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International Reactor Dosimetry File IRDF-2002

Final steps in preparation of the library

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Final Report on IAEA Research Contract No. 11455

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INSTITUTE OF NUCLEAR TECHNIQUES
BUDAPEST UNIVERSITY OF
TECHNOLOGY AND ECONOMICS

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FINAL REPORT

IAEA Research Contract No 11455/R2

E. M. Zsolnay, H. J. Nolthenius*

* The Netherlands

Budapest, 2004 May

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INTRODUCTION

As the result of updating the old reactor dosimetry file IRDF-90, a new reactor dosimetry library "International Reactor Dosimetry File: IRDF-2002" has been developed. The work was done by an international team – under the co-ordination of IAEA NDS – with the following participants:

- International Atomic Energy Agency, Nuclear Data Section (IAEA NDS);
- Institute of Nuclear Techniques (INT), Budapest University of Technology and Economics (BUTE), Budapest, Hungary;
- Institute of Physics and Power Engineering (IPPE), Obninsk, Russia;
- Physikalisch Technische Bundesanstalt (PTB) Braunschweig, Germany;
- Centre d`Etudes Nucleaires (CEA), Bruyeres-le-Chatel, France;
- Sandia National Laboratories (SNL), Albuquerque, USA;
- Pacific Northwest Laboratory (PNL), Richland, USA;
- Nuclear Data Center (NDC), JAERI, Japan.

The library – together with the documentation in the form of a TECDOC – has been completed in February 2004. The new library contains:

- cross sections accompanied with uncertainty information for 66 dosimetry reactions (see in Table 7);
- total cross sections for three cover materials (B, Cd, Gd);
- radiation damage cross sections for some elements and compounds;
- nuclear data for dosimetry application, based on the most up-to-date ENSDF data.

This report presents the final steps in the development of the file IRDF-2002, performed by INT BUTE. The programme of the work – in agreement with the corresponding IAEA contract – was as follows:

1. "Culling of the selected candidate data files by applying criteria such as: magnitude of cross section uncertainty over the complete energy range of reaction, recent evaluation data, graphical display of the energy-dependent cross section should not reveal any unrealistic physical features, the evaluation should clearly indicate the residual product of reaction, format or comments must indicate if the cross section represents the reaction of interest or a sum of reaction products, available comments should include detailed references that permit the underlying calculations to be repeated, etc.
2. Standard benchmark fields (1/E slowing down spectrum in hydrogeneous moderator and the Maxwellian thermal spectrum at specified neutron temperature) will be used to differenciate between candidate evaluations.
3. Preparation of a preliminary version of IRDF-2002; send to NDS.

4. Prepare TECDOC chapters according to statements made in the report of last IRDF meeting. Data and final report to be made available to IAEA-NDS."

1. BACKGROUND

1.1. Detailed analysis and intercomparison of the data of different neutron cross section libraries

As a part of the updating procedure of the International Reactor Dosimetry File IRDF-90 [1], the data of up-to-date reactor dosimetry files (JENDL/D-99 [2], original and up-dated RRDF-98 [3]) and new evaluations in the files ENDF/B-VI (V.8), JEFF-3.0 and CENDL-2 [4] have been analysed, in order to select the best quality cross section and related uncertainty information for the new library, IRDF-2002. The analysis involved the following actions:

- Checking the content and formats of the cross section and uncertainty information in the files of interest;
- Numerical characterization of the cross section data by spectrum averaged cross section values for three theoretical spectrum functions: Maxwellian thermal, 1/E and Watt fission neutron spectrum;
- Intercomparison of the above integral spectrum characteristics;
- Analysis of the uncertainty information of the different libraries (including detailed analysis of the covariance matrices), and intercomparison of the corresponding uncertainty data.

The results, together with the detected errors, discrepancies and shortcomings (related sometimes to the physics and/or to the mathematics content, sometimes to the format of the data) were presented in the form of Progress Reports [5-7] and, communicated to the evaluators of the libraries via IAEA NDS. The evaluators then revised and modified numerous data in the cross section files JENDL/D-99 and IRDF-98, furthermore, a number of new evaluations have been prepared [8-10].

Altogether about 180 different cross section data were analysed (several ones more times, due to the revisions). It has turned out that, for several reactions no better quality cross section evaluations are available in the literature than the data in the file IRDF-90, and only a limited number of new evaluations accompanied by covariance information (the majority of them for RRDF) have been prepared in the energy region from thermal to 20 MeV, during the last decade.

1.2. New evaluations

New cross section evaluations with uncertainty information have been prepared for inclusion in IRDF-2002, by IPPE, Obninsk [8,9] for the following reactions: ^{27}Al(n,p), ^{58}Ni(n,p), ^{103}Rh(n,n'), ^{115}In(n,n'), ^{139}La(n,γ) ^{186}W(n,γ), ^{204}Pb(n,n') and ^{237}Np(n,f). Also these data were analysed in the way described above.

1.3. Candidate cross sections for IRDF-2002

After repeated investigation of the revised data and analysis of the new data [6], a collection was prepared for each of the libraries mentioned above, containing the reactions with cross sections suitable for inclusion in IRDF-2002 [6-7,11]. Table 1 shows the last updated list of these candidates [11]. The cross sections together with the uncertainty information, listed in this Table, are the best quality data available in the open literature at the end of 2002 - first part of 2003 and, the content of the new international reactor dosimetry file has been selected from them.

A file with the corrected data of the candidate cross sections has been compiled and sent to IAEA NDS [12].

There are some reactions interesting for dosimetry applications, but a part of them has shortcomings in the cross section information, while for another part no suitable cross section data were found in the open literature. These reactions are also listed in Table 1.

2. INTERCOMPARISON OF THE CANDIDATE CROSS SECTION DATA WITH EXPERIMENTAL VALUES (C/E) IN STANDARD NEUTRON FIELDS, AND PRELIMINARY SELECTION OF THE DATA FOR IRDF-2002

The next step in preparation of the file IRDF-2002 was the selection of the best quality data from among the candidate cross sections in Table 1. The final decision on the content of the new libray was preceded by a preliminary selection process.

The base of the cross section pre-selection procedure was the comparison of the integral values of the cross sections from the libraries of interest with each other, and with experimental data of standard neutron fields. Another important criterion of the

selection was the quality of the uncertainty information accompanying the cross sections of interest, and the consistency of the data. Therefore, integral (spectrum averaged) cross section values and the related uncertainties were calculated for the reactions of Table 1 in three (standard) neutron fields: Maxwellian thermal, 1/E and ^{235}U fission neutron field [13-14]. The data were then compared with each other and with up-to-date experimental ones (C/E) [15-17]. Based on the results, a recommendation was made on the cross section information to be included in IRDF-2002.

2.1. C/E values and pre-selection of the cross section data in standard thermal and 1/E neutron fields

An intensive review of the literature identified two sources that contained up-to-date experimental data in the Maxwellian thermal and in the 1/E neutron fields. These were the evaluated experimental data of S. F. Mughabgbab [15] and of N. E. Holden [16]. The thermal neutron cross sections in both evaluations of experiments refer to a neutron energy of 0.0253 eV (v_o=2200 m/s), while the resonance integrals were calculated by Mughabgbab with a lower energy limit of 0.5 eV and, with an upper energy limit corresponding to the upper resonance with known scattering width [18]. Holden calculated the resonance integrals from 0.5 eV to 0.1 MeV. In our calculations the thermal neutron cross sections refer to 0.0253 eV neutron energy (v_o=2200 m/s), and the resonance integrals were calculated from 0.5 eV to 1.05 MeV (see also [5] and [6]), with using a multigroup representation (SAND type 640 energy groups) of the the cross sections of interest.

The uncertainty information for the cross sections of interest have been represented by the corresponding relative standard deviation values (with the same energy boundaries as used in the cross section calculations), weighted with a typical MTR spectrum (see Figure 1 [19]).

The results of the comparison can be seen in Table 2. From the data of this Table one can see that, for a few reactions the same cross section information was present both in IRDF-90 and in the other libraries of interest, while the related uncertainty information was sometimes different. This type of reactions are: NA23G, MN55G, CU63G, NB93G,

IN115G, TH232G and PU239F. When the corresponding cross section and uncertainty values were identical, the file IRDF-90 was taken as the source of the data for IRDF-2002.

In case of the reaction FE58G the resonance integral for both cross section files of interest (IRDF-90 and JENDL/D-99) meaningfully deviates from the corresponding data of Mughabgbab, while the JENDL/D-99 value shows a good agreement (as compared with the relevant uncertainties) with the corresponding data of Holden. The situation needs further clarification, eg. by comparison with experimental data in benchmark neutron fields. Nevertheless, for the JENDL/D-99 data generally a better agreement was found with the experimental values than for the corresponding IRDF-90 ones. Taking into consideration the corresponding uncertainty values as well, the JENDL/D-99 data seem to be more realistic, therefore, it has been recommended that the cross section to be included in IRDF-2002 for this reaction should be taken from the file JENDL/D-99.

The other reaction in Table 2, being the subject of selection, is PU239F. As it can be seen, the cross section values for this reaction are practically the same in the considered libraries, but the uncertainties in the file JENDL/D-99 seem to be more reliable, than the corresponding IRDF-90 values. Therefore, again the JENDL/D-99 data have been selected for IRDF-2002.

2.1.1. Characterization of the recommended cross section data

The thermal and epithermal neutron cross sections, recommended for the file IRDF-2002, are presented by Table 3. For numerical characterization of the data the thermal cross sections (σ_L) at 2200 m/s (0.0253 eV) and the resonance integral (IR_L) values from 0.5 eV to 1.05 MeV have been calculated. All the cross section and resonance integral values are compared with the evaluated experimental data of S. F. Mughabgbab [15] and of N. E. Holden [1], as described in the previous chapter.

For representation of the uncertainty information of the selected cross sections, the relative standard deviation values (weighted with an MTR spectrum) were calculated for the thermal and intermediate neutron energy regions separately, using the same energy boundaries as in case of the cross section characterization. The results can be seen in Table 4.

Evaluating the data in Tables 3 and 4, the following statements can be made:

1. The thermal neutron cross sections for the selected reactions in general agree with the evaluated experimental data within one standard deviation of the corresponding library and experimental values.
2. However, the resonance integrals calculated from the library data deviate from the evaluated experimental ones more than one standard deviation of the corresponding library and experimental values, for several reactions (details see below).
3. List of the problems by reactions, related to the data in Tables 3 and 4:

   B10A and LI6T: the uncertainty of the library cross sections in the intermediate neutron energy region is to small (not realistic), as compared with the corresponding C/E values (or the library data deviate significantly from the experimental values).

   NA23G: The uncertainty information contains only a diagonal matrix – new evaluation is required.

   MN55G: The C/E value for the resonance integral is deviating by 16 % from the unity. It is too large deviation also as compared with the related uncertainty

values. New cross section evaluation is needed for this reaction in the intermediate neutron energy region.

FE58G: The C/E value for the resonance integral with the Mughabghab data is deviating by 19 % from unity. At the same time, a large difference is present between the experimental data of the sources considered. Clarification of the situation is needed, as this reaction is one of the most frequently used detectors in the reactor dosimetry. Maybe also new cross section evaluation is needed in the intermediate neutron energy region.

NB93G: The C/E value for the resonance integral is deviating by 17% from unity, furthermore, the uncertainty information contains only a diagonal matrix. New evaluation is needed.

AG109G: The C/E value shows a large deviation from unity both in the thermal and in the intermediate neutron energy regions, furthermore, the evaluated cross section in the library IRDF-90 is given in a rough energy group structure. Re-evaluation of the data is needed. (For this reaction Mughabgbab gave the sum of the cross sections of reactions leading to Ag110(m+g), while the dosimetry libraries contain the cross sections for the reaction leading to Ag110m. Therefore, no comparison with the data of Mughabgbab was possible in this case.)

IN115G: The uncertainty information contains only diagonal matrix. New evaluation is needed.

TA181G: The uncertainty information contains only a diagonal matrix. New evaluation is needed.

AU197G: The available uncertainty information for this reaction is not reliable, it has been withdrawn from ENDF/B-VI. The uncertainty data in IRDF-90 are deriving from the same source. New evaluation is needed.

TH232G: In the uncertainty information below 15 eV diagonal matrix is present. New evaluation is needed.

U235F: The uncertainty information has been declared to be not reliable and has been withdrawn from ENDF/B-VI. The data in the library IRDF-90 have the same origin. New evaluation is needed.

AM241F: No up-to-date experimental data are available for this reaction, therefore, the corresponding C/E values could not be derived.

2.1.2.Conclusions

Based on the results of the cross section selection procedure outlined above, the following conclusions can be drawn, related to the data of Tables 3 and 4:

1. Practically no new cross section evaluations have been made in the low neutron energy region during the last one-two decades, except the reactions ^{139}La(n,γ) and ^{186}W(n, γ), evaluated for the Russian Reactor Dosimetry File [8, 9 and 20].
2. In the thermal neutron energy region the selected cross sections show in most cases a very good agreement with the corresponding evaluated experimental values.
3. At the same time, the resonance integrals for the reactions MN55G, FE58G and NB93G, meaningfully (>10%) deviate from the corresponding experimental data. This deviation is too large even in comparison with the corresponding uncertainty information. Further investigations (eg. testing the data also in benchmark neutron fields) and new cross section evaluations will be needed in these cases.
4. For the reactions NB93G, IN115G, TA181G and for TH232G below 15 eV, the uncertainty information consists of diagonal covariance matrices only. New evaluations with complete covariance information are needed in these cases.
5. Unreliable uncertainty information (withdrawn from ENDF/B-VI) is present in all the investigated cross section libtraries for the reactions AU197G and U235F. Therefore, new cross section evaluations with complete covariance information are needed for these reactions.
6. Before the final decision on the content of IRDF-2002, also a consistency test will have to be made on the pre-selected cross sections in Tables 3 and 4, by comparing the relevant integral data in benchmark neutron fields.

2.2. Pre-selection of the cross section data for IRDF-2002 in the fast neutron energy region

For characterizing and comparing the cross section data of the fast neutron (threshold) reactions present in Table 1, spectrum averaged cross sections were calculated for the theoretical function of the Watt fission spectrum [19], with using a multigroup representation (SAND type 640 energy groups) of the the cross sections of interest. The uncertainty information for the cross sections was represented by the corresponding standard deviation values above 1.05 MeV, weighted with a typical MTR spectrum [19]. The results obtained can be seen in Table 5.

W. Mannhart calculated the responses of activation reactions in the standard neutron field of spontaneous fission of ^{252}Cf and, compared them with experimental data obtained in that neutron field [17]. Spectrum averaged cross sections were calculated together with the related standard deviations, and C/E values were derived. Also qualification of the considered cross section information was given. He investigated the cross section data of the files IRDF-90.v2, JENDL/D-99 and an updated version of RRDF-98, furthermore, the ones of the reactions selected in Table 1 from ENDF/B-VI and JEFF-3.0. His results – with consideration of the differences between the two spectrum functions – show a good agreement with the data in Table 5.

In the present chapter we give a recommendation on the fast neutron cross sections to be included in the file IRDF-2002, and characterize the selected data.

2.2.1. Recommended fast neutron cross sections for IRDF-2002 and characterization of the data

In the fast neutron energy region the applicable standard neutron field for characterization and selection of the cross section data, is the one of the spontaneous fission of ^{252}Cf. Therefore, our earlier findings (see [5,6] and Table 5) were combined with the ones of W. Mannhart [17], and the procedure resulted in the data of Table 6 [14]. This Table contains the list of the 46 fast neutron cross sections recommended to be included in IRDF-2002, together with their characteristics, taken from [17]. The column with the uncertainty values of the calculated average cross sections ($<\sigma_c>$) shows the

standard deviation values in $<\sigma_c>$ due to the cross sections, while the values in brackets give the total standard deviation of $<\sigma_c>$, including the contribution of the uncertainty of the ^{252}Cf spectrum function as well. The uncertainty of the C/E values involves the standard deviations present in the experimental data, in the cross sections of interest and, in the ^{252}Cf spectrum function. So they can be calculated based on the data of column 4 in brackets and, on the data of column 6.

The following conclusions can be drawn regarding the cross section values of this Table:

1. No up-to-date experimental cross section data were available for the reactions P31P, TI0XSC46, TI0XSC48, TI462, TI47NP, TI48NP, TI49NP, CR522, FE542, FE54A, AS752, Y892, IN1152 and PR1412. Therefore, the corresponding C/E values could not be derived. At the same time, a large deviation was found between the cross section values of the reaction TI47NP in the libraries RRDF-98 and IRDF-90. Clarification of the situation is needed.

2. C/E values larger than 5 per cent are present for the following reactions: CU632, RH103N, I1272, TM1692, HG199N, and TH232F. The relatively large deviation between the experimental and calculated cross section values originates from the side of the library cross sections, except the reaction TM1692, where the measured cross section has a large (~ 6%) uncertainty. Improvement of the situation would be useful.

3. Inconsistency is present between the C/E value and the corresponding uncertainties for the reaction MG24P. Clarification of the situation is needed.

4. The uncertainties of the cross sections in Table 6 is in most cases below 4 %. Larger uncertainty values are present in case of the ractions TI47NP , TI48NP, TI48P, TI49NP, NI60P, AS752, PR1412, HG199N, TH232F. During the neutron spectrum adjustment procedure in the energy regions with responses of more detectors, these reactions will have a much smaller weight than the other ones with meaningfully smaller uncertainties.

5. The selected cross sections in Table 6 will have to go through a consistency test as well, by comparing the relevent integral data with experimental values in benchmark neutron fields.

3. FINAL SELECTION OF THE CROSS SECTIONS FOR IRDF-2002

The final selection of the cross sections for the International Reactor Dosimetry File IRDF-2002 was made in frame of a Technical Meeting, held on the project at IAEA NDS, Vienna, from 1 to 3 October 2003 [21]. The selection procedure was based on the following considerations (applied also in the preliminary selection process):

- comparison of the integral values of the candidate cross sections with the corresponding experimental ones in the four standard neutron fields (thermal Maxwellian, 1/E slowing down, ^{252}Cf fission and a 14 MeV neutron field) recommended for the purpose of cross section selection [22];
- quality of the uncertainty information;
- consistency of the data.

The result – final content of the new library – can be seen in Table 7. Characterization of the cross section data present in IRDF-2002 is given in Table 8.

4. PREPARATION OF THE CROSS SECTION DATA IN THE "IRDF" FORMAT

Table 7 gives the list of the cross sections together with their origin, present in the file IRDF-2002. The library is available in two forms:

- point cross section data, and
- group cross section data in the SAND II extended 640 energy group structure.

The group-wise data are given in a simplified ENDF-6 format (referred further on as "IRDF" format). The simplified format means that all the cross section information is given in File 3 (ENDF-6 notation), and the uncertainty information is given in the form of file 33 (ENDF-6 notatin).

The neutron temperature, used to calculate the group cross section data, was 300K.

The conversion of the point cross section data to the 640 group ENDF-6 format was made by IAEA NDS. To obtain the IRDF format library from these data, several additional conversions, corrections and modifications had to be introduced. This chapter presents the work done in this field in frame of the IAEA contract.

The work involved the following actions:

- *Conversion of the ENDF-6 format cross section data to the simplified IRDF format* (converting the data of MF=10 to MF=3, and changing the MT numbers for the reaction cross sections leading to metastable states of the reaction product nuclei).
- *Conversion of the cross section uncertainty information to the simplified IRDF format* (converting the data of MF=40 and MF=32 to MF=33, changing the format of the uncertainty data in accordance with the actions described in the former point).
- *Correction of the lower energy limit of the uncertainty information for several threshold reactions*, in order that the cross section and related uncertainty data should refer to the same energy region.
- *Calculation of integral characteristics for the newly generated group cross section and uncertainty data.*
- *Converting the format of the damage cross sections to the format of the file IRDF-2002, and adding the converted damage data to the cross section library.*
- *Preperation of the covariance matrices of the cross sections in IRDF-2002 in a 27 groups structure*, to be presented in the TECDOC.

Results

Table 9 shows the MAT, MF and MT numbers for the cross sections present in the file IRDF-2002.

The original and corrected lower energy limits of the cross section and uncertainty data for the threshold reactions can be seen in Table 10.

Some integral characteristics of the newly generated group cross section data (derived from the corresponding point values in frame of this project) are given in Table 11. Characteristic cross section values have been calculated in three energy groups: in the thermal neutron energy region the cross sections are given at 2200 m/s; resonance integrals have been calculated from 0.5 eV to 1.05 MeV; and in the fast neutron energy region the cross sections were averaged over the Watt fission spectrum (details see e.g. in [6]). The corresponding uncertainties were calculated in the same energy group structure as applied at the cross sections, except the fast neutron energy region, where the lower energy limit was 1.05 MeV. The results can be seen in Table 12. A typical MTR spectrum (see [19] and Figure 1) was used as weighting spectrum in the uncertainty

calculations. These data can be used for intercomparison with the group cross section values from the source libraries of the selected cross section data.

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6. TABLES AND FIGURES

Table 1. Reactions from the different libraries with cross sections suitable for IRDF-2002

REACTIONS FROM IRDF-90						
LI6T	B10A	MG24P	AL27P	AL27A	P31P	S32P
SC45G	TI46P	TI47NP	TI47P	TI48NP	TI48P	CR522
MN55G	FE54P	FE58G	CO592	CO59G	NI582	NI58P
CU632	CU63G	CU63A	CU652	ZN64P	ZR902	NB932*
NB93N*	RH103N*	AG109G*	IN1152*	IN115N*	I1272	AU1972
AU197G	TH232F	U235F	U238F	U238G	PU239F	
NA23G+	NB93G+	IN115G*+	TH232G+			*45 reactions*
REACTIONS FROM JENDL/D-99						
F192	MG24P	AL27P	AL27A	P31P	TI0XSC46	TI0XSC48
TI462	TI46P	TI48NP	TI48P	TI49NP	CR522	MN55G
FE54P	FE58G	NI582	NI58P	CU632	CU652	Y892
ZR902	IN115N	I1272	TM1692	AU1972	HG199N	U238F
NP237F	PU239F	AM241F	NA232•	TA181G+		
						33 reactions
REACTIONS FROM RRDF-98						
F192	TI462	TI46P	TI47NP	TI48NP	TI48P	TI49NP*
V51A	FE542	FE54A	FE56P	CO59A	CU63A	AS752
NB932	NB93N	LA139G	PR1412	W186G	PB204N	
AL27P•	NI58P•	RH103N•	IN115N•	NP237F •		*25 reactions*
REACTIONS FROM ENDF/B-VI (V.8)						
CR522	NI58P	NI60P	CU632	CU63G	CU652	
NA23G+	NB93G+	IN115G*+	TH232G+			*10 reactions*
REACTIONS FROM JEFF-3.0						
FE56P	NI582	NI58P	NI60P			
						4 reactions

S= 117 reactions

Problematic reactions						
NA23G	TI0XSC47	CR50G	MN552	FE57NP	NB93G	IN115G
EU151G	TA181G	TH232G				*10 reactions*

REMARKS

- New evaluations or updates, 2003.

* Metastable state of the reaction product nuclide.

\+ Diagonal covariance matrix

1) The SAND type short reaction names in the Table have to be interpreted as follows: the chemical symbol and mass number of the target nucleus is followed by the name of the reaction product. The letters A, G, F, 2, N, P, NP and T mean (n,α), (n,γ), (n,f), (n,2n),(n,p), (n,np) and (n,t) reactions, respectively. TI0XSC-46, -47 and-48 indicate the reactions on natural Ti target leading to ^{46}Ti, ^{47}Ti and ^{48}Ti, respectively.

2) No suitable cross section data have been found for the reactions: TI0XSC47, CR50G, MN552, FE57NP and EU151G.

3) Only diagonal covariance matrices are available in all the investigated libraries for the reactions: NA23G, NB93G, IN115G, TA181G and TH232G (below 15 eV).

Table 2. Comparison of the cross section characteristics for some thermal and epithermal neutron reactions, with evaluated experimental data

Reaction code	Library	Calc.cross. sec. σ_L (2200 m/s) (m^2)	Rel. std. of σ_L* (%)	Calc. res. int. IR_L♣ (m^2)	Rel. std. in intermed. E region** (%)	Thermal cross sec. ratio		Res. int. ratio	
						σ_L/σ_M	σ_L/σ_H	IR_L/IR_M	IR_L/IR_H
NA23G+	IRDF-90	5.28-29	2.00	3.16-29	3.15	0.99	---	1.02	1.06
NA23G+	ENDF/B-VI	5.28-29	2.00	3.16-29	3.15	0.99	---	1.02	1.06
MN55G	IRDF-90	1.34E-27	4.18	1.18E-27	3.84	1.00	1.01	0.84	0.84
MN55G	JENDL/D-99	1.34E-27	6.31	1.18E-27	8.04	1.00	1.01	0.84	0.84
FE58G	IRDF-90	1.15E-28	5.07	1.51E-28	5.12	0.88	0.88	0.89	1.16
FE58G	JENDL/D-99	1.30E-28	12.60	1.37E-28	8.75	1.00	1.00	0.81	1.05
CU63G	IRDF-90	4.48E-28	4.11	4.96E-28	3.86	0.99	1.00	1.00	0.99
CU63G	ENDF/B-VI	4.48E-28	4.11	4.95E-28	3.86	0.99	1.00	1.00	0.99
NB93G+	IRDF-90	1.16E-28	10.0	9.92-E28	9.49	1.01	1.05	1.17	1.17
NB93G+	ENDF/B-VI	1.16E-28	10.0	9.92-E28	9.49	1.01	1.05	1.17	1.17
IN115G+	IRDF-90	2.11E-26	6.00	3.28E-25	5.98	1.04	1.03	0.99	0.96
IN115G+	ENDF/B-VI	2.11E-26	6.00	3.28E-25	5.98	1.04	1.03	0.99	0.96
TH232G+	IRDF-90	7.40E-28	4.33	8.57E-27	10.92	1.01	1.00	1.01	1.01
TH232G+	ENDF/B-VI	7.40E-28	4.33	8.57E-27	10.92	1.01	1.00	1.01	1.01
PU239F	IRDF-90	7.48E-26	0.25	2.93E-26	0.26	---	0.99	---	0.98
PU239F	JENDL/D-99	7.47E-26	0.71	2.97E-26	3.82	---	0.99	---	0.99

REMARKS

σ_L and IR_L are calculated values from the corresponding library data.

* Calculated for a typical MTR spectrum from 1E-4 eV to 0.5 eV.

** Calculated for a typical MTR spectrum from 0.5 eV to 1.05 MeV.

♣ Calculated from 0.5 eV to 1.05 MeV.

\+ Diagonal covariance matrix.

σ_M and IR_M are evaluated experimental data of S.F. Mughabghab [15].

σ_H and IR_H are evaluated experimental data of N.E. Holden [16].

Table 3. Thermal neutron cross sections and resonance integrals for the reactions, recommended for IRDF-2002
(Neutron temperature 300 K)

Reaction code	Mat No.	MT No.	Library, source of selection	Libr.cross sec.at 2200 m/s σ_L (m^2)	Evaluated exp. data (2200 m/s) σ_M (m^2)	Evaluated exp. data (2200 m/s) σ_H (m^2)	Cross sec. ratio σ_L/σ_M	Cross sec. ratio σ_L/σ_H	Res. Int. from libr.data IR_L (m^2)	Evaluated res.integral IR_M (m^2)	Evaluated res.integral IR_H (m^2)	Res. int. ratio IR_L/IR_M	Res. int. ratio IR_L/IR_H
LI6T	325	105	IRDF-90	9.41E-26	Not available	9.4(.1)E-26	---	1.00	4.25E-26	Not available	4.22(.04)E-26	---	1.01
B10A	525	107	IRDF-90	3.84E-25	Not available	3.84(.01)E-25	---	1.00	1.72E-25	Not available	1.73(.01)E-25	---	0.99
SC45G	2126	102	IRDF-90	2.72E-27	2.72(.02)E-27	2.7E-27	1.00	1.01	1.19E-27	1.20(.05)E-27	1.20E-27	0.99	0.99
MN55G	2525	102	IRDF-90	1.34E-27	1.336(.005)E-27	1.33(.01)E-27	1.00	1.01	1.18E-27	1.40(.03)E-27	1.40(3)E-27	0.84	0.84
FE58G	2637	102	JENDL/D-99u	1.30E-28	1.30(.03)E-28	1.3(.1)E-28	1.00	1.00	1.37E-28	1.7(.1)E-28	1.3(.2)E-28	0.81	1.05
CO59G	2725	102	IRDF-90	3.73E-27	3.718(.006)E-27	3.72E-27	1.00	1.00	7.45E-27	7.59(.02)E-27	7.4E-27	0.98	1.01
CU63G	2925	102	IRDF-90	4.48E-28	4.52(.02)E-28	4.5(.2)E-28	0.99	1.00	4.96E-28	4.97(.08)E-28	5.0(1)E-28	1.00	0.99
NB93G+	4125	102	IRDF-90	1.16E-28	1.15(.05)E-28	1.1E-28	1.01	1.05	9.92E-28	8.5(.5)E-28	8.5E-28	1.17	1.17
AG109G	4731	102	IRDF-90	4.69E-28	---	4.2E-28	----	1.12	6.56E-27	---	7.0E-27	---	0.94
IN115G*+	4931	102	IRDF-90	2.11E-26	2.02(.02)E-26	2.05E-26	1.04	1.03	3.28E-25	3.3(.1)E-25	3.4E-25	0.99	0.96
LA139G	5712	102	RRDF-98 n	8.89E-28	9.04(.04)E-28	9.2(.2)E-28	0.98	0.97	1.19E-27	1.21(.06)E-27	1.2(.1)E-27	0.98	0.99
TA181G+	7328	102	JENDL/D-99	2.07E-27	2.05(.05)E-27	2.01E-27	1.01	1.04	6.59E-26	6.6(.23)E-26	6.504E-26	1.00	1.01
W186G	7452	102	RRDF-98 n	3.79E-27	3.85(.05)E-27	3.7(.2)E-27	0.98	1.02	4.79E-26	4.85(.15)E-26	5.10(.50)E-26	0.99	0.94
AU197G♣	7925	102	IRDF-90	9.89E-27	9.865(.09)E-27	9.87(.1)E-27	1.00	1.00	1.57E-25	1.55(.028)E-25	1.55(.03)E-25	1.01	1.01
TH232G♦	9040	102	IRDF-90	7.40E-28	7.35(.03)-28	7.37(.04)-28	1.01	1.00	8.57E-27	8.5(.3)E-27	8.5(.3)E-27	1.01	1.01
U235F♥	9228	18	IRDF-90	5.86E-26	Not available	5.86(.02)E-26	---	1.00	2.74E-26	Not available	2.75(.05)E-26	---	1.00
U238G	9237	102	IRDF-90	2.71E-28	2.68(.019)E-28	2.7(.1)E-28	1.01	1.00	2.77E-26	2.77(.03)E-26	2.77(.03)E-26	1.00	1.00
PU239F	9437	18	JENDL/D-99	7.47E-26	Not available	7.52(.03)E-26	---	0.99	2.97E-26	Not available	3.0(.1)E-26	---	0.99
AM241F	9543	18	JENDL/D-99	3.03E-28	Not available	3.15(.1)E-28	---	0.99	7.84E-28	Not available	Not available	---	---

REMARKS

At the evaluated experimental cross section data the values in brackets mean the "absolute" uncertainties (one standard deviation).

\+ Diagonal matrix

♣ For the reaction AU197G the uncertainty information has been withdrawn from ENDF/B-VI (similar old evaluation is present in IRDF-90) .

♦ For the reaction TH232G below 15 eV diagonal matrix is present.

♥ The uncertainty information for the reaction U235F is not reliable, it has been withdrawn from ENDF/B-VI.

n Means new evaluation, u means up-dated data.

The subscripts L, M and H mean library data, and the evaluated experimental data of S.F. Mughabghab [15] and N.E. Holden [16], respectively.

Table 4. Relative standard deviation values averaged over a typical MTR spectrum for the reactions in Table 3, selected for IRDF-2002.

Reaction code	Library, source of selection	Mat.MT No.	Relative std. (%) For the spectrum part	
			Thermal°	Intermediate*
LI6T	IRDF-90	0325.105	0.14	0.14
B10A	IRDF-90	0525.107	0.16	0.16
SC45G	IRDF-90	2126.102	0.73	0.76
MN55G	IRDF-90	2525.102	4.18	3.84
FE58G	JENDL/D-99u	2637.102	12.60	8.75
CO59G	IRDF-90	2725.102	0.66	0.77
CU63G	IRDF-90	2925.102	4.11	3.86
NB93G+	IRDF-90	4125.102	10.00	9.49
AG109G	IRDF-90	4731.102	5.10	6.90
IN115G+	IRDF-90	4931.102	6.00	5.98
LA139G	RRDF-98 u	5712.102	3.87	5.50
TA181G+	JENDL/D-99	7328.102	3.00	3.77
W186G	RRDF-98 u	7452.102	2.31	3.32
AU197G♣	IRDF-90	7925.102	0.14	0.17
TH232G♦	IRDF-90	9040.102	4.33	10.92
U235F♥	IRDF-90	9228.018	0.19	0.27
U238G	IRDF-90	9237.102	0.35	0.37
PU239F	JENDL/D-99	9437.018	0.71	3.82
AM241F	JENDL/D-99	9543.018	2.00	1.56

REMARKS

° From 1E-4 eV to 0.5 eV.

* From 0.5 eV to 1.05 MeV.

u Means up-dated data.

\+ Diagonal matrix

♣ For the reaction AU197G the uncertainty information has been withdrawn from ENDF/B-VI (similar old evaluation is present in IRDF-90) .

♦ For the reaction TH232G below 15 eV diagonal matrix is present.

♥ The uncertainty information for the reaction U235F is not reliable, it has been withdrawn from ENDF/B-VI.

Table 5. Comparison of the cross section characteristics for the fast neutron reactions candidates for IRDF-2002, in the Watt fission neutron spectrum

Reaction code	Library	Cross. sec. $<\sigma_f>$ (m^2)	Rel. std. of $<\sigma_f>$* (%)
F192	JENDL/D-99	6.773E-34	2.92
F192	RRDF-98(u)	5.855E-34	3.02
MG24P	IRDF-90	1.473E-31	2.26
MG24P	JENDL/D-99	1.488E-31	1.24
AL27P	IRDF-90	3.825E-31	3.31
AL27P	JENDL/D-99	4.224E-31	0.72
AL27P	RRDF-98(new)	3.980E-31	2.06
AL27A	IRDF-90	6.860E-32	1.37
AL27A	JENDL/D-99	6.860E-32	1.37
P31P	IRDF-90	2.783E-30	3.60
P31P	JENDL/D-99	2.938E-30	1.34
S32P	IRDF-90	6.345E-30	3.54
TI0XSC46	JENDL/D-99	9.117E-32	2.28
TI0XSC48	JENDL/D-99 (u)	1.971E-32	2.10
TI462	JENDL/D-99	3.621E-34	1.84
TI462	RRDF-98(u)	3.359E-34	4.40
TI46P	IRDF-90	1.002E-30	2.43
TI46P	JENDL/D-99	1.105E-30	2.27
TI46P	RRDF-98(u)	1.118E-30	3.13
TI47NP	IRDF-90	7.958E-34	30.00
TI47NP	RRDF-98(u)	6.380E-34	8.53
TI47P	IRDF-90	1.760E-30	3.69
TI48NP	IRDF-90	1.302E-34	30.00
TI48NP	JENDL/D-99	1.235E-34	2.65
TI48NP	RRDF-98 (u)	1.264E-34	8.59
TI48P	IRDF-90	2.596E-32	2.54
TI48P	JENDL/D-99	2.673E-32	1.85
TI48P	RRDF-98(u)	2.878E-32	5.17
TI49NP	JENDL/D-99	7.668E-35	10.01
TI49NP	RRDF-98(u)	7.657E-35	7.31
V51A	RRDF-98(u)	2.231E-33	3.13
CR522	IRDF-90	3.194E-33	2.68
CR522	JENDL/D-99	3.149E-33	1.29
CR522	ENDF/B-VI	3.248E-33	8.09
FE542	RRDF-98(u)	9.138E-35	4.96
FE54A	RRDF-98(u)	8.122E-32	3.28
FE54P	IRDF-90	7.880E-30	2.13
FE54P	JENDL/D-99 (u)	7.955E-30	0.99
FE56P	RRDF-98(u)	1.022E-31	2.62
CO592	IRDF-90	1.719E-32	2.85
CO59A	RRDF-98(u)	1.498E-32	3.76
NI582	IRDF-90	2.947E-34	3.11
NI582	JENDL/D-99	2.850E-34	0.90
NI582	JEFF-3.0	2.946E-34	2.75
NI58P	IRDF-90	1.038E-29	2.20

Continuation of Table 5.			
NI58P	JENDL/D-99	1.029E-29	0.61
NI58P	RRDF-98(new)	1.055E-29	1.73
NI58P	ENDF/B-VI	1.038E-29	2.45
NI58P	JEFF-3.0	1.054E-29	3.56
NI60P	ENDF/B-VI	1.867E-31	10.15
NI60P	JEFF-3.0	2.111E-31	8.83
CU632	IRDF-90	7.738E-33	1.75
CU632	JENDL/D-99(u)	7.877E-33	1.36
CU632	ENDF/B-VI	7.608E-33	4.43
CU63A	IRDF-90	5.017E-32	2.34
CU63A	RRDF-98(u)	5.128E-32	2.84
CU652	IRDF-90	2.894E-32	1.84
CU652	JENDL/D-99 (u)	3.024E-32	0.92
CU652	ENDF/B-VI	2.894E-32	2.31
ZN64P	IRDF-90	3.774E-30	4.80
AS752	RRDF-98(u)	2.562E-32	6.12
Y892	JENDL/D-99	1.255E-32	1.45
ZR902	IRDF-90	7.536E-33	1.60
ZR902	JENDL/D-99	7.355E-33	0.55
NB932	IRDF-90	3.878E-32	2.80
NB932	RRDF-98	3.839E-32	1.06
NB93N	IRDF-90	1.376E-29	3.01
NB93N	RRDF-98	1.410E-29	2.80
RH103N	IRDF-90	6.968E-29	3.01
RH103N	RRDF-98(nu)	7.061E-29	3.95
IN1152	IRDF-90	7.535E-32	3.57
IN115N	IRDF-90	1.828E-29	2.18
IN115N	JENDL/D-99	1.828E-29	2.18
IN115N	RRDF-98(nu)	1.848E-29	1.71
I1272	IRDF-90	1.045E-31	2.53
I1272	JENDL/D-99	1.090E-31	3.09
PR1412	RRDF-98(u)	9.328E-32	11.68
TM1692	JENDL/D-99	3.458E-31	2.33
AU1972	IRDF-90	3.112E-31	4.28
AU1972	JENDL/D-99	3.140E-31	1.18
HG199N	JENDL/D-99 (u)	2.354E-29	8.08
PB204N	RRDF-98(new)	1.744E-30	4.64
TH232F	IRDF-90	7.372E-30	5.18
U238F	IRDF-90	2.997E-29	0.54
U238F	JENDL/D-99	3.034E-29	2.09

REMARKS

$<\sigma_f>$ Cross section, averaged over the Watt fission spectrum.

* Weighted with a typical MTR spectrum from 1.05 MeV to 20.MeV.

u – update

nu – new update

new – new evaluation

Table 6. Cross section characteristics and C/E values in the ^{252}Cf fission neutron spectrum, for the fast neutron reactions selected for IRDF-2002*

Reaction code	Library	Calc.cross. sec.<σ_c> (mb)	Uncertainty in <σ_c> (%)	Exp. value <σ_e> (mb)	Uncertainty in <σ_e> (%)	C/E
F192	RRDF-98(u)	1.627E-2	2.92 (5.33)	1.612E-2	3.37	1.009±0.064
MG24P	IRDF-90	2.160	2.24 (2.75)	1.996	2.44	1.082±0.040
AL27P	IRDF-90	4.674	3.24 (3.44)	4.880	2.14	0.958±0.039
AL27A	IRDF-90	1.038	1.36 (2.12)	1.016	1.47	1.022±0.026
P31P	JENDL/D-99	32.24	1.40 (1.55)	not available	---	---
S32P	IRDF-90	70.30	3.60 (3.67)	72.54	3.49	0.969±0.049
TI0XSC46	JENDL/D-99	No infor-	mation	is	avail-	able!
TI0XSC48	JENDL/D-99 (u)	No infor-	mation	is	avail-	able!
TI462	JENDL/D-99	1.308E-2	1.86 (8.58)	not available	---	---
TI46P	RRDF-98(u)	13.83	3.05 (3.28)	14.07	1.77	0.983±0.037
TI47NP	RRDF-98(u)	1.941E-2	7.57 (9.58)	not available	---	---
TI47P	IRDF-90	19.38	3.78 (3.83)	19.27	1.66	1.006±0.042
TI48NP	RRDF-98 (u)	4.359E-3	8.20 (11.62)	not available	---	---
TI48P	RRDF-98(u)	0.4268	5.08 (5.32)	0.4247	1.89	1.005±0.057
TI49NP	RRDF-98(u)	2.644E-3	7.18 (10.84)	not available	---	---
V51A	RRDF-98(u)	3.859E-2	3.02 (3.56)	3.900E-2	2.21	0.989±0.041
CR522	JENDL/D-99	9.555E-2	1.29 (5.75)	not available	---	---
FE542	RRDF-98(u)	3.498E-3	4.87 (10.71)	not available	---	---
FE54A	RRDF-98(u)	1.113	3.18 (3.48)	not available	---	---
FE54P	IRDF-90	88.16	2.09 (2.23)	86.84	1.34	1.015±0.026
FE56P	RRDF-98(u)	1.475	2.61 (2.99)	1.465	1.77	1.007±0.035
CO592	IRDF-90	0.4228	2.67 (4.20)	0.405	2.51	1.044±0.051
CO59A	RRDF-98(u)	0.2212	3.54 (3.87)	0.2218	1.88	0.997±0.043
NI582	JENDL/D-99	8.985E-3	0.85 (6.24)	8.952E-3	3.57	1.004±0.072
NI58P	RRDF-98(new)	117.5	1.74 (1.89)	117.5	1.30	1.000±0.023
NI60P	ENDF/B-VI	2.494	10.11 (10.20)	(2.39)	5.44	1.044±0.121
CU632	ENDF/B-VI	0.2056	4.10 (4.81)	0.1844	3.98	1.115±0.078
CU63A	RRDF-98(u)	0.6933	2.83 (3.15)	0.6887	1.96	1.007±0.037
CU652	ENDF/B-VI	0.6777	2.25 (3.69)	0.6582	2.22	1.030±0.044
ZN64P	IRDF-90	42.10	4.87 (4.93)	40.59	1.65	1.037±0.054
AS752	RRDF-98(u)	0.6209	5.76 (6.55)	not available	---	---
Y892	JENDL/D-99	0.344	1.40 (4.47)	not available	---	---
ZR902	IRDF-90	0.2212	1.57 (5.31)	0.2210	2.89	1.001±0.061
NB932	RRDF-98	0.7717	1.03 (2.46)	(0.749)	5.07	1.030±0.058
NB93N	RRDF-98	146.1	2.59 (2.61)	(146)	3.45	1.001±0.043
RH103N	RRDF-98(nu)	725.1	3.94 (3.95)	(809)	2.97	0.896±0.044
IN1152	IRDF-90	1.586	3.23 (4.02)	not available	---	---
IN115N	RRDF-98(nu)	191.8	1.66 (1.70)	197.4	1.37	0.972±0.021
I1272	IRDF-90	2.197	2.28 (3.30)	2.069	2.73	1.062±0.045
PR1412	RRDF-98(u)	1.990	11.03 (11.37)	not available	---	---
TM1692	JENDL/D-99	6.233	2.26 (3.01)	(6.69)	6.28	0.932±0.065
AU1972	IRDF-90	5.747	4.19 (4.65)	5.506	1.83	1.044±0.052
HG199N	JENDL/D-99 (u)	248.6	7.82 (7.83)	298.4	1.81	0.833±0.067
PB204N	RRDF-98(u)	20.39	4.57 (4.67)	(20.58)	4.41	0.978±0.063
TH232F	IRDF-90	78.55	5.09 (5.11)	(89.4)	3.02	0.879±0.052
U238F	JENDL/D-99	319.2	2.00 (2.04)	325.7	1.64	0.980±0.026

Remarks see on the next page

REMARKS

* Data taken from [17]

$<\sigma_c>$ – calculated cross section, averaged over the ^{252}Cf fission neutron spectrum

$<\sigma_e>$ – experimental value, ^{252}Cf fission spectrum-averaged cross section

u – update

nu – new update

new – new evaluation

"Experimental data given in brackets are from single experiments which were not part of the evaluation process" – information from [17]

Table7. Final content of the file IRDF-2002 and source of the data

Reaction	Selected Source	Reaction	Selected Source
^{6}Li(n, t)^{4}He	IRDF-90*	^{65}Cu(n, 2n)^{64}Cu	IRDF-90*
^{10}B(n, α)^{7}Li	IRDF-90	^{64}Zn(n, p)^{64}Cu	IRDF-90
^{19}F(n, 2n)^{18}F	RRDF-98(u)	^{75}As(n, 2n)^{74}As	RRDF-98(u)
^{23}Na(n, γ)^{24}Na+	IRDF-90*	^{89}Y(n, 2n)^{88}Y	JENDL/D-99
^{23}Na(n, 2n)^{22}Na	JENDL/D-99(u)	^{90}Zr(n, 2n)^{89}Zr	IRDF-90
^{24}Mg(n, p)^{24}Na	IRDF-90	^{93}Nb(n, 2n)^{92m}Nb	RRDF-98
^{27}Al(n, p)^{27}Mg	RRDF-98(new)	^{93}Nb(n, n′)^{93m}Nb	RRDF-98
^{27}Al(n, α)^{24}Na	IRDF-90	^{93}Nb(n, γ)^{94}Nb+	IRDF-90*
^{31}P(n, p)^{31}Si	IRDF-90	^{103}Rh(n, n′)^{103m}Rh	RRDF-98(new)
^{32}S(n, p)^{32}P	IRDF-90	^{109}Ag(n, γ)^{110m}Ag	IRDF-90
^{45}Sc(n, γ)^{46}Sc	IRDF-90	^{115}In(n, 2n)^{114m}In	IRDF-90*
^{46}Ti(n, 2n)^{45}Ti	RRDF-98(u)	^{115}In(n, n′)^{115m}In	RRDF-98(new)
^{46}Ti(n, p)^{46}Sc	RRDF-98(u)	^{115}In(n, γ)^{116m}In+	ENDF/B-VI
^{47}Ti(n, x$^{\#}$)^{46}Sc	RRDF-98(u)	^{127}I(n, 2n)^{126}I	IRDF-90
^{47}Ti(n, p)^{47}Sc	IRDF-90	^{139}La(n, γ)^{140}La	RRDF-98(new)
^{48}Ti(n, x$^{\#}$)^{47}Sc	RRDF-98(u)	^{141}Pr(n, 2n)^{140}Pr	RRDF-98(u)
^{48}Ti(n, p)^{48}Sc	RRDF-98(u)	^{169}Tm(n, 2n)^{168}Tm	JENDL/D-99
^{49}Ti(n, x$^{\#}$)^{48}Sc	RRDF-98(u)	^{181}Ta(n, γ)^{182}Ta+	JENDL/D-99
^{51}V(n, α)^{48}Sc	RRDF-98(u)	^{186}W(n, γ)^{187}W	RRDF-98(new)
^{52}Cr(n, 2n)^{51}Cr	IRDF-90	^{197}Au(n, 2n)^{196}Au	IRDF-90
^{55}Mn(n, γ)^{56}Mn	IRDF-90*	^{197}Au(n, γ)^{198}Au	IRDF-90*
^{54}Fe(n, 2n)^{53}Fe	RRDF-98(u)	^{199}Hg(n, n′)^{199m}Hg	JENDL/D-99(u)
^{54}Fe(n, α)^{51}Cr	RRDF-98(u)	^{204}Pb(n, n′)^{204m}Pb	RRDF-98(new)
^{54}Fe(n, p)^{54}Mn	IRDF-90*	^{232}Th(n, γ)^{233}Th+	IRDF-90
^{56}Fe(n, p)^{56}Mn	RRDF-98(u)	^{232}Th(n, f)	IRDF-90
^{58}Fe(n, γ)^{59}Fe	JENDL/D-99(u)	^{235}U(n, f)	IRDF-90
^{59}Co(n, 2n)^{58}Co	IRDF-90	^{238}U(n, f)	JENDL/D-99
^{59}Co(n, α)^{56}Mn	RRDF-98(u)	^{238}U(n, γ)^{239}U	IRDF-90*
^{59}Co(n, γ)^{60}Co	IRDF-90*	^{237}Np(n, f)	RRDF-98(new)
^{58}Ni(n, 2n)^{57}Ni	JEFF 3.0	^{239}Pu(n, f)	JENDL/D-99
^{58}Ni(n, p)^{58}Co	RRDF-98(new)	^{241}Am(n, f)	JENDL/D-99
^{60}Ni(n, p)^{60}Co	ENDF/B-VI	^{nat}B(n, x)#	ENDF/B-VI
^{63}Cu(n, 2n)^{62}Cu	ENDF/B-VI	^{nat}Cd(n, x)#	ENDF/B-VI
^{63}Cu(n, γ)^{64}Cu	IRDF-90*	^{nat}Gd(n, x)#	ENDF/B-VI
^{63}Cu(n, α)^{60}Co	RRDF-98(u)		

\+ Diagonal covariance matrix.
\# Cover material; no covariance information available.
u Up-date.
* ENDF/B-VI Rel 8 (see explanation in the text above).
(n, x$^{\#}$) sum of the reactions (n,np) +(n,pn) +(n,d)

Table 8. Characteristics of the cross sections present in the file IRDF-2002 (neutron temperature 300 K)

Reaction	Selected evaluation	[1)]Calculated library cross section at 2200 m/s s_L (barn)	[1)]Resonance integral from library data IR_L (barn)	[1)]Uncertainty in library data thermal (%)	epithermal (%)	[2)]Calculated average library cross section in ^{252}Cf sf $<s_C>$ (mbarn)	[2)]Uncertainty in $<s_C>$ (%)	C/E
^{6}Li(n, t)	IRDF-90	942	427	0.14	0.14	-	-	thermal: [1)b)]1.00±0.01 epithermal: [1)b)]1.00±0.01
^{10}B(n, α)	IRDF-90	3840	1730	0.16	0.16	-	-	thermal: [1)b)]1.00±0.01 epithermal: [1)b)]0.99±0.01
^{19}F(n, 2n)	RRDF-98(u)	-	-	-	-	1.627E-2	2.92 (5.33)	[2)]1.009±0.064
^{23}Na(n, γ)+	IRDF-90	0.529	0.317	2.00	3.14	-	-	thermal: [1)a)]1.00±0.02 epithermal: [1)a)]0.97±0.04
^{23}Na(n, 2n)	JENDL/D-99(u)	-	-	-	-	8.611E-3	3.90(8.16)	No experimental data in ^{252}Cf fission field
^{24}Mg(n, p)	IRDF-90	-	-	-	-	2.160	2.24 (2.75)	[2)]1.082±0.040
^{27}Al(n, p)	RRDF-98(new)	-	-	-	-	4.912	2.06 (2.37)	[2)]1.007±0.032
^{27}Al(n, α)	IRDF-90	-	-	-	-	1.038	1.36 (2.12)	[2)]1.022±0.026
^{31}P(n, p)	IRDF-90	-	-	-	-	30.68	3.58 (3.65)	No experimental data in ^{252}Cf fission field
^{32}S(n, p)	IRDF-90	-	-	-	-	70.30	3.60 (3.67)	[2)]0.969±0.049
^{45}Sc(n, γ)	IRDF-90	27.3	12.9	0.73	0.76	-	-	thermal: [1)a)]1.00±0.01 epithermal: [1)a)]1.00±0.04
^{46}Ti(n, 2n)	RRDF-98(u)	-	-	-	-	1.218E-2	4.41 (9.55)	No experimental data in ^{252}Cf fission field
^{46}Ti(n, p)	RRDF-98(u)	-	-	-	-	13.83	3.05 (3.28)	[2)]0.983±0.037
^{47}Ti(n, np)‡	RRDF-98(u)	-	-	-	-	1.941E-2	7.57 (9.58)	No experimental data in ^{252}Cf fission field
^{47}Ti(n, p)	IRDF-90	-	-	-	-	19.38	3.78 (3.83)	[2)]1.006±0.042
^{48}Ti(n, np)‡	RRDF-98(u)	-	-	-	-	4.349E-3	8.20 (11.62)	No experimental data in ^{252}Cf fission field
^{48}Ti(n, p)	RRDF-98(u)	-	-	-	-	0.4268	5.08 (5.32)	[2)]1.005±0.057
^{49}Ti(n, np)‡	RRDF-98(u)	-	-	-	-	2.644E-3	7.18 (10.84)	No experimental data in ^{252}Cf fission field
^{51}V(n, α)	RRDF-98(u)	-	-	-	-	3.859E-2	3.02 (3.56)	[2)]0.989±0.041
^{52}Cr(n, 2n)	IRDF-90	-	-	-	-	9.703E-2	2.72 (6.23)	No experimental data in ^{252}Cf fission field

^{55}Mn(n, γ)	IRDF-90	13.4	11.8	4.18	3.84	-	-	thermal: [1)a)]1.00±0.04 [1)b)]1.01±0.04 epithermal: [1)a)]0.84±0.04 [1)b)]0.84±0.04
^{54}Fe(n, 2n)	RRDF-98(u)	-	-	-	-	3.498E-3	4.87 (10.71)	No experimental data in ^{252}Cf fission field
^{54}Fe(n, α)	RRDF-98(u)	-	-	-	-	1.113	3.18 (3.48)	No experimental data in ^{252}Cf fission field
^{54}Fe(n, p)	IRDF-90	-	-	-	-	88.16	2.09 (2.23)	[2)]1.015±0.026
^{56}Fe(n, p)	RRDF-98(u)	-	-	-	-	1.475	2.61 (2.99)	[2)]1.007±0.035
^{58}Fe(n, γ)	JENDL/D-99(u)	1.30	1.37	12.60	8.70	-	-	thermal: [1)a)]1.00±0.13 [1)b)]1.00±0.15 epithermal: [1)a)]0.81±0.08 [1)b)]1.05±0.18
^{59}Co(n, 2n)	IRDF-90	-	-	-	-	0.4228	2.67 (4.20)	[2)]1.044±0.051
^{59}Co(n, α)	RRDF-98(u)	-	-	-	-	0.2212	3.54 (3.87)	[2)]0.997±0.043
^{59}Co(n, γ)	IRDF-90	37.2	76.0	0.66	0.77	-	-	thermal: [1)a)]1.00±0.01 epithermal: [1)a)]1.00± 0.01
^{58}Ni(n, 2n)	JEFF-3.0	-	-	-	-	9.256E-3	2.72 (6.67)	[2)]1.034±0.078
^{58}Ni(n, p)	RRDF-98(new)	-	-	-	-	117.5	1.74 (1.89)	[2)]1.000±0.023
^{60}Ni(n, p)	ENDF/B-VI	-	-	-	-	2.494	10.11 (10.20)	[2)]1.044±0.121
^{63}Cu(n, 2n)	ENDF/B-VI	-	-	-	-	0.2056	4.10 (5.81)	[2)]1.115±0.078
^{63}Cu(n, γ)	IRDF-90	4.47	4.96	4.11	3.86	-	-	thermal: [1)a)]0.99±0.04 [1)b)]0.99±0.06 epithermal: [1)a)]1.00±0.04 [1)b)]0.99±0.04
^{63}Cu(n, α)	RRDF-98(u)	-	-	-	-	0.6933	2.83 (3.15)	[2)]1.007±0.037
^{65}Cu(n, 2n)	IRDF-90	-	-	-	-	0.6779	1.83(3.44)	[2)]1.030±0.042
^{64}Zn(n, p)	IRDF-90	-	-	-	-	42.10	4.87 (4.93)	[2)]1.037±0.054
^{75}As(n, 2n)	RRDF-98(u)	-	-	-	-	0.6209	5.76 (6.55)	No experimental data in ^{252}Cf fission field
^{89}Y(n, 2n)	JENDL/D-99	-	-	-	-	0.344	1.40 (4.47)	No experimental data in ^{252}Cf fission field
^{90}Zr(n, 2n)	IRDF-90	-	-	-	-	0.2212	1.57 (5.31)	[2)]1.001±0.061
^{93}Nb(n, 2n)*	RRDF-98	-	-	-	-	0.7717	1.03 (2.46)	[2)]1.03±0.058
^{93}Nb(n, n')*	RRDF-98	-	-	-	-	146.1	2.59 (2.61)	[2)]1.001±0.043

^{93}Nb(n, γ)+	IRDF-90	1.16	9.91	10.00	9.49	-	-	thermal: [1)a)]1.01±0.11 epithermal: [1)a)]1.17±0.13
^{103}Rh(n, n')*	RRDF-98(u)	-	-	-	-	725.1	3.94 (3.95)	[2)]0.896±0.044
^{109}Ag(n, γ)*	IRDF-90	4.21	68.6	5.10	6.93	-	-	thermal: [1)b)]1.00 epithermal: [1)b)]0.98 No experimental uncertainty
^{115}In(n, 2n)*	IRDF-90	-	-	-	-	1.586	3.23 (4.02)	No experimental data in ^{252}Cf fission field
^{115}In(n, n')*	RRDF-98(u)	-	-	-	-	191.8	1.66 (1.70)	[2)]0.972±0.021
^{115}In(n, γ)*+	IRDF-90	167	2590	6.00	5.98	-	-	thermal: [1)a)]1.04±0.06 epithermal: [1)a)]0.96±0.07
^{127}I(n, 2n)	IRDF-90	-	-	-	-	2.197	2.28 (3.30)	[2)]1.062±0.045
^{139}La(n, γ)	RRDF-98(new)	8.90	11.9	3.87	5.50	-	-	thermal: [1)a)]0.98±0.04 [1)b)]0.98±0.04 epithermal: [1)a)]0.98±0.07 [1)b)]0.99±0.1
^{141}Pr(n, 2n)	RRDF-98(u)	-	-	-	-	1.990	11.03 (11.37)	No experimental data in ^{252}Cf fission field
^{169}Tm(n, 2n)	JENDL/D-99	-	-	-	-	6.233	2.26 (3.01)	[2)]0.932±0.065
^{181}Ta(n, γ)+	JENDL/D-99	20.7	659	3.00	3.77	-	-	thermal [1)a)]1.01±0.04 epithermal: [1)a)]1.00±0.05
^{186}W(n, γ)	RRDF-98(new)	37.9	479	2.31	3.32	-	-	thermal: [1)a)]0.98±0.03 [1)b)]1.02±0.06 epithermal: [1)a)]0.99±0.04 [1)b)]0.94±0.10
^{197}Au(n, 2n)	IRDF-90	-	-	-	-	5.747	4.19 (4.65)	[2)]1.044±0.052
^{197}Au(n, γ)♣	IRDF-90	98.8	1570	0.14	0.17	-	-	thermal: [1)a)]1.00±0.01 [1)b)]1.00±0.01 epithermal: [1)a)]1.01±0.02 [1)b)]1.01±0.02
^{199}Hg(n, n')*	JENDL/D-99(u)	-	-	-	-	248.6	7.82 (7.83)	[2)]0.833±0.067
^{204}Pb(n, n')*	RRDF-98(new)	-	-	-	-	20.39	4.57 (4.67)	[2)]0.978±0.063
^{232}Th(n, γ)+	IRDF-90	7.41	85.6	4.33	10.92	-	-	thermal: [1)a)]1.01±0.04 [1)b)]1.00±0.04 epithermal: [1)a)]1.01±0.12 [1)b)]1.01±0.12
^{232}Th(n, f)	IRDF-90	-	-	-	-	78.55	5.09 (5.11)	0.879±0.052

^{235}U(n, f)♣	IRDF-90	586	272	0.19	0.26	1218	0.32 (0.32)	thermal: [1)b)]1.00±0.004 epithermal:[1)b)]0.99±0.02 [2)]1.007±0.0102
^{238}U(n, f)	JENDL/D-99	-	-	-	-	319.2	2.00 (2.04)	[2)]0.980±0.026
^{238}U(n, γ)	IRDF-90	2.72	277	0.35	0.37	-	-	thermal: [1)a)]1.01±0.01 [1)b)]1.00±0.04 epithermal: [1)a)]1.00±0.01 [1)b)]1.00±0.01
^{237}Np(n, f)	RRDF-98(new)	-	-	-	-	1359	1.72 (1.74)	[2)]0.999±0.024
^{239}Pu(n, f)	JENDL/D-99	747	297	0.71	3.82	1804	2.04 (2.04)	thermal: [1)b)]0.99±0.01 epithermal:[1)b)]0.99±0.05 [2)]0.996±0.025
^{241}Am(n, f)	JENDL/D-99	3.03	7.84	2.00	1.56	1396	2.81(2.90)	thermal: [1)b)]0.99±0.004 epithermal: no exp. data No experimental data in ^{252}Cf fission field

Remarks

1) Calculated data at 300 Kelvin (Zsolnay, E.M., Nolthenius, H.J)

2) Calculated and experimental data from Mannhart, W., Response of activation reactions in the neutron field of spontaneous fission of ^{252}Cf, in Ref. [17].

a) Evaluated experimental data from Mughabghab, S.F., Thermal neutron capture cross sections, resonance integrals and g-factors, INDC(NDS)-440, IAEA, Vienna, February 2003.

b) Evaluated experimental data from: Holden, N.E., Neutron scattering and absorption properties (revised 2003), pp. 198-213 in CRC Handbook of Chemistry and Physics, 84th Edition, Chapter 11, Editor-in-Chief: LIDE, D.R., CRC Press, 2000 NW Corporate Blvd., Boca Raton, Florida 33431, USA (2003).

U Up-date.

\+ Diagonal matrix.

* Metastable state of the product nucleus.

‡ Sum of cross sections of (n, np) + (n, pn) + (n, d) reactions.

♣ Unreliable uncertainty (corresponding data have been withdrawn from ENDF/B-VI).

Column 8 shows the contribution of the energy-dependent library cross section data to the uncertainty of $<\sigma_c>$; values in brackets give the total standard deviation of $<\sigma_c>$, including the contribution of the uncertainty of the ^{252}Cf spectrum function.

Uncertainties given for the C/E values involve the standard deviations of both the calculated and experimental cross-section data.

All uncertainty data in the table are expressed in terms of one standard deviation.

Table 9. Mat, MF and MT numbers of the cross sections in the file IRDF-2002

ACTIVATION AND FISSION REACTIONS

	Group cross sec. data*			Reaction Code	Reaction	Point cross sec. data		
No	Mat	MF	MT			MaT	MF	MT
1	325	3	105	LI6A	6LI(N,T)4HE	325	3	105
2	525	3	107	B10A	10B(N,A)6LI	525	3	107
3	925	3	16	F192	19F(N,2N)18F	925	3	16
4	1125	3	16	NA232	23NA(N,2N)22NA	1125	3	16
5	1125	3	102	NA23G	23NA(N,G)24NA	1125	3	102
6	1225	3	103	MG24P	24MG(N,P)24NA	1225	3	103
7	1325	3	103	AL27P	27AL(N,P)27MG	1325	3	103
8	1325	3	107	AL27A	27AL(N,A)24NA	1325	3	107
9	1525	3	103	P31P	31P(N,P)31SI	1525	3	103
10	1625	3	103	S32P	32S(N,P)32P	1625	3	103
11	2126	3	102	SC45G	45SC(N,G)46SC	2126	3	102
12	2225	3	16	TI462	46TI(N,2N)45TI	2225	3	16
13	2225	3	103	TI46P	46TI(N,P)46SC	2225	3	103
14	2228	3	231	TI47NP	47TI(N,NP)46SC	2228	10	5
15	2228	3	103	TI47P	47TI(N,P)47SC	2228	3	103
16	2231	3	231	TI48NP	48TI(N,NP)47SC	2231	10	5
17	2231	3	103	TI48P	48TI(N,P)48SC	2231	3	103
18	2234	3	231	TI49NP	49TI(N,NP)48SC	2234	10	5
19	2328	3	107	V51A	51V(N,A)48SC	2328	3	107
20	2431	3	16	CR522	52CR(N,2N)51CR	2431	3	16
21	2525	3	102	MN55G	55MN(N,G)56MN	2525	3	102
22	2625	3	16	FE542	54FE(N,2N)53FE	2625	3	16
23	2625	3	103	FE54P	54FE(N,P)54MN	2625	3	103
24	2625	3	107	FE54A	54FE(N,A)51CR	2625	3	107
25	2631	3	103	FE56P	56FE(N,P)56MN	2631	3	103
26	2637	3	102	FE58G	58FE(N,G)59FE	2637	3	102
27	2725	3	16	CO592	59CO(N,2N)58CO	2725	3	16
28	2725	3	102	CO59G	59CO(N,G)60CO	2725	3	102
29	2725	3	107	CO59A	59CO(N,A)56MN	2725	3	107
30	2825	3	16	NI582	58NI(N,2N)57NI	2825	3	16
31	2825	3	103	NI58P	58NI(N,P)58CO	2825	3	103
32	2831	3	103	NI60P	60NI(N,P)60CO	2831	3	103
33	2925	3	16	CU632	63CU(N,2N)62CU	2925	3	16
34	2925	3	102	CU63G	63CU(N,G)64CU	2925	3	102
35	2925	3	107	CU63A	63CU(N,A)60CO	2925	3	107
36	2931	3	16	CU652	65CU(N,2N)64CU	2931	3	16
37	3025	3	103	ZN64P	64ZN(N,P)64CU	3025	3	103
38	3325	3	16	AS752	75AS(N,2N)74AS	3325	3	16
39	3925	3	16	Y892	89Y(N,2N)88Y	3925	3	16
40	4025	3	16	ZR902	90ZR(N,2N)89ZR	4025	3	16
41	4125	3	292	NB932	93NB(N,2N)92NB	4125	3	16
42	4125	3	291	NB93N	93NB(N,N')93NBM	4125	10	4
43	4125	3	102	NB93G	93NB(N,G)94NB	4125	10	16
44	4525	3	51	RH103N	103RH(N,N')103RHM	4525	3	51
45	4731	3	293	AG109G	109AG(N,G)110AGM	4731	10	102
46	4931	3	292	IN1152	115IN(N,2N)114INM	4931	10	16
47	4931	3	291	IN115N	115IN(N,N')115INM	4931	10	4
48	4931	3	293	IN115G	115IN(N,G)116INM	4931	10	102
49	5325	3	16	I1272	127I(N,2N)126I	5325	3	16
50	5728	3	102	LA139G	139LA(N,G)140LA	5728	3	102
51	5925	3	16	PR1412	141PR(N,2N)140PR	5925	3	16
52	6925	3	16	TM1692	169TM(N,2N)168TM	6925	3	16
53	7328	3	102	TA181G	181TA(N,G)182TA	7328	3	102
54	7443	3	102	W186G	186W(N,G)187W	7443	3	102
55	7925	3	16	AU1972	197AU(N,2N)196AU	7925	3	16
56	7925	3	102	AU197G	197AU(N,G)198AU	7925	3	102
57	8034	3	51	HG199N	199HG(N,N')199HGM	8034	3	51
58	8225	3	71	PB204N	204PB(N,N')204PBM	8225	3	71
59	9040	3	18	TH232F	232TH(N,F) FP	9040	3	18
60	9040	3	102	TH232G	232TH(N,G)233TH	9040	3	102
61	9228	3	18	U235F	235U(N,F) FP	9228	3	18
62	9237	3	18	U238F	238U(N,F) FP	9237	3	18
63	9237	3	102	U238G	238U(N,G)239U	9237	3	102
64	9346	3	18	NP237F	237NP(N,F) FP	9346	3	18
65	9437	3	18	PU239F	239PU(N,F) FP	9437	3	18
66	9543	3	18	AM241F	241AM(N,F) FP	9543	3	18

COVER REACTIONS

	Group cross sec. data*			Reaction code	Reaction	Point cross sec. data		
No	Mat	MF	MT			Mat	MF	MT
1	500	3	1	B	B-COVER	500	3	1
2	4800	3	1	CD	CD-COVER	4800	3	1
3	6400	3	1	GD	GD-COVER	6400	3	1

DAMAGE CHARACTERIZATION REACTIONS

	Group cross sec. data*			Reaction Code	Reaction	Point cross sec. data		
No	Mat	MF	MT			MaT	MF	MT
1	1400	3	900	SI0DP	SI-DPA_ASTM	NOT AVAILABLE		
2	2400	3	900	CR0DP	CR-DPA	NOT AVAILABLE		
3	2600	3	900	FE0ASDP	FE-DPA_ASTM	NOT AVAILABLE		
4	2600	3	901	FE0EWDP	ST-DPA_EWGRD	NOT AVAILABLE		
5	2800	3	900	NI0DP	NI-DPA	NOT AVAILABLE		
6	3100	3	900	GA_ASDP	GA_AS-DPA	NOT AVAILABLE		

Table 10. Threshold energies of cross section and of uncertainty data in the file IRDF-2002.

Reaction code	Threshold energies of the cross sections (eV)	Original threshold energies of the uncertainties (eV)	Corrected threshold energies of the uncertainties (eV)
F192	1.090E+07	1.099E+07	1.090E+07
NA232	1.290E+07	1.296E+07	1.290E+07
MG24P	4.900E+06	4.600E+06	4.900E+06
AL27P	1.800E+06	1.896E+06	1.800E+06
AL27A	5.000E+06	5.000E+06	*
P31P	1.200E+06	1.200E+06	*
S32P	9.200E+05	9.200E+05	*
TI462	1.340E+07	1.348E+07	1.340E+07
TI46P	1.600E+06	1.619E+06	1.600E+06
TI47P	6.900E+06	6.900E+06	*
TI47NP	8.400E+06	8.414E+06	8.400E+06
TI48P	3.200E+06	3.279E+06	3.200E+06
TI48NP	9.400E+06	9.414E+06	9.400E+06
TI49NP	9.300E+06	9.138E+06	9.300E+06
V51A	2.000E+06	2.099E+06	2.000E+06
CR522	1.220E+06	1.220E+06	*
FE542	1.360E+07	1.363E+07	1.360E+07
FE54P	6.900E+05	5.000E+05	6.900E+05
FE54A	2.500E+06	2.500E+06	*
FE56P	2.900E+06	2.966E+06	2.900E+06
CO592	1.060E+07	1.060E+07	*
CO59A	3.200E+05	3.340E+05	3.200E+05
NI582	1.240E+07	1.241E+07	1.240E+07
NI58P	1.000E+05	5.000E+05	1.000E+05
NI60P	2.700E+06	2.076E+06	2.700E+06
CU632	1.100E+07	1.103E+07	1.100E+07
CU63A	2.200E+06	2.250E+06	2.200E+06
CU652	1.000E+07	1.000E+07	*
ZN64P	2.100E+05	5.000E+05	2.100E+05
AS752	1.030E+07	1.038E+07	1.030E+07
Y892	1.160E+07	1.161E+07	1.160E+07
ZR902	1.210E+07	1.212E+07	1.210E+07
NB93N	3.000E+04	1.000E+05	3.000E+04
NB932	8.900E+06	9.000E+06	8.900E+06
RH103N	4.000E+04	4.014E+04	4.000E+04
IN115N	3.200E+05	3.392E+05	3.200E+05
IN1152	9.300E+06	1.000E+07	9.300E+07
I1272	9.200E+06	9.465E+06	9.200E+06
PR1412	9.400E+06	9.465E+06	9.400E+06
TM1692	8.100E+06	8.100E+06	*
AU1972	8.100E+06	8.100E+06	*
HG199N	5.250E+05	5.337E+05	5.250E+05
PB204N	2.100E+06	2.197E+06	2.100E+06
TH232F	5.000E+05	5.000E+05	*

* No correction had to be applied

Table 11. Some integral characteristics of the newly generated group cross section data in IRDF-2002

Activation and fission reactions							
No.	MAT	MF	MT	Reaction code	Cross section at 2200 m/s (m^2)	Resonance integral (m^2)	Cross section averaged over the Watt fiss. spectrum (m^2)
1	325	3	105	LI6T	9.420E-26	4.265E-26	3.237E-29
2	525	3	107	B10A	3.843E-25	1.731E-25	4.574E-29
3	925	3	16	F192	0.000E+00	0.000E+00	5.855E-34
4	1125	3	16	NA232	0.000E+00	0.000E+00	2.570E-34
5	1125	3	102	NA23G	5.285E-29	3.171E-29	2.786E-32
6	1225	3	103	MG24P	0.000E+00	0.000E+00	1.473E-31
7	1325	3	103	AL27P	0.000E+00	0.000E+00	3.979E-31
8	1325	3	107	AL27A	0.000E+00	0.000E+00	6.860E-32
9	1525	3	103	P31P	0.000E+00	0.000E+00	2.783E-30
10	1625	3	103	S32P	0.000E+00	1.203E-34	6.340E-30
11	2126	3	102	SC45G	2.732E-27	1.197E-27	5.135E-31
12	2225	3	16	TI462	0.000E+00	0.000E+00	3.359E-34
13	2225	3	103	TI46P	0.000E+00	0.000E+00	1.118E-30
14	2228	3	103	TI47P	0.000E+00	3.216E-32	1.760E-30
15	2228	3	231	TI47NP	0.000E+00	0.000E+00	6.380E-34
16	2231	3	103	TI48P	0.000E+00	0.000E+00	2.878E-32
17	2231	3	231	TI48NP	0.000E+00	0.000E+00	1.264E-34
18	2234	3	231	TI49NP	0.000E+00	0.000E+00	7.657E-35
19	2328	3	107	V51A	0.000E+00	0.000E+00	2.231E-33
20	2431	3	16	CR522	0.000E+00	0.000E+00	3.193E-33
21	2525	3	102	MN55G	1.342E-27	1.180E-27	2.941E-31
22	2625	3	16	FE542	0.000E+00	0.000E+00	9.138E-35
23	2625	3	103	FE54P	0.000E+00	3.332E-33	7.880E-30
24	2625	3	107	FE54A	0.000E+00	0.000E+00	8.122E-32
25	2631	3	103	FE56P	0.000E+00	0.000E+00	1.053E-31
26	2637	3	102	FE58G	1.302E-28	1.369E-28	1.854E-31
27	2725	3	16	CO592	0.000E+00	0.000E+00	1.719E-32
28	2725	3	102	CO59G	3.720E-27	7.603E-27	5.043E-31
29	2725	3	107	CO59A	0.000E+00	3.367E-37	1.498E-32
30	2825	3	16	NI582	0.000E+00	0.000E+00	2.946E-34
31	2825	3	103	NI58P	0.000E+00	1.939E-32	1.055E-29
32	2831	3	103	NI60P	0.000E+00	0.000E+00	1.867E-31
33	2925	3	16	CU632	0.000E+00	0.000E+00	7.609E-33
34	2925	3	102	CU63G	4.474E-28	4.955E-28	1.072E-30
35	2925	3	107	CU63A	0.000E+00	0.000E+00	5.128E-32
36	2931	3	16	CU652	0.000E+00	0.000E+00	2.894E-32
37	3025	3	103	ZN64P	0.000E+00	7.740E-33	3.775E-30
38	3325	3	16	AS752	0.000E+00	0.000E+00	2.561E-32
39	3925	3	16	Y892	0.000E+00	0.000E+00	1.255E-32
40	4025	3	16	ZR902	0.000E+00	0.000E+00	7.536E-33
41	4125	3	102	NB93G	1.156E-28	9.907E-28	2.761E-30
42	4125	3	291	NB93N	0.000E+00	3.251E-30	1.410E-29
43	4125	3	292	NB932	0.000E+00	0.000E+00	3.838E-32
44	4525	3	51	RH103N	0.000E+00	4.087E-29	7.061E-29

Activation and fission reactions (continued)							
No	MAT	MF	MT	Reaction code	Cross section at 2200 m/s (m^2)	Resonance integral (m^2)	Cross section averaged over the Watt fiss. spectrum (m^2)
45	4731	3	293	AG109G	4.214E-28	6.858E-27	9.756E-31
46	4931	3	291	IN115N	0.000E+00	2.112E-30	1.848E-29
47	4931	3	292	IN1152	0.000E+00	0.000E+00	7.535E-32
48	4931	3	293	IN115G	1.665E-26	2.591E-25	1.260E-29
49	5325	3	16	I1272	0.000E+00	0.000E+00	1.044E-31
50	5728	3	102	LA139G	8.895E-28	1.193E-27	6.819E-31
51	5925	3	16	PR1412	0.000E+00	0.000E+00	9.328E-32
52	6925	3	16	TM1692	0.000E+00	0.000E+00	3.458E-31
53	7328	3	102	TA181G	2.069E-27	6.591E-26	8.728E-30
54	7443	3	102	W186G	3.793E-27	4.794E-26	3.292E-30
55	7925	3	16	AU1972	0.000E+00	0.000E+00	3.112E-31
56	7925	3	102	AU197G	9.880E-27	1.566E-25	7.762E-30
57	8034	3	51	HG199N	0.000E+00	2.206E-30	2.354E-29
58	8225	3	71	PB204N	0.000E+00	0.000E+00	1.744E-30
59	9040	3	18	TH232F	0.000E+00	2.437E-32	7.374E-30
60	9040	3	102	TH232G	7.412E-28	8.564E-27	9.313E-30
61	9228	3	18	U235F	5.862E-26	2.719E-26	1.216E-28
62	9237	3	18	U238F	1.183E-33	6.370E-31	3.036E-29
63	9237	3	102	U238G	2.720E-28	2.773E-26	7.083E-30
64	9346	3	18	NP237F	2.154E-30	1.586E-28	1.337E-28
65	9437	3	18	PU239F	7.478E-26	2.974E-26	1.796E-28
66	9543	3	18	AM241F	3.025E-28	7.832E-28	1.362E-28

Cover materials							
No	MAT	MF	MT	Reaction code	Cross section at 2200 m/s (m^2)	Resonance integral (m^2)	Cross section averaged over the Watt fiss. spectrum (m^2)
1	500	3	1	BOTO	7.657E-26	4.042E-26	2.482E-28
2	4800	3	1	CDOTO	2.457E-25	1.733E-26	4.384E-28
3	6400	3	1	GDOTO	4.901E-24	5.728E-26	6.765E-28

Radiation damage (DPA) cross sections							
No	MAT	MF	MT	Reaction code	Cross section at 2200 m/s (m^2)	Resonance integral (m^2)	Cross section averaged over the Watt fiss. spectrum (m^2)
1	1400	3	900	SI0DP	9.841E-30	1.413E-26	9.880E-27
2	2400	3	900	CR0DP	0.000E+00	9.420E-26	9.613E-26
3	2800	3	900	NI0DP	0.000E+00	1.176E-25	8.586E-26
4	2600	3	900	FE0ASDP	1.009E-27	8.646E-26	8.478E-26
5	2600	3	901	FE0EWDP	9.629E-28	8.512E-26	8.404E-26
6	3100	3	900	GA_ASDP	1.245E-26	2.512E-26	6.873E-29

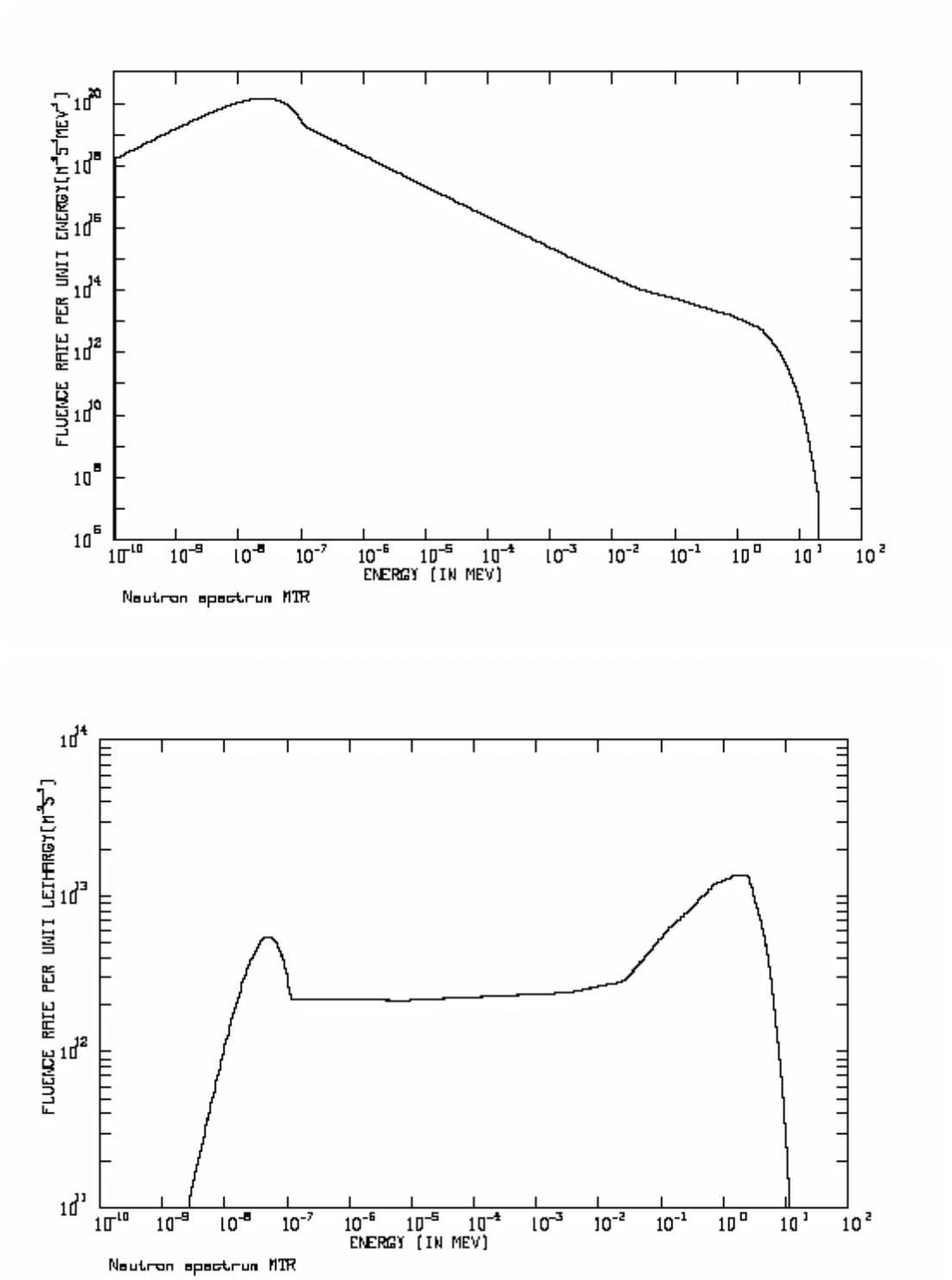

Figure 1. Neutron spectrum MTR – in two different representations – used in the calculations ([19]).

7. CHAPTERS WRITTEN FOR TECDOC ON IRDF-2002

Various chapters have been written for the IAEA-TECDOC on IRDF-2002 (in preparation). These chapters will appear in the TECDOC, and therefore are not produced in this document.

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