

# Precision neutronic calculations of experiments on the neutron transmission through the reflector layers at the BFS critical facilities for expanding the verification database to justify lead cooled fast reactor designs\*

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## Abstract

The paper presents the results of the efforts concerned with expanding the verification database and estimating the calculation uncertainty of the power density in the steel reflector of lead cooled fast reactor designs based on experiments performed in different years at the BFS critical assemblies by analyzing and revising earlier calculation and experimental studies on the transmission of neutrons through the steel reflector layers. The discussion includes experiments at the BFS-66 critical assembly to model neutron and photon fluxes in the reactor core shielding compositions, as well as experiments at the BFS-64 and BFS-80-2 critical assemblies to model the transmission of neutrons and gamma quanta through the reflector layers of various materials. The information provided in earlier materials with the descriptions of the above experiments has been analyzed and expanded through respective data required to prepare precision calculation models for Monte-Carlo neutronic codes. Precision neutronic models have been developed based on actualized and updated data with a detailed description of the BFS heterogeneous structure and experimental devices, and test calculations have been carried out to confirm their efficiency. The calculations of key neutronic characteristics measured at the BFS-66, -64 and -80-2 assemblies were performed using codes based on the Monte Carlo method (MCU-BR, MCNP, MMK-RF, MMK-ROKOKO) with BNAB-RF and MDBBR50 neutron data and the ROSFOND evaluated neutron data library. The developed precision calculation neutronic models of the experiments discussed can be used to justify lead cooled fast reactor designs, to verify neutronic codes and neutron data, and to evaluate the associated uncertainties.

## Keywords

Integral experiments, BFS, precision neutronic models, BNAB-RF, ROSFOND, MCU-BR, MMK-RF, MMK-ROKOKO

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## Introduction

Worldwide, at the present time, there is a common practice of using actively the results of integral and reactor experiments at critical zero-power test facilities to justify reactor plant designs of different types, as well as to assess and improve the accuracy of predicting the performance of in-core and ex-core systems under design (DICE, ICS-BEP, IRPhEP). Carried out with a high quality and evaluated, experiments form the basis for handbooks and banks of recommended evaluated experimental neutronic data which are used extensively to design new experiments, to estimate the efficiency and informational value of new experimental programs, to verify and certify neutron data and codes (RB-061-11), etc.

Properly conducted experiments may acquire the status of a benchmark experiment which shall be understood to mean an experiment of a reference class with as small evaluated uncertainties as possible. The respective benchmark model of an experiment is the most efficient way to record, store and transfer information and data on the neutronic experiment required for verifying and certifying codes and neutron data, improve the accuracy of calculation procedures and algorithms, estimate the accuracy of prediction, determine the target uncertainties of the neutronic performance for the reactor plants under design, and justify particular technical solutions.

In 1990–2013, seven series of experimental programs were performed at BFS-1 and BFS-2 physical test facilities at IPPE using critical configurations with lead in the core or in the reflector (BFS-61, -64, -77, -85, -87, -95, and -113) designed to investigate the neutronic performance of lead cooled fast reactor cores. A calculation and experimental analysis of the experiment data was performed in different years at IPPE for JSC NIKIET, NRC Kurchatov Institute, JSC OKBM Afrikantov, OKB GIDRORESS, ISTC, and NEA/OECD (Dulin et al. 2014).

In the period of 2017–2019, work was initiated as part of the PRORYV Project to develop a system of recommended experimental neutronic data for justifying lead cooled fast neutron reactor designs. The basis for this system is experimental information obtained at the BFS critical facilities. Respective experiments were performed in different years at IPPE to justify sodium and lead cooled fast reactor designs. The database includes evaluations of the experiments conducted at the BFS-61, -64, -77, -85, -87, -95, and -113 critical assemblies.

To expand the verification database for estimating the uncertainty of the power density in the steel reflector of a lead cooled fast reactor plant, earlier calculation and experimental studies on the transmission of neutrons through the steel reflector layers, performed at the BFS-66, -64, and -80 critical assemblies and being outside the scope of the above database of the evaluated PRORYV neutronic experiments, have been analyzed and revised.

## Brief information on the considered BFS experimental programs

The discussion includes experiments at the BFS-66 critical assembly to model neutron and photon fluxes in the reactor core protective compositions, as well as experiments at the BFS-64 and BFS-80-2 critical assemblies to model the neutron and gamma quanta transmission through the reactor core reflector layers of different materials.

The series of the BFS-64 critical assemblies represents a sector model of a fast reactor core with oxide uranium-plutonium fuel and lead coolant and is a modification of the BFS-62 critical assembly (ICS-BEP). The experimental program was aimed at studying the core periphery and reflector properties and identifying the power density field excitation by the control rods as the reactor's inherent non-uniformities. The neutron and gamma quanta spectra were also measured by the proton recoil method. The distributions of the relative  $^{239}\text{Pu}$  and  $^{238}\text{U}$  isotope fission rates were measured: a) along the height of the control rods on the core periphery, and b) along the core radius in the central plane and in the plane passing through the control rods.

The BFS-66-1 assembly, a modification of the BFS-62 assembly, is a model of the BN-600 reactor core with MOX fuel. The assembly of the BFS-66-B series is a model of the BN-800 reactor core with nitride fuel. The BFS-66-B assembly has nine different critical configurations (BFS-66-B1 through B9) which differ from each other in the composition of the axial reflector (three configurations with different uranium reflectors, three steel configurations of different compositions, as well as chromium, nickel and zirconium reflectors).

Axial distributions of the  $^{239}\text{Pu}$ ,  $^{235}\text{U}$  and  $^{238}\text{U}$  fission rates in the tube space near the central channel were measured using the assembly. The purpose of the measurement program is to explore the neutron fields in the non-breeding blankets of fast reactors.

Two modifications of the BFS-80-1 and BFS-80-2 assemblies are models of a lead-bismuth cooled fast reactor core as of the beginning and as of the end of the core life respectively. Radial and axial distributions of the  $^{235}\text{U}$  and  $^{238}\text{U}$  fission rates and neutron spectra were measured using the assemblies by the proton recoil method.

Table 1 presents brief information on the BFS-66, -64 and -80 measurement programs and the performance of the core layouts.

It was found as part of considering in detail earlier studies for analyzing the BFS-66, -64 and -80 experiments that computational modeling used, in most cases, homogeneous models with calculated adjustment factors introduced into the experimental values to lessen the calculation and experiment discrepancies, with some of the experimental measurement values for the spatial distributions of the fissionable isotope fission rates left beyond the analysis scope; combined, these circumstances led to experimental information being lost and distorted.

**Table 1.** Brief description of the BFS-66, -64 and -80 measurement programs.

Assembly	Materials	Application	Measurements
BFS-64	Base	Lead cooled fast reactor sector model with NF simulation	$k_{\text{eff}}$ neutron spectrum and $\gamma$ -quanta
	No CRs		$k_{\text{eff}}$ $\text{H}^{(239)\text{Pu}}$ core and RR, $\text{R}^{(239)\text{Pu}}$
	CRs-1		$k_{\text{eff}}$ $\text{H}^{(239)\text{Pu}}$ and $^{238}\text{U}$ core and RR, $\text{R}^{(239)\text{Pu}}$
	CRs-2		$k_{\text{eff}}$ $\text{H}^{(238)\text{U}}$ RR
BFS-66	66-1, base	Simulation of BN-800 core with full MOX fuel load	$k_{\text{eff}}$ $\text{H}^{(239)\text{Pu}}$ , $^{235}\text{U}$ and $^{238}\text{U}$ , $\text{R}^{(239)\text{Pu}}$ , $^{235}\text{U}$ and $^{238}\text{U}$
	66-B1	Simulation of BN-800 core with NF	$k_{\text{eff}}$ $\text{H}^{(239)\text{Pu}}$ , $^{235}\text{U}$ and $^{238}\text{U}$
	66-B2		
	66-B3		
	66-B4		
	66-B5		
	66-B6		
	66-B7		
	66-B8		
	66-B9		
BFS-80	80-1	Lead-bismuth cooled fast reactor	$k_{\text{eff}}$ $\text{H}^{(235)\text{U}}$ and $^{238}\text{U}$ core and RR, $\text{R}^{(235)\text{U}}$ and $^{238}\text{U}$
	80-2		$k_{\text{eff}}$ neutron and gamma quanta spectrum, $\text{H}^{(235)\text{U}}$ and $^{238}\text{U}$ core and RR, $\text{R}^{(235)\text{U}}$ and $^{238}\text{U}$

**Note:** CR – control rod; FC – fuel cell; CL – coolant; AR – axial reflector; NF – nitride fuel; RR – radial reflector;  $\text{H}^{(239)\text{Pu}}$ ,  $^{235}\text{U}$  and  $^{238}\text{U}$ ,  $\text{R}^{(239)\text{Pu}}$ ,  $^{235}\text{U}$  and  $^{238}\text{U}$  – axial and radial distributions of  $^{239}\text{Pu}$ ,  $^{235}\text{U}$  and  $^{238}\text{U}$  respectively

Information presented in earlier materials with descriptions of the above experiments has been analyzed and expanded by adding respective data required to prepare detailed calculation models for precision neutronic codes. All experimental and calculated data (the latter based on updated models) for the experiments at the BFS-66, -64 and -80 assemblies is presented in accordance with the requirements and the format of the neutronic experiment evaluation and expert review regulations. The experience of using this document has demonstrated its contribution to improving the presentation quality of experimental and calculated data (primarily, by making it possible to elaborate, based on the provided description, an adequate calculation measurement model which could not be done earlier relying only on data from primary information and data sources), as well as the extent to which the experiment estimates can be alienated.

## Calculation results

In conditions of critical assembly cores being heavily heterogeneous, a need arises for using codes which allow reproducing an accurate description of the geometry and presenting in detail the energy dependence of the neutron interaction characteristics. This makes it possible to produce precision codes based on the Monte Carlo method.

Actualized and updated data on the BFS-66, -64 and -80 assemblies were used as the basis for building detailed precision neutronic models with a description of the BFS heterogeneous structure and experimental devices for Monte Carlo based codes, including MCU-BR, MCNP, MMK-RF, and MMK-ROKOKO (Gomin et al. 2000, Andrianova et al. 2016, Zherdev et al. 2018) with

BNAB-RF and MDBBR50 neutron data, and the ROSFOND evaluated neutron data library (Andrianova et al. 2018), which allow calculating key neutronic characteristics measured at the BFS-66, -64 and -80 assemblies, and being important in terms of estimating the uncertainty of the power density in the steel reflector of a lead cooled fast reactor plant.

For all critical assembly configurations that have been considered,  $k_{\text{eff}}$  was calculated for the ROSFOND and BNAB-RF neutron data (Table 2). The difference in the results of the  $k_{\text{eff}}$  calculations performed based on a group approximation (BNAB-RF) and the ROSFOND point-wise nuclear data does not exceed 0.2%; the discrepancy between the calculation and the experiment does not exceed 0.5%  $\Delta k/k$  and lies within the experimental error limits.

**Table 2.** Calculated and experimental criticality values for the BFS-64, -66 and -80 assemblies.

Configuration	$k_{\text{eff}}^{\text{E-exp.}}$	$k_{\text{eff}}^{\text{C}_1\text{-calc.}}$	$k_{\text{eff}}^{\text{C}_2\text{-calc.}}$	$k_{\text{eff}}^{\text{C}_3\text{-calc.}}$
Number of groups	–	Det. path	2224	1324
BFS-64-1, base	1.00114(280)	0.99824(12)	1.00092(12)	0.99815(13)
BFS-64-1 (Pb)	1.00067(280)	0.99768(10)	1.00020(13)	0.99729(12)
BFS-64-1 (CRs-1)	1.00046(280)	0.99807(11)	1.00011(12)	0.99714(13)
BFS-64-1 (CRs-2)	1.00076(280)	0.99815(10)	1.00040(12)	0.99724(12)
BFS-66-1	1.00070(300)	0.99878(10)	0.99974(12)	0.99899(12)
BFS-66-B1	1.00052(300)	0.99548(10)	0.99860(12)	0.99844(12)
BFS-66-B2	1.00088(300)	0.99698(10)	0.99885(13)	0.99991(11)
BFS-66-B3	1.00070(300)	0.99591(11)	0.99805(12)	0.99793(12)
BFS-66-B4	1.00058(300)	0.99674(10)	0.99955(12)	0.99939(12)
BFS-66-B5	1.00033(300)	0.99627(11)	0.99920(12)	0.99994(12)
BFS-66-B6	1.00076(300)	0.99548(10)	0.99858(12)	0.99814(12)
BFS-66-B7	1.00053(300)	0.99652(10)	0.99903(13)	0.99972(13)
BFS-66-B8	1.00060(300)	0.99562(10)	0.99817(12)	0.99832(12)
BFS-66-B9	1.00042(300)	0.99605(10)	0.99850(12)	0.99922(13)
BFS-80-2	1.00045(450)	1.00502(12)	1.005975(12)	0.99974(12)

**Note:**  $k_{\text{eff}}^{\text{C}_1}$  – MMK-RF calculation with the ROSFOND library;  
 $k_{\text{eff}}^{\text{C}_2}$  – MMK-RF calculation using CONSYST-RF with the BNAB-RF neutron data;  
 $k_{\text{eff}}^{\text{C}_3}$  – MMK-RF calculation using CONSYST-RF with the BNAB-93 neutron data.

We shall give examples of calculations for the axial and radial distributions of the BFS relative fission rates performed using the MMK-RF (MMK-ROKOKO) code with the ROSFOND and BNAB-RF neutron data. All axial and radial distribution measurements used small-size fission chambers which were installed with the use of a manipulator into the tube space at different core and reflector axial and radial positions.

Fig. 1 shows the power density field ( $^{235}\text{U}$  fission rate distribution) measurement and calculation results that reflect a non-uniform arrangement of materials in the BFS-64-1 core: Fig. 1a shows a case with a uniform arrangement of materials, and Fig. 1b shows a case with control rods.

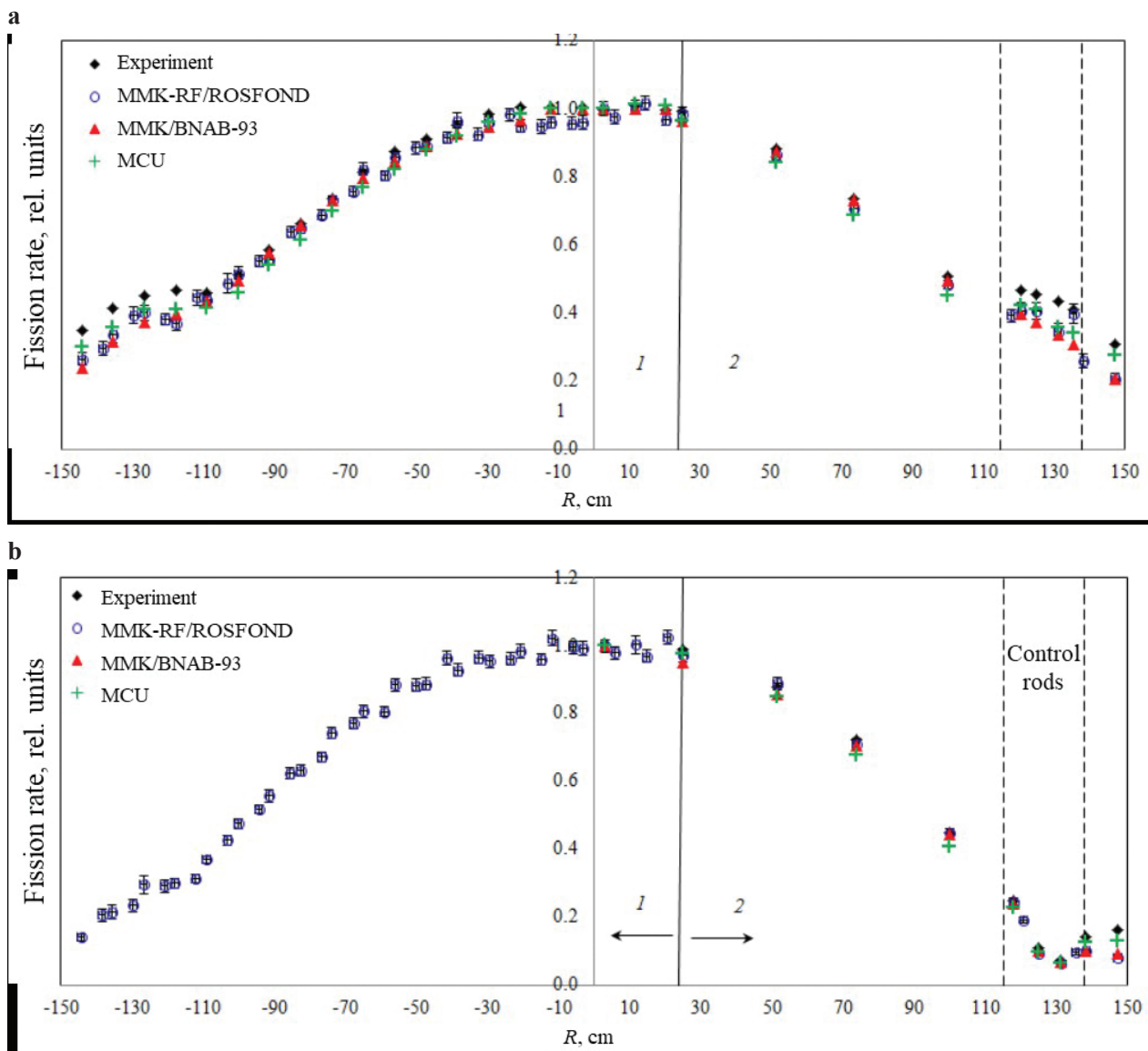
Fig. 2 shows calculated and experimental data for the axial distributions of  $^{235}\text{U}$  (Fig. 2a) and  $^{238}\text{U}$  (Fig. 2b) measured in different channels (gaps between tubes) of the BFS-80-2 assembly. Fig. 2b shows differences in the experimental and calculated values of the  $^{238}\text{U}$  fission rate.

This threshold reaction path is explained by the cell-type structure of the BFS critical assemblies the cores of which represent alternating fuel pellets, coolant pellets, etc.

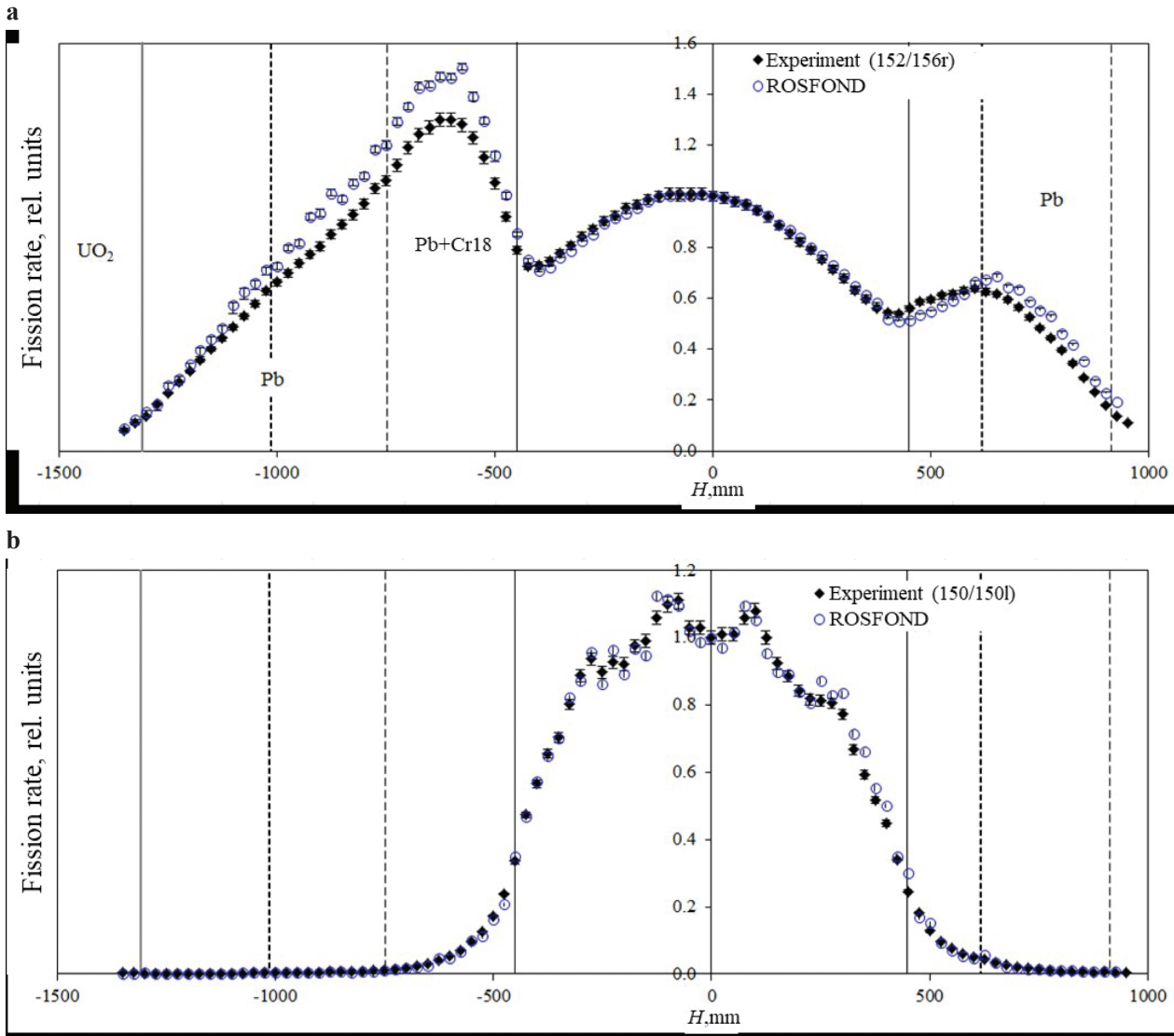
The absolute errors of the fission rate measurements are about 1.5 to 2% for the  $^{235}\text{U}$  and  $^{239}\text{Pu}$  isotopes in the core with an increase to 3 to 4% in the blankets and reflectors, and 2 to 3% in the core and 5 to 7% in the blankets and reflectors for  $^{238}\text{U}$ . The statistical error of the  $^{235}\text{U}$ ,  $^{238}\text{U}$  and  $^{239}\text{Pu}$  fission rate calculation did not exceed the measurement error.

## Analysis of the calculation results

The following discrepancies between the calculation and the experiment, average for all measurements, were estimated in the process of analyzing the calculations of the relative fission rates for the axial and radial distributions



**Figure 1.** Calculated and experimental radial distributions of the relative  $^{235}\text{U}$  fission rate over the BFS-64-1 core: a) without control rods; b) with in-core control rods; 1 – radial direction in the central plane; 2 – radial direction through control rods.



**Figure 2.** Calculated and experimental axial distributions of the relative fission rate over the BFS-80-2 core: a)  $^{235}\text{U}$ ; b)  $^{238}\text{U}$ .

measured using uranium and plutonium fission chambers placed opposite the layers of different materials in the core and in the radial and end blankets:

- in the core, independent of the fuel composition (MOX, uranium or nitride uranium-plutonium fuel), the calculation and experiment discrepancies lie in the experimental error limits and do not exceed 3%;
- the calculation and experiment discrepancies do not exceed 5% in reflectors of the Cr 18% steel grade, and are in the limits of 5 to 10% for the Cr 28% steel grade which complies with the measurement error;
- the calculation and experiment discrepancies in a chromium reflector reach 15%, the calculated values being below the experimental data;
- the calculated values in a steel reflector with a high content of iron are an average of 20% as high as the experimental data;
- the calculated values of the  $^{235}\text{U}$  and  $^{239}\text{Pu}$  fission rates in a nickel reflector are 5 to 10% as high as the

experimental data which does not exceed the measurement error;

- the calculated values of the  $^{235}\text{U}$  and  $^{239}\text{Pu}$  fission rates in the end and radial lead reflectors are an average of 15% and maximally up to 25% as high as the experimental data.

The calculation and experiment discrepancies in the distributions of the  $^{235}\text{U}$  and  $^{239}\text{Pu}$  fission reaction rates in the steel radial blanket of a fast reactor at a level of 20 to 25% were recorded earlier in (Danilychev et al. 2002; Kochetkov et al. 2007; Mitenkova and Novikov 2011a). The results obtained in the study do not contradict to the calculation accuracies for the distributions of the power density field in a steel blanket evaluated earlier in (Blyskavka et al. 2010; Mitenkova and Novikov 2011). An analysis of the BFS-66-B series experiments has however shown that a reduction in the calculation and experiment discrepancies to the measurement error level for the Cr 18% and Cr 28% steel grades can be explained by an overcompensati-



on of errors in determining the Fe, Cr and Ni isotope neutron cross-sections. Further identification of the causes for the observed calculation and experiment discrepancies for steel and lead reflectors requires a specialized analysis involving sensitivity and uncertainty analysis frameworks. Such studies make it possible to update the neutron data for the Fe, Cr, Ni and Pb isotopes based on a calculation analysis of the experiments discussed and to prepare a problem-oriented neutron data library for lead cooled fast reactors.

The performed analysis has shown a high informational value of the earlier experiments at the BFS-66, -64 and -80 critical assemblies. This allows concluding that the developed models can be used to verify codes, form a problem-oriented neutron data library and improving the calculation accuracy of the reactor core performance as well as make it possible to present experiments as benchmark models.

## Conclusion

As a result of the activities on analyzing and revising earlier calculation and experimental studies on the transmission of neutrons through the layers of a steel reflector at the BFS-66, -64 and -80 critical assemblies, the verification database has been expanded for estimating the uncertainty of the power density in the steel reflector of lead cooled fast reactor designs. The calculation analysis performed is the starting point for further calculation and experimental studies to justify and improve the reliability of calculating the reactor performance for advanced lead cooled fast reactor designs. The experiments discussed can be presented as benchmark models for forming neutronic code verification base as well as to evaluate the calculation uncertainty of the power density in the steel reflector of lead cooled fast reactors.

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