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RE-EVALUATION OF MICROSCOPIC AND INTEGRAL CROSS-SECTION DATA FOR IMPORTANT DOSIMETRY REACTIONS

Re-evaluation of the excitation functions for the ^{24}Mg(n,p)^{24}Na, ^{32}S(n,p)^{32}P, ^{60}Ni(n,p)$^{60m+g}$Co, ^{63}Cu(n,2n)^{62}Cu, ^{65}Cu(n,2n)^{64}Cu, ^{64}Zn(n,p)^{64}Cu, ^{115}In(n,2n)^{114m}In, ^{127}I(n,2n)^{126}I, ^{197}Au(n,2n)^{196}Au and ^{199}Hg(n,n')^{199m}Hg reactions

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August 2008

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^{65}Cu(n,2n)^{64}Cu, ^{64}Zn(n,p)^{64}Cu, ^{115}In(n,2n)^{114m}In, ^{127}I(n,2n)^{126}I,
^{197}Au(n,2n)^{196}Au and ^{199}Hg(n,n')^{199m}Hg reactions

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Abstract

Re-evaluations of cross sections and their associated covariance matrices have been carried out for ten dosimetry reactions:

- excitation functions for the ^{63}Cu(n,2n)^{62}Cu, ^{65}Cu(n,2n)^{64}Cu, ^{64}Zn(n,p)^{64}Cu, ^{115}In(n,2n)^{114m}In and ^{199}Hg(n,n')^{199m}Hg reactions were re-evaluated over the neutron energy range from threshold to 20 MeV;
- excitation functions for the ^{24}Mg(n,p)^{24}Na, ^{32}S(n,p)^{32}P and ^{60}Ni(n,p)$^{60m+g}$Co were re-evaluated in the energy range from threshold to 21 MeV;
- excitation functions for the ^{127}I(n,2n)^{126}I and ^{197}Au(n,2n)^{196}Au reactions were re-evaluated in the energy range from threshold to 32 and 40 MeV, respectively.

Benchmark calculations performed for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra show that the integral cross sections derived from the newly evaluated excitation functions exhibit improved agreement with related experimental data when compared with the equivalent data from the IRDF-2002 library.

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1. INTRODUCTION

Cross-section data for $^{24}Mg(n,p)^{24}Na$, $^{32}S(n,p)^{32}P$, $^{60}Ni(n,p)^{60m+g}Co$, $^{63}Cu(n,2n)^{62}Cu$, $^{65}Cu(n,2n)^{64}Cu$, $^{64}Zn(n,p)^{64}Cu$, $^{115}In(n,2n)^{114m}In$, $^{127}I(n,2n)^{126}I$, $^{197}Au(n,2n)^{196}Au$ and $^{199}Hg(n,n')^{199m}Hg$ reactions are needed to solve a wide spectrum of scientific and technical tasks. Activation detectors based on these reactions are commonly used in the field of reactor dosimetry. Furthermore, the $^{24}Mg(n,p)^{24}Na$, $^{32}S(n,p)^{32}P$, $^{63}Cu(n,2n)^{62}Cu$, $^{65}Cu(n,2n)^{64}Cu$ and $^{197}Au(n,2n)^{196}Au$ reactions are often used in experimental nuclear physics as monitor reactions for measurements of unknown cross sections by means of the activation method over the neutron energy range from 13 to 15 MeV. The $^{32}S(n,p)^{32}P$, $^{65}Cu(n,2n)^{64}Cu$ and $^{64}Zn(n,p)^{64}Cu$ reactions lead to the production of therapeutic ^{32}P and ^{64}Cu radionuclides, and are also important for medical applications.

Presentations at recent international symposia on reactor dosimetry [1.1, 1.2] show that the accuracy of the reactor pressure vessel neutron fluence and neutron spectra measurements are limited by our present knowledge of neutron dosimetry reaction cross sections. Evaluated excitation functions for $^{24}Mg(n,p)^{24}Na$, $^{32}S(n,p)^{32}P$, $^{60}Ni(n,p)^{60m+g}Co$, $^{63}Cu(n,2n)^{62}Cu$, $^{65}Cu(n,2n)^{64}Cu$, $^{64}Zn(n,p)^{64}Cu$, $^{115}In(n,2n)^{114m}In$, $^{127}I(n,2n)^{126}I$, $^{197}Au(n,2n)^{196}Au$ and $^{199}Hg(n,n')^{199m}Hg$ reactions from threshold to 20 MeV are given in the new International Reactor Dosimetry File, IRDF-2002 [1.3]. The final selection of evaluated data for IRDF-2002 was performed in October 2003. The most appropriate data for the $^{24}Mg(n,p)^{24}Na$, $^{32}S(n,p)^{32}P$, $^{60}Ni(n,p)^{60m+g}Co$, $^{65}Cu(n,2n)^{64}Cu$, $^{64}Zn(n,p)^{64}Cu$ and $^{197}Au(n,2n)^{196}Au$ reactions were taken from IRDF-90 version 2. Cross-section data for the $^{63}Cu(n,2n)^{62}Cu$ reaction were obtained from the ENDF/B-VI release 8 library, and data for $^{199}Hg(n,n')^{199m}Hg$ reaction were adopted from the JENDL/D-99 dosimetry file. Cross-section data for the $^{115}In(n,2n)^{114m}In$ and $^{127}I(n,2n)^{126}I$ reactions were selected from the CENDL dosimetry file (excitation functions for the $^{115}In(n,2n)^{114m}In$ and $^{127}I(n,2n)^{126}I$ reactions were evaluated in 1989 and 1991, respectively).

The ^{252}Cf spontaneous fission and ^{235}U thermal fission neutron spectra are the best standard benchmark neutron fields for testing threshold reactions. Tests carried out with IRDF-2002 data show that the calculated integral cross sections for the $^{24}Mg(n,p)^{24}Na$, $^{32}S(n,p)^{32}P$, $^{60}Ni(n,p)^{60m+g}Co$, $^{63}Cu(n,2n)^{62}Cu$, $^{65}Cu(n,2n)^{64}Cu$, $^{64}Zn(n,p)^{64}Cu$, $^{127}I(n,2n)^{126}I$, $^{197}Au(n,2n)^{196}Au$ and $^{199}Hg(n,n')^{199m}Hg$ reactions are not in good agreement with integral experimental data [1.4]. The largest discrepancies between the calculated and experimental data in the ^{252}Cf spontaneous fission neutron spectrum are observed for the $^{63}Cu(n,2n)^{62}Cu$ and $^{199}Hg(n,n')^{199m}Hg$ reactions: calculated to measured values (C/E) are 1.115 ± 0.078 and 0.833 ± 0.067, respectively.

Experimental data for ^{252}Cf spontaneous fission and ^{235}U thermal fission neutron spectra are absent for the $^{115}In(n,2n)^{114m}In$ reaction. As mentioned already above, the excitation function for this reaction was evaluated in 1989, and therefore any experimental data obtained at a later date [1.3, 1.4] were not taken into account [1.5, 1.6].

This work is devoted to the re-evaluation of the cross-section data and related uncertainty covariance matrixes of all the dosimetry reactions mentioned above.

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2. METHOD OF EVALUATION OF THE EXCITATION FUNCTIONS FOR DOSIMETRY REACTIONS

2.1. Sources of information used in the evaluation

Two common information sources were used for the ^{24}Mg(n,p)^{24}Na, ^{32}S(n,p)^{32}P, ^{60}Ni(n,p)$^{60m+g}$Co, ^{63}Cu(n,2n)^{62}Cu, ^{65}Cu(n,2n)^{64}Cu, ^{64}Zn(n,p)^{64}Cu, ^{115}In(n,2n)^{114m}In, ^{127}I(n,2n)^{126}I, ^{197}Au(n,2n)^{196}Au and ^{199}Hg(n,n')^{199m}Hg dosimetry reactions: differential and integral experimental data taken mainly from the EXFOR library (January 2007). Data and other relevant information were taken from the original publications when no records were found in EXFOR.

2.2. Analysis of experimental data

All experimental data were analyzed and, if possible, were corrected with respect to the newly recommended cross-section standards for monitor reactions and recommended decay data. Corrections to the experimental data based on the new standards lead to reductions in the discrepancies, and thus resulted in decreases in the uncertainties of the re-evaluated cross sections. The standards used to correct the microscopic experimental data under investigation are given in Table 2.1.

TABLE 2.1. DATA USED AS STANDARDS TO CORRECT THE MICROSCOPIC EXPERIMENTAL CROSS SECTIONS.

Monitor Reaction	Cross section used as standard	Half-life for residual nucleus	Radiation and energy		Emission probability per decay	
^{1}H(n,n)^{1}H	Pronyaev+ [2.1]					
^{6}Li(n,t)^{4}He	Pronyaev+ [2.1]					
^{24}Mg(n,p)^{24}Na	Zolotarev [*]	14.9590 (12) h	Gamma	1368.63 keV	1.0000(1)	[2.6, 2.7]
^{27}Al(n,α)^{24}Na	Zolotarev [2.2]	14.9590 (12) h	Gamma	1368.63 keV	1.0000(1)	[2.6, 2.7]
^{27}Al(n,p)^{27}Mg	Zolotarev+ [2.3]	9.458 (12) m	Gamma Gamma	843.76 keV 1014.44 keV	0.718 (4) 0.280 (4)	[2.6, 2.7] [2.6, 2.7]
^{31}P(n,α)^{28}Al	Gopych+ [2.4]	2.2414 (12) m	Gamma	1778.85 keV	1.000	[2.6, 2.7]
^{32}S(n,p)^{32}P	Zolotarev [*]	14.263 (3) d	Beta+	1710.48 keV	1.000	[2.7]
^{56}Fe(n,p)^{56}Mn	IRDF-2002 [2.5]	2.5789 (1) h	Gamma Gamma	846.754 keV 1810.72 keV	0.9887 (3) 0.2719 (79)	[2.6, 2.7] [2.6, 2.7]
^{58}Ni(n,p)^{58}Co	IRDF-2002 [2.5]	78.86 (6) d	Gamma Gamma	511 keV 810.759 keV	0.298 (4) 0.99450 (10)	[2.7] [2.7]
^{63}Cu(n,2n)^{62}Cu	Zolotarev [*]	9.73 (2) m	Beta+ Gamma Gamma	2925.8 keV 511 keV 1173.02 keV	0.9720 (2) 1.9486 (5) 0.00342 (5)	[2.7] [2.7] [2.6, 2.7]
^{65}Cu(n,2n)^{64}Cu	Zolotarev [*]	12.700 (2) h	Beta+ Beta– Gamma Gamma	653.1 keV 578.7 keV 511 keV 1345.77 keV	0.1740 (22) 0.390 (4) 0.348 (4) 0.00473 (10)	[2.7] [2.7] [2.7] [2.6, 2.7]
^{64}Zn(n,p)^{64}Cu	Zolotarev [*]	12.700 (2) h	Beta+ Beta– Gamma Gamma	653.1 keV 578.7 keV 511 keV 1345.77 keV	0.1740 (22) 0.390 (4) 0.348 (4) 0.00473 (10)	[2.7] [2.7] [2.7] [2.6, 2.7]
^{93}Nb(n,2n)^{92m}Nb	IRDF-2002 [2.5]	10.15 (2) d	Gamma	934.44 keV	0.9907 (4)	[2.6, 2.7]
^{197}Au(n,2n)^{196}Au	Zolotarev [*]	6.183 (10) d	Gamma Gamma Gamma	333.03 keV 355.73 keV 426.10 keV	0.229 (6) 0.870 (4) 0.066 (4)	[2.6, 2.7] [2.6, 2.7] [2.6, 2.7]
^{235}U(n,f)	Pronyaev+ [2.1]					
^{238}U(n,f)	Pronyaev+ [2.1]					

Beta transition: $E_{\beta max}$ values are listed.
[*] cross-section data from this work.

Recommended cross-section data were taken from Ref. [2.8] for the monitor reactions used in measurements of integral cross sections in ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra. Digital data for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were taken from Refs. [2.9,2.10], respectively. Information about the isotopic compositions of the elements was taken from Ref. [2.11].

2.3. Theoretical model calculations for the cross sections of dosimetry reactions

Theoretical model calculations provided an additional source of cross-section information for reactions with inadequate experimental data. Hence, theoretical calculations were carried out to determine the excitation functions of the ^{127}I(n,2n)^{126}I and ^{199}Hg(n,n')^{199m}Hg reactions above 9 MeV.

The optical-statistical method was used for a theoretical description of the excitation function of the ^{127}I(n,2n)^{126}I and ^{199}Hg(n,n')^{199m}Hg reactions, taking into account the contribution of the direct, pre-equilibrium and statistical equilibrium processes in different outgoing channels. These calculations were carried out by means of a modified version of the GNASH code [2.12, 2.13], which includes a subroutine for width fluctuation corrections.

Penetrability coefficients for neutrons were calculated on the basis of the generalized optical model, which estimates the cross sections for the direct excitations of collective low-lying levels. The ECIS coupled-channel deformed optical model code was used for these calculations [2.14], and the optical coefficients of the proton- and alpha-particle penetrabilities were determined by means of the SCAT2 code [2.15].

The data on discrete levels parameters for ^{127}I, ^{199}Hg and all residual nuclei were obtained from Ref. [2.6]. Unknown branching ratios were estimated on the basis of statistical calculations of the possible E1, E2 and M1 gamma-ray transitions. Intensities of such transitions were calculated from the radiation strength functions recommended in Ref. [2.16].

Continuum level densities were represented by means of the Gilbert-Cameron model [2.17] based on the Cook parameters [2.18] (mode IBSF = 1 in the GNASH code). Calculations of the gamma-ray transition probabilities in the continuum region of the excited states of all nuclei under consideration were made in terms of the hypothesis of the domination of the giant dipole resonance with radiative strength function from Kopecky-Uhl systematics [2.19]. Recommended parameters for the giant dipole resonances were taken from Ref. [2.20].

The modified GNASH code was used to calculate the cross sections of the ^{127}I(n,2n)^{126}I and ^{199}Hg(n,n')^{199m}Hg reactions from 9 to 32 MeV and 9 to 20 MeV, respectively.

2.4. Statistical analyses of cross sections from the database

The method of statistical analysis of the correlated data was used to evaluate the excitation functions of the dosimetry reactions, as described in Refs. [2.21, 2.22]. Statistical analyses of the experimental reaction cross sections were carried out using the non-linear regression model. The following rational function was used as the model function (Pade approximation):

$$f(E) = C + \sum_{i=1}^{l_1} \frac{a_i}{E-r_i} + \sum_{k=1}^{l_2} \frac{\alpha_k (E-\varepsilon_r)+\beta_k}{(E-\varepsilon_k)^2+\gamma_k^2} ,$$

where E is the neutron energy, and C, a_i, r_i, α_k, β_k, ε_k and γ_k are the parameters to be determined. The total number of parameters of the Pade approximation is equal to $L = 2l_1 + 4l_2 + 1$.

Parameters of the model function are determined from the minimum of the functional:

$$S(\vec{\beta}) = (\vec{\sigma} - \vec{f})^T (DPD)^{-1} (\vec{\sigma} - \vec{f}),$$

in which the functional to be minimized ($\vec{\beta}$) is the vector of the parameters to be determined; $\vec{\sigma}$ is the vector of cross sections from the database; D is the diagonal matrix of the uncertainty of the cross sections from the database; P is the correlation matrix of the experimental data used to evaluate the excitation function; and the superscript T denotes a transpose.

Technical aspects of the minimization process based on the use of the discrete optimization method and Newton-Gauss algorithm are described in Ref. [2.23]. The algorithm used to minimize $S(\vec{\beta})$ contains two approximations that simplify the calculation scheme appreciably:

1) cross-section data obtained in different experiments are assumed to be uncorrelated;
2) an average correlation coefficient is used to describe the correlations between cross sections measured in one experiment.

The covariance matrix of the uncertainties of the evaluated parameters W($\vec{\beta}$) and the uncertainties of the evaluated function at point $\Delta f(E_{i_k}^k, \vec{\beta})$ are determined from the relationships:

$$W(\vec{\beta}) = \tfrac{s}{n-L} (X^T V^{-1} X)^{-1},$$

$$\Delta f(E_{i_k}, \vec{\beta}) = \sum_{m=1}^{L} \sum_{j=1}^{L} X_{i_k m}^k \, X_{i_k j}^k \, W_{mj},$$

where n is the total number of cross-section data used in the analysis of a reaction, and X is the (n $\times$ L) matrix of the coefficients of sensitivity of the rational function to a change in parameters based on:

$$X_{i_k m} = \frac{\partial f(E_{i_k}, \vec{\beta})}{\partial \beta_m}.$$

The structure of the uncertainties for all experimental data was analyzed to determine the average correlation coefficients. The average correlation coefficient $\vec{p}^k$ for the kth experiment containing information on the n_k values of the reaction excitation function was determined by means of the formulae:

$$\vec{p}^k = \frac{2}{(n_k - 1) n_k} \sum_{i=1}^{n_k - 1} \sum_{j=i+1}^{n_k} \frac{\sum_{m=1}^{l} P_{ij}^m e_i^m e_j^m}{e_i e_j},$$

where $e_i(e_j)$ is the total uncertainty (standard deviation) of the cross section at the ith (jth) point corresponding to a standard deviation of 1σ; e_i^m (e_j^m) is the mth component of the systematic uncertainty of the cross section at the ith (jth) point; P_{ij}^m is the coefficient of the correlation between the mth components of the systematic uncertainties at the ith (jth) points; and l is the number of components of the systematic uncertainty.

This method of statistical analysis of the correlated data was performed by means of the PADE-2 code [2.21].

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3. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE ^{24}Mg(n,p)^{24}Na REACTION

The ^{24}Mg isotopic abundance in natural magnesium is 78.99 ± 0.03 atom percent, and the ^{24}Na obtained via the (n,p) reaction undergoes 100% β^- decay with a half-life of (14.9590 ± 0.0012) hours. 1368.633-keV gamma radiation (I_γ = 1.0000 ± 0.0001) is normally used to determine the ^{24}Mg(n,p)^{24}Na reaction rate. Recommended decay data for the half-life, beta+ and gamma ray emission probabilities per decay of ^{24}Na were taken from Ref. [2.7] of Section 2.

Microscopic experimental data were analyzed during the preparation of the input database assembled in order to evaluate the cross sections and uncertainties for the ^{24}Mg(n,p)^{24}Na reaction [3.1-3.37]. During this procedure, various experimental data [3.6, 3.8-3.10, 3.13-3.18, 3.21-3.27, 3.29, 3.30] were corrected on the basis of the newly recommended cross-section data for the relevant monitor reactions and the recommended decay data (see Table 2.1).

The experimental data of Tewes *et al.* [3.4] in the energy range from 10.1 to 14.0 MeV were re-normalized to a value of (151.0 ± 4.8) mb at 10.1 MeV, as determined from the recent measurements of Mannhart and Schmidt [3.30]. Experimental data in Refs. [3.1, 3.6-3.8, 3.11-3.21, 3.24-3.28, 3.31, 3.33 and 3.37] were obtained by means of Mg samples, and the natural isotopic composition was corrected in order to re-determine the contributions from the ^{25}Mg(n,x)^{24}Na and ^{26}Mg(n,t)^{24}Na reactions (cross-section data for these reactions were taken from Ref. [3.38]). Cross-section data from Refs. [3.31-3.37] were rejected due to their significant deviation from the main bulk of the experimental data. Within these rejected experimental data, the cross-section values reported in Refs. [3.31-3.33, 3.35 and 3.37] comprised only one or two energy points from 14 to 15 MeV.

The excitation function for the ^{24}Mg(n,p)^{24}Na reaction in the energy region from threshold to 21 MeV was evaluated by means of statistical analyses of the experimental cross-section data [3.1-3.30]. Uncertainties in the evaluated excitation function for the ^{24}Mg(n,p)^{24}Na reaction are given in the form of a relative covariance matrix for 49-neutron energy groups (LB = 5). Covariance matrix uncertainties were calculated simultaneously with the recommended cross-section data by means of the PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix in File-33 are as follows:

1.55727E-08	1.67377E-08	1.91246E-08	2.28687E-08
2.38958E-08	2.47166E-08	2.52915E-08	2.57443E-08
2.66240E-08	2.88798E-08	3.03155E-08	3.58359E-08
4.31075E-08	5.20677E-08	6.20313E-08	7.12929E-08
7.97354E-08	3.86395E-07	2.10781E-06	6.80212E-06
2.42172E-05	2.97164E-05	3.88705E-05	5.73034E-05
7.75092E-05	8.22933E-05	9.42793E-05	1.11842E-04
1.20553E-04	1.34987E-04	1.44469E-04	1.74828E-04
1.82501E-04	2.15794E-04	2.22742E-04	2.57857E-04
2.63646E-04	2.75939E-04	3.40886E-04	3.66270E-04
5.30002E-04	5.44725E-04	6.89741E-04	7.78637E-04
8.72319E-04	1.47812E-03	2.89228E-03	4.66436E-03
4.77065E-02			

Evaluated group cross sections and their uncertainties for the excitation function of the ^{24}Mg(n,p)^{24}Na reaction are listed in Table 3.1. Group boundaries are the same as in File-33.

TABLE 3.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE ^{24}Mg(n,p)^{24}Na REACTION IN THE ENERGY RANGE FROM THRESHOLD TO 21 MeV.

Neutron energy (MeV) from – to	Cross-section (mb)	Uncer-tainty (%)	Neutron energy (MeV) from – to	Cross-section (mb)	Uncer-tainty (%)
4.933 – 5.800	0.078	21.80	10.600 – 10.800	158.434	1.19
5.800 – 6.000	1.020	5.54	10.800 – 11.000	159.876	1.09
6.000 – 6.200	3.528	3.90	11.000 – 11.250	162.478	1.09
6.200 – 6.400	5.872	2.77	11.250 – 11.500	170.231	1.29
6.400 – 6.600	22.212	2.61	11.500 – 11.750	184.878	1.35
6.600 – 6.800	43.378	2.68	11.750 – 12.000	166.900	2.00
6.800 – 7.000	35.240	2.12	12.000 – 12.500	176.994	1.21
7.000 – 7.200	46.303	1.97	12.500 – 13.000	192.182	0.81
7.200 – 7.400	51.746	1.80	13.000 – 13.500	196.737	0.86
7.400 – 7.600	56.343	1.93	13.500 – 14.000	194.298	0.70
7.600 – 7.800	83.442	1.78	14.000 – 14.500	190.236	0.69
7.800 – 8.000	95.962	1.31	14.500 – 15.000	172.277	0.75
8.000 – 8.200	113.897	1.04	15.000 – 15.500	158.779	0.94
8.200 – 8.400	119.699	0.94	15.500 – 16.000	147.689	1.08
8.400 – 8.600	116.920	0.91	16.000 – 16.500	137.278	1.15
8.600 – 8.800	121.903	0.91	16.500 – 17.000	127.340	1.19
8.800 – 9.000	123.949	0.93	17.000 – 17.500	117.879	1.22
9.000 – 9.200	126.555	0.94	17.500 – 18.000	108.930	1.26
9.200 – 9.400	120.620	0.96	18.000 – 18.500	100.522	1.37
9.400 – 9.600	132.595	1.01	18.500 – 19.000	92.671	1.57
9.600 – 9.800	124.498	1.15	19.000 – 19.500	85.376	1.85
9.800 – 10.000	137.877	1.41	19.500 – 20.000	78.626	2.21
10.000 – 10.200	150.668	1.59	20.000 – 20.500	72.400	2.64
10.200 – 10.400	156.323	1.62	20.500 – 21.000	66.672	3.12
10.400 – 10.600	157.391	1.40			

One can see from Table 3.1 that the smallest uncertainties in the evaluated cross sections of 0.69% to 0.86% are observed in the neutron energy range from 12.5 to 15.0 MeV. A significant uncertainty of 21.8% in the cross sections from threshold to 5.8 MeV arises from the large uncertainties in the experimental data within this region and the existing discrepancies between these experimental data.

Fig. 3.1 shows the re-evaluated excitation function for the ^{24}Mg(n,p)^{24}Na reaction over the neutron energy range from 4.5 to 12.0 MeV, while Figs. 3.2, 3.3 and 3.4 depict the equivalent data at various energy ranges from 4.5 to 21.0 MeV with the cross sections in IRDF-2002 and the experimental data.

Integral experiments for the ^{24}Mg(n,p)^{24}Na reaction are described in Refs. [3.39-3.58]. Seventeen experiments was carried out in neutron fields with similar spectra to the ^{235}U thermal fission neutron spectrum [3.39-3.55], and four experiments were performed in a ^{252}Cf spontaneous fission neutron spectrum [3.49, 3.56-3.58]. Experimental data obtained for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were corrected with respect to the newly recommended cross sections for the monitor reactions and decay data. Measured integral cross sections for the ^{252}Cf spontaneous fission neutron spectrum [3.49, 3.56-3.58] range from (1.959

± 0.087) mb [3.56, 3.58] to (2.022 ± 0.061) mb [3.57], while a value of (1.996 ± 0.049) mb has been evaluated by Mannhart [3.59].

Measured integral cross sections for the ^{235}U thermal fission neutron spectrum range from 1.082 to 1.508 mb [3.39-3.55]. The lowest value of 1.082 mb was obtained by Shikata in measurements on the JRR-1 reactor [3.39], and no information on the uncertainty is given in this publication. A value of (1.508 ± 0.69) mb was measured by Boldeman [3.40], and has been corrected to give a recommended cross section of (69.08 ± 1.36) mb for the $^{32}S(n,p)^{32}P$ monitor reaction [3.59].

Neutron spectra measurements show that the standard ^{235}U thermal fission neutron spectrum may be obtained from a thermal column with 90%-enriched ^{235}U fission plate. Experimental data obtained from measurements in reactor cores and critical assemblies need to be corrected for differences between the real spectrum and the standard ^{235}U thermal fission neutron spectrum. Determination of this adjustment factor is a significant problem, and represents the major source of uncertainty in the resulting cross section. Therefore, integral experimental data obtained in measurements with perturbed fission spectra were not considered in the evaluation [3.39, 3.42, 3.43, 3.45-3.47, 3.50, 3.51, 3.53 and 3.54].

Experimental data measured in facilities with an enriched ^{235}U fission plate converter agree within 2% [3.40, 3.41, 3.44, 3.48, 3.49], except for the data from Ref. [3.52]. The integral cross section for the $^{24}Mg(n,p)^{24}Na$ reaction given in Ref. [3.52] is (1.372 ± 0.090) mb – this value is significantly lower than all of the other measured cross sections, but also possesses a large uncertainty.

An average weighted cross section of (1.490 ± 0.027) mb has been evaluated on the basis of the experimental measurements of Refs. [3.40, 3.41, 3.44, 3.48 and 3.49]. This value does not contradict the cross section evaluated by Mannhart of (1.451 ± 0.023) mb [3.59], and both recommendations were used in benchmark calculations. The results of tests with the re-evaluated excitation function of the $^{24}Mg(n,p)^{24}Na$ reaction are given in Table 3.2, in which C/E is the ratio of the calculated to experimental cross sections. These data show that the averaged cross sections calculated from the re-evaluated excitation function for the $^{24}Mg(n,p)^{24}Na$ reaction exhibit greater agreement with the integral experimental data than the equivalent data from IRDF-2002 library. The discrepancies between the calculated and experimental data decrease if the re-evaluated cross sections are defined as (1.490 ± 0.027) mb for the ^{235}U thermal fission spectrum and (2.022 ± 0.061) mb for the ^{252}Cf spontaneous fission neutron spectrum. However, more precise measurements of the integral cross sections of the $^{24}Mg(n,p)^{24}Na$ reaction are required for both ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra.

TABLE 3.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR THE $^{24}Mg(n,p)^{24}Na$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	**Average cross section, mb**		**C/E** [*]	**C/E** [3.57]	**C/E** [3.59]
	Calculated	**Measured**			
^{235}U thermal fission neutron spectrum	1.5106 [A]	1.490 ± 0.027 [*]	1.0138		1.0411
	1.5517 [B]	1.451 ± 0.023 [3.59]	1.0414		1.0694
^{252}Cf spontaneous fission neutron spectrum	2.1038 [A]	2.022 ± 0.061 [3.57]		1.0405	1.0540
	2.1599 [B]	1.996 ± 0.049 [3.59]		1.0682	1.0821

[A] Present evaluation.
[B] IRDF-2002 (IRDF-90 version 2).
[*] average weighted cross section from experimental data [3.40, 3.41, 3.44, 3.48, 3.49].

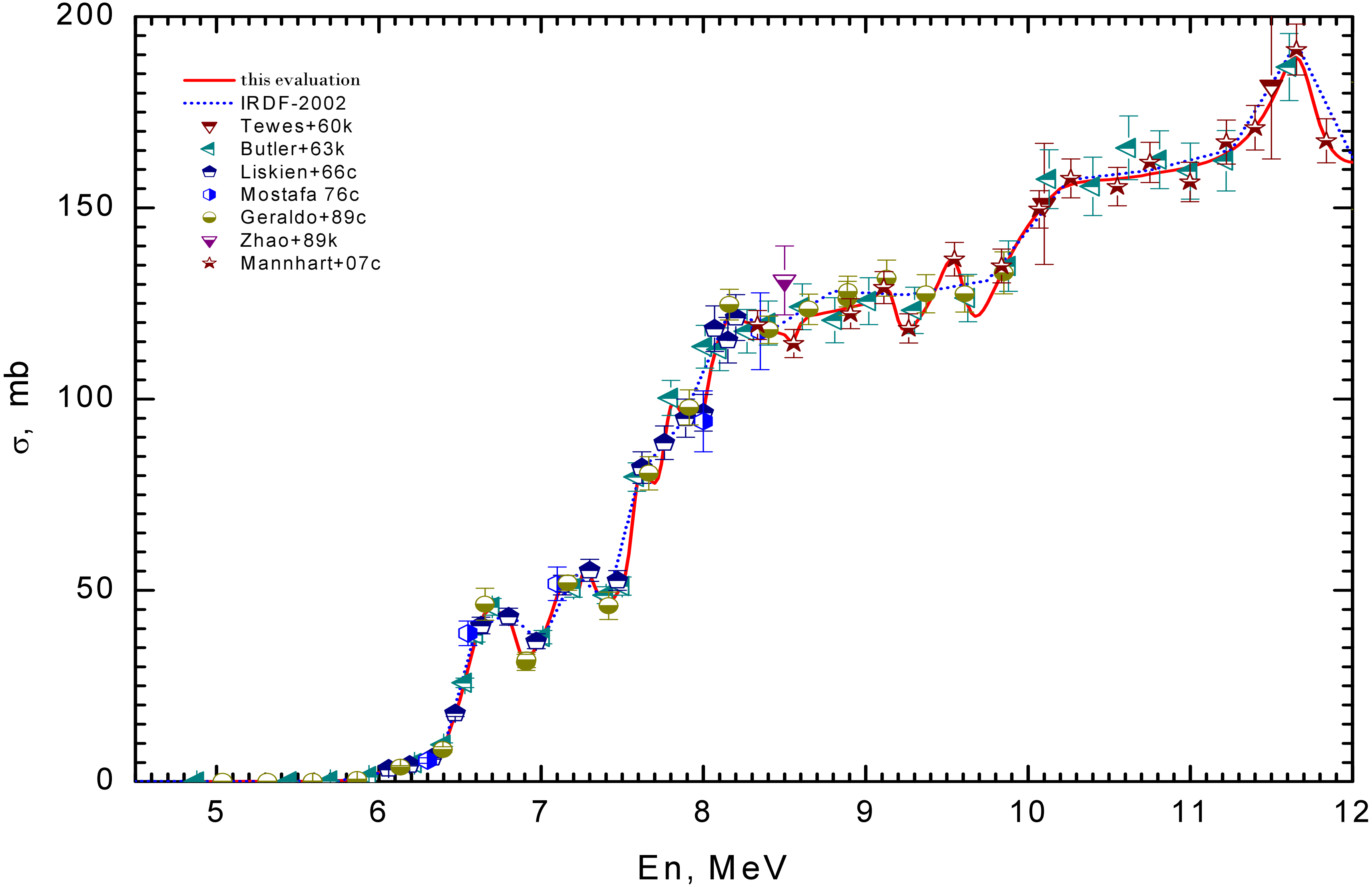

FIG. 3.1. Re-evaluated excitation function of the $^{24}Mg(n,p)^{24}Na$ reaction in the energy range from threshold to 12 MeV in comparison with IRDF-2002 and experimental data.

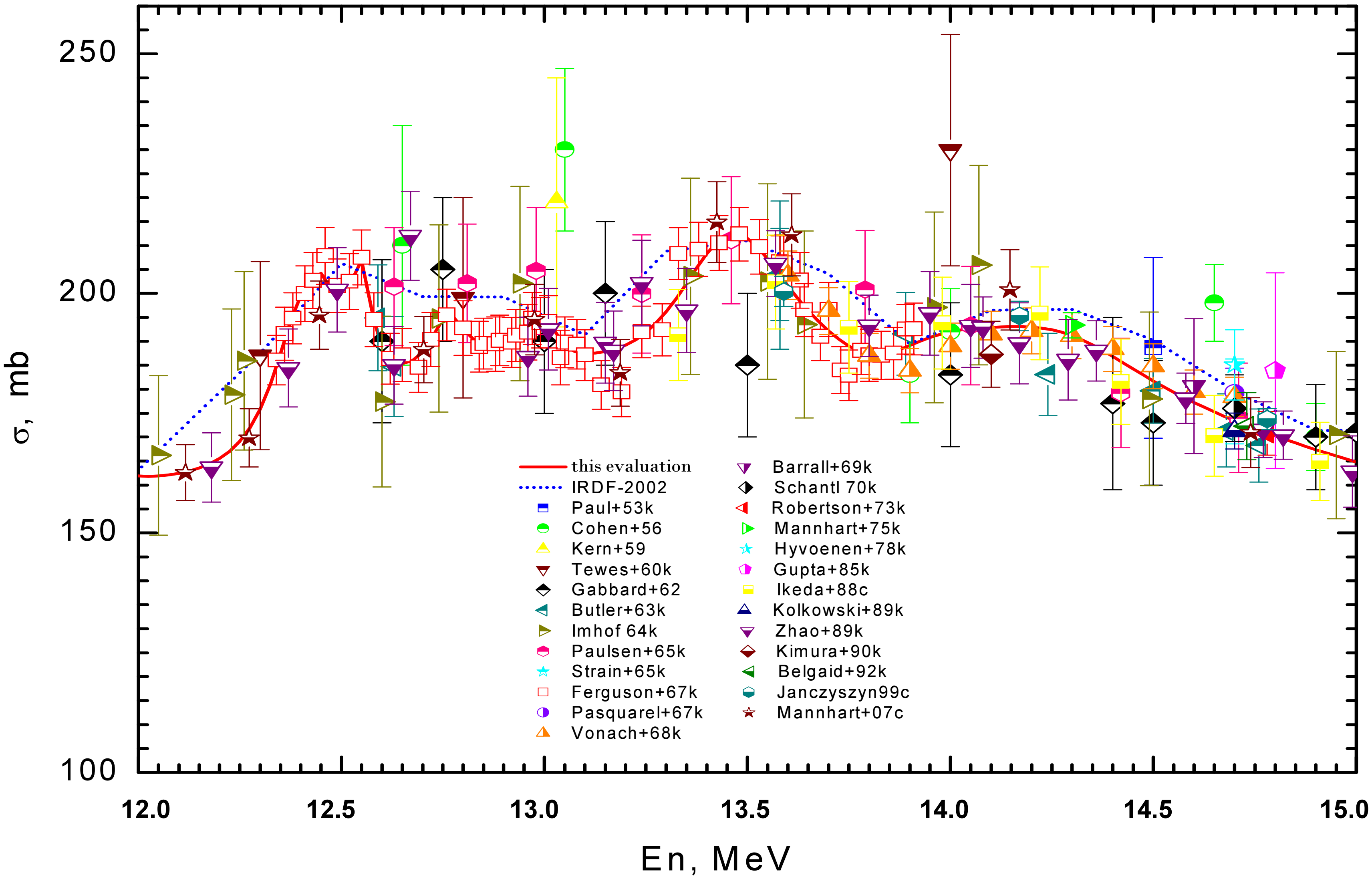

FIG. 3.2. Re-evaluated excitation function of the $^{24}Mg(n,p)^{24}Na$ reaction in the energy range from 12 to 15 MeV in comparison with IRDF-2002 and experimental data.

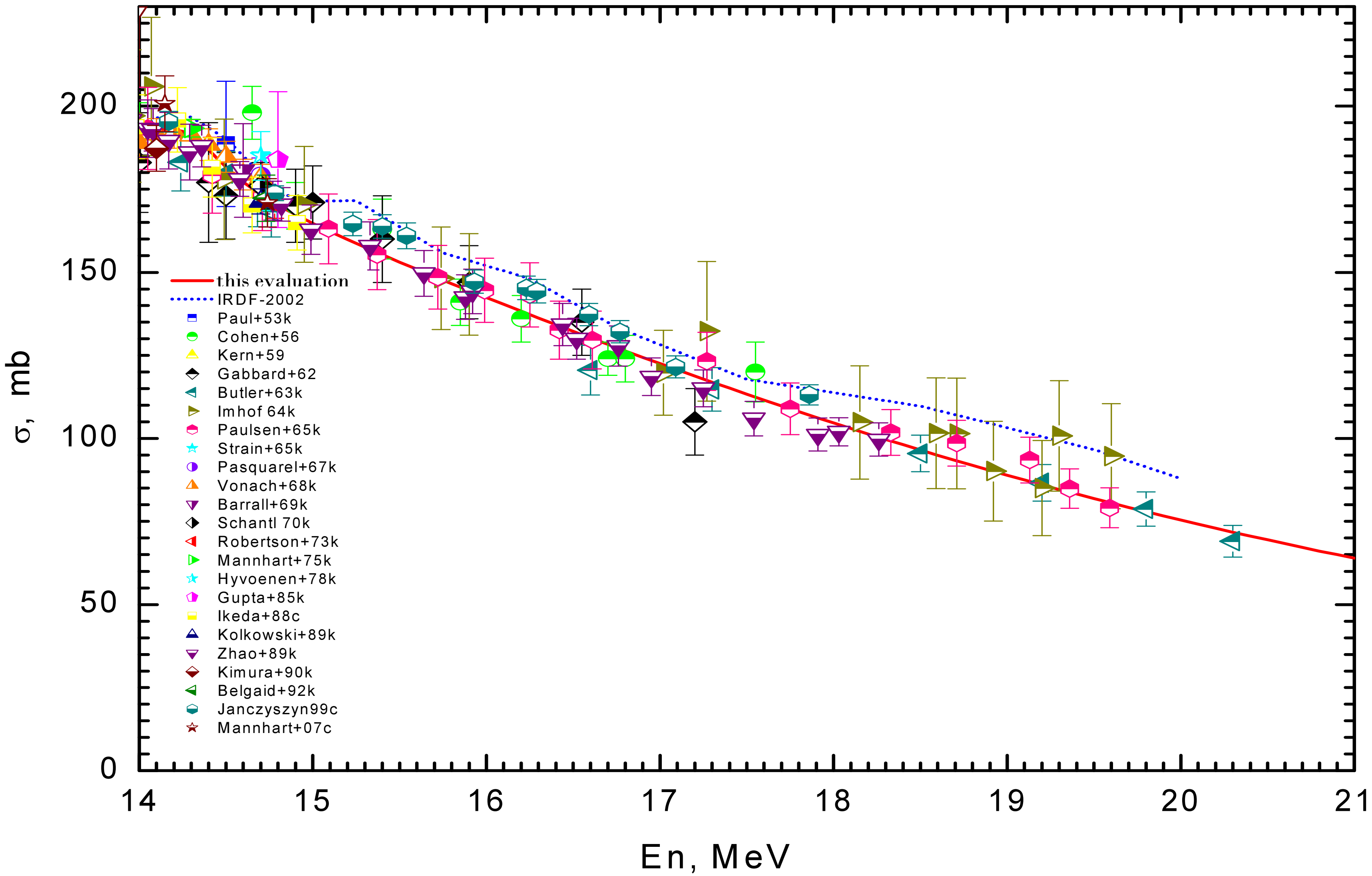

FIG. 3.3. Re-evaluated excitation function of the $^{24}Mg(n,p)^{24}Na$ reaction in the energy range from 14 to 21 MeV in comparison with IRDF-2002 and experimental data.

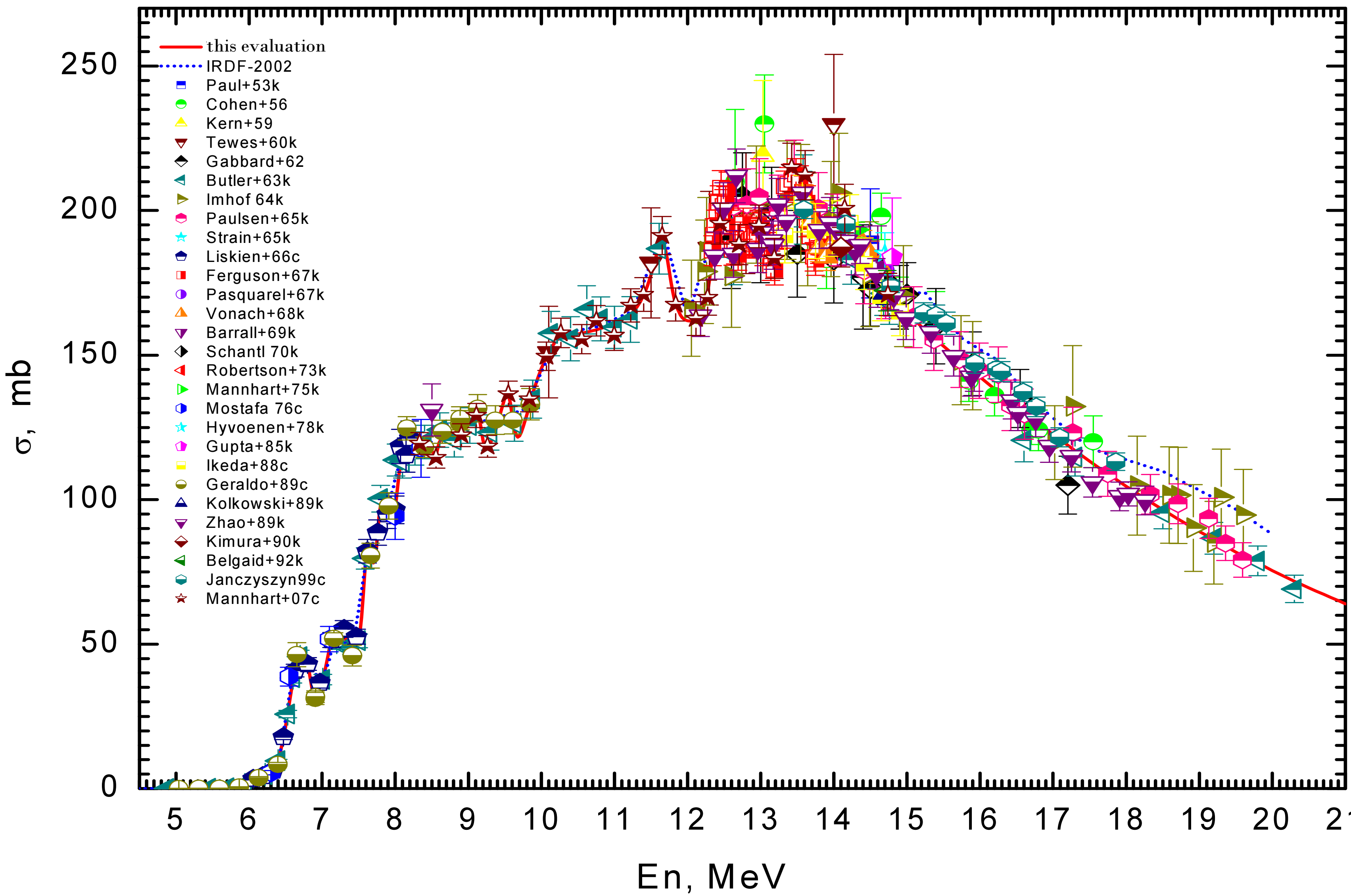

FIG. 3.4. Re-evaluated excitation function of the $^{24}Mg(n,p)^{24}Na$ reaction in the energy range from threshold to 21 MeV in comparison with IRDF-2002 and experimental data.

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4. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE $^{32}S(n,p)^{32}P$ REACTION

The isotopic abundance of ^{32}S in natural sulphur is 95.02 ± 0.09 atom percent, and the ^{32}P obtained via the (n,p) reaction undergoes 100% β^- decay with a half-life of (14.263 ± 0.003) days. Beta particles with an end-point energy of 1710.48 keV (mean energy of 694.9 keV) and total intensity of 100% are used to determine the $^{32}S(n,p)^{32}P$ reaction rate. Recommended decay data for the half-life and beta particle emission probability per decay of ^{32}P were taken from Ref. [2.7] of Section 2.

Microscopic experimental data were analyzed during the preparation of the input database assembled in order to evaluate the cross sections and uncertainties for the $^{32}S(n,p)^{32}P$ reaction [4.1-4.17]. Various experimental data [4.3, 4.4, 4.6, 4.10, 4.11, 4.16, 4.17] were corrected on the basis of the newly recommended cross-section data for the relevant monitor reactions and the recommended decay data (see Table 2.1).

Specific adjustments were also applied to some of the experimental data as outlined below. Total uncertainties between 15 and 21% were assigned to the experimental data of Klema and Hanson [4.1], and a systematic uncertainty of 15% was added to the experimental data of Ricamo *et al.* [4.2]. The uncertainties in the standard cross sections were added to the experimental data of Refs. [4.6, 4.8, 4.14]. Allan obtained two significantly different values for the cross section of (289 ± 25) and (365 ± 25) mb at a neutron energy of 14 MeV [4.5] – the second value contradicts the main bulk of the other experimental data and was set aside from the evaluation. Cross-section data given in Refs. [4.15-4.17] were also rejected due to their significant overestimation of the cross sections of the $^{32}S(n,p)^{32}P$ reaction, with only one experimental value in the energy range from 14 to 15 MeV.

The excitation function for the $^{32}S(n,p)^{32}P$ reaction in the energy region from threshold to 21 MeV was evaluated by means of statistical analyses of the experimental cross-section data [4.1-4.14]. Furthermore, the energy dependence of the cross section from threshold to 1.56 MeV was extrapolated with L = 0 penetrability function for the outgoing p + ^{32}P channel [4.18]. Uncertainties in the evaluated excitation function for the $^{32}S(n,p)^{32}P$ reaction are given in the form of a relative covariance matrix for 49-neutron energy groups (LB = 5). Covariance matrix uncertainties were calculated simultaneously with the recommended cross-section data by means of the PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix in File-33 are as follows:

1.12232E-07	1.14729E-07	1.18410E-07	1.23857E-07
1.30192E-07	1.37164E-07	1.46109E-07	1.53138E-07
1.56481E-07	1.61518E-07	1.84699E-07	1.99948E-07
2.24947E-07	2.38904E-07	2.63635E-07	2.90586E-07
3.07528E-07	3.96799E-07	5.14078E-07	6.63438E-07
8.48014E-07	1.07152E-06	1.33780E-06	1.64978E-06
2.01146E-06	2.68104E-06	8.71278E-06	1.41994E-04
2.86573E-04	3.27802E-04	3.58509E-04	4.02195E-04
4.32464E-04	5.63640E-04	6.98782E-04	8.08989E-04
9.55269E-04	1.05650E-03	1.15169E-03	1.28651E-03
1.34043E-03	1.42786E-03	1.62758E-03	1.70408E-03
1.85140E-03	3.93999E-03	7.38416E-03	1.39707E-02
1.66101E-02			

Evaluated group cross sections and their uncertainties for the excitation function of the $^{32}S(n,p)^{32}P$ reaction are listed in Table 4.1. Group boundaries are the same as in File-33.

TABLE 4.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{32}S(n,p)^{32}P$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 21 MeV

Neutron energy (MeV) from – to	Cross section (mb)	Uncer-tainty (%)	Neutron energy (MeV) from – to	Cross section (mb)	Uncer-tainty (%)
0.958 – 2.000	2.266	10.94	8.000 – 8.500	337.897	2.87
2.000 – 2.200	37.243	3.72	8.500 – 9.000	345.442	2.95
2.200 – 2.400	77.314	3.33	9.000 – 9.500	356.385	3.04
2.400 – 2.600	84.928	3.32	9.500 – 10.000	370.121	3.06
2.600 – 2.800	88.674	3.27	10.000 – 10.500	385.822	2.99
2.800 – 3.000	114.865	3.34	10.500 – 11.000	395.288	2.81
3.000 – 3.200	161.191	3.37	11.000 – 11.500	393.101	2.57
3.200 – 3.400	172.418	3.86	11.500 – 12.000	378.428	2.32
3.400 – 3.600	220.489	3.43	12.000 – 12.500	354.514	2.08
3.600 – 3.800	183.632	3.88	12.500 – 13.000	325.756	1.86
3.800 – 4.000	194.915	4.44	13.000 – 13.500	295.609	1.65
4.000 – 4.200	336.752	4.22	13.500 – 14.000	266.173	1.46
4.200 – 4.400	339.058	3.95	14.000 – 14.500	238.544	1.33
4.400 – 4.600	291.271	3.58	14.500 – 15.000	213.217	1.31
4.600 – 4.800	264.011	3.45	15.000 – 15.500	190.365	1.41
4.800 – 5.000	251.888	3.55	15.500 – 16.000	169.985	1.58
5.000 – 5.200	248.468	3.78	16.000 – 16.500	151.989	1.77
5.200 – 5.400	255.711	3.99	16.500 – 17.000	136.237	1.94
5.400 – 5.600	276.922	3.97	17.000 – 17.500	122.564	2.08
5.600 – 5.800	304.713	3.68	17.500 – 18.000	110.792	2.20
5.800 – 6.000	324.063	3.33	18.000 – 18.500	100.742	2.34
6.000 – 6.500	331.551	3.11	18.500 – 19.000	92.239	2.52
6.500 – 7.000	329.358	2.96	19.000 – 20.000	82.158	2.96
7.000 – 7.500	335.908	2.84	20.000 – 21.000	72.418	3.98
7.500 – 8.000	340.682	2.83			

One can see from Table 4.1 that the smallest uncertainties in the evaluated cross sections between 1.31% and 1.94% are observed in the neutron energy range from 12.5 to 17.0 MeV, while these uncertainties are at their highest near the threshold.

Fig. 4.1 shows the re-evaluated excitation function for the $^{32}S(n,p)^{32}P$ reaction over the neutron energy range from 1.0 to 5.0 MeV, and Fig. 4.2 the equivalent data from threshold to 21.0 MeV compared with the cross sections of IRDF-2002 and the experimental data. Over the neutron energy range from 1.5 to 20 MeV, the re-evaluated excitation function is seen to be in better agreement with the corrected experimental data than the IRDF-2002 evaluation.

Integral experimental data for the $^{32}S(n,p)^{32}P$ reaction are given in Refs. [4.19-4.29]. Eight experiments were carried out in neutron fields with similar spectra to the ^{235}U thermal fission neutron spectrum [4.19-4.26], and five experiments were performed in a ^{252}Cf spontaneous fission neutron spectrum [4.24, 4.25, 4.27-4.29].

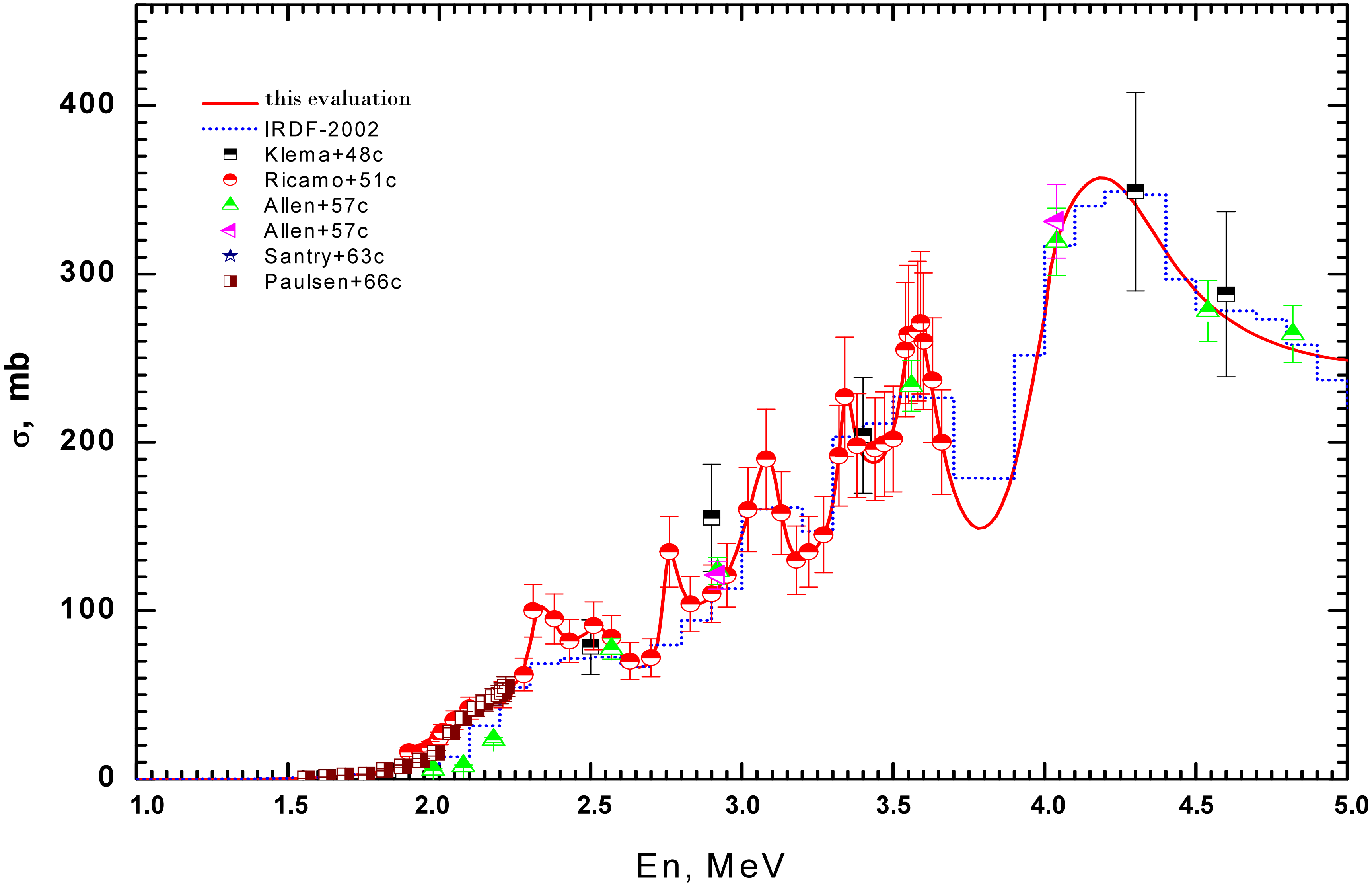

FIG. 4.1. Re-evaluated excitation function of the $^{32}S(n,p)^{32}P$ reaction in the energy range from threshold to 5 MeV in comparison with IRDF-2002 and experimental data.

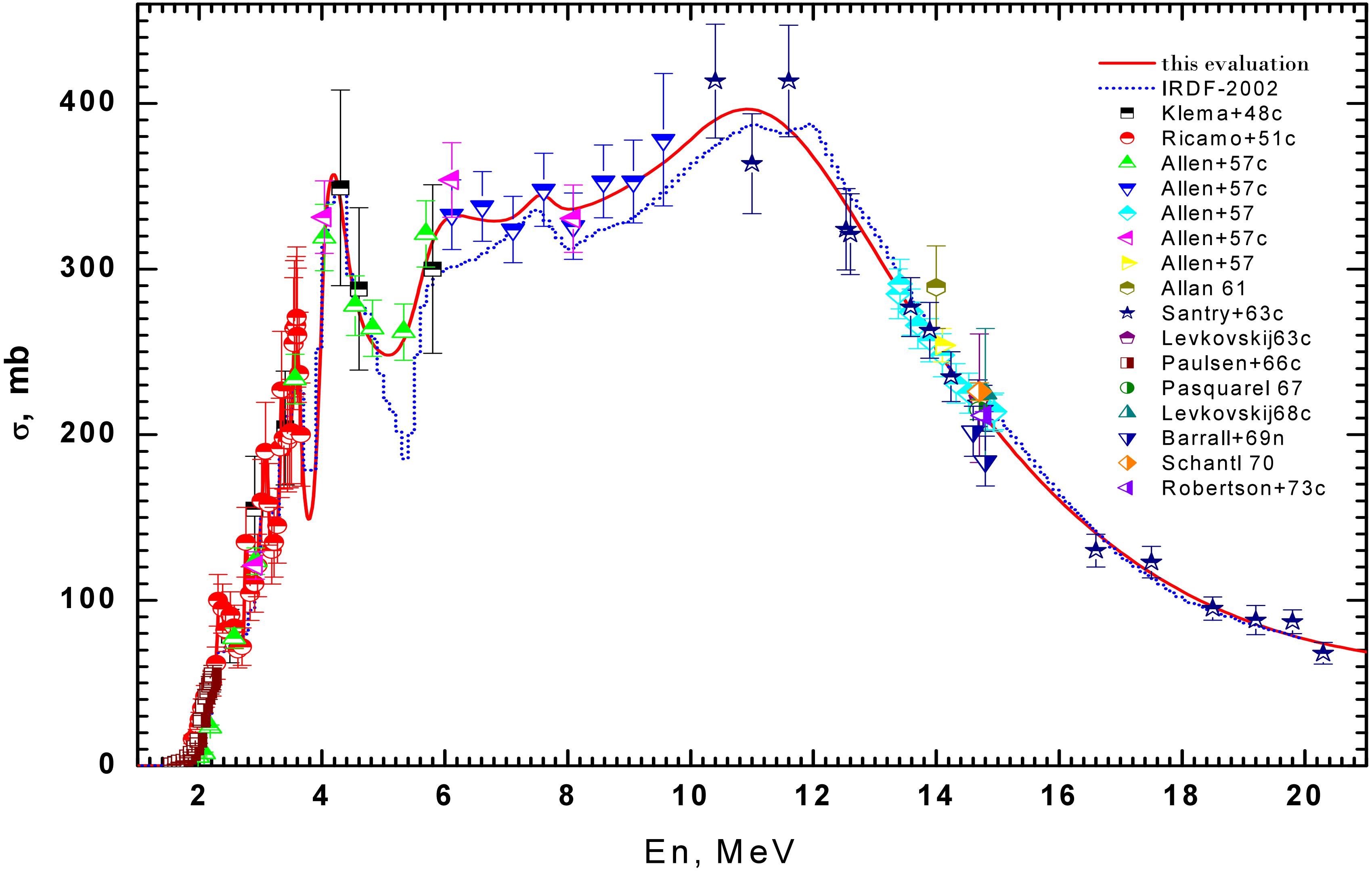

FIG .4.2. Re-evaluated excitation function of the $^{32}S(n,p)^{32}P$ reaction in the energy range from threshold to 20 MeV in comparison with IRDF-2002 and experimental data.

Measured integral cross sections for the ^{235}U thermal fission neutron spectrum extend over a wide range from 41.0 to 74.0 mb [4.19-4.26], with the lowest value of 41 mb being obtained by Lewis and Butler [4.23] although their uncertainties are not quantified. The highest cross-section value of 74.0 ± 3.0 mb was measured by Fabry [4.22] with an enriched ^{235}U fission plate converter. However, the more representative measurements were carried out by Kobayashi and Kimura [4.24, 4.25]: ^{235}U fission spectrum was generated by an enriched ^{235}U fission plate converter in the KUR thermal reactor to give a cross section of (69.41 ± 2.00) mb, in good agreement with an evaluated cross section of (69.080 ± 1.361) mb [4.30].

Measured integral cross sections for the ^{252}Cf spontaneous fission neutron spectrum range from (72.40 ± 4.80) [4.24] to (73.24 ± 2.70) mb [4.27, 4.29], and these data agree well with the evaluated cross section of (72.54 ± 2.53) mb [4.30]. However, the measurements of Dezsö and Csikai give a significantly lower value of (66.35 ± 2.79) mb [4.28] that was not taken into account in the benchmark calculations.

Evaluated excitation functions for the $^{32}S(n,p)^{32}P$ reaction were tested against integral experimental data from Refs. [4.29, 4.30]. Calculated average cross sections for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are compared with the IRDF-2002 and experimental data in Table 4.2. Data calculated from the re-evaluated excitation functions for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra agree well with the experimental data, while discrepancies exist between the IRDF-2002 and experimental data of about 6.4% and 3.2% for the ^{235}U and ^{252}Cf spectra, respectively.

TABLE 4.2. CALCULATED AND MEASURED AVERAGED CROSS SECTIONS FOR THE $^{32}S(n,p)^{32}P$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	**Average cross section, mb**		**C/E [4.30]**
	Calculated	**Measured**	
^{235}U thermal fission neutron spectrum	68.195 [A]	69.080 ± 1.361 [4.30]	0.9872
	64.501 [B]		0.9337
^{252}Cf spontaneous fission neutron spectrum	74.106 [A]	73.240 ± 2.695 [4.29]	1.0216
	70.230 [B]	72.540 ± 2.532 [4.30]	0.9682

[A] present evaluation.
[B] IRDF-2002 (IRDF-90 version 2).

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5. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE ^{60}Ni(n,p)$^{60m+g}$Co REACTION

The isotopic abundance of ^{60}Ni in natural nickel is 26.2231 ± 0.0077 atom percent, and the ^{60}Co obtained via the (n,p) reaction undergoes 100% β^- decay with a half-life of (1925.28 ± 0.28) days. Two of the most intensive gamma rays at 1173.228 keV (I_γ = 0.9985 ± 0.0003) and 1332.492 keV (I_γ = 0.999826 ± 0.000006) are normally used to determine the ^{60}Ni(n,p)$^{60m+g}$Co reaction rate. Recommended decay data for the half-life, and beta and gamma ray emission probabilities per decay of ^{60}Co were taken from Ref. [2.7] of Section 2.

Microscopic experimental data were analyzed during the preparation of the input database assembled in order to evaluate the cross sections and uncertainties for the ^{60}Ni(n,p)$^{60m+g}$Co reaction [5.1-5.32]. During this procedure, various experimental data [5.3, 5.4, 5.6-5.16, 5.18-5.30, 5.33, 5.35-5.38, 5.40-5.44, 5.46] were corrected on the basis of the newly recommended cross-section data for the relevant monitor reactions and the recommended decay data (see Table 2.1).

Other adjustments were also applied to some of the experimental data of Refs. [5.2-5.6, 5.8, 5.9, 5.11-5.13, 5.17, 5.19-5.22, 5.24, 5.25]. Thus, the data of Vonach *et al.* [5.12] were renormalized to the experimental data of Wagner *et al.* [5.14] in the overlapping energy range from 7.71 to 12.00 MeV; prior to the introduction of the new cross-sections standards, the data from Ref. [5.12] were multiplied by a factor Fc = 1.11621. Experimental data in Refs. [5.2-5.6, 5.8, 5.9, 5.11, 5.13, 5.17, 5.19-5.22, 5.24, 5.25] were obtained from Ni samples of natural isotopic composition, and therefore were corrected for the contribution from the ^{61}Ni(n,x)$^{60m+g}$Co reaction with the relevant cross-section data being taken from Ref. [5.33]. Whereas the corrected data of Wang Yongchang *et al.* [5.13] were renormalized to the integral of the experimental data of Konno *et al.* [5.18] in the overlapping neutron energy range from 13.64 to 14.80 MeV, which resulted in a decrease in the cross sections by a factor of Fc = 0.87085 .

The data of Viennot *et al.* [5.15] at 14.83 MeV and Huang Xiaolong *et al.* [5.23] at 7.07 and 11.40 MeV were not included in the evaluation due to their significant underestimation and overestimation of the cross sections of the ^{60}Ni(n,p)$^{60m+g}$Co reaction, respectively. Experimental cross-section data in Refs. [5.26-5.32] were also rejected because of their large deviation from the main bulk of experimental data [5.1-5.25]. Furthermore, only one experimental value in the energy range from 14 to 15 MeV was reported in Refs. [5.26, 5.27, 5.30-5.32].

The excitation function for the ^{60}Ni(n,p)$^{60m+g}$Co reaction in the energy range from threshold to 21 MeV was evaluated by means of statistical analyses of the experimental cross-section data [5.1-5.25] by means of the PADE-2 code. Uncertainties in the evaluated excitation function for the ^{60}Ni(n,p)$^{60m+g}$Co reaction are given in the form of a relative covariance matrix for 35-neutron energy groups (LB = 5). Covariance matrix uncertainties were calculated simultaneously with the recommended cross-section data by means of the PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix in File-33 are as follows:

1.73704E-06	1.79002E-06	1.87221E-06	2.00249E-06
2.15991E-06	2.36667E-06	2.64302E-06	2.91385E-06
3.36594E-06	3.74685E-06	4.38254E-06	5.03985E-06
5.86891E-06	6.97586E-06	8.17859E-06	9.73493E-06
1.19698E-05	1.34917E-05	1.69430E-05	2.02914E-05
2.22877E-05	2.63835E-05	3.10218E-05	4.13754E-05
9.80006E-05	2.64611E-04	5.55947E-04	8.39901E-04
1.15624E-03	2.58009E-03	2.99923E-03	6.33064E-03
1.45176E-02	2.30044E-02	1.35417E-01	

Evaluated group cross sections and their uncertainties for the excitation function of the ^{60}Ni(n,p)$^{60m+g}$Co reaction are given in Table 5.1. Boundaries for the neutron energy groups are the same as in File-33.

TABLE 5.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE ^{60}Ni(n,p)$^{60m+g}$Co REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 21 MeV.

Neutron energy (MeV) from to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from to**	**Cross section (mb)**	**Uncer-tainty (%)**
2.076 – 4.000	0.086	34.76	12.500 – 13.000	157.323	1.50
4.000 – 4.500	1.761	8.70	13.000 – 13.500	157.423	1.30
4.500 – 5.000	6.030	6.86	13.500 – 14.000	153.227	1.13
5.000 – 5.500	12.848	4.59	14.000 – 14.500	144.807	1.04
5.500 – 6.000	21.780	2.84	14.500 – 15.000	133.006	1.07
6.000 – 6.500	32.165	2.05	15.000 – 15.500	119.219	1.29
6.500 – 7.000	43.323	1.91	15.500 – 16.000	104.944	1.74
7.000 – 7.500	54.714	1.86	16.000 – 16.500	91.392	2.40
7.500 – 8.000	66.006	1.78	16.500 – 17.000	79.309	3.16
8.000 – 8.500	77.057	1.75	17.000 – 17.500	69.008	3.89
8.500 – 9.000	87.850	1.82	17.500 – 18.000	60.494	4.49
9.000 – 9.500	98.433	1.94	18.000 – 18.500	53.603	4.94
9.500 – 10.000	108.860	2.05	18.500 – 19.000	48.104	5.29
10.000 – 10.500	119.128	2.09	19.000 – 19.500	43.755	5.69
10.500 – 11.000	129.127	2.05	19.500 – 20.000	40.337	6.30
11.000 – 11.500	138.574	1.96	20.000 – 20.500	37.663	7.26
11.500 – 12.000	146.958	1.83	20.500 – 21.000	35.578	8.58
12.000 – 12.500	153.524	1.67			

The lowest uncertainties of 1.04% to 1.50% in the evaluated cross sections are observed in the neutron energy range from 12.5 to 15.5 MeV, while the large uncertainty of 34.76 % from threshold to 4.00 MeV is caused by the significant discrepancies in the experimental data. Over neutron energies from 5.50 to 16.5 MeV, uncertainties in the cross sections vary between 1.04% and 2.84%, but these values increase dramatically from 3.16% to 8.58% in the neutron energy range from 16.5 to 21 MeV due to the inadequate experimental data.

Fig. 5.1 shows the re-evaluated excitation function for the ^{60}Ni(n,p)$^{60m+g}$Co reaction over the neutron energy range from 0.5 to 21.0 MeV compared with the equivalent cross sections in

IRDF-2002 and the experimental data. The IRDF-2002 evaluation was prepared in 1989, and did not take into account newer experimental measurements [5.14-5.25]. Therefore, the IRDF-2002 study underestimates the ^{60}Ni(n,p)$^{60m+g}$Co reaction cross sections systematically in the energy range from 6 to 14 MeV. Above 16 MeV, the IRDF-2002 evaluation overestimates the cross-section values in comparison with the new re-evaluated data. Below neutron energies of 6 MeV and between 15.0 and 16.0 MeV, both evaluations are in good agreement, although IRDF-2002 recommendations are underestimations from 6 to 14 MeV and overestimations above 16 MeV.

Integral experimental data for the ^{60}Ni(n,p)$^{60m+g}$Co reaction are given in Refs. [5.34-5.36]. Two experiments were carried out in neutron fields similar to the ^{235}U thermal fission neutron spectrum [5.34, 5.35], while one experiment was performed in a ^{252}Cf spontaneous fission neutron spectrum [5.36]. An integral cross section of (2.39 ± 0.13) mb for the ^{252}Cf spontaneous fission neutron spectrum as evaluated by Mannhart has been defined as raw data [5.36], and therefore was not used to test the microscopic data.

Newly recommended cross section of (0.7007 ± 0.0090) mb for the ^{27}Al(n,α)^{24}Na monitor reaction in a ^{235}U thermal fission neutron spectrum has been used to correct available integral measurements. After correction, the highest measured cross section for the ^{60}Ni(n,p)$^{60m+g}$Co reaction in a ^{235}U thermal fission neutron spectrum was determined to be (2.336 ± 0.187) mb by Rochlin [5.34]. Horibe and Chatani used an enriched ^{235}U fission plate converter in their measurements [5.35]. We adopted an averaged cross-section standard of $\langle\sigma\rangle_{U-235}$ = (2.180 ± 0.104) mb from the measured value of Horibe and Chatani [5.35] and corrected on the basis of a new ^{27}Al(n,α)^{24}Na monitor reaction cross section.

The results of tests with the re-evaluated excitation function for the ^{60}Ni(n,p)$^{60m+g}$Co reaction are given in Table 5.2. These data show that the integral cross sections calculated from the re-evaluated excitation function for the ^{235}U thermal fission neutron spectrum agree well with the experimental data, while the equivalent data calculated from IRDF-2002 are significantly discrepant.

TABLE 5.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR THE ^{60}Ni(n,p)$^{60m+g}$Co REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Average cross section, mb		C/E
	Calculated	Measured	
^{235}U thermal fission neutron spectrum	2.1810 [A] 1.9360 [B]	2.180 ± 0.104 [5.35]	1.0005 0.8881
^{252}Cf spontaneous fission neutron spectrum	2.7988 [A] 2.4945 [B]		

[A] present evaluation.
[B] IRDF-2002 (IRDF-90 version 2).

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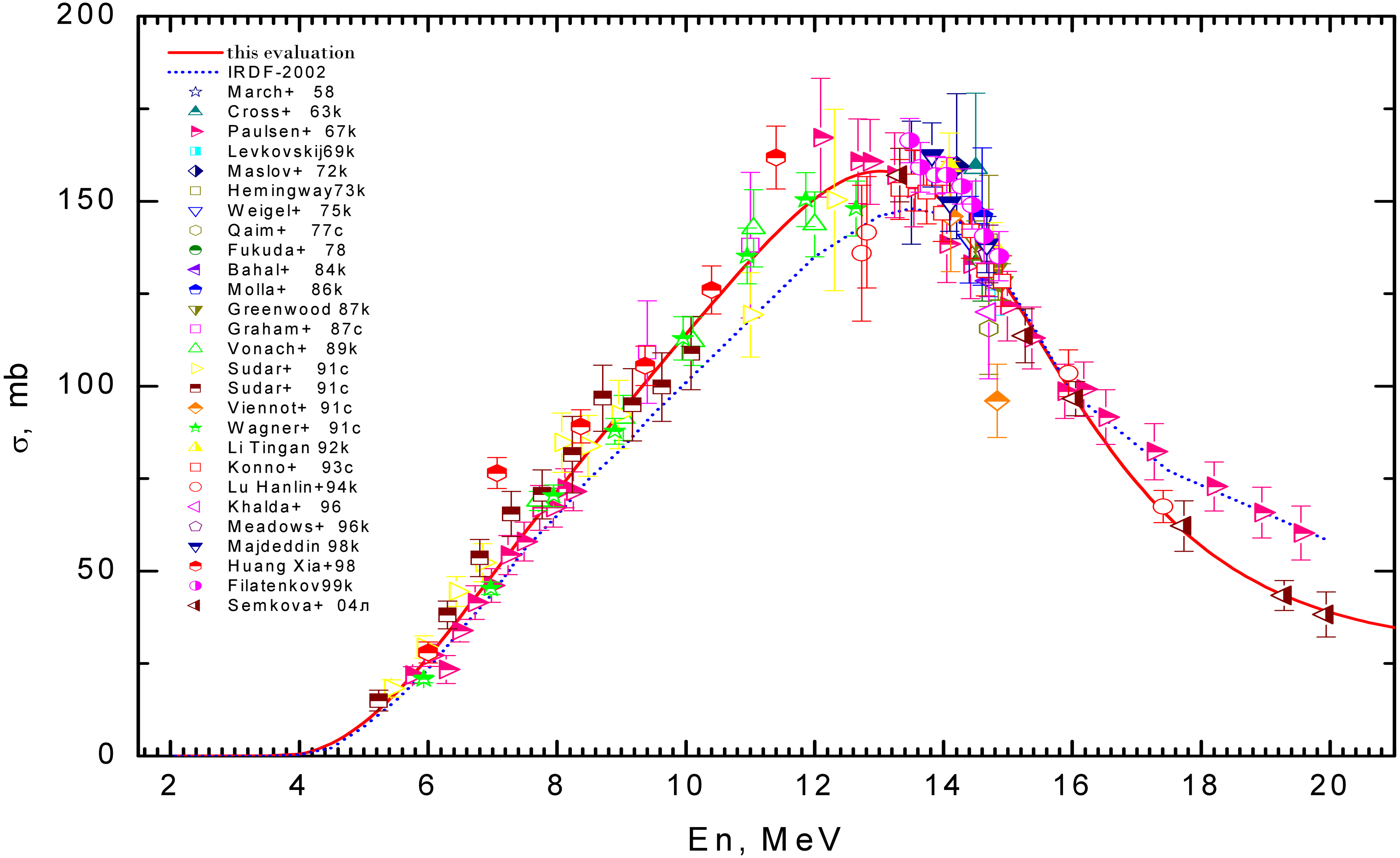

FIG. 5.1. Re-evaluated excitation function of the $^{60}Ni(n,p)^{60m+g}Co$ reaction in the energy range from threshold to 21 MeV in comparison with IRDF-2002 and experimental data.

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6. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE $^{63}Cu(n,2n)^{62}Cu$ REACTION

The ^{63}Cu isotopic abundance in natural copper is 69.17 ± 0.03 atom percent, and the ^{62}Cu obtained via the (n,2n) reaction undergoes 100% β^+ decay with a half-life of (9.73 ± 0.03) minutes. 511-keV annihilation gamma radiation (I_γ = 1.9486 ± 0.0005) can be used to determine the $^{63}Cu(n,2n)^{62}Cu$ reaction rate. Recommended decay data for the half-life, and beta+ and gamma-ray emission probabilities per decay of ^{62}Cu were taken from Ref. [2.7] of Section 2.

Microscopic experimental data were analyzed during the preparation of the input database assembled in order to evaluate the cross sections and uncertainties for the $^{63}Cu(n,2n)^{62}Cu$ reaction [6.1-6.41]. During this procedure, various experimental data in Refs. [6.1, 6.2, 6.9, 6.10, 6.12, 6.16, 6.17, 6.21, 6.22, 6.25-6.30, 6.32, 6.35, 6.37-6.39, 6.41, 6.42] were corrected in terms of the newly recommended cross-section data for the monitor reactions used in the measurements and the recommended decay data (see Table 2.1).

Careful analysis of the experimental cross-section data for the $^{63}Cu(n,2n)^{62}Cu$ reaction between 13 and 15 MeV show that the most reliable data were measured by De Juren and Stooksberry [6.5], Sakisaka *et al*. [6.6], Glover and Weigold [6.9], Grimeland *et al*. [6.15], Qaim [6.21], Valkonen [6.23], Ryves *et al*. [6.26], Ghanbari and Robertson [6.27], and Pansare and Bhoraskar [6.28]. More recent experimental data of Sakane *et al*. [6.29] and Mannhart and Schmidt [6.30] agree within their uncertainties.

Results of the relative measurements of Koehler and Alford [6.11], Rayburn [6.12], Csikai [6.14] and Cuzzocrea *et al*. [6.19] were corrected to the $^{63}Cu(n,2n)^{62}Cu$ cross section obtained from a preliminary evaluation carried out on the most reliable experimental data [6.5, 6.6, 6.9, 6.15, 6.21, 6.26-6.30]. These preliminary results were also used to correct the experimental data of Refs. [6.16, 6.25]. The data of Liskien and Paulsen [6.16] were renormalized to the integral cross section over the energy range from 13.5 to 15.5 MeV, and involved the application of a multiplication factor of Fc = 0.81256 derived from the new cross-section standards. Similarly, the cross-section data of Jarjis [6.25] over an energy range of 12.92 to 14.91 MeV were renormalized to the preliminary evaluated integral cross section between 13.86 and 14.77 MeV, based on a multiplication factor of Fc = 0.98690.

Above a neutron energy of 15 MeV, the most representative experimental data are those of Ryves *et al*. [6.26]. Cross sections given in Ref. [6.26] were obtained in two sets of measurements carried out with different monitor reactions. The results of these measurements agree well over the investigated neutron energy range of 14.67 to 18.95 MeV.

Experimental data of Brolley *et al*. [6.2] and Jarjis [6.25] obtained by means of a Dynamitron accelerator in the energy range from 13.24 to 15.94 MeV were renormalized to the integral cross section calculated from the experimental data of Ryves *et al*. [6.26] in the overlapping energy ranges; total correction factors for these experimental data were Fc = 0.85715 and 0.89123, respectively.

Total uncertainties of 3.9% and 5% were assigned to the cross-section data from Refs. [6.14] and [6.17], respectively, while systematic uncertainties were evaluated and added to the experimental data in Refs. [6.6, 6.21, 6.22, 6.27]. Experimental data for the $^{63}Cu(n,2n)^{62}Cu$ reaction as given in Ref. [6.12] were partially adopted - data above 15 MeV were rejected due to a systematic underestimation of the cross sections. Cross-section data given in Refs. [6.31-6.43] were rejected completely due to their large deviations from the bulk of experimental data – rejected experimental data in Refs. [6.33, 6.39, 6.41. 6.42] had only been measured at one or two energy points from 14 to 15 MeV.

The excitation function for the $^{63}Cu(n,2n)^{62}Cu$ reaction in the energy region from threshold to 20 MeV was evaluated by means of statistical analyses of the experimental cross-section data [6.1-6.30]. Uncertainties in the evaluated excitation function for the $^{63}Cu(n,2n)^{62}Cu$ reaction are given in the form of a relative covariance matrix for 33-neutron energy groups (LB=5). Covariance matrix uncertainties were calculated simultaneously with the recommended cross-section data by means of the PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix in File-33 are as follows:

7.42042E-07	7.59925E-07	7.90851E-07	8.38334E-07
8.81284E-07	9.11081E-07	9.45308E-07	9.84989E-07
1.03138E-06	1.08588E-06	1.15052E-06	1.22803E-06
1.32202E-06	1.43773E-06	1.58233E-06	1.76682E-06
2.00726E-06	2.32906E-06	2.77323E-06	3.40949E-06
4.36207E-06	5.86939E-06	8.41904E-06	1.31240E-05
2.28475E-05	4.64323E-05	1.54121E-04	3.25466E-04
1.37825E-03	1.66307E-03	4.12994E-03	4.93817E-03
2.03005E-02			

All of these eigenvalues are positive.

Evaluated group cross sections and their uncertainties for the $^{63}Cu(n,2n)^{62}Cu$ reaction are listed in Table 6.1. Group boundaries are the same as in File-33. These data show that the smallest uncertainties in the evaluated cross sections of 1.14% to 1.47% are observed in the neutron energy range from 13.8 to 16.0 MeV. A significant uncertainty of 13.8% in the cross sections from threshold to 11.6 MeV arises from the large uncertainties in the experimental data within this region and discrepancies between these experimental data. The evaluated excitation function for the $^{63}Cu(n,2n)^{62}Cu$ reaction in the energy range 13.8 to 16.0 MeV could be recommended as reference cross-section data for activation measurements with short-lived nuclei.

TABLE 6.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{63}Cu(n,2n)^{62}Cu$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 20 MeV.

Neutron energy (MeV) from to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from to**	**Cross section (mb)**	**Uncer-tainty (%)**
11.000 - 11.600	6.531	13.77	15.000 - 15.200	583.632	1.22
11.600 - 12.000	41.444	3.26	15.200 - 15.400	603.779	1.28
12.000 - 12.200	82.501	3.32	15.400 - 15.600	621.877	1.34
12.200 - 12.400	116.588	3.08	15.600 - 15.800	638.098	1.40
12.400 - 12.600	154.085	2.81	15.800 - 16.000	652.681	1.47
12.600 - 12.800	193.680	2.57	16.000 - 16.200	665.845	1.53
12.800 - 13.000	234.178	2.36	16.200 - 16.400	677.791	1.59
13.000 - 13.200	274.579	2.16	16.400 - 16.600	688.697	1.65
13.200 - 13.400	314.094	1.97	16.600 - 16.800	698.720	1.71
13.400 - 13.600	352.159	1.78	16.800 - 17.000	707.998	1.77
13.600 - 13.800	388.391	1.59	17.000 - 17.500	722.673	1.89
13.800 - 14.000	422.568	1.42	17.500 - 18.000	741.368	2.09
14.000 - 14.200	454.590	1.29	18.000 - 18.500	757.824	2.32
14.200 - 14.400	484.442	1.20	18.500 - 19.000	772.085	2.58
14.400 - 14.600	512.174	1.15	19.000 - 19.500	783.364	2.89
14.600 - 14.800	537.875	1.14	19.500 - 20.000	789.948	3.26
14.800 - 15.000	561.652	1.17			

Figs 6.1 and 6.2 show the re-evaluated excitation function for the $^{63}Cu(n,2n)^{62}Cu$ reaction over the neutron energy range from 11.0 to 20.0 MeV and 13.5 to 15.50 MeV respectively, and compare these data with the equivalent cross sections of IRDF-2002 and experimental data. Evaluated cross sections and rejected experimental data are shown in Fig. 6.3. As illustrated in Figs. 6.1-6.3 (except for the neutron energy range from 13.85 to 14.45 MeV), the IRDF-2002 evaluation overestimated the cross-section values systematically in comparison with the newly re-evaluated data. However, over the neutron energy range from 13.85 to 14.45 MeV, both evaluations agreed very well.

Integral experiments for the $^{63}Cu(n,2n)^{62}Cu$ reaction are described in Refs. [6.44-6.48]. Two experiments were carried out in neutron fields with similar spectra to the ^{235}U thermal fission neutron spectrum [6.45, 6.46], and two experiments were performed in a ^{252}Cf spontaneous fission neutron spectrum [6.47, 6.48]. Experimental data obtained for ^{235}U thermal fission neutron spectrum and ^{252}Cf spontaneous fission neutron spectrum were corrected with respect to the newly recommended cross sections for the monitor reactions and decay data.

Cohen [6.44] undertook experiments in which $^{63}Cu(n,2n)^{62}Cu$ cross sections were measured in a Be(d,n) neutron spectrum that spanned an energy range from 2.9 to 18 MeV. Unfortunately, these experimental data cannot be used to verify the evaluated microscopic cross sections due to the absence of well-determined neutron spectra.

Using a cavity technique in a heavy-water reactor, Grundl measured the ratio of the $^{238}U(n,f)/$ $^{63}Cu(n,2n)^{62}Cu$ cross sections [6.45]. The thermal-induced ^{235}U fission neutron spectrum in this facility covered an energy range from 0.8 to 16 MeV, and a cross section of (0.1172 ± 0.0077) mb was determined from the measured ratio. Kobayashi *et al*. measured the average (n,2n) cross section for ^{63}Cu in the centre of the YAYOI fast reactor core [6.46], which is representative of fission neutrons above approximately 2 MeV. When corrected with respect to the new standards, the average cross section obtained in this experiment was (0.1241 ± 0.0171) mb, while the average weighted value from experimental data of Refs. [6.45, 6.46] is equal to (0.1185 ± 0.0070) mb. This average cross section agrees well with the value evaluated by Mannhart of (0.1184 ± 0.0070) mb [6.49].

Experimental data obtained in a ^{252}Cf spontaneous fission neutron spectrum by Dezsö and Csikai [6.47] of (0.294 ± 0.027) mb and by Mannhart [6.48] of (0.1841 ± 0.0070) mb differ by about 60%. The average cross section determined by Dezsö and Csikai was not taken into account in the benchmark calculations because such a large value can not be suitably derived from representative microscopic experimental data. Mannhart recommended an average cross section for the ^{252}Cf spontaneous fission neutron spectrum of (0.1844 ± 0.0073) mb [6.49], which was adopted in this work. Evaluated experimental data from Ref. [6.49] for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were used in the benchmark calculations. Tests were made on the data as given in Table 6.2, where C/E is the ratio of the calculated to experimental cross sections.

The C/E values show significant discrepancies between the experimental data for ^{235}U thermal fission neutron spectrum, the integral cross sections calculated from IRDF-2002 and the re-evaluated excitation functions. Discrepancies between the experimental and calculated data for ^{252}Cf spontaneous fission neutron spectrum are lower but also significant. The lowest discrepancy of 7.6% between the calculated and experimental data is obtained for the ^{252}Cf spontaneous fission neutron spectrum calculations in which the microscopic cross sections from this work were adopted.

TABLE 6.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR THE ^{63}Cu(n,2n)^{62}Cu REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Average cross section, mb		C/E
	Calculated	Measured	
^{235}U thermal fission neutron spectrum	0.090607 [A] 0.094232 [B]	0.1184 ± 0.0070 [6.49]	0.7640 0.7945
^{252}Cf spontaneous fission neutron spectrum	0.19843 [A] 0.20569 [B]	0.1844 ± 0.0073 [6.49]	1.0761 1.1209

[A] present evaluation.
[B] IRDF-2002 (ENDF/B-VI release 8).

The 90% response range of the excitation function of the ^{63}Cu(n,2n)^{62}Cu reaction is practically the same for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra: 11.9 to 16.80 MeV, and 12.0 to 17.3 MeV, respectively. Therefore, the results of the benchmark calculations carried out for the ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are highly contradictory.

Re-evaluation of the ^{63}Cu(n,2n)^{62}Cu excitation function was performed on the basis of old and recent experimental data corrected to the new standards. All microscopic experimental data included in the database agree within their uncertainties, implying that the ^{63}Cu(n,2n)^{62}Cu integral cross section as measured for the ^{235}U thermal fission neutron spectrum is significant overestimated, while the spectrum-averaged cross section measured for the ^{252}Cf spontaneous fission neutron spectrum is underestimated by 5% to 7%. This proposition is supported by an investigation carried out by Mannhart [6.49]. Using the evaluated integral experimental data in the ^{252}Cf spontaneous fission and the ^{235}U thermal fission neutron fields for 29 reactions, Mannhart derived the ratio $<\sigma>_{Cf-252}/<\sigma>_{U-235}$ as a function of the mean energy of the 50% reaction response (E(50%)). As predicted from this evaluated curve, the $<\sigma>_{Cf-252}/<\sigma>_{U-235}$, ratio for the ^{63}Cu(n,2n)^{62}Cu reaction was estimated to be 1.92, compared with a value obtained directly from the experimental data of 1.56 [6.49]. The $<\sigma>_{Cf-252}/<\sigma>_{U-235}$ values calculated from the newly re-evaluated excitation function and IRDF-2002 data are very similar at 2.19 and 2.18, respectively.

New precise measurements of the integral cross sections for the ^{63}Cu(n,2n)^{62}Cu reaction are required to solve the problems outlined above with respect to the ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra.

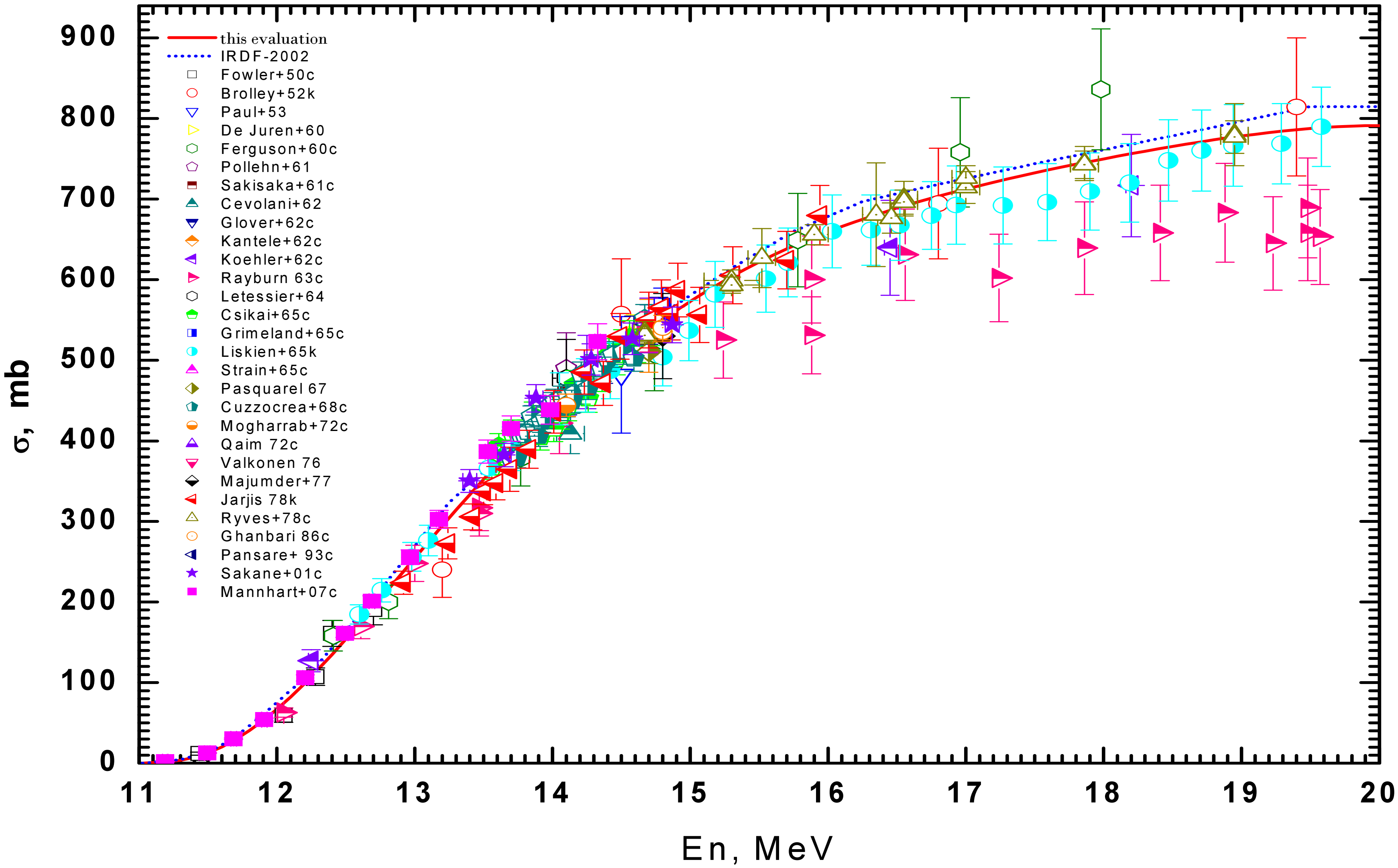

FIG. 6.1. Re-evaluated excitation function of the $^{63}Cu(n,2n)^{62}Cu$ reaction in the energy range from threshold to 20 MeV in comparison with IRDF-2002 and experimental data.

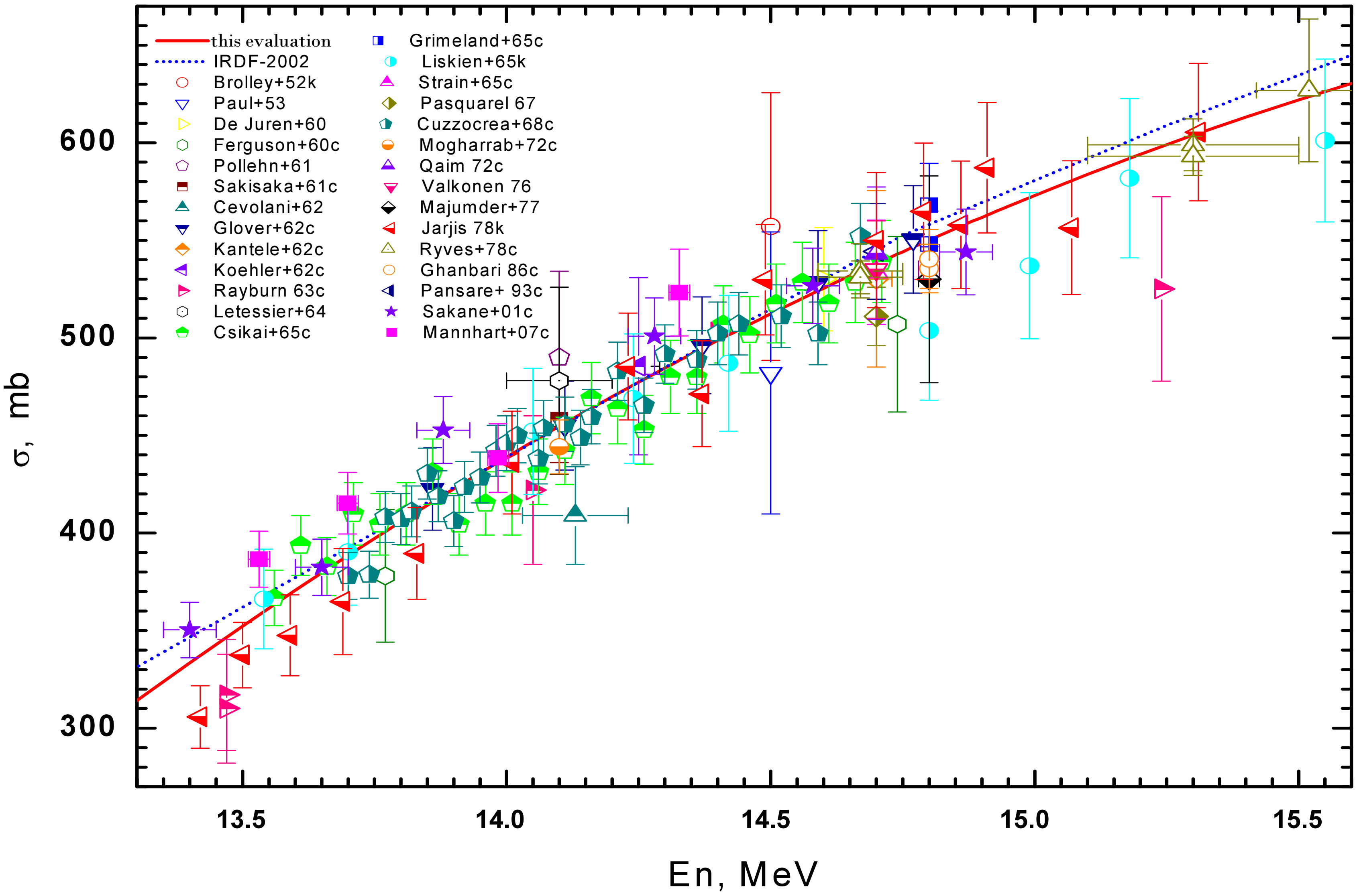

FIG. 6.2. Re-evaluated excitation function of the $^{63}Cu(n,2n)^{62}Cu$ reaction in the energy range from 13.5 to 15.5 MeV in comparison with IRDF-2002 and experimental data.

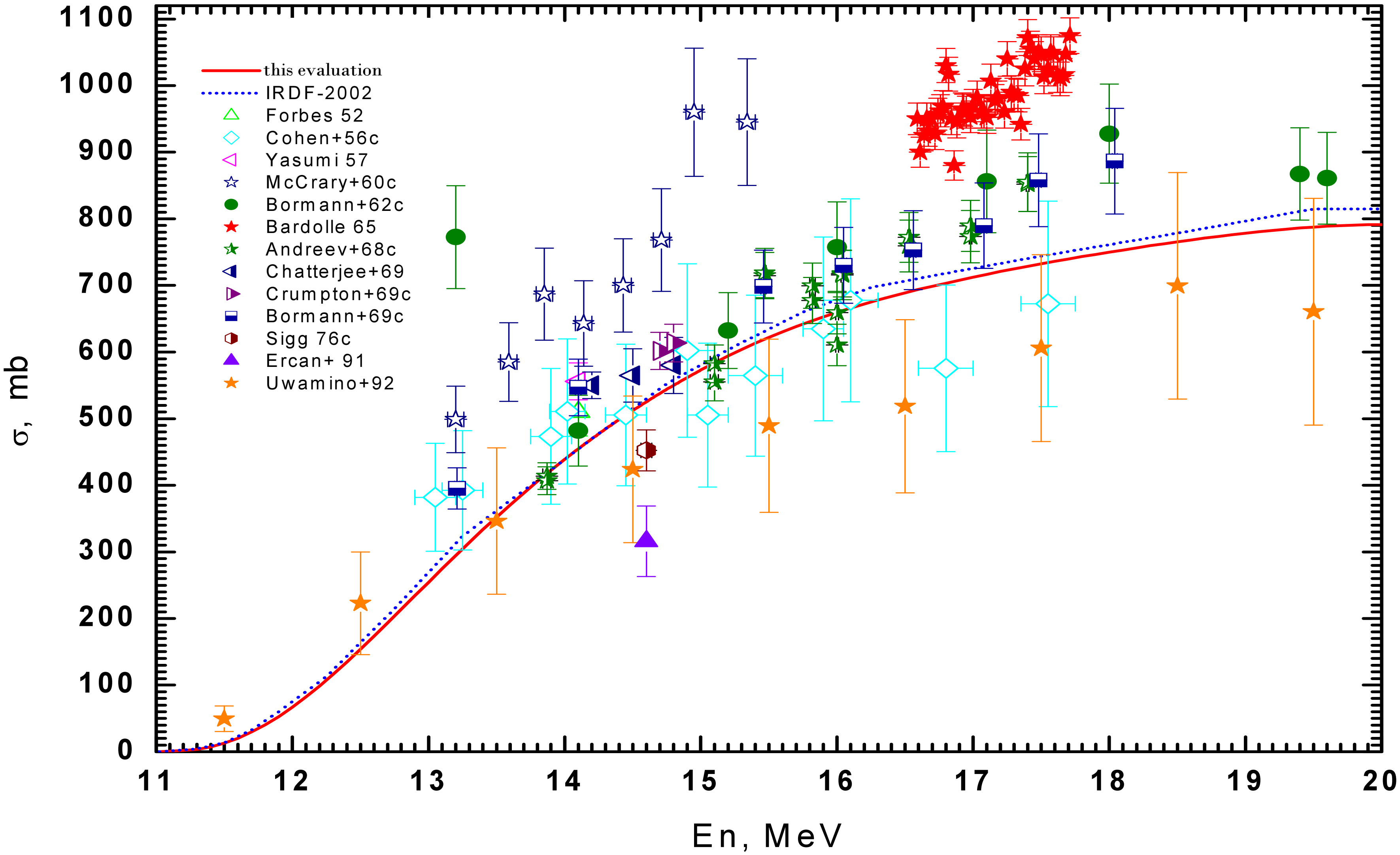

FIG. 6.3. Re-evaluated excitation function of the $^{63}Cu(n,2n)^{62}Cu$ reaction in the energy range from threshold to 20 MeV in comparison with IRDF-2002 and rejected experimental data.

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7. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE $^{65}Cu(n,2n)^{64}Cu$ REACTION

The abundance of the ^{65}Cu isotope in natural copper is 30.83 ± 0.03 atom percent, and ^{64}Cu obtained via the (n,2n) reaction undergoes (61.0 ± 0.3)% electron-capture and (39.0 ± 0.3)% β^- decay with a half-life of (12.700 ± 0.002) hours. 511-keV gamma annihilation radiation (I_γ = 0.348 ± 0.004) is frequently used to determine the $^{64}Cu(n,2n)^{62}Cu$ reaction rate. Recommended decay data for the half-life, and beta and gamma-ray emission probabilities per decay of ^{64}Cu were taken from Ref. [2.7] of Section 2.

Microscopic experimental data were analyzed in the preparation of the input database for the evaluation of the cross sections and uncertainties of the $^{65}Cu(n,2n)^{64}Cu$ reaction [7.1-7.48]. During this procedure, the experimental data of Refs. [7.3, 7.4, 7.6-7.8, 7.10-7.18, 7.20-7.40, 7.43-7.48] were corrected with respect to the newly recommended cross-section standards and decay data (see Table 2.1).

Careful analysis of the experimental cross-section data for the $^{65}Cu(n,2n)^{64}Cu$ reaction between 13 and 15 MeV indicates that the most reliable data in this energy range have been measured by Mannhart and Vonach [7.25], Ryves *et al*. [7.27], Ghanbari and Robertson [7.31], Meadows *et al*. [7.32], Filatenkov *et al*. [7.37], and Mannhart and Schmidt [7.40]. Prestwood and Bayhurst measured relative $^{27}Al(n,\alpha)^{24}Na$ reaction cross sections [7.4], and the experimental data of Refs. [7.7, 7.10, 7.21, 7.22, 7.24, 7.30] are in good agreement with these data after correction with respect to the new standards.

The precise relative measurements of Vonach *et al*. for incident neutron energies of 13.6 to 14.7 MeV were normalized to the absolute cross-section value of 955 mb at 14.70 MeV [7.15]. Re-normalization of the cross-section data of Cuzzocrea *et al*. over an energy range of 13.70 to 14.67 MeV resulted in a re-adjustment of the absolute cross section from 919 to 900.8 mb at 14.11 MeV [7.16]. The $^{65}Cu(n,2n)^{64}Cu$ cross sections at 14.11 and 14.70 MeV were determined from initial evaluations carried out on the basis of the precise experimental data reported in Refs. [7.25, 7.27, 7.31, 7.32, 7.37, 7.40].

Special corrections were applied to the data reported in Refs. [7.2, 7.4, 7.11, 7.12, 7.28, 7.29, 7.33, 7.36]. The experimental data of McCrary and Morgan [7.2], Paulsen and Liskien [7.11], Santry and Butler [7.12], Csikai [7.28], Winkler and Ryves [7.29], Ikeda *et al*. [7.33] and Molla *et al*. [7.36] were re-normalized to the integral cross section calculated from the experimental data of Filatenkov and Chuvaev [7.37] in the overlapping energy ranges. All of the experimental data in Ref. [7.2] were corrected on the basis of a factor of Fc = 0.61795. Corrections made to the data in Refs. [7.11, 7.12, 7.28, 7.29, 7.33, 7.36] in terms of the newly recommended cross-section standards and decay data for ^{64}Cu involved the factors 0.89950, 0.95979, 0.93453, 1.02965, 0.96037 and 0.94159, respectively. A total uncertainty of 8.3% was assigned to the cross-section data of Rayburn [7.6] by taking into account the similar experimental method used in Ref. [7.44]. Systematic uncertainties were evaluated and added to the experimental data of Refs. [7.7, 7.18, 7.24, 7.29], and uncertainties in the standard cross sections were added to the experimental data of Refs. [7.4, 7.12].

At neutron energies above 15 MeV, the most representative experimental data are identified with the studies of Ryves *et al*. [7.27], in which cross sections were obtained in two series of measurements carried out with different monitor reactions. The results of these overlapping studies agree well in the energy range from 14.67 to 18.95 MeV. Data from the measurements of Prestwood and Bayhurst [7.4] were derived on the basis of the $^{238}U(n,f)$ reaction, and were renormalized at 16.5 MeV to a cross section of 1072.6 mb determined from the experimental data of Ryves *et al*. [7.27].

Experimental data given in Refs. [7.6, 7.8] were partially used – data above 16 MeV were rejected due to the systematic underestimation of the cross sections. Cross-section data from Refs. [7.41-7.48] were rejected completely due to their large discrepancies when compared with the bulk of other experimental data; furthermore, cross-section data from Refs. [7.41-7.44, 7.47, 7.48] were only measured at one energy point from 14 to 15 MeV.

The excitation function for the $^{65}Cu(n,2n)^{64}Cu$ reaction in the energy range from threshold to 20 MeV was evaluated by means of a comprehensive statistical analysis of the experimental cross-section data [7.1-7.40]. Uncertainties in the evaluated excitation function are given in the form of a relative covariance matrix for 35-neutron energy groups (LB = 5). This covariance matrix was calculated simultaneously with the recommended cross-section data by means of the PADE-2 code.

The resulting six-digit eigenvalues for the relative covariance matrix of File-33 are as follows:

2.04728E-06	2.06457E-06	2.10040E-06	2.13452E-06
2.20189E-06	2.27085E-06	2.32645E-06	2.39031E-06
2.46534E-06	2.55291E-06	2.65581E-06	2.77928E-06
2.92594E-06	3.10543E-06	3.32520E-06	3.59791E-06
3.94655E-06	4.39463E-06	4.99206E-06	5.80794E-06
6.95579E-06	8.65847E-06	1.12921E-05	1.56717E-05
2.36510E-05	3.97544E-05	7.86225E-05	2.14266E-04
6.08499E-04	1.59686E-03	2.22254E-03	2.73248E-03
6.42193E-03	9.57466E-03	2.09712E-02	

Evaluated group cross sections and their uncertainties for the $^{65}Cu(n,2n)^{64}Cu$ reaction are listed in Table 7.1. Group boundaries are the same as in File-33. While the lowest uncertainties in the evaluated cross sections of 1.48% to 1.75% are observed in the neutron energy range from 13.4 to 15.6 MeV, the most significant uncertainty of 13.73% occurs from threshold to 10.5 MeV due to the large uncertainties and discrepancies between experimental data in this region. The evaluated excitation function in the energy range from 13.5 to 15.5 MeV could be recommended as reference cross-section data for activation measurements.

The re-evaluated excitation function for the $^{65}Cu(n,2n)^{64}Cu$ reaction is shown in Fig. 7.1 (neutron energy range from 10.0 to 20.0 MeV) and Fig. 7.2 (neutron energy range from 13.5 to 15.50 MeV) in comparison with the equivalent data from IRDF-2002 and experimental data; Fig. 7.3 shows the excitation functions and all rejected experimental data.

Only one integral experiment is known to have been performed for the $^{65}Cu(n,2n)^{64}Cu$ reaction in which Mannhart measured the integral cross section in a ^{252}Cf spontaneous fission neutron spectrum [7.49]. A revised experimental value was published in Ref. [7.50] against which the re-evaluated excitation function for the $^{65}Cu(n,2n)^{64}Cu$ reaction has been tested. Calculated average cross sections for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were compared with the experimental data, and the resulting C/E ratios are listed in Table 7.2. The average cross section calculated from the re-evaluated excitation function for the ^{252}Cf spontaneous fission neutron spectrum agrees very well with the related experimental data of

Mannhart [7.50] (C/E = 0.9942), while the integral cross section calculated from the equivalent IRDF-2002 data is approximately 3% higher than the measured value.

TABLE 7.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{65}Cu(n,2n)^{64}Cu$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 20 MeV.

Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)	Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)
10.100 - 10.500	12.094	13.73	14.200 - 14.400	914.059	1.48
10.500 - 11.000	78.969	4.35	14.400 - 14.600	936.070	1.48
11.000 - 11.200	163.604	3.54	14.600 - 14.800	956.110	1.50
11.200 - 11.400	222.221	3.29	14.800 - 15.000	974.162	1.54
11.400 - 11.600	284.205	3.14	15.000 - 15.200	990.544	1.59
11.600 - 11.800	347.588	3.04	15.200 - 15.400	1005.409	1.66
11.800 - 12.000	410.378	2.97	15.400 - 15.600	1018.740	1.75
12.000 - 12.200	471.345	2.87	15.600 - 15.800	1030.861	1.85
12.200 - 12.400	529.521	2.74	15.800 - 16.000	1041.662	1.98
12.400 - 12.600	584.147	2.57	16.000 - 16.500	1057.755	2.23
12.600 - 12.800	635.094	2.39	16.500 - 17.000	1075.662	2.62
12.800 - 13.000	682.044	2.20	17.000 - 17.500	1087.633	3.01
13.000 - 13.200	725.181	2.01	17.500 - 18.000	1093.585	3.33
13.200 - 13.400	764.650	1.85	18.000 - 18.500	1092.641	3.59
13.400 - 13.600	800.510	1.72	18.500 - 19.000	1082.611	3.91
13.600 - 13.800	833.207	1.61	19.000 - 19.500	1058.630	4.46
13.800 - 14.000	862.789	1.54	19.500 - 20.000	1009.103	4.73
14.000 - 14.200	889.659	1.50			

TABLE 7.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR THE $^{65}Cu(n,2n)^{64}Cu$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Average cross section, mb		C/E
	Calculated	Measured	
^{235}U thermal fission neutron spectrum	0.33339 [A] 0.34650 [B]		
^{252}Cf spontaneous fission neutron spectrum	0.65436 [A] 0.67793 [B]	0.6582 ± 0.0146 [7.50]	0.9942 1.0300

[A] present evaluation.
[B] IRDF-2002 (IRDF-90 version 2).

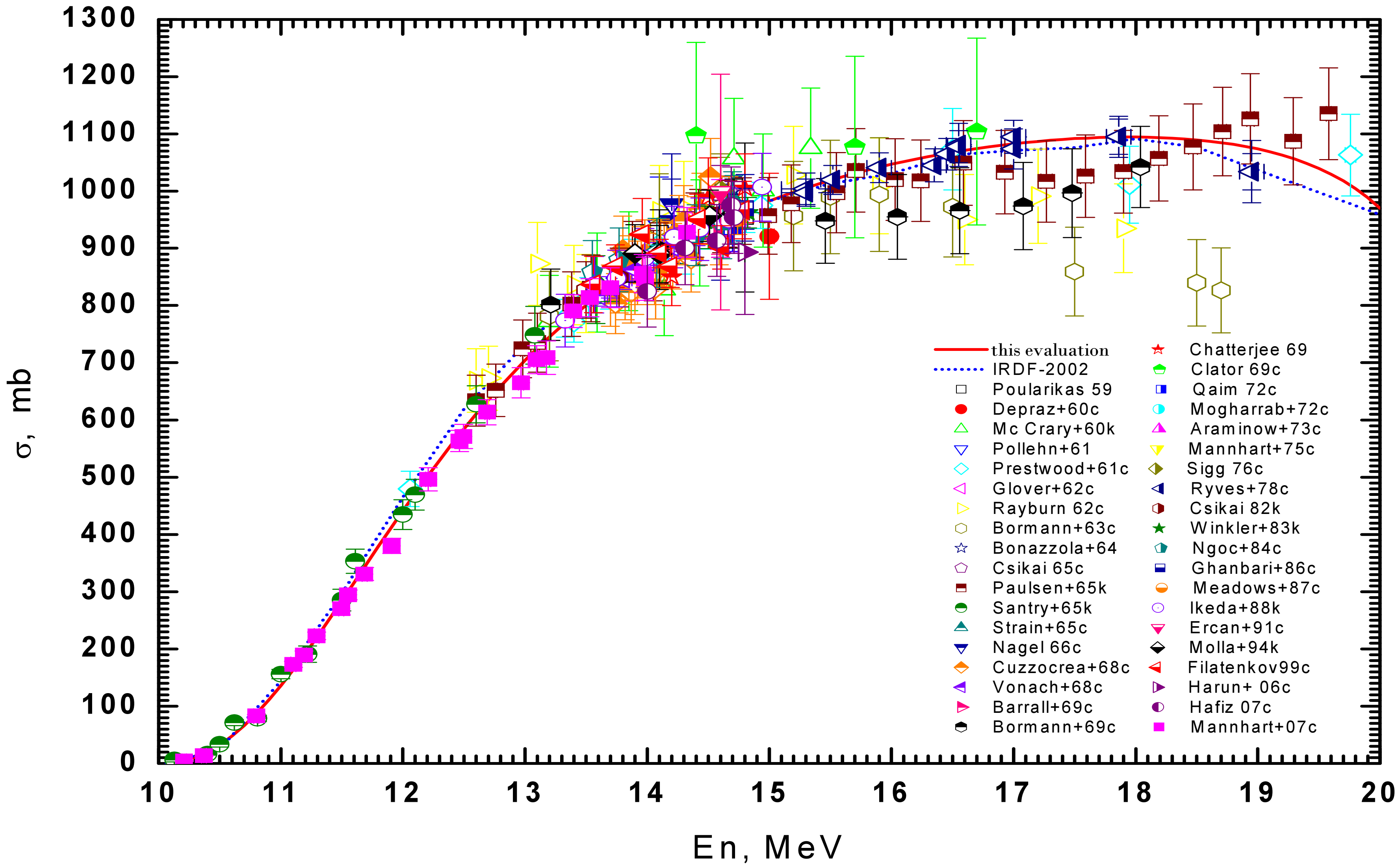

FIG. 7.1. Re-evaluated excitation function of the $^{65}Cu(n,2n)^{64}Cu$ reaction in the energy range from threshold to 20 MeV in comparison with IRDF-2002 and experimental data.

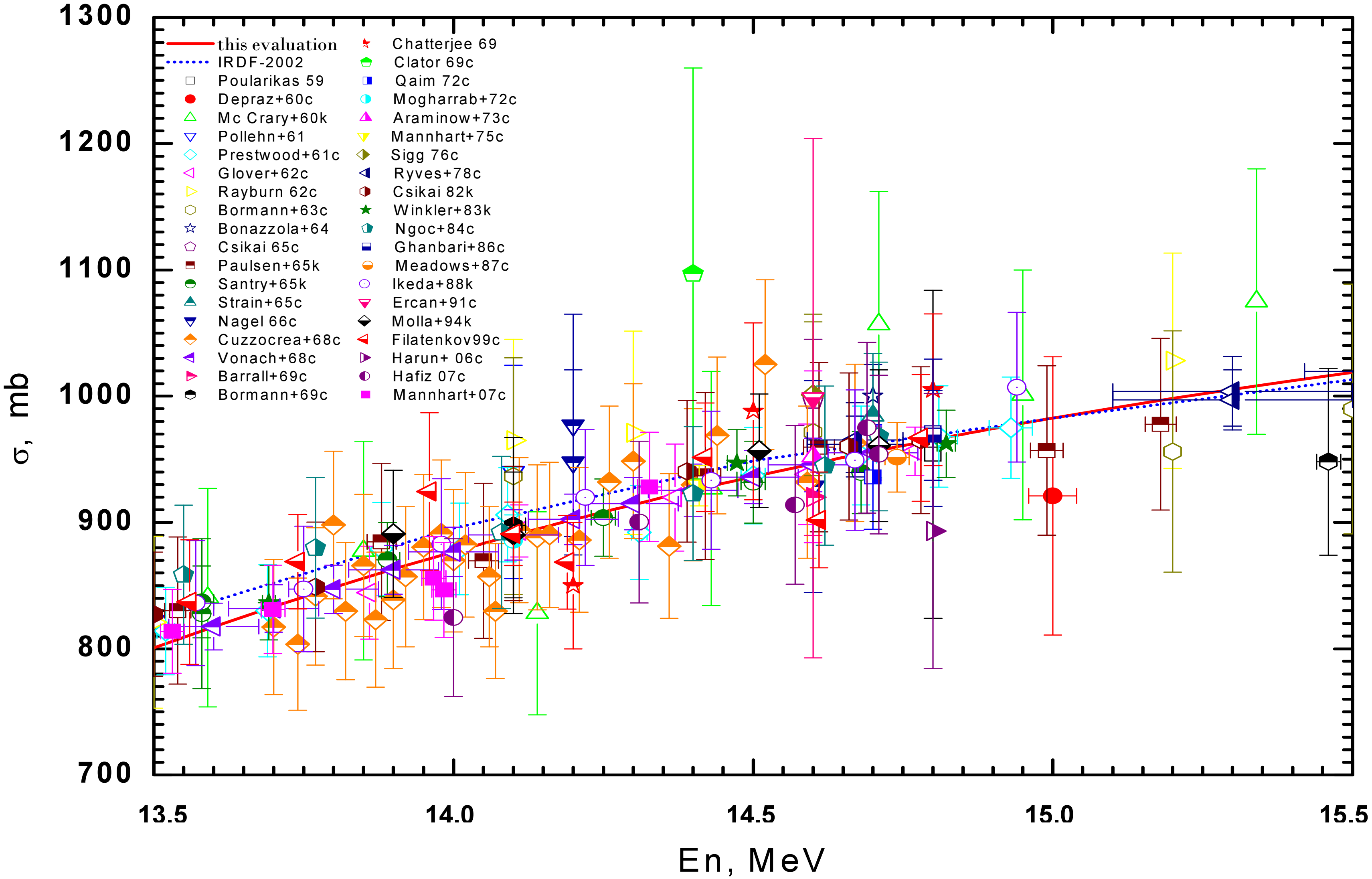

FIG. 7.2. Re-evaluated excitation function of the $^{65}Cu(n,2n)^{64}Cu$ reaction in the energy range from 13.5 to 15.5 MeV in comparison with IRDF-2002 and experimental data.

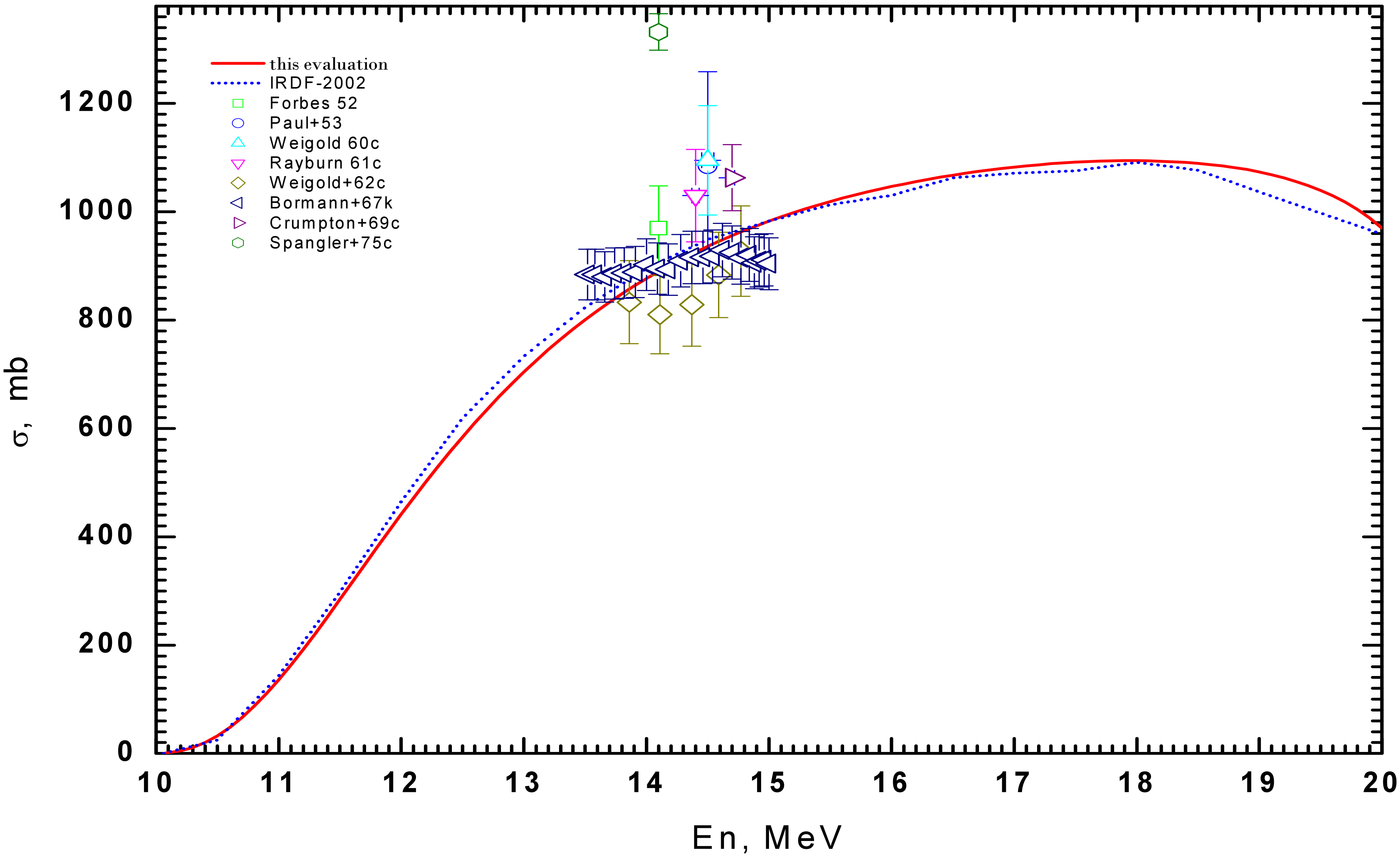

FIG. 7.3. Re-evaluated excitation function of the $^{65}Cu(n,2n)^{64}Cu$ reaction in the energy range from threshold to 20 MeV in comparison with IRDF-2002 and rejected experimental data.

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8. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE $^{64}Zn(n,p)^{64}Cu$ REACTION

The abundance of the ^{64}Zn isotope in natural zinc is 48.63 ± 0.60 atom percent, and the ^{64}Cu obtained via the (n,p) reaction undergoes (61.0 ± 0.3)% electron-capture and (39.0 ± 0.3)% β^- decay with a half-life of (12.700 ± 0.002) hours. 511-keV gamma annihilation radiation (I_γ = 0.348 ± 0.004) was used to determine the $^{64}Zn(n,p)^{64}Cu$ reaction rate. Recommended decay data for the half-life, and beta and gamma-rays emission probabilities per decay of ^{64}Cu were taken from Ref. [2.7] of Section 2.

Microscopic experimental data were analyzed in the preparation of the input database used to evaluate the cross sections and their uncertainties for the $^{64}Zn(n,p)^{64}Cu$ reaction [8.1-8.47]. During the course of this procedure, experimental data from Refs. [8.3, 8.4, 8.6-8.16, 8.18-8.30, 8.33, 8.35-8.38, 8.40-8.44, 8.46] were corrected with respect to the newly recommended cross-section standards and decay data (see Table 2.1).

Corrections were made to the experimental data of Refs. [8.6, 8.11, 8.12, 8.17, 8.25, 8.29-8.32]. Results of the relative measurements of Paulsen and Liskien for incident neutron energies of 0.99 to 2.21 MeV [8.6] were normalized to the absolute cross section of 13.416 mb at 2.20 MeV as determined from the measurements of Smith and Meadows [8.12] with a $^{7}Li(p,n)^{7}Be$ neutron source. Microscopic cross sections measured by Smith and Meadows for the $^{64}Zn(n,p)^{64}Cu$ reaction from 1.159 to 5.576 MeV agree well with the experimental data of Ikeda *et al*. [8.22] and integral experimental data for the ^{252}Cf spontaneous fission neutron spectrum.

The experimental data of Santry and Butler in the energy range from 1.50 to 5.33 MeV [8.11] and the cross-section data of King *et al.* at 2.12 - 4.84 MeV [8.17] were also renormalized to the results of Smith and Meadows with a $^{7}Li(p,n)^{7}Be$ neutron source [8.12]. Correction factors applied to the experimental data from Refs. [8.11] and [8.17] were Fc = 0.91582 and 1.08431, respectively, while a correction factor of Fc = 0.91582 was applied to all the experimental data of Santry and Butler [8.11].

Smith and Meadows used neutrons from the $D(d,n)^{3}He$ reaction [8.12], and their data were renormalized to a cross section of 183.6 mb at 5.384 MeV obtained from measurements carried out with a $^{7}Li(p,n)^{7}Be$ neutron source. After corrections based on the new standard cross sections for the $^{238}U(n,f)$ monitor reaction and recommended emission probability data for ^{64}Cu annihilation gammas, the $D(d,n)^{3}He$ data of Smith and Meadows in the energy range 5.384 to 9.939 MeV were increased by a factor of Fc = 1.15386.

The results of precise measurements carried out by Mannhart and Schmidt are more representative over the neutron energy range of 8.4 to 14.3 MeV [8.33]. Experimental data to be found in Refs. [8.2, 8.16, 8.26, 8.32] have been renormalized by a factor of Fc = 1.15386 as determined from the data of Santry and Butler [8.11], and agree well with the more recent measurements of Mannhart and Schmidt [8.33]. The cross-section data of Viennot *et al*. [8.25], Molla *et al*. [8.30], and Kielan and Marcinkowski [8.31] were renormalized to the integral cross sections calculated from the experimental data of Mannhart and Schmidt [8.33] for the overlapping energy range. Experimental data of Ghorai *et al*. [8.29] were renormalized to evaluated integral cross sections from 14.2 to 16.2 MeV. After corrections against the new neutron cross-section standards, the experimental data of Refs. [8.25, 8.29-8.31] were multiplied by the factors Fc = 0.81195, 0.84351, 0.75385 and 0.93987, respectively.

Huang Xiaolong *et al*. carried out measurements in the energy range 14.65 to 19.02 MeV [8.32] by means of a $T(d,n)^4He$ neutron source, and these data were renormalized to a cross section of 152.9 mb (±2.4%) at 14.65 MeV as evaluated from the experimental data in Refs. [8.4, 8.7, 8.8, 8.14, 8.28].

Specific experimental data were rejected due to large, systematic underestimations of the cross sections above 2.8 MeV [8.35, 8.37, 8.39, 8.47]. The measurements of Bormann and Lammers obtained from a neutron energy of 14.10 to 18.19 MeV were not taken into account from 15.46 to 18.19 MeV due to a systematic overestimation of these particular cross sections [8.9]. Similarly, the data of Ghorai *et al*. at 17.2 MeV [8.29] and Kielan and Marcinkowski [8.31] at 15.9 and 16.6 MeV were also rejected. Cross-section data given in Refs. [8.34-8.47] were rejected completely due to their large deviations from the main bulk of experimental data. Furthermore, cross sections were only measured at one energy point between 14 and 15 MeV in some of the rejected experimental data [8.34, 8.36, 8.40, 8.42-8.46].

The excitation function for the $^{64}Zn(n,p)^{64}Cu$ reaction in the energy region from threshold to 20 MeV was evaluated by means of statistical analyses of the experimental cross-section data [8.1-8.33] by means of the PADE-2 code. Uncertainties in the evaluated excitation function for the $^{64}Zn(n,p)^{64}Cu$ reaction are given in the form of a relative covariance matrix for 49-neutron energy groups (LB = 5). The covariance matrix of uncertainties was calculated simultaneously with the recommended cross sections by means of the PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix of File-33 are as follows:

5.50784E-08	5.71733E-08	5.94423E-08	6.15434E-08
6.41589E-08	6.67452E-08	6.87802E-08	7.09362E-08
7.33156E-08	7.62346E-08	7.97587E-08	8.39517E-08
8.83311E-08	9.24351E-08	9.75569E-08	1.04183E-07
1.11626E-07	1.18053E-07	1.24390E-07	1.34278E-07
1.46587E-07	1.56514E-07	1.69670E-07	1.96551E-07
2.10571E-07	2.58348E-07	2.86453E-07	3.37288E-07
4.18775E-07	5.04883E-07	5.48409E-07	6.51679E-07
7.95195E-07	9.69913E-07	1.51323E-06	6.87735E-06
4.70011E-05	4.54712E-04	8.44040E-04	1.00266E-03
1.39839E-03	1.60043E-03	2.35419E-03	2.47891E-03
5.19930E-03	6.20593E-03	6.78721E-03	2.04942E-02
3.28605E-02			

The evaluated excitation function and related uncertainties for the $^{64}Zn(n,p)^{64}Cu$ reaction are given in Table 8.1. Group boundaries for the neutron energies are the same as in File-33. The lowest uncertainties of 1.81% to 1.92% are observed over the neutron energy range from 13.5 to 14.5 MeV, while the largest uncertainty of 11.81% occurs from threshold to 1.75 MeV and arises from the significant discrepancies in the experimental data. Over the extensive neutron energy range from 2.50 to 13.5 MeV, uncertainties in the cross sections are between 2.00% and 2.93%. Inadequate experimental information at neutron energies from 16 to 20 MeV increased the uncertainties from 3.58% to 8.11%.

TABLE 8.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{64}Zn(n,p)^{64}Cu$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 20 MeV.

Neutron energy (MeV) from to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from to**	**Cross section (mb)**	**Uncer-tainty (%)**
0.500 - 1.750	0.435	11.81	7.750 - 8.000	233.431	2.43
1.750 - 2.000	4.528	4.25	8.000 - 8.500	237.672	2.31
2.000 - 2.250	10.941	3.61	8.500 - 9.000	242.455	2.13
2.250 - 2.500	23.187	3.24	9.000 - 9.500	246.769	2.00
2.500 - 2.750	42.689	2.93	9.500 - 10.000	251.112	2.06
2.750 - 3.000	67.682	2.74	10.000 - 11.000	257.083	2.20
3.000 - 3.250	93.146	2.69	11.000 - 11.500	259.162	2.32
3.250 - 3.500	114.482	2.71	11.500 - 12.000	254.580	2.37
3.500 - 3.750	130.332	2.73	12.000 - 12.500	243.880	2.36
3.750 - 4.000	141.779	2.68	12.500 - 13.000	227.746	2.27
4.000 - 4.250	150.467	2.55	13.000 - 13.500	208.260	2.12
4.250 - 4.500	157.680	2.36	13.500 - 14.000	187.776	1.92
4.500 - 4.750	164.211	2.18	14.000 - 14.500	168.057	1.81
4.750 - 5.000	170.482	2.06	14.500 - 15.000	150.067	2.01
5.000 - 5.250	176.690	2.02	15.000 - 15.500	134.146	2.49
5.250 - 5.500	182.896	2.07	15.500 - 16.000	120.273	3.06
5.500 - 5.750	189.089	2.18	16.000 - 16.500	108.258	3.58
5.750 - 6.000	195.213	2.31	16.500 - 17.000	97.857	3.99
6.000 - 6.250	201.193	2.42	17.000 - 17.500	88.831	4.34
6.250 - 6.500	206.949	2.50	17.500 - 18.000	80.964	4.68
6.500 - 6.750	212.400	2.55	18.000 - 18.500	74.072	5.13
6.750 - 7.000	217.481	2.56	18.500 - 19.000	68.001	5.81
7.000 - 7.250	222.141	2.55	19.000 - 19.500	62.625	6.79
7.250 - 7.500	226.351	2.52	19.500 - 20.000	57.839	8.11
7.500 - 7.750	230.107	2.48			

The re-evaluated excitation function for the $^{64}Zn(n,p)^{64}Cu$ reaction is shown in Fig. 8.1 from 0.5 to 20.0 MeV, and is compared with the equivalent IRDF-2002 excitation function and the experimental data. These evaluated cross sections and the rejected experimental data are given in Fig. 8.2. The IRDF-2002 evaluation did not include the recent experimental data of Mannhart and Schmidt [8.33] and Ikeda *et al*. [8.26]. Therefore, IRDF-2002 underestimated systematically the cross sections of the $^{64}Zn(n,p)^{64}Cu$ reaction in the energy range from 6 to 11 MeV (see Figs. 8.1 and 8.2). Above 14.5 MeV, the IRDF-2002 evaluation overestimated the cross sections in a systematic manner compared with the re-evaluated data. The excitation function re-evaluated from 14 to 20 MeV follows the trend predicted by the renormalized experimental data of Santry and Butler [8.11]. Below 3.6 MeV and in the narrow neutron energy range of 13.8 to 14.2 MeV, both evaluations are in good agreement.

Integral experimental data for the $^{64}Zn(n,p)^{64}Cu$ reaction are given in Refs. [8.48-8.66]. Thirteen experiments were carried out in research reactors and critical assembles with similar spectra to a ^{235}U thermal fission neutron spectrum [8.48-8.60], and six experiments were performed in a ^{252}Cf spontaneous fission neutron spectrum [8.61-8.66].

Measured integral cross sections for the ^{252}Cf spontaneous fission neutron spectrum [8.61-8.66] range from (40.46 ± 1.03) [8.62] to (45.86 ± 2.27) mb [8.61]. The earlier measurements of Dezsö and Csikai [8.61] and Mannhart and Alberts [8.62] may be considered as preliminary data. Representative experimental data for the ^{252}Cf spontaneous fission neutron spectrum were obtained by Kobayashi *et al*. [8.63], Mannhart [8.65] and Benabdallah *et al*. [8.66]. The average weighted integral cross section for the ^{64}Zn(n,p)^{64}Cu reaction was determined to be (42.34 ± 0.94) mb on the basis of these experiments.

Measured integral cross sections for the ^{235}U thermal fission neutron spectrum range from 27.0 to 40.87 mb [8.48-8.60]. The lowest value of (27.0 ± 4.1) mb was obtained by Nasyrov and Sciborskij in their measurements on a critical assembly [8.53] in which no details are given about the procedure adopted to determine the ^{235}U thermal fission cross section from the measured value. Furthermore, no information is given about the monitor reaction and gamma-ray probability used for the annihilation radiation. The highest cross section of (40.87 ± 2.86) mb was obtained by Rochlin [8.48] after correcting for the newly recommended cross section of (0.7007 ± 0.0090) mb for the ^{27}Al(n,α)^{24}Na monitor reaction [8.67]. Unfortunately, no detailed information is given in Ref. [8.48] of the ^{64}Cu decay data used in these measurements.

Measured neutron spectra show that standard ^{235}U thermal fission neutron spectrum may be obtained in facilities consisting of thermal columns of 90%-enriched ^{235}U fission plates. Experimental data obtained in reactor cores and critical assemblies are required to correct for differences between real and standard ^{235}U thermal fission neutron spectra. Determination of this recalculation factor is a significant problem and the main source of uncertainty in the resulting cross sections. Therefore, integral experimental data obtained in measurements with perturbed fission spectra was not taken into consideration [8.48, 8.49, 8.52, 8.53, 8.56, 8.59]. All of the other experimental data were measured in facilities with an enriched ^{235}U fission plate converter. Experimental data given in Refs. [8.51, 8.60] were not considered in the re-evaluation because of the absence of information concerning the gamma-ray probability used for the ^{64}Cu annihilation radiation. After applying corrections based on the new neutron cross-section standards, the well-defined experimental studies in Refs. [8.50, 8.54, 8.55, 8.57, 8.58] gave the following integral cross sections for the ^{64}Zn(n,p)^{64}Cu reaction: (33.94 ± 2.09), (31.61 ± 1.63), (40.18 ± 3.80), (32.67 ± 0.87) and (37.04 ± 4.18) mb, respectively. The lowest values of (31.61 ± 1.63) [8.54] and (32.67 ± 0.87) mb [8.57] were obtained in experiments with only 20%-enriched ^{235}U fission plates, which excludes them from consideration in the analysis of the pure ^{235}U thermal fission neutron spectrum. A well-defined experimental cross section of $<\sigma>_{Cf\text{-}252}$ = (42.34 ± 0.94) mb and $<\sigma>_{U\text{-}235}$ cross sections measured by Boldeman [8.50], Kimura *et al*. [8.55] and Horibe *et al*. [8.58] gave the following $<\sigma>_{Cf\text{-}252}/<\sigma>_{U\text{-}235}$ values of 1.247, 1.054 and 1.143, respectively Mannhart evaluated the experimental data for 29 reactions on the basis of the $<\sigma>_{Cf\text{-}252}/<\sigma>_{U\text{-}235}$ ratio for the ^{64}Zn(n,p)^{64}Cu reaction being equal to 1.1 [8.67]; these results show that the $<\sigma>_{U\text{-}235}$ cross sections measured by Boldeman [8.50] for the ^{64}Zn(n,p)^{64}Cu reaction are significantly underestimated. The average weighted integral cross section $<\sigma>_{U\text{-}235}$ for the ^{64}Zn(n,p)^{64}Cu reaction as determined from the experimental data of Refs. [8.55, 8.58] is (38.89 ± 2.82) mb, and this value gives $<\sigma>_{Cf\text{-}252}/<\sigma>_{U\text{-}235}$ ratio of 1.089 that agrees well with a predicted value of 1.1. Precise measurements of the ^{64}Zn(n,p)^{64}Cu reaction are required to obtain a more precise cross section for the ^{235}U thermal fission neutron spectrum.

An integral cross section of (38.89 ± 2.82) mb has been evaluated for the ^{235}U thermal fission neutron spectrum, and a value of (42.34 ± 0.94) mb was used for the ^{252}Cf spontaneous fission neutron spectrum in the benchmark calculations. The results of tests with the re-evaluated excitation function of the ^{64}Zn(n,p)^{64}Cu reaction are given in Table 8.2 for two standard neutron spectra. These data show that the integral cross sections calculated from the re-evaluated excitation function for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are in good agreement with the experimental values of (38.89 ± 2.82) and (42.34 ± 0.94) mb. Integral cross sections calculated from the IRDF-2002 data also agree with these experimental values. However, the evaluated data recommended in Ref. [8.67] contradict the microscopic cross sections obtained in both of the IRDF evaluations.

TABLE 8.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR THE $^{64}Zn(n,p)^{64}Cu$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Average cross section, mb		C/E [*]	C/E [8.67]
	Calculated	Measured		
^{235}U thermal fission neutron spectrum	38.914 [A]	38.89 ± 2.82 [*]	1.00062	1.09958
	38.399 [B]	35.39 ± 1.07 [8.67]	0.98737	1.08502
^{252}Cf spontaneous fission neutron spectrum	42.718 [A]	42.34 ± 0.94 [*]	1.00893	1.04311
	42.095 [B]	40.59 ± 0.67 [8.67]	0.99421	1.03708

[A] present evaluation.
[B] IRDF-2002 (IRDF-90 version 2).
[*] evaluated experimental cross section from this work.

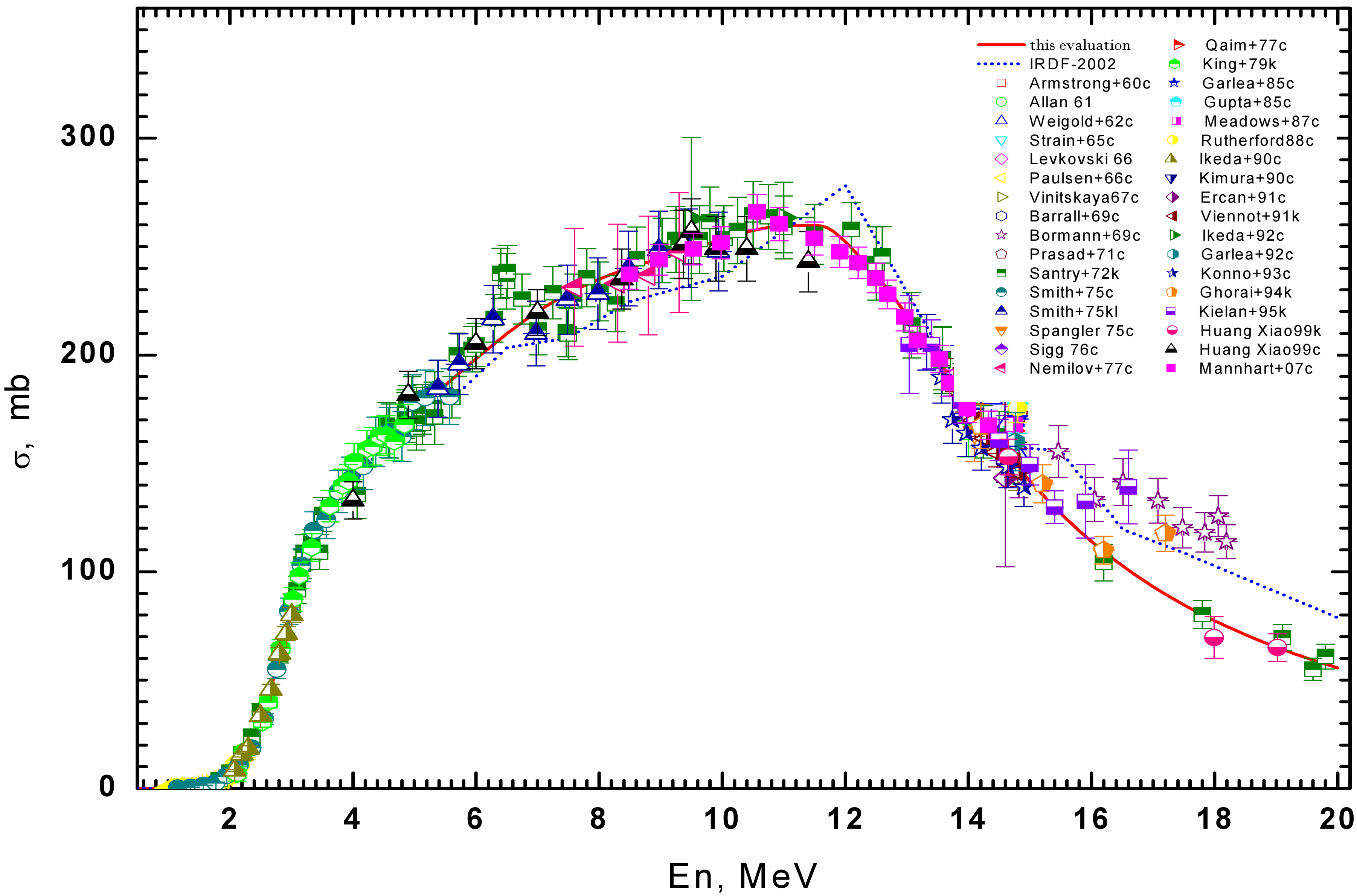

FIG. 8.1. Re-evaluated excitation function of the $^{64}Zn(n,p)^{64}Cu$ reaction in the energy range from threshold to 20 MeV in comparison with IRDF-2002 and experimental data.

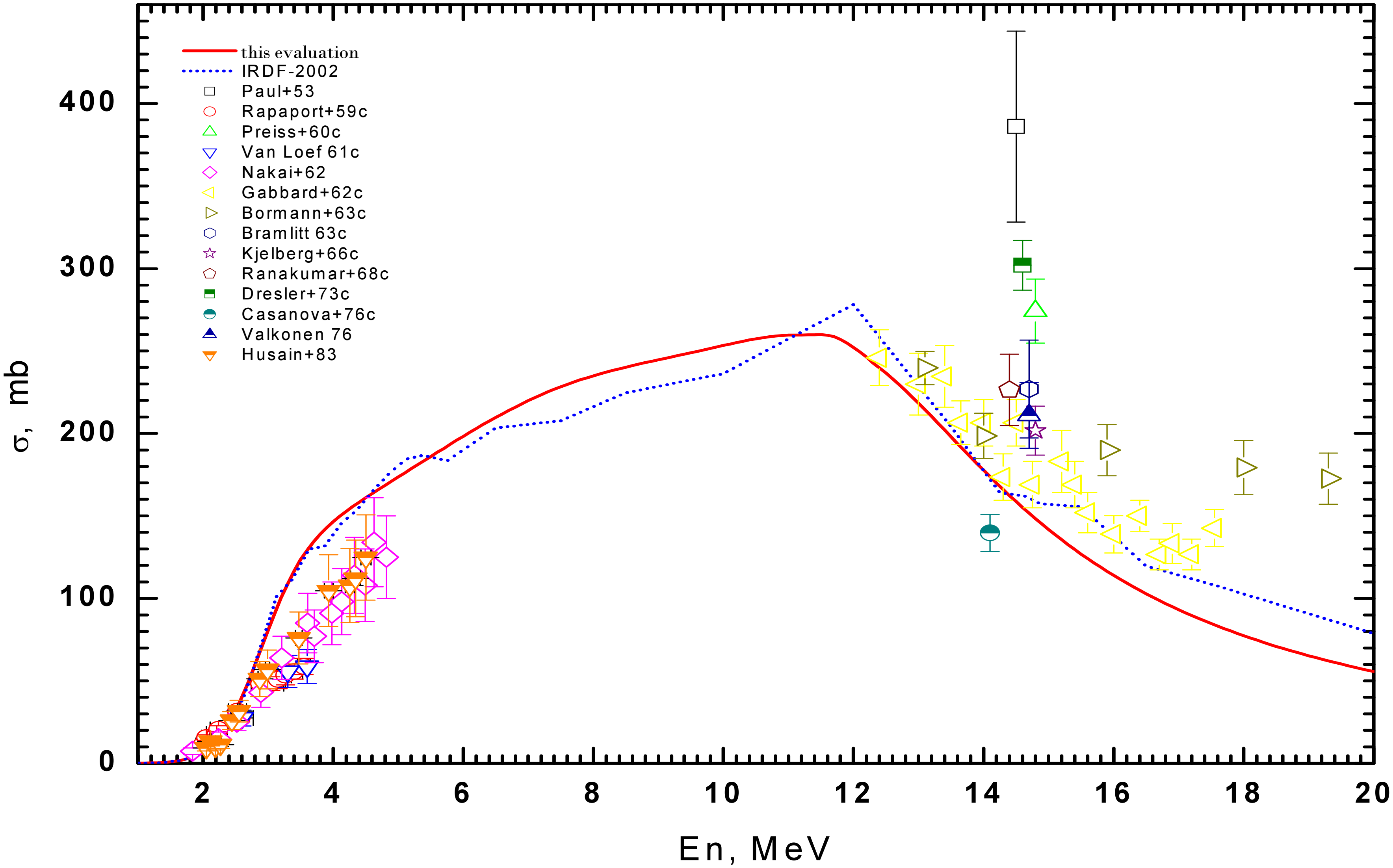

FIG. 8.2. Re-evaluated excitation function of the ^{64}Zn(n,p)^{64}Cu reaction in the energy range from threshold to 20 MeV in comparison with IRDF-2002 and rejected experimental data.

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9. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE ^{115}In(n,2n)^{114m}In REACTION

The isotopic abundance of ^{115}In in natural indium is (95.71 ± 0.05) atom percent, and ^{114m}In from the (n,2n) reaction undergoes (4.4 ± 0.3)% electron-capture + β^+ decay and (95.6 ± 0.3)% IT decay from the first excited state of ^{114}In (190.29 keV and 5^+) with a half-live of (49.51 ± 0.01) days. Decay from the metastable state is accompanied by the emission of 190.27-, 558.43-, 725.24-keV gamma rays. While the 190.27-keV gamma ray arises from the IT decay mode, the 558.43- and 725.24-keV gamma rays are generated in the decay of the ^{114}Cd daughter nucleus from the EC/β^+ decay mode. The ^{115}In(n,2n)^{114m}In reaction rate is usually measured by determining the activity corresponding to the most intensive 190.27-keV gamma ray (I_γ = 0.1556 ± 0.0015). All decay data for ^{114m}In are recommended values from Refs. [2.6, 2.7] of Section 2.

Microscopic experimental data were analyzed during the preparation of the input database for the evaluation of the cross sections and uncertainties for the ^{115}In(n,2n)^{114m}In reaction [9.1-9.21]. The experimental data from Refs. [9.1-9.3, 9.5-9.21] were corrected with respect to the newly recommended neutron cross-section data used, as monitor reactions in the measurements, and the relevant decay data (see Table 2.1).

A series of additional corrections were made to the experimental data of Refs. [9.4, 9.8, 9.10, 9.12]. Thus, the relative measurements of Vonach *et al.* [9.4] were normalized at 14.70 MeV to the absolute cross section of (1386.0 ± 33.3) mb determined from the experimental data of Csikai *et al.* [9.16] and Filatenkov and Chuvaev [9.17]. Similarly, the cross-section data of Ryves *et al.* [9.12] were re-normalized at 14.68 MeV to this same value. The experimental data of Csikai *et al* [9.16] and Vonach *et al.* [9.4] exhibit very similar energy dependence for the excitation function of the ^{115}In(n,2n)^{114m}In reaction over the neutron energy range from 13.00 to 15.00 MeV, and these measurements were used to derive integral cross sections for the correction of the equivalent experimental data of Lu Hanlin *et al.* [9.8] and Santry and Butler [9.l0].

Experimental cross-section data in Refs. [9.18-9.21] were rejected because they underestimate significantly the cross section of the ^{115}In(n,2n)^{114m}In reaction, while the data given in Ref. [9.20] also reproduce those published in Ref. [9.19].

The excitation function for the ^{115}In(n,2n)^{114m}In reaction from threshold to 20 MeV was evaluated by undertaking statistical analyses of the experimental cross-section data by means of the PADE-2 code [9.1-9.17]. Uncertainties in the resulting excitation function are given in the form of a relative covariance matrix for 22-neutron energy groups (LB = 5). Covariance matrix uncertainties were calculated simultaneously with recommended cross-section data by means of PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix in File-33 are as follows:

6.23554E-08	6.27847E-08	6.34214E-08	6.34214E-08
6.60645E-08	6.61285E-08	6.81717E-08	6.90488E-08
7.41560E-08	7.84197E-08	8.61404E-08	1.07485E-07
1.56069E-07	3.04283E-07	1.19516E-06	3.26629E-05
1.73504E-03	2.43384E-03	6.06701E-03	1.19199E-02
1.65514E-02	9.53441E-02		

Evaluated group cross sections and their uncertainties for the ^{115}In(n,2n)^{114m}In reaction are listed in Table 9.1. Group boundaries are the same as in File-33. These data show that the smallest uncertainties in the evaluated cross sections of 2.26% to 2.30% are observed in the neutron energy range from 13.5 to 15.5 MeV. Significant uncertainties of 13.76% to 19.81% in the cross sections from threshold to 10.5 MeV arise from the large uncertainties in the experimental data within this region and the discrepancies between these experimental data; increases in the uncertainties of the excitation function over the neutron energy range from 17.0 to 20.0 MeV can be explained in a similar manner.

Fig. 9.1 shows the re-evaluated excitation function for the ^{115}In(n,2n)^{114m}In reaction for neutron energies between 8.0 and 20.0 MeV in comparison with the IRDF-2002 cross sections and experimental data. At neutron energies from 10.2 to 13.8 MeV, IRDF-2002 gives systematically underestimated cross sections in comparison with the re-evaluated data. More significantly, the newly re-evaluated excitation function exhibits improved agreement with the corrected experimental data.

Integral cross-section data have not been measured for the ^{115}In(n,2n)^{114m}In reaction in ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra. Therefore, only calculated average cross sections for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are given in Table 9.2, which show that integral cross sections calculated from the re-evaluated excitation function are higher for both spectra than equivalent data from the IRDF-2002 library.

TABLE 9.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE ^{115}In(n,2n)^{114m}In REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 20 MeV.

Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)	Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)
9.311 - 9.600	13.303	19.81	14.500 - 15.000	1379.950	2.26
9.600 - 10.000	78.869	18.91	15.000 - 15.500	1398.840	2.28
10.000 - 10.500	237.155	13.76	15.500 - 16.000	1407.170	2.36
10.500 - 11.000	457.123	9.12	16.000 - 16.500	1406.560	2.53
11.000 - 11.500	676.245	6.24	16.500 - 17.000	1398.210	2.76
11.500 - 12.000	867.435	4.48	17.000 - 17.500	1382.950	2.98
12.000 - 12.500	1022.560	3.34	17.500 - 18.000	1361.370	3.18
12.500 - 13.000	1143.150	2.68	18.000 - 18.500	1333.850	3.47
13.000 - 13.500	1234.140	2.38	18.500 - 19.000	1300.550	4.31
13.500 - 14.000	1300.950	2.30	19.000 - 19.500	1261.530	6.36
14.000 - 14.500	1348.280	2.27	19.500 - 20.000	1216.670	10.22

TABLE 9.2. CALCULATED AVERAGE CROSS SECTIONS FOR THE ^{115}In(n,2n)^{114m}In REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Calculated average cross section, mb	
	This evaluation	IRDF-2002
^{235}U thermal fission neutron spectrum	0.9001	0.8796
^{252}Cf spontaneous fission neutron spectrum	1.6328	1.5874

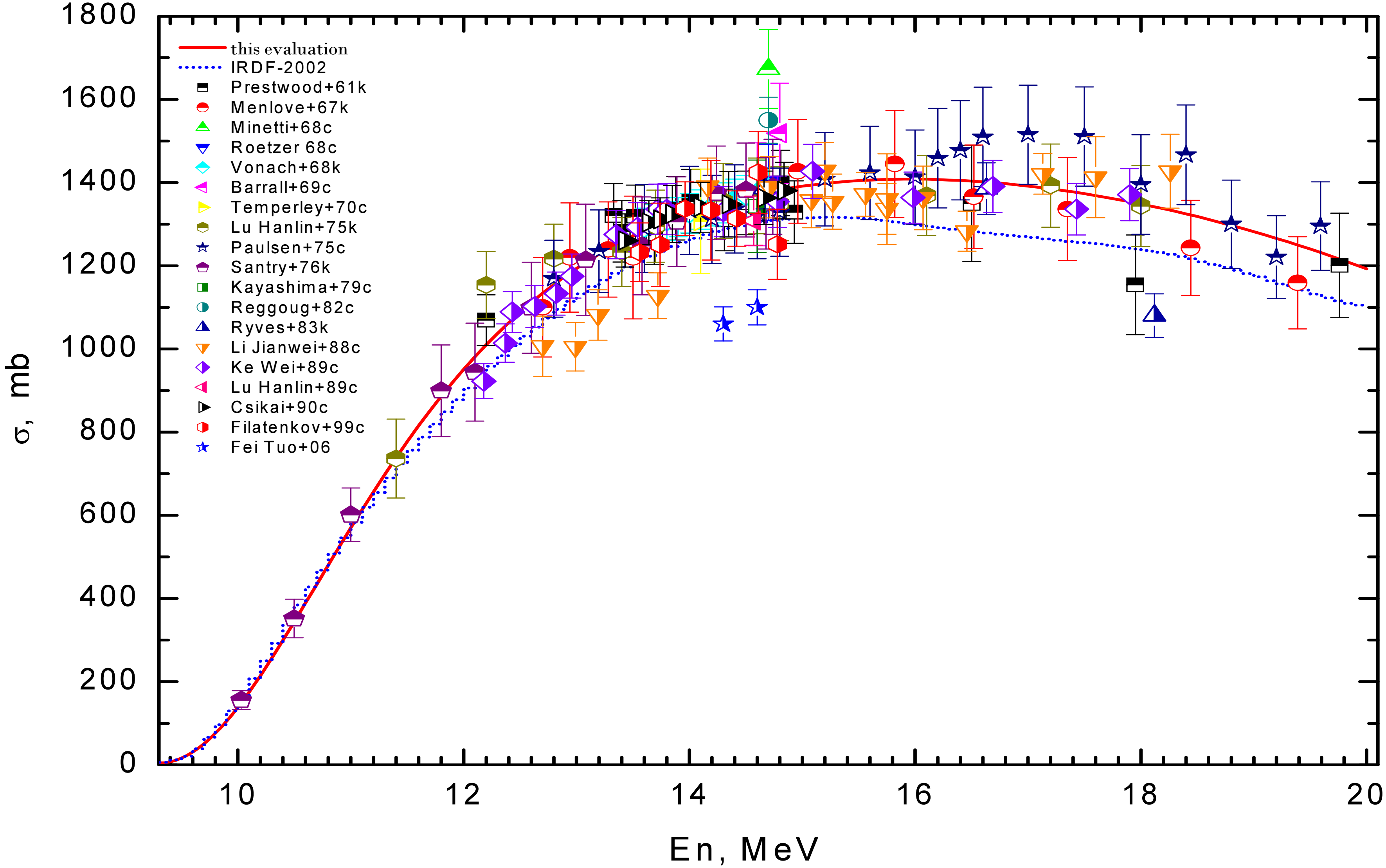

FIG. 9.1. Re-evaluated excitation function of the $^{115}In(n,2n)^{114m}In$ reaction in the energy range from threshold to 20 MeV in comparison with IRDF-2002 and experimental data.

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10. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE $^{127}I(n,2n)^{126}I$ REACTION

The isotopic abundance of ^{127}I in natural iodine is 100 atom percent, and ^{126}I from the (n,2n) reaction undergoes (52.7 ± 0.5)% electron-capture + β^+ decay and (47.3 ± 0.5)% β^- decay with a half-live of (12.93 ± 0.05) days. Decay of ^{126}I is accompanied by the emission 388.633-, 491.243-, 511-, 666.331- and 753.819-keV gamma rays. The $^{127}I(n,2n)^{126}I$ reaction rate is usually measured by determining the activity corresponding to the most intensive gamma rays: 388.633-keV (I_γ = 0.356 ± 0.006) from β^- decay and ^{126}Xe daughter nucleus; 666.331-keV (I_γ = 0.329 ± 0.007) from EC/β^+ decay and ^{126}Te daughter nucleus. All decay data for ^{126}I are recommended values from Ref. [2.7] of Section 2.

Microscopic experimental data were analyzed during the preparation of the input database for the evaluation of the cross sections and uncertainties for the $^{127}I(n,2n)^{126}I$ reaction [10.1-10.13]. During this procedure, the experimental data from Refs. [10.3-10.7] were corrected to the newly recommended neutron cross-section standards and the relevant decay data (see Table 2.1).

The database for the evaluation of the excitation function of the $^{127}I(n,2n)^{126}I$ reaction consisted of microscopic experimental data [10.5, 10.6, 10.8-10.13] and data obtained from GNASH theoretical modelling calculations that agreed well within a wide neutron energy range of 9.5 to 19.6 MeV. Experimental cross-section data reported in Refs. [10.1-10.4, 10.7] were rejected due to their systematic underestimation of the $^{127}I(n,2n)^{126}I$ cross sections in the neutron energy range from 12.3 to 18.3 MeV. The evaluation of the excitation function from 20 to 32 MeV was based only on GNASH calculations.

The excitation function for the $^{127}I(n,2n)^{126}I$ reaction from threshold to 32 MeV was evaluated by means of the generalized least-squares method within the PADE-2 code. Uncertainties in the resulting excitation function are given in the form of a relative covariance matrix for 34-neutron energy groups (LB = 5).

Six-digit eigenvalues for the relative covariance matrix in File-33 are as follows:

1.14192E-07	1.15227E-07	1.16982E-07	1.19457E-07
1.22735E-07	1.26068E-07	1.31268E-07	1.36450E-07
1.43199E-07	1.53115E-07	1.62023E-07	1.82277E-07
1.99829E-07	2.49306E-07	2.85824E-07	3.71735E-07
4.88297E-07	5.64248E-07	6.77032E-07	8.37697E-07
1.01492E-06	1.20140E-06	1.39970E-06	1.61113E-06
3.36624E-06	4.51402E-04	5.53163E-04	1.06557E-02
2.51934E-02	6.11561E-02		

Evaluated group cross sections and their uncertainties for the $^{127}I(n,2n)^{126}I$ reaction are listed in Table 10.1. Group boundaries are the same as in File-33. The smallest uncertainties in the evaluated cross sections of 2.32% to 2.49% are observed in the neutron energy range from 13.0 to 17.5 MeV, while the largest uncertainty of 21.09% occurs from threshold to 9.50 MeV due to inadequate experimental information in this region. Within the neutron energy range from 20 to 32 MeV, uncertainties in the evaluated cross sections are only dependent on uncertainties assigned to the data from theoretical modelling calculations.

Fig. 10.1 shows the re-evaluated excitation function for the $^{127}I(n,2n)^{126}I$ reaction over the energy range from 9.0 to 32.0 MeV in comparison with the IRDF-2002 cross sections and experimental data. Between neutron energies of 9.3 to 10.6 MeV, the IRDF-2002 data underestimate the cross sections systematically, while for neutron energies between 11 and 19 MeV, IRDF-2002 recommends overestimated cross sections. At neutron energies of 10.8 and 20.0 MeV, the cross sections are similar within the new re-evaluation and IRDF-2002.

The integral cross section of the $^{127}I(n,2n)^{126}I$ reaction in a ^{235}U thermal fission neutron spectrum has been measured [10.14-10.16], and the more representative data are given in Ref. [10.16]. Averaged over the ^{252}Cf spontaneous fission neutron spectrum, an integral cross section of (2.069 ± 0.050) mb has been determined by Mannhart [10.18, 10.19]. Selected integral experimental data for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were used to test the re-evaluated excitation function of the $^{127}I(n,2n)^{126}I$ reaction, as listed in Table 10.2. These calculated integral cross sections from the re-evaluated excitation function for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are in good agreement with the experimental data, whereas the data calculated from IRDF-2002 are discrepant.

TABLE 10.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{127}I(n,2n)^{126}I$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 32 MeV.

Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)	Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)
9.216 - 9.500	14.482	21.09	17.500 - 18.000	1549.200	2.78
9.500 - 10.000	123.373	7.58	18.000 - 18.500	1509.610	3.07
10.000 - 10.500	382.356	6.64	18.500 - 19.000	1452.940	3.27
10.500 - 11.000	656.209	5.50	19.000 - 19.500	1382.200	3.39
11.000 - 11.500	890.373	4.03	19.500 - 20.000	1302.060	3.44
11.500 - 12.000	1071.960	3.07	20.000 - 21.000	1175.700	3.47
12.000 - 12.500	1205.930	2.73	21.000 - 22.000	1016.160	3.57
12.500 - 13.000	1302.850	2.60	22.000 - 23.000	880.842	3.80
13.000 - 13.500	1373.330	2.46	23.000 - 24.000	772.447	4.14
13.500 - 14.000	1426.070	2.38	24.000 - 25.000	687.124	4.53
14.000 - 14.500	1467.360	2.39	25.000 - 26.000	619.703	4.88
14.500 - 15.000	1501.270	2.45	26.000 - 27.000	565.681	5.18
15.000 - 15.500	1529.800	2.47	27.000 - 28.000	521.624	5.46
15.500 - 16.000	1553.120	2.41	28.000 - 29.000	485.038	5.74
16.000 - 16.500	1569.740	2.32	29.000 - 30.000	454.132	6.05
16.500 - 17.000	1576.870	2.32	30.000 - 31.000	427.622	6.40
17.000 - 17.500	1571.070	2.49	31.000 - 32.000	404.572	6.79

TABLE 10.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR $^{127}I(n,2n)^{126}I$ IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Average cross section, mb		C/E
	Calculated	Measured	
^{235}U thermal fission neutron spectrum	1.1822 [A]	1.197 ± 0.041 [10.16]	0.9876
	1.2180 [B]		1.0175
^{252}Cf spontaneous fission neutron spectrum	2.1068 [A]	2.069 ± 0.050 [10.19]	1.0183
	2.1968 [B]		1.0618

[A] present evaluation.
[B] IRDF-2002 (CENDL/DOS).

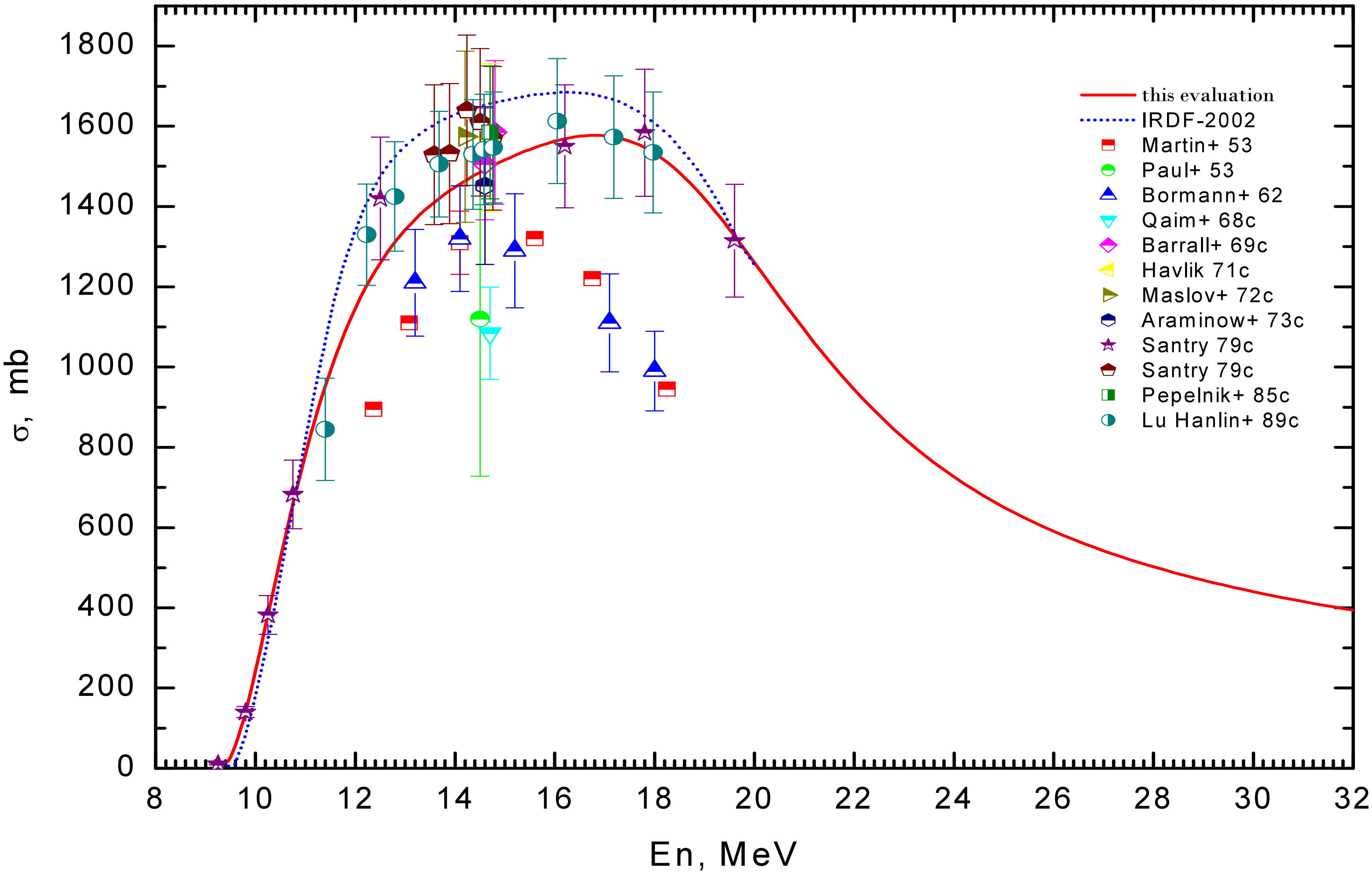

FIG. 10.1. Re-evaluated excitation function of the $^{127}I(n,2n)^{126}I$ reaction in the energy range from threshold to 32 MeV in comparison with IRDF-2002 and experimental data.

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11. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE $^{197}Au(n,2n)^{196}Au$ REACTION

^{197}Au isotopic abundance in natural gold is 100 atom percent, and ^{196}Au from the (n,2n) reaction undergoes (92.80 ± 0.84)% electron-capture and (7.2 ± 0.9)% β^- decay with a half-life of (6.183 ± 0.010) days. The $^{197}Au(n,2n)^{196}Au$ reaction rate is usually measured by determining the activity corresponding to the most intensive gamma rays: 333.03-keV (I_γ = 0.229 ± 0.006), 355.73-keV (I_γ = 0.870 ± 0.008) and 426.10-keV (I_γ = 0.066 ± 0.008). Both the 333.03- and 355.73-keV gamma rays are generated by the decay of daughter ^{196}Hg, while the 426.10-keV gamma ray is emitted in the decay of daughter ^{196}Pt. All decay data for ^{196}Au are recommended values from Refs. [2.6, 2.7] of Section 2.

Microscopic experimental data were analyzed during the preparation of the input database used to evaluate the cross sections and uncertainties for the $^{197}Au(n,2n)^{196}Au$ reaction [11.1-11.39]. The experimental data from Refs. [11.2-11.10, 11.13-11.24, 11.27-11.29, 11.37-11.39] were corrected with respect to the newly recommended neutron cross-section standards and the relevant decay data (see Table 2.1).

A series of specific corrections were made to the experimental data of Refs. [11.1, 11.10, 11.16, 11.20, 11.23, 11.25, 11.26-11.28]:

(a) Neutron energies of 8.40, 9.10, 9.35 and 9.80 MeV as reported by Tewes *et al*. [11.1] were shifted by – 0.25 MeV.

(b) The cross-section data of Paulsen *et al*. [11.10] were obtained with a $^{15}N(d,n)^{16}O$ neutron source, and were renormalized to the integral experimental data of Frehaut *et al*. [11.13] in the overlapping energy range of 10.26 to 11.53 MeV. Cross sections measured by Paulsen *et al*. for incident neutron energies of 12.8 to 19.59 MeV by means of a $T(d,n)^4He$ neutron source were renormalized to the integral experimental data of Greenwood [11.19] over the energy range from 14.5 to 14.9 MeV. Correction factors for these two sets of measurements were Fc = 0.898066 and 1.14644, respectively.

(c) The experimental data of Csikai [11.16], Ikeda *et al*. [11.20] and Filatenkov *et al.* [11.28] were renormalized to the precise measurements of Vonach *et al*. [11.2] and Greenwood [11.19] in the regions of overlapping energy. These data from Refs. [11.16, 11.20, 11.28] were also corrected with respect to the new cross-section standards by multiplying with the coefficients Fc = 0.97027, 1.07128 and 0.95662, respectively.

(d) Wang Xiuyuan *et al*. [11.23] measured relative cross sections for the $^{27}Al(n,\alpha)^{24}Na$ reaction, and these data were renormalized to an absolute measurement of (2135.5 ± 151.2) mb obtained at 14.63 MeV.

(e) Total cross sections for the $^{197}Au(n,2n)^{196}Au$ reaction as given in Ref. [11.25] for incident neutron energies of 32.80 and 38.26 MeV were replaced by recalculated values of the measured metastable cross sections in the energy range of 17.55 to 27.58 MeV with an average isomeric ratio of 0.09971.

(f) Cross sections measured by Uwamino *et al*. [11.26] over a wide energy range from 8.5 to 38.5 MeV were renormalized to the integral experimental data of Frehaut *et al*. [11.13] in the overlapping energy range of 9.5 to 13.8 MeV (Fc = 1.15509).

(e) Data of Belgaid *et al*. [11.27] were obtained by summing the measured metastable and ground state cross sections after correcting with respect to the new standards.

Various experimental cross-section data were rejected [11.30-11.39] due to their discrepancy with respect to the main bulk of the experimental data [11.1-11.29]. Experiments reported in

Refs. [11.30, 11.31, 11.33-11.36, 11.38, 11.39] were set aside because they involved the measurement of only one energy point over the energy range from 14 to 15 MeV.

The excitation function for the $^{197}Au(n,2n)^{196}Au$ reaction from threshold to 40 MeV was evaluated by carrying out statistical analyses of the experimental data by means of the PADE-2 code [11.1-11.29]. Uncertainties in the resulting excitation function were calculated simultaneously, and are given in the form of a relative covariance matrix for 45-neutron energy groups (LB = 5).

Six-digit eigenvalues for the relative covariance matrix in File-33 were as follows:

5.11740E-07	5.13560E-07	5.16557E-07	5.21130E-07
5.28375E-07	5.47078E-07	5.76168E-07	6.30218E-07
6.83781E-07	7.80648E-07	8.99989E-07	1.06192E-06
1.42854E-06	1.80480E-06	2.98749E-06	3.94751E-06
5.98056E-06	9.09072E-06	1.26207E-05	1.61391E-05
1.93342E-05	2.20412E-05	2.42159E-05	2.58905E-05
2.71339E-05	2.80240E-05	2.86343E-05	2.90266E-05
2.92490E-05	2.93225E-05	2.93723E-05	2.94002E-05
5.77001E-05	2.16094E-04	5.25831E-04	9.03750E-04
1.19417E-03	1.51751E-03	1.57719E-03	2.35156E-03
3.46141E-03	4.45408E-03	9.46260E-03	2.52514E-02
3.81316E-02			

Table 11.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{197}Au(n,2n)^{196}Au$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 40 MeV.

Neutron energy (MeV) from to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from to**	**Cross section (mb)**	**Uncer-tainty (%)**
8.112 - 8.500	37.075	15.57	18.000 - 19.000	1510.390	1.95
8.500 - 9.000	183.065	4.26	19.000 - 20.000	1157.560	2.34
9.000 - 9.500	521.409	3.62	20.000 - 21.000	882.910	2.92
9.500 - 10.000	882.812	3.06	21.000 - 22.000	692.433	3.29
10.000 - 10.500	1149.440	3.01	22.000 - 23.000	565.352	3.40
10.500 - 11.000	1363.230	2.80	23.000 - 24.000	480.420	3.42
11.000 - 11.500	1532.500	2.69	24.000 - 25.000	422.833	3.50
11.500 - 12.000	1673.480	2.59	25.000 - 26.000	382.946	3.66
12.000 - 12.500	1797.930	2.45	26.000 - 27.000	354.699	3.83
12.500 - 13.000	1912.410	2.27	27.000 - 28.000	334.265	3.97
13.000 - 13.500	2018.010	1.86	28.000 - 29.000	319.191	4.10
13.500 - 14.000	2103.710	1.25	29.000 - 30.000	307.869	4.22
14.000 - 14.200	2141.480	1.12	30.000 - 31.000	299.226	4.39
14.200 - 14.400	2151.490	1.12	31.000 - 32.000	292.530	4.62
14.400 - 14.600	2157.470	1.08	32.000 - 33.000	287.271	4.93
14.600 - 14.800	2162.230	1.03	33.000 - 34.000	283.087	5.31
14.800 - 15.000	2166.160	1.08	34.000 - 35.000	279.720	5.74
15.000 - 15.500	2168.520	1.44	35.000 - 36.000	276.979	6.21
15.500 - 16.000	2160.800	1.97	36.000 - 37.000	274.724	6.70
16.000 - 16.500	2128.610	2.19	37.000 - 38.000	272.849	7.20
16.500 - 17.000	2058.100	2.20	38.000 - 39.000	271.274	7.71
17.000 - 17.500	1942.260	2.08	39.000 - 40.000	269.938	8.21
17.500 - 18.000	1785.980	1.97			

Evaluated group cross sections and their related uncertainties for the $^{197}Au(n,2n)^{196}Au$ reaction are listed in Table 11.1. Group boundaries of the neutron energy groups are the same as in File-33. The smallest uncertainties in the evaluated cross sections of 1.03% to 1.44% are observed in the neutron energy range from 13.5 to 15.5 MeV, while the most significant uncertainty of 15.57% occurs from threshold to 8.5 MeV due to the large uncertainties and discrepancies between the experimental data in this region. The evaluated excitation function for the $^{197}Au(n,2n)^{196}Au$ reaction in the energy range from 13.5 to 15.5 MeV could be adopted as reference data for activation measurements involving nuclei of intermediate half-life.

Fig. 11.1 shows the re-evaluated excitation function for the $^{197}Au(n,2n)^{196}Au$ reaction over the energy range from 8.0 to 40.0 MeV in comparison with the equivalent IRDF-2002 excitation function and experimental data. These same re-evaluated cross sections and the rejected experimental data are shown in Fig. 11.2. Between neutron energies of 10.2 and 13.8 MeV, the IRDF-2002 data were found to overestimate the cross sections in a systematic manner when compared with the re-evaluated data. Furthermore, the re-normalised experimental data of Frehaut *et al*. [11.13] were found to be in extremely good agreement with the re-evaluated data.

Integral experiments for the $^{197}Au(n,2n)^{196}Au$ reaction are described in Refs. [11.40-11.46]. One of these experiments was carried out in a neutron spectrum that is similar to that of ^{235}U thermal fission [11.40], and six experiments were performed in a ^{252}Cf spontaneous fission neutron spectrum [11.41-11.46]. However, the most representative experimental cross sections for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are believed to originate from Ref. [11.47]. These experimental data were used in benchmark calculations involving the re-evaluated excitation function of the $^{197}Au(n,2n)^{196}Au$ reaction. Calculated and measured average cross sections for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are listed in Table 11.2. These data show that integral cross sections calculated from the re-evaluated excitation function for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are in good agreement with experimental data. The integral cross section calculated from the IRDF-2002 data for the ^{252}Cf spontaneous fission neutron spectrum is about 4.4% higher than the measured value.

Table 11.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR THE $^{197}Au(n,2n)^{196}Au$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	**Average cross section, mb**		**C/E**
	Calculated	**Measured**	
^{235}U thermal fission neutron spectrum	3.3526 [A]	3.392 ± 0.080 [11.47]	0.9884
	3.4762 [B]		1.0248
^{252}Cf spontaneous fission neutron spectrum	5.5312 [A]	5.506 ± 0.101 [11.47]	1.00458
	5.7469 [B]		1.04375

[A] present evaluation.
[B] IRDF-2002 (IRDF-90 version 2).

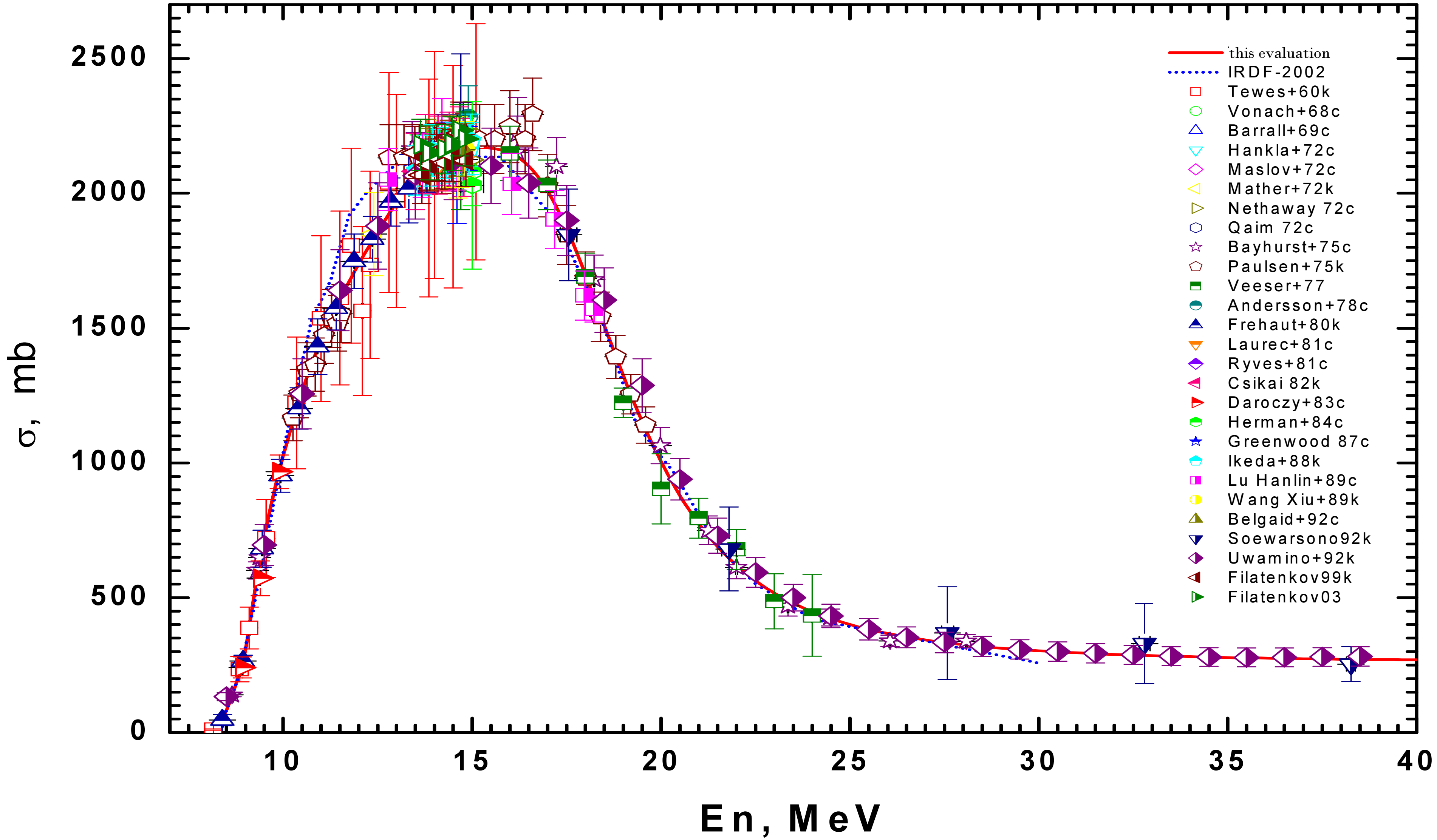

FIG. 11.1. Re-evaluated excitation function of the $^{197}Au(n,2n)^{196}Au$ reaction in the energy range from threshold to 40 MeV in comparison with IRDF-2002 and experimental data.

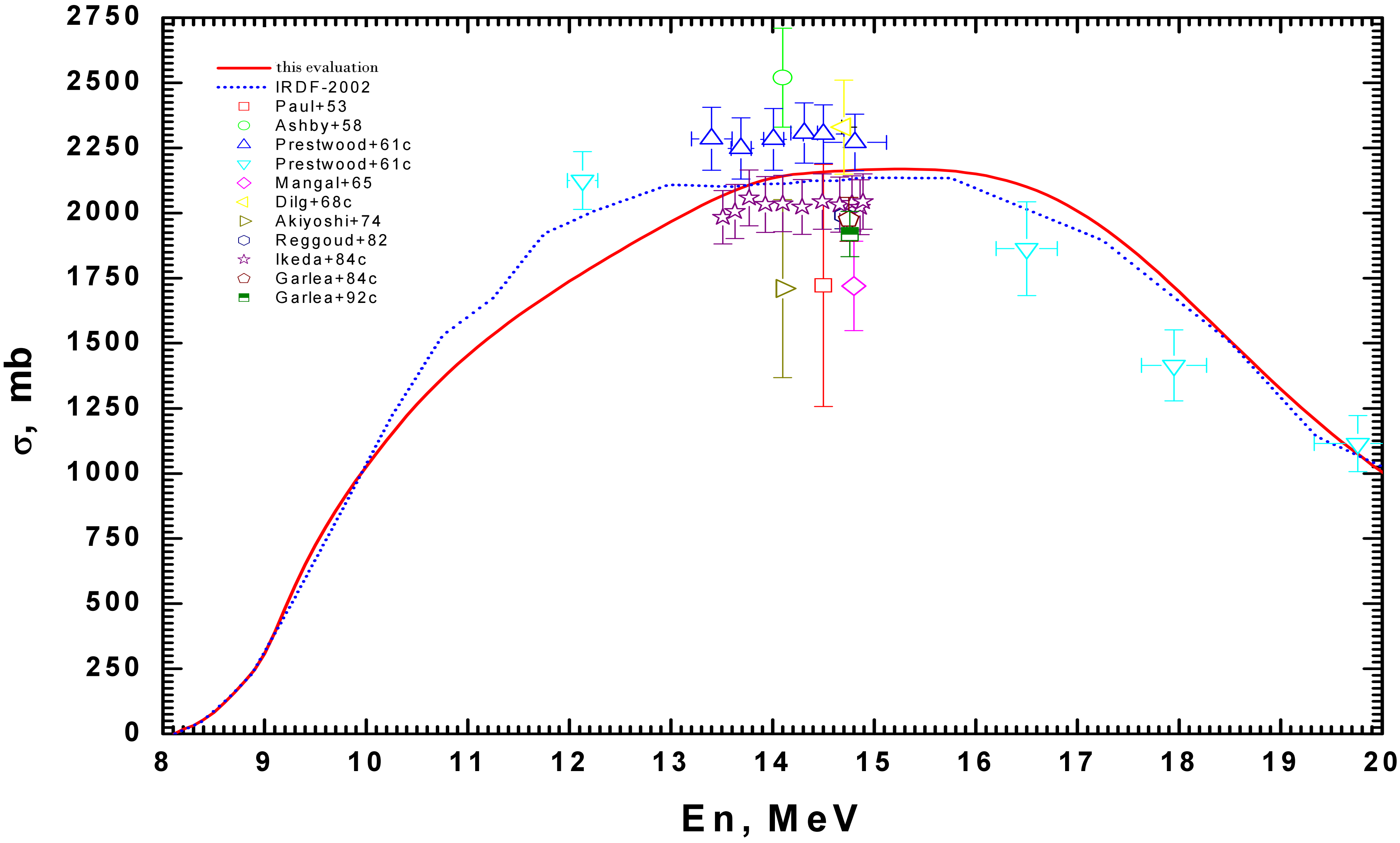

FIG. 11.2. Re-evaluated excitation function of the $^{197}Au(n,2n)^{196}Au$ reaction in the energy range from threshold to 20 MeV in comparison with IRDF-2002 and rejected experimental data.

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12. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE ^{199}Hg(n,n′)^{199m}Hg REACTION

The natural abundance of ^{199}Hg in mercury is (16.87 ± 0.22) atom percent, while ^{199m}Hg formed by the (n,n') reaction undergoes 100% IT decay with a half-live of (42.6 ± 0.2) minutes. This 42.6-min. isomeric state is the seventh excited level in ^{199}Hg, with an energy of 532.48 keV and spin and parity of 13/2+. Decay of the metastable state to the ground level is accompanied by the emission of 118.6-, 158.3-, 255.0-, 374.1- and 413.4-keV gamma rays. The ^{199}Hg(n,n')^{199m}Hg reaction rate is normally determined from the quantification of the most intense gamma rays: 158.3-keV (I_γ = 0.523 ± 0.010) and 374.1-keV (I_γ = 0.138 ± 0.011). All decay data for ^{199m}Hg are taken from Refs. [2.6, 2.7] of Section 2.

Microscopic experimental data were analyzed during the preparation of the input database used to evaluate the cross sections and their uncertainties for the ^{199}Hg(n,n')^{199m}Hg reaction [12.1-12.7]. Some of these experimental data were also corrected [12.3-12.7] to conform with the new standards (see Table 2.1).

Cross-section data measured by Swan and Metzger [12.1] near the threshold for incident neutron energies of 0.57 to 2.22 MeV were renormalized to the experimental data of Sakurai *et al*. [12.6] obtained from measurements with a ^{7}Li(p,n)^{7}Be neutron source. A correction factor of Fc = 2.7151 ± 3.3%, was determined from the ratio of cross-section integrals in the overlapping energy region from 1.120 to 1.850 MeV.

All cross sections measured by Sakurai *et al*. [12.6] over the energy range from 1.79 to 6.27 MeV using neutrons from the D(d,n)^{3}He reaction were renormalized to an absolute cross section of 255.45 mb at 1.790 MeV from measurements carried out with a ^{7}Li(p,n)^{7}Be neutron source. These renormalized data exhibit good consistency with the cross sections measured by Grudzevich *et al*. in the energy range from 7.04 to 9.98 MeV [12.7]. Cross sections determined by Grudzevich *et al*. for incident neutron energies of 9.02, 9.47 and 9.98 MeV were corrected for the contribution from the ^{200}Hg(n,2n)^{199m}Hg reaction as taken from the new evaluation [12.8] (see Fig. 12.1).

Information concerning the excitation function of the ^{199}Hg(n,n')^{199m}Hg reaction above 10 MeV was obtained from experimental data [12.4-12.5] and theoretical modelling calculations undertaken by means of the modified GNASH code.

The database used to evaluate the excitation of the ^{199}Hg(n,n')^{199m}Hg reaction was assembled from microscopic experimental data [12.1, 12.3-12.7] and theoretical modelling calculations. Cross sections that had been determined in Ref. [12.2] were not taken into account during the evaluation process due to their large deviation from the other experimental data. Evaluation of the excitation function of the ^{199}Hg(n,n')^{199m}Hg reaction from threshold to 20 MeV was carried out by means of the generalized least-squares method within the PADE-2 code. Uncertainties in the excitation function are given in the form of a relative covariance matrix for 49-neutron energy groups (LB = 5). The covariance matrix of uncertainties was calculated simultaneously with the recommended cross-section data by means of the PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix in File-33 were as follows:

5.28164E-07	5.33185E-07	5.36611E-07	5.43007E-07
5.46702E-07	5.57031E-07	5.64419E-07	5.79965E-07
5.95040E-07	6.13875E-07	6.40472E-07	6.74286E-07
7.13547E-07	7.64324E-07	8.24211E-07	8.75061E-07
9.53742E-07	1.06221E-06	1.20492E-06	1.37388E-06
1.56046E-06	1.72632E-06	2.61755E-06	4.10670E-06
7.22761E-06	1.14496E-05	1.52121E-05	1.91284E-05
2.72551E-05	4.10732E-05	5.55609E-05	1.85727E-04
4.40740E-04	7.74972E-04	9.81164E-04	1.51328E-03
1.59919E-03	2.23009E-03	2.42447E-03	2.63282E-03
3.01975E-03	4.50000E-03	5.93343E-03	6.70480E-03
8.69636E-03	1.22207E-02	2.41828E-02	4.90999E-02
9.69539E-02			

Evaluated group cross sections and their uncertainties for the ^{199}Hg(n,n')^{199m}Hg reaction are liested in Table 12.1. Group boundaries of the neutron energy groups are the same as in File-33. The smallest uncertainties in the evaluated cross sections of 2.99% to 3.51% are observed in the neutron energy range from 6.0 to 9.0 MeV, while the largest uncertainty of 21.51% occurs from threshold to 0.6 MeV and arises from the significant discrepancies in the experimental data. Inadequate experimental information from 10 to 20 MeV neutron energy result in a steady increase from 4.51% to 12.35% in the cross-section uncertainty, as shown in Table 12.1.

TABLE 12.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE ^{199}Hg(n,n')^{199m}Hg REACTION IN THE ENERGY RANGE FROM THRESHOLD TO 20 MeV.

Neutron energy (MeV) from to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from to**	**Cross section (mb)**	**Uncer-tainty (%)**
0.535 - 0.600	3.837	21.51	5.400 - 5.600	755.648	4.03
0.600 - 0.800	28.453	4.95	5.600 - 5.800	762.254	3.87
0.800 - 1.000	57.263	5.07	5.800 - 6.000	765.004	3.74
1.000 - 1.200	83.327	6.00	6.000 - 6.500	769.370	3.51
1.200 - 1.400	144.416	6.20	6.500 - 7.000	771.448	3.26
1.400 - 1.600	212.059	6.69	7.000 - 7.500	772.603	3.07
1.600 - 1.800	243.698	6.07	7.500 - 8.000	775.692	2.99
1.800 - 2.000	311.361	6.47	8.000 - 8.500	778.714	3.14
2.000 - 2.200	376.814	8.77	8.500 - 9.000	768.013	3.49
2.200 - 2.400	404.106	7.11	9.000 - 9.500	718.715	3.87
2.400 - 2.600	433.973	6.11	9.500 - 10.000	622.867	4.51
2.600 - 2.800	467.780	5.53	10.000 - 10.500	510.321	5.55
2.800 - 3.000	499.076	5.20	10.500 - 11.000	413.375	6.23
3.000 - 3.200	530.636	5.01	11.000 - 11.500	340.381	6.32
3.200 - 3.400	561.994	4.89	11.500 - 12.000	286.544	6.18
3.400 - 3.600	589.275	4.87	12.000 - 13.000	229.658	6.06
3.600 - 3.800	617.008	4.82	13.000 - 14.000	176.696	6.26
3.800 - 4.000	640.275	4.83	14.000 - 14.500	147.689	6.69
4.000 - 4.200	662.505	4.80	14.500 - 15.000	132.296	7.13
4.200 - 4.400	683.662	4.72	15.000 - 16.000	113.734	7.99
4.400 - 4.600	700.120	4.68	16.000 - 17.000	95.100	9.30
4.600 - 4.800	716.680	4.55	17.000 - 18.000	82.409	10.36
4.800 - 5.000	728.612	4.46	18.000 - 19.000	74.508	11.11
5.000 - 5.200	739.594	4.33	19.000 - 20.000	70.535	12.35
5.200 - 5.400	749.748	4.16			

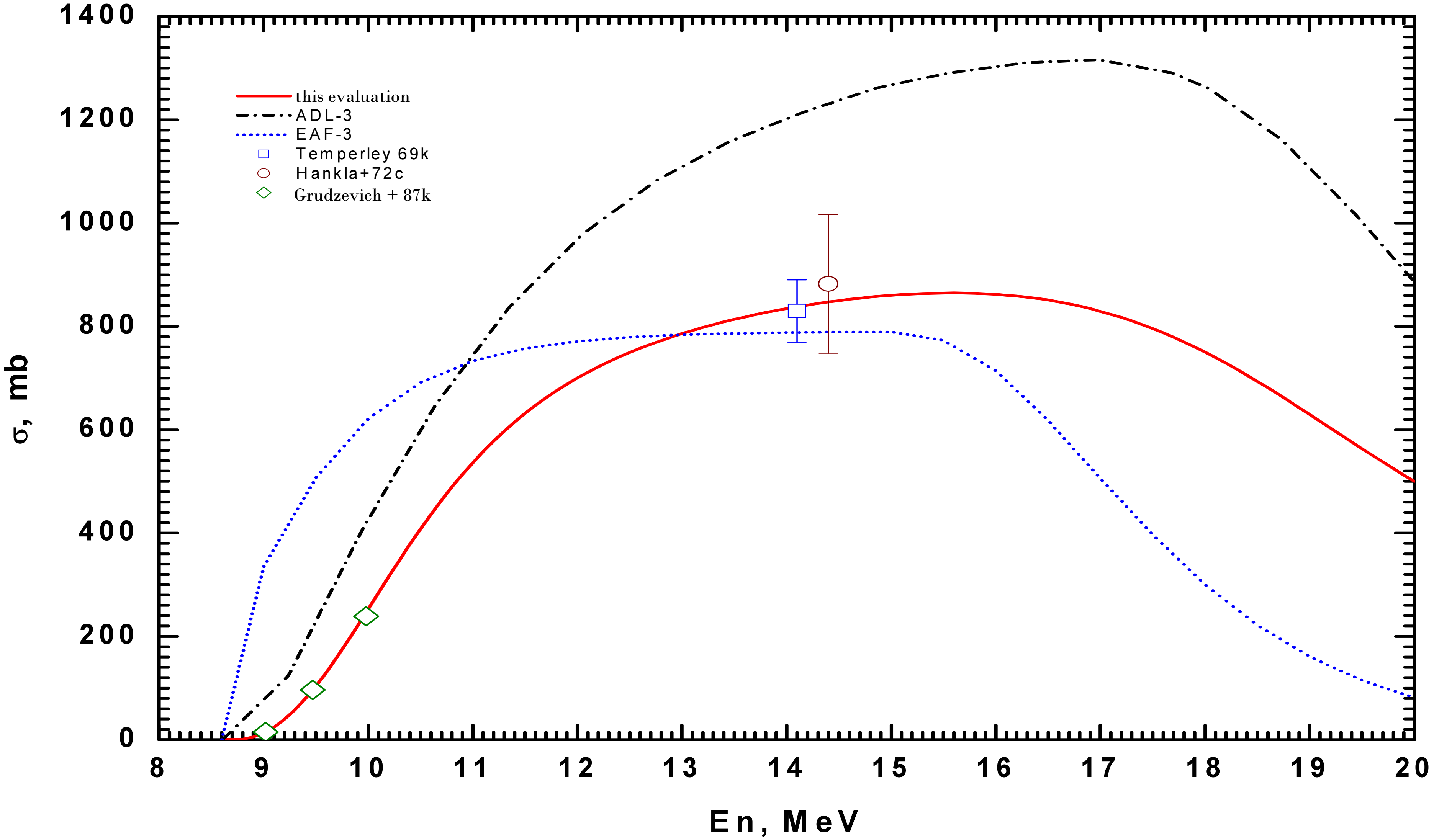

FIG. 12.1. Evaluated excitation function for the $^{200}Hg(n,2n)^{199m}Hg$ reaction from threshold to 20 MeV in comparison with ADL-3, EAF-3 and experimental data.

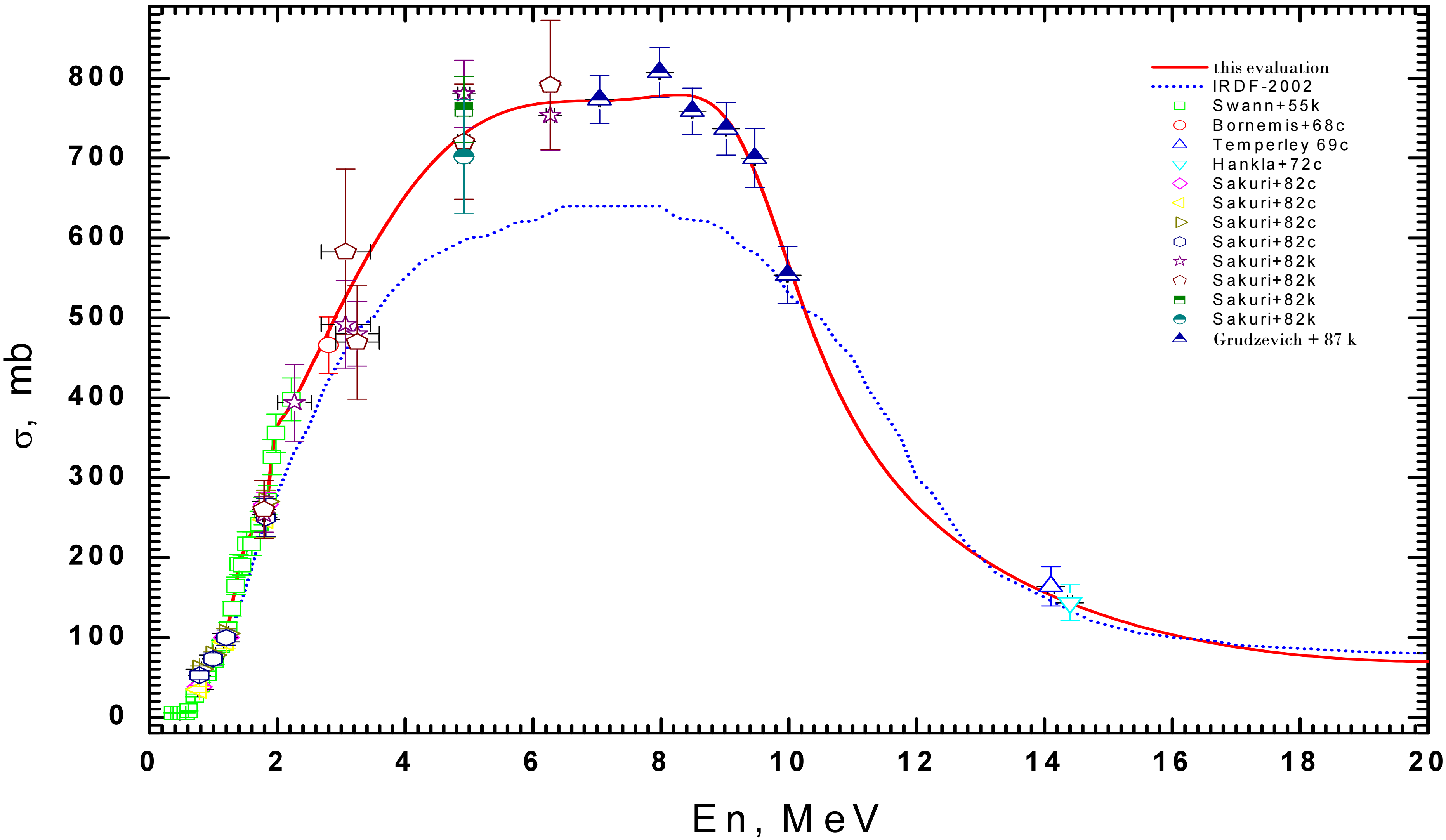

FIG. 12.2. Re-evaluated excitation function of the ^{199}Hg(n,n')^{199m}Hg reaction in the energy range from threshold to 20 MeV in comparison with IRDF-2002 and experimental data.

The re-evaluated excitation function for the ^{199}Hg(n,n')^{199m}Hg reaction is shown in Fig. 12.2 over the neutron energy range from 0.5 to 20.0 MeV in comparison with the IRDF-2002 excitation function and the experimental data. These data show that IRDF-2002 underestimates the cross sections of the ^{199}Hg(n,n')^{199m}Hg reaction systematically from 1.2 to 10.0 MeV, while differences are negligible above 13 MeV.

The integral cross section for the ^{199}Hg(n,n')^{199m}Hg reaction has been measured in ^{235}U thermal fission neutron spectra [12.9-12.12]. A fission converter with a 90-percent enriched ^{235}U plate was used in Refs. [12.9, 12.10], and an averaged cross section for the ^{199}Hg(n,n')^{199m}Hg reaction of (278.7 ± 14.8) mb was derived. Mannhart measured the cross section averaged over a ^{252}Cf spontaneous fission neutron spectrum [12.13], and revised these data to produce a value of (298.4 ± 5.4) mb [12.14]. Selected integral experimental data for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were used to test the re-evaluated excitation function of the ^{199}Hg(n,n')^{199m}Hg reaction, as shown in Table 12.2. These data show that integral cross sections calculated from the re-evaluated excitation function for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are in good agreement with the experimental values. The integral cross sections determined from the IRDF-2002 data are significantly lower that the measured values.

TABLE 12.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR THE ^{199}Hg(n,n')^{199m}Hg REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	**Average cross section, mb**		**C/E**
	Calculated	**Measured**	
^{235}U thermal fission neutron spectrum	285.59 [A] 239.75 [B]	278.70 ± 14.77 [*]	1.0247 0.8602
^{252}Cf spontaneous fission neutron spectrum	296.29 [A] 248.68 [B]	298.40 ± 5.40 [12.14]	0.9929 0.8334

[A] present evaluation.
[B] IRDF-2002 (JENDL/D-99).
[*] average-weighted value obtained from experimental data [12.9, 12.10].

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13. CONCLUSIONS

Re-evaluations of cross sections and their uncertainties have been carried out for ten dosimetry reactions. Excitation functions for the ^{63}Cu(n,2n)^{62}Cu, ^{65}Cu(n,2n)^{64}Cu, ^{64}Zn(n,p)^{64}Cu, ^{115}In(n,2n)^{114m}In and ^{199}Hg(n,n')^{199m}Hg reactions were re-evaluated over the neutron energy range from threshold to 20 MeV, while the excitation functions of the ^{127}I(n,2n)^{126}I and ^{197}Au(n,2n)^{196}Au reactions were re-evaluated in the energy range from threshold to 32 and 40 MeV, respectively. Compared with IRDF-2002, the upper neutron energy boundary for the ^{32}S(n,p)^{32}P, ^{24}Mg(n,p)^{24}Na and ^{60}Ni(n,p)$^{60m+g}$Co reactions was increased from 20 to 21 MeV. Uncertainties in the cross sections for all of these reactions are given in the form of relative covariance matrices.

Benchmark calculations performed for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra show that the integral cross sections calculated from the newly evaluated excitation functions exhibit improved agreement with related experimental data when compared with the equivalent data from the IRDF-2002 library. Thus, the ^{24}Mg(n,p)^{24}Na, ^{32}S(n,p)^{32}P, ^{60}Ni(n,p)$^{60m+g}$Co, ^{63}Cu(n,2n)^{62}Cu, ^{65}Cu(n,2n)^{64}Cu, ^{64}Zn(n,p)^{64}Cu, ^{115}In(n,2n)^{114m}In, ^{127}I(n,2n)^{126}I, ^{197}Au(n,2n)^{196}Au and ^{199}Hg(n,n')^{199m}Hg cross-section files in ENDF-6 format should be considered as suitable candidates in the preparation of an improved version of the International Reactor Dosimetry File (IRDF).

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