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EVALUATION OF CROSS-SECTIONS FOR DOSIMETRY REACTIONS

FINAL REPORT

to the I.A.E.A. for the Contract 5516
1989 - 1991

Cai Dunjiu

Chief Scientific Investigator
Chinese Nuclear Data Centre
Institute of Atomic Energy
Beijing, China

October 1991

IAEA NUCLEAR DATA SECTION, WAGRAMERSTRASSE 5, A-1400 VIENNA

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Foreword

This document summarizes the results of a two-year contract work which was initiated by the IAEA with the Chinese Nuclear Data Centre (CNDC). Cross-sections for 15 dosimetry reactions have been evaluated. It was foreseen that some of these evaluations would be included in the future version of the International Reactor Dosimetry File - IRDF-90 after thorough testing by the leading specialists in the field. These evaluations are independent in the sense that they were made using original systematics and codes developed at the CNDC and some of their own experimental data. They also take into account the latest experimental data from other sources.

This document contains detailed documentation of these evaluations. The data on a PC diskette are available on request from the IAEA Nuclear Data Section, free of charge.

N.P. Kocherov
IAEA Nuclear Data Section

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CONTENTS

Research on the Covariance Matrix of the Evaluated Data

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Abstract

This work ,through comparing the weighted average values for experimental data with the recommended average ones in a variety of small energy interval that group the sets of reaction cross sections,obtains a system of linear equations for the adjusted factors generated by alteration for the experimental data, which makes it possible to solve the question of covariance for the evaluated data considering all physical alteration.The covariance matrices of some dosimetry reaction cross sections are calculated by this method.

Introduction

It takes assumption that we have collected the experimental data of N families (or laboratories) in energy region from E_e to E_f. There are $n_1,n_2,\cdots,n_N$ experimental data in every family,respectively.that is,there are $n=n_1+n_2+\cdots+n_N$ experimental data in total.and we have obtained the evaluated and recommended ones(y'_i ,$i=1,2,\cdots,m$).Like that,how can we get the covariance matrix correlating among them ?

Covariance Matrix and the Least square Method

Assuming that the experimental data vector in the energy region from E_e to E_f and its covariance matrix are

$$X=\begin{bmatrix} X_1 \\ X_2 \\ \cdots \\ X_n \end{bmatrix} \tag{1}$$

$$V_X=\begin{bmatrix} V_{X_1X_1} & V_{X_1X_2} & \cdots & V_{X_1X_n} \\ V_{X_2X_1} & V_{X_2X_2} & \cdots & V_{X_2X_n} \\ \cdots & \cdots & \cdots & \cdots \\ V_{X_nX_1} & V_{X_nX_2} & \cdots & V_{X_nX_n} \end{bmatrix} \tag{2}$$

Now,we divided the energy region into a lot of small intervals.E_e to E_1 is the first small energy region,E_1 to E_2 the second one,$\cdots\cdots$,E_M to E_f the M−th one.there are m_{kg}($k=1,2,\cdots,M$;$g=1,2,\cdots N$) experimental data for g−th family in k−th one.We can calculate the average values in the every one and covariance matrix correlating among them by least square method.[1,2]

$$Y_\alpha=B_\alpha^+X_\alpha=\sum_{X_i\in\{\alpha\}}B_{\alpha i}X_i \tag{3}$$

$$V_{Y_\beta Y_\alpha}=\sum_{X_j\in\{\beta\}}\sum_{X_i\in\{\alpha\}}B_{\beta j}B_{\alpha i}V_{X_jX_i} \tag{4}$$

$\{\alpha\},\{\beta\}$ is a subset that represents the experimental data in a certain small energy region,respectively.where

$$B_\alpha=(C_\alpha A_\alpha^+V_{X_\alpha}^{-1})^+ \tag{5}$$

while

$$C_\alpha=(A_\alpha^+V_{X_\alpha}^{-1}A_\alpha)^{-1} \tag{6}$$

$$A_\alpha=\begin{bmatrix} 1 \\ 1 \\ \cdots \\ 1 \end{bmatrix} \tag{7}$$

Covariance Matrix of the Evaluated data

If the average values for the recommended data in the every small energy region are

$$Y' = \begin{bmatrix} Y'_1 \\ Y'_2 \\ \cdots \\ Y'_M \end{bmatrix} \tag{8}$$

Y_α may differ from Y'_α ($\alpha = 1,2,\cdots M$),obviously,because there are much alteration for the physical reason and the theoretical calculation may be included in evaluated data.

In order to get covarience matrix correlating among Y',we assume that the function relation for Y_α and X_α is

$$Y_\alpha = Y_\alpha(X) \tag{9}$$

Considering the recommended data should relate to the altered experimental ones only,we can write equation (9) into the following expression.

$$Y_\alpha = Y_\alpha(\omega X) = Y_\alpha(\omega_1 X_1, \omega_2 X_2, \cdots, \omega_n X_n) \tag{10}$$

where, $\omega_i (i = 1,2,\cdots\cdots,n)$ is the altered factor for the experimental data .In fact,Y_α is only function in a certain subset $X_i \in \{\alpha\}$.So,equation (10) should be changed into the following form.

$$Y_\alpha = Y_\alpha(\sum_i \omega_i X_i, \{X_i \in \{\alpha\}\}) \tag{11}$$

For the same reason,Y_β is function in another subset $X_i \in \{\beta\}$.The function relation is

$$Y_\beta = Y_\beta(\sum_j \omega_j X_j, \{X_j \in \{\beta\}\}) \tag{12}$$

their differential are

$$\Delta Y_\alpha = \sum_{X_i \in \{\alpha\}} \omega_i \frac{\Delta Y_\alpha}{\Delta X_i} \Delta X_i \tag{13}$$

$$\Delta Y_\beta = \sum_{X_j \in \{\beta\}} \omega_j \frac{\Delta Y_\beta}{\Delta X_j} \Delta X_j \tag{14}$$

Then we can obtain the element for the covariance matrix V_Y from equations (13),(14).

$$V_{Y_\beta Y_\alpha} = < \Delta Y_\beta \Delta Y_\alpha >$$

$$= \sum_{X_j \in \{\beta\}} \sum_{X_i \in \{\alpha\}} \omega_i \frac{\Delta Y_\alpha}{\Delta X_i} < \Delta X_i \Delta X_j > \omega_j \frac{\Delta Y_\beta}{\Delta X_j} \tag{15}$$

From equations (2),(3) and (5),we have

$$V_{X_i X_j} = < \Delta X_i \Delta X_j > \tag{16}$$

$$B_{\beta j} = \frac{\Delta Y_\beta}{\Delta X_j}, \quad B_{\alpha i} = \frac{\Delta Y_\alpha}{\Delta X_i} \tag{17}$$

where

$$X_i \in \{\alpha\}, \quad X_j \in \{\beta\} \tag{18}$$

therefore,we get

$$V_{Y_\beta Y_\alpha} = \sum_{X_j \in \{\beta\}} \sum_{X_i \in \{\alpha\}} B_{\beta j} B_{\alpha i} \omega_i \omega_j V_{X_i X_j} \tag{19}$$

In evidence, the average values for the recommended data in the every small energy region should be characterized by least square method satisfying

$$Y'_\alpha = \sum_{X_i \in \{\alpha\}} B_{\alpha i} \omega_i X_i \tag{20}$$

Y'_α is a average value of the recommended data in subset $X_i \in \{\alpha\}$.ω_i will be obtained through the following method considered the different alteration for the experimental data in the evaluation.

(1) For alteration in a or a few certain energy regions

If alteration is carried out in the energy region from E_j to E_{j+1},that is,$X_j \in \{\gamma\}$.From equation (20),we can get

$$Y'_\gamma = \sum_{X_j \in \{\gamma\}} B_{\gamma j} \omega_j X_j \tag{21}$$

therefore

$$\omega_j = \frac{Y'_\gamma}{\sum_{X_j \in \{\gamma\}} B_{\gamma j} X_j} = \frac{Y'_\gamma}{Y_\gamma} \tag{22}$$

(2) For alteration of every family

From equation (20),we can get a system of linear equations

$$Y_1' = \sum_{X_i \in \{1\}} (\omega_1 \sum_{i=1}^{m_{11}} B_{1i} X_i + \omega_2 \sum_{i=1}^{m_{12}} B_{1i} X_i + \cdots + \omega_N \sum_{i=1}^{m_{1N}} B_{1i} X_i) \quad (23)$$

$$Y_2' = \sum_{X_i \in \{2\}} (\omega_1 \sum_{i=1}^{m_{21}} B_{2i} X_i + \omega_2 \sum_{i=1}^{m_{22}} B_{2i} X_i + \cdots + \omega_N \sum_{i=1}^{m_{2N}} B_{2i} X_i) \quad (24)$$

$$\cdots \quad \cdots \quad \cdots \quad \cdots \quad \cdots \quad \cdots \quad \cdots$$

$$Y_M' = \sum_{X_i \in \{M\}} (\omega_1 \sum_{i=1}^{m_{M1}} B_{Mi} X_i + \omega_2 \sum_{i=1}^{m_{M2}} B_{Mi} X_i + \cdots + \omega_N \sum_{i=1}^{m_{MN}} B_{Mi} X_i) \quad (25)$$

There are three cases for solution of the system of linear equations:

1.M = N,there is one exact solution in this case.

2.M 〈 N,there is not solution .But,we can enlarge M,let M is equal to N in order to get one exact solution.

3.M 〉 N,there are a lot of solutions.In fact,this is a problem of parameter estimation.But,in practice,we usually divide a family data into a few groups (for example,three groups) so as to make N = M .In other words,the ω_i may be changed into ω_{i-1} ,ω_i ,ω_{i+1} and they have relation: $\omega_{i-1} = \omega_i = \omega_{i+1}$.

Finally,we can get the covariance matrix for the evaluated data from equation (19) by the altered factors obtained.

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THE EVLAUATION OF $^{23}Na(n,\gamma)^{24}Na$ REACTION CROSS SECTION

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The evaluation of the neutron capture cross section of ^{23}Na is of considerable importance to the liquid metal fast breeder reactor design and its safe operation. Moreover, Sodium is an important reference material, it can be combined with many other elements to form highly pure salts. And the evaluated data of this cross section could serve as a standard for many other cross section measurements and neutron flux measurement by employing sodium compound. For this reason, the measurement and evaluation of this cross section have been still followed with interest by ones to this day. The measurements[1~4] of the neutron capture cross section of ^{23}Na are listed in Table 1. From Table 1 it can be seen :

- Most of the measurements were carried out for the thermal neutrons, 24 keV neutrons and 14 MeV neutrons. The neutron radiative capture cross section data for other energy regions are quite sparse and the neutron resonance parameters should be adopted to describe the cross section behaviour in the resonance energy region.
- A few of the new experimental data are available. There are only two measurements for thermal neutrons[1,2] and one measurement for 14.7 MeV neutrons reported after 1980.

On the other hand, the " new" evaluated data available are those of ENDF / B-VI (distributed in January 1990) and IRDF-85. The evaluated data of ENDF / B-VI were converted from ENDF / B-V, and those of IRDF-85 were derived from the ENDF / B-V Dosimetry File directly. That means that the sourse of these data files is the same as ENDF / B-V. The evaluation of neutron capture cross section of ^{23}Na for ENDF / B-V was finished by D. C. Larson[42]. As a result ,we take a review to the data given by ENDF / B-VI as the point of departure of our evaluation.

10^{-5} eV ~ 500 keV NEUTRON ENERGY REGION

In Larson's evaluation, resonance parameters were used from 600 eV to 500 keV. Using resonance parameters E_o = 2.81 keV, Γ_n = 376 eV and Γ_γ = 0.353 eV for the large 2.81 keV resonance and Breit–Wigner formula the correct thermal capture value was given, and this form was used to calculate the capture cross section from 10^{-5} to 600 eV. The thermal capture cross section was given as 528 mb.In consideration of facts that no new experimental data of neutron resonance parameters concerned are available after 1980, and the selection of neutron resonance parameters in Larson's evaluation is still reasonable in the light of present point of view, we put our attention on reevaluation of the thermal neutron capture cross section. As shown in Table 1, there are more than 20 measurements for the thermal neutrons. In order to reduce the scope, we have taken those results with a quoted error of less than ± 3 % and made renormalizations for these data (as listed in Table 2). The standard cross sections for thermal neutrons adopted in present evaluation are as follows :

$\sigma_\gamma(^{197}Au)$ = 98.65 ± 0.09 b [43]

$\sigma_n(B)$ = 761 ± 3 b for ANL – BNL Boron [44]

773 ± 3 b for Harwell Boron[44]

A weighted average value of 0.527 ± 0 .006 b (where the error is external error) for thermal neutron capture cross section of ^{23}Na is obtained. This value is in agreement with that of ENDF / B–VI, but is a little lower than the recommended data given by Mughabghab (0.530 ± 0.005 b)[43] and Gryntakis et al. (0.530 ± 0.007 b)[45]. It is comprehensible because the lower new experimental value of 0.513 ± 0.0041 b[1] has taken in our evaluation into account. We believe that the value of 0.527 ± 0.006 b for thermal neutron capture cross section of ^{23}Na is reasonable on the level of our present understanding. This shows that the evaluated data of ENDF / B–VI in 10^{-5} eV ~ 500 keV neutron energy region are acceptable too. Consequently, they are adopted in our evaluation .

0.5 MeV ~ 20 MeV NEUTRON ENERGY REGION

Comparing the ENDF / B–VI data with the experimental data, it can be seen, the ENDF / B–VI data are in good agreement with the experimental data in the neutron energy region from 0.5 to 0.9 MeV, and a obvious discrepancy exists in the high energy region. Especially, the trend of cross section curve of ENDF / B–VI is totally different from the experimental data given by Menlove et al.[12] and Csikai et al.[11]. That is to say, according to the evaluated data given by ENDF / B–VI, the radiative capture cross section of ^{23}Na increase with increasing neutron energy above 14 MeV region, however, the experimental data are tending to decrease. In order to judge what is the proper trend of the excitation curve, we make a systematics calculation by using a systematics formula[46] which is based on the statistical theory and exciton model. The calculation indicates that there is a giant resonance existed above 10 MeV and the cross section trend is tending to decrease above 14 MeV which is in good agreement with Menlove's data[12]. Fig. 1 shows the result of the systematics calculation compared with experimental data.

In view of this situation, a reevaluation based on the experimental data has been carried out.

Eight experimental data sets measured by Magnusson et al.[3], Sigg[5], Holub et al.[7], Csikai et al.[11], Menlove et al.[12], Bame Jr. et al.[22], Perkin[25] and Reese Jr. et al.[35] are selected and renormalized by using the new standard cross section. The new standards adopted are the evaluated data of ENDF / B–VI for ^{235}U fission cross section in the neutron energy region from 500 keV to 20 MeV and the recommended data given by Vonach[47] for ^{27}Al (n,α) reaction cross section in 14 MeV neutron energy region. A data treatment code[48] is used to make the curve fitting with polynomial. Combining the fitted data in 0.9 to 20 MeV region with the ENDF / B–VI data in 0.5 ~ 0.9 MeV region, the evaluated data of neutron capture cross section of ^{23}Na from 0.5 to 20 MeV are obtained and listed in appendix. A comparison of evaluated and measured data is shown in Fig. 2.

COVARIANCE

Uncertainty files for the capture cross section are estimated from the experimental error and the spread in the various data sets.

They are estimated as :

2 % for 10^{-5} ~ 50 eV neutrons

5 % for 50 ~ 600 eV neutrons

10 % for 600 eV ~ 500 keV neutrons

20 % for 500 keV ~ 5 MeV neutrons

25 % for 5 ~ 20 MeV neutrons

DISCUSSION

A comparison of measured and calculated spectrum – averaged neutron capture cross section of ^{23}Na[49~52] is presented in Table 3. The differential cross sections used for their calculations were taken from IRDF–82 or ENDF/B–V. As mentioned above, they are the same as those of ENDF/B–VI.

As can be seen from the table, the agreement between calculated and measured values is rather poor, and the values are contradictory each other for the different benchmark spectra. It shows that more precise knowledge of these spectra and further measurements are still needed. It looks as if our evaluation would be closer to Lamaze's value[51] than that of ENDF/B–VI.

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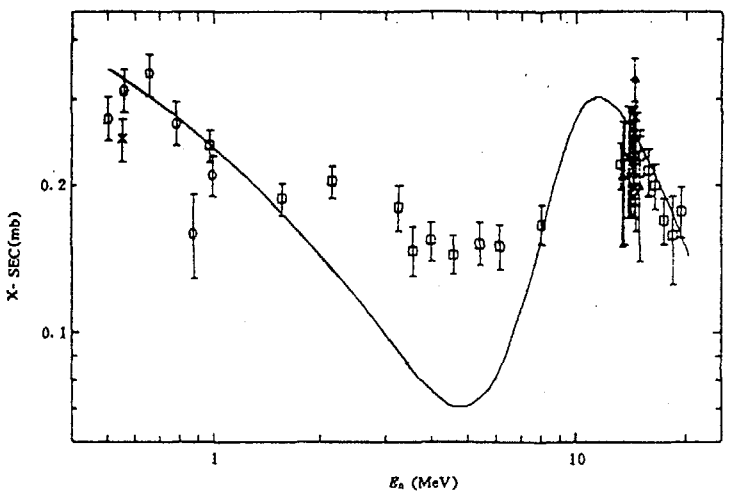

Fig.1 Systematics calculation of $^{23}Na(n,\gamma)$ reaction cross section compared with experimental data

□ Menlove(67) x Reese(53) ⊼ Sigg(76)
○ Bame(59) ◇ Holub(72) z Magnusson(82)
△ Csikai(67) ▲ Perkin(58) — Systematics calculation

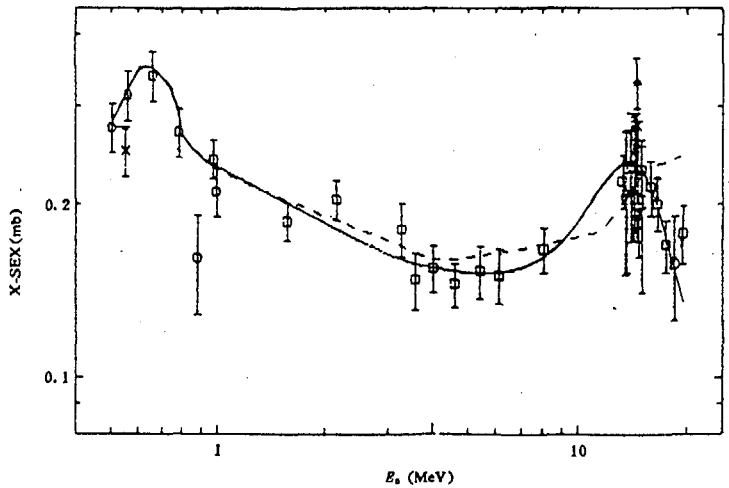

Fig.2 Comparison of evaluated and measured data

□ Menlove(67) x Reese(53) ⊼ Sigg(76) — Recommended Data
○ Bame(59) ◇ Holub(72) z Magnusson(82) ---- ENDF/B–VI
△ Csikai(67) ▲ Perkin(58)

Table 1

Servey of experimental measurements for $^{23}Na(n,\gamma)^{24}Na$ cross section

Yr	Lab	Author	Points	Value & / or Range	Standard	Ref.
88	INW	De Corte+	1	0.513 b at thermal	$^{197}Au\ \sigma_{n,\gamma}$	1
82	JPN	Kaminishi+	1	0.577 b at thermal		2
80	LND	Magnusson+	1	0.190 mb at 14.70 MeV	$^{197}Au\ \sigma_{n,2n}$	3
78	LRL	Heft	1	0.523 b at thermal		4
76	ARK	Sigg	1	0.280 mb at 14.60 MeV	$^{27}Al\ \sigma_{n,\alpha}$	5
75	OAU	Gleason	1	0.54 b at thermal	$^{197}Au\ \sigma_{n,\gamma}$	6
72	RBZ	Holub+	1	0.250 mb at 14.40 MeV	$^{56}Fe\ \sigma_{n,p}$	7
70	NPL	Ryves+	1	0.527 b at thermal	$^{197}Au\ \sigma_{n,\gamma}$	8
70	RPI	Yamamuro+	1	0.50 b at thermal	Cd $\sigma_{n,\gamma}$	9
68	MUA	Hassan+	1	1.65 mb at 24.0 keV	$^{127}I\ \sigma_{n,\gamma}$	10

Table 1

Servey of experimental measurements for $^{23}Na(n,\gamma)^{24}Na$ cross section

Yr	Lab	Author	Points	Value & / or Range	Standard	Ref.
67	DEB	Csikai+	1	0.240 mb at 14.7 MeV	$^{27}Al\ \sigma_{n,\alpha}$	11
67	DEB	Csikai+	7	13.40 MeV to 15.00 MeV		11
67	LOK	Menlove+	17	0.97 MeV to 19.39 MeV	$^{235}U\ \sigma_{n,f}$	12
66	CAD	Le Rigoleur+	18	8.55 keV to 0.134 MeV	$^{7}Li\ \sigma_{n,t}$	13
64	FEI	Bondarenko	1	at thermal	$^{235}U\ \sigma_{n,f}$	14
63	ORL	Macklin+	2	30.0 keV to 65.0 keV	I $\sigma_{n,\gamma}$	15
63	ROS	Alexander+	1	0.49 b at thermal	B σ_{s}	16
63	MUN	Koehler	1	0.50 b at thermal	B σ_{s}	17
62	ROS	Wigner	1	0.49 b at termal		18
61	ANL	Meadows+	1	0.47 b at thermal		19
60	MUN	Wolf	1	0.531 b at thermal	$^{197}Au\ \sigma_{n,\gamma}$	20
59	ORL	Lyon+	1	0.700 mb at 0.195 MeV		21
59	LAS	Bame Jr.+	19	20.00 keV to 0.990 MeV		22
59	HAR	Rose+	1	0.536 b at thermal	B σ_{s}	23
58	HAR	Jowitt+	1	0.536 b at thermal	B σ_{s}	24
58	ALD	Perkin+	1	0.330 b at 14.50 MeV	$^{3}H\ \sigma_{d,\alpha}$	25
58	FEI	Leipunskij+	3	0.200 MeV to 4.00 MeV	$^{127}I\ \sigma_{n,\gamma}$	26
58	FEI	Kononov+	1	1.710 mb at 24.0 keV	$^{127}I\ \sigma_{n,\gamma}$	27
58	LRL	Booth+	1	1.100 mb at 24.0 keV	$^{127}I\ \sigma_{n,\gamma}$	28
57	ORL	Macklin+	1	1.000 mb at 24.0 keV	$^{127}I\ \sigma_{n,\gamma}$	29
56	HAR	Cocking+	1	0.536 b at thermal	$^{197}Au\ \sigma_{n,\gamma}$	30
55	ORL	Brooksband+	1	0.50 b at thermal	$^{197}Au\ \sigma_{n,\gamma}$	31
55	KJL	Grimeland	1	0.51 b at termal	B σ_{s}	32
53	CRC	Bartholomew+	1	0.53 b at thermal	$^{197}Au\ \sigma_{n,\gamma}$	33
53	ANL	Harris+	1	0.503 b at thermal	B σ_{s}	34
53	ORL	Reese Jr.+	1	0.25 mb 500 to 600 keV		35
52	HAR	Littler+	1	0.494 b at thermal	B σ_{s}	36
51	ORL	Pomerance+	1	0.47 b at thermal	$^{197}Au\ \sigma_{n,\gamma}$	37
50	HAR	Colmer+	1	0.50 b at thermal	B σ_{s}	38
49	ANL	Hughes+	1	0.26 mb at 1 MeV	$^{23}Na\ \sigma_{n,\gamma}$	39
47	ANL	Seren+	1	0.63 b at thermal		40
47	UI	Coltman+	1	0.47 b at thermal	B σ_{s}	41

Table 2 Measured thermal neutron cross section for ^{23}Na

Author	(Yr)	Measured value, b	Adjusted value, b	Ref.
De Corte+	(88)	0.513 ± 0.0041	0.513 ± 0.0041	1
Kaminishi+	(82)	0.577 ± 0.008	0.577 ± 0.008	2
Ryves+	(70)	0.5269 ± 0.0045	0.526 ± 0.0045	8
Wolf	(60)	0.531 ± 0.008	0.531 ± 0.008	20
Rose+	(59)	0.536 ± 0.008	0.540 ± 0.008	23
Jowitt+	(58)	0.536 ± 0.008	0.539 ± 0.008	24
Cocking+	(57)	0.536 ± 0.006	0.536 ± 0.006	29
Harris+	(53)	0.503 ± 0.005	0.507 ± 0.005	34
Littler+	(52)	0.494 ± 0.015	0.538 ± 0.015	36
Colmer+	(50)	0.500 ± 0.015	0.544 ± 0.015	38

Table 3 Comparison of measured and calculated spectrum averaged cross sections for ^{23}Na

Benchmark spectra	Measured value, mb	Calculated value, mb
^{252}Cf fiss (NBS)	0.335 ± 0.015 [49]	0.2712 [50]
ISNF (NBS)	1.57 ± 0.10 [51]	1.98 [52]

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EVALUATION OF CAPTURE CROSS SECTION FOR ^{45}SC

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INTRODUCTION

Natural scandium consistes of only ^{45}Sc. The neutron capture cross section of ^{45}Sc is useful for dosimetry study. Unfortunately, only several sets of measured cross sections are available [1~9]. Therefore theory calculation or systematics have to be used in the evaluation. This evaluation was performed mainly based on the resonance parameters of ref.[10], the smooth cross sections of refs.[1 ~ 9] and the systematics [11] of the excitation function for (n,γ) reaction. A set of resonance parameters were given below 100 keV. Zhao's systematics [11] was used to fit limited measured data above 100 keV. The covariance data were estimated based on the errors of the resonance parameters, the measured cross sections and the systematics used in the evaluation.

The detail of evaluation for the resonance region is given in Chapter 2 and for the smooth region in Chapter 3. Chapter 4 reports the evaluation of the covariance data. The comparison and remarks are presented in Chapter 5.

EVALUATION OF RESONANCE PARAMETER

This evaluation is mainly based on a set of the resonance parameters evaluated by JENDL-3 [10]. In the evaluation of JENDL-3, the resonance parameter deduced by Liou et al. from their transmission measurement [12] and those by Kenny et al. from their capture measurement [1] were used. To fit the measured average capture cross sections [1], the capture widths of JENDL-3 were adjusted. The calculated average capture cross sections of ^{45}Sc from the adjusted resonance parameters are given in Tab.1 in comparison with the measured values of Kenny et al. [1], which had been corrected by multiplying a factor of 1:0737 [2]. The average cross sections calculated from the resonance parameters of JENDL-3 and ENDF / B-6 are also shown in the table. It can be found from the table that our values are in good agreement with the measured ones.

Tab.1 Comparison of Average Cross Section

Energy Interval	Average Cross Section , mb			
keV	Measured Value [1]	This Work	JENDL-3	ENDF / B-6
2 – 4	328.9 ± 31.5	330.1	331.1	302.5
4 – 7	203.3 ± 23.3	202.1	178.9	169.4
7 – 10	172.4 ± 18.3	172.7	164.5	235.5
10 – 20	114.0 ± 11.4	114.9	123.0	119.4
20 – 30	76.1 ± 7.8	75.9	72.6	73.0
30 – 40	58.5 ± 5.9	57.9	49.8	48.7
40 – 50	57.4 ± 5.8	56.5	56.0	62.8
50 – 60	46.1 ± 4.6	46.3	46.3	37.1
60 – 70	44.2 ± 4.5	44.2	44.2	51.1
70 – 80	39.5 ± 4.7	39.4	34.7	32.4
80 – 90	35.0 ± 3.5	34.7	30.0	52.7
90 – 100	34.6 ± 3.5	35.0	35.0	33.3

The parameters for two negative energy levels were also taken from JENDL-3. They reproduce the thermal cross section 27.2 b which is as same as 27.2 ± 0.2 b given by BNL-325 (1981) [13].

EVALUATION OF SMOOTH CROSS SECTION (E_n> 100 keV)

In this energy region, only few measured data are available. The systematics of excitation function for (n,γ) reaction [11] was used in this evaluation. By adjusting the parameters in the systematics formula, the curve of cross section

calculated from the systematics reproduces the measured data very well in the energy region of 2 keV to 20 MeV. The comparison of this evaluation with measured data and those of JENDL-3 and ENDF / B-6 is shown in Fig.1.

EVALUATION OF COVARIANCE DATA

The covariance data for dosimetry file play an important role in nuclear engineering. Two parts of covariance data are given.

Long-Range Component

The long-range error component of evaluated cross section is given in file 33 as follows:

1.0×10^{-5}eV ~ 1 keV: given by the error of the thermal cross section of ref.[13].

1 keV ~ 100 keV: given by the normalization error of kenny's measurement.

100 kev ~ 20 MeV: given by systematic uncertainty of the systematics used in this evaluation.

Short-Range Component

The short-range error component is given in the files 32 and 33.

1.0×10^{-5}eV ~ 100 keV: given in file 32. Only the error of the large S-wave resonance was considered. The errors for resonance energy, neutron width and capture width were taken from ref.[13]. The correlation between neutron width and capture width was neglected because neutron width is much larger than capture width for these resonance so that the capture width can be deduced from the capture area measurement directly and does not depend on neutron transmission measurement.

COMPARISON AND REMARKS

The comparison of this evaluation with JENDL-3 and ENDF / B-6 has been shown in Tab.1 and Fig.1. The agreement of this evaluation with measured data looks better than other two evaluations.

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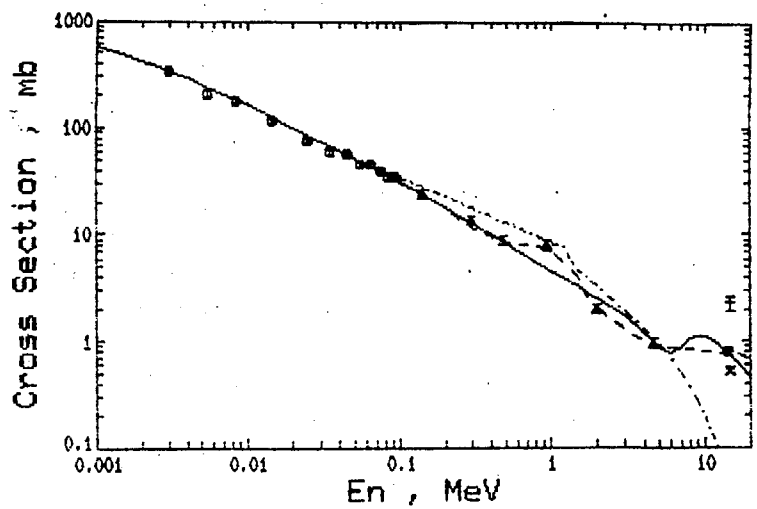

Fig.1 Cross Section of ^{45}Sc (n,gamma) Reaction

□ Kenny et al.
▲ Perkin et al.
✕ Wagner et al.
◇ Budnar et al.
+ Csikai et al.
— This work
—— CENDL - 3
– – ENDF / B-VI

Evaluation of ^{46}Ti(n,p) ^{46}Sc reaction cross section

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The excitation curve of ^{N}Ti(n,x) ^{46}Sc reaction has been evaluated. As we known, the excitation curve of ^{N}Ti(n,x) ^{46}Sc reaction varies quite slowly with neutron energy in 14–15 MeV range due to the effect of (n,np)+ (n,d) reactions for ^{47}Ti in natural sample. While the excitation curve of ^{46}Ti(n,p) reaction using isotopically–enriched sample varies quite fast down with neutron energy above 14 MeV. There were a few data in 14–15 MeV before 1979. With fusion engineering applications and nuclear structure study, the neutron source and the Ge(Li) detector have been developed in a great deal. Therefore recently some accurate experimental data above 14 MeV were reported. So reevaluation of this cross section is necessary.

The excitation curve of ^{46}Ti(n,p) ^{46}Sc reaction by using an enriched ^{46}Ti sample has been evaluated and recommended based on 12 experimental data sets [1–12]in neutron energy range of 4–19 MeV corresponding to 61 experimental points as shown in Table 1.Those experimental data were adjusted by more accurate standard cross sections and by the recent values of decay schemes , half–life , which were taken from Ref [13–17]. The details of adjusting and analyzing data are as follows:

1) The data of Lukic [4],Ghorai [5], Smith [6], Lu Hanlin [8]by natural Ti sample provided 39 energy points data below 10 MeV.The effect of (n,np)+(n,d) reactions for ^{47}Ti could be negligible due to very small ^{46}Sc productions in low neutron energies. Therefore those data below 10 MeV were adopted in the present evaluation.

2) The papers of Bormann [2], Pai [3], and Ikeda [12]provided the experimental data between 12–19 MeV. The newly measured data of Ikeda [12]were available to improve the situation of the data between 13–15 MeV.

3)The papers of Allan [1], Qaim [7], Kobayashi [11], Ribonsky[9] and Molla [10]provided the data around 14 MeV. Among those authors, Kobayashi [11]used the thick LiD target, Which might induce a large low energy scattered background and made the cross section higher in low energy range. The error of measured value of Kobayashi [11]was adjusted .

The evaluation was performed for above mentioned data by using a program of orthogonal polynomial fit on PC computer.

The recommended cross sections are given in the Fig.1 and Table 2 with 0.1, 0.2 or 0.5 MeV energy steps from threshold to 20 MeV . The estimated accuracy is 20 % near threshold, 4–10 % between 6–13 MeV, 3–5 % between 13–16 MeV ; 5–8 % above 16 MeV and 10 % up to 20 MeV.

The uncertainties and their correlations for measured quantities of the ^{46}Ti(n,p) ^{46}Sc reaction among the various data sets were analyzed and considered. The covariance file for the evaluated data was calculated using the code by Shi Zhaomin et al. [18].

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Table 1. Survey of Measured Data for Ti-46(n,p)Sc-46 Reaction

Year	Author	Point	Neutron Energy, MeV	Cross Section, mb	Adjustment Cross Section, mb
61	D.L.Allan [1]	1	14.0	203±21 196±21	
65	M.Bormann [2]	6	12.5- 17.0		
66	H.L.Pai [3]	5	13.6- 19.5		
71	Y.Lukic [4]	7	5.48- 7.04		
71	S.K.Ghorai [5]	3	4.1- 6.1		
75	D.L.Smith [6]	27	4.196- 9.950		
77	S.M.Qaim [7]	1	14.7	221±25	
79	Lu Hanlin [8]	2	4.5, 5.0		
83	I.Ribonsky [9]	1	14.8	266.7±8.5	
86	N.I.Molla [10]	1	14.8	226.2±22.4	
88	K.Kobayashi [11]	1	14.05	267.8±9.3	267.8+9.30 267.8-8.6
89	Y.Ikeda [12]	6	12.33- 14.91		

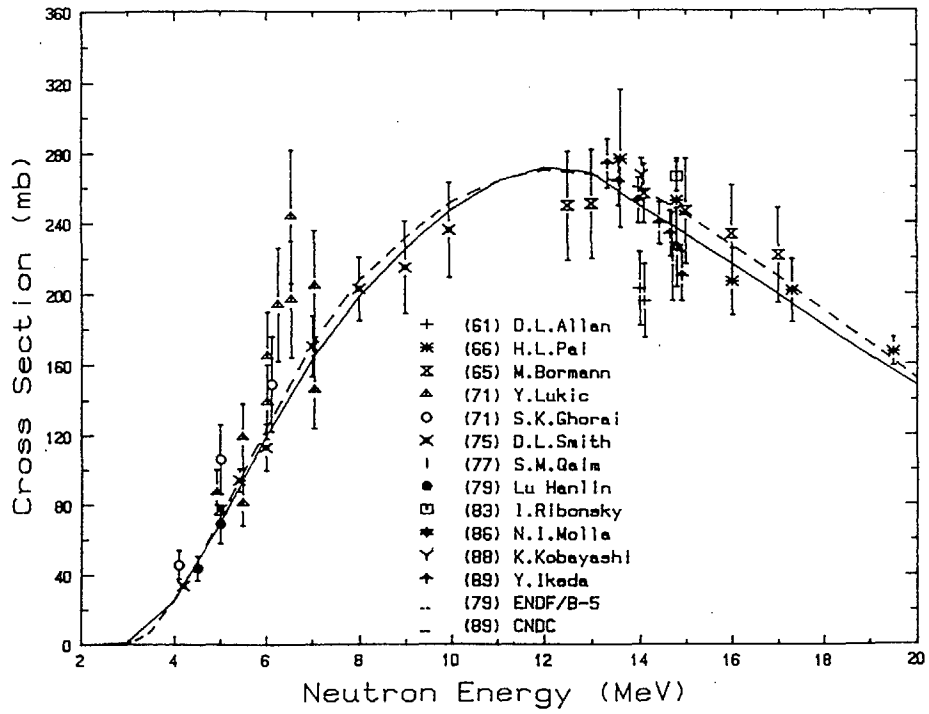

Fig.1 Ti-46(n,p)Sc-46

Table 2 Recommended cross section for Ti-46(n,p)Sc-46 Reaction

Neutron Energy MeV	Cross Section mb	Error mb	Neutron Energy MeV	Cross Section mb	Error mb
2.5	0.0	0.0	12.6	271.3	10.52
3.0	1.66	0.33	12.8	270.3	10.40
3.5	10.33	2.60	13.0	268.8	9.95
4.0	25.37	1.45	13.2	266.8	9.33
4.2	33.03	1.73	13.4	264.4	8.73
4.4	41.41	2.17	13.6	261.5	8.13
4.6	50.37	2.31	13.8	258.3	8.03
4.8	59.79	2.62	14.0	252.1	7.92
5.0	69.53	2.98	14.1	249.5	7.86
5.2	79.46	3.40	14.2	247.9	7.79
5.4	89.48	3.82	14.3	245.9	7.74
5.6	99.48	4.22	14.4	244.2	7.67
5.8	109.4	5.03	14.5	242.7	7.61
6.0	119.1	6.09	14.6	240.5	7.50
6.2	128.6	7.33	14.7	238.7	7.47
6.4	137.8	8.92	14.8	237.6	7.42
6.6	146.7	9.83	14.9	235.7	7.58
6.8	155.3	12.68	15.0	234.2	7.71
7.0	163.5	13.90	15.1	232.1	7.64
7.2	171.3	15.27	15.2	230.6	7.89
7.4	178.8	16.27	15.3	228.9	7.81
7.6	185.8	17.22	15.4	227.3	7.65
7.8	192.5	17.90	15.5	225.6	8.04
8.0	198.8	18.99	15.6	223.9	7.97
8.2	204.8	19.87	15.7	222.2	8.77
8.4	210.5	20.73	15.8	220.1	9.22
8.6	216.0	21.15	15.9	218.8	9.15
8.8	221.2	21.63	16.0	217.5	9.09
9.0	226.1	22.38	16.2	213.7	9.86
9.2	230.8	23.56	16.4	209.9	10.16
9.4	235.3	24.02	16.6	208.4	10.09
9.6	239.6	24.46	16.8	203.1	11.17
9.8	243.7	25.00	17.0	201.0	11.15
10.0	247.6	25.53	17.2	196.5	13.29
10.2	251.3	25.51	17.4	194.1	14.13
10.4	254.8	25.53	17.6	188.3	15.80
10.6	258.0	26.06	17.8	185.2	15.54
10.8	260.9	26.64	18.0	182.4	15.72
11.0	263.5	26.61	18.5	173.5	14.44
11.2	265.9	26.62	19.0	164.5	13.79
11.4	267.9	27.06	19.5	156.7	13.25
11.6	269.5	27.54	20.0	148.4	14.84
11.8	270.7	27.53			
12.0	271.6	27.49			
12.2	271.9	21.75			
12.4	271.8	12.87			

Evaluation of the ^{N}Ti(n,x) ^{46}Sc reaction cross section

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The evaluation of ^{N}Ti(n,x) ^{46}Sc reaction cross section is concerned with two problems.First, several reactions contribute to the same activity of ^{46}Sc for Ti, which contains five isotopic components of ^{46}Ti, ^{47}Ti, ^{48}Ti, ^{49}Ti and ^{50}Ti. The activity of ^{46}Sc can be produced by ^{46}Ti(n,p), ^{47}Ti(n,np), ^{47}Ti(n,d) and ^{48}Ti(n,t). Of these , only the (n,p) reaction for ^{46}Ti and (n,np)+(n,d) reactions for ^{47}Ti are significant. The second problem to be considered is the ^{46}Ti isotopic abundance of sample measured. The natural Ti contains 8 % for ^{46}Ti and 7.3 % for ^{47}Ti, which were known from nuclear wallet cards [1]. At present evaluation, the cross sections of ^{N}Ti(n,x) ^{46}Sc reaction represent the ^{46}Ti(n,p) cross section plus an additional component produced from ^{47}Ti(n,np) and ^{47}Ti(n,d) reactions.

The excitation curve of the ^{N}Ti(n,x) ^{46}Sc reaction has been evaluated and recommended based on 9 experimental data sets in neutron energy range 3–18 MeV corresponding to 77 experimental points in total collected . These data [2–9]are listed in Table 1 and Fig. 1. The collection of the experimental data was lasted to the end of 1989,most microscopic data were included . Many data are retrieved from EXFOR master files of International Atomic Energy Agency, enriched with newly informations as well as IAE experimental result.

The experimental data were analyzed and treated as the follows:

1. The papers of Cross [2], Koehler [3], Liskien [4], Lu Hanlin [8], and Viennot [9]provided the experimental data around 14MeV. Those data points around 14 MeV were adjusted for energy to equivalent 14.7 MeV cross section, which depends on the shape of the excitation curve for ^{N}Ti(n,x) ^{46}Sc reaction. In order to obtain the factors of energy adjustment values ,the data of Zhao Wenrong et al. [14]were used. The data around 14 MeV were also adjusted for nuclear decay schemes , half–life and standard cross section , which were taken from Ref. [10–14]. Some data should be rejected due to the larger discrepancies with others and exceeding the averaged value by three–standard deviation.

The evaluated data were made by the averagement with the weighted factor for the remaining data after finished adjustments mentioned above.The weighted factor was used in the evaluation,which was based on the given errors by authors and quoted errors by us. Present evaluated data value of 14.7 MeV is 281.44 ± 8 mb.

2. Among the experimental data, there are 6 multiple–value sets(about 71 points) in 4–20 MeV, which are available, besides above mentioned 6 ones around 14 MeV. The multiple–value set of D.L.Smith [2] below 10 MeV also were adjusted by standard cross section renewed [11–14]. The evaluation was performed for above mentioned data by using a program of orthogonal polynomial fit on PC computer.

The recommended cross sections are given in the Fig.1 and Table 2 with 0.1, 0.2 or 0.5 MeV energy steps from threshold to 20 MeV . The estimated accuracy is 20% near threshold, 4–10 % between 6–13 MeV, 3–4 % between 13–16 MeV, 5–8 % above 16 MeV and 10 % up to 20 MeV .

The uncertainties and their correlations for measured quantities of the ^{N}Ti(n,x) reactions among the various data sets were analyzed and considered. The covariance file for the evaluated data was calculated using the code by Shi Zhaomin et al. [15]

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Table 1. Survey of Measured Data for Ti-N (n,x)Sc-46 Reaction

Year	Author	Point	Neutron Energy MeV	Cross Section mb	Adjustment Factor 1	Adjustment Factor 2	Adjustment Cross Section, mb
63	W.G.Cross [2]	1	14.5	268±3	0.9983	1.0139	271.3
64	D.R.Koehler [3]	1	14.7	324±97.2	1.0000	1.1487	372.2
65	H.Liskien [4]	17	14.6 (12.61-16.53)	293±21	0.9992		292.8
71	S.T.Ghorai [5]	3	4.1, 5.0, 6.1				
71	Y.Lukic [6]	11	4.1-7.01				
75	D.L.Smith [7]	27	3.07-9.95				
79	Lu Hanlin [8]	9	14.58 (4.5,5, 12.33-17.97)	287.1±14	0.9990	0.9784	280.6
82	M.Viennot [9]	7	14.73 (13.8-14.8)	306±34	1.0003	1.0035	307.2

Table 2 Ti-N (n,x)Sc-46 Cross Section
Recommended Reference Data
Linear-linear Interpolation

Neutron Energy MeV	Cross Section mb	Error mb
2.5	0.0	0.0
3.0	1.68	0.33
3.5	10.33	2.26
4.0	25.37	1.45
4.2	33.03	1.73
4.4	41.41	2.17
4.6	50.37	2.31
4.8	59.79	2.62
5.0	69.53	2.98
5.2	79.46	3.40
5.4	89.48	3.82
5.6	99.48	4.22
5.8	109.40	5.03
6.0	119.10	6.09
6.2	128.60	7.33
6.4	137.80	8.92
6.6	146.70	9.83
6.8	155.30	12.68
7.0	163.50	13.90
7.2	171.30	15.36
7.4	178.80	16.27
7.6	185.90	17.23
7.8	192.80	18.12
8.0	199.30	19.04
8.2	205.50	19.93
8.4	211.40	20.82
8.6	217.10	21.31
8.8	222.60	21.77
9.0	227.80	22.78
9.2	232.80	23.78
9.4	237.60	24.28
9.6	242.30	24.75
9.8	246.70	25.31
10.0	251.0	25.89
10.2	255.0	25.75
10.4	258.9	25.94
10.6	262.6	26.83
10.8	266.0	27.17
11.0	269.2	27.22
11.2	272.1	27.25
11.4	274.8	27.78
11.6	277.3	28.33
11.8	279.4	28.38
12.0	281.2	28.45
12.2	282.8	21.03
12.4	284.0	13.49

Neutron Energy MeV	Cross Section mb	Error mb
12.6	284.9	12.25
12.8	285.5	11.67
13.0	285.9	10.81
13.2	285.9	9.88
13.4	285.7	9.29
13.6	285.2	8.88
13.8	284.5	8.85
14.0	283.7	8.83
14.1	283.6	8.79
14.2	283.5	8.82
14.3	283.4	8.78
14.4	283.4	8.82
14.5	283.4	8.81
14.6	283.3	8.81
14.7	283.2	8.83
14.8	283.0	8.83
14.9	282.8	9.05
15.0	282.2	9.23
15.1	281.9	9.44
15.2	281.9	9.71
15.3	281.8	9.69
15.4	281.8	9.72
15.5	281.7	9.72
15.6	281.6	9.73
15.7	281.6	9.77
15.8	281.5	9.79
15.9	281.3	10.41
16.0	281.2	10.48
16.2	280.8	11.43
16.4	280.9	12.39
16.6	280.8	13.20
16.8	280.7	14.14
17.0	280.6	15.99
17.2	280.6	18.19
17.4	280.5	19.60
17.6	280.4	22.84
17.8	280.4	22.85
18.0	280.4	22.84
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19.0	282.3	22.99
19.5	281.2	28.12
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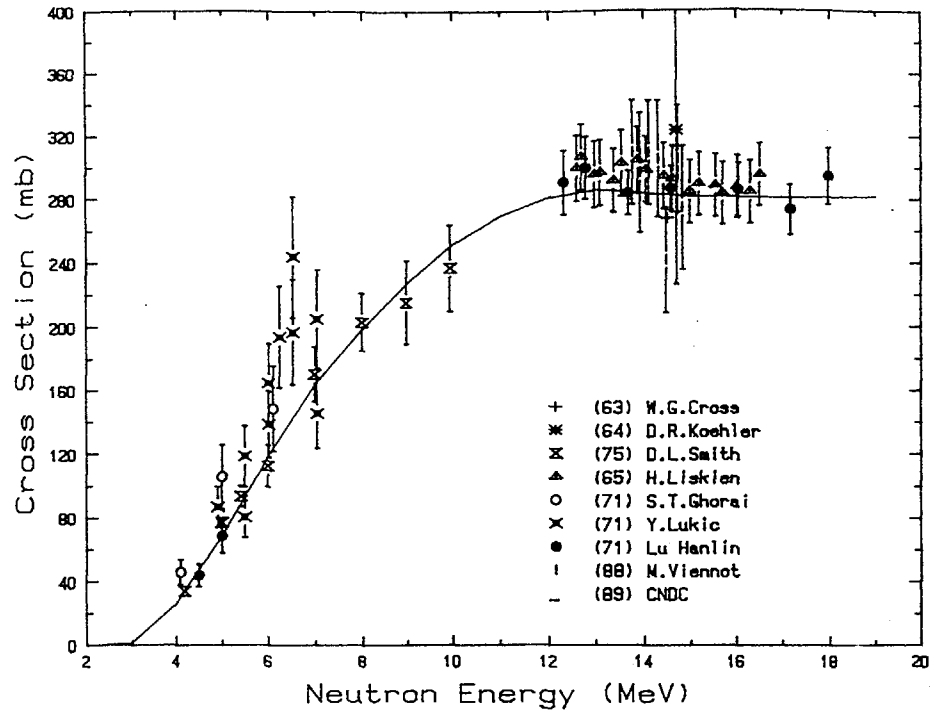

Fig.1 Ti-46(n,x)Sc-46

Evaluation of the $^{48}Ti(n,p)$ ^{48}Sc reaction cross section

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The excitation curve of the $^{48}Ti(n,p)$ ^{48}Sc reaction has been evaluated by C.Philis [1]in 1979. In the evaluation, 14 data sets and about 72 energy points have been collected. Large differences exist among the data collected from different laboratories. For fusion engineering applications and nuclear structure study,the intense pulsed neutron source and the Ge(Li) detector with high resolution have been developed in a great deal . So that the accuracy of the cross section measurement was improved , and then some important results were published. Therefore the reevaluation of this cross section is necessary.

Present work was undertaken to provide more precise and complete excitation curve of the $^{48}Ti(n,p)$ ^{48}Sc reaction in the whole energy range . The excitation curve of the $^{48}Ti(n,p)$ ^{48}Sc reaction has been evaluated and recommended based on 34 experimental data sets corresponding to 131 experimental points in total in neutron energy range 5–19 MeV. These data [2–35]are listed in Table 1 and Fig.1–1,Fig.1–2. Most of the experimental data have been included up to 1989. Many data were retrieved from EXFOR master files of International Atomic Energy Agency, enriched with new informations as well as IAE experimental result.

The experimental data were analyzed and treated as the follows:

1. Among those data sets,20 authors provided the data around 14 MeV only. All of the data including 11 authors of multiple–value set around 14 MeV exist 31 energy points. Those cross sections around 14 MeV were adjusted for energy to equivalent 14.7 MeV cross section, which depends on the shape of the excitation curve for $^{48}Ti(n,p)$ ^{48}Sc reaction. In order to obtain the factors of energy adjustment values,the data of Zhao Wenrong et al. [39]were used. The data around 14 MeV were also adjusted for nuclear decay schemes , half– life and standard cross section , which were taken from Ref. [38,39]. Some data should be rejected due to the larger discrepancies with others and exceeding the averaged value by three standard deviation .

The evaluation data were made by the averagement with the weighted factor for the remaining data after finished adjustments mentioned above. The weighted factors in the evaluation were based on the given errors by authors and quoted errors by us. Present evaluated data is 62.39 ± 0.82 mb at 14.7 MeV , as shown in Fig. 2.

2. In the evaluated excitation function , 13 multiple–value sets (about 96 points) in 4–20 MeV are available , besides above mentioned 33 ones around 14 MeV . The multiple–value set of D.L.Smith [18]below 10 MeV also were adjusted by standard cross section renewed [38–40]. The evaluation was performed for above mentioned data by using a program of orthogonal polynomial fit on PC computer .

The recommended cross sections are given in the Fig.1–1, Fig.1–2 and Table 2 with 0.1 , 0.2 or 0.5 MeV energy steps from threshold to 20 MeV . The estimated accuracy is 30% near threshold , 5–7% below 14 MeV , 1–3 % between 14–16 MeV, 4–6 % above 16 MeV and 7 % up to 20 MeV .

Variance and Covariance Matrix

The uncertainties and their correlations for measured quantities of the $^{48}Ti(n,p)$ reaction among the various data sets were analyzed and considered. The described covariance for the evaluated cross section of $^{48}Ti(n,p)$ reaction from threshold energy to 20 MeV is divided into 8 energy intervals.

The covariance matrix was calculated by the code of Shi Zhaomin et al. [41], the uncertainty correlation matrix for $^{48}Ti(n,p)$ ^{48}Sc reaction cross section evaluation is shown in the table 3 .

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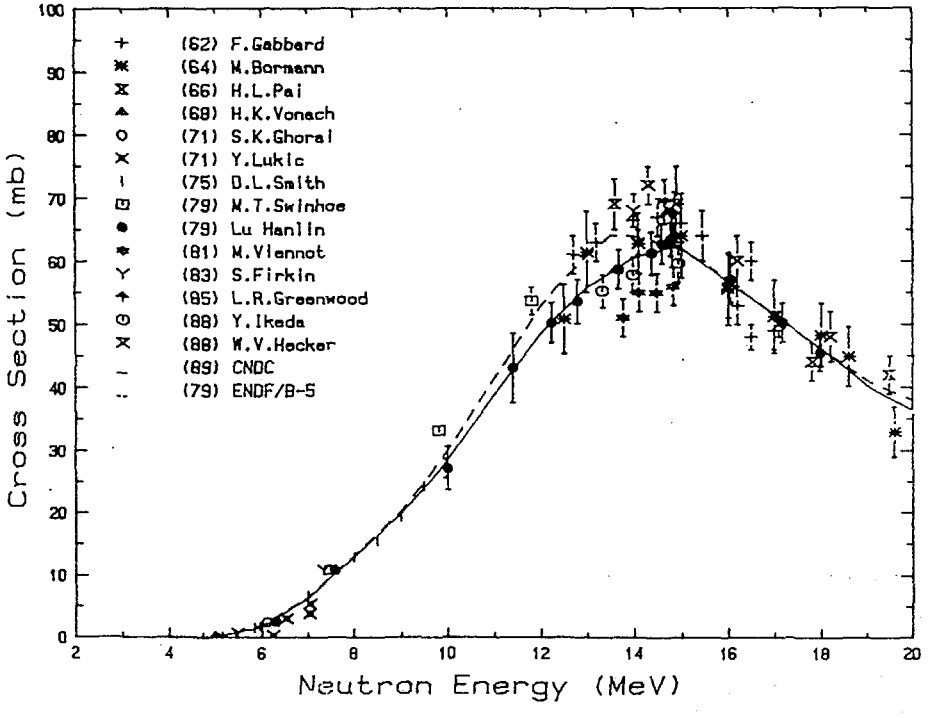

Fig.1-1 Ti-48(n,p)Sc-48

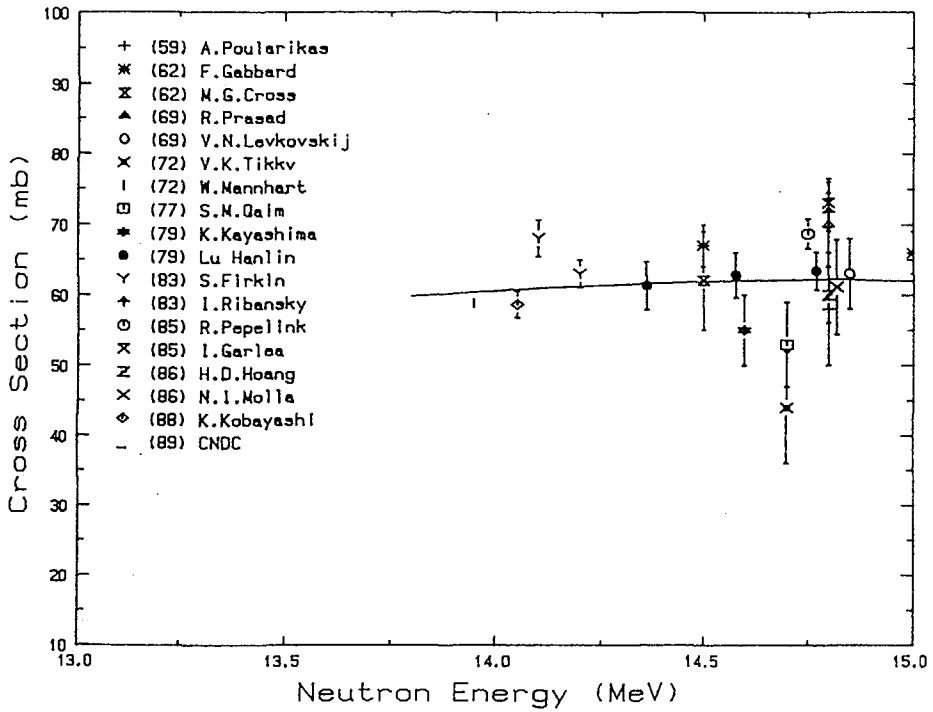

Fig.1-2 Ti-48(n,p)Sc-48

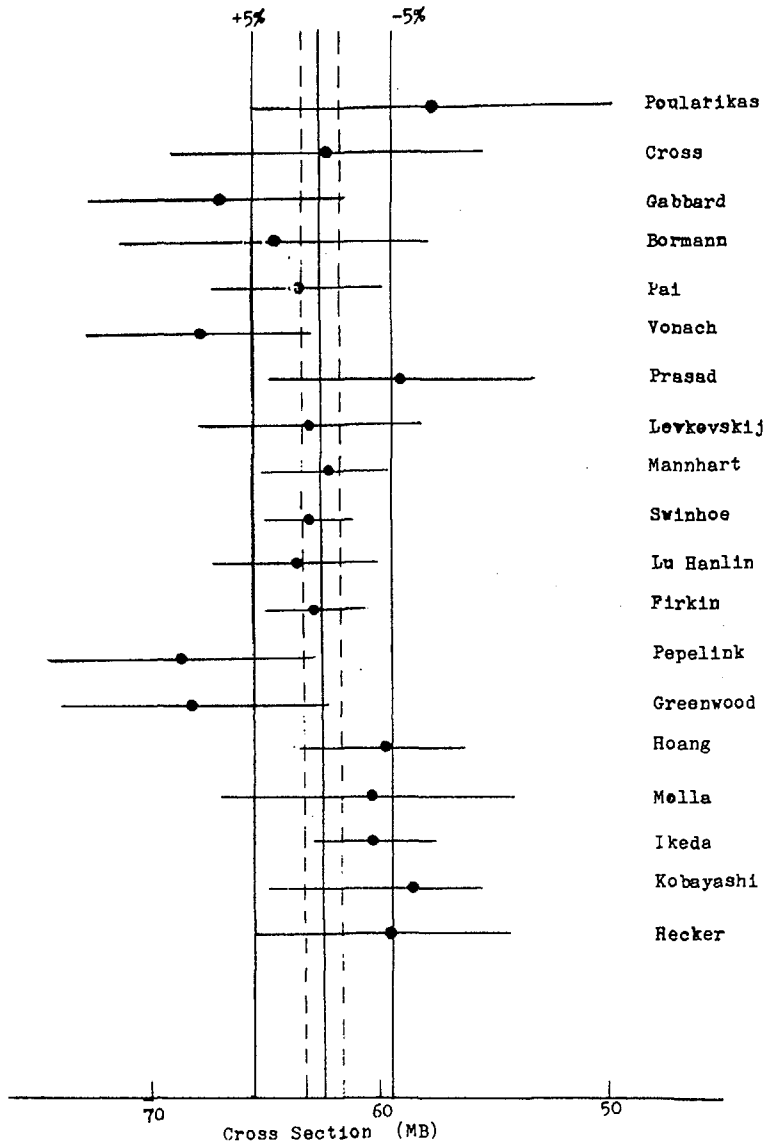

Fig.2 Evaluation of 14.7 MeV Cross Section for $^{48}Ti(n,p)^{48}Sc$

Table 1. Survey of measured data for Ti-48(n,p)Sc-48 Reaction

Year	Author	Point	Energy MeV	Cross Section mb	Adjustment Factor 1	Factor 2	Adjusted Cross Secton, mb
53	Paul [2]	1	14.5	92.7±32.4	0.9976		
59	Poularikas [3]	1	14.8	58±8	1.0024	0.9914	57.6
62	Gabbard [4]	14	14.5 (12.7-17)	62±3	0.9976		66.8
62	Hillman [5]	1	14.5	51±11	0.9976	1.0139	
63	Cross [6]	1	14.5	62±7	0.9976	1.0139	62.2
64	Koehler [7]	1	14.7	32.4±9.7	1.0000	1.1400	
65	Bormann [8]	9	15.0 (12.5-19.6)	64±6.7	1.0072		64.5
65	Strain [9]	1	14.7	132	1.0		
66	Pai [10]	5	14.8 (13.6-19.5)	63±4	1.0024	1.0052	63.4
68	Vonach [11]	12	14.8 (13.6-14.6)	66.5±2.5	1.0	1.0197	67.8
69	Crumpton [12]	1	14.7	80±4	1.0		
69	Prasad [13]	1	14.8	70±6	1.0024	0.8437	59
69	Levkovskij [14]	1	14.8	63±5	1.0024		63
71	Ghorai [15]	1	6.1				
71	Lukic [16]	5	6.53-7.04				
72	Tikku [17]	1	14.7	44±8	1.0	0.9974	
75	Smith [18]	19	4.72-9.95				
75	Mannhart [19]	1	13.95	58.8±0.8	1.0056	1.0325	62.1
75	Spangler [20]	1	14.1	66±7	1.0032	1.0652	

Continued Table 1

Year	Author	Point	Energy MeV	Cross Section mb	Adjustment Factor 1	Factor 2	Adjusted Cross Secton, mb
77	Qaim [21]	1	14.7	53±6	1.0	0.9397	
79	Swinhoe [22]	4	14.1 (5.8-14.1)	63±2	1.0032	1.0041	63
79	Kayashima [23]	1	14.6	55±5	0.9976		
79	Lu Hanlin [24]	14	14.73 (5-18.0)	63.4±3.7	1.0024	1.0000	63.6
81	Viennot [25]	7	14.73 (13.8-14.9)	51±3	1.0024	1.0056	
83	Firkin [26]	6	14.1 (5.47-14.1)	63±2	0.9976		62.8
83	Ribansky [27]	1	14.8	71.7±2.7	1.0024		
85	Pepelink [28]	1	14.7	68.7±2.1	1.0000		68.7
85	Greenwood [29]	5	14.65 (14.5-14.9)	69.4±1.04	0.9980		68.2
85	Garlea [30]	1	14.8	73±3.5	1.0024	0.9651	
86	Hoang [31]	1	14.8	60±4	1.0024	0.9930	57.9
86	Molla [32]	1	14.8	61.1±6.7	1.0024	0.9843	60.3
88	Ikeda [33]	5	14.67 (13.3-14.9)	60.4±2.9	0.9991		60.3
88	Kobayashi [34]	1	14.05	58.6±1.88	1.0028	0.9972	58.6
88	Hecker [35]	5	14.9 (14.3-18.2)	59±6	1.0048	1.0081	59.7

Factor 1: Adjustment for neutron energy

Factor 2: Adjustment for standard cross section , half-life and gamma branching ratio

Table 2 Recommended cross section for Ti-48(n,p)Sc-48 Reaction

Neutron Energy MeV	Cross Section mb	Error mb	Neutron Energy MeV	Cross Section mb	Error mb
4.0	0.0	0.0	13.2	56.91	3.35
4.5	0.03	0.01	13.4	57.96	3.06
5.0	0.08	0.01	13.6	58.92	3.11
5.2	0.16	0.02	13.8	59.79	2.80
5.4	0.33	0.03	14.0	60.56	2.01
5.6	0.61	0.06	14.1	60.91	2.01
5.8	1.03	0.07	14.2	61.22	1.02
6.0	1.59	0.11	14.3	61.49	0.81
6.2	2.31	0.16	14.4	61.72	0.80
6.4	3.18	0.24	14.5	61.91	0.81
6.6	4.17	0.32	14.6	62.05	0.81
6.8	5.28	0.40	14.7	62.13	0.81
7.0	6.47	0.51	14.8	62.16	0.81
7.2	7.73	0.46	14.9	62.12	0.81
7.4	9.02	0.49	15.0	62.02	1.04
7.6	10.33	0.56	15.1	61.84	1.05
7.8	11.65	0.58	15.2	61.59	1.04
8.0	12.97	0.65	15.3	61.26	1.22
8.2	14.30	0.71	15.4	60.85	1.82
8.4	15.65	0.78	15.5	60.36	1.87
8.6	17.01	0.85	15.6	59.79	1.86
8.8	18.42	0.92	15.7	59.14	1.84
9.0	19.88	0.99	15.8	58.41	1.82
9.2	21.41	1.07	15.9	57.61	1.79
9.4	23.03	1.17	16.0	56.74	2.32
9.6	24.74	1.25	16.2	55.83	2.24
9.8	26.54	1.34	16.4	54.17	2.64
10.0	28.45	1.43	16.6	53.62	2.53
10.2	30.41	1.48	16.8	52.61	2.67
10.4	32.46	1.56	17.0	51.62	2.80
10.6	34.55	1.68	17.2	50.28	2.70
10.8	36.68	1.83	17.4	49.24	2.60
11.0	38.80	1.97	17.6	48.20	2.60
11.2	40.89	2.06	17.8	47.69	2.64
11.4	42.94	2.53	18.0	46.13	2.67
11.6	44.91	2.64	18.5	43.39	2.75
11.8	46.79	2.81	19.0	40.26	2.36
12.0	48.57	3.47	19.5	37.84	2.28
12.2	50.23	3.56	20.0	36.37	2.34
12.4	51.78	3.60			
12.6	53.22	3.27			
12.8	54.55	3.32			
13.0	55.78	3.29			

Table 3 Uncertainty correlation matrix for ^{48}Ti(n,p) ^{48}Sc reaction cross section evaluation

1 = 4.5–5.0MeV 2 = 5.0–6.0MeV

3 = 6.0–8.0MeV 4 = 8.0–11.0MeV

5 = 11.0–13.0MeV 6 = 13.0–14.0MeV

7 = 14.0–16.0MeV 8 = 16.0–20.0MeV

	1	2	3	4	5	6	7	8
1	1.000							
2	0.345	1.000						
3	0.349	0.862	1.000					
4	0.346	0.850	0.859	1.000				
5	0.024	0.407	0.165	0.167	1.000			
6	0.019	0.066	0.048	0.062	0.469	1.000		
7	0.019	0.063	0.068	0.065	0.377	0.386	1.000	
8	0.022	0.083	0.252	0.087	0.466	0.429	0.350	1.000

Evaluation of the $^{55}Mn(n,2n)$ ^{54}Mn reaction cross section

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Twenty–four references on $^{55}Mn(n,2n)$ ^{54}Mn reaction cross section data were compiled from literature, but only six references[5,6,11,17,21,23] provided the multiple–value sets in the neutron energy range from 12 to 20 MeV. These results are listed in the Table 1 and plotted in fig.1, except the value of B.Granger [4], which was much bigger than all the others. Eighty–seven cross section values were found from this compilation, Three group data are appeared in the Fig.1, the higher group data included the values of A.Paulsen [5] and V.J.Ashby [1], which are about 9–38% higher than that of the middle group. The middle consentaneous data were given by H.O.Menlove [6], Lu Hanlin [17], L.R.Greenwood [21] and Y.Ikeda [23] in the neutron energy range 12–20MeV. The lower values were given by four authors (R.Wenusch [3], D.A.Salnikov [13], J.Araminowicz [14] and K.Deak [15]), whose data agreed with recommended data of ENDF / B, MAT 1019,1967.

Before evaluation, the adjustments were required for neutron energy to generate equivalent 14.7MeV value and reference cross sections, which were taken from Zhao Wenrong [25] and B.P.Evain [26], if enough informations were provided by the authors. The data of H.O.Menlove, Lu Hanlin and Y.Ikeda have consistant shapes of cross sections, it's taken to make a curve for neutron energy adjustment.

The decay data of ^{54}Mn is very well known and the values of half–life(312.2d) and gamma–intensity(0.99975) for 834keV have not changed to any significant extent for many years. The abundance 100% of isotope ^{55}Mn offers no problems.

After adjustment, the first–pass evaluation was done for twenty–four cross section values at 14.7MeV. Seven data(1,3,4,5,13,14 and 15) were subsequently rejected because they exceeded by three enhanced standard deviations from the ensemble average. A continual evaluation was performed for seventeen remained cross section values. We added a additional error to quoted random error for some results due to not related information provided by the authors or obviously unreasonable. Our evaluation result is 815 ± 13mb and is represented grapghically in Fig.2.

In the twenty–four references, only four consentaneous multiple–value sets(H.O.Menlove, Lu Hanlin, L.R.Greenwood and Y.Ikeda) can be used to produce a shape of $^{55}Mn(n,2n)$ ^{54}Mn reaction in the neutron energy range from 12.17MeV to 19.39MeV. The multiple–value sets were normalized by the evaluated cross section at the 14.7MeV. The shape of theoretical calculation was used as supplement in the blank energy range from threshold to 12 MeV and for extrapolation near 19 MeV, which was used to fit the measured data. The evaluation was made for above mentioned data by used a program of orthogonal polynomial fit. A set of recommended cross section values are given with 0.1MeV or 0.2MeV energy steps from 10.5MeV to 20 MeV energy range in Table 2. The estimated accuracy is about 2%–15%. The correlation matrix for the evaluated cross section is given in Table 3.

Table 1. Compiled data for Mn-55 (n,2n) Mn-54

Ref. No.	Year	Author	Point	Neutron Energy (MeV)	Cross Section (mb)	Adjustment Factor 1	Adjustment Factor 2	Adjusted Cross Section (mb)
1	58	V.J.Ashby	1	14.1	900±70	1.0834		975
2	60	E.Weigold	1	14.5±0.5	825±185	1.0234	0.9178	775
3	61	R.Wenusch	1	14.1	600±120	1.0834	1.0560	686
4	63	B.Granger	1	14.0	1310±327	1.1022		1444
5	65	A.Paulsen	23	14.71±0.27	945±57	0.9990		944
6	67	H.O.Menlove	10	14.96±0.87	854±79	0.9738	*0.9958	828
7	67	J.Csikai	1	14.1	750±112	1.0834	0.9569	778
8	67	H.Vonach	1	14.1±0.1	786±60	1.0834		852
9	69	R.C.Barrall	1	14.8±0.2	750±60	0.9911	1.0000	743
10	69	R.C.Barrall	1	14.6±0.2	785±80	1.0109	0.9536	757
11	69	M.Bormann	8	14.10±0.15	798±78	1.0834	0.9990	837
12	72	G.N.Maslov	1	14.6±0.2	866±65	1.0109	0.9944	875
13	72	D.A.Salnikov	1	14.30±0.14	540±70	1.0419		563
14	73	J.Araminowicz	1	14.6	643±65	1.0109		650
15	75	F.Deak	1	14.7	680±300	1.0000		680
16	76	V.O.Schwerer	1	14.6±0.1	775±80	1.0109	1.0079	790
17	79	Lu Hanlin	15	14.58±0.20	813±29	1.0133	1.0000	821
18	79	K.Kayashima	1	14.6	884±58	1.0109	1.0000	894
19	84	Berrada	1	14.60	730±30	1.0109		738
20	85	B.M.Bahal	1	14.70	741±21	1.0000		741
21	85	L.R.Greenwood	5	14.50	790±16	1.0234	1.0000	816
22	85	J.W.Meadows	1	14.74	756±41	0.9959	1.0000	753
23	88	Y.Ikeda	8	14.67	820±47	1.0044	0.9914	817
24	88	K.Kobayashi	1	14.05	775.4±28.6	1.0232		793

Factor 1: adjustment for neutron energy Factor 2: adjustment for standard cross section, half- life and gamma bronching ratio

Mn-55 (n,2n) Mn-54 Cross Section

Recommended Reference Data

Linear-Linear Interpalation

CROSS SECTION VALUES

Neutron energy (MeV)	Cross section (mb)	Error (mb)	Neutron energy (MeV)	Cross section (mb)	Error (mb)
10.50	8.6	0.86	14.90	833	14
10.60	17.2	1.72	15.00	840	15
10.80	51.6	5.16	15.20	854	17
11.00	97	9.7	15.40	865	22
11.20	148	15	15.60	873	30
11.40	201	20	15.80	879	44
11.60	254	25	16.00	884	57
11.80	305	30	16.20	886	59
12.00	355	20	16.40	888	60
12.20	402	16	16.60	888	62
12.40	447	17	16.80	888	64
12.60	490	19	17.00	887	66
12.80	530	20	17.20	886	67
13.00	568	19	17.40	885	69
13.20	605	18	17.60	883	71
13.40	640	18	17.80	881	72
13.60	673	18	18.00	879	75
13.80	704	18	18.20	874	76
14.00	732	17	18.40	864	77
14.10	746	16	18.60	860	78
14.20	759	15	18.80	850	82
14.30	772	15	19.00	836	84
14.40	783	14	19.20	820	82
14.50	795	14	19.40	804	80
14.60	805	13	19.60	787	79
14.70	815	12	19.80	776	78
14.80	824	13	20.00	773	77

Table 3. Corrleation Matrix for the Evaluated Cross-section of $^{55}Mn(n,2n)^{54}Mn$ Reaction

1 = 10.5–11.0MeV
2 = 11.0–12.0MeV
3 = 12.0–12.5MeV
4 = 12.5–13.0MeV
5 = 13.0–13.5MeV
6 = 13.5–14.0MeV
7 = 14.0–14.5MeV
8 = 14.5–15.0MeV
9 = 15.0–16.0MeV
10 = 16.0–17.0MeV
11 = 17.0–18.0MeV
12 = 18.0–19.0MeV
13 = 19.0–20.0MeV

	1	2	3	4	5	6	7	8	9	10	11	12	13
1	100												
2	35	100											
3	9	11	100										
4	1	2	14	100									
5	2	3	24	33	100								
6	6	8	71	31	75	100							
7	4	5	43	24	53	73	100						
8	1	2	10	25	35	89	86	100					
9	5	6	48	31	49	42	35	54	100				
10	5	6	48	25	47	41	35	54	62	100			
11	5	7	56	27	51	32	38	54	68	67	100		
12	5	7	56	27	51	32	38	54	68	67	61	100	
13	4	6	27	15	26	21	21	30	39	39	39	39	100

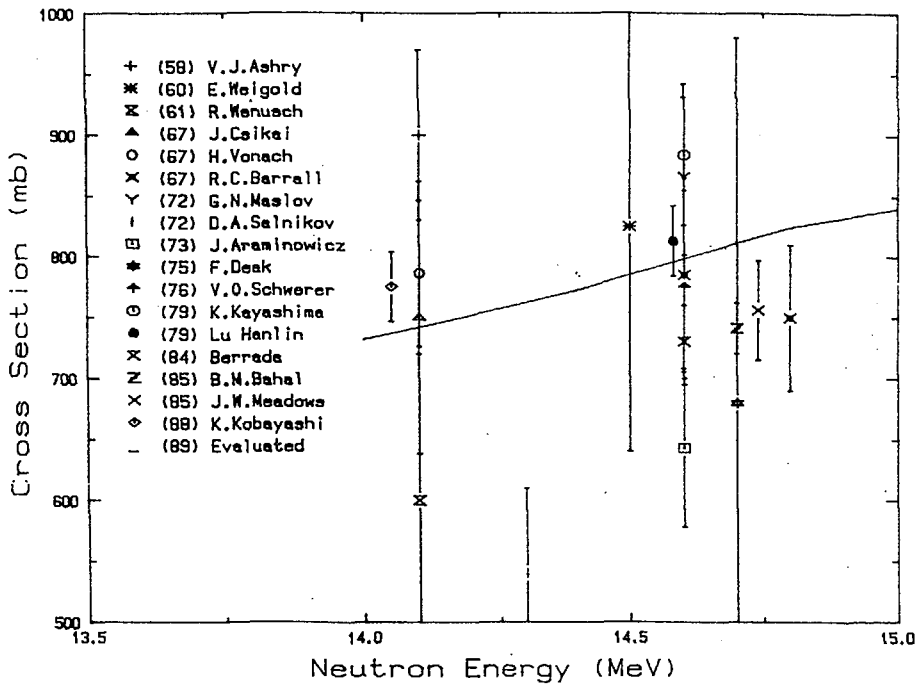

Fig.1(a) Mn-55(n,2n)Mn-54

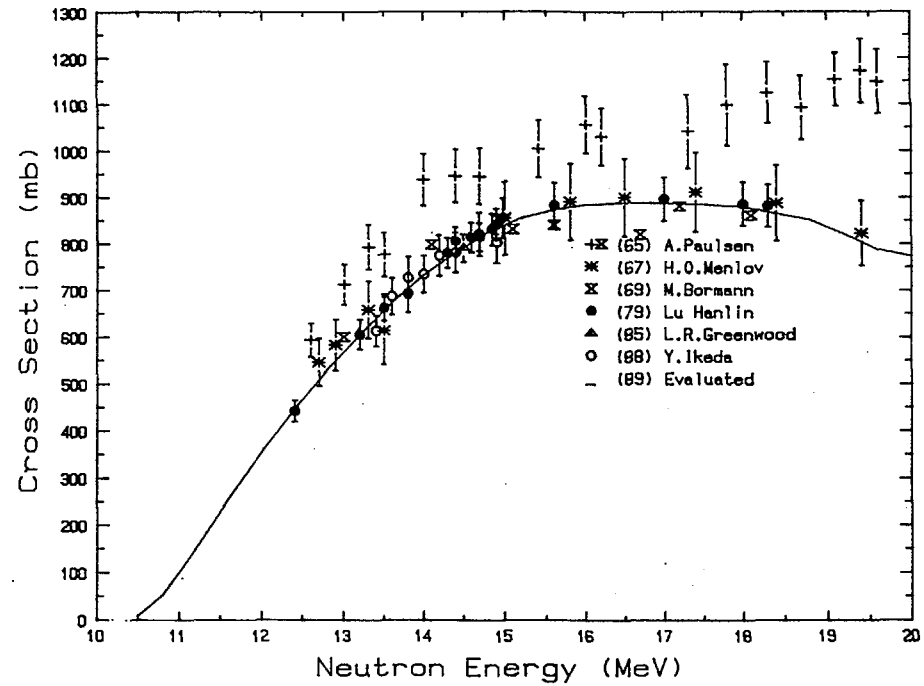

Fig.1(b) Mn-55(n,2n)Mn-54

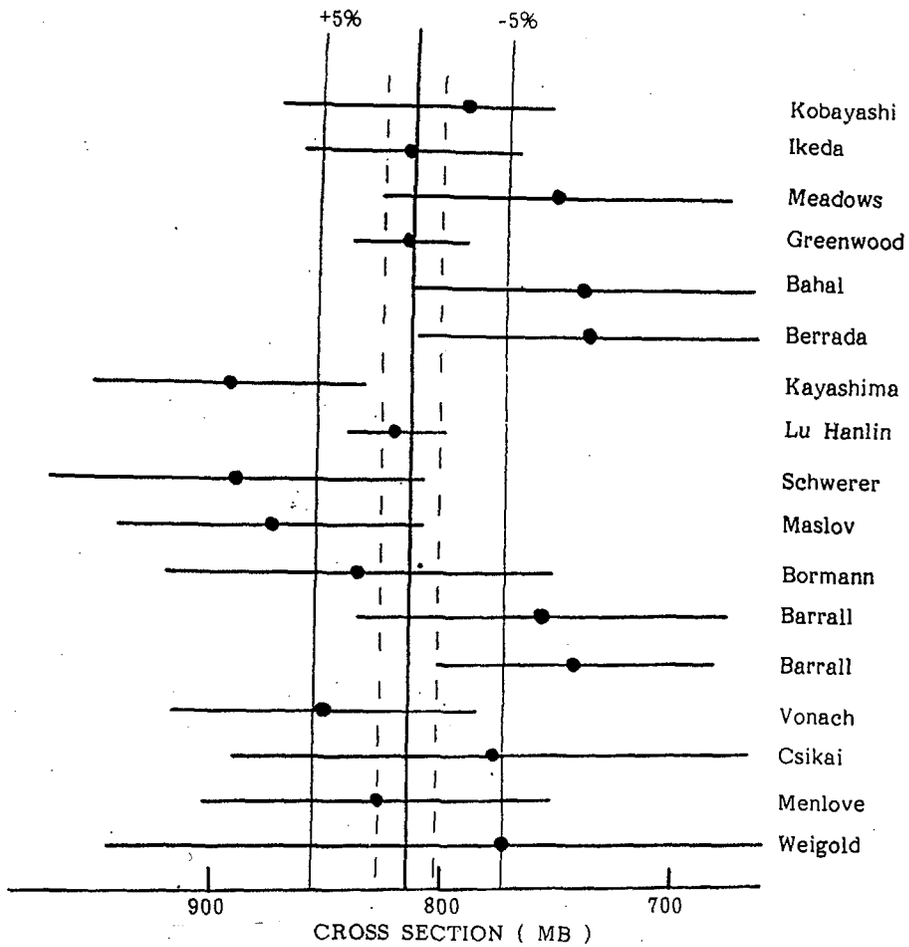

Fig. 2 Evaluation of 14.7 MeV cross section for $^{55}Mn(n,2n)^{54}Mn$

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Evaluation of the $^{54}Fe(n,\alpha)$ ^{51}Cr reaction cross section

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The excitation curve of the $^{54}Fe(n,\alpha)$ ^{51}Cr reaction has been evaluated based on seventeen experimental data sets. These results are listed in Table 1 and Fig.1 [1-16]. A number of experiments provided the data only around 14MeV. Four authors, S.R.Salisbury[4], A.Paulsen[10], Lu Hanlin[12]and Y.Ikeda[19]given multiple-value set in wide neutron energy range.

Some of the evaluations have been made for this reaction. For example,the data of ENDF/B-V are higher than the trend of measured data in the 9-13MeV and 16-20MeV neutron energy range. The latest data(13,14,15) showed that the evaluated data of B.P.Evain[17]must be renewed.

The abundance of isotope ^{54}Fe in natural iron is the value of 5.8%. The half-life of ^{51}Cr is very well known with the value 27.07d. According to recent information, the characteristic gamma ray of 320 keV of the product has a branching ratio of 9.85%.

The present evaluation was made at 14.7MeV using the references from (1) to (16). The collected values were adjusted for some mutual values, including nuclear decay schemes, half-life and standard cross sections, which were taken from Refs. [17,18]if enough information can be obtained in the literature. All of the data were adjusted for energy to equivalent 14.7MeV cross section, which depends on the shape of cross section. We found some information on the shape of cross section from refs.(3,10,12,14,16). But only the data of Lu Hanlin et.al.,[12]span the entire energy range and their results appear to be reasonably consistent. The least-square fit was used to obtain the factors of energy adjustment values.

Some results should be rejected due to the larger discrepancies with others such as D.M.Chittenden's[2]and exceeding by three-standard-deviation for the results of V.H.Pollehn[1], S.M.Qaim[6]and J.J.Singh[8].

The evaluation was made by the averagement of the remained data, after adjustments mentioned above. The weighted factors were used in the evaluation, which were based on the given errors by authors and quoted errors by us. Our final evaluated datum is 91.4 ± 2.3 mb and represented graphically in Fig.2.

Among the references, only the data of S.R.Salisbury, A.Paulsen, Y.Ikeda and Lu Hanlin can be used for the evaluation of excitation function of $^{54}Fe(n,\alpha)^{51}Cr$ reaction. But part data of S.R.Salisbury and A.Paulsen were rejected due to bad shapes for the evaluations in neutron energy ranges of 2.23-4.26MeV and 13.2-16.4MeV respectively. So the total excitation function evaluation was based on the measured data of S.R.Salisbury, A.Paulsen and Fan Peiguo[16]in the neutron energy range 4.5-10.0MeV, the evaluated datum at 14.7MeV and the shape of Lu Hanlin and Y.Ikeda in the neutron energy range 13.0-18.3MeV. The gaps of excitation function in the 10-13MeV and 18-20MeV energy ranges were interpolated and extrapolated respectively. The shape of the theoretical calculation was used to fit the adjacent measured data. The evaluation was performed for above mentioned data by using a program of orthogonal polynomial fit.

The recommended cross sections are given in the Table 2 with 0.1, 0.2 or 0.5MeV energy steps from 4.0 to 20.0MeV energy. The estimated accuracy is 30% near threshold, 4-6% below 14MeV, 2.6% between 14-15MeV, 3-4% above 15MeV and 10% up to 20MeV. The correlation matrix for the evaluated cross section is given in the Table 3.

Table 1. Compiled data for Fe-54 (n,alpha) Cr-51

Ref. No.	Year	Author	Point	Neutron Energy(MeV)	Cross Section (mb)	Adjustment Factor 1	Adjustment Factor 2	Adjusted Cross Section (mb)
1	61	V.H.Pollehn	1	14.1	131±24	1.0252		
2	61	D.M.Chittenden	1	14.8±0.9	270±135	1.0022		
3	63	W.G.Cross	4	14.5	94±10	1.0061	1.0139	95.9
4	65	S.R.Salisbury	9	14.05±0.105	91.6±37.1	1.0272	0.9964	93.5
5	67	R.V.Rao	1	14.4±0.3	90±10	1.0137	0.9451	86.2
6	71	S.M.Qaim	1	14.7±0.3	134±14	1.0000		
7	72	J.J.Singh	1	14.5	139.5±7.5	1.0061		
8	72	G.N.Maslov	1	14.6±0.2	106±7	0.9989	0.9944	105.3
9	78	K.Fukuda	1	14.6	84±7.5	0.9989		83.9
10	79	A.Paulsen	17	14.0±0.3	79.9±5	1.0292	1.1136	88.5*
11	80	O.I.Artem-ev	1	14.8	90±15	1.0022		90.2
12	82	Lu Hanlin	21	14.58±0.20	90:2±4.5	1.0003	1.0152	90.6
13	85	B.M.Bahal	1	14.70	88±6	1.0000		88
14	85	L.R.Greenwood	5	14.5	91.5±	1.0061	1.0000	92.6
15	85	J.W.Meadows	1	14.74±0.02	92.44±6.59	1.0009	1.0000	92.5
16	88	Y.Ikeda	8	14.43	85.3=6.3	1.0130	1.0000	87.9
17	85	Fan Peiguo	1	8.52±0.27	43.2±3.7			

Factor 1: Adjustment for neutron energy

Factor 2: Adjustment for standard cross section, half-life and gamma bronching ratio

* Average value (14.0 MeV and 14.8 Mev)

Table 2

Fe-54 (n,alpha) Cr-51 Cross Section

Recommended Reference Data

Linear-Linear Interpolation

CROSS SECTION VALUES

Neutron energy (MeV)	Cross section (mb)	Error (mb)	Neutron energy (MeV)	Cross section (mb)	Error (mb)
4.0	0.0	0.0	11.2	66.9	4.0
4.2	0.51	0.10	11.4	68.8	4.1
4.4	0.92	0.18	11.6	70.6	4.2
4.6	1.4	0.3	11.8	72.5	4.3
4.8	2.0	0.4	12.0	74.3	4.4
5.0	2.9	0.6	12.2	76.1	4.6
5.2	3.8	0.8	12.4	77.9	4.7
5.4	5.1	1.6	12.6	79.6	4.0
5.6	6.6	1.6	12.8	81.2	4.2
5.8	8.2	1.5	13.0	82.8	3.3
6.0	10.1	1.5	13.2	84.2	3.3
6.2	12.1	1.0	13.4	85.6	2.6
6.4	14.3	0.9	13.6	86.9	2.4
6.6	16.5	0.9	13.8	88.1	2.0
6.8	18.9	0.9	14.0	89.1	2.0
7.0	21.3	1.1	14.1	89.6	2.3
7.2	23.8	1.2	14.2	90.0	2.7
7.4	26.3	1.3	14.3	90.4	2.7
7.6	28.8	1.4	14.4	90.7	2.7
7.8	31.2	1.5	14.5	91.0	2.5
8.0	33.7	1.4	14.6	91.3	2.3
8.2	36.1	1.4	14.7	91.4	2.3
8.4	38.4	1.6	14.8	91.5	2.3
8.6	40.8	1.7	14.9	91.6	2.3
8.8	43.0	1.8	15.0	91.6	2.4
9.0	45.2	1.9	15.5	91.1	2.4
9.2	47.4	1.9	16.0	89.2	2.4
9.4	49.6	2.0	16.5	86.0	2.4
9.6	51.5	2.1	17.0	82.0	2.4
9.8	53.5	2.6	17.5	77.1	2.4
10.0	55.5	2.8	18.0	71.8	2.5
10.2	57.5	2.9	18.5	66.4	3.0
10.4	59.4	3.4	19.0	60.5	4.2
10.6	61.3	3.7	19.5	53.4	5.4
10.8	63.1	3.8	20.0	43.7	4.4
11.0	65.0	3.9			

Table 3. Correlation Matrix for the Evaluated Cross-section of $^{54}Fe(n,\alpha)^{51}Cr$ Reaction

1=4.0–4.5MeV	2=4.5–5.0MeV	3=5.0–5.5MeV
4=5.5–6.0MeV	5=6.0–6.5MeV	6=6.5–7.0MeV
7=7.0–7.5MeV	8=7.5–8.0MeV	9=8.0–8.5MeV
10=8.5–9.0MeV	11=9.0–9.5MeV	12=9.5–10.0MeV
13=10.0–11.0MeV	14=11.0–11.5MeV	15=11.5–12.0Me
16=12.0–12.5MeV	17=12.5–13.0MeV	18=13.0–13.5MeV
19=13.5–14.0MeV	20=14.0–14.5MeV	21=14.5–15.0MeV
22=15.0–16.5MeV	23=16.5–17.0MeV	24=17.0–18.0MeV
25=18.0–19.0MeV	26=19.0–20.0MeV	

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26
1	100																									
2	18	100																								
3	5	29	100																							
4	3	19	22	100																						
5	1	7	8	36	100																					
6	1	7	8	35	37	100																				
7	1	7	8	35	37	36	100																			
8	1	5	5	24	38	36	36	100																		
9	1	5	5	23	24	33	33	33	100																	
10	1	5	5	24	26	24	36	37	33	100																
11	1	5	5	24	26	24	24	37	33	37	100															
12	14	10	3	2	1	1	1	1	2	1	1	100														
13	9	7	2	5	4	5	5	5	10	5	5	16	100													
14	2	3	2	11	12	13	13	13	24	13	13	2	16	100												
15	1	2	2	12	12	13	13	13	19	13	13	1	7	20	100											
16	1	4	4	17	17	19	19	19	31	19	19	2	12	31	45	100										
17	1	3	3	16	16	17	17	18	30	18	18	2	11	29	42	79	100									
18	1	3	3	15	16	17	17	17	30	17	17	2	11	29	43	82	77	100								
19	1	3	3	16	16	17	17	18	29	18	18	2	11	28	37	72	71	65	100							
20	1	3	3	16	17	17	17	18	27	18	18	1	10	26	33	63	62	58	67	100						
21	1	4	4	18	18	19	19	20	29	20	20	2	10	27	35	66	63	60	68	63	100					
22	1	3	3	14	14	15	15	16	26	16	16	1	10	25	32	58	64	61	68	58	60	100				
23	1	3	3	13	14	15	15	15	26	15	15	1	10	24	26	43	42	58	70	60	61	61	100			
24	1	3	3	15	15	17	17	17	29	17	17	2	11	27	27	48	47	43	65	66	67	69	68	100		
25	1	3	3	13	14	15	15	15	27	15	15	2	11	26	26	50	49	45	54	48	58	70	70	78	100	
26	19	14	4	3	1	1	1	1	1	1	1	18	12	2	1	1	1	1	1	1	1	1	1	1	1	100

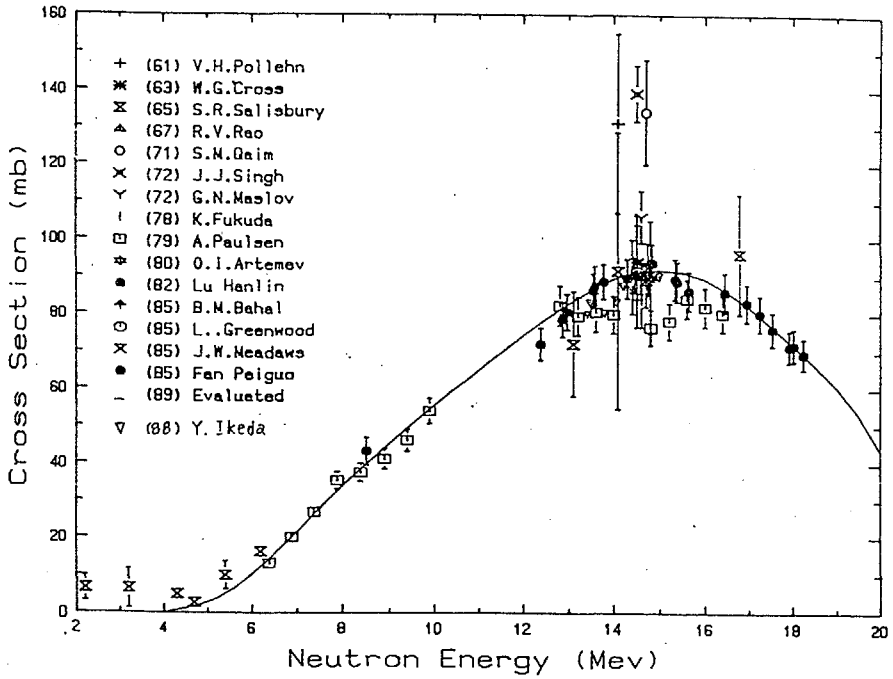

Fig.1 Fe-54(n,a)Cr-51

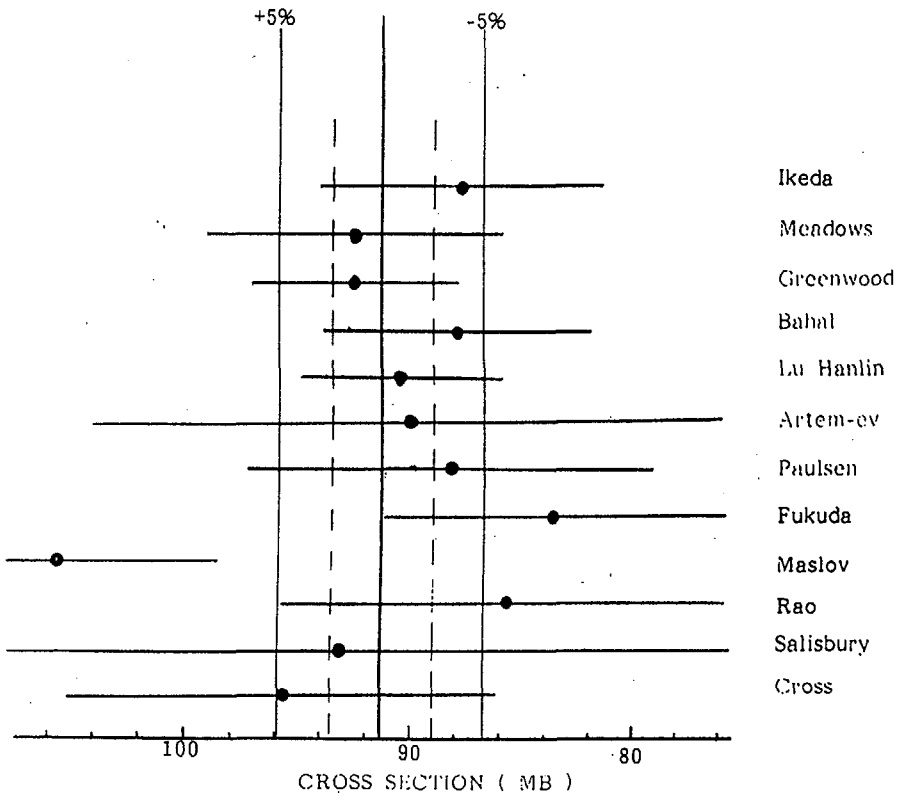

Fig. 2 Evaluation of 14.7 MeV cross section for $^{54}Fe(n,a)^{51}Cr$

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EVALUATION OF $^{58}Fe(n,\gamma)$ ^{59}Fe REACTION CROSS SECTIONS AND ITS COVARIANCE

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ABSTRACT

The evaluations for $^{58}Fe(n,\gamma)$ ^{59}Fe reaction cross sections and its covariance data are completed and compared with ENDF/B-V.

INTRODUCTION

The evaluation of $^{58}Fe(n,\gamma)$ ^{59}Fe reaction cross sections and its covariance for ENDF/B Dosimetry Library was finished by R. Schenter et al. [1] in 1979. No any measured data above 200 keV were available in Schenter's evaluation. Since then, several measurements have been published[2~4]. These new measurements are valuable for the present evaluation.

1 EVALUATION OF CROSS SECTIONS (MF=3)

1.1 Thermal Region (10^{-5}~3eV)

The measured cross sections[5~12] at $E_n = 0.0253$ eV after 1960 are given in Tab. 1.

The newest evaluated value[13] for isotopic abundance f of ^{58}Fe is

$$f = 0.28 \pm 0.01 \qquad (1)$$

which is consistent with both of Schmidt [14] and James [15] new measured data. The data in Tab. 1 have been renormalized to $f = 0.28$. The measured values are also given in Tab. 1. Using these data, the average of latest five measured data is

$$\sigma_{n\gamma} = 1.30 \pm 0.06 \quad b \qquad (2)$$

The error of $\sigma_{n\gamma}$ consists of standard deviation of the average (3.2%) and the error of f (3.5%). The comparison between this evaluation and others[1,16-17] are given in Tab. 2.

According to $1/v$ law, the cross sections in $E_n = 10^{-5}$~3eV can be calculated from the thermal cross section.

Tab. 1 A survey of activation cross sections measurements for $^{58}Fe(n,\gamma)$ ^{59}Fe reaction at $E_n = 0.0253$eV

Author(Year)[Ref.]	$\sigma_{n\gamma}$, b	f, %	$\sigma_{n\gamma}$, b (renorm. to $f = 0.28$)
Cirardi(1963)[5]	1.31	0.31	1.45
Fabry(1965)[6]	1.23±0.035	0.33	1.45±0.04
Carter(1966)[7]	1.09±0.03	?	
Fabry(1967)[8]	1.18±0.03	0.33	1.39±0.04
Ryves(1970)[9]	1.14±0.02	0.33	1.34±0.02
Takine(1978)[10]	1.03±0.10	0.31	1.14±0.11
Nikolow(1980)[11]	1.20±0.05	0.31	1.34±0.06
Simonits(1984)[12]	1.31±0.03	0.28	1.31±0.03

1.2 Resoance Region (3eV~350keV)

The point wise cross sections in this region are calculated from a set of resonance parameters re-evaluated based on the measured data of Ref. [2~3] and Ref. [18~20] by using the program MSBW2[21]. The earlier capture cross section measurement[22] is not adopted by this work because its poor signal-to-background ratio caused a big systematic uncertainty which was obviously underestimated. Moreover, the capture cross sections and capture areas of Ref. [2] have to be multiplied a correction factor 0.9655 because of a mistake in the program used[23].

On the basis of the measurements, neutron resonance parameters for ^{58}Fe are re-evaluated as follows:

The resonance energies and the neutron widths for individual resonances are mainly taken from the high resolution transmission measurement of Garg[20]. The capture widths are given on the basis of Allen [2] and Kappeler [3] measurements. Between these two measurements no systematic deviation is found. From capture areas

$$A_\gamma = g\Gamma_n\Gamma_\gamma/\Gamma \qquad (3)$$

given by Ref. [2] and [3] and supposing $\Gamma_n \gg \Gamma_\gamma$, we can get

$$g\Gamma_\gamma = A_\gamma \qquad (4)$$

the Γ_γ can be calculated in case where J is assigned. All of the resonances with $L > 1$ are supposed to $L = 1$. The J values of p-wave resonances are randomly assigned according to the values of $g\Gamma_\gamma$. Generally, the resonance with large $g\Gamma_\gamma$ is assigned as $J = 2$ and other-

wise $J=1$. For several s-wave resonances, Both Γ_n and Γ_γ given by Allen or by Kappeler through shape fitting are adopted. For two resonances of $E_R=230$ and 359 eV with small neutron widths, all parameters of Schenter evaluation are adopted because Γ_γ can not be obtained from measured capture areas. In the region of $E_n=200\sim350$ keV, no any measured data for capture width are available. The average s-wave capture width which is estimated from re-evaluated resolved resonance parameters

Tab. 2 Evaluated thermal cross section of $^{58}Fe(n,\gamma)$ ^{59}Fe reaction

	ENDF/B-5 1979[1]	JENDL-2 1984[17]	BNL-325 1981[16]	this work
$\sigma_{n\gamma}$, b	1.15	1.28	1.28±0.05	1.30±0.06

$$\bar{\Gamma}_\gamma(L=0)=0.96\pm0.13\ \text{eV} \tag{5}$$

is used as capture width for individual s-wave resonance. For individual p-wave resonance, the $\bar{\Gamma}_\gamma(L=1)$ is adjusted to

$$\bar{\Gamma}_\gamma(L=1)=0.80\ \text{eV} \tag{6}$$

so that average cross sections in resonance region can be linked up with one in smooth region. Adjusted average p-wave capture width is larger than one estimated from resolved resonance parameters and it can be considered as the contribution from d-wave which is important in this energy region. To agree with the measured average cross sections of Allen, parameters of several resonances have been adjusted and about 1mb background added to the calculated cross sections. The comparison of average cross sections from the evaluated point wise cross sections with measured data of Allen are given in Tab. 3.

Tab. 3 Average capture cross sections

Energy range keV	Average caputre cross sections, mb			
	Measured Values[2]	ENDF/B-5[1]	JENDL-2[17]	This work
4.0~5.0	1.8±0.18	9.7	4.8	1.8
6.0~8.0	46.8±2.4	8.1	776.	46.8
10.0~15.0	82.2±8.5	3.3	108.	82.2
15.0~20.0	12.3±0.6	2.7	15.6	12.3
20.0~30.0	3.86±0.29	2.4	40.7	3.86
30.0~40.0	10.6±0.57	5.0	42.1	10.6
40.0~50.0	24.7±0.6	3.7	14.5	24.7
50.0~60.0	4.15±0.29	3.7	22.0	4.15
60.0~80.0	7.72±0.57	2.34	30.3	7.72
80.0~100.0	10.6±2.3	4.0	21.5	10.6
100.0~150.0	6.66±0.38	2.25	2.94	6.66
150.0~200.0	4.73±0.29	7.8	2.48	4.73

1.3 Smooth Region (350keV~20MeV)

No any measured data except Trofimov[4] are available in this region. The systematics formula of Ref. [24] is used to calculate the excitation function of $^{58}Fe(n,\gamma)$ ^{59}Fe reaction. The parameters in the systematics formula are adjusted so that calculated cross sections can pass through Trifumov's data in energy region of $E_n=0.5\sim2$ MeV and $\sigma_{n\gamma}$ (14MeV) = 1 mb given by the systematics study in this target mass region. Moreover, the following parameters of giant dipole resonance are used in the calculation:

$$E_{m1}=16.62\ \text{MeV}\qquad \Gamma_{m1}=4.24\ \text{MeV}$$
$$E_{m2}=19.91\ \text{MeV}\qquad \Gamma_{m2}=4.16\ \text{MeV} \tag{7}$$
$$\sigma_{m1}/\sigma_{m2}=1$$

The comparison with Trofimov's data is shown in Fig. 1.

2 EVALUATION OF COVARIANCE DATA (MF=33)

The covariance should be re-evaluated because of that the new measured data after Schenter's evaluation have been adopted in this evaluation.

1.1 Long Range Component

The medium and long range components of the covariance for the cross sections are evaluated as follows:

1.0E-5~3eV-the uncertainty of the thermal cross section (4.7%);

3eV~100keV-the systematic uncertainties of Allen and Kappeler measurements (2.3%);

100keV~200keV-the systematic uncertainty of Allen measurement (5.3%);

200keV~350keV-the standard deviation of s-wave average capture width;

350keV~7MeV-the systematic uncertainty of Trofimov's measurement (4%);

7MeV~20MeV-the uncertainty of the systematics of $\sigma_{n\gamma}$(14MeV) (30%).

1.2 Short Range Component

The short range components of the covariances are evaluated as follows:

3eV～200keV-the uncertainty of individual s-wave resonance parameters;

200keV～350keV-the standard deviation of capture widths of individual s-wave resonances;

350keV～2MeV-the statistical uncertainties of Trofimov measurement;

2Mev～20MeV-the shape error in the excitation funtion calculated by the systematics.

3 COMPARISON

The comparisons of this evaluation with ENDF/B-5[1] and JENDL-2[17] are shown in Fig. 1.

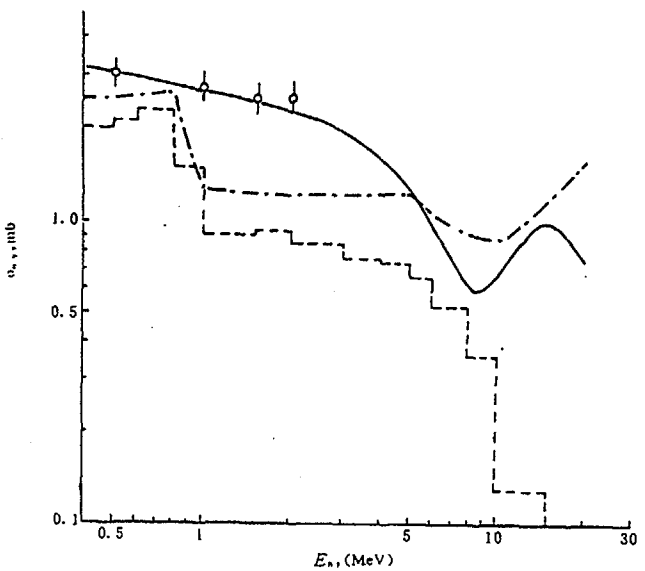

Fig. 1 Cross Sections in Smooth Region

○ —Trofimov;

— · — —ENDF/B—5;

— — —JENDL—2;

—— —this work.

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THE EVALUATION OF $^{59}Co(n,\gamma)^{60}Co$ REACTION CROSS SECTION AND ITS COVARIANCE DATA

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INTRODUCTION

$^{59}Co(n,\gamma)^{60}Co$ reaction cross section is of importance for dosimetry application. The evaluation of cross sections and their covariance for ENDF / B-V were performed by S. Mughabghab in 1977. Since then, there are no any newly measured data. So the present evaluation including the covariance is also carried out based on the same data body.

1 EVALUATION OF CROSS SECTION (MF = 3)

1.1 Cross section in thermal energy region ($1.0\times10^{-5}\sim 1$ eV)

No large differences exist among measured thermal cross sections. See Tab. 1. In order to agree with the resonance parameters, the value of 37.18 ± 0.6 b is adopted in this evaluation.

1.2 Cross section in resonance region

There are only few of measured point-wise cross sections in resonance region. For obtaining the evaluation the cross section must be calculated from the evaluated resonance parameters. In the present case, there are two sets of measured resonance parameters carried out by Spencer et al., one with Electron Linear Accelerator in Oak Ridge National Laboratory[1], and the other with the Van de Graaff Accelerator in Kernforschungszentrum[2]. These two measurements should be considered as independent measurements. It is noted that:

(1) The 30 keV Maxwellian-average capture cross sections calculated with these two sets of resonance parameters are in agreement with each other.

(2) The resonance parameters of these two measurements are consistent essentially.

So these two measurements are combined simply after that the resonance parameter sets are modified according to the following principles:

(1) The resonances, for which the $2g\Gamma_n^o$ or $2g\Gamma_n$ are not given, are not adopted.

(2) The J values are assigned randomly in proportion to the level density factor $2J+1$ to the resonances for which the J values are not given.

(3) $L=1$ are assigned to the resonances for which L values are not given.

(4) The averaged $\Gamma\gamma$ are given for the resonances for which the $\Gamma\gamma$ are not given, i. e.

$$<\Gamma\gamma> = 0.564 \text{ eV} \quad \text{for } J=3$$
$$<\Gamma\gamma> = 0.486 \text{ eV} \quad \text{for } J=4$$

The final resonance parameter set is used to calculate the point-wise cross sections for present evaluation by using the code MLBW [3].

1.3 Cross section in unresolved resonance region (85 keV ~ 1 MeV)

The measured results of Spencer and Macklin[1] are adopted for this evaluation.

1.4 Cross section in the smooth region

The excitation function of systematics predications[4] in the smooth region is adopted and normalized to 1.01 ± 0.13 mb, the average value of two measurments of M. Budnar and F. Rigaud at 14.1 MeV, see Tab. 2.

The results are shown in Fig. 1.

2 COVARIANCE (MF = 33)

1. An uncertainty of 2 % is given to the cross section in the thermal energy region.

2. An uncertainty of 10 % is given for the resonance region based on the following reason:

(1) A systematic uncertainty less than 5 % is given in the Spencer and Macklin's measurement.

(2) The 30 keV Maxwellian-average capture cross section calculated from the measured data of Spencer and Beer is in agreement with that of Spencer and Macklin.

(3) The systematics predication is in agreement with the measured cross section.

3. An uncertainty of 30 % is given for the systematics predications for the smooth energy region of < 4 MeV.

4. An uncertainty of 20 % is given for the energy region of > 4 MeV con-

sidering the uncertainty of the shape of systematics prediction and the uncertainty of the measured value at 14 MeV for normalization.

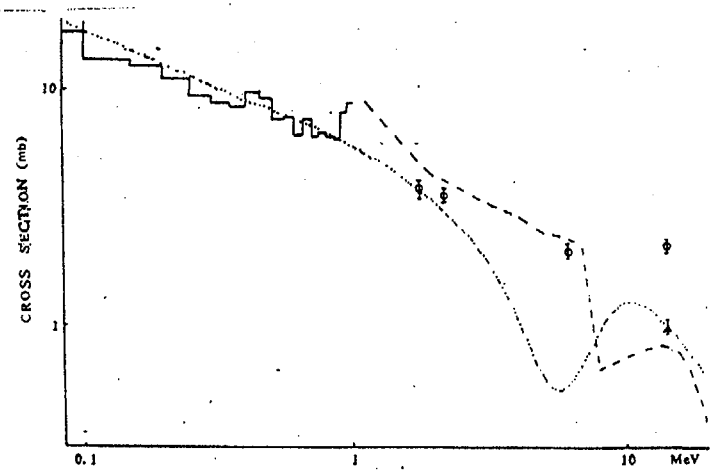

Fig. 1 Comparison of systematics prediction and ENDF / B-V

— P. R. Spencer --- ENDF / B-V — · — · — Systematics

ϕ A. Paulsen Δ M. Budnar et al.

Tab.1 Measured and evaluated thermal cross sections(in barn)

72	MOL	Deworm	DEWORM	37.70 ± 0.4
69	HAR	Silk+	AERE-R-6059	37.31
66	FAR	Carre+	66PARIS 1,479	38.00
61	ANL	Meadows+	NSE 9,132	36.30 ± 0.6
61	BUC	Stefanescu+	61BUCHAR,553	38.40 ± 0.99
60	HAR	Tattersall+	JNE / A 12,32	38.20 ± 0.7
52	FAR	Grimeland+	CR 232,2089	33.90
		ENDF / B-V		37.233
		BNL-325		37.18 ± 0.06
		present work		37.18 ± 0.6

Tab. 2 A survey of capture cross sections at the energy about 14 MeV

79	M. Budnar	1.005 ± 0.155	mb	[6]
71	F. Rigaud	1.02 ± 0.26	mb	[5]

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Evaluation of the $Co^{59}(n,2n)Co^{58,58m}$ reaction cross section

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An accurate knowledge of the $Co^{59}(n,2n)Co^{58,58m}$ reaction cross section is of importance due to the wide use of cobalt as a member of structure materials in fission and fusion reactors and its applicability in neutron dosimetry.Therefore many laboratories were and are interested in measuring the cross section of the reaction,and about forty data sets[1–39] are obtained.

Co^{59} is the sole isotope of the element,so no other reactions lead to Co^{58} production.There is a metastable state,Co^{58m},at an excitation of ~ 25kev in Co^{58}.It has a half life of 9.15 ± 0.10h,and ultimately decays to the ground state of 70.82 ± 0.3d half life via an M3 electromagnetic transition. This total yield can be measured,provided that sufficient time is allowed for the Co^{58m} activity to die away to pure Co^{58g} activity (for example,activity measurements are made several days after the end of the irradiations).Co^{58} decays by β^+ emission and electron capture (EC) to levels in Fe^{58}. The method of choice for the measurements is detection of the characteristic 0.810Mev γ–ray with intensity of 99.44 ± 0.02% .An alternative method is measurement of 0.511Mev γ–ray which follow annihilation of the positrons. Also,there are some works measured total neutron yields by a large liquid scintillator(STANK)[15,17,35].

The 35 sets of measured data available[1–35] are collected and shown in Table 1 and Fig.1.There are considerable spread among these cross section values.But,the inconsistency of measured values has been improved in the 1980s (eighties).

For analysis and comparison,all the measured results are adjusted by the reference cross section(Ref[1] and ENDF / B6),the dependence of cross section on energy and unified nuclear parameters(including intensity of radioactivity,half–life,branching ratio)[Ref(1)].

Lu Hanlin et al.[1–3],Ryves et al.[4],Ikeda et al.[5,6],Meadows et al.[8],Hasan et al.[9],Greenwood et al.[10],Frehaut et al.[15] and Veeser et al.[17] have made efforts to study the reaction for many years and got improved results.And the results of the two latter using the independent method which ,detected the emitted neutrons directly,may be used as check.

According to this situation,the experimental data are treated as the follow ways:

At 14.7Mev neutron energy

Based on the measured results of works[1–5,7–10],the recommended cross section at 14.7Mev is 770 ± 10 mb

For the neutron energy range from threshold to 25Mev

The excitation functions of Lu et al.[1–3],Ikeda[5],Hasan[9],Greenwood[10], Berrada[12],Ghorai[14],Frehaut[15] and Veeser[17] are normalized by the recommended datum at 14.7Mev and shown in Fig. 2.It can be seen that the coincidence among them is better.

By the way,the excitation function of several better measured data have been treated directly by least–square method with the weighted factors in the neutron energy range of 10.45–12Mev and 12–25Mev respectively,the result is combined with the systematic one[40] and shown in Fig. 2. It should be decreased the uncertainty induced by adjusting process because of the energy point of 14.7Mev situating at the steep part of the excitation function.

Covariance matrix is given for the evaluated data considering all altered factors and shown in Table 4.[41].The whole energy region is divided into the ten small energy intervals,they are 10.45–12Mev,12–12.5Mev,12.5–13Mev,13–13.5Mev,13.5–14Mev,14–14.5Mev,14.5–15Mev,15–16Mev,16–21Mev,and 21–25Mev respectively.

For the excitation function of $Co^{59}(n,2n)Co^{58m}$ reaction,we gain it through combining theoretic and systematic calculation with the experimental data because of there only being the ones in range 12.55–19.20 Mev.the last result is shown in Fig. 3.and Table 5 as well as satisfied equation:

$$\sigma_t = \sigma_m + \sigma_g$$

The present evaluation should be better than past ones,because many new improved measured data have been adopted and processed in detail.

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Table 1 Survey of measured cross section for $Co^{59}(n,2n)Co^{58}$ reaction

Year	Lab.	Author	En Mev	Cross Section mb		Adjucted Factor		Method	
				Published	Adjusted	F1	F2	Fluence	Activity
1989	CIAE	Zhao Wenrong+	(12.49–18.26)	✓	✓			AP $.^{27}Al(n,\alpha)$ $.^{56}Fe(n,p)$	Ge(Li) 136 cm^3 NaI(TI) 80 × φ80 mm $4\pi\beta-\gamma$
1990		Li Tingyan+	14.72 (12.84–17.77)	764 ± 43	762.4	0.9979	1.0		
1987		Lu Hanlin+	14.72 (12.84–17.77)	764 ± 43					
1981		Huang Jianzhou+	14.77 (12.49–18.26)	789 ± 29	783.3	0.9928	1.0		
1988	UKNPL	T.B.Ryves+	14.3	753 ± 13	794.1	1.0546	1.0	$.^{56}Fe(n,p)$	Ge(Li) NaI(TI) $4\pi\beta$ pc
1988	JAERI	Y.Ikeda+	14.67 (13.3–14.94)	786 ± 41 752 ± 35	788.8 754.6	1.0035	1.0	Al(n,α) $.^{93}Nb(n,2n)$	Ge(Li) Ge–IN
1985			14.66 (13.5–14.89)	697 ± 37	687.73	1.0052	1.0		
1988	JPNKTO	K.Kobayashi+	14.05	729 ± 29	790.6	1.0954	0.99	Al(n,α)	Ge(Li)
1987	ANL	J.W.Meadows+	14.74	754.3 ± 35	745.1	0.9959	0.9924	$.^{235}U(n,f)$	Ge(Li)
1986	AUSIRK	S.J.Hasan+	14.533 14.822 (13.45–14.822)	766.1 ± 7.27 787.58 ± 8.1	782.3 779.5	1.0212 0.9897	1.00	Al(n,α)	Ge(Li) NaI(TI)
1985	ANL	L.R. Greenwood	14.65 (14.5–14.9)	789 ± 12	787.1	1.0063	0.991	$.^{93}Nb(n,2n)$	Ge(Li)
1985	RUMCIP	I.Garlea+	14.8	828 ± 60	809.8	0.9897	0.9882	$.^{235}U(n,f)$	Ge(Li)
1985			14.8	748 ± 52	710.8	0.9897	0.9602	$.^{238}U(n,f)$	100 cm^3
1984	MORMOH	M.Berrada	14.7 (13.3–14.8)	793	793	1.00	1.00	Al(n,α)	Ge(Li) γ+x spec.
1982	MORMOH	A.Reggoug+	14.7	720 ± 50	720	1.00			Ge–In γ+X Spec.
1980	USAAUB	S.K.Ghorai+	15.3 (13.4–18.1)	768 ± 51	722.3	0.9465	0.9937	Al(n,α)	Ge(Li) 20 cm^3
1980	FRBRC	J.Frehaut+	14.76 (10.9–14.8)	734 ± 50	705.5	0.9938	0.9671	$.^{238}U(n,f)$ STANK(Gd)	
1978	JPNKYU	K.Fukuda+	14.6	752 ± 60	760.9	1.0118			Ge(Li)
1977	LAS	L.R.Veeser+	14.7 16.0 (14.7–24.0)	820 ± 85 838 ± 55		1.00	1.0	H(n,n) STANK(Gd)	
1977	CCPKGU	Ju.E.Kozyr+	14.6	670 ± 90	677.9	1.0118	1.0	—TGF— H(n,n)	NaI(TI) Si
1973	POLLOU	J. Anaminowicz+	14.6	663 ± 67	660.9	1.0118	0.9851	$.^{63}Cu(n,2n)$	NaI(TI)

Table 1 (continued)

Year	Lab.	Author	En Mev	Cross Section mb		Adjucted Factor		Method	
				Published	Adjusted	F1	F2	Fluence	Activity
1972	CCPFEI	O.A. Salnikov+	14.36	570 ± 75	559.8	1.0452		TOF STANK	
1969	USASTF	R.C.Barrall+	14.6	760 ± 60	769	1.0118		Al(n,α)	NaI(TI) 3 × φ3 in
1968	POLWWA	P.Decowski+	14.50 14.80 (12.57–18.2)	683.3 ± 63.7 669 ± 34	710.3 671.4	1.0252 0.9897	1.014	^{65}Cu(n,2n)	NaI(TI)
1967	JPNKTO	S.Okumura	14.73 (13.4–15.0)	1030 ± 17	1026.8	0.9969	1.0	AP	NaI(TI) 7.62 × φ7.62cm
1966	GERHAM	M.Bormann+	15.2 (12.5–19.2)	695 ± 56	663	0.954	1.0	H(n,n) ^{65}Cu(n,2n)	NaI(TI)
1961			14.1 (13.2–19.6)	640 ± 102	714.9	1.0866	1.028	Al(n,α)	
1965	GEL	A.Paulsen+	14.71 (12.6–19.6)	686 ± 41	685.2	0.9989	1.0	H(n,n) AP	NaI(TI)
1965	ORNL	J.E.Strain+	14.7	1040	1030	1.00	0.99	Al(n,α) ^{65}Cu(n,2n)	NaI(TI)
1963	FRSAC	J.Cabe+	14.1	640 ± 70	695.4	1.0866		Cu^{N}(n,2n)	NaI(TI)
1963	FRBRC	S.Granger+	14.0	587 ± 20	648.3	1.1044			NaI(TI)
1963	FRSAC	J.M.F. Jeronymo+	14.9 (13.5–21.7)	508 ± 70	497.7	0.9797	1.0	H(n,n)	NaI(TI) 5.1 × φ4.4 cm
1962	AUSIRK	R.Wenusch+	14.0	630 ± 120	733.6	1.1044	1.054	Al(n,α)	NaI(TI)
1962	AULCBR	E.Weigoid+	14.77 (13.86–14.78)	827 ± 66	821	0.9928		Cu^{N}(n,2n)	NaI(TI) 2 × φ1.75cm
1960			14.5	855 ± 165	807.5	1.0252	0.921	^{65}Cu(n,2n)	
1960	USAKTY	B.D.Kern+	12.62 16.75	350 700		0.886			NaI(TI)
1958	LRL	V.J.Ashby+	14.1	870 ± 70	945	1.0866		STANK(cd)	

Note:

F1:Adjustment Factor for Neutron Energy

F2:Adjustment Factor for Standard Cross Section

F3:Adjustment Factor for Nuclear Decay Data

Method of Neutron Fluence Determination:AP–Associated particle

PT–Proton recoil telescope

Evaluated cross section for En = 14.7Mev: 770 ± 10mb

Table 2 Survey of measured Cross Section for $Co^{59}(n,2n)Co^{58m,g}$ Reaction

Year	Lab.	Author	En Mev	Published Cross Section mb		Branching Ratio		Method	
				σ_m	σ_g	m / T	m / g	Fluence	Activity
1988	UKNPL	T.B.Ryves+	14.3				2.1 ± 0.3	^{56}Fe(n,p)	Ge(Li) NaI(TI) $4\pi\beta$ pc
1986	AUSIRK	S.J.Hasan+	13.893		211.28		2.22	Al(n,α)	Ge(Li) NaI(TI)
1985	GERKIG	B.M.Bahal+	14.7	478 ± 24	231 ± 10			Al(n,α) Al(n,p)	Ge(Li) 98 cm^3
1984	MORMOH	M.Berrada	13.3–14.8	✓				Al(n,α)	Ge(LI) γ+X spec.
1983	BANRAM	N.T.Molla+	14.8	342 ± 42				^{75}As(n,2n)	Ge(Li)
1982	MORMOH	A.Reggoug+	14.7	402 ± 41					Ge–In γ+x spec.
1980	USAAUB	S.K.Ghorai+	13.4–18.1		✓		✓	Al(n,α)	Ge(Li) 20 cm^3
1968	POLWWA	P.Decowski+	12.57–18.2				✓	^{65}Cu(n,2n)	NaI(TI)
1967	JPNKTO	S.Okumura+	13.4–14.96			✓		AP	NaI(TI) 7.62 × φ7.62cm
1966	GERHAM	M.Bormann+	12.55–19.2	✓	✓			^{63}Cu(n,2n) H(n,n)	NaI(TI)
1963	CANCRC	W.G.Cross	14.5				1.55 ± 0.14		
1960	USAARK	I.L.Preiss+	14.8	150 ± 5	145 ± 5			^{63}Cu(n,2n) ^{65}Cu(n,2n) Al(n,α)	

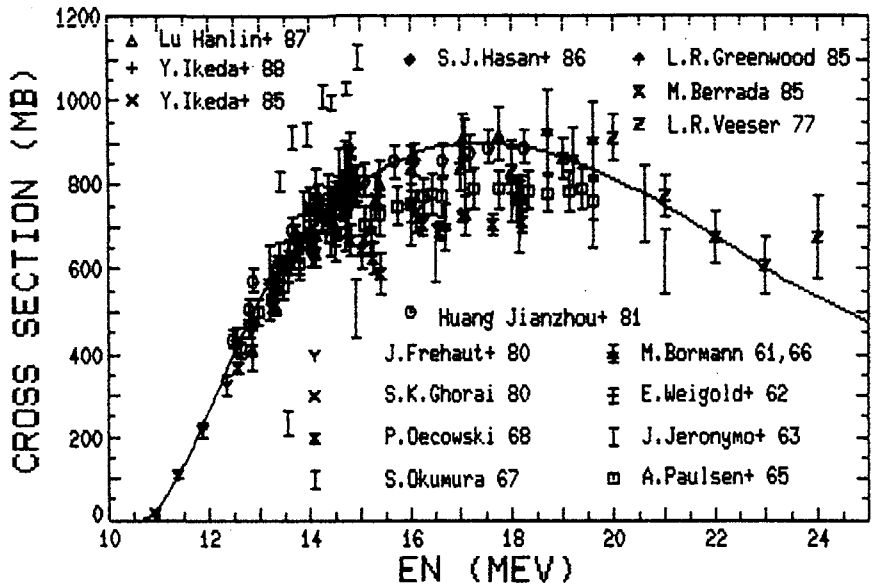

Fig.1 CO-59(n,2n)CO-58 REACTION CROSS SECTION

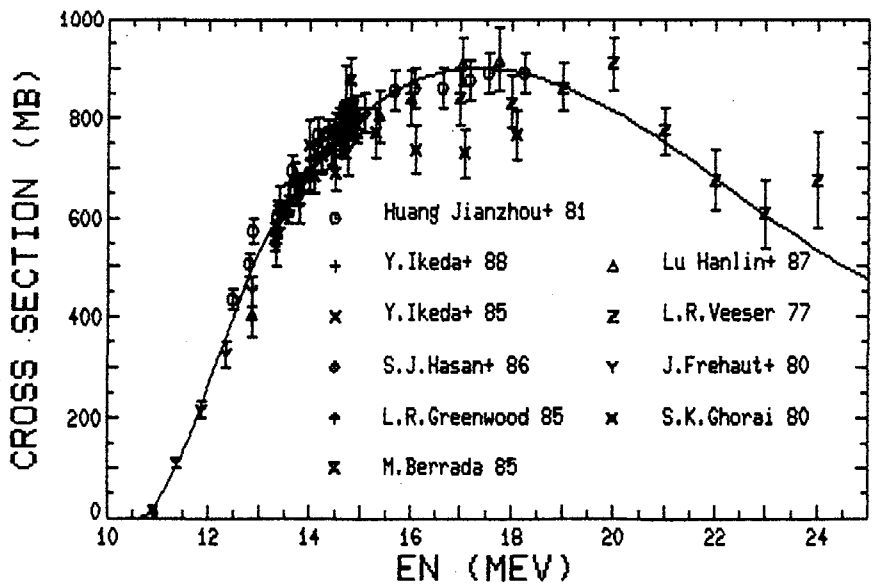

Fig.2 CO-59(n,2n)CO-58 REACTION CROSS SECTION

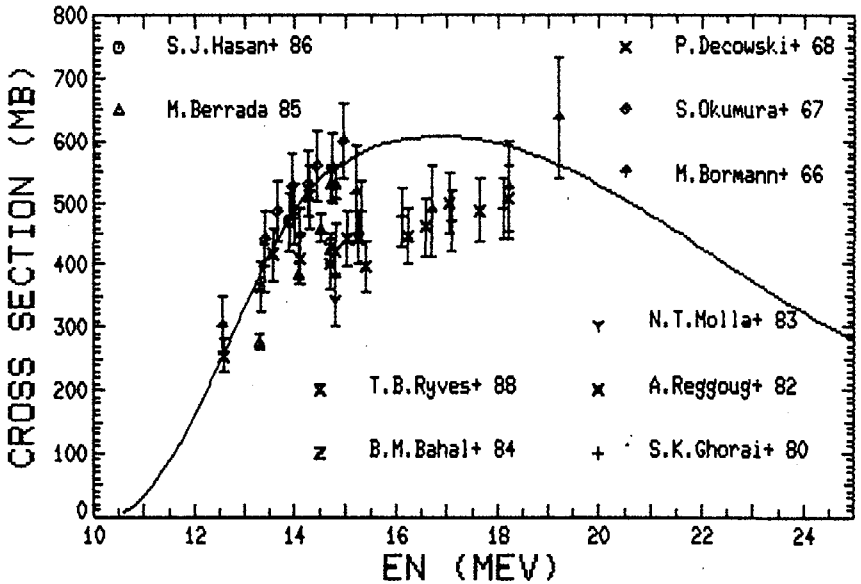

Fig.3 CO-59(n,2n)CO-58m REACTION CROSS SECTION

Table 3. The Recommended CO-59(n,2n)CO-58 Reaction Cross Section

Energy MEV	Cross-Section MB	Uncertainty MB
0.1060E+02	0.8068E+00	0.7261E-01
0.1080E+02	0.4539E+01	0.4085E+00
0.1100E+02	0.2872E+02	0.2585E+01
0.1120E+02	0.6277E+02	0.5649E+01
0.1140E+02	0.1050E+03	0.9450E+01
0.1160E+02	0.1536E+03	0.1152E+02
0.1180E+02	0.2071E+03	0.1553E+02
0.1200E+02	0.2635E+03	0.1976E+02
0.1220E+02	0.3213E+03	0.2088E+02
0.1240E+02	0.3787E+03	0.2462E+02
0.1260E+02	0.4341E+03	0.2822E+02
0.1280E+02	0.4856E+03	0.3156E+02
0.1300E+02	0.5317E+03	0.3456E+02
0.1320E+02	0.5706E+03	0.7418E+01
0.1340E+02	0.6005E+03	0.7806E+01
0.1360E+02	0.6331E+03	0.8230E+01
0.1380E+02	0.6633E+03	0.8623E+01
0.1400E+02	0.6915E+03	0.8989E+01
0.1420E+02	0.7177E+03	0.9330E+01
0.1440E+02	0.7418E+03	0.9643E+01
0.1460E+02	0.7641E+03	0.9933E+01
0.1480E+02	0.7845E+03	0.1020E+02
0.1500E+02	0.8030E+03	0.1044E+02
0.1520E+02	0.8197E+03	0.1066E+02
0.1540E+02	0.8348E+03	0.1085E+02
0.1560E+02	0.8481E+03	0.2544E+02
0.1580E+02	0.8598E+03	0.2579E+02
0.1600E+02	0.8699E+03	0.2610E+02
0.1620E+02	0.8785E+03	0.2635E+02
0.1640E+02	0.8855E+03	0.2656E+02
0.1660E+02	0.8912E+03	0.2674E+02
0.1680E+02	0.8954E+03	0.2686E+02
0.1700E+02	0.8983E+03	0.2695E+02
0.1720E+02	0.8999E+03	0.2700E+02
0.1740E+02	0.9003E+03	0.2701E+02
0.1760E+02	0.8995E+03	0.2698E+02
0.1780E+02	0.8975E+03	0.2692E+02

Table 3. The Recommended CO-59(n,2n)CO-58 Reaction Cross Section

Energy MEV	Cross-Section MB	Uncertainty MB
0.1800E+02	0.8944E+03	0.2683E+02
0.1820E+02	0.8903E+03	0.2671E+02
0.1840E+02	0.8852E+03	0.2656E+02
0.1860E+02	0.8791E+03	0.4395E+02
0.1880E+02	0.8722E+03	0.4361E+02
0.1900E+02	0.8643E+03	0.4322E+02
0.1920E+02	0.8557E+03	0.4278E+02
0.1940E+02	0.8463E+03	0.4231E+02
0.1960E+02	0.8363E+03	0.4181E+02
0.1980E+02	0.8255E+03	0.4127E+02
0.2000E+02	0.8142E+03	0.4071E+02
0.2020E+02	0.8023E+03	0.7221E+02
0.2040E+02	0.7899E+03	0.7109E+02
0.2060E+02	0.7770E+03	0.6993E+02
0.2080E+02	0.7637E+03	0.6873E+02
0.2100E+02	0.7500E+03	0.6750E+02
0.2120E+02	0.7361E+03	0.6625E+02
0.2140E+02	0.7218E+03	0.6496E+02
0.2160E+02	0.7074E+03	0.6367E+02
0.2180E+02	0.6928E+03	0.6235E+02
0.2200E+02	0.6781E+03	0.6103E+02
0.2220E+02	0.6633E+03	0.5970E+02
0.2240E+02	0.6484E+03	0.5836E+02
0.2260E+02	0.6336E+03	0.5702E+02
0.2280E+02	0.6189E+03	0.5570E+02
0.2300E+02	0.6044E+03	0.5440E+02
0.2320E+02	0.5899E+03	0.5309E+02
0.2340E+02	0.5758E+03	0.5182E+02
0.2360E+02	0.5619E+03	0.5057E+02
0.2380E+02	0.5483E+03	0.4935E+02
0.2400E+02	0.5351E+03	0.4816E+02
0.2420E+02	0.5223E+03	0.4701E+02
0.2440E+02	0.5099E+03	0.4589E+02
0.2460E+02	0.4981E+03	0.4483E+02
0.2480E+02	0.4869E+03	0.4382E+02
0.2500E+02	0.4763E+03	0.4287E+02

Table 4. Correlation Matrix for the Evaluated Cross Section of $Co^{59}(n,2n)Co^{58}$ Reaction

1 = 10.6–12.0Mev 2 = 12.0–12.5Mev
3 = 12.5–13.0Mev 4 = 13.0–13.5Mev
5 = 13.5–14.0Mev 6 = 14.0–14.5Mev
7 = 14.5–15.0Mev 8 = 15.0–16.0Mev
9 = 16.0–21.0Mev 10 = 21.0–25.0Mev

	1	2	3	4	5	6	7	8	9	10
1	1.0000									
2	0.0333	1.0000								
3	0.0135	0.6661	1.0000							
4	0.0009	0.5343	0.4567	1.0000						
5	0.0060	0.3581	0.3900	0.4833	1.000					
6	0.0000	0.2507	0.2576	0.3739	0.3773	1.0000				
7	0.0265	0.2630	0.2858	0.2915	0.5073	0.3297	1.0000			
8	0.0000	0.1901	0.2178	0.2553	0.2699	0.2370	0.2358	1.0000		
9	0.0000	0.0852	0.0918	0.1083	0.1078	0.1193	0.1041	0.3468	1.0000	
10	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0077	0.1732	0.0831	1.0000

Table 5. The Recommended CO-59(n,2n)CO-58m Reaction Cross Section

Energy MEV	Cross-Section MB	Uncertainty MB
0.1060E+02	0.5988E+01	0.1198E+01
0.1080E+02	0.1640E+02	0.3280E+01
0.1100E+02	0.3149E+02	0.6298E+01
0.1120E+02	0.5077E+02	0.1015E+02
0.1140E+02	0.7375E+02	0.1475E+02
0.1160E+02	0.9995E+02	0.1499E+02
0.1180E+02	0.1289E+03	0.1933E+02
0.1200E+02	0.1600E+03	0.2400E+02
0.1220E+02	0.1930E+03	0.1930E+02
0.1240E+02	0.2271E+03	0.2271E+02
0.1260E+02	0.2621E+03	0.1573E+02
0.1280E+02	0.2973E+03	0.1784E+02
0.1300E+02	0.3324E+03	0.1994E+02
0.1320E+02	0.3668E+03	0.2201E+02
0.1340E+02	0.4000E+03	0.2400E+02
0.1360E+02	0.4315E+03	0.2589E+02
0.1380E+02	0.4609E+03	0.2765E+02
0.1400E+02	0.4877E+03	0.2926E+02
0.1420E+02	0.5113E+03	0.3068E+02
0.1440E+02	0.5313E+03	0.3188E+02
0.1460E+02	0.5472E+03	0.3283E+02
0.1480E+02	0.5586E+03	0.3352E+02
0.1500E+02	0.5649E+03	0.3389E+02
0.1520E+02	0.5767E+03	0.3460E+02
0.1540E+02	0.5840E+03	0.3504E+02
0.1560E+02	0.5902E+03	0.5902E+02
0.1580E+02	0.5954E+03	0.5954E+02
0.1600E+02	0.5996E+03	0.5996E+02
0.1620E+02	0.6030E+03	0.6030E+02
0.1640E+02	0.6054E+03	0.6054E+02
0.1660E+02	0.6069E+03	0.6069E+02
0.1680E+02	0.6076E+03	0.6076E+02
0.1700E+02	0.6075E+03	0.6075E+02
0.1720E+02	0.6066E+03	0.6066E+02
0.1740E+02	0.6050E+03	0.6050E+02
0.1760E+02	0.6026E+03	0.6026E+02
0.1780E+02	0.5995E+03	0.5995E+02
0.1800E+02	0.5958E+03	0.5958E+02
0.1820E+02	0.5914E+03	0.5914E+02

Table 5.The Recommended CO-59(n,2n)CO-58m Reaction Cross Section

Energy MEV	Cross-Section MB	Uncertainty MB
0.1840E+02	0.5864E+03	0.5864E+02
0.1860E+02	0.5809E+03	0.8713E+02
0.1880E+02	0.5748E+03	0.8622E+02
0.1900E+02	0.5682E+03	0.8523E+02
0.1920E+02	0.5611E+03	0.8417E+02
0.1940E+02	0.5535E+03	0.8303E+02
0.1960E+02	0.5456E+03	0.8184E+02
0.1980E+02	0.5372E+03	0.8058E+02
0.2000E+02	0.5285E+03	0.7928E+02
0.2020E+02	0.5194E+03	0.1039E+03
0.2040E+02	0.5100E+03	0.1020E+03
0.2060E+02	0.5004E+03	0.1001E+03
0.2080E+02	0.4905E+03	0.9810E+02
0.2100E+02	0.4804E+03	0.9608E+02
0.2120E+02	0.4701E+03	0.9402E+02
0.2140E+02	0.4597E+03	0.9194E+02
0.2160E+02	0.4492E+03	0.8984E+02
0.2180E+02	0.4385E+03	0.8770E+02
0.2200E+02	0.4278E+03	0.8556E+02
0.2220E+02	0.4170E+03	0.8340E+02
0.2240E+02	0.4063E+03	0.8126E+02
0.2260E+02	0.3955E+03	0.7910E+02
0.2280E+02	0.3849E+03	0.7698E+02
0.2300E+02	0.3743E+03	0.7486E+02
0.2320E+02	0.3638E+03	0.7276E+02
0.2340E+02	0.3535E+03	0.7070E+02
0.2360E+02	0.3433E+03	0.6866E+02
0.2380E+02	0.3334E+03	0.6668E+02
0.2400E+02	0.3237E+03	0.6474E+02
0.2420E+02	0.3143E+03	0.6286E+02
0.2440E+02	0.3051E+03	0.6102E+02
0.2460E+02	0.2963E+03	0.5926E+02
0.2480E+02	0.2878E+03	0.5756E+02
0.2500E+02	0.2798E+03	0.5596E+02

Evaluation of $^{59}Co(n,\alpha)^{56}Mn$ Reaction

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Cobalt is monoisotope, therefore, this reaction is the only neutron induced process to produce ^{56}Mn. Its half-life of 2.578h and the branch ratio, 98.8%,of the dominant gamma line of 847 keV are well known and very convenient for activation measurements. Maybe these made the cross sections consistent in the range 14–15 MeV.

Twenty-four references [Ref.2–25] were found in literatures. Only the values of the cross sections around 14 MeV are listed in Table 1. For evaluating at 14.7 MeV all collected data were adjusted using the unified standard cross sections and energy dependence, which were taken from Ref.1. The relevant cross section and energy adjusted factors of R_σ and R_E are also given in the Table 1 separately, in which PRS is the simple words for proton recoil spectrometer, while σ_0 andσ represent the original and the adjusted cross sections respectively. The half-life and branch ratio for this reaction were unnecessary to do any revision because nearly the all authors used the same values for these parameters. After adjusting an average of the data was made, and it was found that all adjusted data were fallen in the range of three-standard deviation. So the evaluation was finished using the all adjusted data listed in Table 1, with a weighting factor of reciprocal of squared error. Some errors given by authors were enlarged during the evaluating for getting reasonable weighting factor. However, the evaluated uncertainty was calculated using the original errors. The evaluated value is (31.2 ± 0.5) mb at 14.7 MeV.

The measurements of Huang Jianzhou [Ref.16], Li Tinyan [Ref.25], D.C. Santry [Ref.9], J.M.F.Jerenymo [Ref.8], H.Liskien [Ref.11] and Y.Ikeda [Ref.23] were normalized to the value of the cross section at 14.7 MeV. The evaluation was performed by using a spline function fitting code to fit the normalized data. The evaluated results are tabulated in Table 2 with a step of 0.2 MeV and plotted in Fig.1 along with the other compiled data and Fig.2 together with the normalized six sets of data used in the evaluation. The evaluated uncertainty in the threshold vicinity is over 25% and about 5% in the tailing region of the curve. The errors for the most part of the excitation function amount to 2–3%. The error correlations were determined for the data in each set and the correlation matrix of the evaluated cross section of the reaction is summarized in Table 3.

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Table 1 Cross section and relative parameter at 14.7 MeV

Author	Year	E_n MeV	Neutron Flux	σ_0(mb)	$\Delta\sigma$ mb	R_σ	R_B	σ(mb)
E.B.Paul	53	14.5		39.1	7.8		1.0074	38.8
H.G.Blosser	58	14	^{56}Fe(n,p)	31	3	0.9565	1	32.4
E.Weigold	60	14.5		29	6		1.0074	28.8
I.L.Preiss	60	14.8	Al(n,α)	30	3	1.0159	0.9952	29.7
M.Bormann	61	14.1		29.5	1		1.0032	29.4
F.Gabbard	62	14.5	^{56}Fe(n,p)	34	3	0.9761	1.0074	34.6
J.M.F.Jeronymo	63	14.9		26	3		0.9920	26.2
D.C.Santry	64	14.5	^{32}S(n,p)	30	0.9		1.0074	29.8
C.S.Khurana	65	14.8	PRS	32	5		0.9952	32.2
H.Liskien	65	14.8	PRS	26.9	1.9		0.9952	27.0
E.Frevert	65	14.8		30	3		0.9952	30.1
Levkovskij	68	14.8		32	7		0.9952	32.2
J.C.Robertson	73	14.78	^{56}Fe(n,p)	32.3	1.1	1.0271	0.9968	31.6
S.K.Ghorai	80	14	Al(n,α)	32	3	1.0384	1	30.8
Huang Jianzhou	81	14.58	^{56}Fe(n,p)	30.2	0.9	1.0158	1.0039	29.6
H.M.Agrawal	84	14.62	^{56}Fe(n,p)	33.9	1		1.003	33.8
B.M.Bahal	85	14.7		30.2	1.5		1	30.2
I.Garlea	85	14.8	^{235}U(n,f)	35	5.8		0.9952	35.2
M.Berrada	85	14.8		28.4	1		0.9952	28.5
R.Fischer	86	14.1	α-emission	32.8	1.6		1.0032	32.7
J.W.Meadows	87	14.7	^{238}U(n,f)	30.8	1.4		1	30.8
Y.Ikeda	88	14.66	^{93}Nb(n,2n)	32.3	1.6	1.0107	1.0016	31.9
T.B.Ryves	89	14.7		32.2	2		1	32.2
Li Tingyan	90	14.72	Al(n,α)	31.3	1.9	1	0.9990	31.4
Evaluated Cross Section						(31.2 ± 0.5) mb		

Table 2 Evaluated Cross Section of ^{59}Co(n,α)^{56}Mn Reaction

E_n(MeV)	σ(mb)	$\Delta\sigma$(mb)	E_n(MeV)	σ(mb)	$\Delta\sigma$(mb)	E_n(MeV)	σ(mb)	$\Delta\sigma$(mb)
5.0	0.014	0.007	10.0	17.0	0.5	15.0	30.7	0.7
5.2	0.17	0.05	10.2	17.7	0.6	15.2	30.3	0.8
5.4	0.36	0.03	10.4	18.5	0.6	15.4	29.7	0.7
5.6	0.59	0.03	10.6	19.4	0.6	15.6	29.0	0.7
5.8	0.85	0.04	10.8	20.2	0.6	15.8	28.3	0.7
6.0	1.18	0.04	11.0	21.0	0.7	16.0	27.5	0.7
6.2	1.56	0.06	11.2	21.8	0.7	16.2	26.7	0.8
6.4	2.02	0.08	11.4	22.6	0.7	16.4	25.8	0.8
6.6	2.55	0.10	11.6	23.3	0.7	16.6	24.9	0.8
6.8	3.17	0.16	11.8	24.0	0.8	16.8	24.0	0.7
7.0	3.89	0.16	12.0	24.7	0.8	17.0	23.1	0.7
7.2	4.70	0.21	12.2	25.5	0.8	17.2	22.2	0.8
7.4	5.58	0.25	12.4	26.3	0.8	17.4	21.2	0.7
7.6	6.5	0.2	12.6	27.2	0.8	17.6	20.4	0.7
7.8	7.5	0.3	12.8	28.0	0.8	17.8	19.5	0.7
8.0	8.4	0.3	13.0	28.8	0.7	18.0	18.6	0.7
8.2	9.4	0.3	13.2	29.6	0.7	18.2	17.8	0.7
8.4	10.4	0.4	13.4	30.2	0.8	18.4	17.0	0.7
8.6	11.3	0.4	13.6	30.8	0.8	18.6	16.3	0.8
8.8	12.2	0.4	13.8	31.1	0.8	18.8	15.5	0.8
9.0	13.1	0.5	14.0	31.4	0.7	19.0	14.9	0.7
9.2	13.9	0.5	14.2	31.5	0.6	19.2	14.2	0.9
9.4	14.8	0.5	14.4	31.5	0.6	19.4	13.6	0.8
9.6	15.5	0.5	14.6	31.4	0.5	19.6	13.1	0.8
9.8	16.3	0.5	14.8	31.1	0.5	19.8	12.6	0.8

Table 3. Correlation Matrix for the Evaluated Cross-section of $^{59}Co(n,\alpha)^{56}Mn$ Reaction

1 = 5.0–6.0MeV	2 = 6.0–7.0MeV
3 = 7.0–8.0MeV	4 = 8.0–9.0MeV
5 = 9.0–10.0MeV	6 = 10.0–11.0MeV
7 = 11.0–12.0MeV	8 = 12.0–13.0MeV
9 = 13.0–13.5MeV	10 = 13.5–14.0MeV
11 = 14.0–14.5MeV	12 = 14.5–15.0MeV
13 = 15.0–15.5	14 = 15.5–16.0MeV
15 = 16.0–17.0MeV	16 = 17.0–18.0MeV
17 = 18.0–19.0MeV	18 = 19.0–20.0MeV
19 = 20.0–21.0MeV	

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
1	100																		
2	60	100																	
3	55	61	100																
4	54	56	58	100															
5	30	41	52	52	100														
6	30	41	52	52	54	100													
7	34	35	33	35	52	58	100												
8	35	37	35	36	34	33	51	100											
9	30	32	30	30	31	28	33	44	100										
10	30	32	30	30	29	28	29	53	53	100									
11	30	32	30	31	29	28	33	42	62	67	100								
12	35	37	34	35	34	33	37	45	51	70	72	100							
13	17	18	17	18	17	15	19	32	28	33	34	37	100						
14	18	20	19	21	19	16	21	37	39	37	44	41	49	100					
15	19	21	20	22	20	16	21	42	39	44	46	57	43	64	100				
16	16	17	16	17	16	14	17	30	31	37	38	51	28	59	79	100			
17	16	17	16	16	15	14	17	27	35	34	40	41	20	42	49	69	100		
18	17	18	17	18	15	16	16	16	11	19	17	16	16	27	36	42	48	100	
19	10	10	9	10	10	8	10	20	9	13	20	17	8	14	11	23	42	51	100

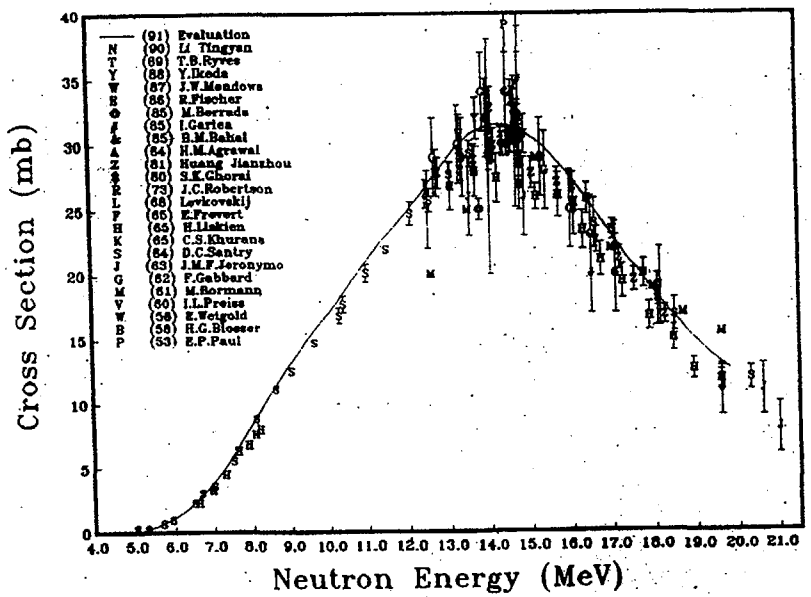

Fig.1 59-Co(n,α)Mn-56 Reaction

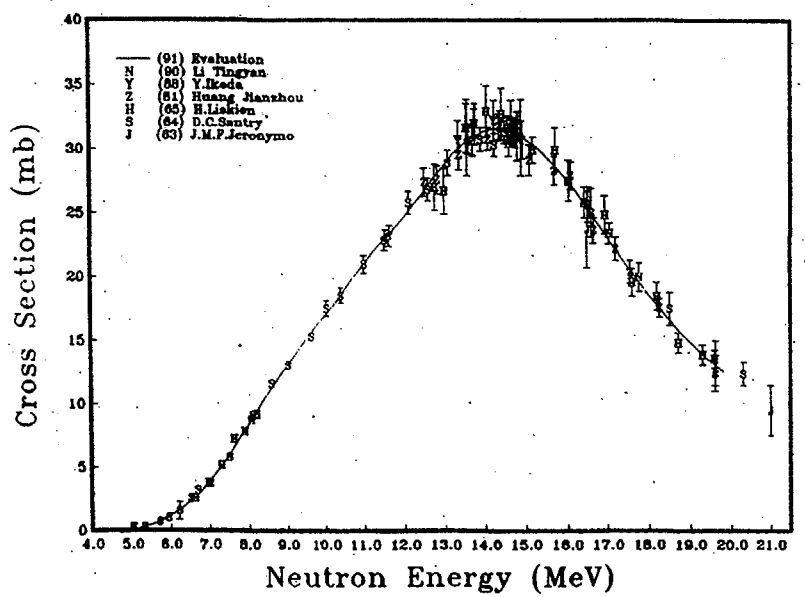

Fig.2 59-Co(n,α)Mn-56 Reaction

Evaluation of $^{109}Ag(n,\gamma)^{110m}Ag$ Reaction Cross Sections and Their Covariance Data

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I. INTRODUCTION

Neutron data of silver are of importance for nuclear application. Stable isotopes ^{107}Ag and ^{109}Ag are formed as fission products. The cross sections of $^{109}Ag(n,\gamma)^{110m}Ag$ are useful in reactor dosimetry study. Unfortunately,a survy shows that very few data only measured at thermal neutron energy [1-3] and around 25 keV [4-5] are available because the half-life of ^{110m}Ag is very long($T_{1/2}$= 250d [6]) and interference came from the neutrons with low energy. So the theory and systematics calculation have to be used in this evaluation.

The semi-classical theory code UNIFY2 of multi-step nuclear reaction [7] was used to calculate the ratio of cross section of $^{109}Ag(n,\gamma)^{110m}Ag$ to $^{109}Ag(n,\gamma)^{110m+g}Ag$. On basis of the ratio and the total capture cross section of ^{109}Ag evaluated through fitting measured data with the systematics of excitation function of (n,γ) reaction [8] and calculating the average cross section in the resonance region, the evaluation of cross section for $^{109}Ag(n,\gamma)^{110m}Ag$ was performed.

Chap.2 describes the evaluation of total neutron capture cross section on ^{109}Ag. The detail of theory calculation on the ratio of cross section of the isomer state to the sum of the isomer and ground states is represented in Chap.3. In the Chap.4, the described is the evaluation of covariance data. The comparison and remarks are given in Chap.5.

II. EVALUATION OF TOTAL NEUTRON CAPTURE CROSS SECTION ON ^{109}Ag

In the neutron energy region below 2.5 keV, a set of evaluated resonance parameters of JENDL-3 [9] was used to calculate cross sections by using the code MSBW [10] . The thermal cross section calculated from the resonance parameters is 90.721 b which is in good agreement with the evaluated value 91.0 ± 1.0 b of Ref.[11] and measured data. Using same set of resonance parameters, the average cross sections were calculated in the neutron energy region from 1.0 eV to 2.5 keV. In the smooth region(above 2.5 keV), Zhao's systematics [8] was used to fit measured data. A large discrepancy among measured data available is observed. Only two measurements of Macklin [12] and Mizumoto et al. [13] were adopted in the fitting. The comparison of the systematics with measured data and evaluated ones of ENDF/B-6 and JENDL-3 is given in Fig.1. From Fig.1, it can be found that this evaluation is in good agreement with measured data and other evaluations below 2 MeV. Above 2 MeV where no measured data are available, this evaluation differ from other evaluations.

III. THEORETICAL CALCULATION OF $R = \frac{\sigma^m}{\sigma^{m+g}}$

The semi-classical theory code UNIFY of multi-step nuclear reaction [7] was used to perform a calculation to obtain the ratio of isomer capture cross section to total capture cross section. In the calculation, the optical model parameters, the giant resonance parameters and level density parameters were taken from Ref.[14]. To the calculation of the ratio, most important parameters are the branching ratio of gamma transition of ^{110}Ag. The data used in this calculation are given in Tab.1.

The result of calculation was normalized to measured data. The comparison of normalized ratio and measured data is shown in Fig.2.

VI. EVALUATION OF COVARIANCE DATA

Because of

$$\sigma^m = \sigma^{m+g}R$$

it follows that

$$\frac{Var(\sigma^m(E_i))}{[\sigma^m(E_i)]^2} = \frac{Var(\sigma^{m+g}(E_i))}{[\sigma^{m+g}(E_i)]^2} + \frac{Var(R(E_i))}{R(E_i)^2}$$

Table 1 Branching Ratio Data for ^{110}Ag

No.	EL,MeV	1	2	3	4	5
1	0.0					
2	0.0011	1.000				
3	0.1176	Isomeric State				
4	0.1916			1.000		
5	0.1987	1.000				
6	0.2369		1.000			
7	0.2371	1.000				
8	0.2672	0.810	0.085			0.105
9	0.2694	0.810	0.085			0.105
10	0.2714			0.207	0.793	
11	0.3045	0.046				0.954

and

$$\frac{Cov(\sigma^{m}(E_i),\sigma^{m}(E_j))}{\sigma^{m}(E_i)\sigma^{m}(E_j)} = \frac{Cov(\sigma^{m+g}(E_i),\sigma^{m+g}(E_j))}{\sigma^{m+g}(E_i)\sigma^{m+g}(E_j)} + \frac{Cov(R(E_i),R(E_j))}{R(E_i)R(E_j)}$$

so that the relative uncertainty of the isomer cross section consists of two components came from the relative uncertainties of total capture cross section and of the ratio. These components are given as follows. The uncertainty of total capture cross sections:

$1.0E^{-5}$eV to 1.0 eV:

given by the error of thermal cross section;

1.0 eV to 0.65 keV:

given by the errors of the gamma width for individual resonance;

0.65 to 2.5 keV:

given by the standard deviation of the average gamma width;

2.5 keV to 2 MeV:

given by the systematical errors of the measurements of Ref.[12] and Ref.[13];

2 MeV to 20 MeV:

given by uncertainty of the systematics.

The uncertainty of the ratio:

given by normalization error.

V. COMPARISON AND REMARKS

The isomer capture cross sections of ^{109}Ag and their covariance data were evaluated in the energy range from $1.0E^{-5}$eV to 20 MeV. The present evaluation is based on experimental information including the measured data of total capture cross section, systematics and theoretical calculation.

The comparison of this evaluation with those of ENDF / B-6 and the limited measured data is shown in Fig.3.

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The author would like to thank Zhou Delin and Cai Dunjiu for their helpful discussion. He also wishes to thank Zhang Jinshang for his help in using UNIFY.

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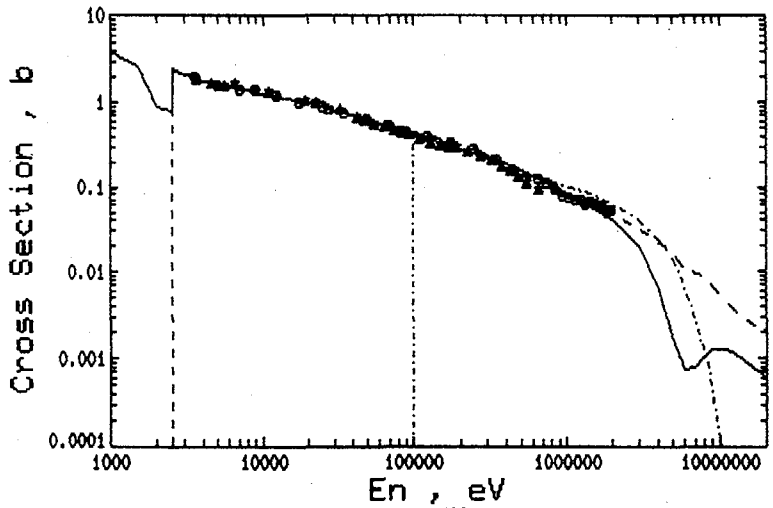

Fig.1 Capture Cross Section of ^{109}Ag
—— this work —·—·— ENDF / B-6 - - - JENDL - 3

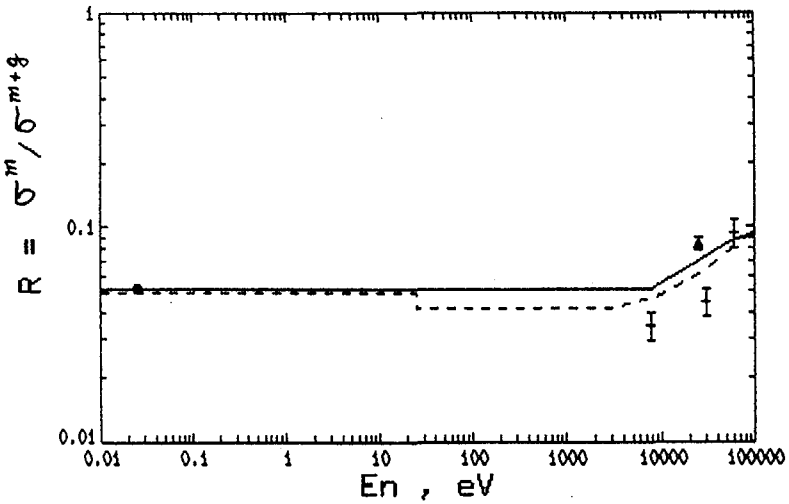

Fig.2 Ratio of Capture Cross Section

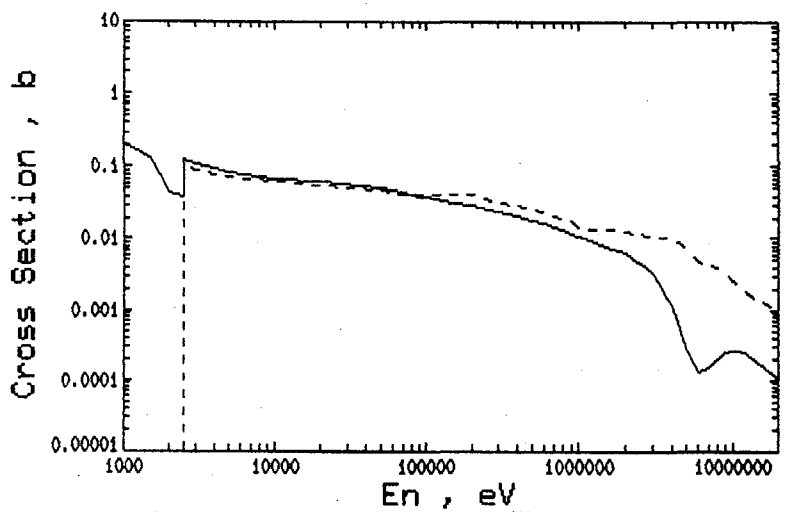

Fig.3 Capture Cross Section of ^{109}Ag(n,gamma) ^{110m}Ag
—— this work ----- ENDF / B-6

Evaluation of the $^{115}In(n,n')$ ^{115m}In reaction cross section

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The excitation function of $^{115}In(n,n')$ ^{115m}In reaction covers considerable wide neutron energy range from threshold of 0.399MeV to 20MeV. So it is an excellent neutron indicator for neutron dosimetry monitor and spectrum measurements.

Thirty three sets of data available [1-33] are shown in Tables 1,2 and Fig.1 which indicate poor consistency of the data, especially in about 14MeV neutron energy range [1-20].

There are two reasons causing the data scattering. The first, part of the primary neutrons from T–D or D–T reaction, which is usually used as neutron source, is scattered through target backing and assembly including the target–cooling system and beam tube, which produce a small low energy component in the neutron field; There is other part of low energy neutrons from (D–d) self build–up target in the target and beam collimator and from D(d,np)D reaction. This kind of neutrons has a small effect on the reactions with high threshold, but has a large influence on $^{115}In(n,n')$ ^{115m}In reaction. Even in the condition of well designed experiments, the activities may increase by several percent due to the effect of low energy neutrons. The second, there is an apparent presence of structure in the excitation curve in neutron energy range of 2–6 MeV, which makes measurements complicated. The positions of structures, Which depend on the energy resolution of the experiment, weren't coincident.

According to this situation and some recent experimental informations including our own measurement results, The $^{115}In(n,n')$ ^{115m}In reaction cross section has been reevaluated.

The experimental data were analyzed and treated as the following ways:

Around 14MeV neutron energy

After analyzing the collected data, we found: Most measurements suffer from large systematic errors because the incident neutron beam is always contaminated by the low energy neutrons; Recently, Lu Hanlin et al. [1] and Ryves et al. [4] have made efforts to reduce and determine precisely the correction induced by these low energy neutrons, and the corrections for neutrons scattering in the sample and target system were calculated using the Monte Carlo method [1-4]. These three data [1,2,4] were used to make a new evaluation at 14.7MeV, the evaluated cross section is 52.36 ± 2 mb, which is about 17% lower than those of the existing evaluation [34-36].

Then, H.O.Menlove's excitation function for En = 13.28–19.39 MeV [21] was normalized as the recommended value at 14.7 MeV and used in present evaluation. The values of Lu and Ryves were adopted also.

For the neutron energy range of 8–14 MeV

The data of D.C.santry [22] and D.L.Smith [23] were adopted. While santry's data were corrected for the effect of low energy neutron. The effect is decreased with neutron energy decreasing in this neutron energy range.

In the neutron energy range between threshold and 8 MeV–"Structure" range

The data of Fan Peiguo, D.C.Santry, D.L.Smith and H.Liskien [24] were adopted. While the D.L.Smith's data were corrected for fission cross section of ^{238}U taken from ENDF / B4 by data of A.B.Smith [37].

The evaluation was performed for above mentioned data by means of a program of spline function fitting for multi–set data.The results are plotted in Fig.1 and summarized in Table 3 with 0.1, 0.2 or 0.5 MeV energy steps from threshold to 20 MeV. The estimated accuracy is 10% near threshold, 2–4% between1–12 MeV, 3–5% between 12–16 MeV and 6–10% up to 20 MeV.

Variance and Covariance Matrix

The uncertainties and their correlations for measured quantities of the $^{115}In(n,n')$ ^{115m}In reaction among the various data sets were analyzed and considered. The described covariance of the $^{115}In(n,n')$ ^{115m}In reaction cross section from threshold energy to 20MeV is divided into 8 energy interval. The covariance of each energy interval depends on its own set uncertainties of all available experimental data and the correlations with other data sets over energy intervals. The correlation matrix for the evaluated data is given in Table 4 [38]

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Table 1 Cross section for In-115(n,n')In-115m reaction around 14 MeV

Year	Author	En MeV	Cross Section, mb		Adjusted Factor			Methods	Correction for low energy neutrons
			published	Adjusted	F1	F2	F3		
89	Lu Hanlin et al.	14.8	50.8±1.8	51.054±2	1.005	1	1	H(n,n),Al(n,α),Asso.α	Fine
80	Fan Peigou et al.	14.63	60.4±3.1	53.06±3.3	.9943			H(n,n),Al(n,α),Asso.α	Fine
83	T.B.Ryves et al.	14.67	53.1±2.2	52.97±2.2	.9976	1	1	Fe-56(n,p)	Fine
64	I.Heertje et al.	14.6	80.0±3.0		.9919				
65	W.Nagel et al.	14.6	50.0±7.8	57.33	.9919	1.0612	1.0893	Fe-56(n,p)	Part
67	H.O.Menlove et al.	14.96	61.6±6.3		1.006		1.0893	Al(n,α); U-235(n,f)	
68	B.Minetti et al.	14.70	125±10					Cu-63(n,2n);Asso.α	
68	H.Roetzer et al.	14.70	83.5±4.2		1	.9636		Al(n,α)	
69	R.C.Barrall et al.	14.60	67.0±7.0	69.03	.9919	.9536	1.0893	Al(n,α);H(n,n)	Part
69	R.C.Barrall et al.	14.80	69.0±5.0	73.54	1.005	.9733	1.0893	Al(n,α);H(n,n)	Part
70	J.K.Temperley et al.	14.10	73.0±8.0	73.75	.9135	1.0745	1.0292	Fe-56(n,p)	
70	P.Decowski et al.	14.52	83.8±1.2		.9829			Zn-64(n,2n)	Part
72	A.Pazsit et al.	14.7	63.0±4.0	65.27 *	1	.9974	1.0388*	Al(n,α)	
76	D.C.Santry et al.	14.74	54.5±2.2	53.74	1.002	.984	1	S-32(n,p)	
76	C.G.Hudson et al.	15.2	50.0±10		1.006	.771		Fe-56(n,p)	
77	G.Magnusson	14.7	63.0±4.0	64.90 *	1	1.0152	1.0148	Al(n,α)	Part
78	P.Andersson et al.	14.9	65.0±4.0	65.39 *	1.006	1	1	H(n,n)	
83	I.Garlea et al.	14.75	78.6±3.6		1.0026			U-235(n,f)	Part
83	Demekhin	14.6	62.10		.9919			Al(n,α)	
84	K.Kudo	14.6	66.2±2.3		.9919			H(n,n)	Part
85	R.Pepelnik et al.	14.7	90.5±4.5					Al(n,α); Nb-93(n,2n)	
88	K.Kobayashi et al.	14.05	65.03±2.47		1.0089			Al(n,α)	Part

Note:

F1---Adjustment Factor for neutron energy[3]
F2---Adjustment Factor for standard cross section[36]
F3---Adjustment Factor for Nuclear Decay Data[38]
* ---from ANL/NDM-89

Evaluated Cross section for En=14.7 MeV

IRDF-82 [34]	61.2	mb
ENDF/B5 [35]	61.2	mb
ANL/NDM-89 [36]	61.75±2.92	mb
Lu Hanlin et al. [1]	52.4±1.5	mb
Present Work	52.36±2	mb

Table 2 Survey of Measurement for excitation function of In-115(n,n')In-115m reaction

Year	Author	Laboratory	En, MeV	Methods	Reference
89	Lu Hanlin et al.	CIAE	13.54 - 14.80	H(n,n), Al(n,α)	1
80	Fan Peiguo et al.	CIAE	0.88 - 8.50	H(n,n), Al(n,α)	2
89	Zhao Wenrong et al.	CIAE	0.88 - 14.80	H(n,n), Al(n,α)	3
83	T.B. Ryves et al.	UKNPL	14.3;14.67	Fe-56(n,p)	4
67	H.O.Menlove et al.	USALOK	13.28 - 19.39	Al(n,α);U-235(n,f)	21
76	D.C.Santry et al.	CANCRC	0.351 - 14.74	S-32(n,p);LC	22
76	D.L.Smith et al.	ANL	0.44 - 10.0	U-235(n,f);U-238(n,f)	23
78	H.Liskien et al.	CBNM	0.403 - 4.1	H(n,n)	24
54	H.C.Martin et al.	LASL	0.44 - 5.26		25
68	H.A.Grench et al.	USALOK	0.363 - 1.017	Au-197(n,γ)	26
69	I.Kimura et al.	KURR1	0.5 - 4.6	H(n,n)	27
73	K.Kobayashi et al.	KURR1	3.37 - 4.89	H(n,n)	28
78	S.Yamamoto et al.	KURR1	5.6 - 7.65	H(n,n)	31
69	P.Decowski et al.	POLIBJ	12.98 - 17.83	Zn-64(n,2n)	29
77	L.Adamski et al.	POLIBJ	0.53 - 1.31	In-115(n,γ)In-116m	30
79	C.F.Ai et al.	CHFSHI	0.99 - 4.23	H(n,n)	32
81	K.Kudo et al.	JPN	2.49;14.6	H(n,n)	18
68	P.Bornemisza et al.	HUNDEB	2.8	Cd-111(n,n')	33

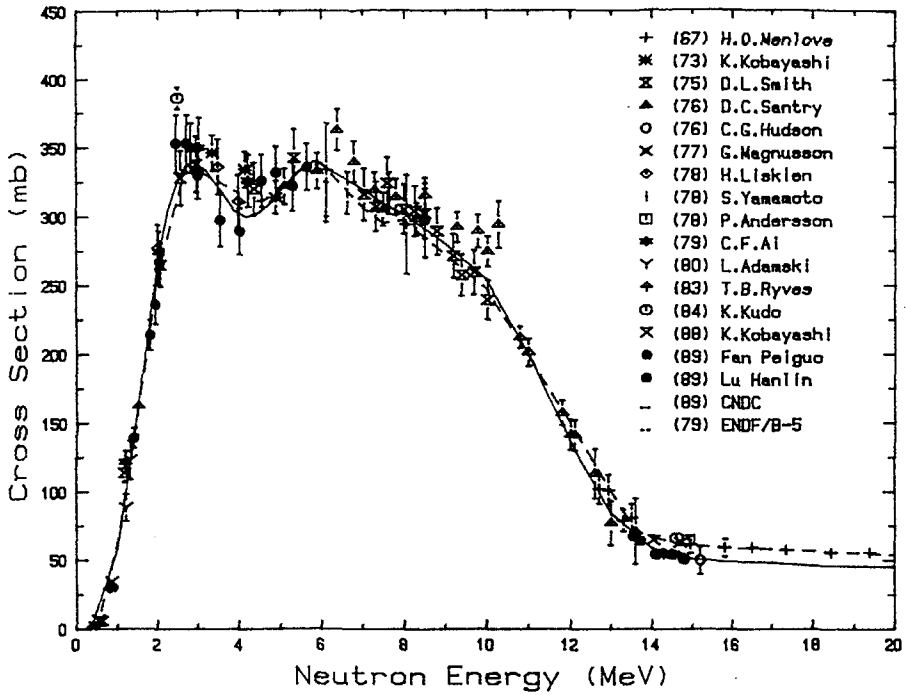

Fig.1 In-115(n,n')In-115m

Table 3 In-115(n,n')In-115m Reaction Cross Section

Recommended Reference Data
Linear-Linear Interpolation

Neutron Energy MeV	Cross Section mb	Error mb	Neutron Energy MeV	Cross Section mb	Error mb
0.34	0.0	0.0	1.6	185.6	1.7
0.4	1.2	0.1	1.7	207.7	2.0
0.6	8.2	0.1	1.8	228.9	2.1
0.8	27.9	0.3	1.9	248.9	2.2
1.0	58.2	0.6	2.0	267.0	2.3
1.1	76.7	0.7	2.1	283.1	2.2
1.2	96.8	0.8	2.2	297.1	2.2
1.3	118.2	1.0	2.3	309.0	2.2
1.4	140.4	1.2	2.4	318.9	2.3
1.5	163.0	1.5	2.5	326.9	2.5

Table 3 In-115(n,n')In-115m Reaction Cross Section

Recommended Reference Data
Linear-Linear Interpolation

Neutron Energy MeV	Cross Section mb	Error mb	Neutron Energy MeV	Cross Section mb	Error mb
2.6	332.9	2.7	6.9	322.5	4.4
2.7	337.1	3.0	7.0	320.0	4.5
2.8	339.6	3.1	7.1	317.7	4.5
2.9	340.3	3.2	7.2	315.5	4.5
3.0	339.5	3.2	7.3	313.5	4.4
3.1	337.5	3.1	7.4	311.6	4.2
3.2	334.4	2.9	7.5	309.7	4.1
3.3	330.6	2.8	7.6	307.9	3.9
3.4	326.2	2.7	7.7	306.3	3.8
3.5	321.5	2.7	7.8	304.6	3.7
3.6	316.7	2.8	7.9	303.0	3.6
3.7	312.1	2.9	8.0	301.5	3.6
3.8	307.9	3.1	8.2	298.3	3.8
3.9	304.4	3.2	8.4	295.2	4.0
4.0	301.8	3.3	8.6	291.9	4.3
4.1	300.2	3.4	8.8	288.3	4.5
4.2	299.6	3.3	9.0	284.4	4.4
4.3	299.9	3.2	9.2	279.9	4.2
4.4	301.1	3.0	9.4	274.9	3.8
4.5	302.9	2.8	9.6	269.1	3.5
4.6	305.3	2.5	9.8	262.3	3.4
4.7	308.1	2.3	10.0	254.4	3.6
4.8	311.4	2.0	10.5	229.8	4.0
4.9	314.9	1.8	11.0	200.1	3.9
5.0	318.5	1.7	11.5	167.9	3.6
5.1	322.2	1.7	12.0	136.0	3.5
5.2	325.8	1.8	12.5	107.1	3.5
5.3	329.3	2.0	13.0	83.8	2.9
5.4	332.4	2.2	13.5	68.1	1.5
5.5	335.2	2.5	14.0	58.5	0.9
5.6	337.4	2.7	14.5	53.1	0.7
5.7	339.1	2.8	15.0	50.6	1.3
5.8	340.1	2.9	15.5	49.7	2.5
5.9	340.4	3.0	16.0	49.4	3.2
6.0	340.1	3.1	16.5	48.6	3.5
6.1	339.3	3.2	17.0	47.6	4.5
6.2	338.1	3.3	17.5	46.6	5.2
6.3	336.5	3.4	18.0	45.8	4.7
6.4	334.6	3.6	18.5	45.4	6.3
6.5	332.4	3.8	19.0	45.5	7.0
6.6	330.0	4.0	19.5	45.6	8.1
6.7	327.5	4.2	20.0	45.8	12.2
6.8	325.0	4.3			

Table 4 Uncertainty correlation matrix for the evaluated Cross sections of $^{115}In(n,n')\ ^{115m}In$ reaction

1 = 0.34–2.0MeV 2 = 2.0–4.0MeV

3 = 4.0–6.0MeV 4 = 6.0–9.0MeV

5 = 9.0–14.0MeV 6 = 14.0–15.0MeV

7 = 15.0–17.0MeV 8 = 17.0–20.0MeV

1	2	3	4	5	6	7	8	
1	1.000							
2	0.714	1.000						
3	0.602	0.156	1.000					
4	0.433	0.502	0.528	1.000				
5	0.384	0.438	0.500	0.658	1.000			
6	0.210	0.269	0.272	0.079	0.291	1.000		
7	0.054	0.053	0.054	0.053	0.079	0.118	1.000	
8	0.051	0.050	0.051	0.051	0.096	0.183	0.654	1.000

Evaluation of the $^{115}In(n,2n)^{114m}In$ Reaction Cross-section

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The excitation function of $^{115}In(n, 2n)^{114m}In$ reaction covers neutron energy range from threshold of 9.228 MeV (including the energy of isomeric state which has 0.190 MeV) to 20 MeV. It is a hopeful neutron indicator for fast neutron spectra and dosimetry applications in fusion reactors, because it has a convenient half-life and high cross sections which are useful for activation studies.

Experimental knowledge of the $^{115}In(n,2n)$ reaction cross sections is entirely based on the results of activation measurements. There are still no experimental results obtained by direct-neutron-detection methods (such as scintillation tank). For both the indium isotopes, the primary activity resulting from the(n,2n) process is due to the decay of metastable state. The interaction with the primary isotope, ^{115}In, results in a metastable state in ^{114}In which decays with 49.5d half-life by means of an E4 transition to the ground state (1^+) with a half-life of 71.9s. The (n,2n) process also directly populates the ground state.

Seventeen sets of data available [1-18] were collected and shown in Table 1 and Fig.1.There is a considerable spread among the cross section values.

For the sake of comparison, all the experimental results were adjusted by the unified nuclear parameters (including intensity of γ-line, half-life, branching ratio), the reference cross section and the dependence of cross section on energy which were taken from [Ref.1]. In several cases, adjustments were made only for the dependence of cross section on energy since some necessary parameters used in data analysis were not given by the authors.

After adjustments, the difference still exist, as shown in Fig .1 , Fig. 2 and Table 1. It is possible that there are systematic differences in the activity determinations of the residual nuclei. The poor resolution of NaI(Tl) detectors affected the accuracy of the activity determination of ^{114m}In nucleus in γ measurements, due to the existence of the ^{116m}In, ^{112m}In and ^{115}Cd nuclei. In addition, the agreement among the measurements by using Ge detector is poor. For example, Reggoug's [7] value is 20 % higher than Lu Hanlin's [1,2] and paulsen's is 20 % lower than Lu's [1,2].

Lu Hanlin et al.[1,2] and Ryves et al.[4,5] have made efforts to improve the measurement and the results, which they obtained, are better (error about 3 %)than the others.

According to this situation, the evaluation was made as the following ways:

At 14.7MeV neutron energy .

Based on the results measured by Ge(Li) detector especially of Lu Hanlin et al.[1, 2],Ryves et al.[4] and Csikai et al.[3], the recommended cross section at 14.7 MeV is 1290 ± 30mb.

For the neutron energy range from threshold to 20 MeV.

The excitation functions of Lu Hanlin et al.[1, 2, 10], Csikai[3],Ryves[4], Santry[9], Menlove[17] and Prestwood[18] were normalized to the recommended datum at 14.7 MeV and shown in Fig.2. It can be seen that the coincidence among them is better.

Then the recommended excitation function was obtained by fitting the measured data with least-square method in the neutron energy range of 10.4-20MeV and by extrapolating from 10.4MeV to 9.228MeV based on the physical trend and systematics calculation. The results are plotted in Fig.1&2 and summarized in Table 2 with 0.2 MeV energy step from threshold to 20 MeV.The estimated uncertainties of the evaluation are: 20% near threshold ,10 % from 11 to 13 MeV, 5~ 10 % between 13-17MeV, and 10~ 15 % above 17 MeV. Since many newly measured data were used in this evaluation, present evaluation should be better than the ones before.

Covariance matrix is given for the evaluated data considering all adjusted factors and shown in Table 3 [28]. Energy region is divided into the fourteen small intervals, they are 9.3-10.0MeV, 10.0-10.5MeV, 10.5-11.5MeV, 11.5-12.2MeV,12.2-12.7MeV, 12.7-13.2MeV, 13.2-13.8MeV, 13.8-14.2MeV, 14.2-14.7MeV, 14.7-15.2MeV,15.2-16.0MeV, 16.0-17.2MeV, 17.2-18.0MeV, 18.0-20.0MeV respectively.

Table 1. Compilation of Measured Cross–section for $^{115}In(n,2n)$ ^{114m}In Reaction Around 14MeV (or Some Energy Range)

Year	Author	Lab.	En MeV	Cross–setion(mb)		Adjusted Factor			Method	
				Published	Adjusted	F_1	F_2	F_3	Fluence	Activity
1989	Lu Hanlin+	CPR.IAE	14.57 (12.18–17.9)	1296 ± 31	1302.4	1.0049	1.0	1.0	AP	Ge(Li)
1989	Csikai+	Hun.IEPKU	14.66 (13.43–14.84)	1329 ± 37		1.002			LONGC $^{93}Nb(n,2n)$	Ge(Li),NaI
1988	Li Jianwei Wu Zhihua+	CPR.FUD	15.75 ± 0.09 (12.7–18.26)	1349 ± 62 1293 ± 52 1345 ± 108	1322.1 1279.1 1328.1	0.9874	0.9926 1.0019	1.0	Al(n,α) $^{56}Fe(n,p)$ PT	Ge(Li)
1983 1980	Ryves+	UK.NPL	14.67 ± 0.03 (14.3–18.12) 14.67	1250 ± 30 1241 ± 36	1255.9 1242.2	1.001 1.001	1.0037	1.0	$^{56}Fe(n,p)$	Ge,4πβ Ge(Li)
1982	Reggoug+	MDR.MOH	14.7	1540 ± 55	1571	1.0	1.02		Al(n,α)	Ge(Li)
1979	Kayashima+	JAN.KYU	14.6	1331 ± 110	1335.7	1.0035			Al(n,α)	Ge(Li)
1976	Santry+	CAN.CRC	14.68 (10.03–14.74)	1419 ± 81	1442	1.001	1.015		$^{32}S(n,p)$ LONGC	NaI
1975	Lu Hanlin+	CPR.IAE	14.6 (11.4–18)	1359 ± 91	1363.8	1.0035	1.0	1.0	Al(n,α)	NaI
1975	Paulsen+	GER CBNM	14.6 (12.8–19.59)	1000 ± 80	1004	1.0035	1.0	1.0	H(n,n), PT Al(n,α)	Ge(Li)
1970	Temperley+	USA.BRL	14.1	1210 ± 140	1349.5	1.0325	1.0802		$^{56}Fe(n,p)$	Ge(Li)
1969	Barrall+	USA.LRL USA.STF	14.8 14.6	1390 ± 110 1330 ± 120	1500 1389	0.9965 1.0035	1.0 0.9611	1.0828	H(n,n) Al(n,α)	NaI
1968	Rotzer+	AJS.IRK	14.7 ± 0.15	1470 ± 120	1489.5	1.0	1.0133		Al(n,α)	NaI,4πpc
1968	Minetti+	ITY.TUR	14.7	1590 ± 90	1590	1.0	1.0		AP, $^{63}Cu(n,2n)$	NaI
1967	Menlove+	USA.LOK USA.STF	14.96 (12.7–19.39)	1264 ± 137	1381.7	0.992		1.1019	$^{235}U(n,f)$ Al(n,α)	NaI
1961	Prestwood+	USA.LA	14.5 (12.2–19.76)	1539 ± 77	1551.6	1.0082			AP, $^{238}U(n,f)$	GM(β)

Note:

F1: Adjustment Factor for Neutron Energy at 14.7MeV.

F2: Adjustment Factor for Standard Cross–section

F3: Adjustment Factor for Nuclear Decay Data

Method of neutron fluence determination: AP: Associated particle,

PT: Proton recoil telescope

Evaluated cross–section for En = 14.7MeV:1290 ± 30 (mb)

Table 2. Evaluated Cross-sections for $^{115}In(n,2n)\ ^{114m}In$ Reaction

En(MeV)	Cross-section(mb)	Uncertainties(mb)	En(MeV)	Cross-section(mb)	Uncertainties(mb)
0.9228E+01	0.0000E+01	0.0000E+00			
0.9400E+01	0.7732E+01	0.1546E+01	0.9600E+01	0.2500E+02	0.5000E+01
0.9800E+01	0.8013E+02	0.1603E+02	0.1000E+02	0.1461E+03	0.2922E+02
0.1020E+02	0.2288E+03	0.4576E+02	0.1040E+02	0.3141E+03	0.6282E+02
0.1060E+02	0.4079E+03	0.8150E+02	0.1080E+02	0.4882E+03	0.9764E+02
0.1100E+02	0.5649E+03	0.1129E+02	0.1120E+02	0.6379E+03	0.6379E+02
0.1140E+02	0.7071E+03	0.7071E+02	0.1160E+02	0.7726E+03	0.7726E+02
0.1180E+02	0.8342E+03	0.8342E+02	0.1200E+02	0.8920E+03	0.8920E+02
0.1220E+02	0.9460E+03	0.9460E+02	0.1240E+02	0.9961E+03	0.1240E+02
0.1260E+02	0.1042E+04	0.1042E+03	0.1280E+02	0.1085E+04	0.1085E+03
0.1300E+02	0.1124E+04	0.6744E+02	0.1320E+02	0.1158E+04	0.6948E+02
0.1340E+02	0.1190E+04	0.7141E+02	0.1360E+02	0.1217E+04	0.7302E+02
0.1380E+02	0.1241E+04	0.7446E+02	0.1400E+02	0.1262E+04	0.7572E+02
0.1420E+02	0.1279E+04	0.7674E+02	0.1440E+02	0.1293E+04	0.7758E+02
0.1460E+02	0.1303E+04	0.7818E+02	0.1480E+02	0.1311E+04	0.7866E+02
0.1500E+02	0.1315E+04	0.7890E+02	0.1520E+02	0.1317E+04	0.7902E+02
0.1540E+02	0.1316E+04	0.7896E+02	0.1560E+02	0.1313E+04	0.7878E+02
0.1580E+02	0.1307E+04	0.7842E+02	0.1600E+02	0.1299E+04	0.7794E+02
0.1620E+02	0.1292E+04	0.7752E+02	0.1640E+02	0.1286E+04	0.7716E+02
0.1660E+02	0.1280E+04	0.7680E+02	0.1680E+02	0.1274E+04	0.7644E+02
0.1700E+02	0.1269E+04	0.7614E+02	0.1720E+02	0.1263E+04	0.1263E+)3
0.1740E+02	0.1257E+04	0.1257E+03	0.1760E+02	0.1254E+04	0.1254E+03
0.1780E+02	0.1246E+04	0.1246E+03	0.1800E+02	0.1239E+04	0.1239E+03
0.1820E+02	0.1230E+04	0.1230E+03	0.1840E+02	0.1220E+04	0.1220E+03
0.1860E+02	0.1206E+04	0.1206E+03	0.1880E+02	0.1193E+04	0.1193E+03
0.1900E+02	0.1178E+04	0.1178E+03	0.1920E+02	0.1159E+04	0.1159E+03
0.1940E+02	0.1146E+04	0.1146E+03	0.1960E+02	0.1127E+04	0.1127E+03
0.1980E+02	0.1112E+04	0.1112E+03	0.2000E+02	0.1103E+04	0.1103E+03

Table 3. Correlation Matrix for the Evaluated Cross-section of $^{115}In(n,2n)^{114m}In$ Reaction

1 = 9.3–10.0MeV
2 = 10.0–10.5MeV
3 = 10.5–11.5MeV
4 = 11.5–12.2MeV
5 = 12.2–12.7MeV
6 = 12.7–13.2MeV
7 = 13.2–13.8MeV
8 = 13.8–14.2MeV
9 = 14.2–14.7MeV
10 = 14.7–15.2MeV
11 = 15.2–16.0MeV
12 = 16.0–17.2MeV
13 = 17.2–18.0MeV
14 = 18.0–20.0MeV

	1	2	3	4	5	6	7	8	9	10	11	12	13	14
1	100													
2	49	100												
3	22	31	100											
4	7	10	22	100										
5	10	9	27	59	100									
6	4	7	20	45	72	100								
7	2	3	11	22	25	29	100							
8	0	0	0	13	11	13	46	100						
9	0	0	4	14	15	18	29	26	100					
10	0	0	4	15	17	21	35	29	46	100				
11	0	0	0	8	14	16	10	8	19	31	100			
12	0	0	3	5	9	12	9	6	34	25	29	100		
13	0	0	1	3	6	8	7	6	16	29	41	37	100	
14	0	0	1	1	1	2	2	1	15	5	5	23	8	100

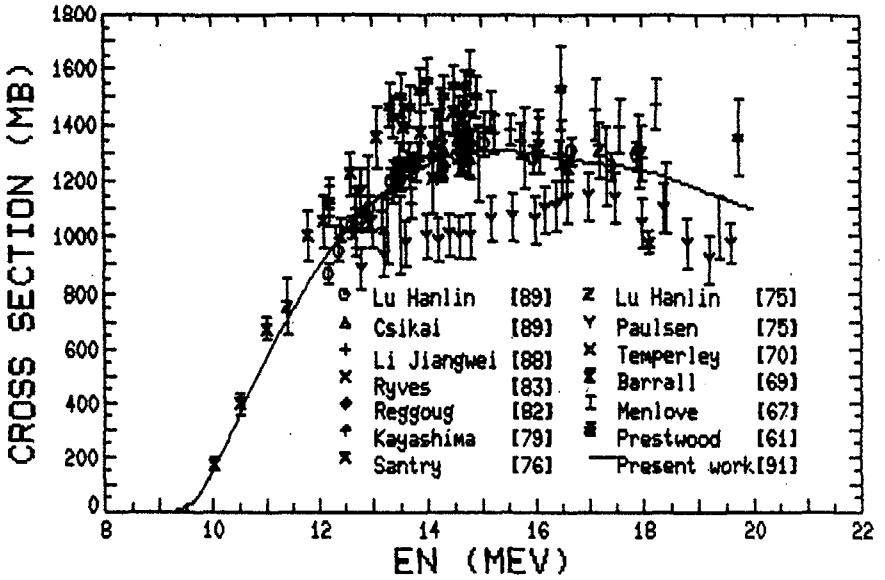

Fig.1 Experimental and evaluated cross-section for In-115(n,2n)In-114m

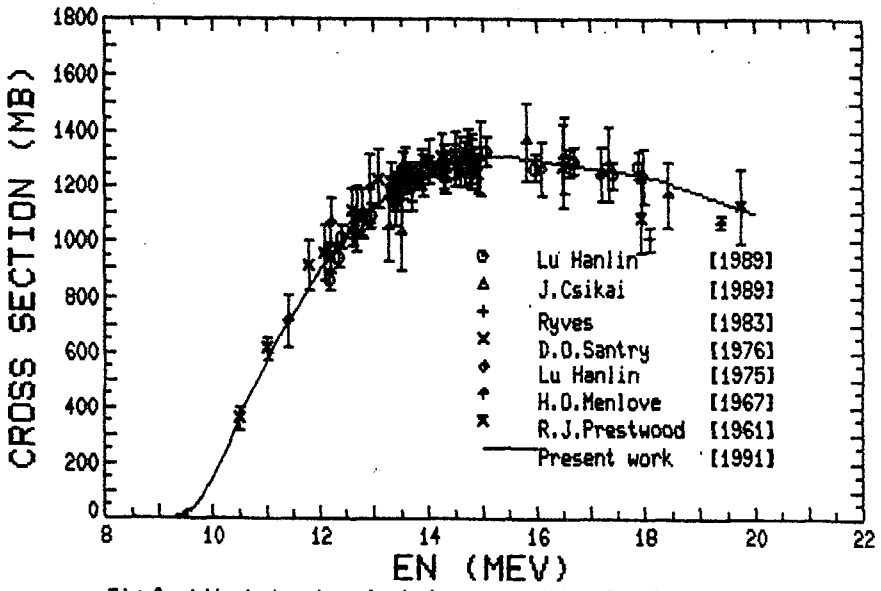

Fig.2 Adjusted and evaluated cross-section for In-115(n,2n)In-114m

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The Evaluation of $^{127}I(n,2n)^{126}I$ Reaction

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Iodine is monoisotope so only one neutron induced reaction leads to the production of ^{126}I. Its half life is 13.02 days [Ref.13]. ^{126}I decays by β^-emisson to the levels of ^{126}Xe and by $\beta^+$and orbital electron capture to the levels of ^{126}Te [Ref.14]. There are two stronger characteristic γ rays emitted: 389 keV (34.1%) and 666 keV (33.1%) [Ref.13], which have been used in measuring the cross sections for the reaction.

Eleven relevant references were compiled except our measurements. They are listed in Table 1 and plotted in Fig.1. The years, authors, measured energy points, cross sections and the methods to determine the neutron flux are given in columns two, three, four and five in Table 1 respectively. The Table also illustrates the branching ratio and half life used by authors for the residual product, and the detectors for activity measurements in the sixth, seventh and eighth columns. In order to get optimum results, the collected data were adjusted for changes in standard cross section, branching ratios of gamma rays,and half life according to the values taken from the references [Ref.13,14,15,16]. The adjusted data around 14 MeV were revised to 14.7 MeV based on the shape of the excitation function given by the reference 16 so as to obtain the evaluated cross section at this energy. Table 2 tabulates the adjusted factors for standard cross section, neutron energy, branching ratio and half life as well as the adjusted cross sections. However only part of the adjusted factors can be obtained because some authors didn't describe the necessary information in their papers. Five data [Ref.1,2,3,4,8] were rejected for the evaluation since they were much over the averaged value of the collected data. The evaluation was performed using the rest of the compiled data in Table 2. The weighting factors are equal to the reciprocal of squared errors. Usually the original errors were kept to get the weighting factor. However, if the collected data were adjusted by large factors, the addition errors were added to them. The evaluated cross section is 1657 ± 41 mb at 14.7 MeV. Because the data of Martin and Bormann were too low, the excitation function was evaluated only using the measurements of Santry and Lu by means of the program of spline function fitting for much–set data. The results from threshold to 20 MeV are summarized in Table 3 and shown in Fig.1.

For comparison, the Fig.1 also shows the data of ENDF / B–V. The present evaluated curve differs from the shape of ENDF / B–V especially in the energy range above 17 MeV.

The error correlations were determined for the data in each set and the correlation matrix of the evaluated cross section for the reaction is given in Table 4.

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Table 1 Collected Data and Relevant Information

No	Year	Author	En(MeV)	σ(mb)	$\Delta\sigma$(mb)	n flux	fr%	$T_{1/2}$	detector
1	1953	Paul	14.5	1120	392	BF_3		13 d	β
2	1953	Martin	12.37	895	54	α Asso.		13.1d	NaI
			13.08	1110	67				
			14.1	1310	79				
			15.6	1320	72				
			16.75	1220	73				
			18.25	945	57				
3	1962	Bormann	13.16	1150		Al(n,α)			
			14.1	1300					
			15.21	1280					
			16.8	1100					
			18.7	990					
4	1968	Qaim	14.7	1128	120	Al(n,α)			
5	1969	Barrall	14.8	1660	140	p recoil	34		NaI
6	1970	W.D.Lu	14.4	1649	80	56Fe(n,P)	34	12.8d	Ge(Li)
7	1971	Havlik	14.7	1640	145	Al(n,α)			NaI
8	1972	Maslov	14.2	1950	200	65Cu(n,p)			NaI
9	1973	Araminowicz	14.6	1487	160	63Cu(n,P)			
10	1973	Santry	9.25	11.6	2.3	^{32}S(n,P)	32	12.88d	Ge(Li)
			9.8	155.	1.6				
			10.25	422	35				
			10.75	754	60				
			12.5	1569	94				
			13.58	1690	101				
			13.8	1720	103				
			14.24	1780	107				
			14.5	1782	107				
			14.76	1766	106				
			16.2	1731	104				
			17.8	1681	100				
			19.6	1424	85				

(continue)

11	1979	Lu Hanlin	11.4	894	116	Al(n,α)	34.1	13d	NaI
			12.23	1407	79				
			12.79	1508	87				
			13.68	1594	60				
			14.36	1619	67				
			14.58	1637	68				
			14.77	1637	68				
			16.05	1707	94				
			17.18	1665	93				
			17.97	1624	94				
12	1985	Pepelnik	14.7	1655	173	^{26}Mg(n,p)			

Table 2 Adjusted Data at 14.7 MeV

No	Author	adjusted factor				adjusted cross section (mb)
		standard	Energy	f_r	$T_{1/2}$	
1	Paul		0.996			1124 ± 392
2	Martin		0.985		1.008	1319 ± 79
3	Bormann		0.985			1319
4	Qaim	1.022	1.0			1104 ± 120
5	Barrall		1.001	0.997		1653 ± 140
6	W.D.Lu	0.894	0.994	0.997	0.977	1893 ± 150
7	Havlik	0.978	1.0			1676 ± 145
8	Moslov	1.01	0.989			1952 ± 200
9	Araminowicz	1.028	0.998		0.919	1577 ± 177
10	Santry	1.013	1.001	0.938	0.991	1648 ± 106
11	Lu Hanlin	1.0	0.998	1.0	1.0	1635 ± 68
12	Pepelnik		1.0			1655 ± 173
Evaluated Cross Section 1657 ± 41 mb						

Table 3 Recommended Cross Section

En(MeV)	σ (mb)	Δσ(mb)	En(MeV)	σ (mb)	Δσ(mb)
9.0	0	0	14.54	1652	42
9.2	2	0.4	14.70	1657	41
9.4	4.4	0.4	14.94	1664	42
9.59	10	2	15.14	1669	43
9.79	80	3	15.34	1674	43
9.99	174	17	15.53	1678	43
10.19	284	23	15.73	1681	44
10.39	407	32	15.93	1684	44
10.58	538	37	16.13	1685	44
10.78	672	42	16.33	1685	45
10.98	806	48	16.52	1684	47
11.18	935	53	16.72	1681	49
11.38	1056	55	16.92	1676	50
11.57	1163	56	17.12	1669	51
11.77	1255	56	17.32	1660	57
11.97	1331	53	17.51	1648	55
12.17	1394	48	17.71	1634	62
12.37	1446	46	17.91	1617	62
12.56	1487	44	18.11	1597	64
12.76	1520	44	18.31	1574	64
12.96	1547	45	18.50	1548	65
13.16	1568	46	18.70	1518	65
13.36	1587	45	18.90	1485	65
13.55	1602	42	19.10	1447	67
13.75	1616	41	19.30	1406	71
13.95	1627	42	19.49	1360	81
14.15	1637	42	19.69	1310	80
14.35	1645	42	19.89	1255	116

Table 4. Correlation Matrix for the Evaluated Cross-section of $^{127}I(n,2n)^{126}I$ Reaction

1 = 9.2–10.19MeV
2 = 10.19–10.58MeV
3 = 10.58–11.18MeV
4 = 11.18–11.57MeV
5 = 11.57–12.37MeV
6 = 12.37–13.55MeV
7 = 13.55–14.15MeV
8 = 14.15–14.54MeV
9 = 14.54–15.14MeV
10 = 15.14–17.12MeV
11 = 17.12–18.11MeV
12 = 18.11–19.89MeV

	1	2	3	4	5	6	7	8	9	10	11	12
1	100											
2	41	100										
3	41	34	100									
4	36	30	30	100								
5	30	25	25	39	100							
6	39	33	33	56	66	100						
7	44	37	37	28	43	51	100					
8	32	27	27	32	37	42	78	100				
9	46	39	39	26	46	55	76	82	100			
10	18	15	15	13	40	39	62	48	74	100		
11	27	23	23	59	47	39	55	49	62	82	100	
12	21	18	18	49	32	27	42	31	41	45	65	100

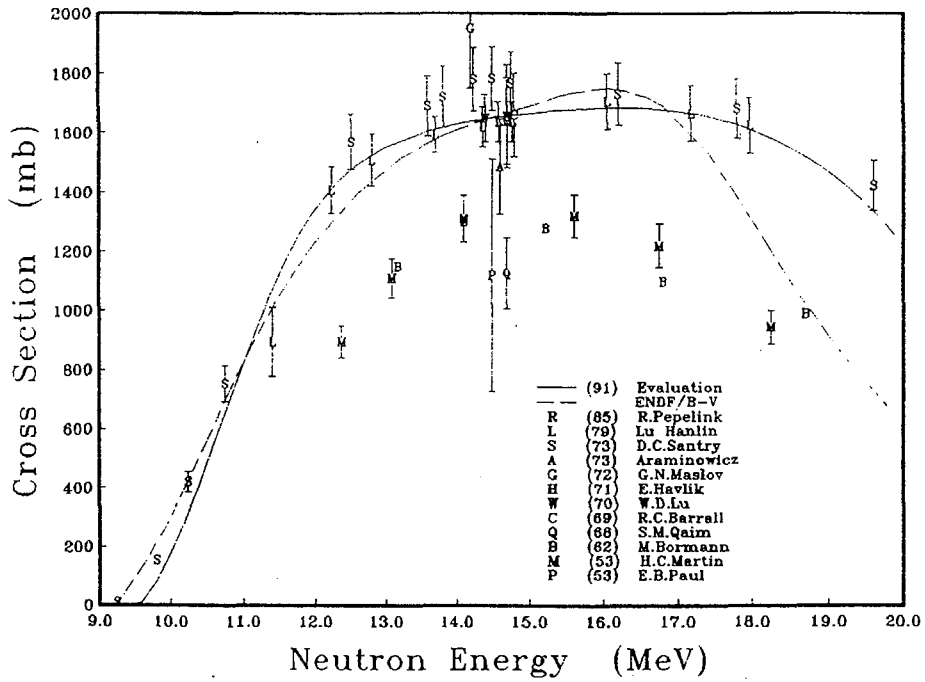

Fig.1 127-I(n,2n)I-126 Reaction

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