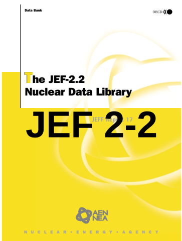

JEF/DOC-672

JUNE 1997

ECN-R-97-008

Status of pseudo fission-product cross sections for fast reactors

Results of the SWG17

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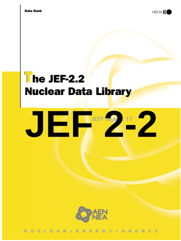


Abstract

Within the framework of the SWG17 benchmark organized by a Working Party of the Nuclear Science Committee of the Nuclear Energy Agency, a comparison of lumped or pseudo fission product cross sections for fast reactors has been made. Four institutions participated with data libraries based on the JEF2.2, EAF-4.2, BROND-2, FONDL-2.1, ADL-3 and JENDL-3.2 evaluated nuclear data files.

Several parameters have been compared with each other: the one-group cross sections and reactivity worths of the lumped nuclide for several partial absorption and scattering cross sections, and the one-group cross sections of the individual fission products. Also graphs of the multi-group cross sections of the lumped nuclide have been compared.

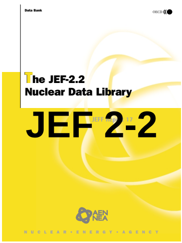
From two contributions based on JEF2.2, it can be concluded that the data processing influences the capture cross section by about 1% and the inelastic scattering cross section by 2%. The differences between the lumped cross sections of the different data libraries are surprisingly small: maximum 6% for capture and 9% for the inelastic scattering. Similar results are obtained for the reactivity effects. Since the reactivity worth of the lumped nuclide is dominated by the capture reaction, the maximum spread in the total reactivity worth is still only 5.5%.

The one-group capture and inelastic scattering cross sections of most of the important individual fission products differ by less then 10% (root mean square values). For the (n,2n) group cross sections, which are rather sensitive to the weighting spectrum in the fast energy range, these differences are several tens of percents.

The final conclusion is that the present status of lumped nuclide cross sections for fast reactors is satisfactory, although some improvements are possible as indicated in this report.

Keywords

Cross sections
Data Libraries
Fast Reactors
Fission products
Pseudo Nuclides



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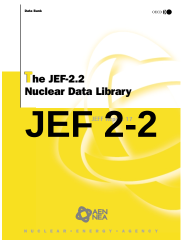


1. INTRODUCTION

This report summarizes the results of the Subworking Group 17 (SWG17) of the Working Party on Evaluation Coordination (WPEC) of the Nuclear Energy Agency (NEA) of the Organisation for Economic Development (OECD) at Paris. The objective of SWG17 was to investigate the status of fission-product cross section evaluations for fast reactor applications by means of comparison of the lumped fission product effects in fast power reactors. The definition of the activities of the subgroup was established at the May 1995 meeting of the WPEC at Paris. The background was that for future development of fast power reactors with extended burnup the reactivity effect of fission products need to be known preferably with an uncertainty of about 5% (1 σ).

An uncertainty of 5% in the reactivity effect is a rather small uncertainty compared to the actual uncertainty in the cross sections (averaged over a fast-reactor flux spectrum) of the individual fission products. However, since the reactivity effect is composed of a very large number of contributions of individual fission products, cancellation of errors may occur, leading to a reduced uncertainty in the lumped effect. This could be true for statistical errors, but not for errors of systematic nature. Since for the bulk of the fission products there are only few experimental data available in the fast energy range, most evaluated data rely heavily on calculations, with adjustments to available differential or in some cases integral data. For the radiative capture cross sections the Hauser-Feshbach theory with width-fluctuation correction is used in all evaluations; a small systematic error could result from different formalisms used for the width fluctuation correction and global (optical-model) parametrizations. However, in general local systematics were used for the parametrization, reducing the effects of systematic errors. Also from experimental data adjustment a small systematic error could result. A rather small contribution results for (n,p) and (n,α) reactions. It is believed that altogether the systematic uncertainty in the lumped capture effect is below 5%.

Another component of the reactivity effect of fast power reactors is due to elastic and particularly inelastic scattering. In this case the important contributions (10 to 15%) to the total reactivity effect result from the lowest levels excited by inelastic scattering. Here only few measurements exist and the evaluations are in general based upon Hauser-Feshbach theory with width fluctuation correction. There has been some concern that in most evaluations a spherical optical model was used and that direct excitation was neglected. Since it is known that for many nuclides in the fission-product mass range direct excitation cannot be neglected at high energies, it was expected that also at lower energies some effect could be seen. Early γ-ray measurements of (n,n') reactions and integral reactivity measurements also indicated that the current evaluations were in general too low at low energies. Another subgroup (SWG 10) deals with these questions and new measurements were made, eg. for Pd and Mo isotopes. Meanwhile, it is clear that in the early evaluations of some fission products the cross sections for the excitation of low-lying states were too low. However the neglect of direct-collective cross section for the excitation of low-lying states. However the neglect of direct-collective effects was not the reason for the too low values at low energies; this was merely due to the optical-model parametrization. A coupled channels optical model generally showed the best results. At higher energies these effects however are important. The expectation was that altogether the uncertainty in the lumped inelastic scattering reactivity effect could be larger than the corresponding capture effect. A figure of 30% would not be surprising.

A comparison of lumped fission-product capture cross sections based upon different nuclear data libraries should reveal a spread in the data from which perhaps some conclusions could be drawn about the uncertainty and about the possible need to enhance evaluation efforts. An important condition is that the different libraries are independent. This is not completely true, since the same basic methodology and often the same experimental data were used. Also in some cases data for one library were adopted in another library. However there are some notable differences between the evaluations. First of all, different evaluators have used different parametrizations and in some evaluations more recent data have been used than in others. As an example, the JEF-2.2 fission product file has been adjusted to integral data, the JENDL-3 data file has undergone extensive re-evaluation with recent data. The BROND-2 file is another source of independent data. Furthermore, new activation files like EAF-4.2 and ADL-3 have been issued recently with emphasis on radiative capture and other charged-particle emission reaction cross sections. These new activation files are very complete, since they contain virtually all fission-product cross sections. Altogether it is believed that the outcome of an comparison could yield valuable indications with regard to the status of and uncertainty of the lumped fission-product effect in fast reactors.

Therefore, a computational benchmark was defined where participants were asked to calculate pseudo fission-product cross sections, using their own multigroup structure and given concentrations for the fissile nuclide ^{239}Pu. A micro flux weighting spectrum, typical for a fast power reactor has been supplied pointwise, but it was not strictly necessary to use it for the generation of multigroup constants, to avoid large amounts of work. The participant could also use its own library with a probably different weighting spectrum. However, the weighting spectrum was obligatory for the condensation to one-group cross sections. In a later phase also a pointwise adjoint flux spectrum was specified to allow for reactivity worth calculations. The results requested were: multi-group pseudo cross sections, one-group pseudo cross sections and reactivity worths, mainly for capture and inelastic scattering. For further interpretation also the one-group cross sections of the individual fission products were requested. All data necessary to perform the benchmark are included in this report. Almost all results are reproduced.

2. BENCHMARK DESCRIPTION

The lumped or pseudo fission-product nuclide accounts for the neutron absorption and scattering of all the individual fission products present in the spent fuel due to the fissioning of one actinide atom. The effective yields of the individual fission products are given in Appendix A for five different actinide atoms. In the benchmark, the yields of ^{239}Pu were used to calculate the cross sections of the lumped nuclide according to:

$$\sigma^x = \sum_{n=1}^{N} \gamma_n \sigma_n^x \tag{2.1}$$

where γ_n is the effective yield or concentration of fission product n and σ_x^n is the one-group microscopic cross section of that fission product for reaction type x. The yields of all the fission products specified in Appendix A sum to 2. If the contributions of some fission products are omitted from the sum in equation 2.1, e.g. due to lack of cross sections, the participant should specify the resulting sum of yields in his calculations. The cross section of the individual fission product n in equation 2.1 is weighted over the total energy range by:

$$\sigma_n^x = \frac{\sum_{g=1}^{G} \Phi_g \sigma_{n,g}^x}{\sum_{g=1}^{G} \Phi_g} \tag{2.2}$$

with Φ_g the fine-group neutron spectrum. This function had to be derived by linear interpolation (lin-lin) from the pointwise neutron spectrum provided in the benchmark description (see figure 2.1 and Appendix B). For scattering cross sections, equation 2.2 reads:

$$\sigma_n^x = \frac{\sum_{g=1}^{G} \Phi_g \sum_{g'=1}^{G} \sigma_{n,g\to g'}^x}{\sum_{g=1}^{G} \Phi_g} \tag{2.3}$$

To be able to solve discrepancies, the participant also had to specify the energy group structure and weighting spectra used to calculate the fine-group microscopic cross sections of the individual fission products ($\sigma_{n,g}^x$ in equation 2.2 and $\sigma_{n,g\to g'}^x$ in equation 2.3). The preferred micro-flux weighting spectrum is the one specified in Appendix B.

The participants had to calculate the contribution of the lumped nuclide to the reactivity effect according to:

$$\rho^x = -\frac{\sum_{g=1}^{G} \sigma_g^x \Phi_g^* \Phi_g}{\sum_{g=1}^{G} \Phi_g^* \Phi_g} \tag{2.4}$$

where Φ_g^* is the adjoint function. For a scattering cross section, the reactivity

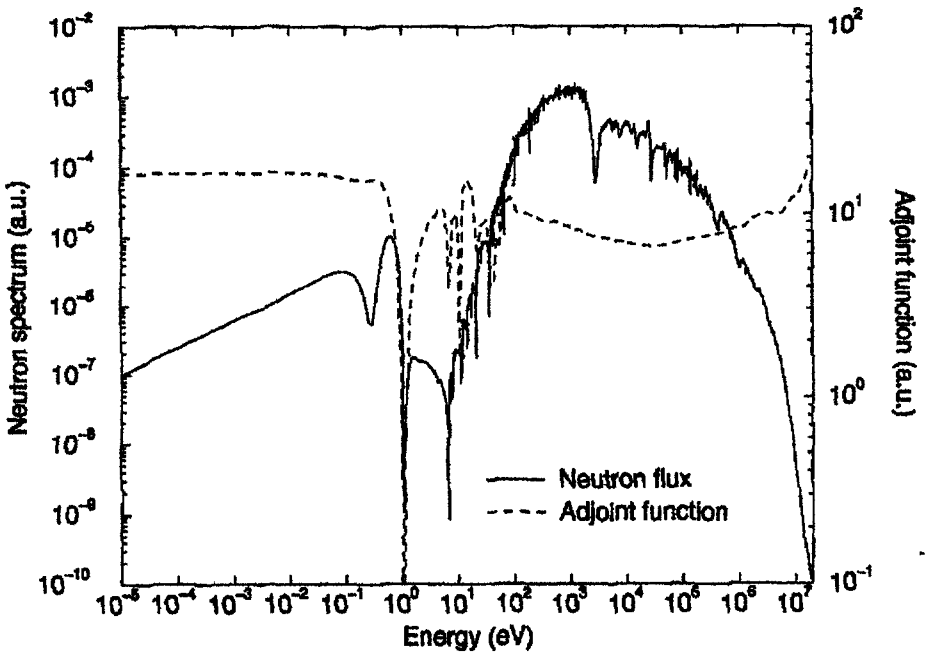


Figure 2.1 *The pointwise neutron spectrum and adjoint function used in the benchmark.*

effect is given by:

$$\rho^x = -\frac{\sum_{g=1}^{G} \sigma^x_{g\to g'} (\Phi^*_g - \Phi^*_{g'})\Phi_g}{\sum_{g=1}^{G} \Phi^*_g \Phi_g} \qquad (2.5)$$

Also the denominator in equations 2.4 and 2.5 had to be specified by the participants. The adjoint function was given pointwise with a linear interpolation scheme (lin-lin), cf. 2.1 and Appendix B.

3. PARTICIPANTS

Contributions to the benchmark were received from CEA (France), ECN (Netherlands), IPPE (Russia) and JNDC (Japan).

3.1 CEA

The contribution of the Commissariat à l'Energié Atomique (CEA) is based on a fine-group version of the JEF2.2 evaluated nuclear data file. The one-group cross sections and the corresponding reactivity worths of 80 individual fission products have been calculated as well as the one-group cross sections and reactivity worths of the lumped nuclide. The sum of yields of all the 80 fission products equals 1.90933.

3.2 ECN

The Netherlands Energy Research Foundation (ECN) contributed results based on the JEF-2.2 evaluated nuclear data file and the EAF-4.2 activation file. The first-mentioned file contains data for elastic and inelastic scattering and partial activation cross sections, the second one contains only partial activation and transmutation cross sections. The radiative capture cross sections in EAF-4.2 for the important fission product nuclides have been adopted from the JEF-2.2 file.

A fine-group library in XMAS structure (172 groups) is used, weighted with a thermal reactor spectrum and based on the JEF-2.2 evaluated nuclear data file. It contains the cross sections of 155 individual fission products with a sum of yields of 1.99936. From this library, one-group cross sections of all individual fission products were calculated as well as the fine-group cross sections and scattering matrices of the lumped nuclide. These were subsequently used to calculate the one-group cross sections and the reactivity worths of the lumped nuclide.

The extensive EAF-4.2 file was used for comparison and to calculate the influence of the weighting spectrum used in the preparation of the fine-group data library. Therefore, two fine-group libraries were made with different weighting spectra: one based on a thermal reactor spectrum (Maxwellian, 1/E and fission spectrum) and one based on the fast reactor spectrum specified in the benchmark. The fine-group cross sections of all individual fission products were used to calculate the fine-group cross sections of the lumped nuclide, which were subsequently condensed to one group.

3.3 IPPE

Three different contributions of the Institute of Physics Power Engineering (IPPE) were received based on libraries from the BROND-2, the FOND-2.1, and the ADL-3 evaluated nuclear data files. The BROND-2 file contains original evaluations made in Russia. The general purpose FOND-2.1 file contains BROND-2, but has been extended with other evaluations. The ADL-3 file is an extended activation/transmutation library. The BROND-2 library contains only 49 of the 162 requested fission products with a total cumulative yield of 1.3689; the extensive FOND-2.1 and ADL-3 libraries contain all 162 nuclides. For this reason, not all the BROND-2 results are reported here.

Two versions of the BROND-2 and FOND-2.1 libraries were used. First, existing fine-group versions with 299 energy groups were used to calculate the one-group (n,γ) cross sections of the individual fission products and the fine-group and one-group (n,γ) cross section and the corresponding reactivity worth of the lumped nuclide. Secondly, broad-group libraries with 28 energy groups condensed with a standard weighting spectrum (not the one specified in this benchmark) were used to calculate the (n,γ) cross section and the corresponding reactivity worth of the lumped nuclide, and the reactivity worth due to inelastic scattering (the inelastic scattering matrix in this FOND-2.1 library contains the contributions of 125 individual fission products with a sum of yields of 1.9340). The broad-group FOND-2.1 library was also used to calculate the one-group radiative capture cross section of the lumped nuclide. This cross section differs about 1% from the one-group cross section calculated by the fine-group version of the FOND-2.1 library and is omitted from the comparison in this report.

A fine-group version of the ADL-3 activation library with 299 energy groups was used to calculate for various reactions the one-group cross sections of the individual fission products and the one-group cross sections and the reactivity worths of the lumped nuclide.

3.4 JNDC

JNDC is a joint contribution of Kawasaki Heavy Industries, Hitachi, Toshiba Corporation and the Japan Atomic Energy Research Institute. Two contributions were received: one based on a 70-group library processed from the JENDL-3.2 evaluated nuclear data file and one based on a 73-group library. In both cases, a typical FBR spectrum was used for the micro flux weighting by the TIMS code. The JENDL-3.2 file contains 140 fission products of the 162 requested ones (see Appendix A) with a sum of yields of 1.99714.

First, a 70-group data library was made containing elastic and inelastic scattering matrices, and absorption cross sections in the energy range up to 10 MeV. This library contains the total absorption cross sections, but no partial capture cross sections. From this library, one-group cross sections for absorption, elastic and inelastic scattering and the (n,2n) reaction of the individual fission products were calculated as well as the one-group cross sections and the corresponding reactivity worths of the lumped nuclide. The results of this library have preference from the view point of reactor calculations.

Secondly, a 73-group data library was made based on the 70 group data library with three energy groups added in the range between 10 and 19.64 MeV. From this library, one-group cross sections were made for all reactions except the partial inelastic scattering ones. Also the fine-group cross sections, the one-group cross sections and the corresponding reactivity worths of the lumped nuclide were calculated. From the view point of nuclear data evaluation, this library has preference.

4. RESULTS LUMPED NUCLIDE

4.1 One-group cross sections

The one-group cross sections of the pseudo fission products are given in Table 4.1 for the elastic scattering (MT=2), inelastic scattering (MT=4), (n,2n) (MT=16), radiative capture (MT=102), (n,p) (MT=103) and (n,α) (MT=107) processes. The last column of the table contains the maximum spread. First we discuss the important capture and inelastic scattering cross sections. The maximum spread is about 6% for capture and about 9% for inelastic scattering. The highest values are for JEF-2.2 and rather low values for JENDL-3.2. Also the total and elastic scattering cross sections are relatively low in JENDL-3.2. Possibly the optical model parametrizations were quite different. For (n,2n), (n,α) and (n,p) there are large discrepancies. This has probably to do with the fact that these threshold cross sections show a steep rise where the fission spectrum declines steeply. Since the shape of the cross section is rather uncertain, this is reflected in the one-group cross section. It is noted that weigthing spectrum differences as well as group structure differences may play a role, cf section 4.4. The contributions of the (n,p) and (n,α) reactions to the total absorption cross section can be negelected. The conclusion is that the 6% maximum spread in the capture cross seems reasonable whereas the 9% spread in the inelastic cross section is smaller than expected.

4.2 Reactivity worths

The corresponding table for the reactivity effects is given in table 4.2. It is noted that the reactivity effects are given here with a normalization that is different from common practice. Thus absolute values have no meaning. On the average the total reactivity effect consists mainly of the radiative capture effect with a 10% contribution of inelastic scattering and very small contributions from the other components (less than 1%). Due to the fact that inelastic scattering cannot be calculated from the activation files a full comparison is only available for JEF-2.2 and JENDL-3.2. The maximum differences in the total reactivity is 5.5% and in the capture and inelastic scattering components 5.8% and 4.8%, respectively.

Table 4.1 *One group cross sections of the pseudo nuclide.*

Lab	File	Flux[a]	2 (b)	4 (b)	16 (mb)	102 (b)	103 (μb)	107 (μb)	Total (b)
CEA	JEF-2.2[b]		*15.2*	*0.590*		*0.571*			*16.4*
ECN	JEF-2.2	T	15.2	0.577	1.23	0.577	12.0	4.4	16.4
ECN	EAF-4.2[c]	T			*1.44*	*0.570*	*11.8*	*11.3*	*16.4*
ECN	EAF-4.2[d]	F			1.57	*0.565*	12.1	11.5	
JNDL	JENDL-3.2	F	14.4	0.531	1.12	0.546			15.5
IPPE	BROND-2[e]			*0.425*	*0.64*	*0.499*			
IPPE	FOND-2.1			0.527	0.78	0.578	12.8	5.32	
IPPE	ADL-3				0.92	0.544	7.5	6.73	
Average[f]			14.8	0.545	1.14	0.561	11.1	7.8	15.9
Maxdiff (%)			5.4	9.0	70	5.9	41	91	5.7

[a] Micro flux weighting spectrum (T is thermal, F is fast reactor weighting).
[b] Not used for average to avoid double counting of JEF-2.2 values.
[c] Not used for average; used to inspect the effect of the micro flux weighting.
[d] (n,γ) cross section not used for average because of relation with JEF-2.2.
[e] Not used for average because of missing nuclides.
[f] Numbers in italics not included in the average.

Table 4.2 *Reactivity effects of the pseudo nuclide (arbitrary units).*

Lab	File	Flux[a]	2 (au)[j]	4 (au)	16 (au)[j]	102 (au)	Total (au)	Total Corr[h]
CEA	JEF-2.2[b]		*-4.25*	*-0.0708*		*-0.580*	*-0.655*	*-0.656*
ECN	JEF-2.2	T	-3.92	-0.0684	-0.694	-0.583	-0.656	-0.656
ECN	EAF-4.2[c]	T				*-0.576*		
ECN	EAF-4.2[d]	F				*-0.571*		-0.642
JNDL	JENDL-3.2	F	-4.67	-0.0652	-0.644	-0.546	-0.622	-0.622
IPPE	BROND-2[e]			*-0.0477*		*-0.504*		
IPPE	FOND-2.1[f]			*-0.0632*		*-0.584*		
IPPE	FOND-2.1[g]			-0.0654		-0.584	-0.649	-0.654
IPPE	ADL-3			-0.0654		-0.550		-0.621
Average[i]			-4.30	-0.0663	-0.669	-0.567	-0.644	-0.639
Maxdiff(%)			17	4.8	7.5	5.8	5.3	5.5

[a] Micro flux weighting spectrum (T is thermal, F is fast reactor weighting).
[b] Not used for average to avoid double counting of JEF-2.2 values.
[c] Not used for average; used to inspect the effect of the micro flux weighting (see also table 4.3).
[d] (n,γ) cross section not used for average because of relation with JEF-2.2.
[e] Not used for average because of missing nuclides.
[f] Not used for average because of missing nuclides
[g] Corrected for missing nuclides (factor 2/1.934 for MT=4).
[h] Missing data supplemented with average value of missing components.
[i] Numbers in italics not included in the average.
[j] Numbers have been multiplied by $1 \cdot 10^3$.

4.3 Cross section plots

In Appendix C all data have been plotted for capture, inelastic scattering and (n,2n) cross sections. In this chapter, a detailed analysis is made mainly based on JEF-2.2, FOND-2.1 and JENDL-3.2.

First the plots of group cross sections for the radiative capture cross section are compared (see figures 4.1 and 4.2). Because the cross sections of the lumped nuclide show a 1/v behaviour above 100 keV, the same curves multiplied with the square root of the energy are shown in figures 4.3 and 4.4, respectively. In the energy range above 100 keV, the various evaluations only differ by a small factor. At lower energies, the resonance structure is visible and some differences between the evaluations is seen from the graphs. At the thermal groups the differences merely reflect the differences in the group structures. This also holds to a certain extent at the highest energies; above 10 MeV there are some differences in the direct-collective capture contributions. Altogether, since in the important energy range above 100 eV, most of the resonance structure has averaged out, there is good reason to believe that the hypothesis of cancellation of errors is correct for fast reactors and that in fact only systematic errors determine the uncertainty in fast capture effect. From the curves 4.3 and 4.4, it is seen that JENDL-3.2 is generally lower than JEF-2.2 and FOND-2.1 above 100 eV. In particular there are differences in the "unresolved range" from 1 to 100 keV.

Secondly, the plots of inelastic scattering cross sections are inspected (see figure 4.5). Here there is remarkable agreement between 200 keV and 10 MeV, but there are some differences below 200 keV. Apparently, cancellation of errors works well above 100 keV, but since there are much less nuclides with levels below 200 keV and since these have different threshold energies, the cross section below 200 keV shows a large spread. As the region up to 200 keV is quite important in fast reactors it could be worthwhile to concentrate on these nuclides. Further analysis shows that the cross section at 10 keV is mainly due to ^{151}Sm and to a lesser extent due to ^{103}Ru. Other important nuclides with thresholds below 200 keV are ^{101}Ru, ^{133}Cs, ^{103}Rh, ^{145}Nd, ^{107}Pd and ^{149}Sm.

The plots for the (n,2n) cross section show a large spread (see figure 4.6), which is partly due to the different group structures and flux weighting spectra used. Still it is curious that at 14 MeV, for which many (n,2n) measurement data are available, the FOND-2.1 library is too low. This needs further investigation. The EAF-4.2 evaluation was adjusted to 14 MeV data or systematics.

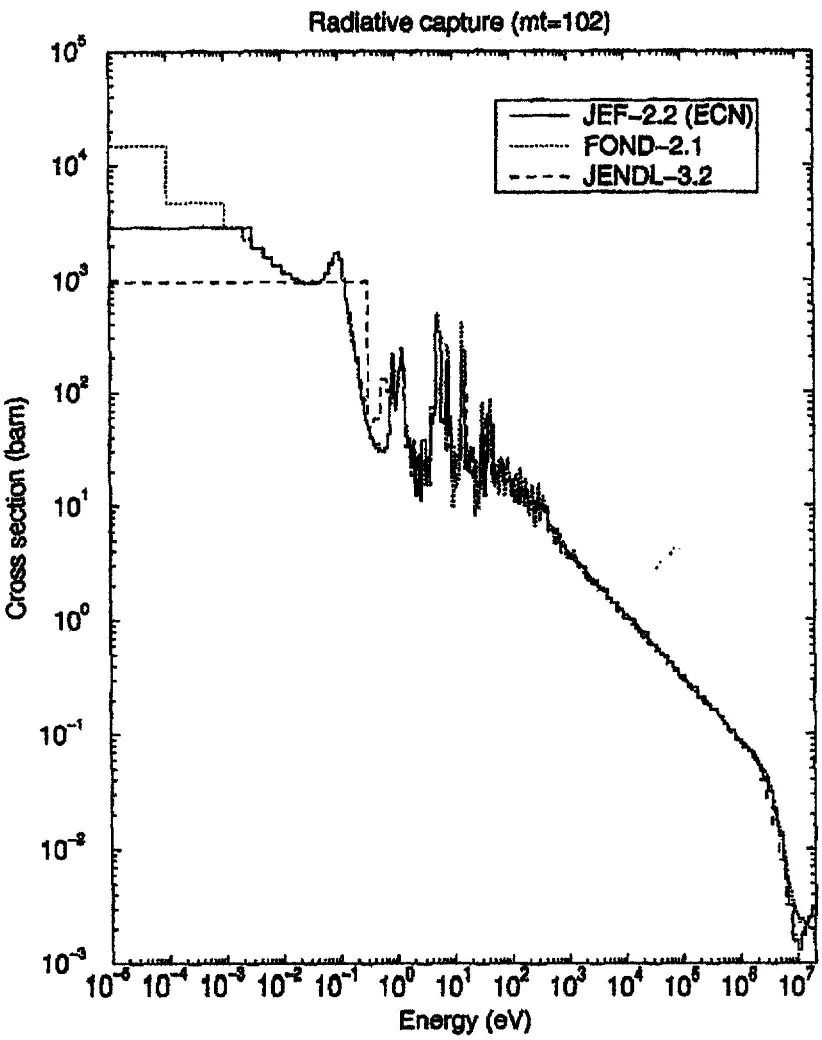
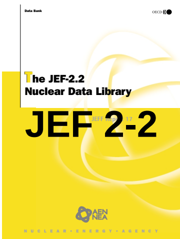


Figure 4.1 *The radiative capture cross section of the lumped nuclide as a function of the energy. Above 1 keV, the pseudo fission product behaves as a 1/v nuclide. The EAF-4.2 results are not shown because they are very close to JEF-2.2. The BROND-2 results are shown in Appendix C where a full comparison is given. The ADL-3 results are not available.*



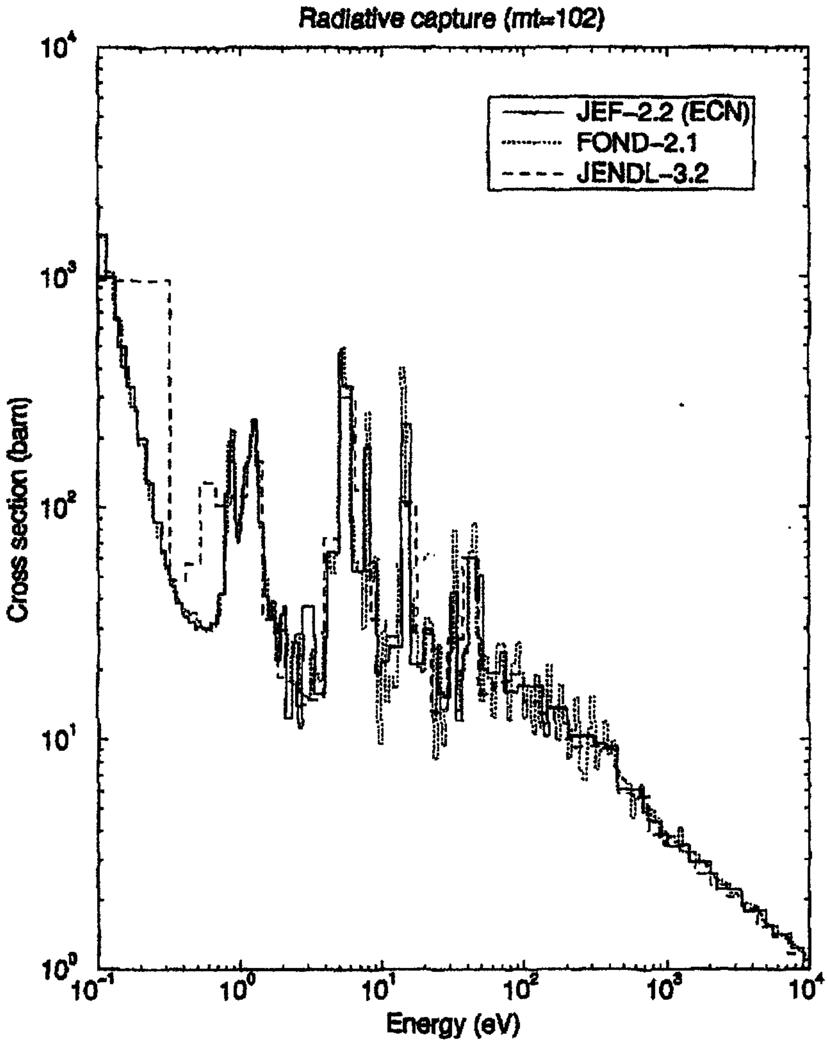


Figure 4.2 *The radiative capture cross section of the lumped nuclide as a function of the energy in the range between 0.1 eV and 10 keV. The main differences are due to the different energy group structures used by the participants. The EAF-4.2 results are very close to JEF-2.2, the BROND-2 results are shown in Appendix C and the ADL-3 results are not available.*

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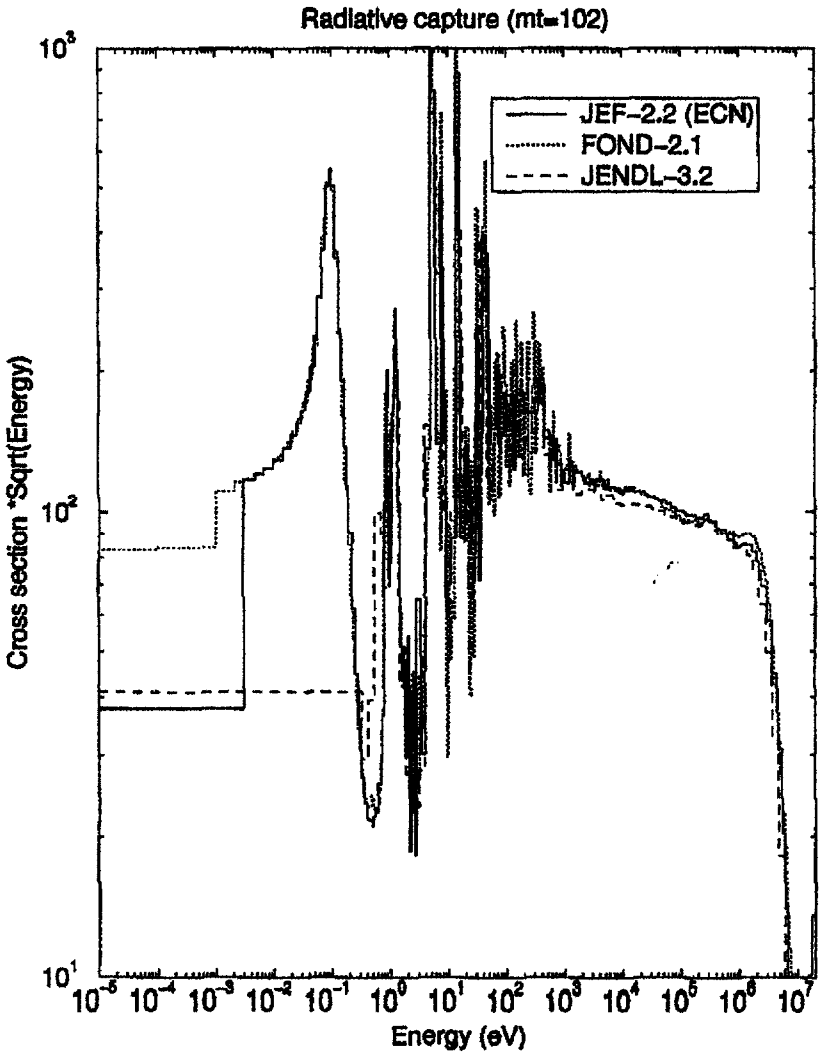

Figure 4.3 *The radiative capture cross section of the lumped nuclide multiplied with the square root of the energy as a function of the energy. The differences in the unresolved resonance range are evident from this figure. Again, the EAF-4.2 results are very close to JEF-2.2, the BROND-2 results are shown in Appendix C and the ADL-3 results are not available.*

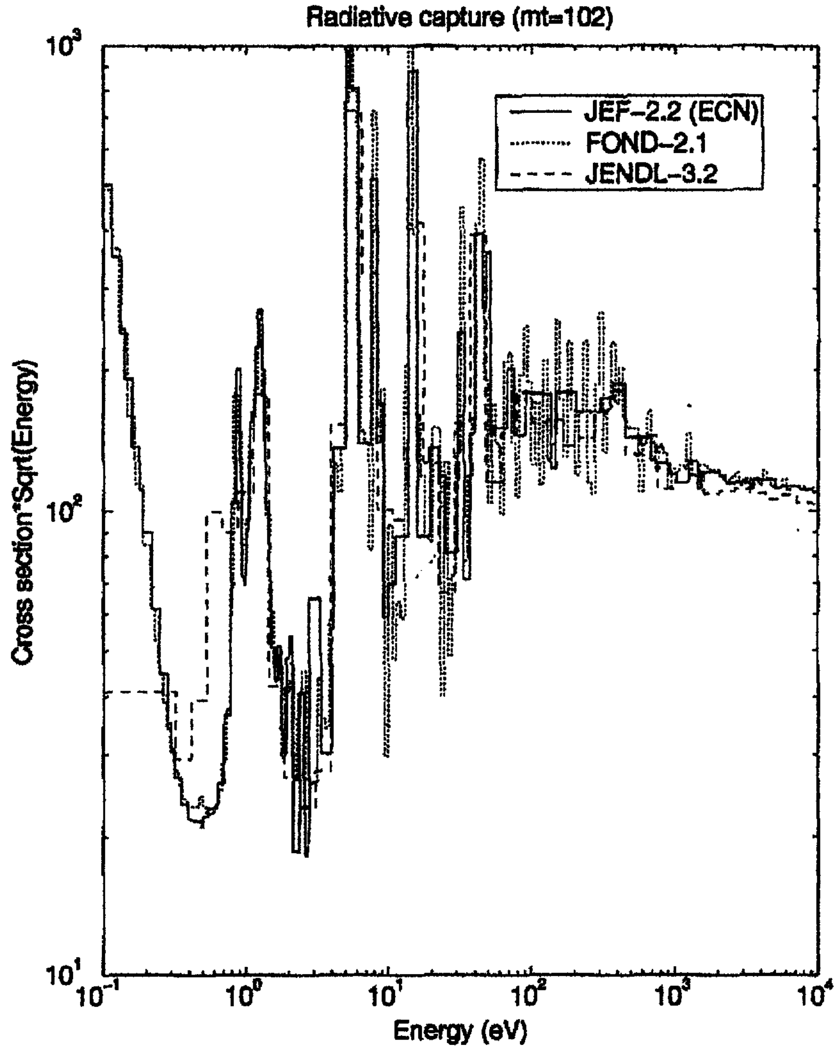

Figure 4.4 *The radiative capture cross section of the lumped nuclide multiplied with the square root of the energy as a function of the energy in the range between 0.1 eV and 10 keV. Again, the EAF-4.2 results are very close to JEF-2.2, the BROND-2 results are shown in Appendix C and the ADL-3 results are not available.*

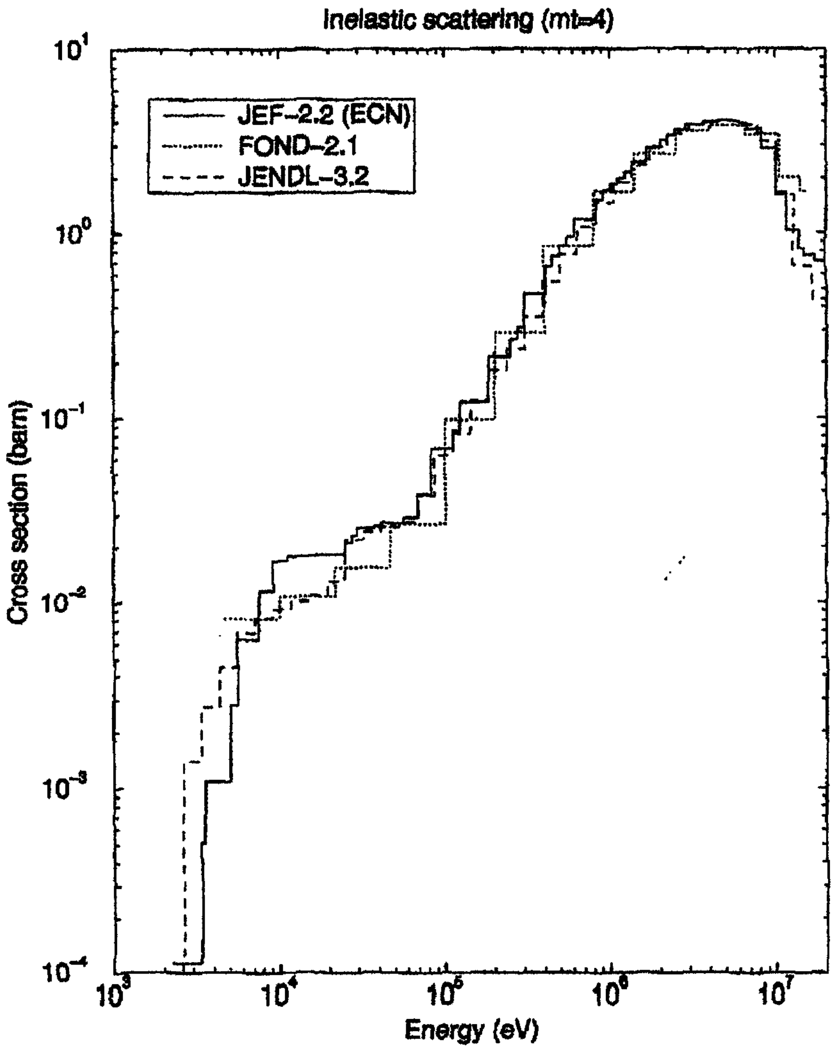

Figure 4.5 *The inelastic scattering cross section of the lumped nuclide as a function of the energy. In the MeV range, the deviations are smallest. At the lower energies (10 keV), the nuclides ^{151}Sm and ^{103}Ru dominate the cross section. Other important nuclides with thresholds below 200 keV are ^{101}Ru, ^{133}Cs, ^{103}Rh, ^{145}Nd, ^{107}Pd and ^{149}Sm.*

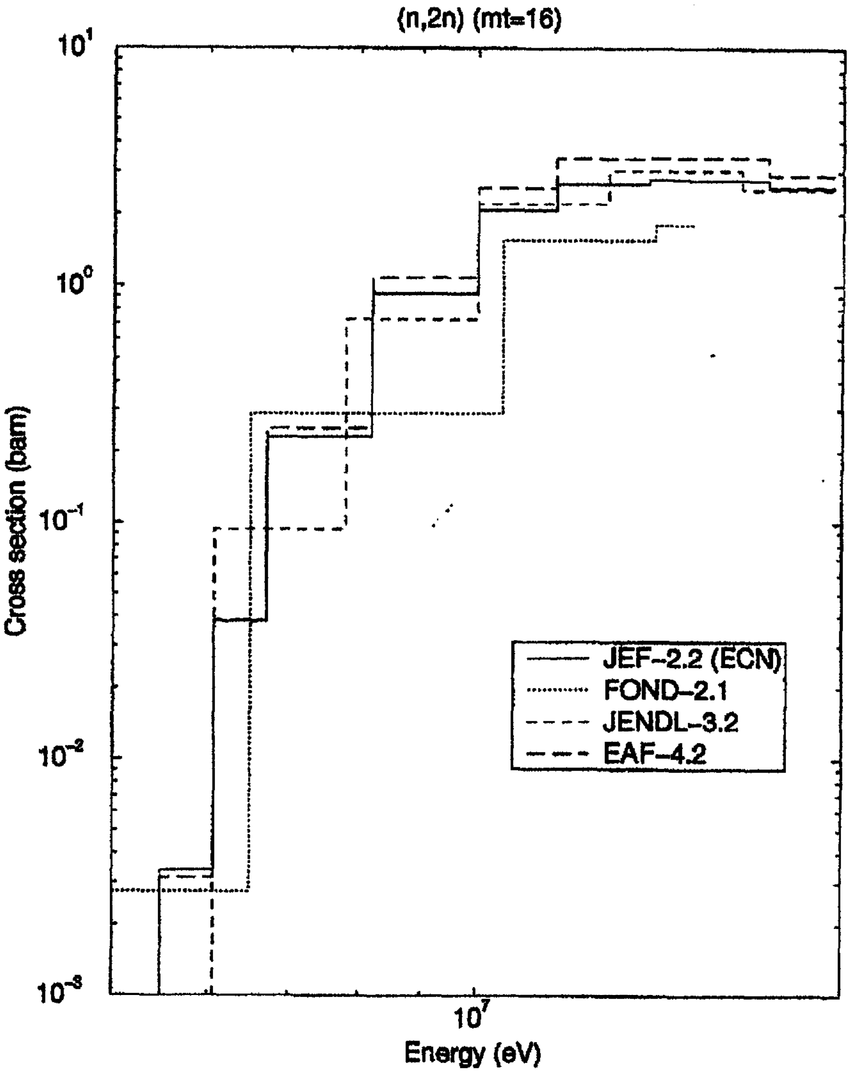

Figure 4.6 *The (n,2n) cross section of the lumped nuclide as a function of the energy.*

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4.4 Influence of the weighting spectrum

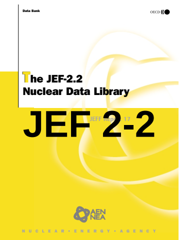
As mentioned in section 3.2, the basic EAF-4.2 file has been collapsed to a fine-group cross section data library in the XMAS group structure by use of two different weighting spectra: a light-water reactor spectrum (Maxwellian, 1/E and fission spectrum) and the fast reactor spectrum shown in figure 2.1. Subsequently the two fine-group libraries were collapsed to one group by the neutron spectrum provided in the benchmark.

Table 4.3 shows the two results. When the thermal weighting spectrum is used, the threshold reactions have up to 8% smaller cross sections and less negative reactivity worths then in case the fast weighting spectrum is used. This effect is quite large. Apparantly, the threshold reactions are very sensitive to the weighting spectrum used. The difference for the radiative capture cross section is, however, less than 1%. For the inelastic cross sections as far as available on EAF-4.2 (for production of metastable states only) the differences are below 2% (not shown in table 4.3). Therefore, the use of a pseudo fission product based on individual nuclide cross sections weighted with a thermal reactor spectrum is justified, provided the number of fine energy groups is sufficiently large.

The results of table 4.3 should also be compared with those of tables 4.1 and 4.2. For the (n,2n) reaction, the EAF-4.2 file gives higher cross sections than JEF-2.2, which is probably due to the missing data in JEF-2.2 (see chapter 5). There is a good agreement for the (n,γ) and (n,p) cross sections. For the (n,α) reaction the EAF-4.2 data seem quite high.

Table 4.3 *One-group cross sections and reactivity worths calculated with the EAF-4.2 activation file processed by two a thermal and fast weighting spectrums.*

MT	Thermal	Fast	Ratio
	One-group cross sections		
16	1.443e-03	1.572e-03	0.918
17	3.220e-05	3.478e-05	0.926
102	5.703e-01	5.653e-01	1.009
103	1.179e-05	1.212e-05	0.973
104	8.235e-07	8.613e-07	0.956
105	7.417e-08	7.950e-08	0.933
106	2.298e-09	2.430e-09	0.946
107	1.129e-05	1.147e-05	0.984
111	1.348e-08	1.430e-08	0.942
	Reactivity worths		
102	-5.760e-01	-5.709e-01	1.009
103	-1.746e-05	-1.792e-05	0.974
104	-1.764e-06	-1.851e-06	0.953
105	-1.694e-07	-1.817e-07	0.932
106	-5.168e-09	-5.476e-09	0.944
107	-1.482e-05	-1.506e-05	0.984
111	-3.073e-08	-3.268e-08	0.940



5. FISSION-PRODUCT CROSS SECTIONS

Table 5.1 gives the one-group radiative capture cross sections for 130 nuclides in order of descending importance (according to the product of concentration and cross section). In the French data set some nuclides are lacking in the lumped fission product (values set to zero). The differences between the CEA and ECN results reflect the processing differences.

Intercomparison of the data for the various nuclides reveals that there are still rather large differences beween the values of the cross sections of different nuclides. The strong resonance absorbers ^{149}Sm, ^{151}Sm, ^{153}Eu, ^{155}Eu have high cross sections also in fast power reactors. The last columns of table 5.1 contain the averages and root mean square values. Note that in this table, JEF-2.2 is counted twice, which may give too optimistic results. From the top-20 nuclides the RMS values are less than 10%, except for the nuclides ^{151}Sm, ^{103}Ru, ^{135}Cs. These are unstable nuclides, for which there are almost no measurements. Integral data are available for these nuclides and have been used in the JEF-2.2 evaluation. From the next 20 nuclides the one with RMS larger than 20% are ^{155}Eu, ^{96}Zr, and ^{95}Zr. It is recommended to have a closer look to these nuclides. More generally, the largest uncertainties come from unstable nuclides. Still, the average RMS value for the top 40 nuclides is below 10%.

Table 5.2 gives the one-group inelastic scattering cross sections for JEF-2.2 and JENDL-3.2 for the most important nuclides in order of descending importance. The activation libraries EAF-4.2 and ADL-3 do not contain total inelastic scattering cross sections. The differences between the values of the cross sections for different nuclides are smaller than in the case of radiative capture. The largest values are obtained for ^{151}Sm and ^{103}Ru, which are nuclides with rather low excitation levels (below 100 keV). The RMS values for the top 40 nuclides are less than 15%, except for the nuclides ^{104}Ru, ^{131}Xe, ^{139}La, ^{151}Sm, ^{144}Ce. These nuclides should be inspected in more detail. The average RMS value for the top 40 nuclides is less than 8%, which is surprisingly small.

From table 5.3 where the (n,2n) one-group cross sections are compared, it becomes eveident that the JEF-2.2 data file does not contain data for all nuclides. The RMS values are in general below 35%, however the cross section difference is rather systematic, with relatively low values for ADL-3. This needs further investigation.

Table 5.1 *One group cross sections for radiative capture (MT=102) of the nuclides in order of descending importance determined by the product of the cross section (FOND-2.1) and yield of the nuclide.*

Institute Library	CEA JEF-2.2	ECN JEF-2.2	IPPE FOND-2.1	IPPE ADL-3	JNDC JENDL-3.2	Average	RMS (%)
^{101}Ru	0.7143	0.7243	0.7616	0.7141	0.7523	0.7333	2.71
^{105}Pd	0.9369	0.9490	0.9161	0.8568	0.9594	0.9236	3.94
^{149}Sm	2.5398	2.5437	2.8700	2.6540	2.2990	2.5813	7.17
^{99}Tc	0.6301	0.6479	0.6561	0.6152	0.5923	0.6283	3.65
^{133}Cs	0.5072	0.5167	0.5184	0.4715	0.4874	0.5003	3.62
^{107}Pd	1.0569	1.0700	1.0399	0.9840	1.0520	1.0406	2.87
^{103}Rh	0.6751	0.6832	0.6526	0.6124	0.6774	0.6601	3.94
^{147}Pm	1.5064	1.5167	1.4309	1.3335	1.2753	1.4126	6.72
^{151}Sm	3.3618	3.3998	2.9413	2.6922	2.1080	2.9006	16.43
^{103}Ru	1.1828	1.1992	1.2140	1.1416	0.5047	1.0485	26.03
^{97}Mo	0.3351	0.3406	0.3568	0.3159	0.3484	0.3394	4.08
^{145}Nd	0.5657	0.5700	0.5124	0.4755	0.5648	0.5377	6.99
^{131}Xe	0.2917	0.2937	0.3323	0.3018	0.3461	0.3131	7.03
^{153}Eu	2.7363	2.7655	2.9288	2.4251	2.5958	2.6903	6.31
^{143}Nd	0.3592	0.3550	0.3370	0.3080	0.3589	0.3437	5.70
^{102}Ru	0.1558	0.1581	0.1796	0.1732	0.1642	0.1662	5.41
^{109}Ag	0.7846	0.7927	0.7203	0.6705	0.6916	0.7319	6.69
^{104}Ru	0.1517	0.1546	0.1658	0.1572	0.1685	0.1596	4.06
^{135}Cs	0.2379	0.2445	0.1361	0.2269	0.2284	0.2148	18.56
^{141}Pr	0.1553	0.1582	0.1540	0.1360	0.1564	0.1520	5.34
^{95}Mo	0.3180	0.3204	0.3351	0.3096	0.3360	0.3238	3.16
^{98}Mo	0.1233	0.1280	0.1179	0.1067	0.1194	0.1191	5.97
^{100}Mo	0.0933	0.0938	0.1011	0.0862	0.1000	0.0949	5.64
^{155}Eu	2.8163	2.8428	2.9559	2.6783	1.3368	2.5260	23.80
^{108}Pd	0.1778	0.1770	0.2507	0.2352	0.2358	0.2153	14.60
^{132}Xe	0.0689	0.0708	0.0755	0.0804	0.0980	0.0787	13.27
^{93}Zr	0.1349	0.1335	0.1042	0.0924	0.1057	0.1141	14.94
^{152}Sm	0.4951	0.4956	0.5077	0.4768	0.4799	0.4910	2.31
^{141}Ce	0.2970	0.2963	0.2960	0.2754	0.2977	0.2925	2.93
^{129}I	0.3644	0.3700	0.3804	0.3566	0.3840	0.3711	2.72
^{106}Ru	0.0922	0.0871	0.0946	0.0887	0.0916	0.0909	2.91
^{96}Zr	0.0343	0.0358	0.0590	0.0550	0.0391	0.0447	22.98
^{106}Pd	0.1994	0.2027	0.2553	0.1959	0.2772	0.2261	14.85
^{127}I	0.6168	0.6199	0.6307	0.5806	0.6028	0.6102	2.83
^{146}Nd	0.0947	0.0956	0.1100	0.0911	0.1076	0.0998	7.57
^{148}Nd	0.1703	0.1681	0.1636	0.1609	0.1469	0.1620	5.08
^{134}Xe	0.0361	0.0360	0.0374	0.0292	0.0272	0.0332	12.53
^{139}La	0.0339	0.0333	0.0390	0.0320	0.0343	0.0345	6.86
^{95}Nb	0.3469	0.3496	0.2705	0.3311	0.3669	0.3330	9.99
^{95}Zr	0.0651	0.0643	0.1229	0.0586	0.1489	0.0919	40.13

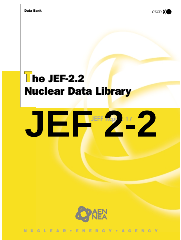


Table 5.1 *Continued.*

Institute Library	CEA JEF-2.2	ECN JEF-2.2	IPPE FOND-2.1	IPPE ADL-3	JNDC JENDL-3.2	Average	RMS (%)
^{110}Pd	0.0970	0.0985	0.2189	0.0920	0.1291	0.1271	37.53
^{111}Cd	0.4605	0.4665	0.4683	0.4350	0.7475	0.5155	22.62
^{147}Sm	1.4571	1.4660	1.4183	1.3243	1.2719	1.3875	5.52
^{150}Nd	0.1728	0.1748	0.1729	0.1668	0.1626	0.1700	2.68
^{154}Eu	3.0951	3.1233	3.1720	2.9430	3.4503	3.1567	5.24
^{143}Pr	0.4094	0.4095	0.4206	0.3847	0.1263	0.3501	32.13
^{157}Gd	1.8214	1.8389	1.6372	1.4989	1.3578	1.6308	11.36
^{92}Zr	0.0365	0.0368	0.0433	0.0425	0.0418	0.0402	7.26
^{91}Zr	0.0796	0.0789	0.0790	0.0720	0.0913	0.0801	7.77
^{156}Gd	0.6203	0.6156	0.7077	0.6631	0.7059	0.6625	6.00
^{94}Zr	0.0229	0.0237	0.0271	0.0262	0.0258	0.0251	6.22
^{147}Nd	1.0075	1.0173	0.7995	0.9482	1.2315	1.0008	13.90
^{137}Cs	0.0274	0.0280	0.0160	0.0264	0.0160	0.0228	24.34
^{142}Ce	0.0436	0.0442	0.0186	0.0196	0.0252	0.0302	37.57
^{144}Ce	0.0493	0.0490	0.0337	0.0470	0.0241	0.0406	24.85
^{159}Tb	0.0000	1.7482	2.0672	1.6488	1.8546	1.8297	8.48
^{154}Sm	0.2479	0.2507	0.2729	0.2514	0.2471	0.2540	3.77
^{83}Kr	0.0000	0.2465	0.2422	0.2232	0.2806	0.2481	8.34
^{144}Nd	0.0800	0.0807	0.0936	0.0758	0.0829	0.0826	7.23
^{128}Te	0.0418	0.0416	0.1029	0.0387	0.0408	0.0531	46.81
^{148}Pm	4.1510	4.1781	4.2530	5.8795	3.2167	4.3357	19.84
^{149}Pm	3.5314	3.5548	3.6200	3.3688	1.2367	3.0623	29.93
^{140}Ce	0.0138	0.0135	0.0123	0.0142	0.0061	0.0120	24.95
^{116}Cd	0.0000	0.1182	0.8856	0.1139	0.0823	0.3000	112.80
^{81}Br	0.0000	0.4293	0.3892	0.4048	0.2777	0.3752	15.48
^{125}Sb	0.3184	0.3217	0.3216	0.0000	0.4493	0.3527	15.82
^{161}Dy	0.0000	2.5911	2.5856	2.4223	0.0000	2.5330	3.09
^{99}Mo	0.5052	0.5132	0.5176	0.4907	0.3780	0.4810	10.87
^{115}In	0.4711	0.4772	0.4755	0.4540	0.6084	0.4972	11.30
^{104}Pd	0.0000	0.2016	0.3311	0.1921	0.2993	0.2560	23.57
^{91}Y	0.0461	0.0467	0.0472	0.0445	0.0891	0.0547	31.47
^{131}I	0.1686	0.1706	0.1743	0.1613	0.2808	0.1911	23.56
^{113}Cd	0.4092	0.4134	0.4128	0.3865	0.5273	0.4299	11.57
^{84}Kr	0.0000	0.0699	0.0664	0.0629	0.0519	0.0628	10.77
^{112}Cd	0.2421	0.2443	0.2535	0.2399	0.2084	0.2377	6.45
^{129}Te	0.0000	0.1387	0.1387	2.7649	0.7658	0.9520	113.18
^{150}Sm	0.4543	0.4583	0.3408	0.3237	0.4341	0.4022	14.41
^{121}Sb	0.4657	0.4707	0.4663	0.4467	0.4414	0.4582	2.57
^{100}Ru	0.0000	0.1996	0.1991	0.1886	0.2067	0.1985	3.26
^{158}Gd	0.0000	0.3337	0.3356	0.3126	0.3417	0.3309	3.32
^{130}Te	0.0143	0.0147	0.0149	0.0137	0.0129	0.0141	5.11
^{105}Rh	0.5905	0.5988	0.6036	0.0000	0.6544	0.6119	4.09
^{140}Ba	0.0000	0.0188	0.0691	0.0185	0.0024	0.0272	92.16
^{87}Rb	0.0247	0.0248	0.0248	0.0213	0.0292	0.0249	10.11
^{133}Xe	0.1351	0.1287	0.1275	0.1184	0.1380	0.1295	5.26

Table 5.1 *Continued.*

Institute Library	CEA JEF-2.2	ECN JEF-2.2	IPPE FOND-2.1	IPPE ADL-3	JNDC JENDL-3.2	Average	RMS (%)
^{114}Cd	0.0000	0.2785	0.2818	0.2611	0.1882	0.2524	15.00
^{155}Gd	0.0000	2.9068	2.9057	2.6694	2.6347	2.7792	4.59
^{148}Sm	0.0000	0.3686	0.3110	0.2907	0.2885	0.3147	10.27
^{90}Sr	0.0131	0.0134	0.0134	0.0126	0.0100	0.0125	10.25
^{140}La	0.3773	0.3788	0.3919	0.3507	0.0000	0.3747	3.99
^{136}Xe	0.0031	0.0031	0.0031	0.0029	0.0011	0.0026	28.79
^{89}Y	0.0000	0.0174	0.0145	0.0225	0.0178	0.0180	16.00
^{127}Te	0.0000	0.4100	0.4100	3.1977	0.8786	1.2241	94.39
^{138}Ba	0.0033	0.0034	0.0036	0.0034	0.0052	0.0038	18.71
^{111}Ag	0.0000	0.7936	0.8141	0.7389	0.0000	0.7822	4.06
^{148}Pm	0.0000	3.9372	4.2530	3.9542	1.9922	3.5342	25.44
^{117}Sn	0.0000	0.2228	0.2243	0.2129	0.2377	0.2244	3.93
^{143}Ce	0.0000	0.3364	0.3512	0.3144	0.0000	0.3340	4.53
^{89}Sr	0.0000	0.0225	0.0226	0.0214	0.0126	0.0198	21.01
^{110}Cd	0.0000	0.2714	0.2658	0.2640	0.2200	0.2553	8.06
^{118}Sn	0.0000	0.1255	0.1264	0.1204	0.0890	0.1153	13.34
^{123}Sb	0.0000	0.2560	0.2558	0.2471	0.2849	0.2610	5.46
^{160}Gd	0.0000	0.2236	0.1696	0.1726	0.2248	0.1977	13.44
^{77}Se	0.0000	0.4126	0.3864	0.3936	0.3970	0.3974	2.41
^{123}Sn	0.0000	0.1232	0.1239	0.1179	0.3672	0.1831	58.09
^{125}Te	0.0000	0.4001	0.4019	0.3696	0.3783	0.3875	3.59
^{80}Se	0.0000	0.0582	0.0493	0.0545	0.0422	0.0510	11.75
^{85}Kr	0.0000	0.0436	0.0066	0.0413	0.0597	0.0378	51.29
^{125}Sn	0.0000	0.3681	0.3791	0.3539	0.0000	0.3670	2.81
^{119}Sn	0.0000	0.0643	0.0630	0.0587	0.1847	0.0927	57.35
^{136}Ba	0.0000	0.0531	0.0505	0.0453	0.0700	0.0547	16.90
^{137}Ba	0.0000	0.0649	0.0680	0.0642	0.0822	0.0698	10.42
^{124}Sn	0.0000	0.0265	0.0281	0.0288	0.0150	0.0246	22.89
^{120}Sn	0.0000	0.0454	0.0454	0.0585	0.0456	0.0487	11.63
^{86}Kr	0.0000	0.0039	0.0034	0.0040	0.0028	0.0036	13.46
^{136}Cs	0.0000	0.3023	0.3187	0.2888	0.2535	0.2908	8.26
^{85}Rb	0.2193	0.2245	0.1979	0.2063	0.2767	0.2250	12.24
^{78}Se	0.0000	0.0806	0.0682	0.0706	0.0917	0.0778	11.95
^{130}Xe	0.0000	0.1265	0.1267	0.1461	0.2729	0.1680	36.33
^{128}Xe	0.0000	0.1905	0.1886	0.2026	0.2600	0.2104	13.82
^{96}Mo	0.0000	0.0859	0.0901	0.0873	0.0893	0.0882	1.88
^{160}Dy	0.0000	2.2301	2.2255	2.1165	0.0000	2.1907	2.40
^{82}Se	0.0000	0.0093	0.0094	0.0092	0.0288	0.0142	59.66
^{126}Sn	0.0000	0.0070	0.0070	0.0071	0.0085	0.0074	8.26
^{162}Dy	0.0000	0.9439	0.9456	0.8971	0.0000	0.9289	2.42
^{122}Sn	0.0000	0.0234	0.0238	0.0231	0.0276	0.0245	7.56
^{88}Sr	0.0000	0.0010	0.0010	0.0011	0.0041	0.0018	73.81
^{156}Eu	0.0000	0.0696	0.0713	0.0650	0.7202	0.2315	121.87
^{154}Gd	0.0000	1.1355	1.3159	1.2452	0.9733	1.1675	11.07
^{134}Ba	0.0000	0.1164	0.1120	0.1038	0.2081	0.1351	31.40

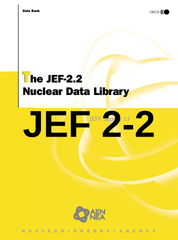


Table 5.2 *One group cross sections for inelastic scattering (MT=4). of the nuclides in order of descending importance determined by the product of the cross section (JENDL-3.2) and yield of the nuclide.*

Institute Library	CEA JEF-2.2	ECN JEF-2.2	JNDC JENDL-3.2	Average	RMS (%)
^{101}Ru	0.6176	0.6012	0.4748	0.5646	11.30
^{133}Cs	0.4675	0.4576	0.4411	0.4554	2.39
^{99}Tc	0.4633	0.4487	0.3599	0.4239	10.78
^{103}Rh	0.4129	0.4028	0.4041	0.4066	1.11
^{135}Cs	0.3057	0.2998	0.2615	0.2890	6.78
^{104}Ru	0.2127	0.2072	0.2919	0.2373	16.30
^{105}Pd	0.5106	0.4970	0.3870	0.4649	11.90
^{100}Mo	0.2716	0.2641	0.2852	0.2736	3.19
^{102}Ru	0.1940	0.1884	0.2426	0.2083	11.68
^{141}Pr	0.3522	0.3448	0.3577	0.3516	1.50
^{145}Nd	0.5086	0.4985	0.5236	0.5103	2.02
^{103}Ru	1.0791	1.0630	0.9858	1.0426	3.91
^{97}Mo	0.2611	0.2549	0.2947	0.2703	6.47
^{131}Xe	0.5059	0.4955	0.3559	0.4524	15.12
^{107}Pd	0.5094	0.4961	0.4284	0.4780	7.42
^{149}Sm	0.9357	0.9174	0.9073	0.9202	1.28
^{139}La	0.2745	0.2693	0.1850	0.2429	16.90
^{98}Mo	0.1898	0.1880	0.1956	0.1911	1.68
^{106}Ru	0.2860	0.2792	0.3048	0.2900	3.74
^{137}Cs	0.1718	0.1685	0.1565	0.1656	3.96
^{134}Xe	0.1670	0.1656	0.1268	0.1532	12.17
^{151}Sm	1.9461	1.9303	1.1939	1.6901	20.76
^{142}Ce	0.2104	0.2064	0.1706	0.1958	9.13
^{132}Xe	0.2088	0.2054	0.1559	0.1900	12.71
^{147}Pm	0.5153	0.5027	0.4834	0.5005	2.62
^{93}Zr	0.2349	0.2301	0.2062	0.2237	5.61
^{146}Nd	0.3134	0.3060	0.3054	0.3082	1.18
^{95}Mo	0.3383	0.3319	0.3328	0.3343	0.85
^{148}Nd	0.3780	0.3695	0.3943	0.3806	2.71
^{109}Ag	0.4323	0.4218	0.4105	0.4215	2.11
^{94}Zr	0.1380	0.1368	0.1543	0.1431	5.57
^{136}Xe	0.1156	0.1144	0.0924	0.1075	9.91
^{108}Pd	0.2604	0.2547	0.2531	0.2561	1.23
^{150}Nd	0.6038	0.5902	0.5478	0.5806	4.11
^{144}Ce	0.4548	0.4433	0.2091	0.3691	30.67
^{143}Nd	0.1570	0.1554	0.1380	0.1501	5.74
^{152}Sm	0.7550	0.7364	0.6112	0.7009	9.11
^{92}Zr	0.1322	0.1310	0.1510	0.1381	6.63
^{138}Ba	0.1087	0.1076	0.0849	0.1004	10.91
^{96}Zr	0.0883	0.0874	0.0838	0.0865	2.26

Table 5.2 *Continued.*

Institute Library	CEA JEF-2.2	ECN JEF-2.2	JNDC JENDL-3.2	Average	RMS (%)
^{96}Zr	0.0883	0.0874	0.0838	0.0865	2.26
^{129}I	0.4695	0.4605	0.4174	0.4491	5.06
^{140}Ce	0.0684	0.0676	0.0780	0.0713	6.61
^{153}Eu	1.0290	1.0051	0.7759	0.9367	12.18
^{110}Pd	0.2897	0.2838	0.2884	0.2873	0.88
^{130}Te	0.1155	0.1145	0.1292	0.1198	5.60
^{90}Sr	0.1056	0.1047	0.1303	0.1135	10.47
^{106}Pd	0.2263	0.2208	0.2036	0.2169	4.46
^{154}Sm	0.5118	0.4997	0.6552	0.5556	12.71
^{127}I	0.5195	0.5089	0.4693	0.4992	4.33
^{87}Rb	0.1884	0.1840	0.1813	0.1846	1.58
^{144}Nd	0.2105	0.2071	0.2132	0.2103	1.18
^{83}Kr	0.0000	0.6619	0.5107	0.5863	12.90
^{143}Pr	0.3973	0.3895	0.4662	0.4177	8.26
^{95}Zr	0.1093	0.1083	0.0956	0.1044	5.97
^{91}Y	0.1741	0.1713	0.1875	0.1776	3.99
^{91}Zr	0.0851	0.0845	0.0913	0.0870	3.51
^{155}Eu	0.7267	0.7078	0.6595	0.6980	4.05
^{111}Cd	0.4115	0.4014	0.3348	0.3826	8,90
^{134}Cs	0.7659	0.7546	0.8764	0.7990	6.87
^{141}Ce	0.1437	0.1421	0.1078	0.1312	12.62
^{147}Nd	0.6425	0.6295	0.8092	0.6938	11.79
^{128}Te	0.1524	0.1506	0.1433	0.1488	2.65
^{95}Nb	0.1814	0.1794	0.1316	0.1641	14.03
^{89}Y	0.0000	0.1013	0.0827	0.0920	10.13
^{156}Gd	0.5917	0.5773	0.5054	0.5581	6.76
^{157}Gd	0.8340	0.8158	0.8253	0.8251	0.90
^{88}Sr	0.0000	0.0404	0.0505	0.0455	11.16
^{147}Sm	0.5573	0.5452	0.5243	0.5423	2.51
^{99}Mo	0.5006	0.4874	0.6881	0.5587	16.41
^{84}Kr	0.0000	0.1638	0.1147	0.1392	17.62
^{140}Ba	0.0000	0.1311	0.1522	0.1416	7.44
^{131}I	0.2701	0.2646	0.2820	0.2722	2.66
^{86}Kr	0.0000	0.1114	0.0655	0.0885	25.98
^{133}Xe	0.1951	0.1925	0.2571	0.2149	13.88
^{158}Gd	0.0000	0.5067	0.5507	0.5287	4.16
^{81}Br	0.0000	0.2173	0.3164	0.2669	18.56
^{89}Sr	0.0000	0.0366	0.0790	0.0578	36.74
^{129}Te	0.0000	0.1390	0.1850	0.1620	14.20
^{119}Sn	0.0000	0.5773	0.6247	0.6010	3.94
^{85}Kr	0.0000	0.0537	0.0620	0.0579	7.16
^{150}Sm	0.3317	0.3243	0.3734	0.3431	6.30
^{82}Se	0.0000	0.1211	0.1439	0.1325	8.59
^{121}Sb	0.4425	0.4348	0.4908	0.4560	5.44
^{100}Ru	0.0000	0.1609	0.2071	0.1840	12.55

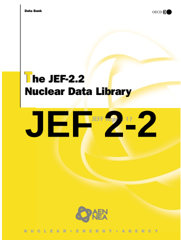


Table 5.2 *Continued.*

Institute Library	CEA JEF-2.2	ECN JEF-2.2	JNDC JENDL-3.2	Average	RMS (%)
^{126}Sn	0.0000	0.0667	0.1018	0.0842	20.81
^{113}Cd	0.3898	0.3776	0.3305	0.3659	6.99
^{125}Sb	0.1746	0.1714	0.1819	0.1759	2.50
^{159}Tb	0.0000	0.9638	0.6133	0.7885	22.23
^{112}Cd	0.1488	0.1457	0.1849	0.1598	11.15
^{104}Pd	0.0000	0.2035	0.1922	0.1979	2.86
^{154}Eu	0.9158	0.8918	0.4550	0.7542	28.08
^{156}Eu	0.0000	1.3320	1.1121	1.2221	9.00
^{117}Sn	0.0000	0.3214	0.3463	0.3339	3.73
^{148}Sm	0.0000	0.1982	0.2462	0.2222	10.81
^{105}Rh	0.2872	0.2806	0.4240	0.3306	20.00
^{114}Cd	0.0000	0.1390	0.2030	0.1710	18.72
^{160}Gd	0.0000	0.5105	0.5664	0.5385	5.19
^{80}Se	0.0000	0.1235	0.1525	0.1380	10.51
^{116}Cd	0.0000	0.1597	0.2119	0.1858	14.04
^{149}Pm	0.5045	0.4923	0.6892	0.5620	16.03
^{124}Sn	0.0000	0.0645	0.1035	0.0840	23.23
^{115}In	0.1190	0.1177	0.1393	0.1253	7.89
^{136}Ba	0.0000	0.1298	0.1353	0.1326	2.07
^{127}Te	0.0000	0.1215	0.2295	0.1755	30.78
^{153}Sm	0.0000	1.4055	1.3639	1.3847	1.50
^{148}Pm	0.6098	0.5942	0.4980	0.5673	8.71
^{137}Ba	0.0000	0.1851	0.1415	0.1633	13.37
^{123}Sb	0.0000	0.2208	0.2576	0.2392	7.70
^{110}Cd	0.0000	0.1255	0.1674	0.1464	14.30
^{155}Gd	0.0000	0.7726	0.7637	0.7681	0.57
^{122}Sn	0.0000	0.0727	0.0983	0.0855	14.99
^{120}Sn	0.0000	0.0670	0.1002	0.0836	19.87
^{125}Te	0.0000	0.5385	0.5546	0.5466	1.46
^{118}Sn	0.0000	0.0608	0.0959	0.0784	22.36
^{123}Sn	0.0000	0.0907	0.1487	0.1197	24.24
^{78}Se	0.0000	0.1294	0.1652	0.1473	12.15
^{77}Se	0.0000	0.3267	0.4345	0.3806	14.16
^{96}Mo	0.0000	0.1160	0.1833	0.1496	22.48
^{85}Rb	0.2460	0.2403	0.3057	0.2640	11.20
^{130}Xe	0.0000	0.2370	0.1742	0.2056	15.27
^{148}Pm	0.0000	0.5942	0.7662	0.6802	12.64
^{136}Cs	0.0000	0.1827	0.3319	0.2573	29.00
^{142}Nd	0.0000	0.0744	0.0974	0.0859	13.37
^{128}Xe	0.0000	0.2641	0.2022	0.2331	13.27
^{134}Ba	0.0000	0.1694	0.1688	0.1691	0.18
^{126}Te	0.0000	0.1412	0.1532	0.1472	4.09
^{90}Zr	0.0000	0.0501	0.0590	0.0545	8.14
^{82}Kr	0.0000	0.1753	0.1342	0.1548	13.29
^{115}Sn	0.0000	0.1916	0.2001	0.1958	2.16

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Table 5.3 *One group cross sections for (n,2n) reactions (MT=16). of the nuclides in order of descending importance determined by the product of the cross section (ADL-3) and yield of the nuclide.*

Institute Library	CEA JEF-2.2	ECN JEF-2.2	IPPE ADL-3	JNDC JENDL-3.2	Average	RMS (%)
^{142}Ce	1.1818	1.0179	1.0874	0.9124	1.0499	9.37
^{145}Nd	3.9818	3.8276	1.5629	3.2449	3.1543	30.40
^{143}Nd	2.6339	2.5023	1.0868	1.8441	2.0168	30.48
^{101}Ru	1.0978	0.9987	0.6108	1.0233	0.9326	20.31
^{131}Xe	2.0694	1.9330	0.9326	1.1105	1.5114	32.83
^{144}Ce	1.6126	1.4160	1.2582	1.2309	1.3794	11.02
^{97}Mo	0.0000	0.0000	0.5574	0.5398	0.5486	1.61
^{137}Cs	0.4922	0.3927	0.4355	0.3799	0.4251	10.32
^{136}Xe	0.7068	0.5802	0.3998	0.5135	0.5501	20.20
^{93}Zr	0.8254	0.7369	0.6553	1.0138	0.8078	16.50
^{107}Pd	2.2136	2.0801	0.8043	1.2667	1.5912	36.52
^{141}Ce	2.0727	1.9468	2.0871	3.2376	2.3361	22.40
^{96}Zr	0.6988	0.6158	0.4580	0.5241	0.5742	15.88
^{105}Pd	1.1214	1.0192	0.4507	0.7754	0.8417	30.70
^{134}Xe	0.6174	0.4972	0.2839	0.2728	0.4178	34.91
^{135}Cs	0.3927	0.2867	0.2716	0.2671	0.3045	16.89
^{100}Mo	0.5894	0.5017	0.3109	0.3065	0.4271	28.67
^{149}Sm	4.1049	3.9494	1.5415	2.6303	3.0565	34.21
^{151}Sm	2.9146	2.7685	2.4655	2.4093	2.6395	7.94
^{146}Nd	1.1233	1.0230	0.6855	0.6612	0.8733	23.27
^{103}Ru	2.2548	2.1410	1.0694	1.6210	1.7716	26.56
^{139}La	0.5170	0.4278	0.2676	0.2497	0.3655	30.54
^{104}Ru	0.0000	0.0000	0.2434	0.2903	0.2669	8.78
^{148}Nd	1.4448	1.3391	0.8781	0.8036	1.1164	25.02
^{133}Cs	0.4921	0.4005	0.2254	0.2128	0.3327	35.53
^{98}Mo	0.4841	0.3964	0.2514	0.2423	0.3435	29.58
^{94}Zr	0.0000	0.0000	0.3308	0.3609	0.3458	4.35
^{138}Ba	0.4936	0.3937	0.2701	0.2665	0.3560	26.55
^{95}Zr	1.0420	0.9499	0.7379	1.2580	0.9969	18.73
^{132}Xe	0.4985	0.3889	0.2375	0.2064	0.3328	35.45
^{147}Pm	1.0495	0.9402	0.7492	0.5796	0.8296	21.69
^{102}Ru	0.0000	0.0000	0.1723	0.2058	0.1890	8.86
^{99}Tc	0.4543	0.3697	0.1973	0.2010	0.3056	36.18
^{106}Ru	0.0000	0.0000	0.2971	0.3716	0.3344	11.15
^{140}Ce	0.0000	0.0000	0.2043	0.1914	0.1978	3.27
^{150}Nd	1.3084	1.1096	0.8256	0.6928	0.9841	24.41
^{90}Sr	0.0000	0.0000	0.4331	0.4335	0.4333	0.04
^{103}Rh	0.3539	0.2761	0.1669	0.1389	0.2340	36.82
^{141}Pr	0.3735	0.2902	0.1736	0.1826	0.2550	32.30
^{95}Mo	0.0000	0.0000	0.3382	0.3408	0.3395	0.38

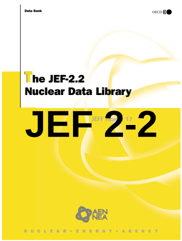


Table 5.3 *One group cross sections for (n,2n) reactions (MT=16).*

Institute Library	CEA JEF-2.2	ECN JEF-2.2	IPPE ADL-3	JNDC JENDL-3.2	Average	RMS (%)
^{140}Ba	0.0000	2.4793	1.7190	1.5348	1.9110	21.39
^{92}Zr	0.0000	0.0000	0.2253	0.2366	0.2309	2.44
^{130}Te	0.0000	0.0000	0.3499	0.3008	0.3254	7.54
^{91}Zr	0.4528	0.4000	0.3370	0.3101	0.3750	14.81
^{144}Nd	0.9323	0.8359	0.5527	0.4850	0.7015	26.70
^{147}Nd	0.0000	0.0000	3.1135	4.9516	4.0325	22.79
^{108}Pd	0.4641	0.3716	0.1749	0.2133	0.3060	38.36
^{89}Sr	0.0000	0.0000	0.7110	1.2691	0.9901	28.18
^{152}Sm	0.8242	0.7408	0.4670	0.3803	0.6031	30.57
^{91}Y	0.0000	0.0000	0.4003	0.3647	0.3825	4.66
^{143}Pr	0.0000	0.0000	0.8734	0.7758	0.8246	5.92
^{109}Ag	0.3879	0.3136	0.1785	0.1790	0.2648	33.96
^{95}Nb	0.4757	0.3922	0.2893	0.2776	0.3587	22.56
^{111}Cd	0.8347	0.7483	0.5800	0.7638	0.7317	12.77
^{128}Te	0.0000	0.0000	0.2773	0.2267	0.2520	10.02
^{140}La	0.0000	0.0000	3.9566	0.0000	3.9566	0.00
^{110}Pd	0.5779	0.4859	0.2284	0.3032	0.3988	34.98
^{133}Xe	0.0000	0.0000	0.9399	1.2794	1.1096	15.30
^{129}I	0.3965	0.3107	0.2297	0.2434	0.2950	22.41
^{129}Te	0.0000	0.0000	0.8330	1.5979	1.2155	31.46
^{85}Kr	0.0000	0.0000	0.2869	0.6447	0.4658	38.40
^{154}Sm	0.0000	0.0000	0.5825	0.4901	0.5363	8.61
^{153}Eu	0.7206	0.6040	0.3774	0.2563	0.4896	37.31
^{99}Mo	0.0000	0.0000	1.5544	2.1806	1.8675	16.77
^{106}Pd	0.3844	0.2929	0.1378	0.1692	0.2461	40.11
^{157}Gd	0.0000	0.0000	1.5178	1.2578	1.3878	9.37
^{143}Ce	0.0000	0.0000	3.4956	0.0000	3.4956	0.00
^{134}Cs	0.0000	0.0000	0.8867	0.9711	0.9289	4.54
^{147}Sm	2.6939	2.5681	0.9539	0.8208	1.7592	49.70
^{126}Sn	0.0000	0.0000	0.3984	0.4161	0.4073	2.17
^{87}Rb	0.0000	0.0000	0.0959	0.1044	0.1002	4.25
^{155}Eu	0.6982	0.5957	0.5030	0.3643	0.5403	22.74
^{86}Kr	0.0000	0.1518	0.1141	0.0885	0.1181	22.03
^{127}I	0.3880	0.3255	0.2162	0.2331	0.2907	24.05
^{154}Eu	1.0676	0.9558	1.6655	1.2818	1.2427	21.79
^{113}Cd	0.6090	0.5296	0.9032	1.1662	0.8020	31.44
^{88}Sr	0.0000	0.0000	0.0553	0.0483	0.0518	6.80
^{83}Kr	0.0000	0.6797	0.2187	0.3625	0.4203	45.82
^{89}Y	0.0000	0.0720	0.0490	0.0433	0.0548	22.62
^{156}Gd	0.6199	0.5188	0.3587	0.3074	0.4512	27.65
^{150}Sm	0.0000	0.0000	0.5997	0.3996	0.4997	20.02
^{119}Sn	0.0000	0.0000	0.8365	1.3121	1.0743	22.14
^{158}Gd	0.0000	0.0000	0.5982	0.5008	0.5495	8.87
^{131}I	0.0000	0.0000	0.2435	0.3070	0.2752	11.54
^{125}Sb	0.0000	0.0000	0.2931	0.2520	0.2726	7.52

Table 5.3 *One group cross sections for (n,2n) reactions (MT=16).*

Institute	CEA	ECN	IPPE	JNDC	Average	RMS
Library	JEF-2.2	JEF-2.2	ADL-3	JENDL-3.2		(%)
^{132}Te	0.0000	0.0000	0.3962	0.0000	0.3962	0.00
^{124}Sn	0.0000	0.0000	0.3355	0.3282	0.3318	1.11
^{84}Kr	0.0000	0.1263	0.0705	0.0628	0.0865	32.69
^{156}Eu	0.0000	0.0000	1.9911	1.3876	1.6894	17.86
^{148}Pm	0.0000	0.0000	2.0836	2.1593	2.1214	1.78
^{148}Sm	0.0000	0.0000	0.4725	0.4138	0.4431	6.62
^{137}Ba	0.0000	0.0000	0.6610	0.8709	0.7660	13.70
^{117}Sn	0.0000	0.0000	0.5407	0.8909	0.7158	24.46
^{127}Te	0.0000	0.0000	0.7215	1.1535	0.9375	23.04
^{161}Dy	0.0000	0.0000	1.4646	0.0000	1.4646	0.00
^{82}Se	0.0000	0.0000	0.1208	0.1422	0.1315	8.16
^{123}Sn	0.0000	0.0000	0.7066	2.2047	1.4556	51.46
^{160}Gd	0.0000	0.0000	0.7977	0.7502	0.7740	3.07
^{159}Tb	0.0000	0.6496	0.4991	0.4648	0.5378	14.92
^{112}Cd	0.0000	0.0000	0.1476	0.1672	0.1574	6.22
^{114}Cd	0.0000	0.0000	0.1915	0.2356	0.2135	10.32
^{122}Sn	0.0000	0.0000	0.2762	0.2667	0.2714	1.75
^{136}Ba	0.0000	0.0000	0.2321	0.1867	0.2094	10.85
^{100}Ru	0.0000	0.0000	0.1242	0.1511	0.1376	9.75
^{153}Sm	0.0000	0.0000	2.5744	1.7546	2.1645	18.94
^{116}Cd	0.0000	0.0000	0.2617	0.2859	0.2738	4.42
^{115}In	0.0000	0.0000	0.1825	0.1617	0.1721	6.05
^{149}Pm	0.0000	0.0000	0.7706	0.7371	0.7539	2.22
^{120}Sn	0.0000	0.0000	0.2113	0.2052	0.2082	1.46
^{121}Sb	0.0000	0.0000	0.2142	0.1598	0.1870	14.55
^{104}Pd	0.0000	0.2287	0.1014	0.1155	0.1485	38.36
^{155}Gd	0.0000	0.0000	1.3381	0.9688	1.1534	16.01
^{148}Pm	0.0000	0.0000	2.8793	2.1593	2.5193	14.29
^{136}Cs	0.0000	0.0000	1.2428	1.1047	1.1737	5.88
^{125}Te	0.0000	0.0000	0.9264	1.0398	0.9831	5.77
^{118}Sn	0.0000	0.0000	0.1708	0.1529	0.1618	5.54
^{81}Br	0.0000	0.0000	0.0754	0.0808	0.0781	3.44
^{125}Sn	0.0000	0.0000	0.9031	0.0000	0.9031	0.00
^{80}Se	0.0000	0.0000	0.0954	0.1055	0.1004	5.00
^{103}Rh	0.0000	0.0000	0.1620	0.2340	0.1979	18.19
^{123}Sb	0.0000	0.0000	0.2339	0.2306	0.2322	0.70
^{115}Cd	0.0000	0.0000	0.8746	0.0000	0.8746	0.00
^{111}Ag	0.0000	0.0000	0.2272	0.0000	0.2272	0.00
^{110}Cd	0.0000	0.0000	0.1120	0.0957	0.1038	7.88
^{96}Mo	0.0000	0.0000	0.1709	0.1648	0.1678	1.81
^{77}Se	0.0000	0.0000	0.2817	0.5030	0.3923	28.20
^{160}Tb	0.0000	0.0000	1.7044	0.0000	1.7044	0.00
^{130}Xe	0.0000	0.2918	0.1753	0.1914	0.2195	23.49
^{142}Nd	0.0000	0.0000	0.1202	0.1387	0.1295	7.14
^{78}Se	0.0000	0.0000	0.0655	0.0669	0.0662	1.05

6. CONCLUSIONS

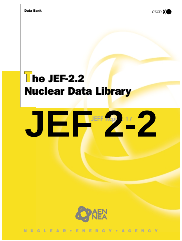
From the previous sections we may conclude that the considered data bases give a maximum spread in lumped one-group cross sections averaged over the spectrum of a large fast power reactor of about 6% for capture and 9% for inelastic scattering. The total reactivity effect has a maximum spread of 5.5%. These values indicate that the uncertainty in the reactivity effect is of the same order. Inspection of the differences did not reveal large problems in the libraries for capture and inelastic scattering cross sections. Slight improvements could be obtained by having closer inspection of the capture cross sections of the unstable nuclides ^{151}Sm, ^{103}Ru and ^{135}Cs. From the inelastic scattering cross sections some improvements could be obtained by investigating the data for nuclides with low-lying levels, below 200 keV (e.g. ^{101}Ru, ^{133}Cs, ^{103}Rh, ^{145}Nd, ^{103}Ru, ^{107}Pd and ^{149}Sm).

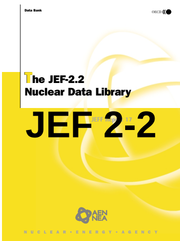
However, improvements for individual fission products will not easily solve the remaining discrepancies, since there are also systematic differences. These are probably caused by the evaluation method. For capture the unresolved resonance treatment with the difficulties in parametrization of strength functions, level spacing and radiation widths due to missed resonances, and the treatment of width fluctuations could be important. Also the background correction in the resolved energy range can play a role.

For the inelastic cross sections, the optical model choice and parametrization as well as width-fluctuation factor treatment could be different in various evaluations. It is not easy to assess such differences in the evaluation methods.

A possible point to be further investigated is the systematic difference of 5 to 6% in the total, elastic and capture cross sections between JENDL-3.2 and JEF-2.2. A common reason could be the optical-model parametrization or the systematics used for the (p-wave) strength functions. More generally, it is suggested to compare the yield-averaged values of S_0, S_1, Γ_γ and D_0 of the various evaluations, to assess possible reasons for systematic differences. These parameters determine the unresolved resonance range and are difficult to extract from resolved resonance parameters if there is a significant fraction of missed (p-wave) resonances. The optical model should match the s- and p-wave strength functions. In the high-energy part of the resolved-resonance range, a correction for missed resonances should be made. All these points are well known to the evaluators, but in order to make progress it is necessary to re-investigate such possible sources of differences.

Finally, in spite of the above mentioned problems, the status of lumped fission-product cross sections is satisfactory, although some improvements are possible as indicated in this report.


The JEF-2.2
Nuclear Data Library
JEF 2-2



APPENDIX A. NUCLIDE CONCENTRATIONS

Table A.1 *Nuclides and concentrations in the lumped fission product. For the benchmark, the yields of ^{239}Pu were used.*

Nuclide	Identification	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu
^{76}Ge	320760	.00015	.00001	.00011	.00001	.00000
^{75}As	330750	.00007	.00000	.00000	.00000	.00000
^{77}Se	340770	.00029	.00004	.00013	.00014	.00010
^{78}Se	340780	.00056	.00014	.00035	.00029	.00019
^{79}Se	340790	.00091	.00040	.00053	.00053	.00037
^{80}Se	340800	.00151	.00087	.00090	.00091	.00068
^{82}Se	340820	.00390	.00250	.00235	.00209	.00171
^{81}Br	350810	.00246	.00157	.00137	.00148	.00109
^{82}Br	350820	.00000	.00000	.00000	.00000	.00000
^{82}Kr	360820	.00006	.00003	.00006	.00003	.00002
^{83}Kr	360830	.00601	.00390	.00348	.00312	.00245
^{84}Kr	360840	.01084	.00559	.00565	.00458	.00377
^{85}Kr	360850	.01380	.00715	.00655	.00589	.00501
^{86}Kr	360860	.01928	.01382	.00878	.00793	.00674
^{85}Rb	370850	.00029	.00015	.00014	.00012	.00010
^{86}Rb	370860	.00001	.00000	.00000	.00000	.00000
^{87}Rb	370870	.02548	.01700	.01150	.01019	.00884
^{86}Sr	380860	.00004	.00000	.00001	.00000	.00000
^{88}Sr	380880	.03641	.02102	.01421	.01281	.01135
^{89}Sr	380890	.01278	.00956	.00543	.00499	.00424
^{90}Sr	380900	.05075	.03115	.01875	.01964	.01752
^{89}Y	390890	.03156	.02202	.01285	.01110	.00960
^{90}Y	390900	.00001	.00001	.00001	.00001	.00000
^{91}Y	390910	.01763	.01332	.00864	.00847	.00746
^{90}Zr	400900	.00043	.00026	.00016	.00016	.00014
^{91}Zr	400910	.03532	.02499	.01660	.01537	.01376
^{92}Zr	400920	.05639	.04312	.03061	.02930	.02579
^{93}Zr	400930	.05964	.04700	.03791	.03531	.03153
^{94}Zr	400940	.06449	.05107	.04274	.04232	.03773
^{95}Zr	400950	.02315	.02063	.01757	.01773	.01701
^{96}Zr	400960	.06553	.05744	.04939	.05027	.04737
^{95}Nb	410950	.01146	.00991	.00854	.00839	.00811
^{95}Mo	420950	.02925	.02417	.02118	.01998	.01955
^{96}Mo	420960	.00060	.00025	.00025	.00023	.00020
^{97}Mo	420970	.05848	.05804	.05200	.05280	.05080
^{98}Mo	420980	.06136	.06127	.05650	.05568	.05495
^{99}Mo	420990	.00091	.00112	.00099	.00108	.00101
^{100}Mo	421000	.06355	.06151	.06439	.06041	.05951
^{99}Tc	430990	.05349	.06007	.05500	.05552	.05294
^{100}Ru	441000	.00152	.00165	.00153	.00149	.00144
^{101}Ru	441010	.05327	.05944	.06387	.05848	.05876

Table A.1 *Table continued.*

Nuclide	Identification	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu
^{102}Ru	441020	.04816	.06361	.06834	.06165	.06099
^{103}Ru	441030	.00796	.01479	.01592	.01549	.01526
^{104}Ru	441040	.02366	.05702	.06499	.05791	.05900
^{106}Ru	441060	.00365	.02337	.03378	.04131	.04267
^{103}Rh	451030	.02509	.04346	.04810	.04391	.04404
^{105}Rh	451050	.00013	.00034	.00048	.00055	.00056
^{104}Pd	461040	.00067	.00112	.00126	.00112	.00113
^{105}Pd	461050	.01426	.03348	.04837	.05230	.05430
^{106}Pd	461060	.00149	.00679	.01105	.01148	.01200
^{107}Pd	461070	.00172	.01234	.03083	.04078	.04110
^{108}Pd	461080	.00078	.00684	.02415	.03321	.03912
^{110}Pd	461100	.00045	.00139	.00923	.01200	.01691
^{109}Ag	471090	.00054	.00201	.01632	.02139	.02758
^{111}Ag	471110	.00001	.00004	.00021	.00030	.00045
^{110}Cd	481100	.00001	.00005	.00042	.00052	.00067
^{111}Cd	481110	.00027	.00073	.00410	.00558	.00854
^{112}Cd	481120	.00039	.00071	.00135	.00288	.00433
^{113}Cd	481130	.00034	.00055	.00091	.00150	.00222
^{114}Cd	481140	.00034	.00046	.00100	.00102	.00103
^{115}Cd	481150	.00000	.00001	.00001	.00001	.00001
^{115m}Cd	481151	.00001	.00001	.00008	.00008	.00007
^{116}Cd	481160	.00036	.00035	.00064	.00085	.00100
^{115}In	491150	.00020	.00037	.00089	.00086	.00097
^{115}Sn	501150	.00001	.00002	.00004	.00004	.00005
^{116}Sn	501160	.00000	.00001	.00002	.00002	.00002
^{117}Sn	501170	.00037	.00034	.00064	.00085	.00090
^{118}Sn	501180	.00039	.00034	.00065	.00080	.00087
^{119}Sn	501190	.00042	.00034	.00066	.00080	.00086
^{120}Sn	501200	.00044	.00034	.00067	.00085	.00085
^{122}Sn	501220	.00048	.00037	.00069	.00095	.00093
^{123}Sn	501230	.00030	.00022	.00039	.00063	.00057
^{124}Sn	501240	.00060	.00042	.00120	.00120	.00111
^{125}Sn	501250	.00004	.00005	.00011	.00010	.00007
^{126}Sn	501260	.00104	.00100	.00303	.00316	.00240
^{121}Sb	511210	.00045	.00035	.00066	.00083	.00088
^{122}Sb	511220	.00000	.00000	.00000	.00000	.00000
^{123}Sb	511230	.00024	.00017	.00030	.00046	.00043
^{124}Sb	511240	.00000	.00000	.00000	.00000	.00000
^{125}Sb	511250	.00063	.00066	.00163	.00138	.00095
^{126}Sb	511260	.00001	.00000	.00001	.00001	.00000
^{127}Sb	511270	.00005	.00004	.00012	.00010	.00008

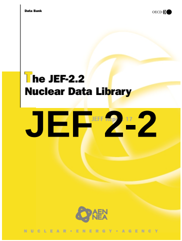


Table A.1 *Table continued.*

Nuclide	Identification	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu
^{122}Te	521220	.00001	.00001	.00002	.00002	.00002
^{123}Te	521230	.00000	.00000	.00000	.00000	.00000
^{124}Te	521240	.00000	.00000	.00000	.00000	.00000
^{125}Te	521250	.00005	.00005	.00012	.00009	.00007
^{125m}Te	521251	.00001	.00001	.00001	.00001	.00001
^{126}Te	521260	.00006	.00001	.00008	.00005	.00002
^{127m}Te	521271	.00017	.00013	.00045	.00033	.00028
^{128}Te	521280	.00321	.00506	.00790	.00618	.00520
^{129m}Te	521291	.00108	.00189	.00230	.00215	.00179
^{130}Te	521300	.01439	.01959	.01956	.01889	.01534
^{132}Te	521320	.00089	.00095	.00104	.00104	.00094
^{127}I	531270	.00174	.00132	.00446	.00318	.00277
^{129}I	531290	.00442	.00713	.00897	.00783	.00664
^{131}I	531310	.00153	.00185	.00218	.00184	.00150
^{128}Xe	541280	.00005	.00003	.00012	.00008	.00007
^{129}Xe	541290	.00000	.00000	.00000	.00000	.00000
^{130}Xe	541300	.00008	.00012	.00018	.00013	.00011
^{131}Xe	541310	.03063	.03391	.04139	.03224	.02683
^{132}Xe	541320	.04761	.04546	.05273	.04823	.04405
^{133}Xe	541330	.00206	.00184	.00223	.00199	.00191
^{134}Xe	541340	.07155	.06172	.07229	.06270	.06102
^{136}Xe	541360	.05993	.06292	.06776	.07121	.07112
^{133}Cs	551330	.06293	.05152	.06441	.05353	.05231
^{134}Cs	551340	.00169	.00120	.00153	.00123	.00121
^{135}Cs	551350	.06393	.05627	.07424	.06772	.06705
^{136}Cs	551360	.00004	.00003	.00009	.00007	.00005
^{137}Cs	551370	.06228	.06949	.06342	.06948	.07181
^{134}Ba	561340	.00014	.00008	.00011	.00008	.00008
^{135}Ba	561350	.00000	.00000	.00000	.00000	.00000
^{136}Ba	561360	.00024	.00016	.00082	.00056	.00031
^{137}Ba	561370	.00047	.00051	.00057	.00050	.00052
^{138}Ba	561380	.06692	.06477	.04936	.06465	.06738
^{140}Ba	561400	.00433	.00493	.00414	.00456	.00461
^{139}La	571390	.07087	.06083	.06053	.05802	.06093
^{140}La	571400	.00057	.00065	.00054	.00060	.00061
^{140}Ce	581400	.05322	.05506	.04787	.04868	.05031
^{141}Ce	581410	.01178	.01124	.01176	.01027	.01074
^{142}Ce	581420	.05878	.05054	.04924	.05302	.04990
^{143}Ce	581430	.00048	.00042	.00039	.00047	.00050
^{144}Ce	581440	.03849	.03424	.02642	.03057	.03336
^{141}Pr	591410	.04854	.04277	.04611	.03757	.04006

Table A.1 *Table continued.*

Nuclide	Identification	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu
^{143}Pr	591430	.00467	.00403	.00372	.00452	.00481
^{142}Nd	601420	.00028	.00024	.00026	.00021	.00022
^{143}Nd	601430	.05270	.04163	.03970	.04464	.04854
^{144}Nd	601440	.01301	.01104	.00876	.00975	.01075
^{145}Nd	601450	.03778	.04086	.03010	.03268	.03477
^{146}Nd	601460	.03016	.03897	.02557	.02847	.03051
^{147}Nd	601470	.00133	.00186	.00141	.00161	.00176
^{148}Nd	601480	.01691	.02359	.01715	.01814	.01929
^{150}Nd	601500	.00715	.01464	.01039	.01157	.01290
^{147}Pm	611470	.01651	.02131	.01664	.01768	.01979
^{148}Pm	611480	.00004	.00005	.00004	.00004	.00005
^{148m}Pm	611481	.00020	.00025	.00019	.00020	.00023
^{149}Pm	611490	.00015	.00025	.00019	.00023	.00024
^{147}Sm	621470	.00135	.00168	.00133	.00137	.00155
^{148}Sm	621480	.00086	.00104	.00083	.00084	.00096
^{149}Sm	621490	.01017	.01620	.01265	.01404	.01520
^{150}Sm	621500	.00076	.00117	.00093	.00099	.00109
^{151}Sm	621510	.00400	.00928	.00761	.00851	.00911
^{152}Sm	621520	.00350	.00751	.00759	.00850	.00930
^{153}Sm	621530	.00002	.00006	.00007	.00008	.00009
^{154}Sm	621540	.00097	.00213	.00321	.00405	.00447
^{153}Eu	631530	.00182	.00390	.00463	.00508	.00574
^{154}Eu	631540	.00020	.00042	.00051	.00054	.00061
^{155}Eu	631550	.00034	.00105	.00214	.00267	.00295
^{156}Eu	631560	.00002	.00009	.00020	.00027	.00030
^{154}Gd	641540	.00000	.00001	.00001	.00001	.00001
^{155}Gd	641550	.00001	.00004	.00009	.00011	.00012
^{156}Gd	641560	.00017	.00071	.00166	.00215	.00246
^{157}Gd	641570	.00007	.00027	.00093	.00122	.00139
^{158}Gd	641580	.00006	.00021	.00089	.00111	.00135
^{160}Gd	641600	.00001	.00003	.00033	.00033	.00039
^{159}Tb	651590	.00003	.00008	.00043	.00053	.00062
^{160}Tb	651600	.00000	.00000	.00002	.00002	.00002
^{161}Tb	651610	.00000	.00000	.00001	.00002	.00001
^{160}Dy	661600	.00000	.00000	.00001	.00002	.00002
^{161}Dy	661610	.00000	.00001	.00020	.00029	.00020
^{162}Dy	661620	.00000	.00000	.00002	.00003	.00002
^{163}Dy	661630	.00000	.00000	.00000	.00000	.00000
^{164}Dy	661640	.00000	.00000	.00000	.00000	.00000
Total		2.00002	2.00002	2.00004	2.00002	2.00003