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VALIDATION AND BENCHMARK TESTING OF ACTINIDE NUCLEAR DATA*)

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November 1986

*) Work performed under Research Contract No. 3692/RI/RB with the International Atomic Energy Agency, Vienna.

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Contents:

page

Text

Tables

References:

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Abstract

Resonance integrals and fission spectrum averaged cross sections are calculated for the actinides of the libraries ENDF/B-IV, ENDF/B-V actinides, 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

Transactinium isotopes (Z > 89) are becoming more and more important in nuclear technology. They play important roles in the nuclear fuel cycles of both thermal and fast reactors and have found increasing areas of application in science and industry.

Actinide nuclear data are useful for the calculation of: poisoning and absorption effects, decay heat and gamma source terms, delayed neutron yields and spectra, Na-void and Doppler coefficients, breeding ratio and doubling time, neutron source terms and prompt neutron effects.

The objective of this study is to infer the accuracy of the actinide nuclear data through integral testing by considering $\frac{1}{E}$ spectrum averaged cross sections and fission spectrum averaged cross sections.

In both cases the integral cross sections of four different libraries ENDF/B-IV (Ref. 1) and V (Ref. 2), INDL/A-83 (Ref. 3) and JENDL-2 (Ref. 4) were compared with each other and with experimental values, when available.

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Infinite Dilution Resonance Integral

The infinite dilution resonance integral is defined by

$$I_i = \int_{E_2}^{E_1} \sigma_i(E) \frac{dE}{E} \qquad (1)$$

for a $\frac{1}{E}$ spectrum, E_1 and E_2 being the upper and lower limits of the $\frac{1}{E}$ flux. For the present calculations E_1 = 20 MeV and E_2 = 0.5 ev. The microscopic cross section $\sigma_i(E)$ can be the total cross section, the fission cross section or the capture cross section.

The infinite dilution resonance integral was calculated as defined by eq. (1) for the four libraries already mentioned.

The code used to process the libraries is the NJOY system (Ref. 5). The calculations were compared with the results obtained by processing the system LINEAR (Ref. 6) → RECENT (Ref. 7) → GROUPIE (Ref. 8) and with calculated published results (Ref. 9). The objective of this comparison was to verify if the calculated values of the resonance integral were independent of the codes used to calculate them. The values obtained are shown on Tables I to IV. We observe that the results are within 1% difference, except the cases that are detailed below where discrepancies exist:

- In ENDF/B-V the resonance integral data of Pu-238 (Table I) seem to be different from data of Ref. 9.

- In ENDF/B-IV the fission resonance integral for Am-241 (observing the other evaluations it seems that something is wrong when using LINEAR → RECENT → GROUPIE).

For the cases mentioned above no format error were detected by the checking codes and we could not find any processing error. We concluded that some parameter is being interpreted differently by NJOY and by LINEAR → RECENT → GROUPIE. In the following tables we will use the result calculated by NJOY.

For the MAT identification number of the nuclides we refer the reader to Table V.

The resonance integrals obtained using the data from the ENDF/B-IV and V, INDL/A-83 and JENDL-2 libraries are shown in Tables VI to VIII.

We observe that:

a) Total resonance integral varies within 10% for the different evaluations except for:

Am-242g with differences greater than 100%; Am-241, Pa-233, Am-243, Am-242m, Pu-238, Pu-242 greater than 10%.

b) Capture resonance integral varies within 10% for the different evaluations except for:

Am-242g, with differences larger than 100%; Pa-233, U-235, Pu-241, Pu-242, Am-241, Am-242m, Am-243, Cm-243, with deviations larger than 10%.

c) Fission resonance integral varies within 10% with the different evaluations except for:

U-236, Pu-242, Am-242g, Am-243, Cm-244, with differences larger than 100%; Pa-233, U-234, Np-237, Am-242m, Cm-242, Cm-247, Cm-246, which show deviations larger than 10%.

For the experimental values, the compilation of E.M. Gryntakis and J.I. Kim (Ref. 10) and an EXFOR (Ref. 11) retrieval were consulted. The experimental values shown on Tables IX and X were obtained by calculating the mean of the experimental data encountered in the literature. The mean was calculated using the least squares technique where each experimental value was weighted by the inverse of its experimental error. The complete data used to calculate the mean experimental value are on Appendix I. All experimental data without indication of error were ignored in the calculation of this value. The recommended value taken from (Ref. 12) is also given on the third column. The other columns

give the deviation from the experimental value for different evaluations (ENDF/B-V, JENDL-2 and INDL/A-83).

We observe that experimental data are scarce and that among the available data a very small proportion agrees within 10%.

Fission Spectrum Averaged Cross Section

The fission spectrum averaged cross section were calculated as

$$I = \int_0^\infty \chi(E)\ \sigma_i(E)\ dE \qquad (2)$$

where $\chi(E)$ is the fission spectrum and $\sigma_i(E)$ can be the total cross section, the fission cross section, or the capture cross section. This way of weighting the cross section practically eliminates most of the resonance range. The limits used for the integral are 20 MeV and 0.1×10^{-4} eV which is the energy range for the libraries.

The spectrum used for U-235 and for Cf-252 are those recommended by the National Bureau of Standards as given in the IRDF-82 library (Ref. 13).

On Table XI are shown the results of processing LINEAR → RECENT → GROUPIE and NJOY. Again the data for Pu-240 and Pu-241 seems to have problems which we could not identify. Comparing with the other evaluations the NJOY results seems better.

The calculated values for the four libraries already mentioned are shown in Tables XII through XVII.

At high energies the situation seems to be better than in the range of resonances. For the total cross section the discrepancies are lower, about half of the values being within approximately 5%.

The data are more discrepant in the following cases:

- Total cross section: discrepancies of about 30% for Pu-238, Am-242g.

- Fission cross section: discrepancies of more than 20% for Pa-233, Am-242g, Am-243, Cm-242.

- Capture cross section: most data have discrepancies greater than 20%.

On Table XVIII the calculated values are compared with the experimental values. The complete data collection used to calculate the mean experimental data are on Appendix I. To calculate this mean, all experimental data without indication of error were ignored. For actinides with Z > 94 most data agree within 5% except for Th-232 from INDL/A-83.

Concluding Remarks

There are no experimental data for several actinides (see blank spaces on tables with experimental data), making it impossible to confirm the quality of the evaluated data.

We can summarize our conclusions as follows:

a) Comparison with experimental data

a1. Fission resonance integral

There are no reliable experimental data for Th-232, Pa-233, U-234, U-236, U-238, Np-237, Pu-240, Pu-242, Am-241, Am-242g and Cm-242.

U-233 : JENDL-2 agrees well with experimental data

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

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

Pu-239 : INDL/A-83 and JENDL-2 agree well with experimental data

Pu-241 : JENDL-2 and INDL/A-83 agree well with experimental data

Am-242m : Differences greater than 10%

Am-243 : Good agreement for JENDL-2 but disagreement with INDL/A-83

Cm-243 : Differences greater than 15%

Cm-244 : Differences greater than 20%

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

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

Cm-247 : Good agreement for ENDF/B-V and differences greater than 17% for INDL/A-83

a2. Capture resonance integral

There are no reliable experimental data for U-234, Np-237, Pu-241, Am-242g, Am-242m.

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

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

U-233 : Good agreement for JENDL-2

U-235 : Good agreement for JENDL-2 and INDL/A-83

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

U-238 : Good agreement for JENDL-2

Pu-238 : The three evaluations are very discrepant (95%)

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

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

Pu-242 : Good agreement for ENDF/B-V and INDL/A-83, differences greater than 10% for JENDL-2

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

Am-243 : Differences of about 20% for the three evaluations

Cm-242 : Differences of about 20% for the three evaluations

Cm-243 : Differences greater than 15% for the three evaluations

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

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

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

Cm-247 : Differences of about 30% but within the experimental error

a3. Cf-252 fission spectrum averaged cross section

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

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

INDL/A-83 shows good agreement for U-235, Np-237, Pu-239, Pu-240, Pu-241 and a large difference for Th-232.

a4. U-235 fission spectrum averaged cross sections

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

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

For the nuclides not mentioned in a3. of a4. either there are no experimental values or there are no evaluated values.

b) Comparison between evaluations

b1. Total resonance integral

Pu-238 : INDL/A-83 seems much lower than the other evaluations

Am-241 : JENDL-2 seems much lower than the other evaluations

Am-242g : Great difference between ENDF/B-V and JENDL-2

Am-242m : Difference greater than 20% between ENDF/B-V and JENDL-2

b2. Fission resonance integral

Pa-233 : JENDL-2 is about 50% greater than the other evaluations

Np-237 : INDL/A-83 is lower than all the other evaluations

Pa-242 : INDL/A-83 is about 300% higher than the other evaluations

Am-242g : JENDL-2 is about 150% higher than ENDF/B-V

Am-242m : JENDL-2 is about 20% lower than ENDF/B-V

Am-243 : JENDL-2 gives a very high value

Cm-242 : ENDF/B-V is about 80% lower than the other evaluations

Cm-246 : INDL/A-83 is about 50% lower than ENDF/B-V

b3. Capture resonance integral

Pu-241 : INDL/A-83 is about 30% lower than JENDL-2

Am-241 : JENDL-2 gives a very low value

Am-242g : There is a very large difference between ENDF/B-V and JENDL-2

Am-242m : JENDL-2 is about 40% lower than ENDF/B-V

Am-243 : A very low value is given in JENDL-2 and INDL/A-83

Cm-243 : ENDF/B-V is about 20% lower than the other evaluations

b4. Fission spectrum averaged total cross section

Tables XII and XV show a mean value of about 7.7 except for the following cases:

U-234 : from ENDF/B-V

Pu-238 : from JENDL-2

Am-242g : from ENDF/B-V

Am-242m : from ENDF/B-V

Cm-246 : from INDL/A-83

Cm-247 : from INDL/A-83

b5. Fission spectrum averaged fission cross section

The situation is not clear in the following cases:

Pa-233 : Very different values for all libraries

Am-242g : ENDF/B-V

Am-242m, Cm-242, Cm-243: The values for all libraries are very different

b6. Fission spectrum averaged capture cross section

In general the data are very discrepant (more than 20%) for most of the actinides.

It would be very interesting, if a more detailed analysis could be done, for the cases where large discrepancies exist between the two processing codes used in this work, (e.g. Am-241 in ENDF/B-IV).

	σ_{total}			$\sigma_{fission}$			$\sigma_{(n,\gamma)}$		
NUCLIDE	REF 9	NJOY	LINEAR RECENT GROUPIE	REF 9	NJOY	LINEAR RECENT GROUPIE	REF 9	NJOY	LINEAR RECENT GROUPIE
Th-232	341.4			0.6185			83.96		
Pa-233	1029.6	1030.8	1030.01	2.947	2.947	2.948	856.4	857.3	856.8
U-233	1068.8			756.0			136.55		
U-234	932.8	934.6	933.3	6.539	6.542	6.544	660.2	660.5	660.5
U-235	600.6			281.7			139.15		
U-236	646.0	648.1	646.3	7.768	7.772	7.772	347.0	347.3	347.1
U-238	637.2			2.0324			279.5		
Np-237	838.2	840.3	838.6	6.87	6.87	6.86	640.4	641.	640.7
Pu-238	481.7	456.1	453.6	33.07	30.80	31.3	162.29	153.8	152.1
Pu-239	677.3			303.5			193.93		
Pu-240	8788.			8.830			7971.		
Pu-241	973.9			588.4			196.89		
Pu-242	1668.3	1670.4	1669.3	5.568	5.552	5.567	1272.6	1273.7	1273.2
Am-241	1629.7	1631.3	1627.4	13.431	13.438	13.428	1423.5	1424.4	1421.04
Am-242g		899.62	896.40		623.5	621.4		72.51	72.03
Am-242m	2339.3	2364.4	2340.4	1883.4	1903.3	1884.2	286.3	289.8	286.4
Am-243	2035.3	2037.9	2036.2	6.151	6.152	6.154	1818.4	1821.1	1819.1
Cm-242		331.9	330.85		6.25	6.25		111.73	111.29
Cm-243	2384.7	2402.8	2385.7	1951.8	1966.1	1952.5	248.42	250.9	248.52
Cm-244	965.3	967.5	965.8	18.697	18.72	18.7	593.5	594.3	593.7
Cm-245	1105.6	1111.6	1111.6	837.0	833.5	833.4	108.63	117.58	117.55
Cm-246	316.3	317.2	316.4	10.419	10.42	10.424	103.8	104.03	103.9
Cm-247		1411.3	1410.2		749.9	751.10		492.5	492.14

Table I - Infinite dilution resonance integral using ENDF/B-V data done with three different codes (barns).

	σ_{total}		$\sigma_{fission}$		$\sigma_{(n,\gamma)}$	
NUCLIDE	NJOY	LINEAR RECENT GRUPIE	NJOY	LINEAR RECENT GROUPIE	NJOY	LINEAR RECENT GROUPIE
Th-232	313.2	312.6	0.5928	0.593	85.57	85.57
Pa-233	1030.8	1030.0	2.947	2.948	857.3	856.8
U-233	1068.5	1069.5	762.4	763.1	134.65	134.76
U-234	978.1	978.1	5.580	5.585	631.7	631.6
U-235	603.4	602.5	283.5	283.4	138.86	138.82
U-236	633.4	633.5	3.425	3.427	347.1	347.1
U-238	610.6	609.3	2.057	2.056	277.8	277.7
Np-237	840.1	838.4	6.848	6.834	640.96	640.7
Pu-238	436.5	434.5	30.89	31.4	144.81	143.13
Pu-239	678.8	678.1	303.8	303.6	194.16	194.1
Pu-240	9409.5	9406.8	9.544	9.537	8450.6	8448.6
Pu-241	894.8	893.4	586.6	586.6	125.7	125.68
Pu-242	1511.5	1510.5	5.839	5.837	1126.9	1126.6
Am-241	1847.6	1812.8	13.87	21.01	1640.7	1616.7
Am-243	1560.3	1559.5	4.365	4.364	1362.9	1362.3
Cm-244	1029.4	1029.3	44.88	44.892	593.8	593.4

Table II - Infinite dilution resonance integral with ENDF/B-IV data done with two different codes (barns).

	σ_{total}		$\sigma_{fission}$		$\sigma_{(n,\gamma)}$	
NUCLIDE	NJOY	LINEAR RECENT GROUPIE	NJOY	LINEAR RECENT GROUPIE	NJOY	LINEAR RECENT GROUPIE
Th-232	336.93	335.38	0.6267	0.6271	80.32	80.16
Pa-233	1070.62	1063.92	2.965	2.974	882.1	876.3
U-235	610.2	608.1	276.8	276.6	143.6	143.5
Np-237	849.49	850.2	5.832	5.822	654.1	654.2
Pu-238	404.57	404.78	31.54	31.53	143.6	143.5
Pu-239	665.2	662.9	304.5	306.4	182.5	182.5
Pu-240	9292.4	9300.3	9.359	9.352	8420.6	8430.0
Pu-241	905.5	895.5	565.2	564.6	140.2	140.2
Pu-242	1521.7	1519.6	23.34	23.30	1169.8	1170.1
Am-241	1625.9	1626.6	13.767	13.764	1439.4	1439.4
Am-243	2032.0	2029.3	5.934	5.926	1815.0	1813.2
Cm-242	328.39	327.65	11.62	11.60	115.9	115.2
Cm-243	2334.0	2328.5	1876.0	1872.9	294.0	293.2
Cm-245	1115.0	1109.6	823.63	821.11	116.7	115.8
Cm-246	334.2	334.7	6.936	6.947	111.0	110.7
Cm-247	1342.99	1337.52	663.6	661.52	495.6	493.9

Table III - Infinite dilution resonance integral with INDL/A-83 data done with the different codes (barns).

	σ_{total}		$\sigma_{fission}$		$\sigma_{(n,\gamma)}$	
NUCLIDE	NJOY	LINEAR RECENT GROUPIE	NJOY	LINEAR RECENT GROUPIE	NJOY	LINEAR RECENT GROUPIE
Th-232	335.3	334.1	0.6364	0.6365	79.91	79.90
Pa-233	968.1	968.5	4.682	4.673	779.2	778.61
U-233	1089.8	1088.7	771.32	771.29	138.65	138.59
U-234	949.2	949.9	6.438	6.437	609.0	609.0
U-235	613.3	614.0	275.3	278.75	153.42	153.32
U-236	646.5	646.8	7.61	7.61	347.1	347.01
U-238	633.3	652.4	2.053	1.262	279.0	280.23
Np-237	876.5	875.6	6.257	6.258	662.4	662.2
Pu-238	491.5	492.0	32.43	32.44	156.3	156.3
Pu-239	667.2	675.6	294.71	301.57	191.1	195.3
Pu-240	9410.3	9410.3	10.08	10.09	8448.0	8449.5
Pu-241	960.7	959.6	590.10	590.39	186.9	186.9
Pu-242	1478.3	1478.9	6.35	6.35	1117.0	1116.7
Am-241	1497.3	1496.1	14.69	14.69	1298.9	1298.5
Am-242g	1826.3	1834.8	1259.12	1265.45	391.10	392.59
Am-242m	1906.9	1907.2	1528.5	1528.6	206.9	206.9
Am-243	2043.1	2041.5	11.37	11.36	1817.6	1816.6
Cm-242	350.5	351.2	11.08	11.093	116.2	116.3
Cm-243	2294.6	2293.4	1812.5	1813.5	298.7	298.7
Cm-244	963.2	963.7	18.38	18.39	593.5	593.4
Cm-245	1084.2	1085.7	799.31	799.91	107.8	107.8

Table IV - Infinite dilution resonance integral with JENDL-2 data done with two different codes (barns).

NUCLIDE	ENDF/B-IV MAT NUMBER	ENDF/B-V MAT NUMBER	INDL/A-83 MAT NUMBER	JENDL-2 MAT NUMBER
Th-232	1296		9090	2903
Pa-233	1297	1391	9193	2911
U-233	1260			2921
U-234	1043	1394		2922
U-235	1261		9211	2923
U-236	1163	1396		2924
U-238	1262			2925
Np-237	1263	1337	9337	2931
Pu-238	1050	1338	9438	2942
Pu-239	1264		9421	2943
Pu-240	1265		9431	2944
Pu-241	1266		9441	2945
Pu-242	1161	1342	9450	2946
Am-241	1056	1361	9541	2951
Am242g		8542		2952
Am-242m		1369		2953
Am-243	1057	1363	9530	2954
Cm-242		8642	9662	2961
Cm-243		1343	9663	2962
Cm-244	1162	1344		2963
Cm-245		1345	9665	2964
Cm-246		1346	9666	
Cm-247			9647	

Table V - Identification number of the nuclides.

Nuclide	Origin	Reference	Mat number
Th-232	Romania	INDC (RUM)-10	9090
Pa-233	Romania	INDC (ROM)-12	9193
U-235	Minsk	INDC (CCP)-257	9211
Np-237	Cadarache	INDC (FR)-42	9337
Pu-238	Cadarache	INDC (FR)-57	9438
Pu-239	Minsk	INDC (CCP)-166	9421
Pu-240	Minsk	INDC (CCP)-215	9431
Pu-241	Minsk	INDC (CCP)-256	9441
Pu-242	Minsk	INDC (CCP)-256	9450
Am-241	Cadarache	79 Knoxville	9541
Am-243	Harwell	unpublished	9530
Cm-242	Bologna	INDC (ITY)-7	9662
Cm-243	Bologna	CNEN-RT/FI(81)23	9663
Cm-245	Bologna	CNEN-RT/FI(81)24	9665
Cm-246	Bologna	INDC (ITY)-8	9666
Cm-247	Bologna	INDC (ITY)-9	9647

Some of the nuclides in the INDL/A-83 library are evaluations taken from the JENDL-2 library. They were processed as being from JENDL-2. They are:

Nuclide	MAT number in INDL/A-83	MAT number in JENDL-2
U-235	9235	2923
U-238	9238	2925
Pu-239	9439	2943
Pu-240	9445	2944
Pu-241	9448	2945

Table V - (continuation) - Identification number of the nuclides

NUCLIDE	ENDF/B-V	ENDF/B-IV	INDL/A-83	JENDL-2	MAXIMUM % DIFFERENCE
Th-232		313.2	336.93	335.3	7.6
Pa-233	1030.8	1030.8	1070.6	968.1	10.6
U-233		1068.5		1089.8	2.0
U-234	934.6	978.1		949.2	4.7
U-235		603.4	610.2	613.3	1.7
U-236	648.1	633.4		646.5	2.3
U-238		610.6		633.3	3.7
Np-237	840.3	840.1	849.4	876.5	4.3
Pu-238	456.1	436.5	404.5	491.5	21.5
Pu-239		678.8	665.2	667.2	2.0
Pu-240		9409.5	9292.4	9410.3	1.3
Pu-241		894.8	905.5	960.7	7.4
Pu-242	1670.4	1511.5	1521.7	1478.3	13.0
Am-241	1631.3	1847.6	1625.9	1497.3	23.4
Am-242g	899.62			1826.3	103.0
Am-242m	2364.4			1906.9	24.0
Am-243	2037.9	1560.3	2032.0	2043.1	30.9
Cm-242	331.9		328.3	350.5	6.7
Cm-243	2402.8		2334.0	2294.6	4.7
Cm-244	967.5	1029.4		963.2	6.9
Cm-245	1111.6		1115.0	1084.2	2.8
Cm-246	317.2		334.2		5.4
Cm-247	1411.3		1342.9		5.1

Table VI - Total resonance integral (barns).

NUCLIDE	ENDF/B-V	ENDF/B-IV	INDL/A-83	JENDL-2	MAXIMUM % DIFFERENCE
Th-232		0.5928	0.6267	0.6364	7.4
Pa-233	2.947	2.947	2.965	4.682	58.9
U-233		762.4		771.3	1.2
U-234	6.542	5.580		6.438	17.2
U-235		283.5	276.8	275.3	3.0
U-236	7.772	3.425		7.61	126.9
U-238		2.057		2.053	0.0
Np-237	6.87	6.848	5.832	6.257	17.8
Pu-238	30.80	30.89	31.54	32.43	5.3
Pu-239		303.8	304.5	294.71	3.3
Pu-240		9.544	9.359	10.08	7.7
Pu-241		586.6	565.2	590.10	4.4
Pu-242	5.552	5.839	23.34	6.35	320.4
Am-241	13.438	13.87	13.767	14.69	9.3
Am-242g	623.5			1259.12	145.2
Am-242m	1903.3			1528.5	24.5
Am-243	6.152	4.365	5.934	11.37	160.5
Cm-242	6.25		11.62	11.08	85.9
Cm-243	1966.1		1876.0	1812.5	8.5
Cm-244	18.72	44.88		18.38	144.2
Cm-245	833.5		823.63	799.31	4.3
Cm-246	10.42		6.936		50.3
Cm-247	749.9		663.6		13.

Table VII - Fission resonance integral (barns).

NUCLIDE	ENDF/B-V	ENDF/B-IV	INDL/A-83	JENDL-2	MAXIMUM % DIFFERENCE
Th-232		85.57	80.32	79.91	7.1
Pa-233	857.3	857.3	882.1	779.2	13.2
U-233		134.65		138.65	3.0
U-234	660.5	631.7		609.0	8.5
U-235		138.86	143.6	153.42	10.5
U-236	347.3	347.1		347.1	0.0
U-238		277.8		279.0	0.4
Np-237	641.0	640.96	654.1	662.4	3.4
Pu-238	153.8	144.81	143.6	156.3	8.9
Pu-239		194.16	182.5	191.1	6.4
Pu-240		8450.6	8420.6	8448.0	0.4
Pu-241		125.7	140.2	186.9	48.7
Pu-242	1273.7	1126.9	1169.8	1117.0	14.0
Am-241	1424.4	1640.7	1439.4	1298.9	26.3
Am-242g	72.51			391.10	439.4
Am-242m	289.8			206.9	40.1
Am-243	1821.1	1362.9	1815.0	1817.6	33.6
Cm-242	111.73		115.9	116.2	4.0
Cm-243	250.9		294.0	298.7	19.1
Cm-244	594.3	593.8		593.5	0.1
Cm-245	117.58		116.7	107.8	9.1
Cm-246	104.03		111.0		6.7
Cm-247	492.5		495.6		0.6

Table VIII-Capture resonance integral (barns).

NUCLIDE	EXP. VALUE	RECOMMENDED VALUE	* ENDF/B-V	* INDL/A-83	* JENDL-2
Th-232		0.0746 ± 0.016			
Pa-233					
U-233	773.9 ± 13.4	783.4 ± 7.8			0.3
U-234		5.96			
U-235	280.1 ± 3.2	276.3 ± 2.8		1.2	1.7
U-236		2			
U-238		0.0013 ± 0.0002			
Np-237		6.5 ± 1.2			
Pu-238	32 ± 5	24.2 ± 2.7	3.8	1.4	1.3
Pu-239	323.8 ± 6.5	312.2 ± 8.2		6.0	9.0
Pu-240		5			
Pu-241	532.6 ± 13.6	558 ± 18		6.1	10.8
Pu-242		5			
Am-241					
Am-242g		300			
Am-242m	2260 ± 200	2260 ± 200	15.8		32.4
Am-243	12.0 ± 0.8	13 ± 2.5	48.7	50.6	5.3
Cm-242		33			
Cm-243	1560. ± 98	1527 ± 142	26.0	20.3	16.2
Cm-244	15.1 ± 0.6	13.4 ± 1.5	24.0		21.7
Cm-245	878.4 ± 46.8	805 ± 80	5.1	6.2	9.0
Cm-246	10.4 ± 0.4	11.3 ± 1.2	0.3	33.3	
Cm-247	806.4 ± 37.4	754 ± 60	7.0	17.7	

Table IX - Fission resonance integral comparison with experimental data (barns) (* these values are | Evaluated - Exp | / Exp in % for the respective libraries)

NUCLIDE	EXP. VALUE	RECOMMENDED VALUE	* ENDF/B-V	*INDL/A-83	*JENDL-2
Th-232	79.4 ± 0.8	82.3 ± 2.4		1.1	0.6
Pa-233	817.0 ± 17.9	865 ± 35	4.9	8.0	4.6
U-233	143.4 ± 4.4	138.1 ± 4.6			3.3
U-234		678 ± 38			
U-235	140.2 ± 2.4	141.8 ± 4.2		2.4	9.4
U-236	353.6 ± 9.2	358 ± 8	1.8		1.8
U-238	272.5 ± 2.9	276.3 ± 2.7			2.4
Np-237		821.5 ± 58.0			
Pu-238	3276 ± 229	154 ± 9	95.3	95.6	95.2
Pu-239	221 ± 11	191 ± 16		17.4	13.5
Pu-240	9579 ± 367	8460 ± 305		12.1	11.8
Pu-241		161 ± 13			
Pu-242	1275.6 ± 20.0	1131 ± 57	0.2	8.3	12.4
Am-241	1119.2 ± 31.5	1330 ± 117	27.3	28.6	16.1
Am-242g					
Am-242m		1100 ± 500			
Am-243	2259.3 ± 42.6	2200 ± 15	19.4	19.7	19.6
Cm-242	150 ± 40	156 ± 35	25.5	22.7	22.5
Cm-243	215.7 ± 20.3	214 ± 17	16.3	36.3	38.5
Cm-244	642.6 ± 29.4	632.6 ± 32	7.5		7.6
Cm-245	104.3 ± 8.0	101 ± 8	12.8	11.9	3.4
Cm-246	120.5 ± 6.3	121.3 ± 7.5	13.7	7.9	
Cm-247	800 ± 400	650 ± 250	38.4	38.1	

Table X - Capture resonance integral comparison with experimental data (barns) (* these values are |Evaluated - Exp|/Exp in % for the respective libraries).

	σ_{total}		$\sigma_{fission}$		$\sigma_{(n,\gamma)}$	
NUCLIDE	NJOY	LINEAR RECENT GROUPIE	NJOY	LINEAR RECENT GROUPIE	NJOY	LINEAR RECENT GROUPIE
Th-232	7.544	7.536	0.07586	0.07494	0.09501	0.09574
Pa-233	7.455	7.491	0.6083	0.6038	0.1565	0.1590
U-235	7.667	7.669	1.237	1.236	0.08844	0.08950
Np-237	7.548	7.561	1.308	1.298	0.1844	0.1881
Pu-238	7.790	7.803	1.983	1.975	0.07334	0.07441
Pu-239	7.717	7.704	1.795	1.795	0.05335	0.05420
Pu-240	7.818	7.783	1.341	1.332	0.08546	0.08654
Pu-241	7.990	8.006	1.621	1.622	0.06649	0.06715
Pu-242	8.041	8.024	1.196	1.191	0.07115	0.07166
Am-241	7.635	7.648	1.349	1.341	0.3028	0.3081
Am-243	7.663	7.676	1.124	1.116	0.2350	0.2386
Cm-242	8.056	8.075	1.664	1.660	0.02963	0.03032
Cm-243	7.955	7.976	2.167	2.170	0.01696	0.01728
Cm-245	8.100	8.128	1.980	1.977	0.04658	0.04729
Cm-246	8.742	8.791	1.342	1.338	0.02284	0.02333
Cm-247	8.556	8.592	2.265	2.269	0.05013	0.051113

Table XI - Cf-252 fission spectrum averaged cross sections (barns) done with two different codes (Data for INDL/A-83).

NUCLIDE	ENDF/B-V	ENDF/B-IV	INDL/A-83	JENDL-2	MAXIMUM % DIFFERENCE
Th-232		7.240	7.544	7.552	4.3
Pa-233	7.294	7.294	7.455	7.585	4.0
U-233		7.254		7.676	5.8
U-234	8.499	8.437		7.444	14.2
U-235		7.607	7.667	7.631	0.8
U-236	8.071	8.015		7.738	4.3
U-238		7.796		7.780	0.2
Np-237	7.765	7.636	7.548	7.784	3.1
Pu-238	7.575	7.509	7.790	10.77	43.4
Pu-239		7.719	7.717	7.699	0.3
Pu-240		7.580	7.818	7.831	3.3
Pu-241		8.188	7.990	7.828	5.1
Pu-242	7.934	7.523	8.041	7.830	6.9
Am-241	7.977	8.259	7.635	7.767	8.2
Am-242g	10.20			7.741	31.8
Am-242m	7.040			7.743	10.0
Am-243	7.926	7.988	7.663	7.839	4.2
Cm-242	7.408		8.056	7.807	8.8
Cm-243	8.219		7.955	7.762	5.9
Cm-244	7.528	7.632		7.616	1.4
Cm-245	8.218		8.100	7.635	7.6
Cm-246	7.768		8.742		12.5
Cm-247	7.911		8.556		8.2

Table XII - Cf-252 fission spectrum averaged total cross section (barns).

NUCLIDE	ENDF/B-V	ENDF/B-IV	INDL/A-83	JENDL-2	MAXIMUM % DIFFERENCE
Th-232		0.07443	0.07586	0.08162	9.7
Pa-233	0.4798	0.4798	0.6083	0.9964	107.7
U-233		1.833		1.885	2.8
U-234	1.232	1.157		1.208	6.5
U-235		1.241	1.237	1.248	0.9
U-236	0.5992	0.5875		0.6056	3.1
U-238		0.3154		0.3233	2.5
Np-237	1.352	1.351	1.308	1.297	4.2
Pu-238	1.983	2.078	1.983	2.017	4.8
Pu-239		1.789	1.795	1.818	1.6
Pu-240		1.336	1.341	1.367	2.3
Pu-241		1.650	1.621	1.621	1.8
Pu-242	1.129	1.234	1.196	1.138	10.4
Am-241	1.474	1.277	1.349	1.519	19.0
Am-242g	0.001502			1.759	117010.5
Am-242m	2.214			1.866	18.7
Am-243	1.205	1.023	1.124	1.284	25.5
Cm-242	1.028		1.664	1.798	74.9
Cm-243	2.073		2.167	2.372	14.4
Cm-244	1.614	1.755		1.554	12.9
Cm-245	1.978		1.980	1.878	5.4
Cm-246	1.386		1.342		3.3
Cm-247	2.065		2.265		9.7

Table XIII - Cf-252 fission spectrum averaged fission cross sections (barns).

NUCLIDE	ENDF/B-V	ENDF/B-IV	INDL/A-83	JENDL-2	MAXIMUM % DIFFERENCE
Th-232		0.09839	0.09501	0.08348	17.9
Pa-233	0.1786	0.1786	0.1565	0.1178	51.6
U-233		0.06086		0.08034	32.0
U-234	0.1705	0.1704		0.1031	65.4
U-235		0.09582	0.08844	0.1274	44.1
U-236	0.1702	0.1702		0.1377	23.6
U-238		0.07136		0.06495	9.9
Np-237	0.1627	0.1629	0.1844	0.1676	13.3
Pu-238	0.1423	0.04100	0.07334	0.3148	667.8
Pu-239		0.03976	0.05335	0.05901	48.4
Pu-240		0.08211	0.08546	0.09012	9.8
Pu-241		0.1073	0.06640	0.1455	119.0
Pu-242	0.07040	0.05541	0.07115	0.08512	53.6
Am-241	0.2542	0.06190	0.3028	0.2904	389.2
Am-242g	0.0002470			0.2020	81681.4
Am-242m	0.01868			0.1041	457.3
Am-243	0.07300	0.06175	0.2350	0.1884	280.6
Cm-242	0.02408		0.02963	0.08468	251.7
Cm-243	0.01477		0.01696	0.03328	125.3
Cm-244	0.1193	0.06487		0.1191	83.9
Cm-245	0.04074		0.04658	0.04413	14.3
Cm-246	0.04219		0.02284		84.7
Cm-247	0.04056		0.05013		23.6

Table XIV - Cf-252 fission spectrum averaged capture cross sections (barns).

NUCLIDE	ENDF/B-V	ENDF/B-IV	INDL/A-83	JENDL-2	MAXIMUM % DIFFERENCE
Th-232		7.249	7.557	7.564	4.4
Pa-233	7.304	7.304	7.482	7.596	4.0
U-233		7.262		7.686	5.8
U-234	8.518	8.459		7.454	14.3
U-235		7.617	7.680	7.643	0.8
U-236	8.096	8.033		7.752	4.4
U-238		7.809		7.795	0.2
Np-237	7.782	7.648	7.561	7.798	3.1
Pu-238	7.588	7.527	7.806	10.79	43.4
Pu-239		7.730	7.723	7.711	0.3
Pu-240		7.595	7.826	7.846	3.3
Pu-241		8.209	8.008	7.843	4.7
Pu-242	7.945	7.539	8.057	7.849	6.9
Am-241	7.987	8.261	7.651	7.777	8.0
Am-242g	10.20			7.757	31.5
Am-242m	7.029			7.760	10.4
Am-243	7.944	7.998	7.683	7.849	4.1
Cm-242	7.419		8.067	7.817	8.7
Cm-243	8.232		7.968	7.778	5.8
Cm-244	7.538	7.651		7.624	1.5
Cm-245	8.237		8.115	7.643	7.8
Cm-246	7.778		8.762		12.7
Cm-247	7.933		8.586		8.2

Table XV - U-235 fission spectrum averaged total cross section (barns).

NUCLIDE	ENDF/B-V	ENDF/B-IV	INDL/A-83	JENDL-2	MAXIMUM % DIFFERENCE
Th-232		0.07153	0.07274	0.07846	9.7
Pa-233	0.4680	0.4680	0.6037	0.9906	111.7
U-233		1.836		1.888	2.8
U-234	1.226	1.152		1.199	6.4
U-235		1.241	1.238	1.248	0.8
U-236	0.5899	0.5772		0.5959	3.2
U-238		0.3066		0.3148	2.7
Np-237	1.347	1.345	1.303	1.293	4.2
Pu-238	1.976	2.068	1.978	2.011	4.7
Pu-239		1.787	1.794	1.818	1.7
Pu-240		1.334	1.335	1.364	2.3
Pu-241		1.651	1.624	1.624	1.7
Pu-242	1.125	1.224	1.198	1.135	10.1
Am-241	1.465	1.263	1.342	1.510	19.6
Am-242g	0.001929			1.756	90931.6
Am-242m	2.217			1.866	18.8
Am-243	1.193	1.019	1.116	1.274	25.0
Cm-242	1.006		1.662	1.798	78.7
Cm-243	2.081		2.170	2.379	14.3
Cm-244	1.607	1.743		1.554	12.2
Cm-245	1.980		1.983	1.887	5.1
Cm-246	1.378		1.342		2.7
Cm-247	2.064		2.276		10.3

Table XVI - U-235 fission spectrum averaged fission cross section (barns).

NUCLIDE	ENDF/B-V	ENDF/B-IV	INDL/A-83	JENDL-2	MAXIMUM % DIFFERENCE
Th-232		0.1006	0.09734	0.08575	17.3
Pa-233	0.1839	0.1840	0.1619	0.1222	50.6
U-233		0.06175		0.08233	33.3
U-234	0.1745	0.1744		0.1060	64.6
U-235		0.09817	0.09093	0.1310	44.1
U-236	0.1742	0.1742		0.1415	23.1
U-238		0.07334		0.06675	9.9
Np-237	0.1677	0.1678	0.1891	0.1733	12.8
Pu-238	0.1456	0.04209	0.07509	0.319	657.9
Pu-239		0.04090	0.05447	0.06075	48.5
Pu-240		0.08423	0.08758	0.09274	10.1
Pu-241		0.1094	0.06878	0.149	116.8
Pu-242	0.07256	0.05675	0.07236	0.08776	54.6
Am-241	0.2602	0.06498	0.3098	0.2981	376.8
Am-242g	0.0003172			0.2067	65063.9
Am-242m	0.01963			0.1065	442.5
Am-243	0.0762	0.06366	0.2419	0.1947	280.0
Cm-242	0.02498		0.03085	0.08703	248.4
Cm-243	0.0155		0.01760	0.03451	122.7
Cm-244	0.1228	0.06632		0.1222	85.2
Cm-245	0.04177		0.04786	0.04518	14.6
Cm-246	0.04329		0.02361		83.4
Cm-247	0.0416		0.05155		23.9

Table XVII - U-235 fission spectrum averaged capture cross section (barns).

Cf-252 Spectrum				
NUCLIDE	EXP	* ENDF/B-V	* INDL/A-83	* JENDL-2
Th-232	0.0883 ± 0.0024		14.1	7.6
U-233	1.938 ± 0.020			2.7
U-235	1.225 ± 0.009		1.1	1.9
U-238	0.328 ± 0.003			1.4
Np-237	1.409 ± 0.017	4.1	7.2	8.0
Pu-239	1.841 ± 0.017		2.5	1.3
Pu-240	1.337 ± 0.032		0.3	2.2
Pu-241	1.616 ± 0.080		0.5	0.3
U-235 Spectrum				
Th-232	0.0777 ± 0.0027		6.4	1.0
U-238	0.308 ± 0.015			2.2

Table XVIII - Fission spectrum averaged fission cross sections (barns). Comparison with experimental values. (* these values are |Evaluated - Exp|/Exp in % for the respective libraries).

APPENDIX I

The following tables give the experimental values, which were processed by the program ASIGMA *, used to calculate the mean. The limits of the calculated resonance integral are 20 Mev and 0.5 eV. When the limits of the experimental values were different from these limits we renormalized the experimental value adding or subtracting the corresponding calculated value (numbers with * in the tables).

* "Aplicação do Método dos Mínimos Quadrados no Ajuste de Seções de Choque", by Alexandre D. Caldeira, Nota Interna, in preparation.

EXP.VALUE		ERROR	REF.
70.00	←	5.00	52
85.00	←	8.50	25
82.70	←	1.80	29
82.50	←	3.00	28
88.00	←	3.00	20
83.00	←	5.00	27
81.20	←	3.40	30
83.00	←	6.00	26
87.00	←	4.00	21
87.00	←	2.00	29
96.00	←	6.00	24
59.00	←	6.00	24
64.00	←	7.00	24
88.00	←	5.00	24
68.00	←	3.00	24
63.00	←	2.00	24
79.00	←	4.00	31
* 89.47	←	4.00	16
* 93.16	←	6.00	16
* 72.54	←	4.50	17

MEAN VALUE		STD.DEV.
79.40	←	.76

TABLE I.I -EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS) FOR TH-232

EXP.VALUE		ERROR	REF.
930.00	←	135.00	33
857.00	←	35.00	69
* 836.99	←	35.00	35
* 921.19	←	90.00	34
* 849.19	←	43.00	35
* 840.19	←	43.00	35
* 377.10	←	67.00	88

MEAN VALUE		STD.DEV.
817.00	←	17.92

TABLE I.II -EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS) FOR PA-233

EXP.VALUE	ERROR	REF.
798.00	← 26.00	28
838.00	← 40.00	38
771.00	← 49.00	39
850.00	← 90.00	41
830.00	← 60.00	42
* 738.70	← 36.00	37
* 728.70	← 24.00	37
* 824.70	← 90.00	32

MEAN VALUE	STD.DEV.
773.92	← 13.42

TABLE I.III-EXPERIMENTAL FISSION RESONANCE INTEGRAL (IN BARNS)
FOR U-233

EXP.VALUE	ERROR	REF.
135.00	← 8.00	39
146.00	← 8.00	42
* 147.75	← 7.00	36

MEAN VALUE	STD.DEV.
143.36	← 4.40

TABLE I.IV -EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS)
FOR U-233

EXP.VALUE	ERROR	REF.
271.00	← 25.00	43
274.00	← 11.00	44
278.00	← 9.00	45
275.00	← 16.00	39
292.00	← 14.00	42
277.30	← 5.40	46
284.00	← 25.00	43
280.00	← 7.00	89
* 288.67	← 14.00	37
* 302.67	← 15.00	17

MEAN VALUE	STD.DEV.
280.12	← 3.21

TABLE I.V -EXPERIMENTAL FISSION RESONANCE INTEGRAL (IN BARNS)
FOR U-235

EXP.VALUE	ERROR	REF.
136.00	← 8.00	39
143.00	← 7.00	48
150.00	← 6.00	42
* 137.86	← 3.00	47

MEAN VALUE	STD.DEV.
140.22	← 2.39

TABLE I.VI -EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS)
FOR U-235

EXP.VALUE	ERROR	REF.
450.00	± 30.00	49
417.00	± 25.00	50
419.00	± 70.00	50
381.00	± 20.00	22
259.00	± 22.00	18
* 332.10	± 15.00	88

MEAN VALUE	STD.DEV.
353.55	± 9.16

TABLE I.VII-EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS) FOR U-236

EXP.VALUE	ERROR	REF.
282.00	± 20.00	52
286.00	± 25.00	15
282.00	± 8.00	53
281.00	± 10.00	23
267.00	± 5.00	20
269.40	± 5.20	46
* 265.19	± 21.80	17
* 278.19	± 10.00	71
* 289.89	± 40.00	51

MEAN VALUE	STD.DEV.
272.47	± 2.89

TABLE I.VIII-EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS) FOR U-238

EXP.VALUE	ERROR	REF.
3260.00	± 280.00	55
3310.00	± 400.00	55

MEAN VALUE	STD.DEV.
3276.44	± 229.38

TABLE I.IX -EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS) FOR PU-238

EXP.VALUE	ERROR	REF.
387.00	± 22.00	44
301.00	± 10.00	45
327.00	± 22.00	40
366.00	± 26.00	56
330.00	± 30.00	41
328.00	± 22.00	67
320.00	± 19.00	72
327.00	± 22.00	14
434.00	± 61.00	57

MEAN VALUE	STD.DEV.
323.80	± 6.49

TABLE I.X -EXPERIMENTAL FISSION RESONANCE INTEGRAL (IN BARNS) FOR PU-239

EXP.VALUE	ERROR	REF.
8904.00	← 550.00	58
11000.00	←4000.00	68
11731.00	←1000.00	19
11459.00	←1000.00	15
* 8654.54	← 700.00	59

MEAN VALUE	STD.DEV.
9579.37	← 367.38

TABLE I.XI -EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS)
FOR PU-240

EXP.VALUE	ERROR	REF.
557.00	← 33.00	44
550.00	← 40.00	41
* 524.08	← 16.00	37

MEAN VALUE	STD.DEV.
532.60	← 13.55

TABLE I.XII-EXPERIMENTAL FISSION RESONANCE INTEGRAL (IN BARNS)
FOR PU-241

EXP.VALUE	ERROR	REF.
1275.00	← 30.00	55
1275.00	← 30.00	55
1280.00	← 60.00	60

MEAN VALUE	STD.DEV.
1275.56	← 20.00

TABLE I.XIII-EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS)
FOR PU-242

EXP.VALUE	ERROR	REF.
2100.00	← 200.00	62
850.00	← 60.00	61
1140.00	← 40.00	54
1570.00	← 110.00	72

MEAN VALUE	STD.DEV.
1119.20	← 31.46

TABLE I.XIV-EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS)
FOR AM-241

EXP.VALUE	ERROR	REF.
9.00	← 1.00	67
17.10	← 1.30	72

MEAN VALUE	STD.DEV.
12.01	← .79

TABLE I.XV -EXPERIMENTAL FISSION RESONANCE INTEGRAL (IN BARNS)
FOR AM-243

EXP.VALUE	ERROR	REF.
2290.00	← 50.00	55
2300.00	← 200.00	62
2200.00	← 150.00	72
2130.00	← 110.00	55

MEAN VALUE	STD.DEV.
2259.26	← 42.56

TABLE I.XVI-EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS)
FOR AM-243

EXP.VALUE	ERROR	REF.
1860.00	← 400.00	63
1480.00	← 150.00	73
* 1590.99	← 136.00	70

MEAN VALUE	STD.DEV.
1559.95	← 97.70

TABLE I.XVII-EXPERIMENTAL FISSION RESONANCE INTEGRAL (IN BARNS)
FOR CM-243

EXP.VALUE	ERROR	REF.
12.50	← 2.50	63
13.40	← 1.50	67
13.40	← 1.00	14
* 18.05	← 1.00	66

MEAN VALUE	STD.DEV.
15.13	← .62

TABLE I.XVIII-EXPERIMENTAL FISSION RESONANCE INTEGRAL (IN BARNS)
FOR CM-244

EXP.VALUE	ERROR	REF.
650.00	← 50.00	61
650.00	← 50.00	63
626.00	← 53.00	74

MEAN VALUE	STD.DEV.
642.61	← 29.41

TABLE I.XIX-EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS)
FOR CM-244

EXP.VALUE	ERROR	REF.
770.00	← 150.00	63
805.00	← 80.00	67
802.00	← 80.00	14
* 1160.67	← 100.00	65

MEAN VALUE	STD.DEV.
878.41	← 46.78

TABLE I.XX -EXPERIMENTAL FISSION RESONANCE INTEGRAL (IN BARNS)
FOR CM-245

EXP.VALUE	ERROR	REF.
108.00	← 61.00	74
* 104.28	← 8.00	64

MEAN VALUE	STD.DEV.
104.32	← 7.96

TABLE I.XXI-EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS)
FOR CM-245

EXP.VALUE	ERROR	REF.
13.30	← 1.50	14
13.30	← 1.50	67
* 10.00	← .40	66

MEAN VALUE	STD.DEV.
10.41	← .37

TABLE I.XXII-EXPERIMENTAL FISSION RESONANCE INTEGRAL (IN BARNS)
FOR CM-246

EXP.VALUE	ERROR	REF.
118.00	← 15.00	74
* 121.03	← 7.00	64

MEAN VALUE	STD.DEV.
120.49	← 6.34

TABLE I.XXIII-EXPERIMENTAL CAPTURE RESONANCE INTEGRAL (IN BARNS) FOR CM-246

EXP.VALUE	ERROR	REF.
935.00	← 190.00	63
730.00	← 70.00	67
* 1061.90	← 110.00	65
* 783.60	← 50.00	66

MEAN VALUE	STD.DEV.
806.35	← 37.41

TABLE I.XXIV-EXPERIMENTAL FISSION RESONANCE INTEGRAL (IN BARNS) FOR CM-247

EXP.VALUE	ERROR	REF.
.089	← .003	80
.085	← .005	84

MEAN VALUE	STD.DEV.
.088	← .002

TABLE I.XXV -CF-252 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR TH-232

EXP.VALUE	ERROR	REF.
1.893	← .046	80
1.947	← .031	87
1.947	← .031	82

MEAN VALUE	STD.DEV.
1.936	← .020

TABLE I.XXVI-CF-252 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR U-233

EXP.VALUE		ERROR	REF.
1.207	←	.052	75
1.215	←	.022	77
1.216	←	.019	76
1.052	←	.031	85
1.266	←	.019	86
1.266	←	.019	87

MEAN VALUE		STD.DEV.
1.225	←	.009

TABLE I.XXVII-CF-252 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR U-235

EXP.VALUE		ERROR	REF.
.324	←	.014	75
.308	←	.017	79
.311	←	.014	84
.288	←	.007	85
.347	←	.006	86
.347	←	.006	87
.326	←	.007	78

MEAN VALUE		STD.DEV.
.328	←	.003

TABLE I.XXVIII-CF-252 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR U-238

EXP.VALUE		ERROR	REF.
1.380	←	.100	79
1.442	←	.023	82
1.366	←	.027	78

MEAN VALUE		STD.DEV.
1.409	←	.017

TABLE I.XXIX-CF-252 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR NP-237

EXP.VALUE		ERROR	REF.
1.790	←	.041	77
1.861	←	.030	87
1.861	←	.030	82
1.824	←	.035	78

MEAN VALUE		STD.DEV.
1.841	←	.017

TABLE I.XXX -CF-252 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR PU-239

EXP.VALUE		ERROR	REF.
1.337	←	.032	80

TABLE I.XXXI-CF-252 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR PU-240

EXP.VALUE		ERROR	REF.
1.616	←	.080	80

TABLE I.XXXII-CF-252 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR PU-241

EXP.VALUE		ERROR	REF.
.077	←	.004	76
.076	←	.004	76

MEAN VALUE		STD.DEV.
.076	←	.003

TABLE I.XXXIII-U-235 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR TH-232

EXP.VALUE		ERROR	REF.
.308	←	.015	83

TABLE I.XXXIV-U-235 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR U-238

EXP.VALUE		ERROR	REF.
1.337	←	.032	80

TABLE I.XXXI-CF-252 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR PU-240

EXP.VALUE		ERROR	REF.
1.616	←	.080	80

TABLE I.XXXII-CF-252 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR PU-241

EXP.VALUE		ERROR	REF.
.077	←	.004	76
.076	←	.004	76

MEAN VALUE		STD.DEV.
.076	←	.003

TABLE I.XXXIII-U-235 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR TH-232

EXP.VALUE		ERROR	REF.
.308	←	.015	83

TABLE I.XXXIV-U-235 SPECTRUM AVERAGE FISSION CROSS SECTION (IN BARNS) FOR U-2238

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