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Validation of Neutron Data for Pb and Bi Using Critical Experiments

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A series of critical assemblies at the BFS facility was created for the validation of the neutron data for Pb and Bi in fast reactor spectrum. It is planned to continue the experiments in the nearest future. The analysis of the experiments has shown that the use of neutron data from files of various national libraries results in significant differences in the calculated values of basic neutronic characteristics.

***KEY WORDS:** fast reactor spectrum, critical assemblies at BFS facility, uranium and plutonium fuel, lead or lead-bismuth coolant, k_{eff}, spectral indices, void reactivity effect, experiment vs. calculation comparison*

I. Introduction

The experience of accurate estimation of neutron data for Pb and Bi has increased recently in connection with the Accelerator-driven system (ADS) project and the new generation fast reactors 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 agreement with the experimental data was achieved after the use of new neutron data for Pb in the ABBN-93.1[1)] library.

The studies of neutron data available has not shown any advantages with the new versions in libraries ENDF/B-VI[2)] and JENDL-3.2[2)] vs. the estimations obtained based on ABBN-93.1. In the energy range of 100 keV – 500 keV, the difference reaching 10% in 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.

In this connection, several benchmarks were assembled at BFS with uranium and plutonium fuel and lead or lead-bismuth coolant. It is planned to continue these experiments, and to use the results of analysis for the validation and correction of neutron data.

II. Experimental Program for Critical Tests

In addition to the experiments carried out previously at critical assembly BFS-61 (model of a small fast reactor with mixed Pu-U nitride fuel and lead coolant), the following critical assemblies have been investigated in recent years:

- BFS-77 had a central sub-core (∅ ~ 85 cm) corresponding to that of BREST-300[3)] reactor (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 ~ 1200); 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 need of a more precise estimation of neutron data for Bi necessitated the creation of benchmarks with sub-cores arranged consecutively (with basic Pb or Pb-Bi). Two critical assemblies were studied:

- BFS-85 (the uranium oxide core with enrichment of ~ 60%, two variants of radial reflector and central insertion (∅ ~ 25 cm from Pb, then - from Pb-Bi); the experimental 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). The layout of BSF-85 assembly is shown in **Fig. I.**

- BFS-87 (mixed Pu-U nitride core containing Pu in fuel of ~ 30%, two variants of reflector – Pb and Pb-Bi, measurement of spectral indices at the center and distribution of response rate, measurement of β_{eff}).

III. Analysis of the Results

Estimated calculation values of the criticality parameter for BFS-61, obtained using TRIGEX[4)] code with constants of ABBN-93 are presented in **Table 1.**

The uncertainty values attributed to the estimated results were obtained as ¼ from the estimated correction for heterogeneity, which appears to be prevailing in this case. The estimates obtained were substantiated by the precision calculations by Monte-Carlo method by MMK-KENO[5)] code (see **Table 2**). In these calculations, each critical assembly was represented with all geometrical features (including the core support barrel of BFS and its supporting slab).

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The calculations by MMK-KENO code were carried out with constants of ABBN-93 in 299-groups, with P5-approximation for the scattering matrix (statistical accuracy of calculation of ± 0.0005).

As a comparison, Table 2 shows the calculation results with constants for lead from the evaluated data library JENDL-3.2. These calculations are lower than those obtained with ABBN-93, their predictions underestimating the BFS-61 assembly criticality by 0.4-0.5%. It should be noted that the group constant system of ABBN-93 for lead uses the evaluation from the previous version of the estimated data library JENDL-3[2].

The calculation analysis of measured spectral indices was also carried out in the frame of TRIGEX program, with accounting for the space and resonance-related heterogeneous effects which were calculated by FFCP[6] code.

Table 3 presents the estimated values of spectral indices vs. measurement results. The said results have been given with the uncertainties estimated by the experimenter taking into account the uncertainty of corrections introduced into the calculation.

Results of the void reactivity effect analysis for BFS-77 assembly are presented in **Table 4**.

Table 5 shows the constant components of the calculation uncertainty for fast reactor BREST-300 after the first phase of the analysis of results; obviously, additional experiments and analysis thereof are necessary (in particular, for Pb-Bi).

Abbreviations used:

BFS – Fast critical facility; BR – breeding ratio.

Table 1 Estimated calculated values of criticality parameter

BFS assembly No	**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 2 Precision calculations of criticality parameter

BFS Assembly No.	Calculations of k_{eff} by the MMK-KENO code	
	Heterogeneous calculation using ABBN-93 (JENDL-3 for Pb)	Heterogeneous calculation using JENDL-3.2
61-0	0.9995	0.9959 (-0.4%)
61-1	0.9971	0.9935 (-0.4%)
61-2	0.9977	0.9926 (-0.5%)

Table 3 Calculation-vs.-experiment for spectral indices (in indices capture is marked C and fission is marked F)

Functional	Introduced corrections	Estimated calculated values	Calc./Exp.
C8/F5 (1)	1.000	0.128	1.001 ± 0.024
C8/F5 (2)	1.000	0.137	1.025 ± 0.023
C8/F5 (3)	1.000	0.144	1.001 ± 0.021
F8/F5 (1)	1.000	0.0305	0.968 ± 0.030
F239/F235	0.999	1.059	1.002 ± 0.015
F240/F239	0.992	0.271	1.050 ± 0.020
F241/F239	1.000	1.261	1.002 ± 0.015

Table 4 Void reactivity effect (in portions, cents)

Total number of tubes	Number of tubes per portion	Experiment – portion and per one tube	Calculation - portion and per one tube
		Upwards	
19	19	-16.2 (-0.85)	-15.8 (-0.83)
34	15	-11.5 (-0.77)	-12.9 (-0.86)
56	22	-16.9 (-0.77)	-13.8 (-0.63)
85	29	-25.1 (-0.87)	-21.3 (-0.73)
		Core	
19	19	+1.4 (+0.074)	-9.6 (-0.505)
34	15	+1.9 (+0.127)	-6.5 (-0.433)
56	22	-0.3 (-0.014)	-7.4 (-0.336)
85	29	-1.2 (-0.041)	-10.2 (-0.352)

Table 5 Estimated constant components of calculation uncertainty

Parameter	Constant component of uncertainty	
	Without macro-experiment (the micro-data level)	With account for experiment at critical assemblies
k – eff	2.3	1.1
BGA (+0.11)	53.2 (± 0.06)	26.3 (± 0.03)
VOID (+0.98 Δk/k)	41.3 (± 0.48 Δk/k)	38.6 (± 0.48 Δk/k)
DOPL (-0.48 Δk/k)	12.1	12.1

IV. Conclusions

- Non-reliable neutron data for recent advanced Reactor core designs with lead or lead-bismuth coolant so as for ADS Projects was the reason to re-estimate old experiments and to start the new experiments at BFS facilities.
- Two critical assemblies benchmark and mock-up type of BREST-300 Design and four critical assemblies benchmark types for comparing lead vs. lead-bismuth coolants in U oxide and U-Pu fuel were investigated.
- The program of investigations included measurements of spectral indices, central coefficients of reactivity, void effect of coolant, control rods worth, distribution of fission rates, Minor Actinides and measurement of β_{eff}.
- The calculational analysis of the experimental results are under way now. The calculations carried out using as design

code TRIGEX with corrections on heterogeneity and others effects so as precision Monte-Carlo code MMK-KENO. Different types of Data Libraries were used such as ENDF/B-VI, JENDL-3.2 and ABBN-93.1. The differences found are associated with the energy range, for which experimental information is lacking.

• Based upon carried out calculations, there are plans to prolong the experimental program.

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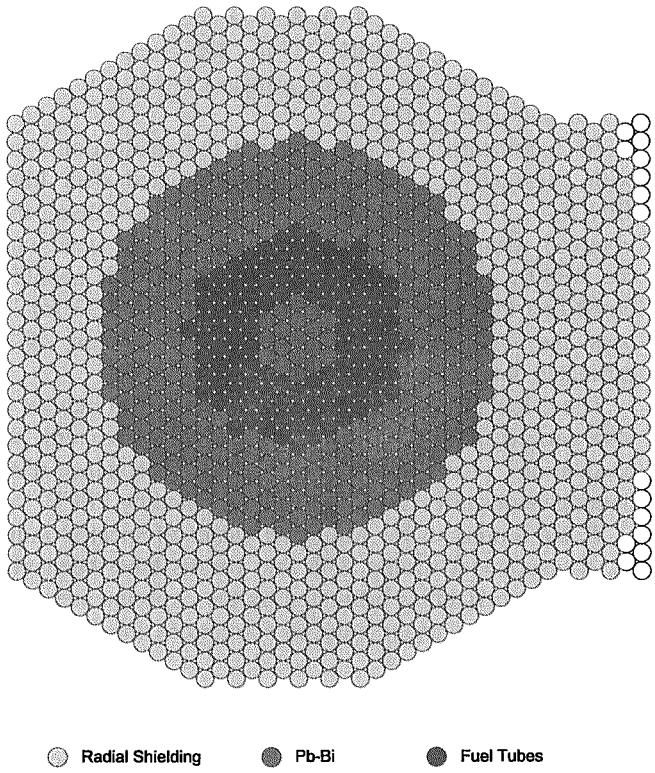

Fig. I Layout of BSF-85 assembly