INDC(NDS)-0616
Distr. G+RD

International Atomic Energy Agency

INDC International Nuclear Data Committee

SUMMARY DESCRIPTION OF THE NEW INTERNATIONAL REACTOR DOSIMETRY AND FUSION FILE (IRDFF release 1.0)

Eva M. Zsolnay[1]
Roberto Capote Noy[2]
Henk J. Nolthenius[3]
Andrej Trkov[4]

[1] Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary
[2] International Atomic Energy Agency (IAEA), NAPC-Nuclear Data Section, Vienna, Austria
[3] The Netherlands
[4] Josef Stefan Institute, Ljubljana, Slovenia

May 2012

IAEA Nuclear Data Section, Vienna International Centre, P.O. Box 100, 1400 Vienna, Austria

Selected INDC documents may be downloaded in electronic form from
http://www-nds.iaea.org/publications/
or sent as an e-mail attachment.
Requests for hardcopy or e-mail transmittal should be directed to
services@iaeand.iaea.org
or to:
Nuclear Data Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna
Austria

Printed by the IAEA in Austria

May 2012

INDC(NDS)-0616
Distr. G+RD

SUMMARY DESCRIPTION OF THE NEW INTERNATIONAL REACTOR DOSIMETRY AND FUSION FILE (IRDFF release 1.0)

Eva M. Zsolnay[1]
Roberto Capote Noy[2]
Henk J. Nolthenius[3]
Andrej Trkov[4]

[1] Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary
[2] International Atomic Energy Agency (IAEA), NAPC-Nuclear Data Section, Vienna, Austria
[3] The Netherlands
[4] Josef Stefan Institute, Ljubljana, Slovenia

Abstract: A new cross section library, the "International Reactor Dosimetry and Fusion File release1.0 (IRDFF 1.0)" has been developed for reactor dosimetry and fusion applications. It contains cross sections with uncertainties (covariance matrices) for 74 dosimetry reactions and for B, Cd and Gd cover materials in the form of point-wise linearly interpolable data from 10^{-11} MeV up to at least 60 MeV, in ENDF-6 format. For reactor dosimetry applications these pointwise data were processed and made available in the SAND II type 640 group format (10^{-10} MeV to 20 MeV). 33 cross section datasets have been adopted with minor modifications from the file IRDF-2002; the others are new evaluations. This report summarizes the characterization of the new cross section evaluations in Maxwellian thermal (300K), 1/E and ^{252}Cf(sf) neutron fields, and presents the most important features and data of the resulted version of the new international reactor and fusion dosimetry cross section file.

IRDFF 1.0 data are available at the IAEA/Nuclear Data Section web: http://www-nds.iaea.org/IRDFF.

May 2012

TABLE OF CONTENTS

1. INTRODUCTION **7**

2. CHARACTERIZATION AND PRELIMINARY VALIDATION OF THE NEW CROSS SECTION EVALUATIONS 8

2.1. The method of characterization and validation of the new evaluations 8

2.2. Results, discussion and conclusions 9

3. THE NEW INTERNATIONAL REACTOR DOSIMETRY AND FUSION FILE IRDFF RELEASE 1.0 AND CHARACTERIZATION OF ITS DATA **13**

4. REFERENCES **15**

TABLES **17**

REFERENCES TO THE TABLES **41**

1. INTRODUCTION

The International Reactor Dosimetry File (IRDF-2002) was released on the web by the International Atomic Energy Agency (IAEA) in 2005 (http://www-nds.iaea.org/irdf2002) and the corresponding Technical Report was published in 2006 [1]. This library contained cross section data with uncertainty information in the form of covariance matrices for 66 reactions applied in the field of reactor dosimetry, and characteristic nuclear data for these reactions. Data on total and capture cross sections were also included for the cover materials B, Cd, Gd applied in some irradiations of dosimetry detectors. The cross section data cover the energy region from 10^{-10} MeV to 20 MeV. The library at the time of its preparation contained the best quality data available for reactor dosimetry applications; nevertheless, there remained several shortcomings in the file [2] (diagonal covariance matrices for some reaction cross sections, lack of experimental data, etc.). New data needs emerged from the fusion research community asking for a significant extension of IRDF-2002 library to include new reactions suitable for fusion studies and dosimetry[1], and to increase the upper energy limit of the cross section evaluations up to 60 MeV [3-5].

After releasing IRDF-2002, new cross section evaluations have become available, e.g. within the European JEFF-3.1 [6] and the US ENDF/B-VII [7-8] libraries, the IAEA neutron cross section standards file [9], and the recent evaluations of K.I. Zolotarev [10-13].

A Consultants' Meeting was held at the IAEA headquarters in Vienna, Austria on 5-7 May 2010 to review the requirements for improving and extending the content of the library IRDF-2002, taking into account the above mentioned points of view [14]. One of the primary objectives of the meeting was to discuss the selection and the appropriate methods for characterizing and implementing the new cross section evaluations in the current dosimetry file IRDF-2002, to obtain an updated cross section library suitable both for reactor and fusion dosimetry applications. The participants decided the actions needed for reaching the above aims; furthermore, recommendations and a preliminary work plan have also been proposed (see Ref.[14]).

To meet the demands of the fusion community the energy limits of all the cross section data intended to be put in the new library have been extended to at least 60 MeV, and a new library was assembled by adding the improved, extended and partially validated data set to IRDF-2002 [15]. The library obtained was named the **I**nternational **R**eactor **D**osimetry and **F**usion **F**ile: **IRDFF** release 1.0. This report summarizes the characterization (preliminary validation) of the new cross section evaluations in Maxwellian thermal, 1/E and ^{252}Cf spontaneous fission neutron fields, and presents the most important features and data of the resulted version of the new international reactor and fusion dosimetry cross section file, IRDFF release 1.0.

[1] See the IFMIF (International Fusion Materials Interaction Facility) project, the results of which will be used in the development of the DEMO fusion reactor demonstration power plant [3-5].

2. CHARACTERIZATION AND PRELIMINARY VALIDATION OF THE NEW CROSS SECTION EVALUATIONS

The cross section sets of the new evaluations were available in the present work as ENDF-6 formatted point-wise linearly interpolable data from 10^{-11} MeV up to at least 60 MeV (ENDF-6 files MF=3,33 for the stable; MF=10,40 for the radioactive products). The extension of the energy range was described in [15]. In the neutron energy region above 20 MeV there are no reference neutron fields available, therefore no validation of the cross section sets is being made at those energies.

2.1.The method of characterization and validation of the new evaluations

The data taken from the new cross section evaluations mentioned in the Introduction had to be analysed and validated in reference neutron fields [16] (Maxwellian thermal, 1/E and ^{252}Cf(sf)) in order to determine whether they are suitable for improving and extending the International Reactor Dosimetry File IRDF-2002. The analysis and validation procedure (characterization of the cross section data) involved the following actions:

1) Converting the linearly interpolable point cross section data into the extended SAND II type 640 group format using a flat weighting spectrum and codes PREPRO-2010 [17] and NJOY [18]. The later was used to process ENDF-6 point-wise files containing ENDF-6 file MF=32 uncertainty data

2) Critical review of the format and contents of both the flat-weighted 640 group cross section files and the original ENDF-6 formatted files.

3) Checking the neutron temperature of the cross section data: all cross section sets have to be given at 300 K neutron temperature to be comparable with the experimental data of S.F. Mughabghab [19] and to be suitable for reactor dosimetry applications. This requirement is especially important for capture and fission reactions.

4) Investigating the uncertainty and correlation information (covariance matrices) of the new cross section evaluations.

5) Calculating the integral values with uncertainties (spectrum average cross sections) in the following reference neutron fields recommended in [16] for reactor dosimetry using SAND II type 640 group data derived in point 1 :

   − Maxwellian thermal neutron spectrum (neutron temperature: 300 K) using evaluated experimental data of Refs. [19-21];
   − 1/E neutron spectrum (Resonance Integral, IR) using evaluated experimental data of Ref. [19];
   − ^{252}Cf spontaneous fission neutron field using evaluated experimental data of Ref. [22].

   The preliminary validation procedure involved the comparison of the **C**alculated integral data obtained from the library cross sections with the corresponding **E**xperimental values (C/E).

6) Comparison of the C/E data together with their uncertainties of the newly evaluated cross sections with the corresponding ones of the IRDF-2002 library.

7) Characterizing the integral cross-section data of the new evaluations also in the ^{235}U(n_{th},f) prompt fission neutron spectrum induced by thermal neutrons and taken from the ENDF-B/VII.1 library [8].

To get reliable results, the most recent evaluated experimental data (of spectrum averaged cross sections) were selected from the literature and applied in the calculations. The C/E values used for the cross section validation in the thermal and 1/E neutron energy regions are based on the evaluated experimental data of S.F. Mughabghab [19], with some data retrieved from the KAYZERO database of integral constants for Neutron Activation Analysis [20] as described in Ref. [21]. The C/E values used for the cross section validation in the fast neutron energy region up to 20 MeV are based on the evaluated experimental data of W. Mannhart [22] measured in the ^{252}Cf(sf) neutron field. The ^{252}Cf(sf) reference neutron spectrum was taken from the IRDF-2002 database [1]. The capture reactions are not characterized in the ^{252}Cf(sf) field as the important part (from dosimetry point of view) of their cross section is in the thermal and 1/E neutron energy regions.

For characterization of the cross section data in the 1/E neutron energy region the resonance integral (IR) was calculated from 0.5 eV to 20 MeV to correspond to the experimental data evaluated by S.F. Mughabghab [19]. In order that the new evaluations should be comparable with the corresponding IRDF-2002 library data, the resonance integral values of the latter ones had to be recalculated for the above mentioned neutron energy interval as the original IRDF-2002 resonance integral data refer to the neutron energy region from 0.5 eV to 1.05 MeV [1].

2.2. Results, discussion and conclusions

The above described investigations involved the format checking and modifications as well as the validation in the reference neutron fields [16] of 44 new cross section evaluations including reaction data for three cover materials: B, Cd and Gd. The neutron temperature was 300 K in all cases.

Results and discussion

1) Extension of the upper energy limit of the point-wise linearly interpolable cross section data up to 60 MeV in the ENDF-6 format for new evaluations was described in Ref.[15]. A similar work was undertaken for those evaluations adopted from the IRDF-2002 library. High energy data used for the extension was retrieved from the TENDL-2010 library [23]. TENDL-2010 cross sections were renormalized to the evaluated cross section at the highest available neutron energy.

2) Format problems and covariance information in the new evaluations:

a) The files with the new cross-section evaluations consisted of point-wise linearly interpolable cross section data in ENDF-6 files MF=3 and 10, the latter one for the radioactive product nuclides. The uncertainty of the data was given in the ENDF-6 format files MF=33 or MF=40, correspondingly. Original sources of these point data files comprised in a number of cases uncertainty information in ENDF-6 file MF=32 for capture and for some of the fission dosimetry reactions. To simplify the application of the uncertainty information for neutron metrology, these data were converted into the ENDF-6 file MF=33 or MF=40 before further use.

b) Full covariance matrices were found in all selected new evaluations. A rough investigation of these covariance matrices resulted in no negative eigenvalues of those matrices; their detailed analysis is left for a future work.

c) The size of the covariance matrices for the energy region below 20 MeV was much larger for the reactions ^{55}Mn(n,g), ^{197}Au(n,g), ^{232}Th(n,g), ^{235}U(n,f), ^{238}U(n,f), ^{238}U(n,g) and ^{239}Pu(n,f) than in cases of the other reactions. This is due to the fact that during the conversion of these covariance data (from the ENDF-6 file MF=32 to MF=33) a finer energy group was used. This fine energy structure for the covariance information may be important in the neutron metrology, if thick detector materials are used, resulting in significant neutron selfshielding [24].

3) The results of the characterization (preliminary validation) of the new cross section evaluations are presented in Table 1. This table contains the calculated spectrum averaged cross section in Maxwellian thermal (300K), 1/E and ^{252}Cf(sf) reference neutron spectra [16]. C/E values were obtained using the corresponding experimental data [19-22]. From the data of the Table it can be seen that eight new reactions – as compared with the content of the library IRDF-2002 – are present among the new cross section evaluations. Three of them - ^{59}Co(n,3n), ^{169}Tm(n,3n) and ^{209}Bi(n,3n) - are for high energy neutron dosimetry. Due to their high threshold energy the cross sections of these reactions can not be characterized in the ^{252}Cf spontaneous fission neutron spectrum.

For the reactions ^{60}Ni(n,p) and $^{92}Mo(n,p)^{92m}Nb$ only raw experimental data (based only on one measurement) are available in the literature for the ^{252}Cf spontaneous fission neutron spectrum [22]. Therefore the C/E values are not considered to be reliable in these cases.

In case of the reaction ^{6}Li(n,t) no evaluated experimental data are available for the resonance integral up to 20 MeV in the literature. Therefore, the resonance integral of the evaluated cross sections of this reaction could not be characterized in the same neutron energy region as in case of the other reactions.

The data of the reaction $^{115}In(n,\gamma)^{116m}In$ are derived from the file IRDF-2002, only the branching ratio between the metastable and ground states was changed in the present evaluation. The covariance data are the same as in case of IRDF-2002. The cross section of the reaction $^{139}La(n,\gamma)$ is also not a new evaluation, its data are derived from IRDF-2002. Nevertheless, we included this reaction in the Table, because its cross sections are characterized also by data derived from the KAYZERO (k_0) database of the neutron activation analysis [20]. The k_0 based data was also used for additional characterization of the cross sections in case of the reactions $^{55}Mn(n,\gamma)$, $^{58}Fe(n,\gamma)$ and $^{115}In(n,\gamma)^{116m}In$ [21]. The resonance integral data in these cases refer to the neutron energy region from 0.55 eV to 2.0 MeV [21] (the results can be seen in Table 1).

Cross section data of the reaction ^{241}Am(n,f) are derived from the file IRDF-2002. A large change (almost a factor of two) in the value of the resonance integral is observed, due to extension of the upper neutron energy limit up to 20 MeV in current calculations (in IRDF-2002 the upper energy limit of the calculated resonance integral was 1.05 MeV).

4) To show the improvement in the IRDF-2002 library cross section data by introducing the new cross section evaluations, we compared the C/E values of the corresponding reaction cross sections for the old/new data sets in Table 2. The resonance integrals refer to the neutron energy region from 0.5 eV to 20 MeV to be comparable with Mughabghab data [19]. The results show that the agreement between the calculated integral cross sections and the corresponding experimental ones (C/E) is generally better in case of the new evaluations than it was in the library IRDF-2002 with the exception of the reaction ^{60}Ni(n,p) in the ^{252}Cf(sf) reference neutron spectrum. For this reaction a much better C/E value has been obtained for the IRDF-2002 data than for the newly evaluated cross section. However, the uncertainty value in the former case is rather large (11.6 %); furthermore, the experimental value used in the calculations is a raw value based on one measurement only [1,22].

5) The integral values of the new cross section evaluations averaged over the $^{235}U(n_{th,}f)$ prompt fission neutron spectrum have also been calculated and compared with the corresponding experimental data [25]. The intercomparison was made for those cross-section evaluations, which have reliable experimental data in the literature [25] and their calculated integral cross sections have an acceptable uncertainty value (usually 10% or lower) in the energy region of interest. Therefore, only the reactions having the mean energy E(50%) of their integrated response below 6 MeV in the $^{235}U(n_{th,}f)$ spectrum have been taken into account. The results obtained are presented in Table 3. The data of the Table show that the uncertainty of the calculated spectrum average cross sections over the ENDF-B/VII.1 $^{235}U(n_{th,}f)$ prompt fission neutron spectrum is generally much larger than the one obtained in the $^{252}Cf(sf)$ neutron spectrum. This situation reflects the higher uncertainty of the ENDF-B/VII.1 $^{235}U(n_{th,}f)$ spectrum [8] compared to the $^{252}Cf(sf)$ reference spectrum at higher neutron energies.

Conclusions

Reviewing the data of Tables 1-3, the following conclusions can be drawn:

1) Altogether 41 new dosimetry cross section evaluations accompanied with their uncertainty information in the form of covariance matrices have been analysed and validated (in the extended SAND II type 640 group format) in three reference neutron fields for reactor dosimetry [16] (Maxwellian thermal, 1/E and $^{252}Cf(sf)$ neutron fields). Newly evaluated cross sections of three cover materials B, Cd and Gd (without uncertainty information) have also been investigated, but there is a lack of experimental data to characterize these cross sections (see Table 1).

2) The results of the characterization (preliminary validation) of the new cross section data are summarized in Table 1. The Table shows that eight new reactions (marked **N**) are present as compared with the content of the library IRDF-2002. The new reactions are: $^{55}Mn(n,\gamma)$, $^{59}Co(n,3n)$, $^{59}Co(n,p)$, $^{67}Zn(n,p)$, $^{92}Mo(n,p)^{92m}Nb$, $^{115}In(n,n')^{115m}In$, $^{169}Tm(n,3n)$ and $^{209}Bi(n,3n)$. Three of them - $^{59}Co(n,3n)$, $^{169}Tm(n,3n)$ and $^{209}Bi(n,3n)$ - can be used only for high energy neutron dosimetry (above 20 MeV).

3) The calculated integral values of the newly evaluated cross sections are generally in good agreement with the corresponding experimental data (C/E is in the vicinity of 1.00, see Table 1). However, there are some cases where the integral of the library cross section data showed a large deviation from the corresponding experimental value:

    a) For the reactions $^{60}Ni(n,p)$ and $^{92}Mo(n,p)$ raw experimental data (based only on one measurement) are available in the literature for the ^{252}Cf spontaneous fission neutron spectrum [22]. Therefore the C/E value is not reliable in these cases (see the data of Table 1). At the same time, the C/E value for the reaction $^{60}Ni(n,p)$ is acceptable for the data of the file IRDF-2002 (see Table 2). But, it has to be mentioned, that the uncertainty of this C/E value is rather large (11.6 %). New integral (SPA) measurements are needed to resolve this discrepancy.

    b) No integral experimental data are available in the ^{252}Cf spontaneous fission neutron spectrum for the cross sections of several reactions: $^{6}Li(n,t)$, $^{10}B(n,\alpha)$, $^{67}Zn(n,p)$, $^{89}Y(n,2n)$, $^{115}In(n,2n)^{114m}In$, and $^{241}Am(n,f)$ (Table 1). Therefore, the cross sections of these reactions cannot be characterized in the above neutron field. New measurements and updated evaluations for these reactions are welcome. The reactions $^{59}Co(n,3n)$, $^{169}Tm(n,3n)$ and $^{169}Tm(n,3n)$ have a high threshold energy; therefore they could be used for high energy neutron dosimetry above 20 MeV. However, new reference spectra are needed to characterize these reactions for neutron energies above 20 MeV.

c) No evaluated experimental data are available for the resonance integral for the reaction ^{6}Li(n,t).

d) The resonance integrals of the reactions ^{58}Fe(n,γ) and ^{237}Np(n,f) in the new evaluations are discrepant with Mughabghab experimental values [19] (Table 1). In case of the reaction ^{58}Fe(n,γ) new evaluated data agree well with values derived from KAYZERO database [20,21]. Further investigation of discrepancies is needed. The reaction ^{237}Np(n,f) is used for fast neutron dosimetry, therefore the detected deviation in the resonance integral derived from the library cross section data does not affect its reactor dosimetry application.

4) The data of the k_0 database of the neutron activation analysis [20] were used for additional characterization of the cross sections of the following reactions: ^{55}Mn(n,γ), ^{58}Fe(n,γ), ^{115}In(n,γ)^{116m}In and ^{139}La(n,γ) (Table 1). The resonance integral in these cases refers to the neutron energy interval from 0.55 eV to 2.0 MeV [21]. The obtained results generally agree within one standard deviation with the corresponding ones referring to the neutron energy region between 0.5 eV and 20 MeV evaluated by Mughabghab [19]. Exceptions are the experimental values of the resonance integral in case of the reaction ^{58}Fe(n,γ), and of the thermal cross sections of the reactions ^{55}Mn(n,γ) and ^{139}La(n,γ) [21]. In these cases the data based on the k_0 database [20] are discrepant with evaluated values of S.F. Mughabghab [19]. Further investigation of these discrepancies is warranted.

5) The C/E values for the corresponding reaction cross sections in the new IRDFF library and in the IRDF-2002 library have been compared in Table 2. The resonance integrals refer also here to the neutron energy interval between 0.5 eV and 20 MeV. The results show that the agreement between the calculated library cross sections and the corresponding experimental data is generally better in case of the new evaluations than it was in the library IRDF-2002. That means that the quality of the cross section data in the International Reactor Dosimetry File IRDF-2002 can be improved with the new cross section evaluations considered in this Report.

6) The integral cross sections averaged over the ENDF-B/VII.1 ^{235}U(n_{th},f) prompt fission neutron spectrum [8] have an unacceptable large uncertainty in case of the detectors with high threshold energies, therefore only reactions having the mean energy E(50%) of their integrated response below 6 MeV have been taken into account (Table 3). This fact is due to the large uncertainty of the evaluated ENDF/B-VII.1 ^{235}U(n_{th},f) neutron spectrum in the high neutron energy region. It should be noted that the ^{235}U(n_{th},f) spectrum is not considered in the neutron metrology as a reference one [16].

7) The size of the covariance matrices for the energy region below 20 MeV was much larger for the reactions ^{55}Mn(n,g), ^{197}Au(n,g), ^{232}Th(n,g), ^{235}U(n,f), ^{238}U(n,f), ^{238}U(n,g) and ^{239}Pu(n,f) than in cases of the other reactions. This is due to the fact that a finer energy structure was used during the conversion of the covariance data (from the ENDF-6 file MF=32 to MF=33) in case of these reactions than for the other ones. Larger dimensions introduce an additional uncertainty in the positive definitness of such matrices. Further investigations are warranted.

Based on the above results, several data of the old reactor dosimetry library IRDF-2002 have been replaced by the new evaluations and numerous reaction cross sections from the new datasets have been added to IRDF-2002. In this way a new library named IRDFF release 1.0 was obtained. The content and characterization of this new cross section library is presented in the next chapter.

3. THE NEW INTERNATIONAL REACTOR DOSIMETRY AND FUSION FILE IRDFF release 1.0 AND CHARACTERIZATION OF ITS DATA

The new International Reactor Dosimetry and Fusion File IRDFF release 1.0 contains cross section data with uncertainty information in the form of covariance matrices for 74 dosimetry reactions, and specific nuclear data for these reactions. Besides the dosimetry reactions there are available cross section data (without uncertainty information) also for three cover materials B, Cd and Gd used during the irradiation of some specific detectors. The primary cross section sets in the library are available as point-wise linearly interpolable data from 10^{-11} MeV at least up to 60 MeV (in ENDF-6 format). The neutron temperature is 300 K in all cases (the exact value of the most probable velocity of the Maxwellian distribution at the temperature 300 K is 2223 m/s). These cross section data have been processed and they are also available in the extended SAND II type 640 group format in the neutron energy region from 10^{-10} MeV to 20 MeV for reactor dosimetry applications.

Part of the cross section data in the new library is derived from the former International Reactor Dosimetry File IRDF-2002 (33 reactions), 50% of the old reactions were replaced by new evaluations (33), furthermore also eight new reactions were added to the file. Extension of the 20 MeV upper energy limit of the point-wise linearly interpolable cross section data up to 60 MeV for all reactions was done using the information of the TENDL-2010 library [23], and described in Ref.[15]. The content and the most important basic data of the new cross section library are summarized in Table 4.

Characterization of the cross section data present in the new library IRDFF release 1.0 is summarized in Table 5 for three reference neutron fields [16] of reactor dosimetry (Maxwellian thermal, 1/E and ^{252}Cf spontaneous fission neutron fields). The Table lists the cross sections included in the new library, together with their integral characteristics and the ratios of the corresponding calculated and experimental integral cross section data (C/E). The uncertainties of the C/E values involve the standard deviations both of the calculated (including the uncertainty of the neutron spectra) and of the experimental cross section data. The resonance integral values are calculated in all cases from 0.5 eV up to 20 MeV (in the file IRDF-2002 the resonance integral referred to the neutron energy region from 0.5 eV to 1.05 MeV, but for the new library also these data have been recalculated to the energy interval from 0.5 eV to 20 MeV). The capture reactions are not characterized in the ^{252}Cf spontaneous fission neutron spectrum, as the important part of their cross sections for dosimetry is in the low neutron energy region. For the reaction $^{45}Sc(n,\gamma)$ the cross section data in the new library agree with the ones of IRDF-2002, but the uncertainty data (a part of the covariance information) were modified by IAEA NDS. Similarly, the data of the reaction $^{115}In(n,\gamma)^{116m}In$ are copied from IRDF-2002, only the branching ratio between the metastable and ground state was changed in the present evaluation. The covariance data are the same as in case of IRDF-2002.

Detailed analysis of the cross section sets derived from the file IRDF-2002 can be found in the IAEA Technical Report no. 452 [1], while the preliminary validation (detailed analysis) of the new cross section evaluations was presented in chapter 2 of this report.

The thermal cross sections and the resonance integrals of the neutron capture and thermal fission reactions present in the new library are characterized in Table 6. The results show that the calculated and experimental thermal cross section data have a good agreement (see the corresponding C/E values), while for the resonance integrals the C/E values are discrepant for ^{58}Fe(n,γ) (if Mughabghab evaluation [19] is considered), ^{93}Nb(n,γ) and ^{237}Np(n,f) reactions. This discrepancy will have to be treated in the next revision of the library.

As a result of the activities described in chapter 2 of this report, the quality of the cross section data in the new International Reactor Dosimetry and Fusion File IRDFF release 1.0 has been meaningfully improved and its content updated. The upper energy limit of the cross section data in the library has been extended as compared with the International Reactor Dosimetry File IRDF-2002.

The following shortcomings of the validation process and of the new library are noted:

1) No experimental cross section values are available in the ^{252}Cf spontaneous fission neutron (reference) field for the following reactions: ^{23}Na(n,2n), ^{31}P(n,p), ^{46}Ti(n,2n), ^{47}Ti(n,x)^{46}Sc, ^{48}Ti(n,x)^{47}Sc, ^{49}Ti(n,x)^{48}Sc, ^{52}Cr(n,2n), ^{54}Fe(n,2n), ^{54}Fe(n,α), ^{59}Co(n,3n), ^{67}Zn(n,p), ^{75}As(n,2n), ^{89}Y(n,2n), ^{115}In(n,2n)^{114m}In, ^{141}Pr(n,2n), ^{169}Tm(n,3n), ^{209}Bi(n,3n), and ^{241}Am(n,f).

2) No evaluated experimental cross section data are available in the literature for the resonance integral up to 20 MeV for the reaction ^{6}Li(n,t).

3) The reactions ^{59}Co(n,3n), ^{169}Tm(n,3n) and ^{209}Bi(n,3n) are for high energy neutron dosimetry. Due to their high threshold energy the cross sections of these reactions cannot be characterized in the ^{252}Cf spontaneous fission neutron spectrum. Experimental data would be needed for these reactions measured in a new reference neutron spectrum with energies above 20 MeV.

4) For the reactions ^{60}Ni(n,p), ^{92}Mo(n,p)^{92m}Nb and ^{103}Rh(n,n')^{103m}Rh the C/E value in the ^{252}Cf spontaneous fission neutron spectrum highly deviates from unity. In the first two cases it may be due to the fact, that raw experimental cross section data (based only on one measurement) are available in the literature for the SPA cross section measured in ^{252}Cf spontaneous fission neutron spectrum. Evaluated experimental data (based on several independent measurements) would be needed to characterize the library cross sections of these reactions. For the reaction ^{103}Rh(n,n')^{103m}Rh further investigations are needed to find the reason of the discrepancy.

5) The resonance integral has a large deviation from the recommended experimental values [19] for the following reactions: ^{58}Fe(n,γ), ^{93}Nb(n,γ) and ^{237}Np(n,f), but ^{58}Fe(n,γ) data agree well with KAYZERO database [20]. New measurements and revision of the resonance parameters in ^{93}Nb(n,γ) and ^{237}Np(n,f) evaluations are recommended.

6) The covariance matrices of the new evaluations as a result of a rough investigation do not have negative eigenvalues. Further detailed analysis is in progress.

7) Diagonal covariance matrices (i.e. only uncertainties without correlations) are still present for the cross sections of the following reactions: ^{23}Na(n,γ), ^{58}Fe(n,γ), ^{93}Nb(n,γ), ^{115}In(n,γ),and ^{181}Ta(n,γ). Further improvement of covariance data is needed.

8) Newly evaluated cross sections of three cover materials B, Cd and Gd have also been investigated, but the lack of experimental data made impossible their characterization. At the same time, there is no uncertainty information available for the newly evaluated cross sections of these cover materials.

4. REFERENCES

[1] O. Bersillon, L.R. Greenwood, P.J. Griffin, W. Mannhart, H. J. Nolthenius, R. Paviotti-Corcuera, K.I. Zolotarev, E.M. Zsolnay, P.K. McLaughlin, and A. Trkov, International Reactor Dosimetry File 2002 (IRDF-2002), 2006, International Atomic Energy Agency, Vienna, Austria, Technical Reports Series No. **452**, December 2006. Database available online at http://www-nds.iaea.org/irdf2002.

[2] E.M. Zsolnay, H.J. Nolthenius, and A.L. Nichols: Investigation of New Reaction Cross-Section Evaluations in Order to Update and Extend the IRDF-2002 Reactor Dosimetry Library. In: Proc of the 13th International Symposium on Reactor Dosimetry, Akersloot, The Netherlands, 25-30 May 2008. „Reactor Dosimetry State of Art 2008". (Eds.:W. Voorbraak, L. Debarberis, P. D'hondt, J. Wagemans). World Scientific, 2009.

[3] U. Fischer, *et al.*, Nuclear Data for Fusion Energy Technologies: Requests, Status and Development Needs. In AIP Proc. Int. Conf. Nuclear Data for Science and Technology (Eds.: R.C. Haight, M.B. Chadwick, T. Kawano and P. Talou), **AIP769**, pp. 1478-1485 (AIP, Melville, New York, 2005).

[4] P. Batistoni *et al.*, Neutronics Design and Supporting Experimental Activities in the EU, *Fusion Eng. Design* **81** (2006) pp.1169-1181.

[5] R. Garabedian *et al.*, Design of the DEMO Fusion Reactor following ITER. *Inst. Stand. Technol.* **114** (2009) pp. 229-236

[6] A.J. Koning *et al.*, The JEFF-3.1 Nuclear Data Library. *JEFF Report* **21**, NEA/OECD, Paris, France (2006) ISBN 92-64-02314-3.

[7] M.B. Chadwick *et al.*, ENDF/B-VII.0: Next Generation Evaluated Nuclear Data Library for Nuclear Science and Technology, *Nuclear Data Sheets* **107** (2006) pp.2931-3060.

[8] M. Chadwick *et al.*, ENDF/B-VII.1. Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data. *Nuclear Data Sheets* **112** (2011) pp.2887-2996.

[9] A. D. Carlson *et al.*, International Evaluation of Neutron Cross Section Standards. *Nuclear Data Sheets* **110** (2009) 3215.

[10] K.I. Zolotarev: Re-evaluation of Microscopic and Integral Cross Section Data for Important Dosimetry Reactions. Technical report **INDC(NDS)-0526** (IAEA, Vienna, August 2008).

[11] K.I. Zolotarev: Evaluation of Cross-Section Data from Threshold to 40-60 MeV for Specific Neutron Reactions Important for Neutron Dosimetry Applications. Technical report **INDC(NDS)-0546** (IAEA, Vienna, April 2009).

[12] K.I. Zolotarev: Evaluation of Cross-Section Data from Threshold to 40-60 MeV for Some Neutron Reactions Important for Fusion Dosimetry Applications. Technical report **INDC(NDS)-0584** (IAEA, Vienna, November 2010).

[13] K.I. Zolotarev: Evaluation of selected (n,n'), (n,γ), (n,p), (n,2n) and (n,3n) Excitation Functions for Fission and Fusion Reactors Dosimetry Applications. Progress Report on IAEA Research Contract 16242, submitted to the IAEA on August 2011 (unpublished).

[14] M.A. Kellett and L.R. Greenwood, Summary Report of Consultants' Meeting at IAEA Headquarters, Vienna, Austria, 5-7 May 2010: „Improvement and Extensions to IRDF-2002. Technical report **INDC(NDS)-0575** (Vienna, Austria, December 2010).

[15] R. Capote, K.I. Zolotarev, V.G. Pronyaev, A.Trkov, "Updating and extending the IRDF-2002 Dosimetry Library". Presented at the 14th Int. Symp. on Reactor Dosimetry, held on May 22-27, 2010, Bretton Woods, NH, USA; to be published in *Journal of ASTM International* **9** (2012).

[16] "Supplementary Workshop on Testing of the IRDF-2002 File" chaired by B. Osmera and J.G.Williams. In: WORKSHOP SESSION III. Dosimetry for Irradiation Facilities and Test and Research Reactors. Proc. of the 11th International Symposium on Reactor Dosimetry.

Reactor Dosimetry in the 21st Century. Brussels, Belgium 18-23 August 2002 (Editors: J. Wagemans. H.A. Abderrahim, P. D'hondt, and C. De Raedt). World Scientific (New Jersey, London, Singapore, Hong Kong, 2003), pp.718-719.

[17] D.E. Cullen, **PREPRO 2010**, ENDF/B Pre-processing Codes, International Atomic Energy Agency, Vienna, Nov. 2010; available online at http://www-nds.iaea.org/ndspub/endf/prepro/.

[18] R.E. MacFarlane and D.M. Muir, "NJOY-99.0: Code System for Producing Pointwise and Multigroup Neutron and Photon Cross Sections from ENDF/B Data", Technical report PSR-480, Los Alamos National Laboratory (2000). See also OECD/NEA compilation of NJOY updates; available online at http://www.nea.fr/.

[19] S.F. Mughabghab, „Atlas of Neutron Resonances", Elsevier, ISBN-10:0-444-52035-X. Amsterdam-Tokio (2006).

[20] F. De Corte and A. Simonits, Recommended nuclear data for use in the k_0 standardization of neutron activation analysis, *Atom. Data & Nucl. Data Tables* 85 (2003) pp. 47–67.

[21] I. Kodeli and A. Trkov, Validation of the IRDF-2002 dosimetry library. Nucl. Instr. and Meth. In Physics Research **A577** (2007).

[22] W. Mannhart: "Response of Activation Reactions in the Neutron Field of Californium-252 Spontaneous Fission". In: International Reactor Dosimetry File 2002 (IRDF-2002) International Atomic Energy Agency, Technical Reports Series no. **452**,Vienna, 2006

[23] D. Rochman and A.J. Konig, TENDL-2010: Reaching completeness and accuracy, OECD/NEA, JEFF document **JEF/DOC-1349**, November 2010; TENDL-2010 nuclear data library available online at: http://www.talys.eu/tendl-2010

[24] G. Zerovnik *et al.*, Influence of resonance parameters' correlations on the resonance integral uncertainty; ^{55}Mn case. Nucl. Instr. and Meth. in Physics Research **A 632** (2011) pp. 137-141.

[25] W. Mannhart: Status of the Evaluation of the Neutron Spectrum of the ^{235}U+nth. In: IAEA Consultant's Meeting on International Neutron Cross-Section Standards: Measurements and Evaluation Techniques, Technical Report **INDC(NDS)-0540** (IAEA, Vienna, 2008). Presentation available at http://www.nds.iaea.org/standards/CM2010/Mannhart-U235-PFNS-CM2008.pdf

TABLES
(with own References)

Table 1. CHARACTERIZATION OF THE NEW CROSS SECTION EVALUATIONS (NEUTRON TEMPERATURE 300K)

Serial No.	Reaction name and evaluation	Maxwellian thermal (300 K)		1/E (Resonance Integral)		^{252}Cf spontaneous fission neutron spectrum		C/E
		Calc. libr. integr. data (barn)	Exp. data [1] (barn)	Calc. libr. integr.data (barn)	Exp. data [1] (barn)	Calc. libr. integr.data (mb)	Exp. data [2] (mb)	
1.	$^{6}Li(n,t)$ [3,4,10]	938.47 ± 0.14 %	940.00 ± 0.42%	423.93 ± 0.13 %	**No data ••**	321.4 ± 0.86%	**No exp.data**	Thermal: 0.998±0.44 % Res. Int. : **No exp. data ••** $^{252}Cf(sf)$: **No exp. data**
2.	$^{10}B(n,\alpha)$ [3,4,10]	3842.60 ± 0.33 %	3837 ± 0.23 %	1724.5 ± 0.33 %	1722 ± 0.29 %	446.3 ± 1.0%	**No exp.data**	Thermal: 1.001±0.40 % Res. Int.: 1.001±0.44 % $^{252}Cf(sf)$: **No exp. data**
3.	$^{24}Mg(n,p)$ [7]	---	---	---	---	2.105 ± 1.78 %	1.996 ± 2.44 %	1.055±3.02 %
4.	$^{27}Al(n,p)$ [5]	---	---	---	---	4.751 ± 2.35 %	4.880 ± 2.14 %	0.974±3.18%
5.	$^{27}Al(n,\alpha)$ [6]	---	---	---	---	1.019 ± 1.77 %	1.016 ± 1.47 %	1.003±2.30 %
6.	$^{32}S(n,p)$ [7]	---	---	---	---	74.11 ± 2.59 %	72.54 ± 3.49 %	1.003±4.35%
7.	$^{45}Sc(n,\gamma)$ [18] ♠	27.208 ± 6.86 %	27.200 ± 0.74 %	11.888 ± 7.61 %	12.10 ± 4.13 %	---	---	Thermal: 1.000±6.90 % Res. Int.: 0.982±8.66 %
8.	$^{47}Ti(n,p)$ [8]	---	---	---	---	19.56 ±2.80 %	19.27 ±1.66 %	1.015±3.26 %
9.	$^{55}Mn(n,2n)$ [6] **N**	---	---	---	---	0.416 ± 3.75 %	0.407 ± 2.33 %	1.022±4.42 %
10.	$^{55}Mn(n,\gamma)$ [3,9]	13.278 ± 1.11 %	13.60 ± 0,37 % 13.18* ± 0.92 %	13.526 ± 3.82 % 13.246** ± 3.82 %	13.400 ± 3.73 % 13.879** ± 2.76 %	---	---	Thermal: 0.994±1.17 % Thermal*: 1.007±1.44 % Res. Int.: 1.009±5.34 % Res. Int. **: 0.954±4.71 %

Serial No.	Reaction name and evaluation	Maxwellian thermal (300 K)		1/E (Resonance Integral)		^{252}Cf spontaneous fission neutron spectrum		C/E
		Calc. libr. integr. data (barn)	Exp. data [1] (barn)	Calc. libr. integr.data (barn)	Exp. data [1] (barn)	Calc. libr. integr.data (mb)	Exp. data [2] (mb)	
11.	^{58}Fe(n,γ) [12]	1.315 ± 5.07 %	1.320 ± 2.27 % 1.300* ± 2.66 %	1.275 ± 4.93 %. 1.246** ± 4.93 %	1.500 ± 4.67 % 1.267** ± 2.84 %	---	---	Thermal :0.996±5.55% **Res. Int.: 0.850 ±6.79%** Thermal*: 1.011±5.73 % Res. Int. **: 0.983±5.69 %
12.	^{59}Co(n,2n) [6]	---	---	---	---	0.408 ± 3.63 %	0.405 ± 2.51 %	1.007±4.41 %
13.	^{59}Co(n,3n) [16] **N**	---	---	---	---	---	**No data**	**High energy dosimetry!**
14.	^{59}Co(n,p) [6] **N**	---	---	---	---	1.715 ±3.65 %	1.690 [6] ± 2.48 %	1.015±4.41 %
15.	^{60}Ni(n,p) [7]	---	---	---	---	2.803 ± 2.27 %	2.390 ± 5.44 % **Raw data!**	**1.173 ±5.89 % Raw data in ^{252}Cf(sf) neutron field!**
16.	^{63}Cu(n,2n) [7]	---	---	---	---	0.198 ± 4.39 %	0.184 ± 3.98 %	1.076±5.93 %
17.	^{65}Cu(n,2n) [7]	---	---	---	---	0.654 ± 3.50 %	0.658 ± 2.22 %	0.994±4.14 %
18.	^{64}Zn(n,p) [7]	---	---	---	---	42.72 ± 1.86 %	42.20 [7, 14] ± 2.3 %	1.012±2.96 %
19.	^{67}Zn(n,p) [15] **N**	---	---	---	---	1.106 ± 5.31 %	**No data**	**No experimental data!**
20.	^{89}Y(n,2n) [16]	---	---	---	---	0.345 ± 4.42 %	**No data**	**No experimental data!**
21.	^{90}Zr(n,2n) [6]	---	---	---	---	0.217 ± 5.17 %	0.221 ± 2.89 %	0.982±5.92 %

Serial No.	Reaction name and evaluation	Maxwellian thermal (300 K)		1/E (Resonance Integral)		^{252}Cf spontaneous fission neutron spectrum		C/E
		Calc. libr. integr. data (barn)	Exp. Data [1] (barn)	Calc. libr. integr.data (barn)	Exp. data [1] (barn)	Calc. libr. integr.data (mb)	Exp. data [2] (mb)	
22.	^{92}Mo(n,p)^{92m}Nb [15] **N**	---	---	---	---	7.835 ± 3.75 %	15.17 [15] ± 4.4 % **Raw data**	**0.516±5.78 %** **Raw data in ^{252}Cf(sf) neutron field!**
23.	^{93}Nb(n,2n)^{92m}Nb [16]	---	---	---	---	0.791 ± 2.38 %	0.791 [16] ± 4.5 %	1.000±5.09 %
24.	^{113}In(n,n')^{113m}In [15] **N**	---	---	---	---	158.1 ± 1.24 %	161,2 [15] ± 2.04 %	0.981±2.39 %
25.	^{115}In(n,n')^{115m}In [17]	---	---	---	---	190.64 ± 1.70 %	197.40 ± 1.37 %	0.966±2.18 %
26.	^{115}In(n,2n)^{114m}In [7]	---	---	---	---	1.633 ± 5.51 %	**No data**	**No experimental data!**
27.	^{115}In(n,γ) ^{116m}In [18]$^{+}$	159.8 ± 6.00 %	162.3 ± 0.43 % 160,24***** ±6.23 %	2538.10 ±5.99 % 2538****** ±5.99 %	2650 ±3.77 % 2692****** ±6.51 %	---	---	Thermal: 0.985±6.15 % Thermal*****: 0.997±8.65 % Res. Int.: 0.958±7.08 % Res. Int. ******: 0.943±8.85 %
28.	^{127}I(n,2n) [7]	---	---	---	---	2.107 ± 3.83 %	2.069 [7] ± 2.42 %	1.018±4.53 %
29.	^{139}La(n,γ) [18] ♣	9.041 ± 3.85 %	9.040 ± 0.44 % 9.420***** ± 1.78 %	12.109 ± 5.54 % 11.915****** ± 5.41 %	12.100 ± 4.96 % 11.681****** ± 3.00 %	---	---	Thermal: 1.000±3.88 % Thermal*****: 0.960±4.24 % Res. Int.: 1.000±7.44 % Res. Int. ******: 1.020±6.30%
30.	^{169}Tm(n,2n) [16]	---	---	---	---	6.268 ± 3.1 %	6.384 [16] ± 6.28 %	0.982±7 %
31	^{169}Tm(n,3n) [15] **N**	---	---	---	---	---	**No data**	**High energy dosimetry!**

Serial No.	Reaction name and evaluation	Maxwellian thermal (300 K)		1/E (Resonance Integral)		^{252}Cf spontaneous fission neutron spectrum		C/E
		Calc. libr. integr. data (barn)	Exp. data [1] (barn)	Calc. libr. integr.data (barn)	Exp. data [1] (barn)	Calc. libr. integr.data (mb)	Exp. Data [2] (mb)	
32.	^{186}W(n,γ) [3,9]	38.095 ± 5.92 %	38.100 ± 1.31 %	484.20 ± 4.43 %	480.00 ± 3.13 %	---	---	Thermal: 1.000±6.06 % Res. Int.: 1.009±5.42 %
33.	^{197}Au(n,2n) [7]	---	---	---	---	5.531 ± 2.75 %	5.506 [7] ± 1.83 %	1.005±3.30 %
34.	^{197}Au(n,γ) [4,10, 19]	98.70 ± 0.48 %	98.650 ± 0.10 %	1570.5 ± 2.07 %	1550.0 ± 1,81 %	---	---	Thermal: 1.000±0.49 % Res. Int.: 1.013±2.75 %
35.	^{199}Hg(n.n’)^{199m}Hg [7]	---	---	---	---	296.3 ± 3.66 %	298.4 ± 1.81 %	0.993±4.08 %
36	^{209}Bi(n,3n) [16] **N**	---	---	---	---	---	**No data**	**High energy dosimetry!**
37.	^{232}Th(n,f) [9, 20, 21]	---	---	---	---	77.56 ± 2.14 %	76.90 ± 3.8 % [14,21]	1.009±4.36 %
38.	^{232}Th(n,γ) [9,20]	7.338 ± 1.24 %	7.350 ± 0.40 %	84.325 ± 1.90 %	83.300 ± 1.80 %	---	---	Thermal: 0.998±1.30 % Res. Int.: 1.012±2.62 %
39	^{235}U(n,f) [4,10,19,21]	585.10 ± 0.35 %	582.6 ± 0.20 %	276.02 ± 0.29 %	275.00 ± 1.82 %	1224.8 ± 0.42 %	1210 ± 1.20 %	Thermal: 1.004±0.40 % Res. Int.: 1.004±1.84 % 1.012±1.27 %
40.	^{237}Np(n,f) [22]	0.021 ± 10.00 %	0.020 ± 5.00 %	6.929 ± 3.81 %	4.70 ± 4.26 %	1360 ± 1.71 %	1361 ± 1.59 %	Thermal: 1.050±11.18 % **Res. Int.: 1.474±5.72 %** 0.999±2.33 % **Fast neutron dosimetry!**
41.	^{238}U(n,f) [4,10,19,21]	---	---	---	---	318.5 ± 0.66 %	325.7 ± 1.64 %	0.978±1.77 % **Fast neutron dosimetry!**

Serial No.	Reaction name and evaluation	Maxwellian thermal (300 K)		1/E (Resonance Integral)		^{252}Cf spontaneous fission neutron spectrum		C/E
		Calc. libr. integr. data (barn)	Exp. data [1] (barn)	Calc. libr. integr.data (barn)	Exp. data [1] (barn)	Calc. libr. integr.data (mb)	Exp. data [2] (mb)	
42.	^{238}U(n,γ) [4,10,19,21]	2.686 ± 1.84 %	2.680 ± 0.71 %	275.59 ± 1.45 %	277.11 ± 1.08 % [11]	---	---	Thermal: 1.002±1.97 % Res. Int.: 0.995±1.81 %
43.	^{239}Pu(n,f) [4,10,19,21]	748.20 ± 1.08 %	748.10 ± 0.27 %	302.81 ± 0.72 %	303.00 ± 3.30 %	1796 ± 0.46 %	1812 ± 1.37 %	Thermal: 1.000±1.11 % Res. Int.: 0.999±3.38 % 0.991±1.44 %
44.	^{241}Am(n,f) [18] $^{\times}$	3.018 ± 2.00 %	3.20 ± 2.81 %	13.86 ± 1.74 %	14.40 ± 6.94 %	1397.2 ± 2.85 %	**No data**	Thermal: 0.943±3.45 % Res. Int.: 0.962±7.15 % ^{252}Cf(sf) : **No exp. data** **Fast neutron dosimetry!**

COVER MATERIALS

Serial No.	Reaction name and evaluation	Maxwellian thermal (300 K)		1/E (Resonance Integral)		C/E
		Calc. libr. integr. data (barn)	Exp. data [1] (barn)	Calc. libr. integr.data (barn)	Exp. Data [1] (barn)	
1.	B – Absorber [3]	765.06 **No uncert**.	767 ± 1.041 %	343.2 **No uncert.**	---	Thermal: 0.997 **No uncertainty!**
2.	Cd – Absorber [20,23]	2487.5 **No uncert**.	2520 ± 1.98 %	66.878 **No uncert.**	70.00 ±14.28 %	Thermal: 0.987 RI: 0.950 **No uncertainty!**
3.	Gd – Absorber [3]	45945 **No uncert**.	48890 ± 0.13 %	389.11 **No uncert**.	390.00 ± 2.56 %	Thermal: 0.940 RI: 0.998 **No uncertainty**

REMARKS to TABLE 1.

The calculated thermal cross section values in this Table refer to 300 K.

The calculated and experimental values of the resonance integral (1/E region) in this Table refer to the neutron energy region from 0.5 eV to 20 MeV (to be consistent with S.F. Mughabghab [1]), except the cases flagged with the mark **.

The capture reactions are not characterized in the ^{252}Cf spontaneous fission neutron spectrum, as the important part of their cross sections for dosimetry is in the low neutron energy region.

The C/E data without any remark refer to the ^{252}Cf spontaneous fission neutron field. The evaluated experimental cross section values in the ^{252}Cf spontaneous fission spectrum are deriving from W. Mannhart [2], otherwise the Reference is given besides the corresponding experimental value.

N New reaction.

●● In case of the reaction ^{6}Li(n,t) for the resonance integral no evaluated experimental data are available up to 20 MeV.

♠ For the reaction ^{45}Sc(n,γ) the cross section data agree with the ones of IRDF-2002, but the uncertainty data (conversion of a part of the covariance information) were modified by IAEA NDS.

+ The data of the reaction ^{115}In(n,γ)^{116m}In are taken from IRDF-2002, only the branching ratio between the metastable and ground state was changed in the present evaluation. The covariance data are the same as in case of IRDF-2002.

♣ The reaction ^{139}La(n,γ) is not a new evaluation, data are taken from IRDF-2002. Its cross sections are characterized also by data derived from the k_0 database of the neutron activation analysis [11].

***** The experimental cross section in the thermal neutron energy region is derived from the k_0 database of the neutron activation analysis [11].

****** The resonance integral is calculated and the corresponding experimental value of it (derived from the k_0 database of the neutron activation analysis [11]) is given in the neutron energy region from 0.55 eV to 2 MeV.

$^{\times}$ Not new evaluation, it is from the file IRDF-2002 (see details in the text).

Table 2. COMPARISON OF THE C/E VALUES FOR THE NEW CROSS SECTION EVALUATIONS IRDFF release 1.0

WITH THE CORRESPONDING DATA OF THE LIBRARY IRDF-2002

Serial No.	Reaction name and reference for the new evaluations	C/E of integral data on Maxwellian thermal (300K), 1/E (Resonance Integral) and ^{252}Cf(sf) neutron fields		Remarks
		For the new evaluations	For IRDF-2002 data [18]	
1.	^{6}Li(n,t) [3,4,10]	Thermal: 0.998 ±0.44 % Res. Int.: **No exp. data***	Thermal: 1.002±0.44 % Res.Int.: **No exp. data***	**No evaluated exp. data for the Res. Int. up to 20 MeV**. The thermal cross sections are the same in the two evaluations.THE NEW EVALUATION IS ACCEPTED.
2.	^{10}B(n,α) [3,4,10]	Thermal: 1.001±0.40 % Res. Int.: 1.001±0.44 %	Thermal: 1.001±0.28 % Res. Int.: 1.002±0.33 %	THE TWO EVALUATIONS ARE THE SAME!
3.	^{24}Mg(n,p) [7]	1.055±3.02 %	1.082± 3.70 %	NEW EVALUATION IS BETTER!
4.	^{27}Al(n,p) [5]	0.974±3.18 %	1.007± 0.70 %	UNCERTAINTY IN IRDF-2002 WAS UNDERESTIMATED!
5.	^{27}Al(n,α) [6]	1.003±2.30 %	1.022±2.54 %	NEW EVALUATION IS BETTER!
6.	^{32}S(n,p) [7]	1.003±4.35 %	0.969±5.06 %	NEW EVALUATION IS BETTER!
7.	^{47}Ti(n,p) [8]	1.015±3.26 %	0.983±3.76 %	THE TWO EVAL. ARE THE SAME!
8.	^{55}Mn(n,γ) [3,9]	Thermal: 0.994±1.17 % Res. Int.: 1.009±5.34 %	Thermal: 1.003±4.19 % **Res. Int.: 0.878±14.87 %**	NEW EVALUATION IS BETTER!
9.	^{58}Fe(n,γ) [12]	Thermal: 0.996±5.55 % Res. Int.: 0.983±5.34 % ♣	Thermal: 0.985±**12.80 %** **Res. Int. 0.905±10.02%**	NEW EVALUATION IS BETTER!
10.	^{59}Co(n,2n) [8]	1.007±4.41 %	1.044±4.88 %	NEW EVALUATION IS BETTER!
11.	^{60}Ni(n,p) [7]	**1.173 ±5.89 %**	1.044±**11.6%**	**Raw exp.data** (based only on one measurement) **in ^{252}Cf(sf) neutron field !**
12.	^{63}Cu(n,2n) [7]	1.076±5.93 %	**1.115±7.00 %**	NEW EVALUATION IS BETTER!
13.	^{65}Cu(n,2n) [7]	0.994±4.14 %	1.030±4.10 %	NEW EVALUATION IS BETTER!
14.	^{64}Zn(n,p) [7]	1.012±2.96 %	1.037±5.21 %	NEW EVALUATION IS BETTER!

Serial No.	Reaction name and reference for the new evaluations	C/E of integral data on Maxwellian thermal (300K), 1/E (Resonance Integral) and ^{252}Cf(sf) neutron fields		Remarks
		For the new evaluations	For IRDF-2002 data [18]	
15.	^{89}Y(n,2n) [16]	**No exp. data for ^{252}Cf(sf) neutron field**	**No exp. data for ^{252}Cf(sf) neutron field**	**No characterization is possible !** The calculated ^{252}Cf(sf) averaged integral cross sections are the same in the two evaluations! (See Table 1 and [18]).
16.	^{90}Zr(n,2n) [6]	0.982±5.92 %	1.001±6.10 %	NEW EVALUATION. IS ACCEPTED, AS DOCUMENTED IN INDC(NDS)-0546 [6].
17.	^{93}Nb(n,2n)^{92m}Nb [16]	1.000±5.09 %	1.030±5.63 %	NEW EVALUATION IS BETTER!
18.	^{115}In(n,n') ^{115m}In [17]	0.966±2.18 %	0.972±2.16 %	THE TWO EVALUATIONS ARE THE SAME!
19.	^{115}In(n,2n)^{114m}In [7]	**No exp. data for ^{252}Cf(sf) neutron field**	**No exp. data for ^{252}Cf(sf) neutron field**	**No characterization is possible !** About 3 % diff. between the two calculated ^{252}Cf(sf) averaged integral cross sections.
20.	^{115}In(n,γ) ^{116m}In [18]$^{+}$	Thermal: 0.985±6.15 % Res. Int.: 0.958±7.08 %	Thermal: 1.029±6.01 % Res. Int.: 0.978±7.08 %	NEW EVALUATION IS PERHAPS BETTER!
21.	^{127}I(n,2n) [7]	1.018±4.53 %	1.062±4.24 %	NEW EVALUATION IS BETTER!
22.	^{169}Tm(n,2n) [16]	0.928±7 %	0.932±6.97 %	NEW EVALUATION IS BETTER!
23.	^{186}W(n,γ) [3,9]	Thermal: 1.000±6.06 % Res. Int.: 1.009±5.42 %	Thermal: 1.010±2.66 % Res. Int.: 0.994±4.56 %	NEW EVAL. ACCEPTED! (C/E is better, uncertainty is higher in this case.)
24.	^{197}Au(n,2n) [7]	1.005±3.30 %	1.044±4.98 %	NEW EVALUATION IS BETTER!
25.	^{197}Au(n,γ) [4,10,19]	Thermal: 1.000±0.49 % Res. Int.: 1.013±2.75 %	Thermal: 1.001±0.10 % Res. Int.: 1.008±1.82 %	THE TWO EVALUATIONS ARE ABOUT THE SAME!
26.	^{199}Hg(n.n')^{199m}Hg [7]	0.993±4.08 %	**0.833±8.04 %**	NEW EVALUATION IS BETTER!
27.	^{232}Th(n,f) [9,20,21]	1.009±4.36 %	**0.879±5.92 %**	NEW EVALUATION IS BETTER!
28.	^{232}Th(n,γ) [9,20]	Thermal: 0.998±1.30 % Res. Int.: 1.012±2.62 %	Thermal: 1.008±4.35 % Res. Int.: 1.032±**11.07%**	NEW EVALUATION IS BETTER!

Serial No.	Reaction name and reference for the new evaluations	C/E of integral data on Maxwellian thermal (300K), 1/E (Resonance Integral) and ^{252}Cf(sf) neutron fields		Remarks
		For the new evaluations	For IRDF-2002 data [18]	
29	^{235}U(n,f) [4,10,19,21]	Thermal: 1.004±0.40 % Res. Int.: 1.004±1.84 %	Thermal: 1.006±0.27 % Res. Int.: 1.004±1.84 %	THE TWO EVAL. ARE THE SAME!
30	^{237}Np(n,f) [22]	0.999±2.33 %	0.999±2.40 %	THE TWO EVALUATIONS ARE THE SAME!
31.	^{238}U(n,f) [4,10,19,21]	0.978±1.77 %	0.980±2.65 %	THE TWO EVALUATIONS ARE ABOUT THE SAME!
32.	^{238}U(n,γ) [4,10,19,21]	Thermal: 1.002±1.97 % Res. Int.: 0.995±1.81 %	Thermal: 1.015±1.22 % Res. Int.: 1.004±1.66 %	THE TWO EVALUATIONS ARE ABOUT THE SAME!
33.	^{239}Pu(n,f) [4,10,19,21]	Thermal: 1.000±1.11 % Res. Int.: 0.999±3.38 %	Thermal: 0.998±0.76 % Res. Int.: 0.999±5.31 %	THE TWO EVALUATIONS ARE ABOUT THE SAME!

REMARKS to TABLE 2.

1) The C/E values are based on the evaluated experimental data of S.F. Mughabghab [1] in the thermal and 1/E neutron energy regions, while in case of the threshold detectors on the evaluated experimental data in the ^{252}Cf spontaneous fission spectrum of W. Mannhart [2]. If the experimental data in the latter case are taken from other sources, it is indicated in Table 1.

2) The calculated values of the resonance integral (1/E energy region) in this Table refer to the neutron energy region from 0.5 eV to 20 MeV (to be consistent with S.F. Mughabghab experimental data [1]). It means that the resonance integrals for the IRDF-2002 cross section data have been recalculated in the above mentioned neutron energy region (the original energy interval considered in IRDF-2002 was 0.5 eV to 1.05 MeV [18]).

3) The C/E values without any remark refer to the ^{252}Cf spontaneous fission neutron spectrum.

\+ The data of the reaction ^{115}In (n,γ)^{116m}In are derived from IRDF-2002, only the branching ratio between the metastable and ground states was changed by IAEA NDS in the present evaluation. The covariance data are the same as in case of IRDF-2002.

* In case of the reaction ^{6}Li(n,t) no evaluated experimental data for the resonance integral are available up to 20 MeV.

♣ The C/E value based on the experimental data of the k_0 database of the neutron activation analysis [11] was accepted in this case.

Table 3. COMPARISON OF THE LIBRARY AND EXPERIMENTAL VALUES OF THE NEW CROSS SECTION EVALUATIONS AVERAGED OVER THE ENDF-B/VII.1 $^{235}U(n_{th},f)$ PROMPT FISSION NEUTRON SPECTRUM

Reaction name and evaluation	^{235}U fiss. spectrum		C/E
	Calc. libr. integral data (mb) [3]	Exp. data (mb)	
^{27}Al(n,p) [5]	3.956 ± 12.57 %	3.902 [24] ±1.77 %	1.014 ± 12.69 %
^{32}S(n,p) [7]	68.17 ± 6.91 %	69.080 [24] ±1.97 %	0.987 ± 7.18 %
47Ti(n,p) [8]	18.137 ± 6.51 %	17.84 [24] ± 1.99 %	1.017 ± 6.81 %
^{59}Co(n,p) [6]	1.416 ±13.57 %	1.396 [24] ±2.36%	1.014 ± 13.77 %
^{64}Zn(n,p) [7]	38.90 ± 7.16 %	35.39 [24] ±3.02 %	1.099 ± 7.77 %
^{115}In (n,n') ^{115m}In [17]	187.21 ±3.34 %	187.80 [24] ±1.23 %	0.997 ± 3.56 %
^{197}Au(n,γ) [4,10,19]	75.71 ±3.51 %	74.0 [14] ±4.0 %	1.023 ± 5.32 %
^{199}Hg(n.n')^{199m}Hg [7]	285.6 ± 5.27 %	278.0 [14] 5.30%	1.027 ± 7.47 %
^{232}Th(n,f) [9,20,21]	74.15 ± 4.64 %	74.5 [14] ±4.2 %	0.995 ± 6.25%
^{235}U(n,f) [4,10,19, 21]	1222.4 ± 0.43 %	1217.0 [24] ±1.12%	1.004 ± 1.181%
^{237}Np(n,f) [22]	1355.7 ±2.10 %	1350.0 [24] ±1.78 %	1.004 ± 2.75%
^{238}U(n,f) [4.10,19, 21]	309.20 ± 3.49 %	309.4 [24] ±1.13 %	0.999 ± 3.66%
^{239}Pu(n,f) [4,10,19, 21]	1795.4 ±0.51 %	1831.0 [24] ±1.70%	0.980 ± 1.78%

Remark to Table 3

The intercomparison was made for the new cross section evaluations having reliable experimental data in the literature [3]. Only reactions having the mean neutron energy E(50%) of their integrated response [14,24] below 6 MeV in the specified fission neutron field have been taken into consideration. The selection of reactions was done trying to keep the total uncertainty of calculated reaction rates for this neutron spectrum below 15%.

Table 4. BASIC DATA AND SOURCES OF THE CROSS SECTIONS IN THE NEW LIBRARY IRDFF release 1.0

Serial No. in IRDFF rel. 1.0	Reaction Code	Reaction	Mat	MF	MT	Source of the cross section data in IRDFF release 1.0	Status and origin of the cross section data
1.	Li6T	**^{6}Li(n,t)^{4}He**	325	3	105	**[3,4,10,]**	**R**
2.	B10A	**^{10}B(n,α)^{7}Li**	525	3	107	**[3,4,10]**	**R**
3.	F192	**^{19}F(n,2n)^{18}F**	925	3	16	**[18]**	**IRDF-2002**
4.	Na232	**^{23}Na(n,2n)^{22}Na**	1125	3	16	**[18]**	**IRDF-2002**
5.	Na23G	**^{23}Na(n,γ)^{24}Na**	1125	3	102	**[18]**	**IRDF-2002**
6.	Mg24P	**^{24}Mg(n,p)^{24}Na**	1225	3	103	**[7]**	**R**
7.	Al27P	**^{27}Al(n,p)^{27}Mg**	1325	3	103	**[5]**	**R**
8.	Al27A	**^{27}Al(n,α)^{24}Na**	1325	3	107	**[6]**	**R**
9.	P31P	**^{31}P(n,p)^{31}Si**	1525	3	103	**[18]**	**IRDF-2002**
10.	S32P	**^{32}S(n,p)^{32}P**	1625	3	103	**[7]**	**R**
11.	Sc45G	**^{45}Sc(n,γ)^{46}Sc**	2125	3	102	**[18]**	**IRDF-2002**
12.	Ti462	**^{46}Ti(n,2n)^{45}Ti**	2225	3	16	**[18]**	**IRDF-2002**
13.	Ti46P	**^{46}Ti(n,p)^{46}Sc**	2225	3	103	**[18]**	**IRDF-2002**
14.	Ti47NP	**^{47}Ti(n,x)^{46}Sc**	2228	10	5	**[18]**	**IRDF-2002**
15.	Ti47P	**^{47}Ti(n,p)^{47}Sc**	2228	3	103	**[8]**	**R**
16.	Ti48NP	**^{48}Ti(n,x)^{47}Sc**	2231	10	5	**[18]**	**IRDF-2002**
17.	Ti48P	**^{48}Ti(n,p)^{48}Sc**	2231	3	103	**[18]**	**IRDF-2002**
18.	Ti49NP	**^{49}Ti(n,x)^{48}Sc**	2234	10	5	**[18]**	**IRDF-2002**
19.	V51A	**^{51}V(n,α)^{48}Sc**	2328	3	107	**[18]**	**IRDF-2002**
20.	Cr522	**^{52}Cr(n,2n)^{51}Cr**	2431	3	16	**[18]**	**IRDF-2002**
21.	Mn55G	**^{55}Mn(n,γ)^{56}Mn**	2525	3	102	**[3,9]**	**R**
22.	Mn552	**^{55}Mn(n,2n)^{54}Mn**	2525	3	16	**[6]**	**N**
23.	Fe542	**^{54}Fe(n,2n)^{53}Fe**	2625	3	16	**[18]**	**IRDF-2002**
24.	Fe54P	**^{54}Fe(n,p)^{54}Mn**	2625	3	103	**[18]**	**IRDF-2002**
25.	Fe54A	**^{54}Fe(n,α)^{51}Cr**	2625	3	107	**[18]**	**IRDF-2002**
26.	Fe56P	**^{56}Fe(n,p)^{56}Mn**	2631	3	103	**[18]**	**IRDF-2002**
27.	Fe58G	**^{58}Fe(n,γ)^{59}Fe**	2637	3	102	**[12]**	**R**
28.	Co592	**^{59}Co(n,2n)^{58}Co**	2725	3	16	**[6]**	**R**
29.	Co593	**^{59}Co(n,3n)^{57}Co**	2725	3	17	**[16]**	**N**
30.	Co59G	**^{59}Co(n,γ)^{60}Co**	2725	3	102	**[18]**	**IRDF-2002**
31.	Co59P	**^{59}Co(n,p)^{59}Fe**	2725	3	103	**[6]**	**N**
32.	Co59A	**^{59}Co(n,α)^{56}Mn**	2725	3	107	**[18]**	**IRDF-2002**
33.	Ni582	**^{58}Ni(n,2n)^{57}Ni**	2825	3	16	**[18]**	**IRDF-2002**

Serial No. in IRDFF rel. 1.0	Reaction Code	Reaction	Mat	MF	MT	Source of the cross section data in IRDFF release 1.0	Status and origin of the cross section data
34.	Ni58P	**^{58}Ni(n,p)^{58}Co**	2825	3	103	**[18]**	**IRDF-2002**
35.	Ni60P	**^{60}Ni(n,p)^{60}Co**	2831	3	103	**[7]**	**R**
36.	Cu632	**^{63}Cu(n,2n)^{62}Cu**	2925	3	16	**[7]**	**R**
37.	Cu63G	**^{63}Cu(n,γ)^{64}Cu**	2925	3	102	**[18]**	**IRDF-2002**
38.	Cu63A	**^{63}Cu(n,α)^{60}Co**	2925	3	107	**[18]**	**IRDF-2002**
39.	Cu652	**^{65}Cu(n,2n)^{64}Cu**	2931	3	16	**[7]**	**R**
40.	Zn64P	**^{64}Zn(n,p)^{64}Cu**	3025	3	103	**[7]**	**R**
41.	Zn67P	**^{67}Zn(n,p)^{67}Cu**	3034	3	103	**[15]**	**N**
42.	As752	**^{75}As(n,2n)^{74}As**	3325	3	16	**[18]**	**IRDF-2002**
43.	Y892	**^{89}Y(n,2n)^{88}Y**	3925	3	16	**[16]**	**R**
44.	Zr902	**^{90}Zr(n,2n)^{89}Zr**	4025	3	16	**[6]**	**R**
45.	Mo92P	**^{92}Mo(n,p) ^{92m}Nb**	4225	10	103	**[15]**	**N**
46.	Nb932	**^{93}Nb(n,2n)^{92m}Nb**	4125	10	16	**[16]**	**R**
47.	Nb93N	**^{93}Nb(n,n')^{93m}Nb**	4125	10	4	**[18]**	**IRDF-2002**
48.	Nb93G	**^{93}Nb(n,γ)^{94}Nb**	4125	3	102	**[18]**	**IRDF-2002**
49.	Rh103N	**^{103}Rh(n,n')^{103m}Rh**	4525	10	4	**[18]**	**IRDF-2002**
50.	Ag109G	**^{109}Ag(n,γ)^{110m}Ag**	4731	10	102	**[18]**	**IRDF-2002**
51.	In113N	**^{113}In(n,n')^{113m}In**	4925	10	4	**[15]**	**N**
52.	In1152	**^{115}In(n,2n)^{114m}In**	4931	10	16	**[17]**	**R**
53.	In115N	**^{115}In(n,n')^{115m}In**	4931	10	4	**[17]**	**R**
54.	In115G	**^{115}In(n,γ)^{116m}In**	4931	10	102	**[3]**	**R**
55.	I1272	**^{127}I(n,2n)^{126}I**	5325	3	16	**[17]**	**R**
56.	La139G	**^{139}La(n,γ)^{140}La**	5728	3	102	**[18]**	**IRDF-2002**
57.	Pr1412	**^{141}Pr(n,2n)^{140}Pr**	5925	3	16	**[18]**	**IRDF-2002**
58.	Tm1692	**^{169}Tm(n,2n)^{168}Tm**	6925	3	16	**[16]**	**R**
59.	Tm1693	**^{169}Tm(n,3n)^{167}Tm**	6925	3	17	**[15]**	**N**
60.	Ta181G	**^{181}Ta(n,γ)^{182}Ta**	7328	3	102	**[18]**	**IRDF-2002**
61.	W186G	**^{186}W(n,γ)^{187}W**	7443	3	102	**[3,9]**	**R**
62.	Au1972	**^{197}Au(n,2n)^{196}Au**	7925	3	16	**[7]**	**R**
63.	Au197G	**^{197}Au(n,γ)^{198}Au**	7925	3	102	**[4,10,19]**	**R**
64.	Hg199N	**^{199}Hg(n,n')^{199m}Hg**	8034	10	4	**[7]**	**R**
65.	Pb204N	**^{204}Pb(n,n')^{204m}Pb**	8225	10	4	**[18]**	**IRDF-2002**
66.	Bi2093	**^{209}Bi(n,3n)^{207}Bi**	8325	3	17	**[16]**	**N**
67.	Th232F	**^{232}Th(n,f)FP**	9040	3	18	**[9,20,21]**	**R**
68.	Th232G	**^{232}Th(n,γ)^{233}Th**	9040	3	102	**[9,20]**	**R**
69.	U235F	**^{235}U(n,f)FP**	9228	3	18	**[4,10,19,21]**	**R**
70.	U238F	**^{238}U(n,f)FP**	9237	3	18	**[4,10,19,21]**	**R**

Serial No. in IRDFF release 1.0	Reaction Code	Reaction	Mat	MF	MT	Source of the cross section data in IRDFF release 1.0	Status and origin of the cross section data
71.	U238G	**^{238}U(n,γ)^{239}U**	9237	3	102	**[4,10,19,21]**	**R**
72.	Np237F	**^{237}Np(n,f)FP**	9346	3	18	**[22]**	**R**
73.	Pu39F	**^{239}Pu(n,f)FP**	9437	3	18	**[4,10,19,21]**	**R**
74.	Am241F	**^{241}Am(n,f)FP**	9543	3	18	**[18]**	**IRDF-2002**
Cover materials without uncertainties							
Serial No.	Reaction code	Name of cover material	Mat	MF	MT	Source of the cross section data in IRDFF release 1.0	Status and origin of the cross section data
1	---	**B-COVER**	500	3	001	**[3]**	**R**
2	---	**CD-COVER**	4800	3	001	**[20,23]**	**R**
3	---	**GD-COVER**	6400	3	001	**[3]**	**R**

Explanations to Table 4

R =replacement of IRDF-2002 data by new evaluations.
N = new evaluation, not present in IRDF-2002.

Table 5. CROSS SECTIONS IN THE FILE IRDFF release 1.0 AND THEIR CHARACTERISTICS (NEUTRON TEMPERATURE 300 K)

Serial No. in IRDFF release 1.0	Reaction name	Selected evaluation	Calculated library cross section Maxwellian (300K) σ_L (barn) and uncertainty	Resonance integral from library data IR_L (barn) and uncertainty	Calculated average library cross section in ^{252}Cf spontaneous fission $<\sigma_{Cf}>$ (mb) and uncertainty	C/E
1	$^{6}Li(n,t)^{4}He$	ENDF/B-VII.1 [3,4,10]	938.47±0.14 %	423.93±0.13 %	321.3 ± 0.86%	Thermal: 0.998±0.44 % Res.Int.: **No exp. data** **No exp. data in $^{252}Cf(sf)$ field**
2	$^{10}B(n,\alpha)^{7}Li$	ENDF/B-VII.1 [3,4,10]	3842.6±0.33 %	1724.5±0.33 %	446.3 ± 0.56%	Thermal: 1.001±0.44 % Res.Int.: 1.001±0.44 % **No exp. data in $^{252}Cf(sf)$ field**
3	$^{19}F(n,2n)^{18}F$	IRDF-2002 [18]	---	---	1.627E-2 ± 2.92 %	1.009±6.34 %
4	$^{23}Na(n,2n)^{22}Na$	IRDF-2002 [18]	---	---	8.611E-3 ± 3.90 %	**No exp. data in $^{252}Cf(sf)$ field**
5	$^{23}Na(n,\gamma)^{24}Na$	IRDF-2002 [18]	0.528±2.00 %	0.317±3.00 %	---	Thermal: 1.022±2.15 % Res.Int.: 1.018±4.39 %
6	$^{24}Mg(n,p)^{24}Na$	ZOLOTAREV [7]	---	---	2.105 ± 1.78 %	1.055±3.02 %
7	$^{27}Al(n,p)^{27}Mg$	ZOLOTAREV [5]	---	---	4.751 ± 2.35 %	0.974±3.18%
8	$^{27}Al(n,\alpha)^{24}Na$	ZOLOTAREV [6]	---	---	1.019 ± 1.77 %	1.003±2.30 %
9	$^{31}P(n,p)^{31}Si$	IRDF-2002 [18]	---	---	30.68 ± 3.58 %	**No exp. data in $^{252}Cf(sf)$ field**
10	$^{32}S(n,p)^{32}P$	ZOLOTAREV [7]	---	---	74.11 ± 2.59 %	1.003±4.35%
11	$^{45}Sc(n,\gamma)^{46}Sc$	IRDF-2002 [18]	27.208±6.86 %	11.89±7.61 %	---	Thermal: 1.000±6.90 % Res.Int.: 0.982±8.66 %
12	$^{46}Ti(n,2n)^{45}Ti$	IRDF-2002 [18]	---	---	1.218E-2 ± 4.41 %	**No exp. data in $^{252}Cf(sf)$ field**
13	$^{46}Ti(n,p)^{46}Sc$	IRDF-2002 [18]	---	---	13.83 ± 3.05 %	0.983±3.76 %
14	$^{47}Ti(n,x)^{46}Sc$	IRDF-2002 [18]	---	---	1.941E-2 ± 7.57 %	**No exp. data in $^{252}Cf(sf)$ field**
15	$^{47}Ti(n,p)^{47}Sc$	ZOLOTAREV [8]	---	---	19.56 ± 2.80 %	1.015±3.26 %
16	$^{48}Ti(n,x)^{47}Sc$	IRDF-2002 [18]	---	---	4.349E-3 ± 8.2 %	**No exp. data in $^{252}Cf(sf)$ field**
17	$^{48}Ti(n,p)^{48}Sc$	IRDF-2002 [18]	---	---	0.268 ± 5.08 %	1.005±5.67 %

Serial No. in IRDFF release 1.0	Reaction name	Selected evaluation	Calculated library cross section Maxwellian (300K) σ_L (barn) and uncertainty	Resonance integral from library data IR_L (barn) and uncertainty	Calculated average library cross section in ^{252}Cf spontaneous fission <σ_c> (mb) and uncertainty	C/E
18	^{49}Ti(n,x)^{48}Sc	IRDF-2002 [18]	---	---	2.644E-3 ± 7.18 %	**No exp. data in ^{252}Cf(sf) field**
19	^{51}V(n,α)^{48}Sc	IRDF-2002 [18]	---	---	3.859E-2 ± 3.02 %	0.989±4.15 %
20	^{52}Cr(n,2n)^{51}Cr	IRDF-2002 [18]	---	---	9.703E-2 ± 2.72 %	**No exp. data in ^{252}Cf(sf) field**
21	^{55}Mn(n,γ)^{56}Mn	ENDF/B-VII.1 [3] TRKOV *et al.* [9]	13.278±1.11 %	13.526±3.82 %	---	Thermal: 0.991±1.21 % Res.Int.: 1.009±5.34 %
22	^{55}Mn(n,2n)^{54}Mn	ZOLOTAREV [6]	---	---	0.416 ± 3.75 %	1.022±4.42 %
23	^{54}Fe(n,2n)^{53}Fe	IRDF-2002 [18]	---	---	3.498E-3 ± 4.87 %	**No exp. data in ^{252}Cf(sf) field**
24	^{54}Fe(n,p)^{54}Mn	IRDF-2002 [18]	---	---	88.16 ± 2.09 %	1.015±2.56 %
25	^{54}Fe(n,α)^{51}Cr	IRDF-2002 [18]	---	---	1.113 ± 3.18 %	**No exp. data in ^{252}Cf(sf) field**
26	^{56}Fe(n,p)^{56}Mn	IRDF-2002 [18]	---	---	1.475 ± 2.61 %	1.007±3.48 %
27	^{58}Fe(n,γ)^{59}Fe	JEFF3.1 [12]	1.315±5.07 %	1.275±4.93 % 1.246± 4.93 % (*)	---	Thermal: 0.996±5.55 % **Res.Int.: 0.850±6.79%** Res.Int.: 0.983±5.69% (*)
28	^{59}Co(n,2n)^{58}Co	ZOLOTAREV [6]	---	---	0.408 ± 3.63 %	1.007±4.41 %
29	^{59}Co(n,3n)^{57}Co	ZOLOTAREV [16]	---	---	---	**High energy dosimetry!**
30	^{59}Co(n,γ)^{60}Co	IRDF-2002 [18]	37.176 ±0.65 %	75.801±0.77 %	---	Thermal: 1.000±0.70 % Res.Int.: 1.024±2.81 %
31	^{59}Co(n,p)^{59}Fe	ZOLOTAREV [6]	---	---	1.715 ± 3.65 %	1.015±4.41 %
32	^{59}Co(n,α)^{56}Mn	IRDF-2002 [18]	---	---	0.221 ± 3.54 %	0.997±4.31 %
33	^{58}Ni(n,2n)^{57}Ni	IRDF-2002 [18]	---	---	9.256E-3 ± 2.72 %	1.034±7.54 %
34	^{58}Ni(n,p)^{58}Co	IRDF-2002 [18]	---	---	117.5 ± 1.74 %	1.000±2.30 %
35	^{60}Ni(n,p)^{60}Co	ZOLOTAREV [7]	---	---	2.803 ± 2.27 %	**1.173 ±5.89 %** **Raw exp. data in ^{252}Cf(sf)**
36	^{63}Cu(n,2n)^{62}Cu	ZOLOTAREV [7]	---	---	0.654 ± 3.50 %	0.994±4.14 %

Serial No. in IRDFF release 1.0	Reaction name	Selected evaluation	Calculated library cross section Maxwellian (300K) σ_L (barn) and uncertainty	Resonance integral from library data IR_L (barn) and uncertainty	Calculated average library cross section in ^{252}Cf spontaneous fission $\langle\sigma_c\rangle$ (mb) and uncertainty	C/E
37	$^{63}Cu(n,\gamma)^{64}Cu$	IRDF-2002 [18]	4.470±4.14 %	4.988±4.23 %	---	Thermal: 0.993±4.13 % Res.Int.: 1.004±4.53 %
38	$^{63}Cu(n,\alpha)^{60}Co$	IRDF-2002 [18]	---	---	0.6933 ± 2.83 %	1.007±3.67 %
39	$^{65}Cu(n,2n)^{64}Cu$	ZOLOTAREV [7]	---	---	0.654 ± 3.50 %	0.994±4.14 %
40	$^{64}Zn(n,p)^{64}Cu$	ZOLOTAREV [7]	---	---	42.72 ± 1.86 %	1.012±2.96 %
41	$^{67}Zn(n,p)^{67}Cu$	ZOLOTAREV [15]	---	---	1.106 ± 5.31 %	**No exp. data in ^{252}Cf(sf) field**
42	$^{75}As(n,2n)^{74}As$	IRDF-2002 [18]	---	---	0.6209 ± 5.76 %	**No exp. data in ^{252}Cf(sf) field**
43	$^{89}Y(n,2n)^{88}Y$	ZOLOTAREV [16]	---	---	0.345 ± 4.42 %	**No exp. data in ^{252}Cf(sf) field**
44	$^{90}Zr(n,2n)^{89}Zr$	ZOLOTAREV [6]	---	---	0.217 ± 5.17 %	0.982±5.92 %
45	$^{92}Mo(n,p)\ ^{92m}Nb$	ZOLOTAREV [15]	---	---	7.835 ± 3.75 %	**0.516±5.78 %** **Raw exp. data in ^{252}Cf(sf) neutron field**
46	$^{93}Nb(n,2n)^{92m}Nb$	ZOLOTAREV [16]	---	---	0.791 ± 2.38 %	1.000±5.09 %
47	$^{93}Nb(n,n')^{93m}Nb$	IRDF-2002 [18]	---	---	146.1 ± 2.59 %	1.000±4.30 %
48	$^{93}Nb(n,\gamma)^{94}Nb$	IRDF-2002 [18]	1.156±10.00 %	9.924±9.83 %	---	Thermal: 1.005±10.94 % **Res.Int.: 1.196 ±10.95 %**
49	$^{103}Rh(n,n')^{103}Rh$	IRDF-2002 [18]	---	---	725.1 ± 3.94 %	**0.896±4.91 %**
50	$^{109}Ag(n,\gamma)^{110m}Ag$	IRDF-2002 [18]	4.211±5.10 %	68.483±6.96 %	---	Thermal: 0.954±5.24 % **Res.Int.: 1.052±8.26 %**
51	$^{113}In(n,n')^{113m}In$	ZOLOTAREV [15]	---	---	158.1 ± 1.24 %	0.981±2.39 %
52	$^{115}In(n,2n)^{114m}In$	ZOLOTAREV [7]	---	---	1.633 ± 5.51 %	**No exp. data in ^{252}Cf(sf) field**
53	$^{115}In(n,n')^{115m}In$	ZOLOTAREV [17]	---	---	190.64 ± 1.70 %	0.966±2.18 %
54	$^{115}In(n,\gamma)^{116m}In$	IRDF-2002 [18]	159.8±6.00 %	2538.1±5.99 %	---	Thermal: 0.984±6.01 % Res.Int.: 0.958±7.08 %

Serial No. in IRDFF release 1.0	Reaction name	Selected evaluation	Calculated library cross section Maxwellian (300K) σ_L (barn) and uncertainty	Resonance integral from library data IRL (barn) and uncertainty	Calculated spectrum average cross section in ^{252}Cf spontaneous fission $<\sigma_C>$ (mb) and uncertainty [%]	C/E
55	$^{127}I(n,2n)^{126}I$	ZOLOTAREV [7]	---	---	2.107 ± 3.83 %	1.018±4.53 %
56	$^{139}La(n,\gamma)^{140}La$	IRDF-2002 [18]	9.041±3.85 %	12.109±5.54 %	---	Thermal: 1.000±3.90 % Res.Int.: 1.001±7.44 %
57	$^{141}Pr(n,2n)^{140}Pr$	IRDF-2002 [18]	---	---	1.990 ± 11.03 %	**No exp. data in ^{252}Cf(sf) field**
58	$^{169}Tm(n,2n)^{168}Tm$	ZOLOTAREV [16]	---	---	6.268 ± 3.1 %	0.982±7 %
59	$^{169}Tm(n,3n)^{167}Tm$	ZOLOTAREV [15]	---	---	---	**High energy dosimetry!**
60	$^{181}Ta(n,\gamma)^{182}Ta$	IRDF-2002 [18]	20.679±3.00 %	660.01±3.79 %	---	Thermal: 0.993±4.13 % Res.Int.: 1.008±4.86 %
61	$^{186}W(n,\gamma)^{187}W$	ENDF/B-VII.1 [3] + [9]	38.095±5.92 %	484.20±4.43 %	---	Thermal: 1.000±6.10 % Res.Int.: 1.009±5.42 %
62	$^{197}Au(n,2n)^{196}Au$	ZOLOTAREV [7]	---	---	5.531 ± 2.75 %	1.005±3.30 %
63	$^{197}Au(n,\gamma)^{198}Au$	INT. CROSS SEC. STANDARDS [4,10,19]	98.70±0.48 %	1573.5±2.07 %	---	Thermal: 1.005±0.50 % Res.Int.: 1.015±2.75 %
64	$^{199}Hg(n,n')^{199m}Hg$	ZOLOTAREV [7]	---	---	296.3 ± 3.66 %	0.993±4.08 %
65	$^{204}Pb(n,n')^{204m}Pb$	IRDF-2002 [18]	---	---	20.39 ± 4.57 %	0.978±6.44 %
66	$^{209}Bi(n,3n)^{207}Bi$	ZOLOTAREV [16]	---	---	---	**High energy dosimetry!**
67	$^{232}Th(n,f)FP$	ENDF/B-VII.0 [20] + [9] AND [21]	---	---	77.56 ± 2.14 %	1.009±4.36 %
68	$^{232}Th(n,\gamma)^{233}Th$	ENDF/B-VII.0 [20] + [9]	7.338±1.24 %	84.325±1.90 %	---	Thermal: 0.998±1.30 % Res.Int.: 1.012±2.62 %
69	$^{235}U(n,f)FP$	INT. CROSS SEC. STANDARDS [4,10,19,21]	585.1±0.35 %	276.0±0.29 %	1224.8 ± 0.42 %	Thermal: 1.004±0.40 % Res.Int.: 1.004±1.84 %
70	$^{238}U(n,f)FP$	INT. CROSS SEC. STANDARDS [4,10,19,21]	---	---	318.5 ± 0.66 %	0.978±1.77 % **Fast neutron dosimetry!**

Serial No. in IRDFF release 1.0	Reaction name	Selected evaluation	Calculated library cross section Maxwellian (300K) σ_L (barn) and uncertainty	Resonance integral from library data IR_L (barn) and uncertainty	Calculated average library cross section in ^{252}Cf spontaneous fission $<\sigma_c>$ (mb) and uncertainty	C/E
71	$^{238}U(n,\gamma)^{239}U$	INT. CROSS SEC. STANDARDS [4,10,19,21]	2.686±1.84 %	275.59±1.45 %	---	Thermal: 1.002±1.97 % Res.Int.: 0.995±1.81 %
72	$^{237}Np(n,f)FP$	ZOLOTAREV [22]	0.021±10.00 %	6.929±3.81 %	1360 ± 1.71 %	Thermal: 1.050±11.18 % **Res.Int.: 1.474±5.72 %** 0.999±2.33 % **Fast neutron dosimetry!**
73	$^{239}Pu(n,f)FP$	INT. CROSS SEC. STANDARDS [4,10,19,21]	748.20±1.08 %	302.81±0.72 %	1796 ± 0.46 %	Thermal: 1.000±1.11 % Res.Int.: 0.999±3.38 % 0.991±1.44 %
74	$^{241}Am(n,f)FP$	IRDF-2002 [18]	3.018±2.00 %	13.86±1.74 %	1397.2 ± 2.85%	Thermal: 0.943±3.45 % Res.Int.: 0.962±7.15 % **No exp. data in ^{252}Cf(sf) field** **Fast neutron dosimetry!**

COVER MATERIALS

Serial No. in IRDFF release 1.0	Reaction name	Selected evaluation	Calculated library cross section Maxwellian (300K) σ_L (barn) and uncertainty	Resonance integral from library data IR_L (barn) and uncertainty	C/E
1	B – Absorber	ENDF/B-VII.1 [3]	765.06 **No uncertainty!**	343.2 **No uncertainty!**	Thermal: 0.997 RI: No exp. data **No uncertainty!**
2	Cd – Absorber	ENDF/B-VII.0 [3] + [23]	2487.5 **No uncertainty!**	66.878 **No uncertainty!**	Thermal: 0.987 RI: 0.955 **No uncertainty!**
3	Gd – Absorber	ENDF/B-VII.1 [3]	45945 **No uncertainty!**	389.11 **No uncertainty!**	Thermal: 0.940 RI: 0.998 **No uncertainty!**

REMARKS to Table 5

The calculated thermal neutron cross section values in this Table refer to 300 K.

The calculated and experimental values of the resonance integral (1/E region) in this Table refer to the neutron energy region from 0.5 eV to 20 MeV (to be consistent with S.F. Mughabghab [1]), except when marked (*). For the latter case the limits of 0.55eV-2MeV are used to be consistent with KAYZERO database [11]

The capture (and some other thermal neutron induced) reactions are not characterized in the ^{252}Cf spontaneous fission neutron spectrum, as the important part of their cross sections for dosimetry is in the low neutron energy region.

The C/E data without any remark refer to the ^{252}Cf spontaneous fission neutron field. The evaluated experimental cross section values in the ^{252}Cf spontaneous fission spectrum are taken from W. Mannhart [2], otherwise the Reference is given in Table 1.

Table 6. CHARACTERIZATION OF THE CROSS SECTION AND RESONANCE INTEGRAL DATA FOR THE CAPTURE AND THERMAL FISSION REACTIONS IN THE FILE IRDFF release 1.0

Serial No.	Reaction name and evaluation	Maxwellian thermal (300K)		1/E (Resonance Integral)		C/E
		Calc. libr. integr. data (barn)	Exp. Data [1] (barn)	Calc. libr. integr.data (barn)	Exp. data [1] (barn)	
1.	$^{6}Li(n,t)$ [3,4,10]	938.47 ± 0.14 %	940.00 ± 0.42 %	423.93 ± 0.13 %	**No data***	Thermal: 0.998±0.44 % Res. Int.: **No data***
2.	$^{10}B(n,\alpha)$ [3,4,10]	3842.6 ± 0.33 %	3837.00 ± 0.23 %	1724.5 ± 0.33 %	1722.0 ± 0.29 %	Thermal: 1.001±0.44 % Res. Int.: 1.001±0.44 %
3.	$^{23}Na(n,\gamma)$ [18]	0.528 ± 2.00 %	0.517 ± 0.77 %	0.317 ± 3.00 %	0.311 ± 3.21 %	Thermal: 1.022±2.15 % Res. Int.: 1.018±4.39 %
4.	$^{45}Sc(n,\gamma)$ [18] **	27.208 ± 6.86 %	27.20 ± 0.74 %	11.888 ± 7.61 %	12.10 ± 4.13 %	Thermal: 1.000±6.90 % Res. Int.: 0.982±8.66 %
5.	$^{55}Mn(n,\gamma)$ [3,9]	13.278 ± 1.11 %	13.40 ± 0.37 %	13.526 ± 3.82 %	13.40 ± 3.73 %	Thermal: 0.991±1.21 % Res. Int.: 1.009±5.34 %
6.	$^{58}Fe(n,\gamma)$ [12]	1.315 ± 5.07 %	1.32 ± 2.27 % 1.300 ± 2.66 % ♦	1.275 ± 4.93 % 1.246 ♦♦ ± 4.93 %	1.50 ± 4.67 % 1.267 ± 2.84 % ♦	Thermal: 0.996± 5.55% **Res. Int.: 0.850±6.79 %** Thermal♦: 1.011±5.73 % Res. Int.♦: 0.983±5.69 %
7.	$^{59}Co(n,\gamma)$ [18]	37.176 ± 0.65 %	37.18 ± 0.16 %	75.801 ± 0.77 %	74.00 ± 2.70 %	Thermal: 1.000±0.70 % Res. Int.: 1.024±2.81 %
8.	$^{63}Cu(n,\gamma)$ [18]	4.470 ± 4.14 %	4.50 ± 0.44 %	4.988 ± 4.23 %	4.97 ± 1.61 %	Thermal: 0.993±4.13 % Res. Int.: 1.004±4.53 %
9.	$^{93}Nb(n,\gamma)$ [18]	1.156 ± 10.00 %	1.15 ± 4.35 %	9.924 ± 9.83 %	8.30 ± 4.82 %	Thermal: 1.005±10.94 % **Res. Int.:1.196±10.95 %**
10	$^{109}Ag(n,\gamma)^{110m}Ag$ [18]	4.211 ± 5.10 %	3.95 ± 1.27 %	68.483 ± 6.96 %	65.10 ± 4.45 %	Thermal: 0.954±5.24 % Res. Int.: 1.052±8.26 %

Serial No.	Reaction name and evaluation	Maxwellian thermal (300K)		1/E (Resonance Integral)		C/E
		Calc. libr. integr. data (barn)	Exp. Data [1] (barn)	Calc. libr. integr.data (barn)	Exp. data [1] (barn)	
11	^{115}In (n,γ) ^{116m}In [18] $^{+}$	159.8 ± 6.00 %	162.30 ± 0.43 %	2538.10 ± 5.99 %	2650.00 ± 3.77 %	Thermal: 0.984±6.01 % Res. Int.: 0.958±7.08 %
12	^{139}La(n,γ) [18]	9.041 ± 3.85 %	9.04 ± 0.44 %	12.109 ± 5.54 %	12.10 ± 4.96 %	Thermal: 1.00±3.90 % Res. Int.: 1.001±7.44 %
13	^{181}Ta(n,γ) [18]	20.679 ± 3.00 %	20.50 ± 2.44 %	660.01 ± 3.79 %	655.00 ± 3.05 %	Thermal: 0.993±4.13 % Res. Int.: 1.008±4.86 %
14.	^{186}W(n,γ) [3,9]	38.095 ± 5.92 %	38.10 ± 1.31 %	484.20 ± 4.43 %	480.00 ± 3.12 %	Thermal: 1.00±6.10 % Res. Int.: 1.009±5.42 %
15.	^{197}Au(n,γ) [4,10,19]	98.70 ± 0.48 %	98.65 ± 0.09 %	1573.50 ± 2.07 %	1550.00 ± 1.81 %	Thermal: 1.005±0.50 % Res. Int.: 1.015±2.75 %
16.	^{232}Th(n,γ) [9,20]	7.338 ± 1.24 %	7.35 ± 0.40 %	84.325 ± 1.90 %	83.30 ± 1.80 %	Thermal: 0.998±1.30 % Res. Int.: 1.012±2.62 %
17.	^{235}U(n,f) [4,10,19,21]	585.10 ± 0.35 %	582.60 ±1.89 %	276.02 ± 0.29 %	275.00 ± 1.82 %	Thermal: 1.004±0.40 % Res. Int.: 1.004±1.84 %
18.	^{238}U(n,γ) [4,10,19,21]	2.686 ± 1.84 %	2.68 ± 0.71 %	275.59 ± 1.45 %	277.00 ± 1.08 %	Thermal: 1.002±2.00 % Res. Int.: 0.995±1.81 %
19.	^{237}Np(n,f) [22]	0.021 ± 10.00 %	0.02 ± 5.00 %	6.929 ± 3.81 %	4.70 ± 4.89 %	Thermal: 1.050± 10.01% **Res. Int.: 1.474±6.20 %** **Fast neutron dosimetry!**
20.	^{239}Pu(n,f) [4,10,19,21]	748.20 ± 1.08 %	748.10 ± 0.27 %	302.81 ± 0.72 %	303.00 ± 3.30 %	Thermal: 1.000±1.11 % Res. Int.: 0.999±3.38 %
21.	^{241}Am(n,f) [18]	3.018 ± 2.00 %	3.20 ± 2.81 %	13.86 ± 1.74 %	14.40 ± 6.94 %	Thermal: 0.943±3.45 % Res. Int.: 0.962±7.15 % **Fast neutron dosimetry!**

Remarks

In the present Table the resonance integral data refer to the neutron energy region from 0.5 eV to 20 MeV (except the case flagged ♦♦ below)

***** In case of the reaction ^{6}Li(n,t) for the resonance integral no evaluated experimental data are available up to 20 MeV.

♦ Experimental data taken from KAYZERO database (see Ref.[11]).

♦♦ The resonance integral was calculated from 0.55eV to 2.0 MeV to be comparable with the experimental value used in [11].

****** For the reaction ^{45}Sc(n,γ) the cross section data agree with the ones of IRDF-2002, but the uncertainty data (a part of the covariance information) were modified by IAEA NDS.

$^{+}$ The data of the reaction ^{115}In(n,γ) are derived from IRDF-2002, only the branching ratio between the metastable and ground states was changed in the present evaluation by IAEA NDS. The covariance data are the same as in case of IRDF-2002

REFERENCES TO THE TABLES

[1] S.F. Mughabghab: Atlas of Neutron Resonances, Elsevier, ISBN-10:0-444-52035-X. Amsterdam-Tokio (2006).

[2] W. Mannhart: ”Response of Activation Reactions in the Neutron Field of Californium-252 Spontaneous Fission” In: International Reactor Dosimetry File 2002 (IRDF-2002) International Atomic Energy Agency, Technical Reports Series no. **452**,Vienna, 2006

[3] M. Chadwick, P. Oblozinsky, M. Herman *et al.*,"ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data", *Nucl. Data Sheets* **112** (2011) 2887–2996.

[4] S.A. Badikov, Chen Zhenpeng, A.D. Carlson, E.V. Gai, G.M. Hale, F.-J. Hambsch, H.M. Hofmann, T. Kawano, N.M. Larson, V.G. Pronyaev, D.L. Smith, Soo-Youl Oh, S. Tagesen and H. Vonach, International Evaluation of Neutron Cross-Section Standards, IAEA Technical Report **STI/PUB/1291** (IAEA, Vienna, December 2007).

[5] K. I. Zolotarev, P.K. Zolotarev, and J. Csikai, J., “Al-27 Russian Reactor Dosimetry File RRDF-2009, Revision 3, MAT 1325, MF 3,33, MT 103,” adopted from material 1325 BROND-3 Al-27 Evaluation, October 2009.

[6] K.I. Zolotarev: Evaluation of Cross-Section Data from Threshold to 40-60 MeV for Specific Neutron Reactions Important for Neutron Dosimetry Applications. Technical report **INDC(NDS)-0546** (IAEA, Vienna, April 2009).

[7] K.I. Zolotarev: Re-evaluation of Microscopic and Integral Cross Section Data for Important Dosimetry Reactions. Technical report **INDC(NDS)-0526** (IAEA, Vienna, August 2008).

[8] K.I. Zolotarev, “Ti-47 Russian Reactor Dosimetry File RRDF-2002, Rev.2, MAT 2228, MF 3,33, MT 103,” October 2009.

[9] A. Trkov, R. Capote, E.Sh. Soukhovitskii, L.C. Leal, M. Sin, I. Kodeli, and D.W. Muir, "Covariances of evaluated nuclear cross-section data for ^{232}Th, ^{180,182,183,184,186}W and ^{55}Mn", *Nucl. Data Sheets* **112** (2011) 3098-3119

[10] A. D. Carlson, V. G. Pronyaev, D. L. Smith, N. M. Larson, Z.-P. Chen, G. M. Hale, F. J. Hambsch, E. V.Gai, S.-Y. Oh, S. A. Badikov, T. Kawano, H. M. Hofmann, H. Vonach, and S. Tagesen, “International Evaluation of Neutron Cross Section Standards,” *Nucl. Data Sheets* **110** (2009) 3215.

[11] I. Kodeli and A. Trkov, Validation of the IRDF-2002 dosimetry library. *Nucl. Instr. and Meth. In Physics Research* **A577** (2007) 664–681.

[12] A.J. Koning, M.C. Duijvestijn, M.C. Moxon, and A.Trkov, “Fe-58 Evaluation, JEFF3.1 library, MAT2637,” retrieved from http://www-nds.iaea.org (Accessed May 2011).

[13] K. Shibata et al., “JENDL-4.0: A New Library for Nuclear Science and Engineering,” *J. Nucl. Sci. Technol.* **48** (2011) 1.

[14] R. Capote, K.I. Zolotarev, V.G. Pronyaev and A.Trkov, Updating and Extending the IRDF-2002 Dosimetry Library. Presented at the 14th International Symposium on Reactor Dosimetry, held on May 22-27, 2010, Bretton Woods, New Hampshire, USA; to be published in *Journal of ASTM International* **9** (2012).

[15] K.I. Zolotarev: Evaluation of selected (n,n’), (n,γ), (n,p), (n,2n) and (n,3n) excitation functions for fission and fusion reactor dosimetry. Progress Report on IAEA Research Contract No 16242 (unpublished).

[16] K.I. Zolotarev: Evaluation of Cross-Section Data from Threshold to 40-60 MeV for Some Neutron Reactions Important for Fusion Dosimetry Applications. Technical report **INDC(NDS)-0584** (IAEA, Vienna, November 2010).

[17] K.I. Zolotarev and P.K. Zolotarev, “In-115 Russian Reactor Dosimetry File RRDF-2010, MAT 4931, MF 3,33, MT 51,” adopted from Material 4931 BROND-3 In-115 Evaluation, March 2009.

[18] O. Bersillon, L. R. Greenwood, P. J. Griffin, W. Mannhart, H. J. Nolthenius, R. Paviotti-Corcuera, K. I. Zolotarev, E.M. Zsolnay, P.K. McLaughlin, and A. Trkov, International Reactor Dosimetry File 2002 (IRDF-2002), 2006, International Atomic Energy Agency, Vienna,

Austria, Technical Reports Series No. **452**, December 2006. Database available online at http://www-nds.iaea.org/irdf2002.

[19] K.I. Zolotarev, V.G. Pronyaev, and L. Leal, Private communication to R. Capote (IAEA), August 2008.

[20] M. Chadwick, P. Oblozinsky, M. Herman *et al*, "ENDF/B-VII.0: Next generation evaluated nuclear data library for nuclear science and technology", *Nucl. Data Sheets* **107** (2006), pp. 2931-3060.

[21] L.C. Leal, G. Arbanas *et al.*, "Resonance Region Covariance Analysis Method and New Covariance Data for 232Th, 233U, 235U, 238U and 239Pu". *Nucl. Data Sheets* **109** (2008) pp. 2868-2873.

[22] K.I. Zolotarev, S.A.Badikov, A.I.Blokhin, and A.V. Ignatyuk, "Np-237 Russian Reactor Dosimetry File RRDF-2009, Revision 3, MAT 9346, MF 3,33, MT 18," adopted from Material 9346 BROND-3 Np-237 Evaluation, October 2009.

[23] P.Schillebeeckx, "Resonance Evaluations of Cadmium Isotopes" (private communication to R. Capote and A. Trkov), December 2009.

[24] W. Mannhart, U235 experimental average cross-section data tabulated in „Status of the Evaluation of the Neutron Spectrum of the 235U+nth". In: IAEA Consultant's Meeting on International Neutron Cross-Section Standards: Measurements and Evaluation Techniques, IAEA Technical Report, **INDC(NDS)-0540** (IAEA, Vienna, 2008). Presentation available online at http://www.nds.iaea.org/standards/CM2010/Mannhart-U235-PFNS-CM2008.pdf

Nuclear Data Section
International Atomic Energy Agency
Vienna International Centre, P.O. Box 100
A-1400 Vienna
Austria

e-mail: services@iaeand.iaea.org
fax: (43-1) 26007
telephone: (43-1) 2600-21710
Web: http://www-nds.iaea.org