Verification of Neutron Data for Main Reactor Materials from RUSFOND Library based on Integral Experiments

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Abstract. In this work the modern state of the library of evaluated nuclear data files RUSFOND for the main reactor materials, U235, U238, Pu239, Fe, Cr, Ni, Na, Pb, etc., is given. Calculations are performed and comparison with experimental data is done for the following characteristics: (i) Removal cross-sections under the threshold of fission of U-238 etc... (ii) Average cross-sections with different standard neutron fission spectra; (iii) Criticality of fast uranium and plutonium systems. Calculations are performed using continuous energy cross-sections and a Monte-Carlo code.

1 Introduction

National neutron data libraries were recently modified in USA, Europe Japan, China and Russia. These data libraries are now being tested for further use in applications. The neutron data were mainly tested using criticality calculations for the benchmarks selected by a special procedure from the ICSBEP Handbook. Other tests included 1) calculations of average cross sections on standard and some specially selected spectra, and 2) calculations of the cross sections of neutron removal under the thresholds of detector reactions such as (n,fis)U-238, (n,fis)Np-237, (n,p)Al-27, etc. These results are presented in this work.

2 Neutron data libraries

The following neutron data libraries were used: the Russian library ROSFOND presented in 2008 and corrected in 2010 [1]; the Japanese library JENDL-4.0 presented in 2011 [2]; the US library ENDF/B-VII.1 presented in 2011 [3]; the European library JEFF-3.1.1 presented in 2009 [4]. The main reactor nuclides were selected as follows:

- Fuel materials – U-235, U-238, Pu-239;
- Structural materials – Fe, Cr, Ni;
- Coolant materials – Na, Pb.

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3 Benchmarks used

The Mughabghab evaluation of resonance integrals (the spectrum of $\sim 1/E$) was used as a reference [5]. Average cross sections calculated on the U-235 and Cf-252 fission spectra as well as removal cross sections from the EXFOR Library were taken as reference values [6]. The input data and corresponding benchmark criticalities for selected benchmarks were taken from ICSBEP Handbook [7]. The criteria for selecting benchmarks were 1) simplicity of benchmark models, 2) representativeness with respect to the H/fuel ratio (from low to high values), 3) description completeness, and others [8]. 62 models with the HEU type fuel, 49 models with the LEU type fuel, and 102 models with the Pu fuel were selected. Neutron data for U-235, U-238 and Pu-239 were preliminary tested in criticality calculations of the Pruvost test models [9].

4 Results

The comparison of the results on resonance integrals (the spectrum of $\sim 1/E$) for Na and Pb are presented in Table 1, and for U-235, U-238 and Pu-239 in Table 2. In Tables 1-2 the following notation is used: “LMFBR” stands for the neutron spectrum of a typical sodium-cooled fast breeder reactor, “U235” stands for the fission spectrum of U-235. 30-group cross sections (the energy structure accepted in the ICSBEP Handbook is applied) were used in verification. They were prepared by NJOY [10]. The experimental values of the removal cross sections presented in Table 3 were taken from Ref. [11]. The comparison of the results of criticality calculations of the Pruvost test models was done on the next stage of verification. The calculations were performed by MCNP5 [12]. Data for generating input files for MCNP were taken from Ref. [9]. The dependence of the calculation values of criticality and corresponding C/E ratio values on the content of fuel nuclides is presented in Fig.1 for the case of homogeneous mixture of HEU and water with water reflector, and for the case of homogeneous mixture of PU and water with water reflector. The next stage of verification was done on the base of criticality calculations for the benchmark set selected from the ICSBEP Handbook. The C/E ratio for the sodium benchmark (MCF004) is presented in Fig. 2. Similar results for the set of the lead benchmarks are presented in Fig. 2 both. The C/E ratios for the HEU type benchmarks are presented in Fig. 3, for the LEU type benchmarks in Fig. 4, and for the PU type benchmarks in Fig. 5. Criticality calculations were performed by MCNP5. The reference (experimental) values were taken from the ICSBEP Handbook. For convenience, all final results are collected in Table 4.

Table 1. The comparison of the average cross sections for coolant nuclide.

Na	<i>capture</i>	<i>total</i>	<i>inel</i>	Pb	<i>capture</i>	<i>total</i>	<i>inel</i>
$\phi(LMFBR)$				$\phi(LMFBR)$			
RF2010	1.000	1.000	1.000	RF2010	1.000	1.000	1.000
E-VII.1	1.000	1.000	1.000	E-VII.1	1.000	1.000	0.992
JEFF-3.1.2	0.931	1.037	0.814	JEFF-3.1.2	1.000	1.000	1.000
JENDL-4.0	0.946	1.029	1.083	JENDL-4.0	1.104	1.019	1.005
$\phi(U-235)$				$\phi(U-235)$			
RF2010	1.000	1.000	1.000	RF2010	1.000	1.000	1.000
E-VII.1	1.001	1.000	1.000	E-VII.1	1.001	1.000	0.990
JEFF-3.1.2	0.977	1.062	0.732	JEFF-3.1.2	1.000	1.000	1.000
JENDL-4.0	1.039	1.001	1.016	JENDL-4.0	1.306	0.991	1.006

Table 2. The comparison of the average cross sections for fission nuclide.

U-235	<i>capture</i>	<i>inel</i>	<i>fission</i>	<i>nu-bar</i>	U-238	<i>capture</i>	<i>inel</i>	<i>fission</i>	<i>nu-bar</i>
<i>$\phi(LMFBR)$</i>					<i>$\phi(LMFBR)$</i>				
RF2010	1.000	1.000	1.000	1.000	RF2010	1.000	1.000	1.000	1.000
E-VII.1	1.000	1.008	0.994	1.001	E-VII.1	1.000	1.052	1.000	0.993
JEFF-3.1.2	1.003	0.999	0.997	1.000	JEFF-3.1.2	1.001	0.952	0.998	1.001
JENDL-4.0	0.956	0.998	0.984	1.000	JENDL-4.0	0.999	0.990	0.999	0.994
<i>EXP $\phi(U-235)$</i>			1.000		<i>EXP $\phi(U-235)$</i>			1.000	
RF2010	1.000	1.000	1.006	1.000	RF2010	1.000	1.000	0.985	1.000
E-VII.1	1.001	1.023	1.003	1.002	E-VII.1	1.000	1.013	0.985	0.992
JEFF-3.1.2	1.000	0.999	0.998	1.000	JEFF-3.1.2	0.988	1.028	0.980	1.000
JENDL-4.0	0.945	0.960	0.998	1.003	JENDL-4.0	1.011	1.000	0.981	0.994
<i>EXP $\phi(Cf-252)$</i>			1.000		<i>EXP $\phi(Cf-252)$</i>			1.000	
RF2010	1.000	1.000	0.994	1.000	RF2010	1.000	1.000	0.947	1.000
E-VII.1	1.000	1.024	0.991	1.001	E-VII.1	1.000	1.013	0.947	0.992
JEFF-3.1.2	1.000	0.999	0.986	1.000	JEFF-3.1.2	0.988	1.030	0.941	1.000
JENDL-4.0	0.944	0.961	0.986	1.003	JENDL-4.0	1.010	1.001	0.942	0.994
Pu-239	<i>capture</i>	<i>inel</i>	<i>fission</i>	<i>nu-bar</i>					
<i>$\phi(LMFBR)$</i>									
RF2010	1.000	1.000	1.000	1.000					
E-VII.1	0.984	1.341	1.006	1.001					
JEFF-3.1.2	1.000	1.000	1.000	1.000					
JENDL-4.0	0.979	1.249	1.009	1.003					
<i>EXP $\phi(U-235)$</i>			1.000						
RF2010	1.000	1.000	0.976	1.000					
E-VII.1	0.755	1.185	0.975	1.001					
JEFF-3.1.2	1.000	1.000	0.976	1.000					
JENDL-4.0	0.987	1.174	0.979	0.997					
<i>EXP $\phi(Cf-252)$</i>			1.000						
RF2010	1.000	1.000	0.978	1.000					
ENDF/B-VII.1	0.759	1.182	0.977	1.001					
JEFF-3.1.2	1.000	1.000	0.978	1.000					
JENDL-4.0	0.986	1.171	0.981	0.997					

Table 3. The comparison of the removal cross sections.

Na	<i>(n,f)Np237</i>	<i>(n,f)U238</i>	<i>(n,p)Al27</i>	Pb	<i>(n,f)Np237</i>	<i>(n,f)U238</i>	<i>(n,p)Al27</i>
EXP		1.000		EXP	1.000	1.000	1.000
RF2010	1.000	0.831	1.000	RF2010	1.300	1.182	1.060
E-VII.1	1.000	0.831	0.999	E-VII.1	1.251	1.145	1.036
JEFF-3.1.2	1.005	0.720	0.732	JEFF-3.1.2	1.300	1.182	1.060
JENDL-4.0	1.113	0.851	0.999	JENDL-4.0	1.298	1.188	1.067
U-235	<i>(n,f)Np237</i>	<i>(n,f)U238</i>	<i>(n,p)Al27</i>	U-238	<i>(n,f)Np237</i>	<i>(n,f)U238</i>	<i>(n,p)Al27</i>
EXP		1.000		EXP		1.000	
RF2010	1.000	1.151	1.000	RF2010	1.000	1.007	1.000
E-VII.1	1.009	1.180	1.058	E-VII.1	1.002	1.007	1.000
JEFF-3.1.2	0.989	1.139	0.992	JEFF-3.1.2	1.017	1.058	1.108
JENDL-4.0	0.975	1.169	1.045	JENDL-4.0	0.963	0.994	1.072
Pu-239	<i>(n,f)Np237</i>	<i>(n,f)U238</i>	<i>(n,p)Al27</i>				
EXP		1.000					
RF2010	1.000	0.819	1.000				
E-VII.1	1.196	0.944	1.087				
JEFF-3.1.2	1.000	0.819	1.000				
JENDL-4.0	1.245	1.001	1.122				

Table 4. The average benchmark uncertainties and the results for C/E-1 for benchmark set (in pcm).

Benchmark type	Number of benchmarks	Average benchmark uncertainty	RF2010	E-VII.1	JEFF-3.1.1	JENDL-4.0
<i>Cr capture</i>	1	180	904	766	874	1121
<i>Fe capture</i>	6	44	133	222	252	114
<i>Fe reflector*</i>	23	26	-39	2	-26	-25
<i>N capture</i>	1	40	-118	-64	-115	-153
<i>N reflector</i>	6	26	2	30	1	4
<i>Na</i>	1	15	-23	29	41	57
<i>Pb</i>	10	20	6	-46	-7	-3
<i>HEU fast</i>	7	21	-1	4	-30	-12
<i>HEU thermal</i>	55	39	-11	-6	-6	4
<i>LEU fast</i>	14	35	1	-20	-51	-71
<i>LEU thermal</i>	35	26	9	6	11	-7
<i>PU fast</i>	8	21	-14	4	0	-11
<i>PU thermal</i>	94	44	34	44	15	45

* PMF42 (cases 3-15) and PMF45 (cases 2-4) are not included.

5 Discussion

More information on Cr, Fe and Ni is presented in the next poster. The impact of the choice of Na evaluation is presented in Fig. 2. It is seen that using neutron data for Na from RF2010 instead of JEFF-3.1.1 leads to the bias in calculation criticality by $\sim 0.5\%$. The best agreement for the lead benchmarks is observed when using the JENDL-4.0 library (see Fig. 2). Calculation results for thermal systems (HEU, LEU and Pu) based on different neutron libraries are in better agreement with each other than with corresponding benchmark values (see Fig.3-5). For the set of the fast benchmarks discrepancies between calculation and reference values can be, in some cases, as much as 1% and even greater (see Fig 6-7).

Fig. 1: Pruvost curves for U-235 and Pu-239

Fig. 2: Na and Pb benchmarks.

Fig. 3: Thermal HEU benchmarks.

Fig. 4: Thermal LEU benchmarks.

Fig. 5: Thermal Pu benchmarks.

Fig. 6: Fast HEU and LEU benchmarks.

Fig. 7: Fast Pu benchmarks.

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