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VERIFICATION OF MINOR ACTINIDE NEUTRON DATA ON FAST CRITICAL ASSEMBLY EXPERIMENTS.

V.A. Doulin, A.M.Tsiboulia

Physics and Power Engineering Institute, Obninsk, Russia

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The aim of this work is to demonstrate the verification possibility of the modern minor actinides (MA) evaluated neutron cross sections by comparison of its reaction rates and reactivities measured in different fast critical assemblies with calculated values. The absence of sufficient amount of neptunium leads to impossibility of providing of neptunium critical experiments. So the measurements of MA fission rates and reactivities are the only (except of neptunium) possibility to verify the fission and capture cross sections in the integral experiments. The fitness of such way may be tested for neptunium where the activation experiments are possible. Indeed the discrepancy between the experimental fission, capture rates and the well evaluated reactivity worth ratios values and corresponding calculated ones have to be self-satisfied . If it occurred then the verification of the evaluated neutron cross sections for another (nonactivated) MA is possible too.

Two years ago at BFS facility in IPPE Obninsk about 10 Kg of neptunium dioxide was packed in SS cans and used in experiments. The investigation of the inserted neptunium influence on the above mentioned functionals and on K_{eff} value permit to increase its verification possibility .

We represent below as an example the evaluated experimental data obtaining procedure and its comparison results with the calculation using some modern Minor Actinides (MA) evaluated neutron cross sections.

1. EXPERIMENTS.

1.1. FCA-9 assemblies.

Actinide integral measurements were carried out on set of FCA assemblies to test the fission and capture cross sections of minor actinides (MA). The assemblies built for this purpose cover the systematic change of the neutron spectrum shape. The "softest" neutron spectrum was in FCA-9-1 assembly, the "hardest" was in FCA-9-7 assembly The integral data measured are :

- the central fission rate ratio (FRR) of Np-237, Pu-238, Pu-239 ,Am-241, Am-243 and Cm-244 relative to the fission in U-235;
- the central sample worth (CSR) of Np-237, Pu-238, Pu-239, Pu-240, Am-241 and Am-243.

All MA samples were about 20 grams in weight, the Pu-239 and U-235 ones were several times more. The exact description of the assemblies, experimental devices and obtained results of measurements were given in [1].

1.2. BFS assemblies.

The core of BFS-67 assembly was composed with 96% enriched metal plutonium, depleted uranium dioxide, sodium and stainless steel. This composition was similar to the SUPERPHENIX core.

About half of uranium dioxide in this composition was replaced with sodium for constructing of BFS-69 assembly core.

For both BFS assemblies the spectra were similar to the spectrum averaged over FCA assemblies. The integral data measured are:

(1) - FRR of Np-237 and Pu-239 relative to fission in U-235 ;

(2) - CSW of Np-237 ,Pu-239 and U-235.

All samples sizes were less than in FCA experiments. The Np and Pu samples were of three different size. The exact description of the assemblies, experimental devices and obtained results of measurements were given in [2].

2. ANALYSIS OF EXPERIMENTS.

The starting point of neutron data testing is homogeneous calculation of FRR and first order perturbation theory using ABBN approach [3] . Evaluation of experiments means taking in account the heterogeneous structure of core cells, finite sizes of samples and group constant correction at calculation of CSW.

Heterogeneous structure of critical assembly's cell is taken into account by using the integral-transport approximation. Undisturbed group fluxes and adjoint fluxes are obtained from solutions of corresponding integral-transport equations in the cell approximation. Criticality is attained by modification of a neutron leakage. Perturbation of collision probabilities are taking in account too by calculation results using perturbation theory (first type of correction). Consideration of a detailed energy structure of adjoint solution gives the additional contribution into the reactivity worth ratio (second type of correction).

Using samples of finite size, which are placed in the clearances between pellets of the facility results in the distortion of neutron flux on the sample position. The resonance self-shielding in the sample can be described by substitution of group factors of the self-shielding by the generalized factors, and by taking into account distortions of spectrum because of multiplication and slowing down of neutrons (third type of correction) .

Numerical calculations of all these effects are carried out using HEEPCM code. The accuracy of calculations of the heterogeneous and bilinear effects was verified by means of comparison with calculation results obtained from the TULPE code, where another method is used . In addition, a special program of measurements on series of BFS critical assemblies was fulfilled for investigation of calculation accuracy for such heterogeneous effects.

3. RESULTS OF EXPERIMENTS.

3.1. FCA assemblies.

The results of measurements and evaluation are presented in Table 3.1 for CSW ratios of Pu - 239 / U - 235 and in Table 3.2 for CSW ratios of Np-237/Pu-239.

Table 3.1

Assemb. FCA	EXP. virgin	Correction of 1 and 2 types	Correction of 3 type for Pu	Correction of 3 type for U	EXP. evaluated
CSW ratio Pu-239 / U-235					
9-1	1.476	-.023	-.146	+.109	+1.416
9-2	1.617	-.010	-.127	+.124	+1.604
9-3	1.713	-.009	-.008	+.016	+1.712
9-4	1.708	-.006	-.079	+.089	+1.712
9-5	1.750	0	-.075	+.081	+1.756
9.7	1.745	-.002	-.065	+.060	+1.738

Table 3.2

Assemb. FCA	EXP. virgin	Correction of 1 and 2 types	Correction of 3 type for Pu	Correction of 3 type for U	EXP. evaluated
CSW ratio Np-237 / Pu - 239					
9-1	-.865	+.179	-.220	-.073	-.979
9-1	-.242	+.072	-.042	-.013	-.225
9-3	-.014	+.021	-.014	+.005	-.002
9-4	+.054	+.006	-.005	+.006	+.061
9-5	+.158	+.002	-.004	+.010	+.166
9-7	+.117	+.001	-.003	+.007	+.122

3.2 BFS assemblies.

The results of measurements and evaluation are presented in Table 3.3 for CSW ratios of Pu-239/U-235. The table contains the results for three different types of Pu samples. The results of measurements and evaluation are presented in Table 3.4 for CSW ratios of Np-237/U-235. The table contains the results for three different types of Np samples.

Table 3.3

Assembly	Type of sample	EXP VIRGIN	ZERO SIZE of samples	correction of 1 and 2 types	EXP evaluated
67-1	Pu - A	1.350 ± .020	1.327 ± .021		
	Pu - B	1.406 ± .008	1.365 ± .009		
	Pu - C	1.404 ± .005	1.352 ± .007		
	averaged value		1.353 ± .019	+.011	1.364 ± .019
69-1	Pu - A	1.591 ± .018	1.560 ± .020		
	Pu - B	1.607 ± .007	1.552 ± .008		
	Pu - C	1.592 ± .004	1.528 ± .006		
	averaged value		1.541 ± .019	-.002	1.539 ± .019

Table 3.4

Assembly		VIRGIN EXP	ZERO SIZE of samples	correction of 1 and 2 types	evaluated experiment
67-1	Np - A	-.240 ± .011	-.250 ± .011		
	Np - B	-.228 ± .006	-.240 ± .008		
	Np - C	-.228 ± .005	-.245 ± .007		
	averaged value		-.245 ± .010	+.023	-.222 ± .011
69-1	Np - A	-.120 ± .010	-.128 ± .010		
	Np - B	-.119 ± .006	-.130 ± .007		
	Np - C	-.114± .003	-.131 ± .004		
	averaged value		-.130 ± .005	+.021	-.109 ± .006

Notes: A) Pu size : Nl = 0.0044 (barn $^{-1}$); Np -size : Nl = 0.00220 (barn $^{-1}$)
B) Pu size : Nl = 0.0076 (barn $^{-1}$); Np size: Nl = 0.00285 (barn $^{-1}$)
C) Pu size : Nl = 0.0091 (barn $^{-1}$); Np size Nl = 0.00455 (barn $^{-1}$)

4. COMPARISON OF CALCULATED AND EXPERIMENTAL DATA

Simple homogeneous spherical models of all considered experiments were constructed. Calculations were carried out in 28 group S_4-approximation by means of CRAB-1 code. The ABBN-90 group constant set was used, but neptunium-237 cross sections were variable. All calculation models were exactly critical. Criticality was achieved by adjustment of core radius. Experimental data were reduced to conditions of calculation models (corrections on finite dimensions of samples and on heterogeneity were introduced). This reduced data are given in the Tables 4.1 and 4.2 in comparison with results of calculations performed by using different Np-237 neutron data. For reactivity worth ratios differences between calculation and experimental data are because percentage measure of differences are not adequate for the value with changed sign. For fission cross section ratios differences are given in the percent.

Better description of experimental fission ratio was achieved by using JENDL-3.2 data. Discrepancies between experiment and calculation are not systematic and can be fully explained by integral experiment uncertainty. The ABBN-90 set describes these data practically with the same accuracy (see Figure 1 where discrepancies of experimental and calculated data are plotted dependent on value of reactivity worth ratio).

From consideration of reactivity worth ratios another picture is appeared. The ABBN-90 set leads to systematic deviations from experimental data. So old ENDF/B-5 evaluation is not appropriate. The more modern data JENDL-3.2 and other allows to describe reactivity worth ratios in the interval from -0.5 to +0.5 with about the same accuracy. Discrepancies between experimental and calculated data in this region can be explained by experimental uncertainties. But in the case of FCA-9-1 assembly (with the more soft neutron spectrum) JENDL-3.2 set leads to large negative ^{237}Np reactivity worth value. This discrepancy may be eliminated by correction of the capture cross section of ^{237}Np in keV-energy region.

Table 4.1. Comparison of experimental and calculated CSR data (ρ_{237} / ρ_{235}).

Assembly	Experiment	(Calculation - Experiment)*100			
		ABBN-90	ENDF/B-VI	JEF-2	JENDL-3.2
FCA-9-1	- 1.390	- 3.6	- 1.5	- 5.0	- 15.0
FCA-9-2	- 0.361	5.7	- 1.0	- 5.5	- 9.8
FCA-9-3	-0 .003	12.9	3.6	- 1.2	- 2.6
FCA-9-4	0.104	15.0	5.9	1.4	3.8
FCA-9-5	0.291	14.2	5.1	0.3	3.2
FCA-9-6	0.433	17.7	9.0	3.6	6.9
FCA-9-7	0.204	14.1	7.5	4.0	6.7
FCA-10-1	- 0.002	6.9	0.8	- 2.1	- 2.1
BFS-67-1	- 0.222	8.8	5.0	3.3	1.9
BFS-69-1	- 0.109	8.8	4.0	2.0	1.0

Table 4.2. Comparison of experimental and calculated FRR data (f_{237} / f_{235}).

Assembly	Experiment	(calculation / experiment - 1) × 100			
		ABBN-90	ENDF/B-VI	JEF-2	JENDL-3.2
FCA-9-1	0.209	1.0	- 1.9	-2.9	1.0
FCA-9-2	0.344	- 1.2	- 3.8	- 4.7	- 1.5
FCA-9-3	0.437	- 2.1	- 4.6	- 5.5	- 2.5
FCA-9-4	0.409	4.2	0.7	0.0	2.9
FCA-9-5	0.494	2.4	- 0.9	-1.4	1.2
FCA-9-6	0.599	- 0.2	- 3.0	- 3.8	- 1.3
FCA-9-7	0.429	5.4	1.6	0.9	3.7
FCA-10-1	0.357	- 2.2	- 5.3	- 5.9	- 3.1
BFS-67-1	0.242	2.5	- 0.4	-1.2	2.5
BFS-69-1	0.322	0.6	- 2.2	-3.5	0.5

REFERENCES

1. T. Mukaiyama et. all, Actinide Integral Measurements on FCA for Evaluating and Improving Their Cross Section Data, 1985, NEACRP - A - 684.
2. S.P.Belov, J.C.Cabrillat Investigation of MA Transmutation Problem In Benchmark Experiments at BFS Facility with Neptunium in Fuel Composition.
3. Abagyan L.R. et al. Group constants for reactor and shielding calculations (Russian). M., Energoatomizdat, 1981.
4. Dulin V.A. Sov. J. Atomic Energy, 1989, v.66, is. 2, p. 79.