Benchmark-experiments for Pb and Bi neutron data testing

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Abstract

The expedience of accurate estimation of neutron data for Pb and Bi has increased recently in connection with the Accelerator-driven system (ADS) projects and the new generation fast reactors under development, which shall use lead or lead-bismuth coolant. Still the significant difference (10%) in the energy range of 100 keV – 500 keV, for the σ^{tot} from various data sets has been observed . The differences found are associated with the energy range, for which experimental information is lacking. The situation with Bi data is not better. In this connection, several benchmarks were assembled at BFS with uranium and plutonium fuel and lead or lead-bismuth coolant.

The scope of the investigations included the measurements of the spectral indexes, distributions of the fission rates of the main isotopes, small samples worths and coolant voiding. The special program was connected with minor actinides. The influence of the plutonium isotope composition was investigated at the assemblies with reactor and weapon grade Pu.

Calculations of the measured parameters were carried out using the most modern versions of nuclear data libraries.

All the results of these experiments and their analysis have prepared for the construction of the benchmarks and planed as the candidates for the International data base **IRPhEP**.

KEYWORDS: Fast critical assembly, experiments, Pb and Pb-Bi coolants, neutron data, calculation results, reactor experiment benchmarks data base

1. Introduction

The experience of estimation of neutron data for Pb and Bi has increased recently in connection with the new reactor system designs under development, which shall use lead or lead-bismuth coolant. The experiments carried out previously at critical assembly BFS-61 - the benchmark of fast reactor with plutonium fuel and lead coolant – have shown a significant difference between the calculated and experimental values for both spectral indices, and for k_{eff}.

The significant difference till 10% in the energy range of 100 keV – 500 keV, for the σ^{tot} has been observed A considerable difference has been found also in the estimation of cross-sections for inelastic moderation processes. The differences found are associated with the energy range, for which experimental information is lacking.

At BFS facilities the peculiarities of the Pb and Bi neutron data were investigated in series of critical assemblies for the wide energy diapason of fast neutron spectrum.

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2. The Description of the Experiments

The following critical assemblies were investigated (See Table 1).

BFS-61 - models of a small fast reactor with mixed Pu-U nitride fuel and lead coolant. Measurements on cores with various radial reflector compositions were carried out.

BFS-77 had a central sub-core (∅ ~ 85 cm) corresponding to that of BREST-300 reactor [1] (mixed nitride fuel, lead coolant, BR for the core of ≈ 1); the experimental program was intended mostly for the measurement of central functionals (spectral indices, capture in uranium vs. fission of plutonium, central coefficients of reactivity, void effect of coolant, etc.).

BFS-64 sector model of the BREST-300 reactor with radial reflector from Pb-Bi (central angle ~ 120^0); core diameter > 2 m, core height ~ 1 m. The program of experiments dealt with measurement of effectiveness of the control and protection system (CPS) elements made from various materials, radial and axial distribution of the response speed, and radial dependence of the coolant's void effect.

The needs of more correct estimation of Bi neutron data triggered the creation of the benchmarks with inserts on the base of Pb and Pb-Bi compositions. Two types of critical assemblies were investigated:

BFS-85 was the uranium oxide core with enrichment of ~ 60%, two variants (Pb, Pb-Bi) of radial reflector and central insertion (∅ ~ 25 cm); the experimenting programs included measurement of ratios of fission rates for chambers with U-238, U-235, Pu-239, Np-237, Am-241, Am-243, Cm-244 distribution of fission rate for chambers with U-238, U-235, Pu-239).

BFS-87 was mixed Pu-U nitride core containing Pu in fuel of ~ 30% with two variants of reflector – Pb and Pb-Bi, the experimenting programs included measurement of spectral indices at the center, distribution of fission rates and measurement of β_{eff}.

The problem of minor actinides (MA) transmutation in the advanced reactor designs with heavy coolants also was important, and in addition there were no sufficient amount of the experiments in the correspondent energy range.

The investigations of the MA constants were carried out at **BFS-95-1** and **BFS-95-2** critical assemblies - the benchmarks with plutonium nitride fuel and lead coolant.

Table 1: The parameters of investigated critical assemblies.

Assembly	**BFS-61**	**BFS-77**	**BFS-64**	**BFS-85**	**BFS-87**
Model	U-Pu-N	BREST-300	BREST -300	Pb/Bi data	Pb/Bi data
Description	Behchmark	Behchmark	Mock-up	Behchmark	Behchmark
Fuel	U-Pu-C	U-Pu-C	U-Pu-C	U-C	U-Pu-C
Coolant	Pb	Pb	Pb	Pb / Pb-Bi	Pb / Pb-Bi
Dimensions:					
R (cm)	45	42 +13*	110	11** +20	37
H (cm)	86	109	109	47	66
Enrichment %	15	11	11	60	25

* - uranium driver

**** -** Pb or Pb-Bi central insertion

3. Computer codes

The analytical treatment of the experimental results obtained at BFS critical assemblies were carried out by the basic code used for fast reactor analysis: **TRIGEX** [2], with code of constants preparation **CONSYST/ABBN** [3] and with **ABBN-93** [4] cross section data. Heterogeneous structure of the BFS assemblies was taken into account, and all others necessary corrections to **TRIGEX** code homogeneous diffusion calculation results are evaluated using **FFCP** [5] and **TWODANT** [6] codes. Monte Carlo precision calculations were made on the basis of **MMK-KENO** [3] code.

3.1 Cross sections data base

The set of group cross section data ABBN-93, developed at the SSC RF IPPE was taken as a basis fro the analysis of critical assemblies. The set of cross section data ABBN-93 – is the new Russian system of the group cross section data with traditional (28 groups) and multi-group (299 groups) division. This system has been designed for the analysis of neutron and photon patterns and their functionals in the core, blanket and shielding of different types of nuclear reactors. ABBN-93 system is based on the files of evaluated nuclear cross sections data recommended by the experienced specialists. These files extracted from the libraries of evaluated nuclear data, namely ENDF/B-VI, JENDL-3, BROND-2 and some other sources, are included into the Russian library of evaluated data FOND-2 [7] (SSC RF IPPE).

The cross sections and codes system CONSYST/ABBN (SSC RF IPPE) provides different applications of the ABBN cross section data. The key codes of CONSYST/ABBN system are CONSYST, PRECON1 and PRECONS Fortran codes. PRECON1 and PRECONS codes couple the group cross section data of ABBN with such Russian computer codes as SYNTES (VNIIAES), JARFR (SSC KI), FACT-BR (NIKIET) and TRIGEX (SSC RF IPPE). CONSYST code provides cross section data for the kinetics codes such as ANISN and TWODANT, as well as for codes intended for precise calculations of the reactor characteristics using Monte Carlo method, for instance MMK-KENO and MCNP. In MCNP-4B calculations by Monte Carlo method ML45 (NIKIET) library also is used was generated on the base of ENDF/B-6 data (see. Table 2). For the analysis of the systems with heavy coolants the ABBN/BREST (SSC RF IPPE) data library also was generated on the base of ENDF/B-6 data and used .

Table 2: The basic libraries used for the preparation of the nuclear data of the main isotopes for the calculations by MCNP-4B code.

Nuclide	Library	Nuclide	Library
Pb204	JENDL-3.2	Np^{237}	ENDF/B-VI rev.5
Pb206	ENDF/B-VI rev.5	Pu^{238}	JENDL-3.2
Pb207	ENDF/B-VI rev.5	Pu^{239}	ENDF/B-VI rev.5
Pb208	ENDF/B-VI rev.5	Pu^{240}	ENDF/B-VI rev.5
U^{234}	JENDL-3.2	Pu^{241}	ENDF/B-VI rev.5
U^{235}	ENDF/B-VI rev.5	Pu^{242}	JEF-2.2
U^{236}	JENDL-3.2	Am^{241}	ENDF/B-VI rev.5
U^{238}	ENDF/B-VI rev.5	Am^{243}	JEF-2.2

4. Analytical treatment of experimental results

By now, analysis of BFS-61 critical assembly experiments has been completed, and analysis has been started on the experiments performed on other critical assemblies (preliminary results of this analysis are given below).

4.1 Criticality

Results of K_{eff} calculations for series of critical assemblies are presented on Fig. 1. In the Table 2 on example of BFS-61 assembly the scale of the effects were taken into account in the calculation analysis is shown. It is seen, that the modern version of Russian constants system ABBN-93 is well predict criticality of described assemblies. We shall remark that Pb cross-sections in this constants system are obtained with using of JENDL-3 estimation. To receive conception on existing uncertainty of the neutron data for lead it is possible by comparing with other estimations. On Figure1 and in Table 3 the comparison results of K_{eff} calculations with using estimation of Pb cross-sections from JENDL-3.2 file are presented. JENDL-3.2 estimation as it is possible to see from obtained data gives a little bit major differences from experiment. It is possible to mark that in assemblies (BFS-61, BFS-77) containing lead only in core the last estimation gives lower K_{eff} values in comparison with ABBN-93. In the case of presence of lead in reflector situation more complicated and requires more detailed analysis. At present the obtained results constrain us from replacement of lead cross-sections in ABBN-93 on later estimations. The criticality of the assemblies containing in reflector a lead-bismuth alloy also is not bad agreed with experiment. We shall remark that calculations were implemented with application of the bismuth constants were obtained on the basis of the data, contained in JEF-2.2 file. The results obtained independently with using of the various modern computational programs based on Monte-Carlo method (MCNP, MMK-KENO) also are sufficiently well agreed.

Table 2: Estimated calculated values of criticality parameter.

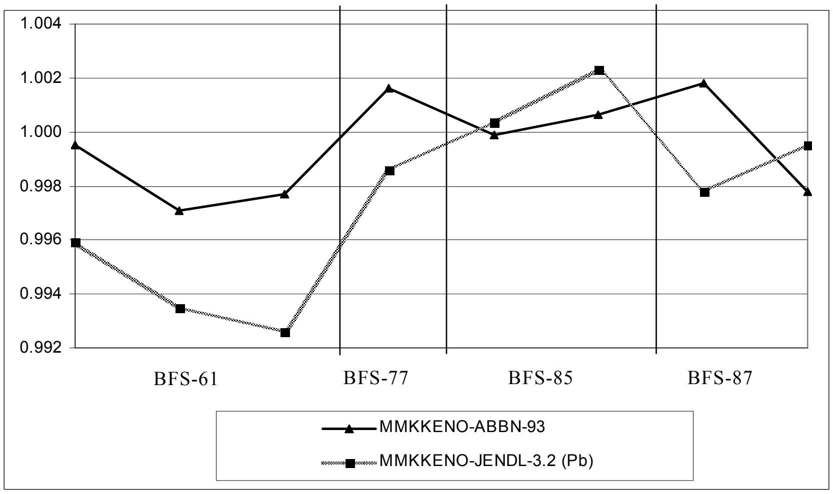
BFS assembly	**61-0**	**61-1**	**61-2**
Basic calculation	0.9624	0.9653	0.9677
Heterogeneous Correction	+2.54%	+2.36%	+2.26%
Kinetic Correction	+1.0%	+0.9%	+0.74%
Other corrections	+0.1%	+0.1%	+0.74%
Estimated Value	0.998 ± 0.006	0.998 ± 0.006	0.998 ± 0.006

Table 3: Precision calculations of criticality parameter.

BFS No.	MCNP-4B ML-45	MMK-KENO code	
		ABBN-93 (JENDL-3 for Pb)	ABBN-93+ JENDL-3.2 for Pb
61-0	-	0.9995(5)*	0.9959 (5)
61-1	-	0.9971(5)	0.9935 (5)
61-2	-	0.9977(5)	0.9926 (5)
77-1	1.00025(13)	1.0016(6)	-
77-1a	1.00074(13)	-	-

* - statistical errors

Figure 1: The results of K_{eff} calculations.



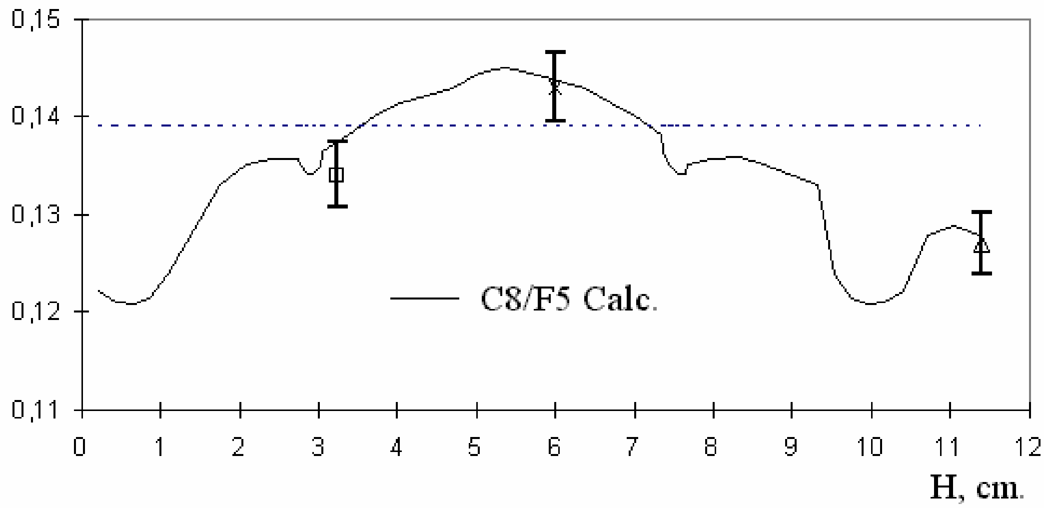
4.2 Spectral indexes.

Results of calculation of main spectral indexes (f8/f5, f9/f5, c8/f5) are well enough agreed with experimental data that indicate a correctness of calculation of neutron spectrum. Comparative results are presented in Table 4. Under the analysis of spectral indexes the taking into account of heterogeneous structure of BFS fuel tubes is played essential role. On Fig. 2 calculation and experimental results of distribution of spectral index c8/f5 through the core cell of BFS-61-0 assembly are presented. **For some actinides (^{238}Pu, ^{240}Pu and ^{241}Am) discrepancies can reach up to 10%** that, apparently, indicate the present accuracy of knowledge of fission cross-sections of these nuclides in region of energy typical for fast reactors with lead coolant and for ADS blankets.

Table 4: Calculation to experiment ratio for spectral indexes.

Index	**БФС-61**	**БФС-77**	**БФС-77**
	TRIGEX ABBN-93	TRIGEX ABBN-93	*MCNP-4B ML-45*
C238/F235	1.001 ± 0.024	1.056±0.050	1.013±0.050
F238/F235	0.968 ± 0.030	1.011±0.030	0.985±0.030
F239/F235	1.002 ± 0.015	0.996±0.014	0.995±0.014
F240/F239	1.050 ± 0.020	1.043±0.033	1.103±0.033
FNp^{237}/F239	-	1.079±0.042	1.039±0.042
FPu^{238}/F239	-	1.056±0.022	1.063±0.022
FAm^{241}/F239	-	0.985±0.039	0.907±0.039
FAm^{243}/F239	-	1.15±0.08	1.016±0.080

Figure 2: Comparison of calculation and experimental distribution of C8/F5 alonge BFS-61 core cell.



5. Conclusions

The experimental program of investigation of neutron-physical characteristics of critical assemblies with uranium-lead and plutonium-lead cores and with Pb and Pb-Bi reflectors represents wide interest from the point of view of check of the neutron data and methods of calculation of perspective fast reactors with heavy coolant and also ADS systems. Preliminary results of the carried out calculation analysis indicate that accepted in the Russian constant system ABBN-93 library allows quite good describe experimental data. Group constants for lead used in ABBN-93 are obtained with using of JENDL-3 files, Bi – with JEF-2.2. Causes interest clarification of reasons more larger discrepancies under using of later estimations, for example, JENDL-3.2. As a whole work on analysis of experiments is not completed fully. Within the framework of international cooperation it is supposed to make efforts to evaluation of experiments, creation on their basis benchmarks, implementation of calculation analysis with attracting a wide set of the nuclear data libraries and codes, and all the results of these experiments and their analysis planed as the candidates for the International data base **IRPhEP**.

Acknowledgements

The authors wish to thank the Alexander Lopatkin (NIKIET) for the presented results of calculations by MCNP-4B code.

References

1) I.P. Matveenko, A. M. Tsiboulya, G. N. Mantourov, et al. "Experimental investigations of BREST-OD-300 reactor characteristics at BFS facility". In: Working materials for IAEA-AG-1076, IWG-FR/104, Vienna, Austria, p.p. 204-239 (2001).
2) A.S. Seryogin, T.S. Kislitsyna. "Abstract of TRIGEX-CONSYST-ABBN90 Code Package" - Preprint IPPE-2655, Obninsk (1997).
3) A.A. Blyskavka, G. N. Manturov, M. N. Nikolaev, et al. "The CONSYST//MMKKENO Code Package for Nuclear Reactors Calculation with Monte-Carlo Method" - Preprint IPPE-2887, Obninsk (2001).
4) G. N. Manturov, M. N. Nikolaev, A. M. Tsiboulia, et al.," ABBN-93 Group Constant System Part 1: Nuclear Constants for Calculation of Neutron and Photon Radiation Fields," *VANT, Ser. Nuclear Constants*, **v.1**, p. 59 (1996) [in Russian].
5) A.A. Bezborodov, B.G. Ryazanov, M. M. Savoskin, et al., "Evaluation of Heterogeneous Effects by Method of the First Collision Probability in Fast Neutron Critical Assemblies," *VANT, Ser. Physics and Technology of Nuclear Reactors*, **v.2**, p.8 (1986) [in Russian].
6) TWODANT-SYS, RSICC Peripheral Computer Code Collection ID CCC-547, ORNL.
7) V.N.Koscheev, M.N.Nikolaev, A.M.Tsiboulia, G.V.Savoskina. The FOND-2.2 Evaluated Neutron Data Library. IAEA-NDS-199, Rev.1. 2002.