EXPERIMENTAL STUDIES OF MA NUCLEAR DATA CORRECTION ON CRITICAL ASSEMBLIES

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Abstract

The wide list of experimental results on MA investigations at BFS critical assemblies with different types of neutron spectrum and BN-350 reactor irradiation results are presented for the verification of neutron data libraries. Heterogeneous structure of assembly cells, peculiarities of group constant preparation for reactivity coefficients calculation, definite size of samples were taken into account during analysis using ABBN-93 nuclear data library.

Introduction

The problem of refining of minor actinides (MA) fission cross-section and, especially, MA capture cross-section is rather actual, since at present their uncertainties are several times more then it is required. For the last twenty years fission and capture cross-sections of ^{235}U were studied well, their uncertainty is estimated ~2%. On the contrary, the cross-sections of MA are known with larger uncertainty (5-15%).

Together with new microdata on MA, the experiments of integral type at critical assemblies and reactors are necessary. These measurements are:

- ratio of fission cross-section by chambers;
- central reactivity (worths) coefficients;
- Doppler-effect;
- direct introducing MA into fuel of the core and study of its influence on neutron characteristics;
- irradiation in power reactors.

In the paper the wide experimental programme, carried out at BFS critical assemblies and BN-350 power reactor, is presented for the verification of neutron libraries.

Heterogeneous structure of assembly cells, peculiarities of group constant preparation for reactivity coefficients calculation, definite size of samples were taken into account during analysis. ABBN-93 nuclear data library was used.

The aim of the work

The creation of the experimental data base on neutron-physical parameters of MA were measured at critical assemblies with fast and medium neutron spectrum, and also at BN-350 reactor, for the further adjustment of neutron data and codes.

Experimental facilities

The BFS-1, BFS-2 facilities are used as for simulation of systems with fast neutrons as well as with intermediate and thermal spectrum for the testing of design characteristics, neutron data and codes.

The BFS-2 critical facility was designed in the IPPE for the full-size simulation of cores and shielding of large fast reactors with a unit power up to 3 000 MWe. It was put into operation in October 1969 (see Figure 1). Now it is the biggest working critical facility in the world.

About 10 000 tubes are installed inside of the vessel with an effective diameter ~5 m and a height ~3.3 m. They are made of stainless steel, also aluminium one are available. Their outside diameter is 50 mm and the wall thickness is 1 mm. The hexagonal lattice pitch of the grid is 51 mm.

The space between the tubes can be filled with round or triangular stainless steel or polythene sticks. This space can be also used for inserting different small-size detectors for experimental investigations of reaction rate distributions and others.

The facility equipped by graphite column 1 450×1 600×2 600 mm sizes and metal column, by forced cooling system, fuel rod reloading unit also for oscillation and movement the samples and chambers. The maximum power level is 1 000 watt and temperature till 80°C.

At the BFS facilities there are the large amount of fissile materials (metal and dioxide of uranium 36% and 90% enrichments, weapon and reactor grade metal plutonium, about 9 t), about 280 t of fertile materials, including uranium dioxide and metal, thorium dioxide and 120 t of structural materials (stainless steel, Al, Ni, Nb, Zr, C, B_4C, Al_2O_3.) and 9 t of coolant materials (sodium, lead). Besides, the availability of 11 kg of NpO_2 in BFS pellets, give the possibility to make the mock-up placement of this actinide in the reactor core.

All the reactor materials are in form of pellets with diameter 47 mm and 10-0.1 mm thickness. The pellets some of the reactor materials covered by Al or SS.

Figure 1. **The facility BFS -2**

Measured parameters

At critical assemblies were studied:

- The ration of fission cross sections of MA: ${}^{i}\sigma_f$ / ${}^{U235}\sigma_f$ (i → ^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm, ^{245}Cm, ^{238}Pu, ^{240}Pu, ^{241}Pu) by fission chambers.
- ${}^{Np237}\sigma_c$ / ${}^{238}\sigma_c$ by activation method (foils).

- Spectral indexes ${}^{Pu239}\sigma_f$ / ${}^{235}\sigma_f$, ${}^{U238}\sigma_f$ / ${}^{235}\sigma_f$, ${}^{U238}\sigma_c$ / ${}^{235}\sigma_f$.
- Small sample worths (^{237}Np, ^{241}Am, ^{240}Pu, ^{241}Pu).
- Doppler effect for ^{237}Np, ^{238}U. The sample of ^{241}Am will be prepared soon.
- Sodium void reactivity effect (SVRE).
- Control rod mock-up worths (CRW).
- Adding the Np in the fuel composition of the assembly with aim to investigate his influence at neutronic characteristics such as SVRE, CRW, spectral indexes etc.).

At BN-350 power reactor

The samples of ^{237}Np, ^{241}Am, ^{240}Pu, ^{241}Pu were irradiated and analysed.

Characteristics of investigated reactor models at critical assemblies

The list of reactor models been investigated and planed to be investigated in critical experiments rather significant:

1. Fast reactor with sodium coolant and oxide fuel.
2. Fast reactor with sodium coolant and metal fuel (KALIMER design).
3. Fast reactor with sodium coolant and fuel in inert matrix.
4. Fast reactor with Pb, Pb-Bi coolant and nitride fuel (the programme is on the way).
5. Fast reactor with sodium coolant and nitride fuel (the programme will start this year).
6. Molten-salt reactor with graphite reflector (the programme is in preparation).

This wide list of the models is due to the need of MA cross-section investigations in wide diapason of neutron spectrums of fast reactors designs planed to be investigated (see Figure 2).

Except this, to avoid the uncertainties cause by errors in inelastic cross-sections of Pb and Bi, the special benchmarks were assembled for neutron data testing of these elements.

The description of investigated critical assemblies and the results of measurements

The models of fast reactors with sodium coolant and MOX fuel investigated on critical assemblies BFS-67 (1, 2, 3), BFS-69 (1 and 2), BFS-71 (1 and 2) and BFS-62 (5 and 6)

All of these critical assemblies have different sizes, amount of sub-zones and differ by presence of control rod mock-ups. The specific differences of investigated critical assemblies and measured values are presented in Table 1.

At all of these assemblies the influence of placement of all available NpO_2 (11 kg) in form of BFS pellets, on the main neutronic parameters was investigated.

Figure 2. **Neutron spectrums of investigated reactors (299 energy group)**

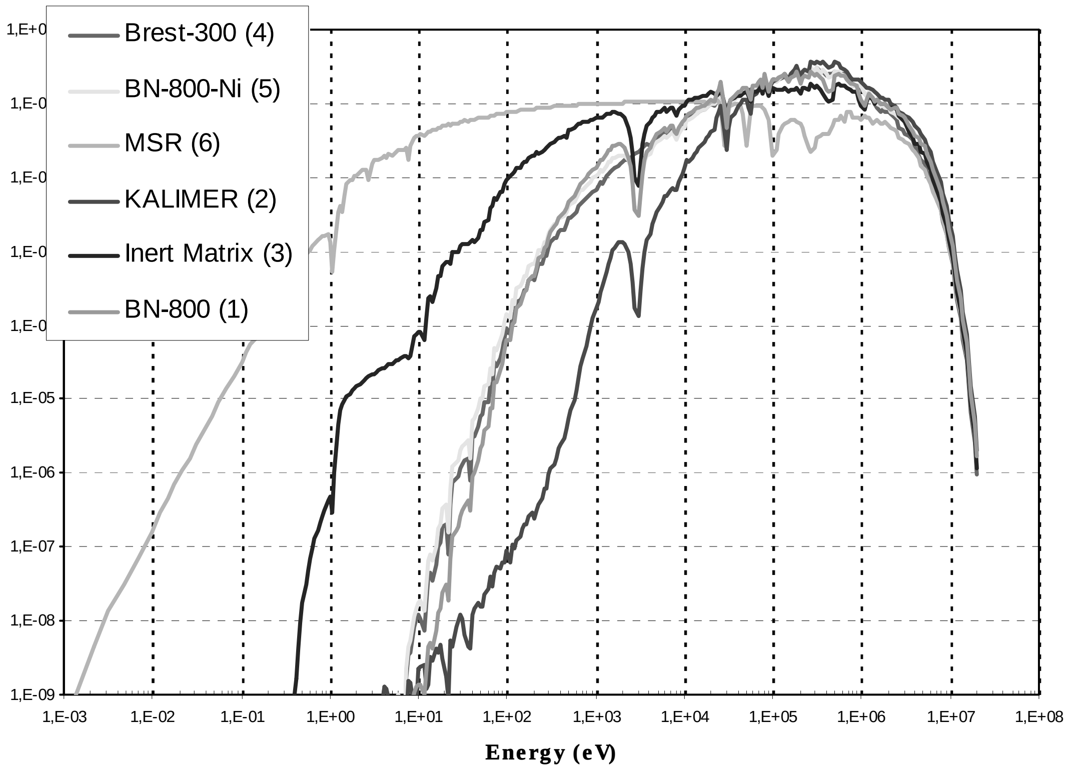


Table 1. **Critical assemblies with sodium coolant and MOX fuel**

Series	**BFS-67**	**BFS-69**	**BFS-71**	**BFS-62**
Modification	-1 / -2 / - 3	-1 / -2	-1 / -2	-5 / -6 /-7
Model	SuperPhenix	CAPRA	Max Enrichment	BN-600 MOX
Characteristic	Benchmark	Benchmark	Benchmark	4 zones model with CR mock-up
Assembly Sizes: R (cm) – H (cm)	35+30* 76	38+14* 75	37 77	73+10+14+8 105
Fissile Enrichment (%)	19	40	57	15:18:17:21
Sizes of Np insertion: R(cm) – H(cm)	0 / 15/ 21 0 / 38 /38	0 / 16 0 / 45	0 / 16 0 / 46	0 / 18 / 12 0 / 31 /31
Amount of Np insert (%)	0 / 13 / 7	0 / 14	0 / 16	0 / 11 / 17
Measured parameters				
Spectral indexes	+	+	+	+
Cross-section ratios of MA	+	+	+	+
SVRE	+	+	+	+
Doppler-effect	-	+	-	+
Power realise	-	+	+	-
Small samples worths	+	+	+	+
CRW	+	+	-	-

* - uranium driver

At whole it should be mention the worsening of the safety parameters due to disposal of Np in the core: the decreasing of control rod worth (see Table 2) and increasing of SVRE positive value (see Table 3). [1]

Table 2. **The absorption worth at BFS-67 series**

Absorber	**Absorption worth** (%β_{eff})		
	BFS-67-1	**BFS-67-3**	**BFS-67-2**
B_4C (80%)	-2.16 ± 0.12	-2.13 ± 0.13	-2.06 ± 0.12
B_4C nat.	-0.879 ± 0.039	-0.825 ± 0.036	-0.787 ± 0.035

Table 3. **Results of SVRE measurements at BFS-67**

				SVRE, (%β_{eff})		
№	FR voided	Cells voided	Na (kg) voided	BFS-67-1	BFS-67-3	BFS-67-2
1	31	3+4+5+6	6.920	6.2 ± 0.3	–	18.8 ± 0.8
2	1	4+5	0.112	0.157±0.010	0.324±0.017	0,398±0.008

In the Table 4 ratios of fission and capture cross-sections measured by fission chambers and by foils at critical assembly with Np (BFS-71-2) and without Np (BFS-71-1) are presented. The level of experimental errors also is shown. It is possible to underline the influence of Np placement on F28/F25 spectral index.

Table 4. **The comparison of measured rate reaction ratios**

	БФС-71-1	**БФС-71-2 (с Np)**
F28/F25	0.0510 ± 0.0008	0.0550 ± 0.0009
F49/F25	1.084 ± 0.021	1.105 ± 0.022
F37/F49	0.317 ± 0.007	0.334 ± 0.007
F48/F49	0.711 ± 0.013	0.720 ± 0.013
F40/F49	0.334 ± 0.006	0.340 ± 0.006
F41/F49	1.227 ± 0.025	1.216 ± 0.025
F42/F49	0.244 ± 0.006	0.257 ± 0.006
F51/F49	0.290 ± 0.006	0.306 ± 0.006
F53/F49	0.220 ± 0.004	0.230 ± 0.004
F64/F49	0.355 ± 0.005	0.373 ± 0.005
C37/C28	5.38 ± 0.17	4.86 ± 0.16

* - F25→last figures of isotope number

| | → last figures of element proton number(2-for ^{235}U)

F – fission, C – capture

The models of fast reactors with sodium coolant and metal fuel were investigated at BFS-55, BFS-73 and BFS-75 critical assemblies

At these critical assemblies (Table 5), only rate reactions ratios and small samples worths for the nuclear data testing were measured.

Table 5. **Critical assemblies with sodium coolant and metal fuel**

Assembly	***BFS-55***	***BFS-73***	***BFS-75***
Model	-	KALIMER	KALIMER
Characteristic	Benchmark	Benchmark	Benchmark
Fuel	Pu	U	U
Assembly sizes: R (cm) – H (cm)	51 102	55 98	26+18 103
Fissile enrichment (%)	10	19	15:20
Measured parameters			
	Fission rate ratios		
F37/F49	-	-	+
F40/F49	+	+	+
F41/F49	+	+	-
F48/F49	-	-	+
F51/F49	-	-	+
F53/F49	-	-	+
F64/F49	-	-	+
	Small samples worths		
^{240}Pu / ^{235}U	+	+	+
^{241}Pu / ^{235}U	+	+	+
^{241}Am / ^{235}U	-	-	+

In common the results of the analysis of this measurements are acceptable.

Fast reactor models with Pb and Pb-Bi coolant and nitride fuel are 4 studied at BFS-61, BFS-77, BFS-85 and BFS-87 critical assemblies

Till now days the nitride fuel was modelling by means of metal fuel (U or Pu) with introduction of some materials (for example C), to be available at BFS facilities, for adjusting of the neutron spectrum of the model and the design.

At these assemblies (see Table 6), also only rate reactions ratios and small samples worths of MA worths for the nuclear data testing were measured. The introduction of Np in the cores is not carried out.

Table 6. **Critical assemblies with Pb and Pb-Bi coolant and nitride fuel**

Assembly	***BFS-61***	***BFS-77***	***BFS-87-1***
Model	–	BREST-300	–
Characteristic	Benchmark	Benchmark	Benchmark
Fuel	U-Pu-C	U-Pu-C	U-Pu-C
Assembly sizes: R (cm) – H (cm)	45 86	40+13* 110	37 66
Fissile enrichment (%)	15	11	25

Table 6. **Critical assemblies with Pb and Pb-Bi coolant and nitride fuel** (contd.)

Measured parameters			
	Fission rate ratios		
F37/F49	+	+	+
F40/F49	+	-	-
F41/F49	+	-	-
F42/F49	+	-	-
F48/F49	-	+	-
F51/F49	-	+	-
F53/F49	+	+	-
F64/F49	+	+	-
C37/C25	-	+	-
	Small samples worths		
^{240}Pu /^{235}U	-	+	-
^{241}Pu /^{235}U	-	+	-
^{241}Am /^{235}U	-	+	-
^{237}Np /^{235}U	-	+	-

* - uranium driver

The results of the analysis of the first experiments with Pb showed the significant differences in C/E values not only for cross section ratios but for Keff too. After correction of inelastic cross section for Pb, the situation changed to better side (see Table 7), but for the evidence the additional special benchmarks were assembled with significant amount of Pb and Pb-Bi in the core and in the blankets (BFS-85 and BFS-87 critical assemblies).

Table 7. **The comparison of experimental and calculational indexes at BFS-61** [2]

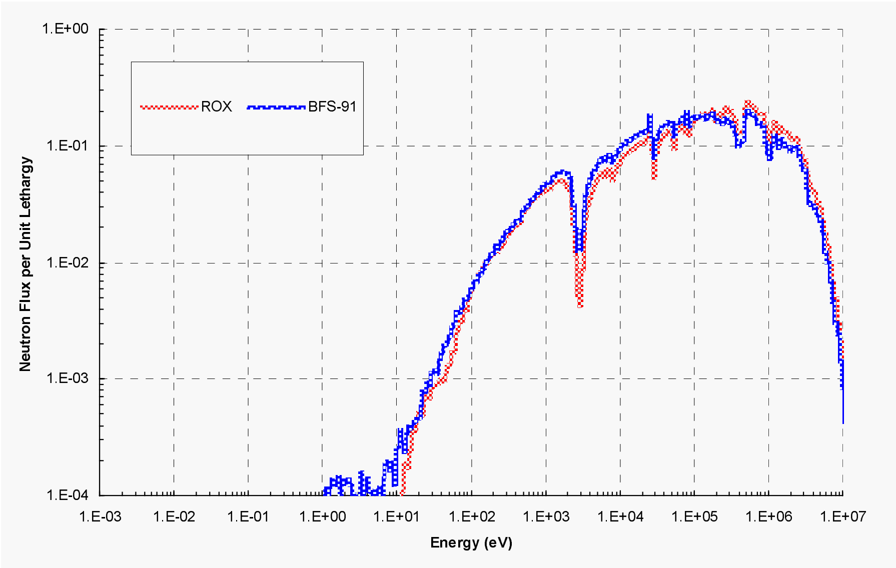
INDEX	Calculation	C/E
F238/F235	0.0305	0.968 ± 0.030
F239/F235	1.059	1.002 ± 0.015
F240/F239	0.271	1.050 ± 0.020
F241/F239	1.261	1.002 ± 0.015

Fast reactor with sodium coolant and fuel with inert matrix

It should be mention that except previously have carried out experiments at BFS-58 critical assembly, the new series of BFS-91 critical assemblies were accomplished. At this series the Doppler effect, spectral indexes and small sample worths were measured.

The investigation on the ROX-fuel problem was carried out as on the model with fast neutron spectrum (BFS-91-1 critical assembly, see Figure 3), so as on the model with more softer neutron spectrum – BFS-91-2 critical assembly.

Figure 3. **Calculated neutron spectrum**



Measurements in BN-350 reactor

BN-350 is the first in the world industrial fast power reactor. A large cycle of neutronics experiments was fulfilled during its operation (i.e. beginning from 1973). So a large scale experimental programme was fulfilled with different actinide samples irradiated in different parts of BN-350 core and blanket These experiments were used for testing of MA cross sections. [3]

It should be mention that the list of experiments carried out at BFS facilities and at BN-350 reactor specially were selected to supplement each other.

The results of the experiments at BN-350 reactor and their analysis will be present in another paper at this Meeting.

Conclusions

- The studying of MA for traditional cores with oxide fuel and sodium coolant is finished.
- The work on investigations of cores with Pb and Pb-Bi coolant is in process now.
- The investigation of MSR is under preparation now.
- After this it is seems that experimental data base will be fulfilled.
- During the work on this Programme the recommendation on cross section correction for ^{237}Np and ^{241}Am capture in the energy diapason of neutrons for the fast reactor MA burner already have been received
- The analysis of MSR physics shows that for the calculations of this reactor the knowledge of data of wide actinides spectrum is demand. Not only data of Np, Am and Pu isotopes, but Cm, Bk and Cf also cause the significant input, and that is should be tested in critical experiments.

References

[1] S. Belov *et al.* (1996), *Investigation of MA Transmutation Problem in Benchmark Experiments at BFS Facility with Neptunium in Fuel Composition*, PHYSOR-96, Mito, Japan, September.

[2] A.M. Tsiboulia *et al.* (2001), *Validation of Neutron Data for Pb and Bi Using Critical Experiments*, ND2001, Japan.

[3] ABBN-90, Multigroup *Constants Set for Calculation of Neutron and Photon Radiation Fields and Functionals*. Including CONSYST2 Program. RSICC Peripheral Routine Program Collection DLC-182.