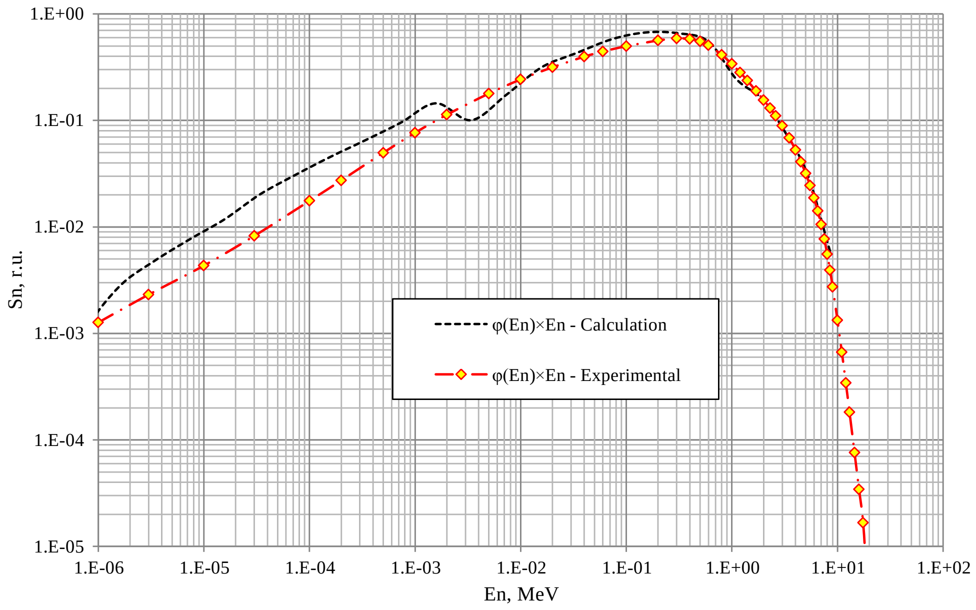


FIG. 10. Neutron spectrum in cell D04 (Д04).

In conclusion, let us compare the calculation and experimental neutron flux density normalized for a thermal power of 60 MW. *FIG.11* shows compared radial distribution of the neutron flux density obtained in calculation and experiment.

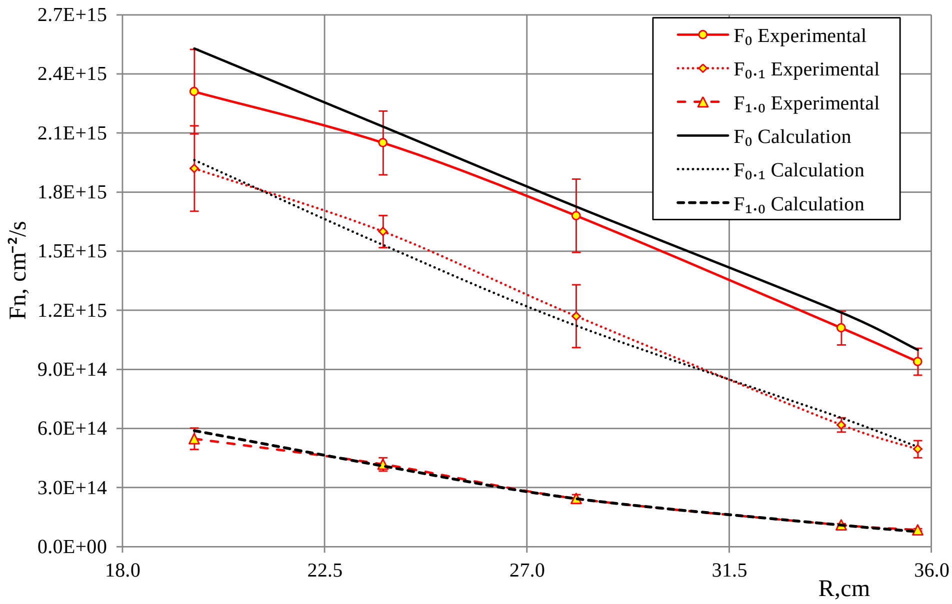


FIG. 11. Radial distribution of the integral neutron flux density and neutron flux density (E>0.1 MeV and E>1.0 MeV) in the core mid-plane.

The given data show that the calculation and experimental data agree well, and the difference between them does not exceed the experimental uncertainty in determining the neutron flux density (5÷10%).

To sum up, we can conclude that the improvement of the standard calculation model resulted in a better agreement between the calculation and experimental data in the BOR-60 blanket.

Therefore, prediction of irradiation conditions of the experimental rigs will be improved at the core boundary and blanket bulk.

Conclusion

The conditions of the experiment to determine the neutron spectra in the BOR-60 blanket were analyzed and reproduced. The obtained data can be used to verify the codes, models and procedures applied in support of BOR-60 operation and experimental research.

The spatial-energy distribution of neutrons in the BOR-60 blanket was determined in the experiment and calculation showing that the calculation and experimental data agree within the experimental uncertainty.

The latest version of the BOR-60 calculation model for TRIGEX enables rather precise prediction of the BOR-60 neutronics in the blanket, which will favorably affect the representativeness of the calculation and experiments in support of conducted experimental research.

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