International Atomic Energy Agency

INDC(BZL)-29
Distr.: G+Spec.

INDC INTERNATIONAL NUCLEAR DATA COMMITTEE

VALIDATION OF ACTINIDE NUCLEAR DATA FROM ENDF/B-V, INDL/A-83 AND JENDL-2

R. Paviotti Corcuera, Marisa de Moraes

Instituto de Estudos Avancados
Centro Técnico Aeroespacial
12225 Sao José dos Campos, SP, Brazil

1987

IAEA NUCLEAR DATA SECTION, WAGRAMERSTRASSE 5, A-1400 VIENNA

INDC(BZL)-29
Distr.: G+Spec.

VALIDATION OF ACTINIDE NUCLEAR DATA FROM ENDF/B-V, INDL/A-83 AND JENDL-2

R. Paviotti Corcuera, Marisa de Moraes

Instituto de Estudos Avancados
Centro Técnico Aeroespacial
12225 Sao José dos Campos, SP, Brazil

1987

Printed by the IAEA in Austria
November 1988

88-05670

VALIDATION OF ACTINIDE NUCLEAR DATA FROM ENDF/B-V, INDL/A-83 AND JENDL-2 *

R. Paviotti Corcuera and Marisa de Moraes
INSTITUTO DE ESTUDOS AVANÇADOS
CENTRO TÉCNICO AEROESPACIAL
12225 SÃO JOSÉ DOS CAMPOS, SP, BRASIL

ABSTRACT

Resonance integrals and fission spectrum averaged cross sections are calculated for the actinides of ENDF/B-V, INDL/A-83 and JENDL-2. The results are compared with each other and with experimental data when available. The experimental data are scarce and there exist large differences among data from different libraries.

Introduction

It is the aim of this study to infer the global accuracy of evaluated actinide nuclear data through integral testing by considering 1/E weighting spectrum and fission spectrum averaged cross sections.

The 1/E function will give strong weight to the resonance region and the fission spectrum will almost eliminate the resonance region and will make appear the regions at high energies ($E > 0.5$ Mev).

For these two types of weighting functions three evaluations were considered and comparisons were done between themselves and with experimental data when available. The evaluations considered were ENDF/B-V Actinides, INDL/A-83 and JENDL-2 (Rev. 1).

---

* Work performed under research contract no. 3692/R2/RB with the International Atomic Energy Agency.

Infinite Dilution Resonance Integral

The infinite dilution resonance integral is defined by

$$I_i = \int_{E_2}^{E_1} \sigma_i(E) \frac{dE}{E} .$$

For the present calculation E_1 = 20 Mev and E_2 = 0.5 ev. The microscopic cross sections $\sigma_i(E)$ considered are fission and capture.

The results obtained for ENDF/B-V, INDL/A-83 and JENDL-2 are shown on table 1. On the same table an experimental value is given, this value was obtained by calculating the mean of all the experimental data obtained from the literature (Gryntakis and Kim, 1982 and EXFOR library). each value being weighted by the inverse of its experimental error (Corcuera et al, 1985; Corcuera and Trkov, 1987).

For completeness of this study, "recommended values" are also reported (Gryntakis et al 1984). It should be pointed out however that evaluated values were also introduced to calculate the mean given as a "recommended value". In the present study we have preferred to compare the evaluated values to purely experimental results.

On table 1 the percentual difference between evaluated and experimental values is also shown.

Fission Spectrum Averaged Cross Sections

The fission spectrum averaged cross sections were calculated as

$$I = \int_0^{\infty} \chi(E)\sigma_i(E)dE ,$$

where $\chi(E)$ is the fission spectrum and $\sigma_i(E)$ can be the fission cross section or the capture cross section.

In this study two spectrum were used, the ^{235}U and the ^{252}Cf, both of them in two representations, the one recommended by the

Table 1 - Infinite dilution resonance integral (barns). Comparison with experimental data

	CAPTURE RESONANCE INTEGRAL								FISSION RESONANCE INTEGRAL							
NUCLIDE	EXP. VALUE	RECOMENDED VALUE	ENDF/B-V	%DIF.*	INDL/A-83	% DIF.*	JENDL-2	% DIF.*	EXP. VALUE	RECOMENDED VALUE	ENDF/B-V	%DIF.*	INDL/A-83	%DIF.*	JENDL-2	%DIF.*
^{232}Th	79.4 ± 0.8	82.3 ± 2.4			80.32	1.1	79.91	0.6		0.0746 ± 0.016			0.627		0.6364	
^{233}Pa	817.0 ± 17.9	865 ± 35	857.3	4.9	882.1	8.0	779.2	4.6			2.947		2.965		4.682	
^{233}U	143.4 ± 4.4	138.1 ± 4.6	.				138.65	3.3	773.9 ± 13.4	783.4 ± 7.8					771.3	0.3
^{234}U		678 ± 38	660.5				609.0			5.96	6.542				6.438	
^{235}U	140.2 ± 2.4	141.8 ± 4.2			143.6	2.4	153.42	9.4	280.1 ± 3.2	276.3 ± 2.8			276.8	1.2	275.3	1.8
^{236}U	353.6 ± 9.2	358 ± 8	347.3	1.8			347.1	1.8		2	7.772				7.61	
^{238}U	272.5 ± 2.9	276.3 ± 2.7					279.0	2.4		0.0013 ± 0.0002					2.053	
^{237}Np		821.5 ± 58.0	662.6		654.1		662.4			6.5 ± 1.2	6.87		5.832		6.257	
^{238}Pu	3276 ± 229	154 ± 9	153.8	95.3	143.6	95.6	156.3	95.2	32 ± 5	24.2 ± 2.7	30.80	3.8	31.54	1.4	32.43	1.3
^{239}Pu	221 ± 11	191 ± 16			185.6	16.0	191.1	13.5	323.8 ± 6.5	312.2 ± 8.2			305.6	5.6	294.71	9.0
^{240}Pu	9579 ± 367	8460 ± 305			8422.7	12.1	8848.0	11.8		5			9.694		10.08	
^{241}Pu		161 ± 13			140.2		186.9		532.6 ± 13.6	558 ± 18			565.2	6.1	590.10	10.8
^{242}Pu	1275.6 ± 20.0	1131 ± 57	1273.7	0.2	1169.8	8.3	1117.0	12.4		5	5.552		23.4		6.35	
^{241}Am	1119.2 ± 31.5	1330 ± 117	1424.3	27.3	1439.4	28.6	1298.9	16.1			13.438		13.767		14.69	
$^{242}Am_g$			72.51				391.10			300	623.5				1259.12	
$^{242}Am_m$		1100 ± 500	289.8				206.9		2260 ± 200	2260 ± 200	1903.3	15.8			1528.5	32.4
^{243}Am	2259.3 ± 42.6	2200 ± 15	1822.8	19.3	1815.0	19.7	1817.6	19.6	12.0 ± 0.8	13 ± 2.5	6.152	48.7	5.934	50.6	11.37	5.3
^{242}Cm	150 ± 40	156 ± 35	111.73	25.5	115.9	22.7	116.2	22.5		33	6.25		11.62		11.08	
^{243}Cm	215.7 ± 20.3	214 ± 17	250.9	16.3	294.0	36.3	298.7	38.5	1560. ± 98	1527 ± 142	1966.1	26.0	1876.0	20.3	1812.5	16.3
^{244}Cm	642.6 ± 29.4	632.6 ± 32	594.3	7.5			593.5	7.6	15.1 ± 0.6	13.4 ± 1.5	18.72	24			18.38	21.7
^{245}Cm	104.3 ± 8.0	101 ± 8	109.1	4.6	116.7	11.9	107.8	3.4	878.4 ± 46.8	805 ± 80	840.5	4.3	823.63	6.2	799.31	9.0
^{246}Cm	120.5 ± 6.3	121.3 ± 7.5	104.03	13.7	111.0	7.9			10.4 ± 0.4	11.3 ± 1.2	10.42	0.3	6.936	33.3		
^{247}Cm	800 ± 400	650 ± 250	492.5	38.4	495.6	38.1			806.4 ± 37.4	754 ± 60	749.9	7.0	663.6	17.7		

* These values correspond to |evaluated-Exp|/Exp in % for the respective library.

National Bureau of standards (NBS) as given in the IRDF-85 library and the Madland-Nix (1982) representation.

Tables 2 and 3 show the fission spectrum averaged fission and capture cross sections weighted by the ^{252}Cf and the ^{235}U fission spectrum recommended by the NBS. On these tables the maximum percentual difference between evaluations is also shown.

The Madland-Nix representation was used to weight the cross section and it was found that it makes lower the capture cross section by 7% to 13% and higher the fission cross section by 5% to 10% (Corcuera et al, 1986).

The comparison with experimental data when available is on table 4 for both types of representation of the fission spectrum.

In order to verify the calculation procedures two independent ways were used, the NJOY system (MacFarlane et al, 1982) and the LINEAR-RECENT-GROUPIE system (Cullen D.E., 1978-1980).

The results obtained with both procedures were within 1% (Corcuera et al., 1985, 1986).

Concluding Remarks

For many actinides there are no experimental data, making it very difficult to validate the evaluated data.

We can summarize our conclusions as follows:

a) Comparison with experimental data (Table 1)

a.1) Fission resonance integral

There are no reliable experimental data for ^{232}Th, ^{233}Pa, ^{234}U, ^{236}U, ^{238}U, ^{237}Np, ^{240}Pu, ^{242}Pu, ^{241}Am, $^{242}Am_g$ and ^{242}Cm.

^{233}U : JENDL-2 agrees well with experimental data

^{235}U : JENDL-2 and INDL/A-83 agree well with experimental data

Table 2 - ^{252}Cf fission spectrum averaged cross sections (barns)

	fission				capture			
NUCLIDE	ENDF/B-V	INDL/A-83	JENDL-2	*	ENDF/B-V	INDL/A-83	JENDL-2	*
^{232}Th		0.07586	0.08162	7		0.09501	0.08348	14
^{233}Pa	0.4798	0.6083	0.9964	>100	0.1786	0.1565	0.1178	52
^{233}U			1.885				0.08034	
^{234}U	1.232		1.208	2	0.1705		0.1031	65
^{235}U		1.237	1.248	1		0.08844	0.1274	23
^{236}U	0.5992		0.6056	1	0.1702		0.1377	24
^{238}U			0.3233				0.06495	
^{237}Np	1.352	1.308	1.297	4	0.1629	0.1844	0.1676	13
^{238}Pu	1.983	1.983	2.017	2	0.1423	0.07334	0.3148	>100
^{239}Pu		1.802	1.818	1		0.05476	0.05901	11
^{240}Pu		1.345	1.367	2		0.08547	0.09012	6
^{241}Pu		1.621	1.621	1		0.06649	0.1455	>100
^{242}Pu	1.129	1.196	1.138	2	0.07040	0.07115	0.08512	21
^{241}Am	1.474	1.349	1.519	13	0.2542	0.3028	0.2904	19
$^{242}Am_g$	0.001502		1.759	>100	0.002470		0.2020	>100
$^{242}Am_m$	2.214		1.866	18	0.01868		0.1041	>100
^{243}Am	1.205	1.124	1.284	14	0.1369	0.2350	0.1884	>100
^{242}Cm	1.028	1.664	1.798	75	0.02408	0.02963	0.08468	>100
^{243}Cm	2.073	2.167	2.372	14	0.01477	0.01696	0.03328	>100
^{244}Cm	1.614		1.554	4	0.1193		0.1191	0.2
^{245}Cm	1.829	1.980	1.878	5	0.04064	0.04658	0.04413	14
^{246}Cm	1.386	1.342		3	0.04219	0.02284		85
^{247}Cm	2.065	2.265		10	0.04056	0.05013		24

* This column represents the maximum percentual difference between evaluations.

Table 3 - ^{235}U fission spectrum averaged cross sections (barns)

	fission				capture			
NUCLIDE	ENDF/B-V	INDL/A-83	JENDL-2	*	ENDF/B-V	INDL/A-83	JENDL-2	*
^{232}Th		0.07274	0.07846	8		0.09734	0.08575	14
^{233}Pa	0.4680	0.6037	0.9906	>100	0.1839	0.1619	0.1222	50
^{233}U			1.888				0.08233	
^{234}U	1.226		1.199	2	0.1745		0.1060	65
^{235}U		1.238	1.248	1		0.09093	0.1310	31
^{236}U	0.5899		0.5959	1	0.1742		0.1415	23
^{238}U			0.3148				0.06675	
^{237}Np	1.347	1.303	1.293	4	0.1678	0.1891	0.1733	13
^{238}Pu	1.976	1.978	2.011	2	0.1456	0.07509	0.319	>100
^{239}Pu		1.802	1.818	1		0.05617	0.06075	12
^{240}Pu		1.340	1.364	2		0.08759	0.09274	6
^{241}Pu		1.624	1.624	1		0.06878	0.149	> 100
^{242}Pu	1.125	1.198	1.135	2	0.07256	0.07236	0.08776	20
^{241}Am	1.465	1.342	1.510	12	0.2602	0.3098	0.2981	19
$^{242}Am_g$	0.001929		1.756	>100	0.0003172		0.2067	>100
$^{242}Am_m$	2.217		1.866	19	0.01963		0.1065	>100
^{243}Am	1.193	1.116	1.274	14	0.1415	0.2419	0.1947	>100
^{242}Cm	1.006	1.662	1.798	78	0.02498	0.03085	0.08703	>100
^{243}Cm	2.081	2.170	2.379	14	0.0155	0.01760	0.03451	>100
^{244}Cm	1.607		1.554	3	0.1228		0.1222	0.5
^{245}Cm	1.831	1.983	1.887	5	0.04165	0.04786	0.04518	15
^{246}Cm	1.378	1.342		3	0.04329	0.02361		83
^{247}Cm	2.064	2.276		10	0.0416	0.05155		24

* This column represents the maximum percentual difference between evaluations.

Table 4: Fission spectrum averaged fission cross sections (barns). Comparison with experimental data.

^{252}Cf		NBS SPECTRUM			MADLAND-NIX SPECTRUM		
NUCLIDE	EXP.	* INDL/A-83	* JENDL-2	* ENDF/B-V	* INDL/A-83	* JENDL-2	* ENDF/B-V
^{232}Th	0.0883 ± 0.0024	14.1	7.6		5.1	1.9	
^{233}U	1.938 ± 0.020		2.7			3.3	
^{235}U	1.225 ± 0.009	1.0	1.9		1.1	2.0	
^{238}U	0.328 ± 0.003		1.4			7.6	
^{237}Np	1.409 ± 0.017	7.2	8.0	4.1	4.0	4.8	0.6
^{239}Pu	1.841 ± 0.017	2.1	1.3		1.5	0.7	
^{240}Pu	1.337 ± 0.032	0.6	2.2		4.3	5.8	
^{241}Pu	1.616 ± 0.080	0.3	0.3		0.1	0.1	
^{235}U		NBS SPECTRUM			MADLAND-NIX SPECTRUM		
NUCLIDE	EXP.	* INDL/A-83	* JENDL-2	* ENDF/B-V	* INDL/A-83	* JENDL-2	* ENDF/B-V
^{232}Th	0.0777 ± 0.0027	6.4	1.0		4.4	3.1	
^{238}U	0.308 ± 0.015		2.2			4.4	

* these values are $|\text{evaluated} - \text{Exp}|/\text{Exp}$ in % for the respective libraries.

^{238}Pu : ENDF/B-V, INDL/A-83 and JENDL-2 agree well with experimental data

^{239}Pu : INDL/A-83 agree well with experimental data and JENDL-2 show a difference of 9%

^{241}Pu : Good agreement for JENDL-2 and INDL/A-83

$^{242}Am_m$: Differences greater than 15% but within the experimental error

^{243}Am : Good agreement for JENDL-2 and differences of about 50% for the other evaluations

^{243}Cm : Differences greater than 15%

^{244}Cm : Differences greater than 20%

^{245}Cm : Good agreement for ENDF/B-V, INDL/A-83 and JENDL-2

^{246}Cm : Good agreement for ENDF/B-V and differences larger than 30% for INDL/A-83

^{247}Cm : Good agreement for ENDF/B-V and differences larger than 17% for INDL/A-83

a.2) Capture resonance integral

There are no reliable experimental data for ^{234}U, ^{237}Np, ^{241}Pu, $^{242}Am_g$, $^{242}Am_m$.

^{232}Th : Good agreement for INDL/A-83 and JENDL-2

^{233}Pa : Good agreement for ENDF/B-V, INDL/A-83 and JENDL-2

^{233}U : Good agreement for JENDL-2

^{235}U : Good agreement for INDL/A-83 and JENDL-2 is 9% higher

^{236}U : Good agreement for ENDF/B-V and JENDL-2

^{238}U : Good agreement for JENDL-2

^{238}Pu : The three evaluations are very discrepant (95%)

^{239}Pu : Differences greater than 10% for INDL/A-83 and JENDL-2

^{240}Pu : Differences greater than 10% for INDL/A-83 and JENDL-2

^{242}Pu : Good agreement for ENDF/B-V and differences greather than 8% for JENDL-2 and INDL/A-83

^{241}Am : Differences greater than 15% for ENDF/B-V, INDL/A-83 and JENDL-2

^{243}Am : Differences of about 20% for the three evaluations

^{242}Cm : Differences of about 20% for the three evaluations

^{243}Cm : Differences greater than 15% for the three evaluations

^{244}Cm : Good agreement for ENDF/B-V and JENDL-2

^{245}Cm : Good agreement for JENDL-2 and ENDF/B-V, differences greater than 10% for INDL/A-83

^{246}Cm : Good agreement for INDL/A-83, differences greater than 10% for ENDF/B-V

^{247}Cm : Differences of about 30% but within the experimental error.

a.3) ^{252}Cf fission spectrum averaged cross section

JENDL-2 shows good agreement following nuclides: ^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{237}Np, ^{239}Pu, ^{240}Pu, ^{241}Pu.

ENDF/B-V shows good agreement for ^{237}Np.

INDL/A-83 shows good agreement for ^{235}U, ^{237}Np, ^{239}Pu, ^{240}Pu, ^{241}Pu and a large difference for ^{232}Th when weighted by the NBS spectrum.

a.4) ^{235}U fission spectrum averaged cross section

JENDL-2 shows good agreement for ^{232}Th and ^{238}U.

INDL/A-83 shows good agreement for ^{232}Th.

For the nuclides not mentioned above either three are no experimental values or there are no evaluated values.

In the case of the very few experimental values available we can say that the use of the Madland-Nix representation seems to improve a little the results but to confirm this trend more experimental data would be needed.

b) No experimental data available, comparison between evaluations (table 1)

b.1) Fission resonance integral

^{232}Th : Good agreement

^{233}Pa : JENDL-2 is about 50% higher than the other evaluations

^{234}U : Good agreement between JENDL-2 and ENDF/B-V

^{236}U : Good agreement between JENDL-2 and ENDF/B-V

^{237}Np : INDL/A-83 is about 18% lower than the other evaluations

^{240}Pu : Reasonable agreement between evaluations

^{242}Pu : INDL/A-83 is about 300% higher than the other evaluations

^{241}Am : Good agreement

^{242}Am$_g$: Discrepancies larger than 100%

^{242}Cm : Discrepancies larger than 80%

b.2) Capture resonance integral

^{234}U : Reasonable agreement between evaluations

^{237}Np : Reasonable agreement between evaluations

^{241}Pu : Discrepancies greater than 30%

^{242}Am$_g$: Discrepancies greater than 100%

^{242}Am$_m$: Discrepancies greater than 40%

b.3) Fission spectrum averaged cross sections

On Table 2 and 3 the maximum percentual difference between evaluations has been indicated. We observe that there exist large differences between evaluations and that for the fission spectrum averaged capture cross section the discrepancies are larger.

For the fission spectrum averaged fission cross section there exist agreement within 5% for ^{234}U, ^{236}U, ^{242}Pu, ^{244}Cm, ^{245}Cm; and

^{246}Cm. For the nuclides not mentioned the discrepancies between evaluations are larger.

For the fission spectrum averaged capture cross section there is good agreement for ^{240}Pu and ^{244}Cm, but for the other nuclides the differences are larger, sometimes passing over 100%.

References

Corcuera R.P. and de Moraes M. (1985) Report IEAv-023/85 Validation and Benchmark Testing of Actinide Nuclear Data, Instituto de Estudos Avançados, Centro Técnico Aeroespacial - São José dos Campos, Brasil.

Corcuera R.P. and de Moraes M. (1986) Nota Técnica IEAv-012/86 "Comments on the Discrepancies Found in Running NJOY and Linear-Recent-Groupie for Some Actinides. Centro Técnico Aeroespacial - Brasil.

Corcuera R.P. and de Moraes M. (1986) Nota Técnica IEAv-011/86 "Comparison of Averaged Actinide Cross Sections Weighted by the Madland-Nix and the NBS Spectrum". Centro Técnico Aeroespacial - Brasil.

Corcuera R.P. and Trkov A. (1987) Intercomparison of Derived Integral Data from Evaluated Data Libraries of Actinides, to be published as an INDC report of the Nuclar Data Section IAEA.

Cullen D.E. (1978) "Program LINEAR (Version 79-1) "UCRL-50400, Vol. 17 Part A. USA.

Cullen D.E. (1979) "Program RECENT (Version 79-1) "UCRL-50400, Vol. 17, Part C. USA.

Cullen D.E. (1980) "Program GROUPIE (Version 79-1) "UCRL-50400, Vol. 17, Part D. USA.

ENDF/B-V Actinides (1984) Rev. 2. National Neutron Cross Section Center, BNL, Upton, N.Y. USA.

EXFOR Experimental Format for exchange of data. Data supplied by Nuclear Data Section IAEA (1984) Vienna.

Gryntakis E.M. and Kim J.I. (1982) "A Compilation of Resonance Integrals", Institut Fuer Radiochemie, Tu Munchen, 8046 Garching, FRG.

Gryntakis E.M. et al, (1984) "Thermal Neutron Cross Sections and Infinite Dilution Resonance Integrals", NDS - IAEA report to be published. NDS. IAEA Vienna.

INDL/A-83 (1983) IAEA Nuclear Data Library for Evaluated Neutron Reaction Data of Actinides. IAEA-NDS-12 Rev. 7.

IRDF-85 International Reactor Dosimetry File-Nuclear Data Section - IAEA-NDS-41 Rev. 1.

JENDL-2 (Rev. 1) Japanese Evaluated Nuclear Data Library, Version 2 Rev. 1. IAEA-NDS-18 Rev. 2.

Madland D.G and Nix J.R (1982) Nucl. Sci. Engng. 81, 213.

MacFarlane et al. (1982) Report LA-9303-M, Vols I and II (ENDF-324).