No. 6

Supplement (1992)

CNIC-00690

CNDC-0010

INDC(CPR)-028/L

COMMUNICATION OF NUCLEAR DATA PROGRESS

China Nuclear Information Center

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Atomic Energy Press

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Beijing, Oct., 1992

EDITORIAL NOTE

This is a supplement to No. 6 of *Communication of Nuclear Data Progress* (CNDP), in which the second and final part of CENDL–2 (Chinese Evaluated Nuclear Data Library, Version 2.0) papers is published. It includes the evaluation reports of 17 elements and isotopes, they are as follows :

^{16}O, ^{23}Na, ^{31}P, Ca, Zn, Mn, ^{59}Co, Cd, In, Hf, ^{232}Th, ^{235}U, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{249}Bk, ^{249}Cf.

We hope that our readers and colleagues will not spare their comments, in order to improve the publication.

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CONTENTS

THE EVALUATION OF OXYGEN COMPLETE NEUTRON DATA FOR CENDL–2

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INTRODUCTION

The oxygen complete neutron data are very important for fission reactor and shielding calculation, because it widely exists in the fuel and shielding materials. The data are also useful for fusion study. So its data were evaluated for CENDL–2.

The natural oxygen consists of three isotopes, ^{16}O, ^{17}O, ^{18}O, their abundances are 99.76%, 0.038%, 0.20% respectively[1]. Because the abundance of ^{16}O is very high, so usually the data for natural O are regarded as the same as ^{16}O.

The energy possible reactions for ^{16}O in the energy region below 20 MeV are given in table 1.

This is a summary report for the valuation. The evaluation of experimental data and covariance data are descriped in sections 2, 3 respectively, the theoretical calculatioin and comprehensive recommendation are given in sections 4, 5 respectively, and finally, the data are compared with other evaluations and discussed in section 6.

Table 1 The energy possible reactions, their Q values and threshold energies

	Reaction	Q–values (MeV)	Threshold (MeV)
1	n,total	0	0
2	n,n	0	0
3	n,n′	−6.0494	6.2692
4	n,2n	−15.664	16.652
5	n,n′α	−7.162	7.614
6	n,n′p	−12.127	12.892
7	n,γ	4.1440	0
8	n,p	−9.636	10.244
9	n,d	−9.9030	10.527
10	n,t	−14.479	15.392
11	n,^{5}He	−14.617	15.539
12	n,α	−2.2159	2.356

1 EXPERIMENTAL DATA EVALUATION

1.1 Total Cross Section

There are about 10 sets of measured data for total cross section, but only two latest ones[2,3] were taken, Cierjacks′ new measurement is particularly important, for its high resolution and precision (about 1%~2.5%). The resolution time (5.5 ps / m) was much improved, and correspondingly, the energies of resonances can be determined more acurately.

Since there are too many energy points (21059) in Cierjacks′ data for storing in library and application of users, the points were selected in the case of keeping peak and valley points. Then the data were fitted with spline function by using spline program[33].

For the purpose of adjusting the optical model parameters in theoretical calculation, the data were also averaged over a quite large energy interval, so that the structures disappear and the curve becomes some what smooth.

1.2 Elastic Cross Section

There are 36 sets of measured data for integral and differential elastic cross section. Among them, 11 sets of data[4~14] were taken for integral cross section, and 7 sets for differential one[7, 14~22].

The new data in the energy region 9.2~ 14.9 MeV for differential cross section measured by Glendenning et al. are very significant for filling the gap, in which the data were very scare before. Besides the measurement was performed by using tandem, with better time and energy resolution ($\Delta t = 0.3$ ns, $\Delta E / E = 2.5\sim 11\%$), smaller data error, many enough energy points (11) and angles (~ 20). With this important supplement, the experimental differential cross section covers the whole energy region from 0.1 to 15 MeV, and is good enough to take it as recommended one.

1.3 (n,p) Cross Section

There are 8 sets of experimental data[23~30] for (n,p) cross section, all of them were measured in earlier years. The data measured by Kantele[23], Schantl [27], Prasad[28], Matra[29] were renormalized by using new evaluated cross sections of ^{63}Cu(n,2n)[31], ^{27}Al(n,p)[32], ^{56}Fe(n,p)[32]. The data were fitted by using spline program with knot optimization[33]. The result as well as the comparison with ENDF / B−6, JENDL−3, BROND−2 is shown in Fig. 1.

1.4 (n,α) Cross Section

There are 7 sets of experimental data[34~40] for (n,α) cross section in the energy region from 3.6 to 15 MeV. The A. S. Divatia's data were got from the measurement of inverse reaction ^{13}C(α,n)^{16}O cross section. The data were fitted with spline program[33]. It should be pointed out that the all α particles were recorded in these measurements, so in fact, the measured cross section is a sum of (n,α) and (n,n′α) cross sections. As the threshold energy for (n,n′α) is 7.614 MeV, the (n,n′α) cross section becomes quite large when $E_n > 10$ MeV, so the measured data can be considered as (n,α) cross section only in the case of $E_n < 10$ MeV.

1.5 Nonelastic Cross Section and Others

There are only three sets of experimental data for nonelastic scattering cross section[41~43]. The data points are very few, the errors are quite large and the data are not very reliable. To supplement the data, the indirectly measured data were also calculated by substracting the measured elastic cross section [4~14] from the energy average total cross section[3]. All of these data are used as reference for theoretical calculation and data recommendation.

There are very few measured data for (n,2n)[45], (n,d)[46], (n,γ)[47, 48] cross sections. All of them were used as reference for parameter adjusting in theoretical calculation.

There are also some measured data for γ–production cross section, especialy for some fixed energy γ line production cross section[49~54]. All of them were used as reference in the theoretical calculation.

2 COVARIANCE DATA EVALUATION

The covariance data were evaluated for the reaction cross sections, the experimental data of which are more enough to be taken as recommendation.

2.1 Total Cross Section

As mentioned above, only one set of data[3] was adopted in the energy region from 3.13 to 20 MeV. The information about the error given in the paper is : statistical 0.3~ 1%; background correction 1~ 3%; dead time < 0.1%; multi–scattering negligible and sample error no information. The data were normalized to their early measured data[55, 56] at 3.5 MeV. It is clear that the main systematic error came from the normalization, which is depended on the error of their early work. But there is no exact information about the data error at 3.5 MeV, in the paper only saying that it is less than 3%. Comparing this data around 3.5 MeV with F. G. Perey's[57] and R. B. Schwartz's[58] data, it can be seen that the statistical error is 1.3% and the systematic difference between this data and Perey's data is 1.1%, which can be regarded as systematic error. So the total, used as normalization error, is 1.7%.

2.2 (n,p) Cross Section

According to the situation, as shown in Fig. 1, it is devided into three regions :

1) 16.5~20 MeV

In this region, there are three sets of data, and there are systematic differences between them. Drawing lines through each sets of data, the average distance between them (devided by 2) was taken as systematic error in this region.

2) 12.6~16.5 MeV

In this region, there are 8 sets of data, but only three of them were measured in whole region, the rest only around 14 MeV, and there is no systematic difference among them. Carefully analyzing the data shows that Seeman's[24] data not only have more energy points but also smaller error 4% (others are 8%, 10% respectively). So the curve is mainly determined by this set of data, and so the covariance. Analyzing the data shows that the total error is about 4.5% and the systematic component is 1.5% for detector efficiency and 1.0% for sample volume, so the total systematic error is 1.8%.

3) $E_{th} \sim 12.6$ MeV

There are three sets of data, but the curve mainly determined by DE. Jurent's data. The systematic components of the error are 5% for both neutron flux and neutron attenuation correction, which are considered as medium–range error.

2.3 (n,α) Cross Section

It is devided into three regions : 3 ~ 6.7 MeV, 6.7 ~ 11 MeV and 11 ~ 20 MeV.

1) 3~6.7 MeV

Only one set of data[39] is in this region, the systematic error components are 2% for current integration and 12% for detector efficiency, which are considered as medium–range error.

2) 6.7~11 MeV

There is mainly one set of data[36] in this region, its total error is about 10%, and the normalization error is less than 5%, 4% was taken as systematic error.

3) 11~20 MeV

In this region, as mentioned above, the (n,α) cross section could not be given by measured data, so the covariance data also could not be given through analyzing the experimental data.

3 THEORETICAL CALCULATION

The data, including neutron data and γ–production data, in the energy region from 5 to 20 MeV were calculated by using optical model, compound statistical and pre–equilibrium theory[57].

For purpose of adjusting parameters, the total cross section data[3] were averaged over large energy interval, the elastic differential cross section in the energy region 6~15 MeV were selected[6, 7, 14], and evaluated (n,p), (n,α) experimental data were used. Also some measured data for nonelastic, (n,2n), (n,d) cross sections were taken as references.

Due to oxygen is a quite light nuclide, so it's difficult to calculate the data by using statistical theory. The parameters were adjusted as much as possible, so that the result could fit experimental better. The comparison between calculated and measured data for total cross section is shown in Fig. 2.

The detail on theoretical calculation will be given in another report[58].

4 COMPREHENSIVE RECOMMENDATION

Taking an overall view of the experimental data, theoretical calculated data and others, the comprehensive recommendation is as follows :

4.1 The calculated data by using R–matrix[59] were taken for total, elastic and (n,α) cross sections below 6 MeV;

4.2 The evaluated experimental data were taken for total (6~20 MeV), (n,p) (6~ 20 MeV), (n,α) cross section (6~ 10 MeV) and elastic angular distribution (6~15 MeV).

4.3 The calculated data by using statistical theory[58] were taken for all other data, except elastic and nonelastic scattering cross sections.

4.4 To make the data consistent in physics, the nonelastic cross section was obtained by summing all cross sections of nonelastic channels, and the elastic scattering cross section was obtained by substracting nonelastic cross section from total cross section. The calculations were completed by using program CRECTJ5[60].

4.5 Also using CRECTJ5, isotropic angular distributions, which are required in the format but could not given out in the calculation, were supplemented. And also the data transformation matrix of elastic scattering was added by using program TRP[61].

4.6 To insure the correctness in format and physics, the data were checked by using ENDF utility program CHECKER, PSYCHE, FIZCON, and all problems found were corrected.

5 COMPARISON WITH OTHER NEW EVALUATED DATA FROM JENDL–3, ENDF / B–6 AND BROND–2

The recommended data were compared with other new evaluated data from JENDL–3, ENDF / –6 and BROND–2.

5.1 There is little difference between present data and other libraries for total and (n,p) cross sections (Figs. 3, 1), which were recommended based on experimental data and have higher precision.

5.2 Comparing with others, present elastic scattering cross section is highest one, and the nonelastic cross section is lowest one in the high energy region (Figs. 4, 5). This is caused by making consistance in the condition of keeping total cross section almost the same and taking into account the experimental data.

5.3 There is large difference between present data and others for inelastic scattering cross section in high energy region ($E_n > 12$ MeV) (Fig. 6), which were calculated theoretically. The reason is that a large (n,n′α) cross section ($E_{th} = 7.614$ MeV) is given in our case, but is not in ENDF / B–6 and BROND–2, and is very small in JENDL–3. Comparing with experimental data, in our case, the sum of (n,α) and (n,n′α) cross sections is comparable with corresponding measured data, but others are too small, so a quite large (n,n′α) cross section may be reasonable. Correspondingly, there is also a difference for inelastic γ–production cross section (Fig. 7). This problem is being studyed further.

5.4 For differential elastic scattering cross section, there are some differences between present data and others, but not too large, two examples are given in Figs. 8, 9.

5.5 In present evaluation, some covariance data are given for total and (n,p), (n,α) reaction cross sections (although they are not complete), no such data given in other libraries (at least for the time being). Also comparing with JENDL–3, more γ–production cross section and spectrum data are given in our evaluation, while no γ–production spectrum data given at all in ENDF / B–6 and BROND–2.

The authors would like to acknowledge Dr. Su Zongdi for his helpful comments and discussions on the theoretical calculation and Dr. Chen Zhenpeng for their calculation and his discussing on R matrix calculation. Also thanks goes to Dr. P. G. Young for his information and discussing on the evaluation.

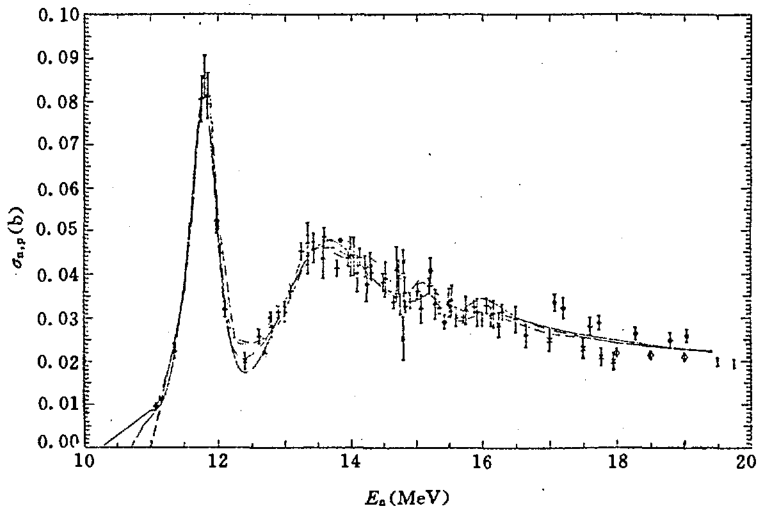

Fig. 1 Recommended (n,p) cross section

J. Kantele (USAARK), 1962, K. W. Seeman (USAKAP), 1962,
J. Martin (USALAS), 1954, DE. Jurent (USALAS), 1962,
M. Bormann (GERHAM), 1967, W. Schantl (AUSIRK), 1965,
R. Prasad (INDMUA), 1965, B. Matra (INDBOS), 1965,
——— Present data, ---- JENDL−3,
— · — ENDF / B−6, — · · — BROND−2

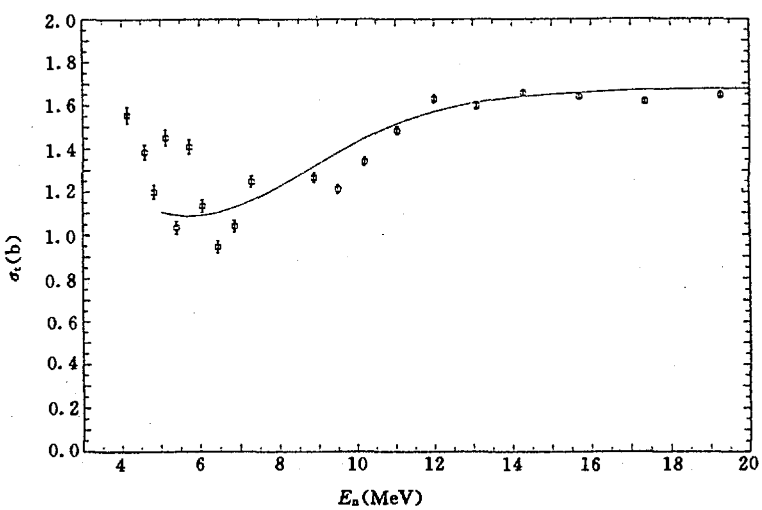

Fig. 2 Total cross section : comparison between experimental and theoretical data

S. Cierjacks (GERKFK), 1968, ——— Theoretical calculated data

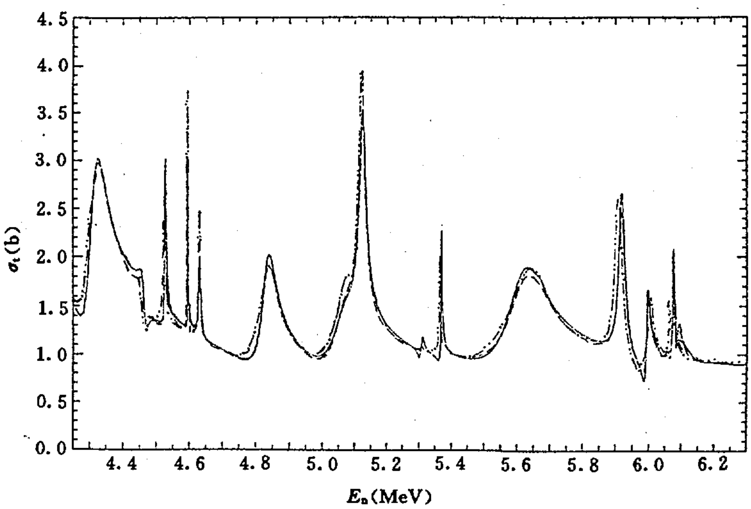

Fig. 3−1 Total cross section : comparison with others evaluated data

—— Present data, - - - JENDL−3, —·— ENDF / B−6, —· · — BROND−2

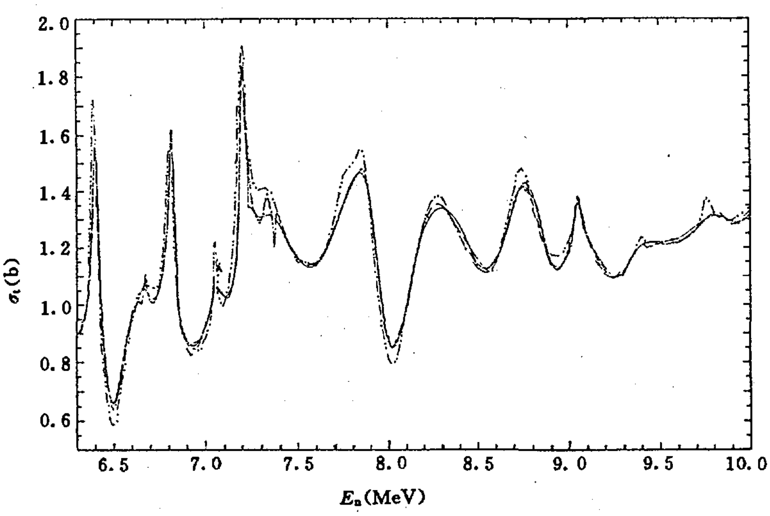

Fig. 3−2 Total cross section : comparison with others evaluated data

—— Present data, - - - - JENDL−3,

—·— ENDF / B−6, —· · — BROND−2

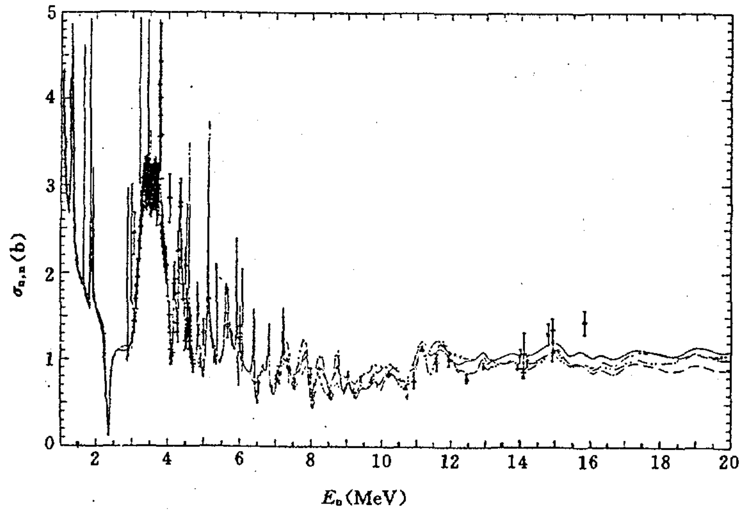

Fig. 4 Elastic scattering cross section

Φ W. E. Kinney (USAORL), 1972, ⚠ S. G. Glendinning (USATNL), 1982,

Ɨ D. Lister (USACOL), 1966, ⁑ R. R. Bauer (USALRL), 1963,

◇ C. F. Chase (OSALOK), 1961, ▲ N. A. Bostrorn (USATNL), 1972,

✝ M. Donald (CANOTU), 1966, ✱ P. L. Beach (USAOHO), 1967,

Ɨ B. Lundberg (SWDFOA), 1963, Others, D. Meier (SWTETH), 1969,

M. Baba (JPNTOH), 1985,

——— CENDL–2, - - - - JENDL–3,

— • — ENDF / B–6, — • • — BROND–2

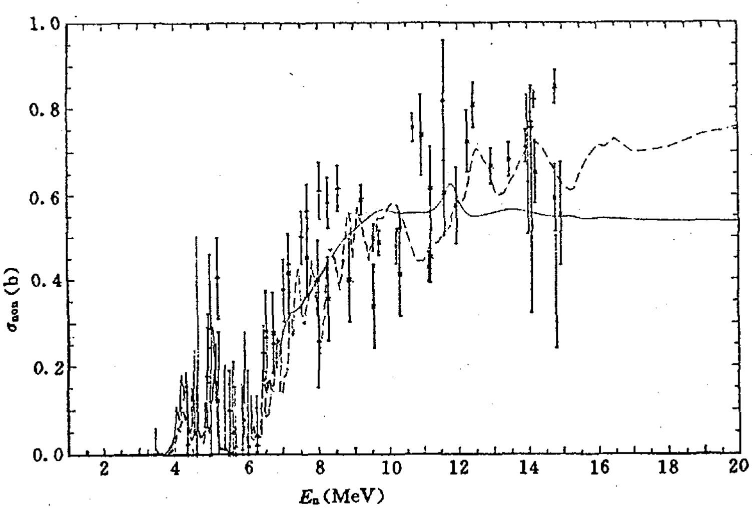

Fig. 5 Nonelastic cross section

Φ R. W. Bauer (USALRL), 1963, ⚠ L. F. Chase (USALOK), 1961,

Ɨ Brarati Pal (INDBOS), 1980, ⁑ A. Chatterjee (INDBOS), 1967,

Others : Total–elastic, ——— CENDL–2, - - - - JENDL–3

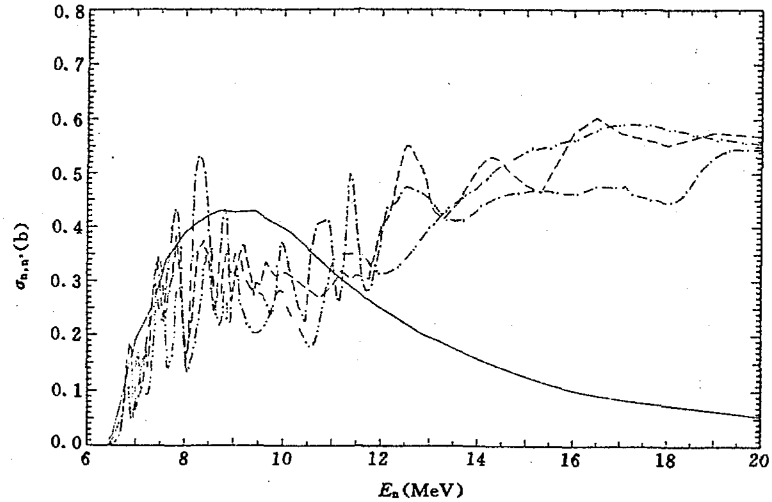

Fig. 6 Inelastic cross section

——— CENDL-2, ---- JENDL-3,

— • — ENDF / B-6, — • • — BROND-2

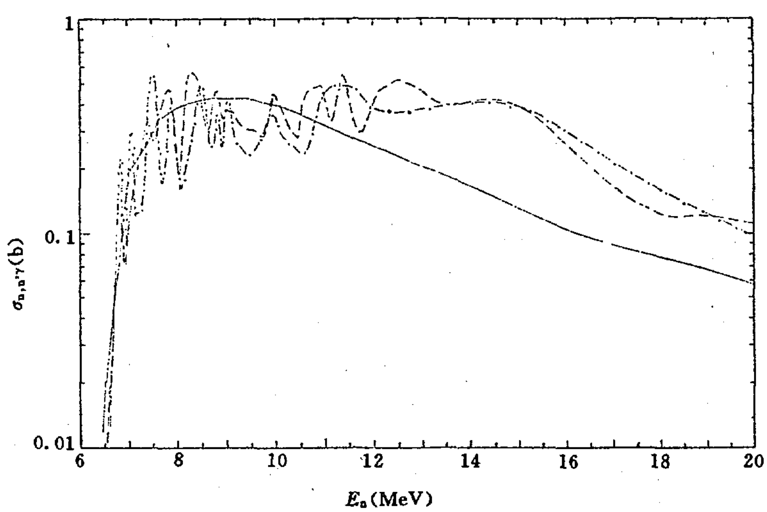

Fig. 7 Inelastic γ-production cross section

——— CENDL-2, ---- ENDF / B-6, — • — BROND-2

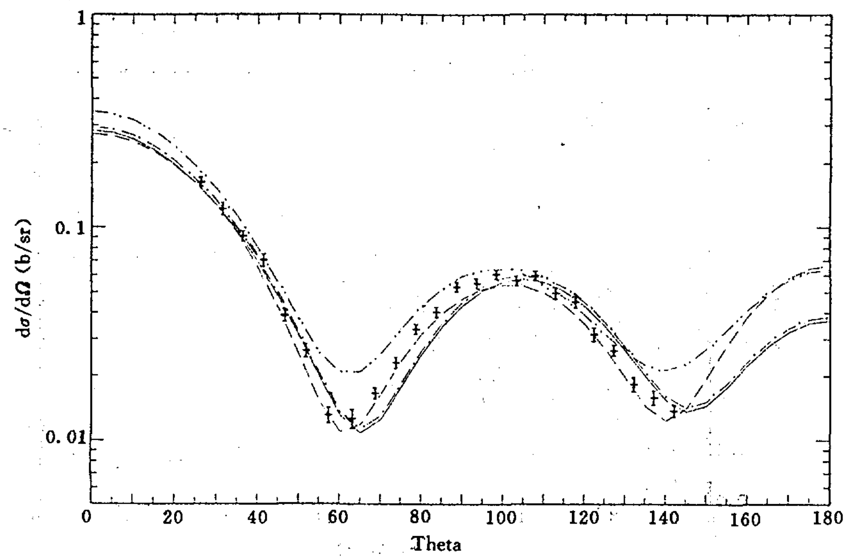

Fig. 8–1 Elastic scattering differential cross section at 9.211 MeV

Ŧ S. G. Glendinning (TNL), 1982, ——— CENDL–2,

- - - - JENDL–3, — • — ENDF / B–6, — • • — BROND–2

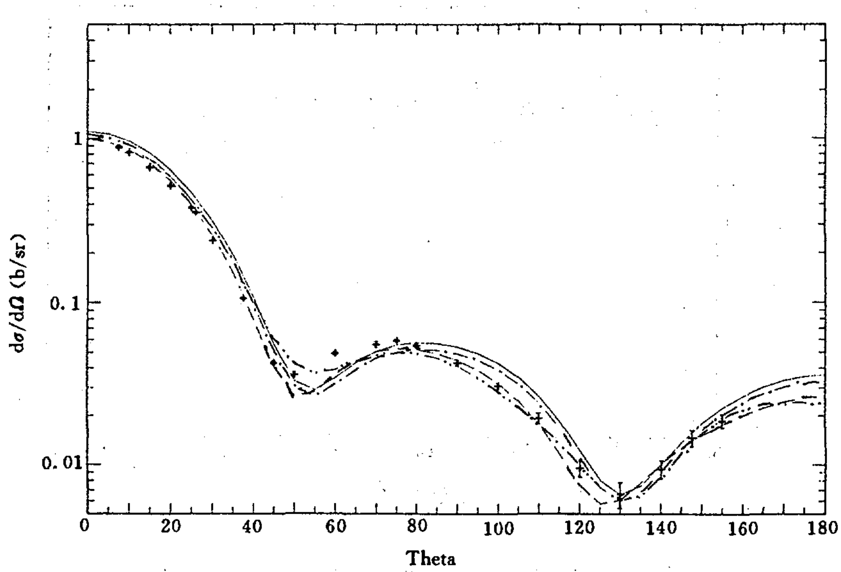

Fig. 8–2 Elastic scattering differential cross section at 14.14 MeV

Ŧ D. Meir (SWTETH), 1969, ——— CENDL–2,

- - - - JENDL–3, — • — ENDF / B–6, — • • — BROND–2

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EVALUATION OF NEUTRON NUCLEAR DATA OF SODIUM

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INTRODUCTION

The neutron cross section of Sodium are of considerable importance to the liquid metal fast breeder reactor design and safe operation. Moreover, Sodium is an important reference material, it can be combined with many other elements to form highly pure compounds. So measurement and evaluation of its cross sections have been still followed with interest to this day.

The evaluation for CENDL−1 was made by Xu Zhizheng in 1983, and the evaluation was revised for CENDL−2 by Wu Zhihua in 1991.

1 RESONANCE PARAMETERS

Resonancc parameters were taken from JENDL−3, in which the resolved parameters by MLBW formula were given in the energy region from 10^{-5} eV to 350 keV. Parameters were mainly taken from the recommended data of BNL Maghabghab[1] and the data for some levels were modified so that the calculated total cross sections for ^{23}Na could fit the experimental data.

2 TOTAL CROSS SECTION

There are four sets of data measured with white neutron source and TOF technique. From 300 keV to 20 MeV the evaluation is based on the data of Larson[2]. The other three are utilized for comparison purposes. See Fig. 1.

3 ELASTIC SCATTERING CROSS SECTION

This cross section results from substraction of the nonelastic cross section from the total. The nonelastic cross section is formed as the sum of total inelastic, capture, 2n, 3n and particle emission cross sections. See Fig. 2.

4 INELASTIC SCATTERING CROSS SECTION

Inelastic scattering to 0.4 MeV level (MT=51) is the largest contributor to this cross section. The high resolution data of Larson[3] from threshold to 2 MeV was used but renormalized down by 8 percent. The data of Donati[4], Diskens[5], Perey[6] and Freeman[7] were also used. The model theory calculation fitting these measurements were adopted in this evaluation. See Fig. 3.

5 CAPTURE CROSS SECTION

Most of the measurements were carried out for the thermal neutrons, 24 keV neutrons and 14 MeV neutrons. For other energy regions the data are quite sparse. After 1980, only three new experimental data are available, but there are some new evaluated data, such as ENDF / B–6, IRDF–85 and Yuan's evaluation[8]. The Yuan's is choosed for this file. Yuan's file from 10^{-5} eV to 0.9 MeV neutron energy region is the same as ENDF / B–6 or IRDF–85. From 0.9 MeV to 20 MeV his recommended values were re–evaluated by fitting eight experimental data. Before data fitting, new standards for ^{235}U fission cross section and ^{27}Al(n,α) cross section had been adopted to renormalize the experimental data. See Fig. 4.

6 (n,2n) CROSS SECTION

The existing measurements exhibit serious discrepancies specially at neutron energies above 15 MeV. These inconsistencies have resulted in a large difference between most recent evaluations ENDF / B–6, IRDF–85 and JENDL–3. Wagner[9] in 1990 reported the (n,2n) cross section of 114.5$\pm$ 6.6 mb at 19.45 MeV. Lu Hanlin[10] also reported a set of experimental data and gave a set of fitted recommended values which are much close to the data of Wagner[9] and Ikeda[11]. Lu's recommended data were adopted in this file. See Fig. 5. .

7 (n,p), (n,α) CROSS SECTIONS

From threshold to 12 MeV, the data of Williamson[12] and Bass[13] were used. Above 12 MeV, the data of model calculation were used. See Fig. 6 and Fig. 7.

8 AUGULAR DISTRIBUTIONS

For MT = 2, the data of various authors Fasoli[14], Fasoli[15], Kuijper[16], Perey[6] and Towle[17] were used in adjusting the optical parameters. All the data recommended in this file are calculated by the above mentioned parameters.

9 DISCUSSION

The comparision of CENDL–2, ENDF / B–6 and JENDL–3 are shown in figures. For total cross section and scattering cross section, three files are similar because they are based mainly on same measurements.

For capture cross section, CENDL–2 and ENDF / B–6 are based on same measured data, but JENDL–3 is calculated by CASTHY code. See Fig. 4.

For (n,2n) cross section, there are three different trends, CENDL–2 is in the middle, and it is closer to recent measured values. See Fig. 5.

From CINDA 91, we've learned that some new measurements for resonance parameters, and for (n,p) and (n,α) cross sections are under edition to EXFOR. They should be investigated in more detail in our futher studies.

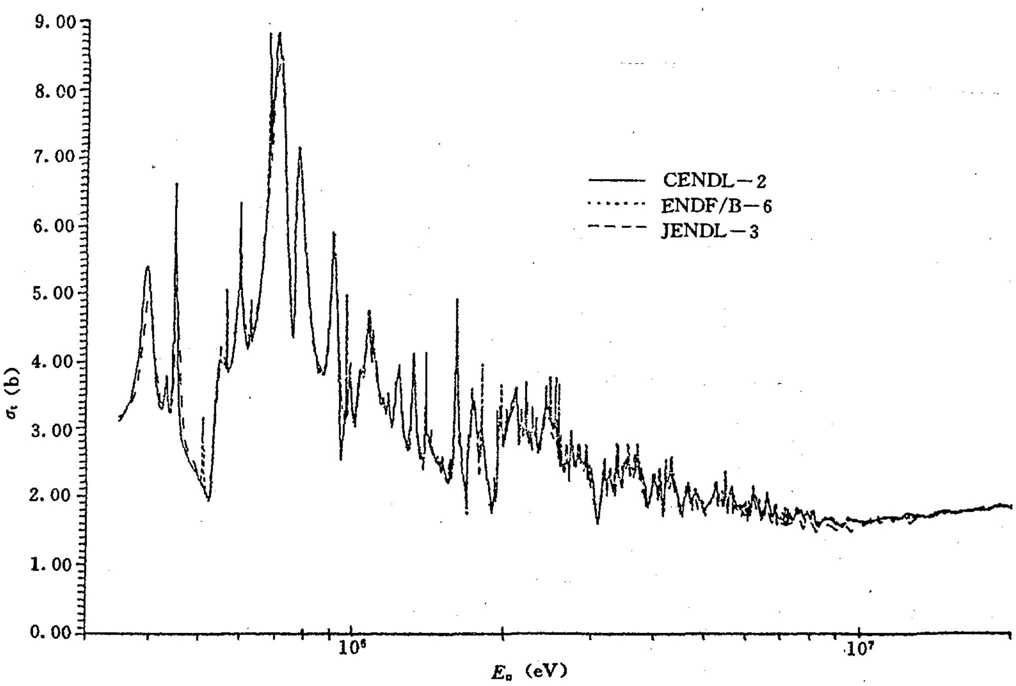

Fig. 1 The total cross section of Na

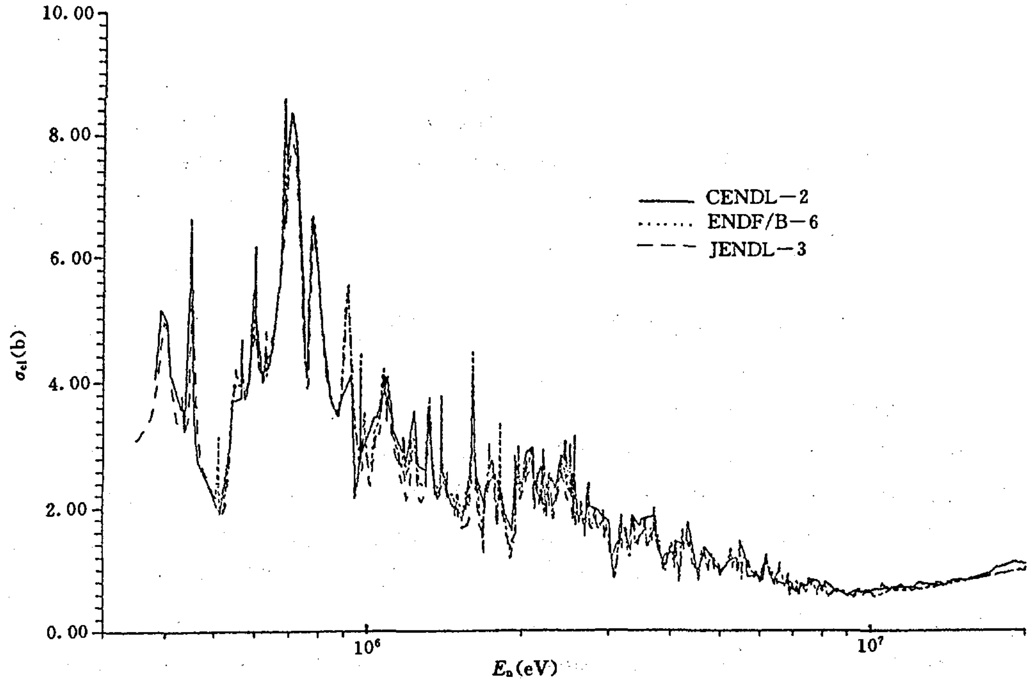

Fig. 2 The elastic scattering cross section of Na

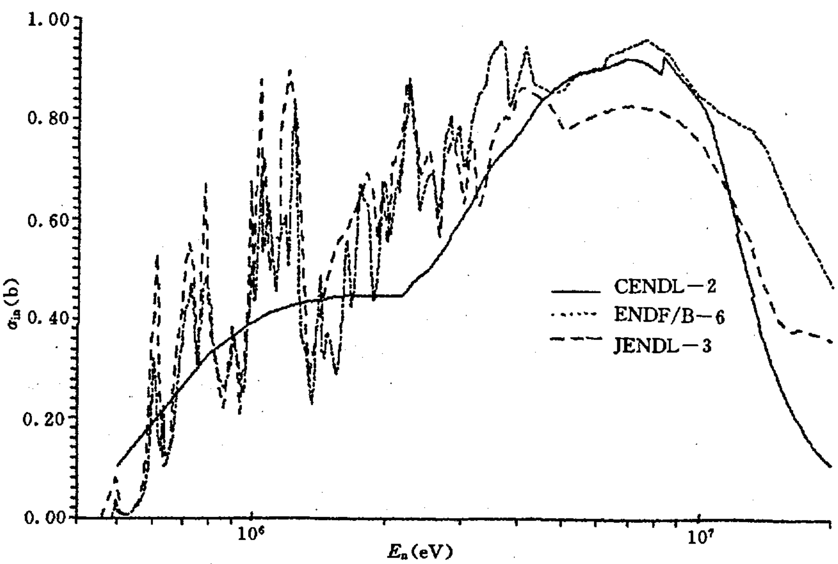

Fig. 3 The inelastic scattering cross section of Na

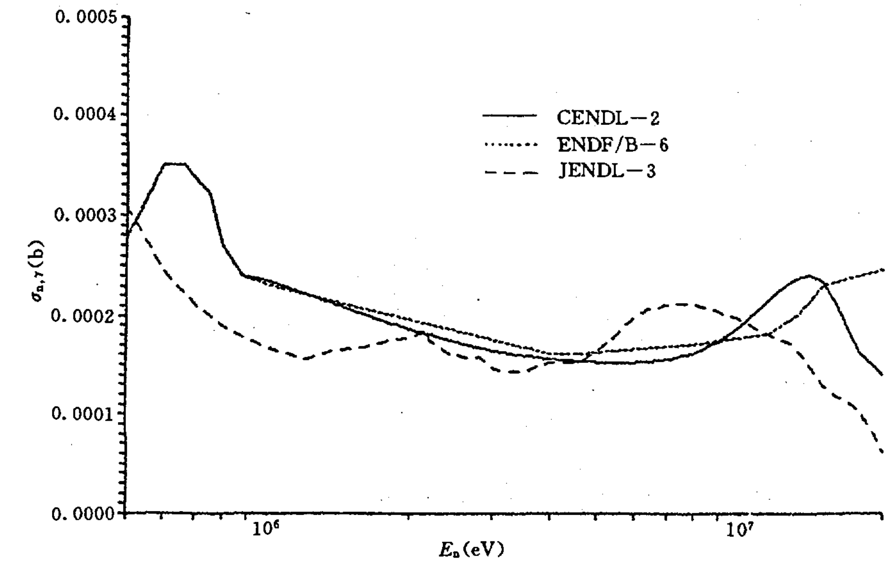

Fig. 4 The capture cross section of Na

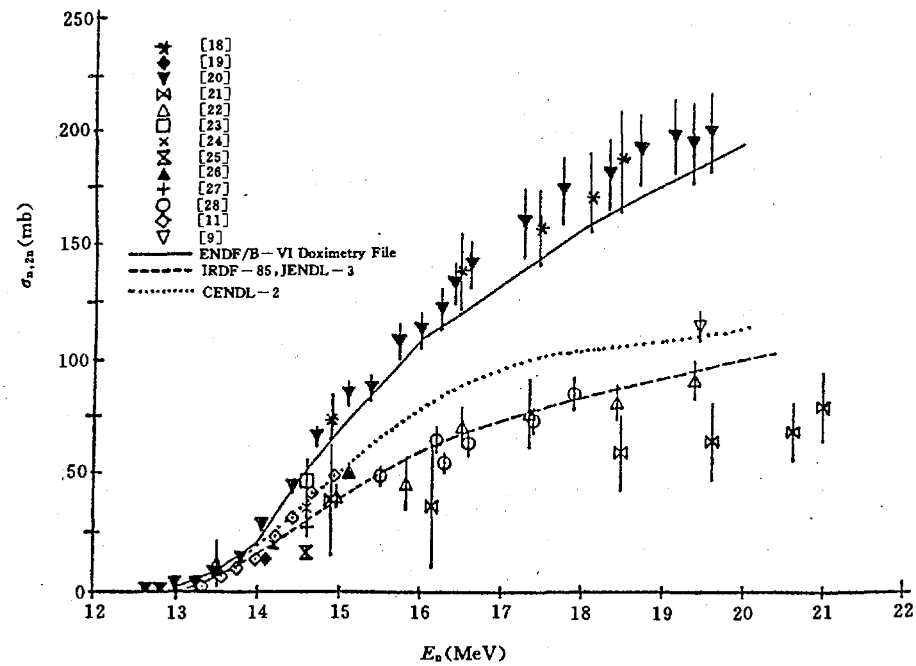

Fig. 5 $^{23}Na(n,2n)^{22}Na$ cross section

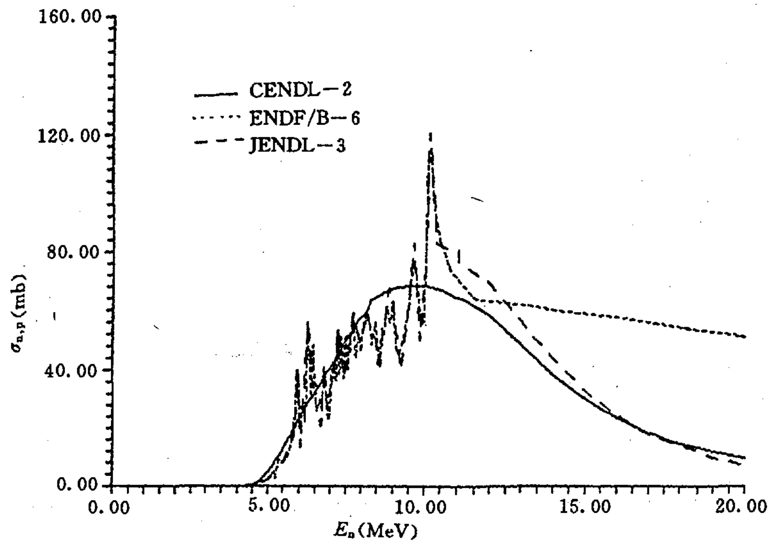

Fig. 6 The (n,p) cross section of Na

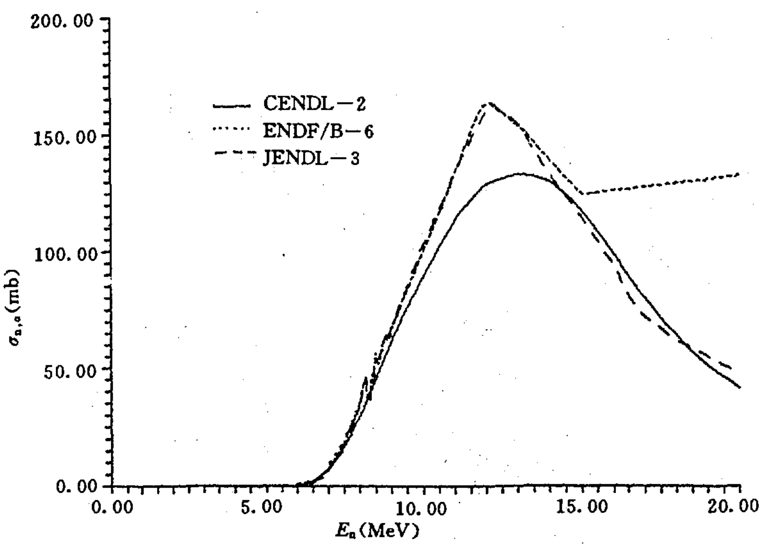

Fig. 7 The (n,α) cross section of Na

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THE EVALUATION OF ^{31}P

COMPLETE DATA FOR CENDL-2

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A complete evaluation is given for neutron data of phosphor in the incident neutron energy region from 10^{-5} eV to 20 MeV (MAT = 2150).

All experimental data available up to 1985 were almost collected and adopted as the base of the evaluation. The results of nuclear model calculation was used for complement. All theorectical calculations were completed by using code MUP2[1].

File 1 General information

MT = 451 Description of the evaluation and related references.

File 2 Resonance parameters

MT = 151 Only the effective scattering radius was given. The data was taken from ENDF-1978.

File 3 Neutron cross section

The cross sections at 2200M / S are :

Total	3.3040 b
Elastic	3.1390 b
Capture	0.1741 b

MT = 1 Total cross section

The data were taken from Ref. [2] from 10^{-5} eV to 450 keV; based on Ref. [3] at 0.0253 eV; on Refs. [4~7], [4, 8, 9] from 450 keV to 15 MeV; taken from theoretically calculated data from 15 to 20 MeV.

MT = 2 Elastic scattering cross section

This was obtained by subtracting the nonelastic cross section from the total cross section.

MT = 3 Nonelastic scattering cross section

This was obtained by summing the all nonelastic cross sections, including (n,2n), (n,x), (n,n′x), (n,xx) and (n,γ) cross sectins, which are in agreement with the experimental data[18].

MT = 4 Total inelastic scattering cross section

This was obtained by summing 40 discrete and a continuous excited state inelastic scattering cross sections.

MT = 16 (n,2n) reaction cross section

Based on the available data[10~17, 19] and so on. Generally these data are consistent with each other. The theoretic calculation results agreed with the experimental data. Here is only (n,2n) cross section, the (n,2nx) cross section was included in the total inelastic scattering cross section.

MT = 22, 23 (n,n′α), (n,n′t) reaction cross section

The theoretic calculation results were adopted.

MT = 28 (n,n′p) reaction cross section

The calculated results were normalized to the data[20].

MT = 51~90 Inelastic scattering to 40 excited state cross sections

The theoretic calculation results were adopted. The level parameters used are:

No.	Energy (MeV)	Spin-parity
G. S.	0.0	0.5^+
1	1.2662	1.5^+
2	2.2337	2.5^+
3	3.1341	0.5^+
4	3.2950	2.5^+
5	3.4146	3.5^+
6	3.5058	1.5^+
7	4.1903	2.5^+
8	4.2607	1.5^+
9	4.4309	3.5^-
10	4.5936	1.5^+
11	4.6338	3.5^+
12	4.7831	2.5^+
13	5.0149	1.5^+

14	5.0152	0.5^+
15	5.1154	2.5^+
16	5.2661	0.5^+
17	5.4431	4.5^+
18	5.5293	3.5^+
19	5.5692	1.5^+
20	5.6723	2.5^+
21	5.7731	2.5^+
22	5.8923	4.5^+
23	5.9879	1.5^-
24	6.0478	3.5^+
25	6.0801	4.5^+
26	6.2331	1.5^+
27	6.3366	0.5^+
28	6.3808	1.5^+
29	6.3986	2.5^-
30	6.4537	5.5^+
31	6.4608	1.5^+
32	6.4958	0.5^-
33	6.5006	4.5^+
34	6.5942	2.5^-
35	6.6103	1.5^-
36	6.7929	4.5^-
37	6.8251	5.5^-
38	6.8423	2.5^-
39	6.9092	1.5^-
40	6.9317	2.5^+

MT = 91 Inelastic scattering cross section to the continuous states, it was taken from the theoretic calculation results with modification.

MT = 102 Capture cross section

From 10^{-5} eV to 450 keV it was taken from Ref. [2], from 450 keV to 20 MeV the theoretic calculation results were adopted. At 0.0253 eV it was based on Ref. [3] data.

MT = 103 (n,p) reaction cross section

It was based on the available experimental data[21~25], except that from 9 to 14 MeV and from 15 to 20 MeV it was taken from the theoretic calculation data.

MT = 104, 106 (n,d), (n,^{3}He) reaction cross sections

The theoretic calculation results were adopted.

MT = 105 (n,t) reaction cross section

The theoretic calculation result was adopted.

MT = 107 (n,α) reaction cross section

It was based on the available experimental data[10, 26~30], except that from 9 to 13 MeV it was taken from the theoretic calculation results.

MT = 112 (n,pα) reaction cross section

The theoretic calculation results were adopted.

MT = 251, 252, 253 The theoretic calculation results were adopted.

File 4 Angular distributions of secondary neutrons

MT = 2 Elastic scattering

Taken from theroretic calculation data, which essentialy agreed with available experimental data, all distributions were represented by Legendre coefficients in the c. m. system. The transformation matrix from c. m. system to l. m. system is given.

MT = 16, 22, 28, 32 and 91 continuous secondary neutron

Assumed to be isotropic

MT = 51~90 Inelastic scattering to 40 excited satates

Theoretical results were taken.

File 5 Energy distributions of secondary neutrons

MT = 16, 22, 28, 32 and 91 Taken from the theoretic calculation data, represented by tabulated normalization possibility.

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EVALUATION OF NEUTRON NUCLEAR DATA OF NATURAL CALCIUM FOR CENDL–2

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ABSTRACT

Neutron nuclear data of natural calcium for CENDL–2 has been evaluated in the energy region from 10^{-5} eV to 20 MeV. Evaluated quantities are the total, nonelastic scattering, elastic and inelastic scattering, radiation capture, (n,p) (n,t), (n,2n), (n,α) reaction cross sections and the angular distributions of elastic and inelastic cross sections. Some of the data were calculated with the program AUJP[1] based on optical model and the program MUP2[2] based on

Hauser–Feshbach model and pre–equilibrium evaporation model.

INTRODUCTION

Natural calcium is an indispensable component of structure material of reactor. Therefore the neutron nuclear data of natural calcium are important for improvement of the reactor design. Except the total cross section, there are not so many experiment data of natural calcium. A few data sets on the nonelastic scattering, elastic and inelastic cross section and the capture cross sections were measured. There are no or almost no experimental data on channels of (n,t) (n,2n), (n,p), (n,α), (n,np), (n,nα) reactions for natural calcium. All the missing data (including the data of energy angular distribution of secondary neutrons) were calculated using program AUJP and MUP2, which was based on optical, Hauser–Feshbach and pre–equilibrium evaporation models. All the data of natural calcium were recommended in ENDF / B–5 format.

1 THE TOTAL CROSS SECTION

There are a lot of data on the total cross section since 1968. Most of them were measured with white light neutron source and TOF method. The original measured data below 10 MeV in a certain laboratory in different energy ranges were adopted.

In the energy range of 10^{-5} eV to 1.6 keV, we calculated the value using the resonance parameters and potential scattering radius given by S. F. Mughabghab et al.[3] and multiple–level B–W formula and 1 / ν law. The result has been compared with the value measured by A. Abdel et al.[4]

In the energy range of 1.6 keV to 0.55 MeV, the recent measured data[5] were recommended. The uncertainty is about 2.0 %.

In the energy range of 0.56 to 14 MeV, we adopted the data measured by R. B. Schwartz[6] in 1974. The uncertainty is about 1.5 % to 2.0 %.

In the energy range of 14.0~20.0 MeV, the data measured by Lorsout et al.[7] were recommended. See Fig. 1.

2 NONELASTIC SCATTERING CROSS SECTION

Our recommended data were given by summing of all the reaction cross sections except the elastic scattering. The experimental data[8~12] were used for adjusting the parameters in the calculation. See Fig. 2.

3 THE CROSS SECTIONS OF THE ELASTIC AND INELASTIC SCATTERING

All the elastic scattering cross sections were given by the difference between the total and the nonelastic scattering cross sections. For the inelastic scattering cross section we only found the measured data[13~15] of ^{40}Ca for the levels of 3.35, 3.75 MeV and 3.90 MeV. Therefore we use the calculated data as recommendation value and compared them with ^{40}Ca experimental data. See Fig. 3.

4 CAPTURE CROSS SECTION

Below 1.5 MeV the excitation curve of (n,γ) reaction for natural calcium has structures. We only found the experimental data at thermal and 14 MeV [16~18]. Therefore in the energy range of 10^{-5} eV to 1.2 MeV the cross section (0.43 ± 0.04) was calculated using the cross section at thermal point and the resonance parameters given by S. F. Mughabghab et al. and the $1/v$ law. In the energy region of 1.2 to 20 MeV the cross sections were calculated using the progaram MUP2. The experiment data at 14 MeV (0.53 ± 0.09 bar) were used to determine the parameters in the program. See Fig. 4.

5 THE (n,p) REACTION

Only cross section of calcium isotopes at 14 MeV could be found[19, 20]. From them we took the cross section (459 ± 21) mb as the natural calcium corresponding value at 14.1 MeV. All the calculated cross sections of (n,p) reaction were recommended. See Fig. 5.

6 THE (n,t) REACTION

Fitted cross sections of ^{40}Ca for the (n,t) reaction in the energy range of 14.9 to 19.6 MeV[21] was recommended as (n,t) cross section of natural calcium. The value was consistent with the systematics data given by Zhou Delin[22] and there was large difference comparing with the value calculated using program MUP2. See Fig. 6.

7 THE (n,2n) REACTION

Although the measured (n,2n) cross section of ^{40}Ca exists[23], the natural calcium corresponding data can not be replaced by the ^{40}Ca data because their thresholds are quite different. Therefore we recommended the theoretical data. See Fig. 7.

8 THE (n,α), (n,n$'\alpha$), (n,n$'$p) REACTIONS

For these reactions only the experimental cross sections of ^{40}Ca[24~26] could be found at a few energy points. So all the recommended data were taken from model calculations.

9 THE ANGULAR DISTRIBUTION OF THE ELASTIC AND INELASTIC SCATTERING CROSS SECTION AND SECONDARY NEUTRON ENERGY

For angular distribution of elastic scattering cross sections 13 reference literature have been collected and 8 of them are shown on Table 1. In the energy range below 4 MeV the data measured by A. B. Smith[27] and R. O. Lame[28] were recommended. Above the energy the theoretical calculation data were used. For the angular distribution of inelastic scattering cross section and secondary neutron energy, no experimental data have been found and the theoretical calculated data were used.

Table 1

Year	Author	E (MeV)	Uncertainty	Technical
1961	R. O. Lane[28]	0.03~1.81	10 %	4 Detectors
1982	A. B. Smith[27]	1.60~8.35	5 %	TOF
1969	B. Holmquvist[29]	6.09~8.05	5 %	TOF
1970	F. G. Perey[14]	5.5 ~8.52	5 %	TOF
1977	J. C. Ferrer[30]	11.0	3~ 5 %	TOF
1966	A. J. Frasca[31]	14.0	5~10 %	TOF
1977	J. Rapaport[32]	20.0	5~10 %	TOF

10 CONCLUSION AND DISCUSSION

Our recommended data sets compared with ENDF / B–6 and JENDL–3 are shown in Fig. 1 to 12. From these results one sees :

10.1 The evaluated cross sections of the elastic scattering and the non–elastic scattering are well consistent with the experimental value, especially at the 14.5 MeV point. The systematical adjusting for the data set have improved the results.

10.2 In the neutron energy below 4 MeV, the data of the elastic scattering angular distribution were adopted from the data measured by A. B. Smith et al.[27], which improved the recommended data.

10.3 For the recommended data of total cross section in the higher neutron energy, the new data measured by Lorsont et al.[7] were used.

The (n,t) reaction cross sections of natural calcium were replaced by ^{40}Ca corresponding data, which were well consistent with the systematically calculated values[22]. We thought that would be more reasonable.

ACKNOWLEDGMENTS

The authors would like to thank Dr. Zhou Delin for his systematic calculation program and his advice throughout this work. We also thank Dr. Liu Tingjin and Dr. Yu Baosheng for their useful help in translating the programs to us.

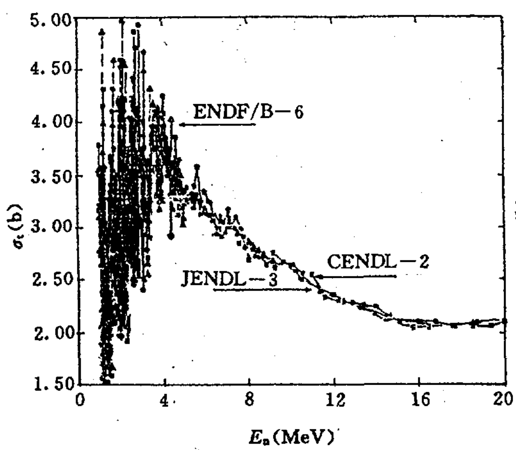

Fig. 1 The total cross section of natural Calcium

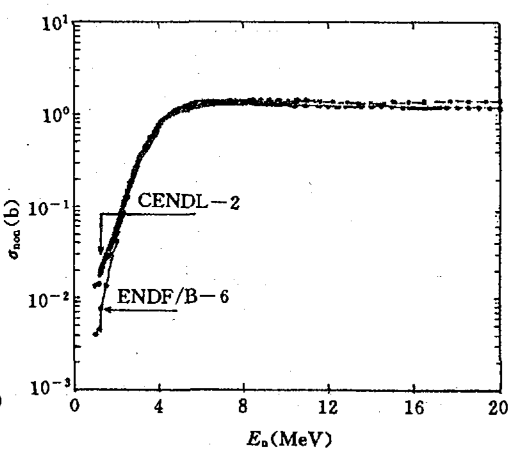

Fig. 2 The nonelastic scattering cross section

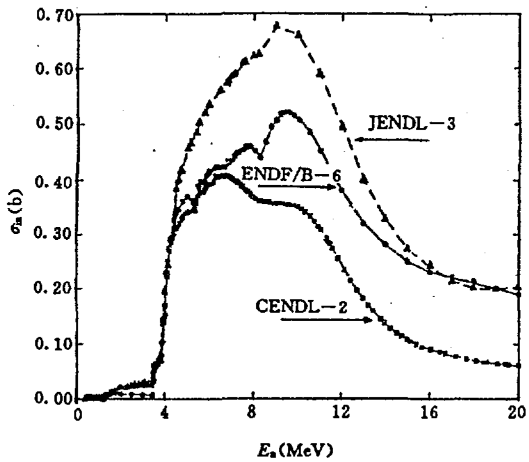

Fig. 3 The inelastic scattering cross section

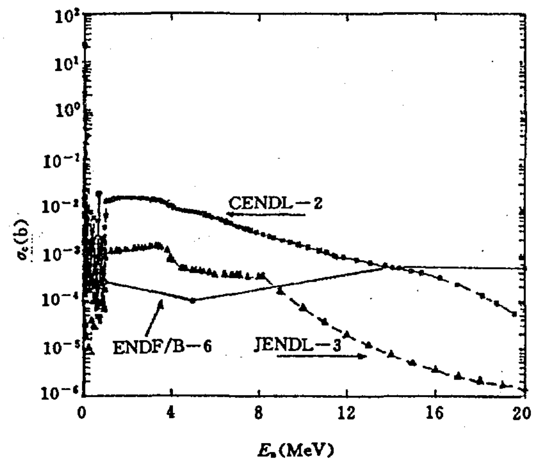

Fig. 4 The capture cross section

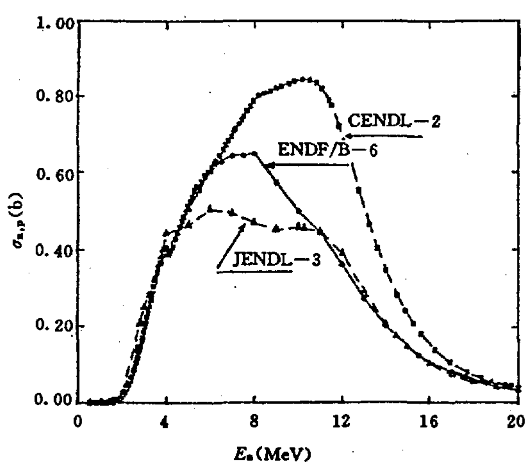

Fig. 5 The (n,p) reaction cross section

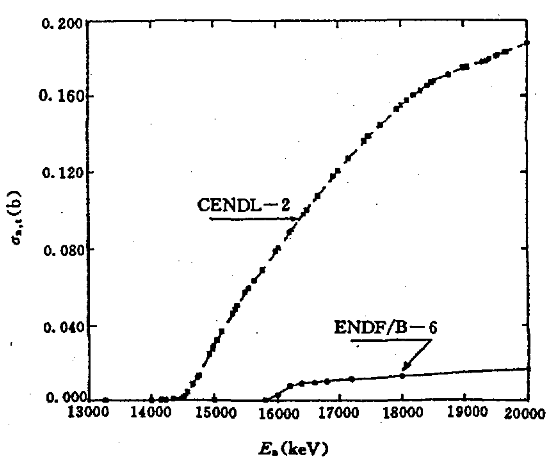

Fig. 6 The (n,t) reaction cross section

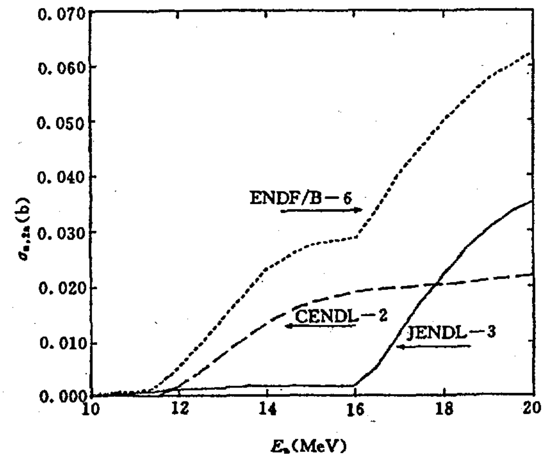

Fig. 7 The (n,2n) reaction cross section

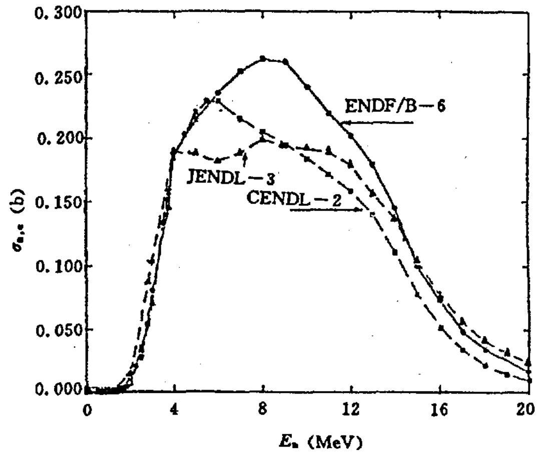

Fig. 8 The (n,α) reaction cross section

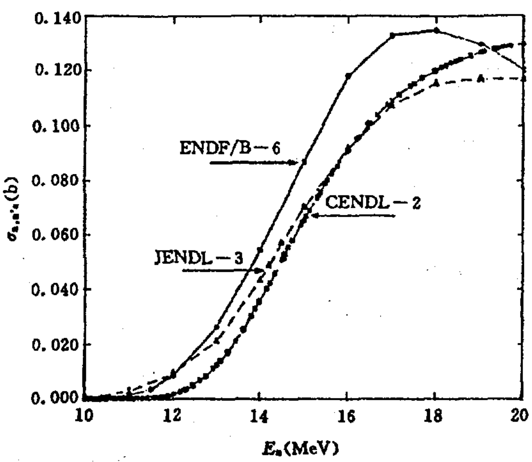

Fig. 9 The (n,n′α) reaction cross section

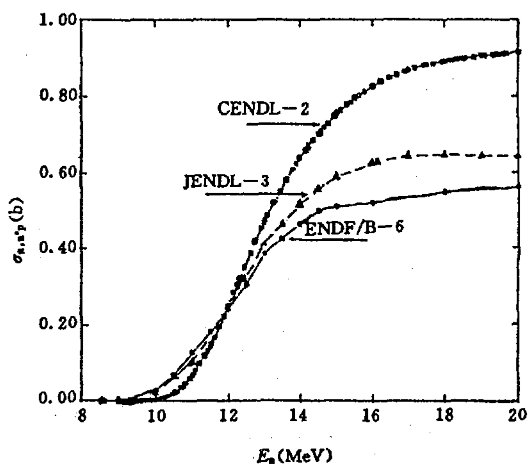

Fig. 10 The (n,n′p) reaction cross section

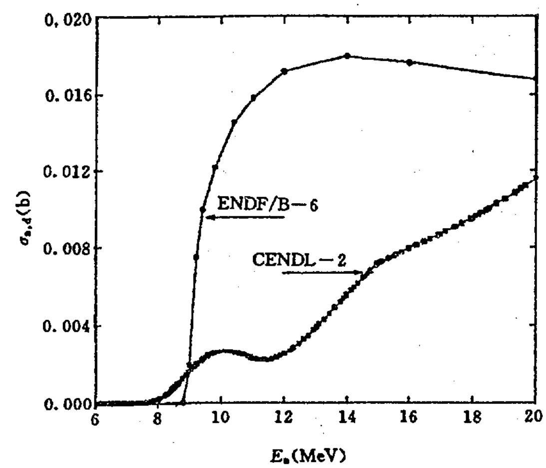

Fig. 11 The (n,d) reaction cross section

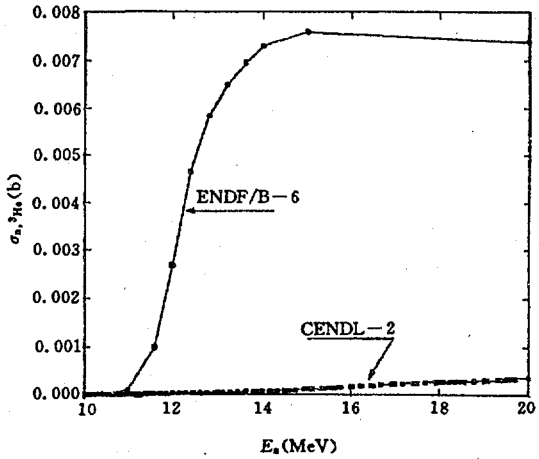

Fig. 12 The (n,^{3}He) reaction cross section

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THE EVALUATION OF Mn DATA FOR 10^{-5} eV TO 20 MeV NEUTRONS *

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1 THERMAL CROSS SECTIONS AND RESONANCE PARAMETERS

The resonance region covers 10^{-5} eV to 80 keV. However, the list of resonances extends to 100 keV. There is no unresolved region. The resonance parameters and the thermal (0.0253 eV) cross sections were taken from Ref. [1]. Their value are as follows :

Total cross section = 15.5 b
Scattering cross section = 2.2 b
Capture cross sesction = 13.3 b

2 FAST NEUTRON CROSS SECTIONS

2.1 Total Cross Section

The data in the energy range 80 ~ 257 keV was based on the experimental data of Ref. [2]; 257 ~ 500 keV on Ref. [3]; 0.5 ~ 20 MeV on Ref. [4]. Some points were discarded to reduce the data points without sacrificing important structural information.

2.2 Elastic Scattering Cross Section

The data were obtained by subtracting nonelastic from total cross section, which are in agreement with the experimental data of Refs. [6, 7].

2.3 Total Inelastic Cross Section

They were calculated by using nuclear model code MUP2[5], and are in agreement with the experimental data of Refs. [8 ~ 13].

2.4 (n,2n) Cross Section

From threshold to 12 MeV and 18 ~ 20 MeV they were calculated by using MUP2 code. From 12 to 18 MeV, the experimental data[14 ~ 21] were used and they were fitted with spline function code SPF[47].

2.5 (n,3n) Cross Section

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* This evaluation was completed in 1986.

It was calculated by using MUP2 code.

2.6 (n,n′α) and (n,n′p) Cross Sections

They were calculated by using MUP2 code.

2.7 Inelastic Scattering Cross Sections to Discrete Levels

These cross sections were calculated by using MUP2 and are in agreement with experimental data of Refs. [9, 10].

2.8 Inelastic Scattering Cross Section to Continuous State

This was calculated by using MUP2 code.

2.9 Radiative Capture Cross Section

From 80 keV to 20 MeV, the experimental data[22~31] were used and they were fitted with spline function.

2.10 (n,p) Cross Section

The data were calculated by using MUP2 code and are in agreement with the experimental data of Refs. [32, 33]

2.11 (n,d) Cross Section

The data were calculated with MUP2 code, and are in agreement with the experimental data at 14.1 MeV[34].

2.12 (n,t) Cross Section

The data were calculated by using MUP2 code.

2.13 (n,^{3}He) Cross Section

The data were calculated with the MUP2 code, and were normalized to experimental data of Refs. [35, 36].

2.14 (n,α) Cross Section

From threshold to 12 MeV the data were calculated by using MUP2 code. From 12 to 20 MeV they were obtained by fitting the experimental data of Refs. [37~41] with SPF code[47].

3 ANGULAR DISTRIBUTION OF SECONDARY NEUTRONS

3.1 Elastic Scattering Cross Section

Calculated data[5] were adopted, and the calculations are in very good agreement with the measured data of Refs. [42~46].

3.2 Angular Distributions of (n,2n), (n,3n), (n,n'α), (n,n'p) Reactions

They are assumed to be isotropic in the center of mass system.

4 ENERGY DISTRIBUTION OF SECONDARY NEUTRONS

All of them were calculated by using MUP2 code.

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THE EVALUATION OF COBALT DATA FOR 10^{-5} eV TO 20 MeV NEUTRONS*

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INTRODUCTION

Cobalt is a constituent of structural materials in the nuclear power systems, and also an adequate material for threshold activation detector and fast neutron dosimetry. So the cobalt neutron data are very important. But there are large discrepancies between the various reported results for some reactions in the sixties and seventies.

The more precise measurements were given in the recent years. So it is necessary to renew the evaluation of cobalt data. In this evaluation the following improvements were made :

A. The data measured in recent years were added in, such as the inelastic scattering cross sections to the discrete levels measured by Leclaire et al.[1]. Because the Ge(Li) detector was used, the more discrete states were determined. The data of (n,p) and (n,2n) cross sections reported in the international conference on Nuclear Data for Basic and Applied Science, Sanda Fe, USA (1985) are more consistent.

B. The nuclear model calculations[2] were completed carefully. Based on the evaluated measured data, the parameters were adjusted and best ones were found, which fit the experimental data best. So the reliabilities of calculated data are increased.

This evaluation covers the incident neutron energy E_n from 10^{-5} eV to 20 MeV.

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* This evaluation was completed in 1986. The comparisons between this evaluation and ENDF / B–6 and JENDL–3 were carried out in 1992.

1 THE THERMAL CROSS SECTIONS AND RESONANCE PARAMETERS

The thermal (0.0253 eV) cross sections and resonance parameters are both obtained from Ref. [3]. The spins of the weak resonances are assigned randomly in order to obey the (2J+1) law for level density. The resonance region covers 10^{-5} eV to 80 keV. However, the list of the resonances are extended to 100 keV and care has been taken to connect smoothly with the data above 80 keV. There is no unresolved region as the resonance peak can be resolved clearly up to 80 keV.

The values of thermal cross sections are as follows :

Total cross section = 43.18 b

Scattering cross section = 6.00 b

Capture cross section = 37.18 b

2 FAST NEUTRON CROSS SECTIONS

2.1 Total Cross Section

As mentioned above, from 10^{-5} eV to 80 keV, the total cross sections are represented by S– and P–wave resonance parameters. But actually, the resonances exist up to about 5 MeV. There were only two papers about the precise measurement : 3804 points in the energy region 588 eV to 220 keV by Garge et al.[4] and 6178 points in the region 0.36 to 32 MeV by Cierjacks[5]. The data were measured in the sixties, but they are still the best precise to date. So these data are adoped. Some points were discarded in such a fashion as to reduce the data points required without sacrificing important structural information. In the region of 220～360 keV, the precise measured data are absent. Those data are taken from ENDF / B–4, which were calculated vai R–matrix.

Above 5 MeV, in addition to the data of Cierjacks, the data of Refs. [6～10] were used too, there are no systematic deviation in the various Refs. The recommendation in this energy region is based on the spline function fit.

2.2 Elastic Scattering Cross Section

From 80 keV to 20 MeV, the data were obtained by subtracting the nonelastic scattering cross sections from the evaluated total cross sections. The results agree with the measured data of Refs. [11～16] roughly.

The comparison between this evaluation and that of ENDF / B–6, JENDL−3 and measured data are shown in Fig. 1.

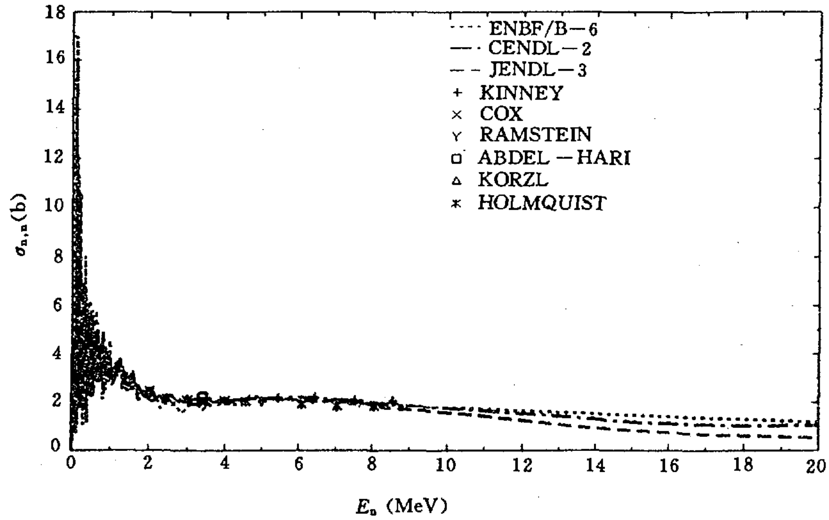

Fig. 1 The comparison of (n,n) cross sections between this evaluation and those of ENDF / B−6, JENDL−3 and measured data

2.3 Inelastic Cross Section

(1) The total inelastic cross sections

There are no experimental data for the neutron energy region 3∼ 14 MeV and only a few measured data in other energy regions. So the evaluation is mainly based on the nuclear model calculations. The results agree with the measured data of Ref. [17] in the region below 2.2 MeV. However, the results are lower than the data of Ref. [17] in the region of 2.2∼ 2.7 MeV. The evaluation agree with Ref. [18] in the region of 16∼20 MeV.

(2) The inelastic scattering cross section to the discrete levels.

The 11 discrete levels were given in ENDF / B−5 which was based on the experimental data of Guenther et al.[19] in the neutron energy region 1.8∼ 3.1 MeV and Etemad[20] in the neutron energy region 2.0∼4.5 MeV. In these energy regions, more precise data measured by Leclaire[1] were published in 1978. By using the Ge(Li) detector and TOF technique, some levels unresolved in the earlier papers were resolved. The 21 discrete levels were given in the energy region from 1.099 to 2.962 MeV. In the same incident neutron energy, the highest discrete levels measured by Leclaire are higher than that of Refs. [19,20]. Some example measured by Leclaire are shown in table 1.

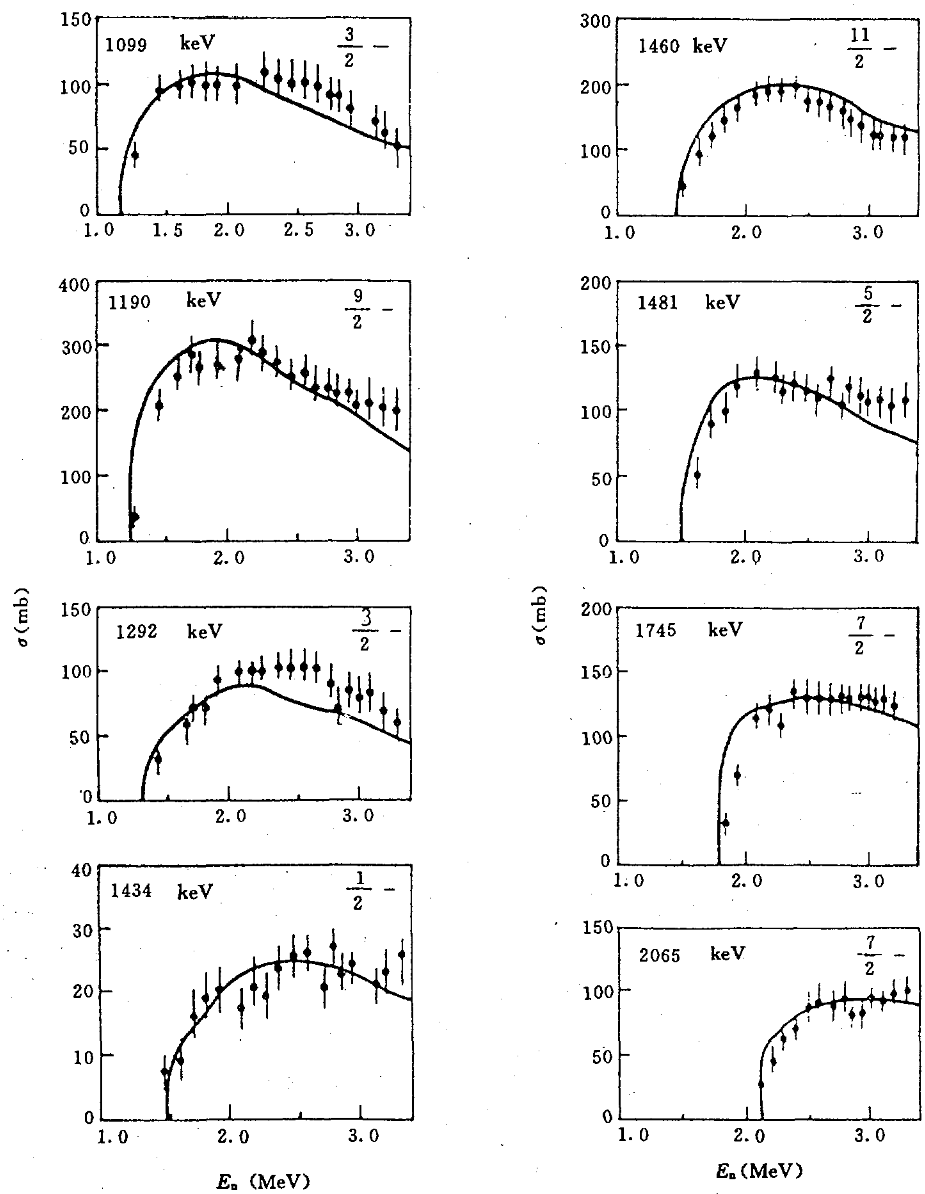

Fig. 2a The cross sections of inelastic neutron scattering to the excited states from 1099 to 2065 keV are compared with measured data of Leclaire (1978)

I Leclaire, ——— Present work

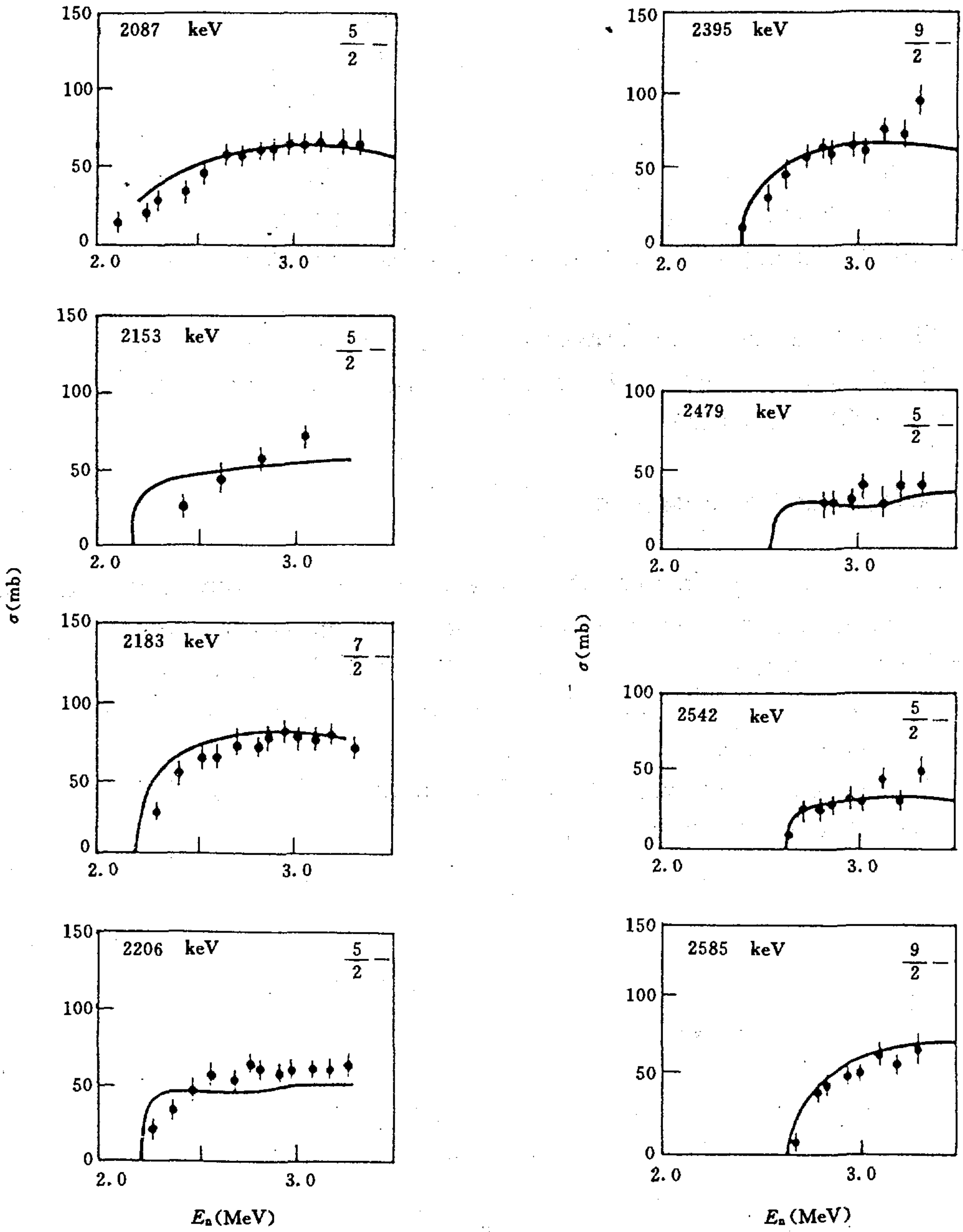

Fig. 2b The cross sections of inelastic neutron scattering to the excited states from 2087 to 2585 keV are compared with the data of Leclaire (1978)

Table 1 The measured highest discrete levels

E_n (MeV)	Leclaire	Guenther	Almen
2	1.744	1.460	1.100
3	2.962	2.087	1.744

In this evaluation, they were obtained from the calculations based on the Hauser–Feshbach theory with the width–fluctuation correction. The results agree with the Ref. [1] very well as shown in Fig. 2a and Fig. 2b.

(3) The inelastic scattering cross sections to the continuum region

The data were calculated based on the pre–equilibrium theory. The effective threshold is 2.82 MeV.

2.4 (n,2n) Reaction Cross Section

There are a large discrepancies between the various reported results measured in the sixties, for instance, the maximum value is seven times larger than the minimum. But the data measured in the seventies are consistent with each other, except a few exceptions. This evaluation are based on the data published later than the seventies[21~29].

The comparison between this evaluation and ENDF / B–6, JENDL–3 and measured data are shown in Fig. 3.

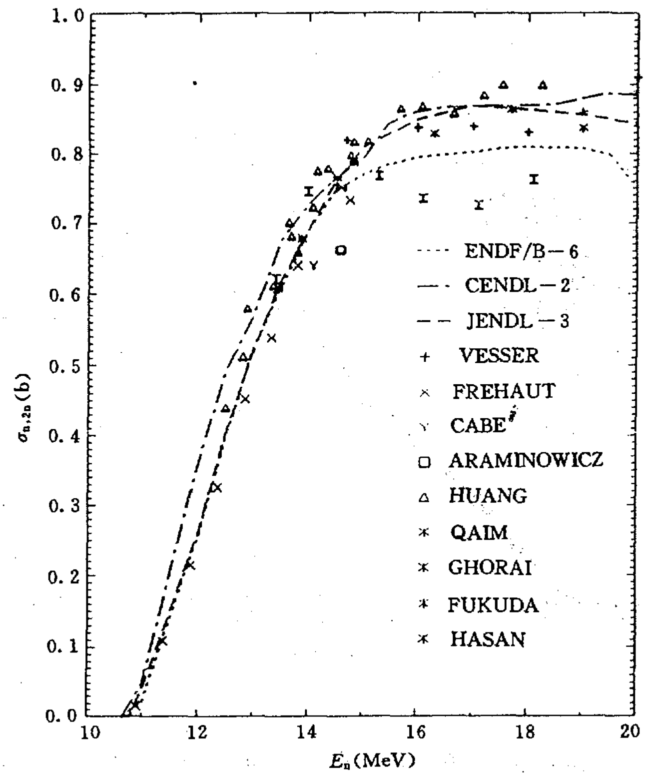

Fig. 3 The comparison of (n,2n) cross sections between this evaluation and that of ENDF / B-6, JENDL-3 and measured data

2.5 (n,p) Reaction Cross Section

Most earlier measurements of (n,p) reaction cross section were made at around 14.5 MeV, except Smith's[30] measurements in the region from 2.5 to 10 MeV. Above 15 MeV, recently several measurements[21, 31] were made at around 14.5 MeV and in the region 14~ 18 MeV. These data agree with each other. This evaluation is based on the results of Refs. [21, 30, 31].

The comparison between this evaluation and ENDF / B-6, JENDL-3 and measured data are shown in Fig. 4.

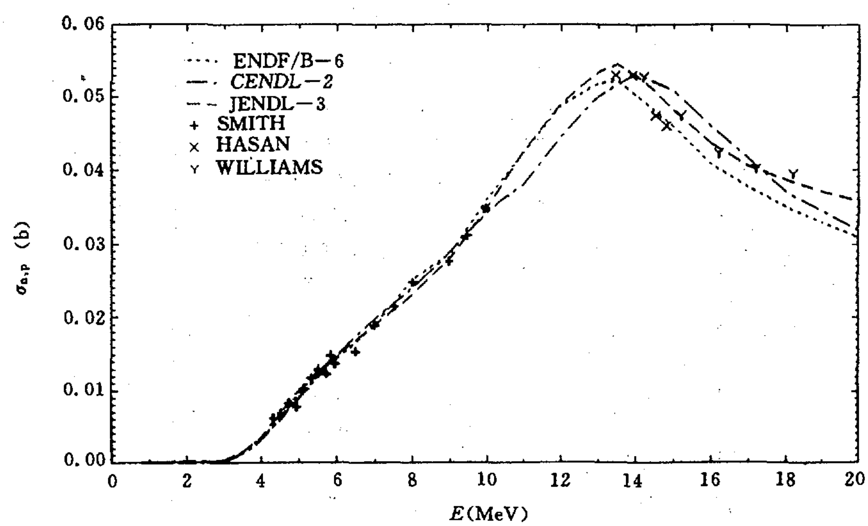

Fig. 4 The comparison of (n,p) cross sections between this evaluation and that of ENDF / B−6, JENDL−3 and measured data

2.6 (n,α) Reaction Cross Section

The evaluated data were obtained from the spline fitting the experimental data of Refs. [24, 28, 32∼35]. In the region below 5 MeV, the cross section is negligible due to the effect of Coulomb barrier. The comparison between this evaluation and ENDF / B−6, JENDL−3 and measured data are shown in Fig. 5.

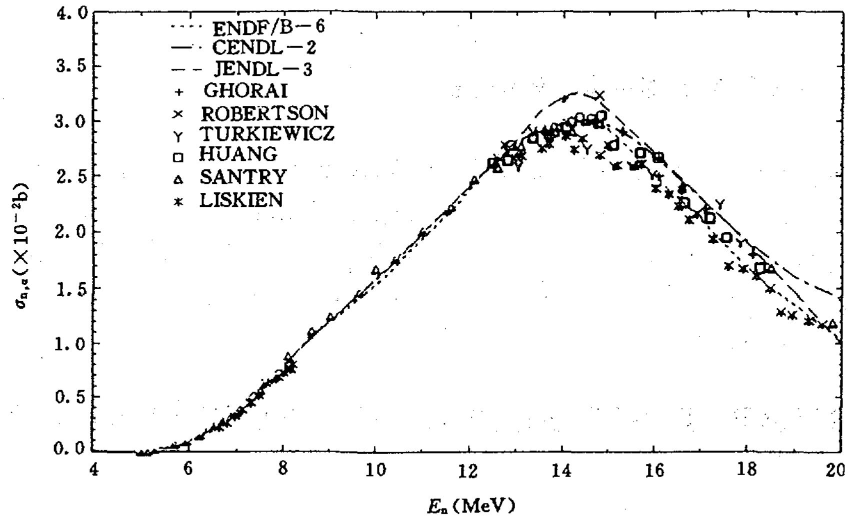

Fig. 5 The comparison of (n,α) cross sections between this evaluation and that of ENDF / B–6, JENDL–3 and measured data

2.7 The Radiative Capture Cross Section

The evaluated data from 80 to 1000 keV are based on the spline fitting measured data of Ref. [36]. The experimental data from 1 to 20 MeV are scarce. The data of Refs. [36~39] provide a reference for the evaluation.

The comparison between this evaluation and ENDF / B–6, JENDL–3 and measured data are shown in Fig. 6.

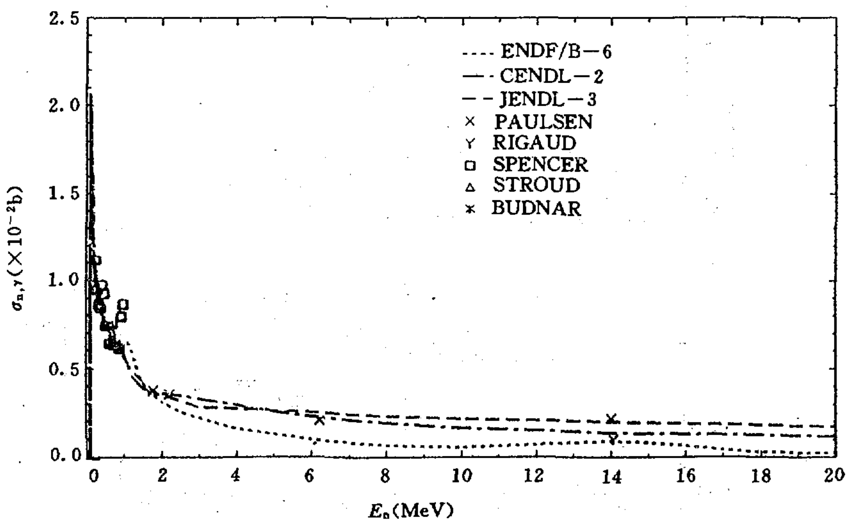

Fig. 6 The comparison of (n,γ) cross sections between this evaluation and that of ENDF / B–6, JENDL–3 and measured data

2.8 Other Small Cross Section Reactions

(n,t), (n,d), (n,^{3}He), (n,n′p) and (n,n′α) reaction cross sections are all in the order of mb or smaller. The shortage of experimental data, large measurement error, and great discrepancy between the values from the various experimenters are typical case for these cross sections. These evaluations are based on the nuclear model calculations.

3 ANGULAR DISTRIBUTION OF SECONDARY NEUTRONS

3.1 Elastic Scattering Angular Distribution

Based on the measured data of Refs. [11, 12, 15, 19, 40～42] the differential cross sections were calculated by using nuclear model theory. Several evaluated angular distribution curves compared with the measured value are shown in Fig. 7.

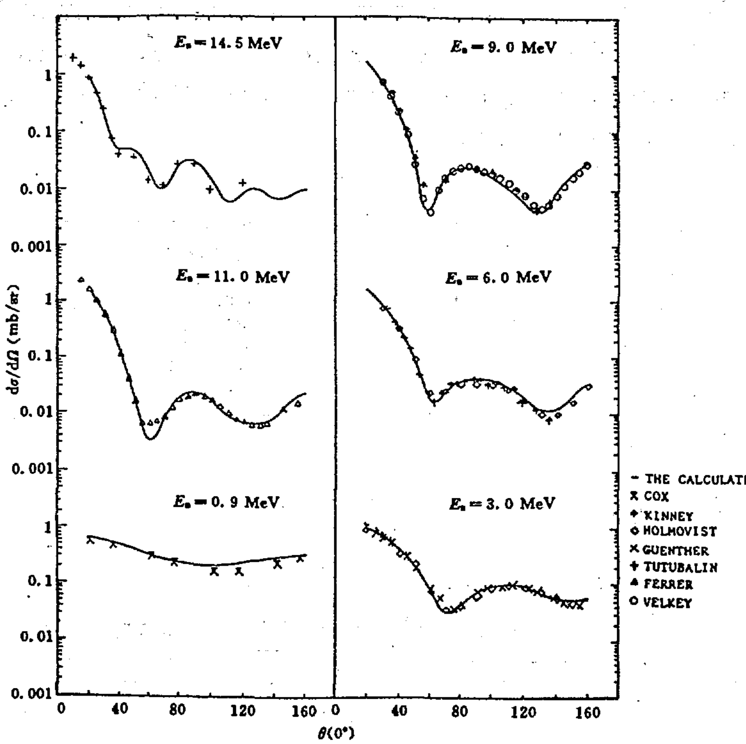

Fig. 7 The evaluated angular distribution curves are compared with the experimental data

3.2 Inelastic Scattering Angular Distributions

The angular distributions for inelastic scattering to the discrete levels were calculated by using nuclear model code, and for that to the continuum, isotropic angular distribution in the center of mass system is assumed.

3.3 Angular Distributions of (n,2n), (n,3n), (n,nα), (n,np), (n,nd) and (n,nt) Reactions

Isotropic angular distributions in the center of mass system are assumed.

4 SECONDARY NEUTRON ENERGY DISTRIBUTION

The data were obtained from the nuclear model calculations.

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THE EVALUATION OF NATURAL CADMIUM NEUTRON DATA

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ABSTRACT

A comprehensive evaluated neutron nuclear data set for cadmium, in the energy extending from 10^{-5} eV to 20.0 MeV, is described. The measured data, the application of theoretical models, and the evaluation method are outlined. Attention is given to uncertainty specification and comparisons are made with the previous evaluation. The corresponding numerical file, in ENDF / B–5 format, has been stored in CENDL–2.

INTRODUCTION

Cadmium is one of the important materials in control rod of nuclear reactor. At the earlier stage, measured data were sparse, the measurements were carried out in a short energy range. Recommended values were almost given by theoretical calculation. Since 1970 the measurement method has been greatly improved, and some new results are available, so that it is possible to give recommended values from the combination of experimental data and theoretical calculation.

The natural cadmium contains eight isotopes. For simplification's sake, six isotopes, (Isotopes ^{106}Cd and ^{108}Cd small abundance are put into ^{110}Cd.) are taken for the calculation of reaction cross sections. But in resonance energy region, we still consider that the natural cadmium includes eight isotopes, so as to obtain more accurate results.

1 RESONANCE PARAMETERS

Both CNDC[1] and BNL325(4th Ed.)[2] gave the detailed evaluation for cadmium. These two sets of parameters are similar. Two problems have been noticed. Firstly, natural cadmium has eight isotopes, and the high energy limit of each isotope in the resolved resonance region are different. The way of dealing with this problem is to use the energy limit as high as possible. In present work, 10 keV was used. On the other hand a point–wise representation of the total cross sections in some energy range of the isotopes ^{106}Cd and ^{108}Cd is used, for which not enough experimental information is available. Secondly, this evaluation is mainly based upon the work of BNL325(4th Ed.) and complimented with several parameters from Ref. [3]. Some calculated data using these parameters are compared with experimental data as follows :

	This file	Recommended exp. data
Thermal cross section (b)	2512.86	2520.0 ± 50
Resonance integal (b)	81.42	69.6 ± 5.7

2 REACTION CROSS SECTIONS

2.1 Total Cross Section

In the 0.05 ~ 20 MeV energy range the experimental data of Ref. [4] and Ref. [5] are in agreement very well with each other in the cited errors. These two measurements are adopted in this evaluation (Fig. 1). In the lower energy range (1 ~ 50 keV), the measurements[6~8] were done before 1970. The agreement of these measurements are not so good. Model theory calculation fitting these measurements is adopted in this evaluation.

2.2 Elastic Scattering Cross Section

The experimental information of this reaction for natural cadmium in 0.3 ~ 4 MeV energy range is mainly provided by Vonach[9] and Smith[10]. The energy regions of these measurements are overlappe with each others (Fig. 2). The trend of the data is the same, the cross sections were obtained from differential elastic scattering measurements. These data were used in our theoretical calculation to adjust the parameters. These adjusted parameters are then used in the calculation for the rest energy range.

2.3 Nonelastic Scattering Cross Section

There are a few experimental data in the energy range (1 ~ 7 MeV), with large uncertainties[11~14]. These data have been used to check the model theory calculation, and the calculated results are adopted in this evaluation.

2.4 Inelastic Scattering Cross Sections

Only Smith[10] measured these cross sections in the 2~4 MeV energy region for some discrete energy levels. These data have been used for model theory calculation and the calculated results were adopted in this evaluation.

2.5 Radiative–Capture Cross Section

From 0.3 MeV to 4 MeV energy region, there are only three measurements for capture cross section of Cd[15~17]. Our recommended values are based of the fitted values and extrapolated to 20 MeV based on ENDF / B–6 (Fig. 3).

2.6 (n,2n) Reaction Cross Sections

Only one measured datum of this cross section for natural cadmium at 14 MeV is available[18]. The model theory calculations are adopted for present evaluation.

2.7 Other Reaction Cross Sections

For natural cadmium's (n,3n) and the charged–particle–emission reactions such as (n,α), (n,p), (n,np), (n,d), only a few measured data are available, and the values are small, so our recommended values are based on model theory calculation.

3 ELASTIC SCATTERING ANGULAR DISTRIBUTION

The experimental data were taken from the measurements of Vonach[9] and Smith[10] in 0.3~4 MeV energy range (Fig. 4). The measured data were transferred from laboratory system to center–of–mass system, and then fitted with Legendre polynomials. The data for the unmeasured energy region were complemented by theory calculations.

4 THEORITICAL CALCULATIONS

The model theory parameters are taken from Refs. [19~23] and calculations were carried out using codes AUJP and MUP2. The calculated results including total, nonelastic scattering, elastic scattering and (n,γ) as well as the angular distribution agree with measured data quite well.

5 DISCUSSION

As the calculated values agree quite well with the measured data, both of them can be used as recommended values. In this evaluation, the experimental data were used for total cross section, (n,γ) and (n,n) cross sections. For other cross sections the calculated values were used.

The comparisons of our results with ENDF / B–6 are shown in Fig. 5.

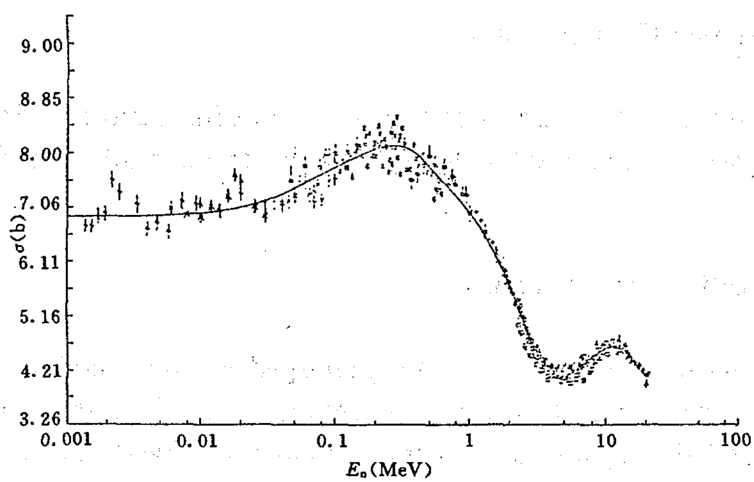

Fig. 1 The total cross section of Cd

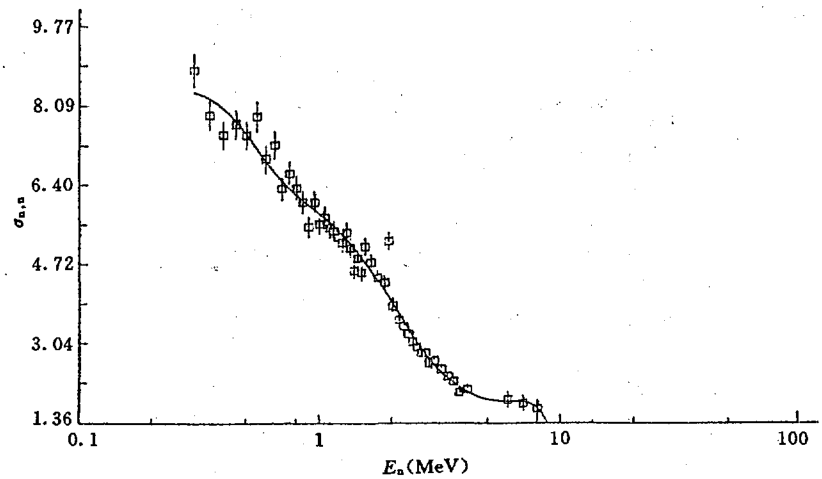

Fig. 2 The (n,n) cross section of Cd

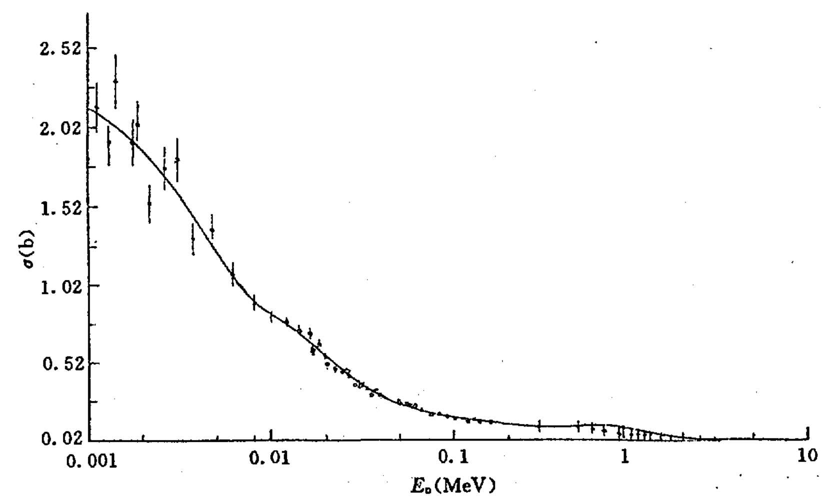

Fig. 3 The (n,γ) cross section of Cd

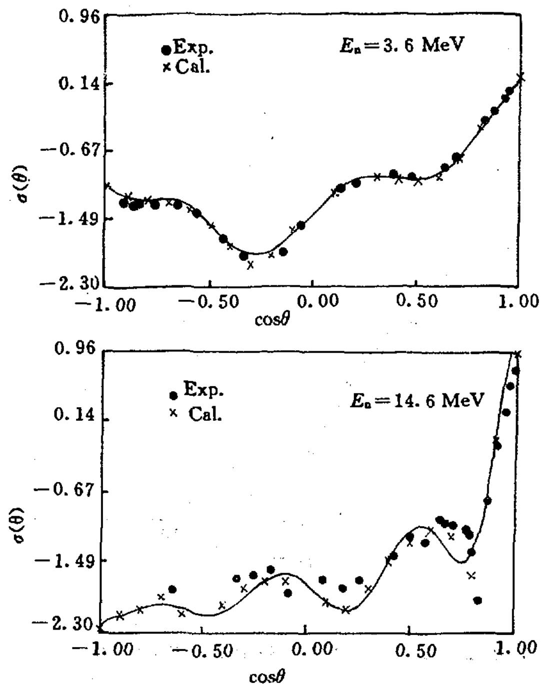

Fig. 4 The Angular distribution of elastic scattering on Cd for 3.6, 14.6 MeV neutron

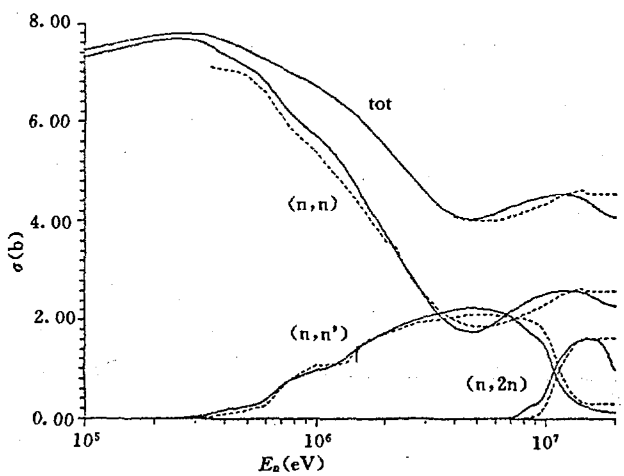

Fig. 5 Some cross sections for Cd from CENDL−2 and ENDF / B−6

——— CENDL−2, ---- ENDF / B−6

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EVALUATED NEUTRON DATA FILE FOR INDIUM

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INTRODUCTION

Natural indium consists of two isotopes ^{113}In (4.3 ± 0.2) %, ^{115}In (95.7 ± 0.2) %. The ^{113}In is stable, and the other is a long life beta decay nucleus, its half life is (4.41 ± 0.25) 10^{14} years. These two isotopes appear as fission products with high yields. The metal Indium is deformable in mechanical process, so it has been widely used in nuclear dosimetry.

Most early evaluations, such as Refs.[1~ 5] were out–of–date. The evaluation in ENDF / B–4 was done by Schmittroth in 1973. After that, many special files about Indium isotopes were completed in ENDF / B–5[6~ 8] in 1979 and ENDF / B–6 in 1990. This evaluation for CENDL–2 was accomplished in 1989.

1 RESONANCE PARAMETERS

There were two evaluated sets of resonance parameters, the CNDC[9] in 1981 and the BNL 325[10] in 1980. The CNDC data were adopted. Its resolved resonance region is up to 2 keV. Some calculation data using these parameters with experimental data have been compared as follows :

	This file	Recommended exp. data
Thermal cross section (b)	199.5	193.8 ± 1.5[10, 11]
Resonance integal (b)	3160.2	3133.0 ± 75[11]
Act. cross section at 5×10^{-5} eV (b)	4270	4080.0 ± 370[12]

The unresolved resonance region is from 2~ 50 keV. The early measured data of total cross sections seem too low (6.2~6.4 b)[13]. More recently data is between 7.0~7.2 b[14, 15].

2 TOTAL CROSS SECTION

For natural indium more than twenty sets of experimental data are available, but none of the data were obtained with the white–source facilities.

Transmission measurements were performed by Smith[16, 17] Foster[18], and Poenitz[14] in the energy ranges from 47 keV to 14 MeV. These measurements are consistent with each other and they were taken as our main references. The measurements of Peterson[19] from 17.5~ 28.9 MeV, Kent[20], Malmberg[21], Coon[22] around 3.66 MeV, 6.70 MeV and 14 MeV respectively, were also considered.

The recommended values from 50 keV to 20 MeV were based on a least squares fit to these data. The uncertainly is about 2 %, but below 200 keV the uncertainly is great than 5 % and above 15 MeV the uncertaintly is 4 %.See Fig. 1.

3 ELASTICAL SCATTERING CROSS SECTION

From 50 keV to 6 MeV energy region, the measured data of Vonach[23], Cox[24], Smith[17] and Holmqvist[25] were adopted. Above 6 MeV, the theoretically calculated cross sections were adopted. See Fig. 2.

4 INELASTIC AND NONELASTIC SCATTERING CROSS SECTIONS

Five sets of experimental data for inelastic scattering from 0.5 to 9 MeV were used to get a fitting curve. But there are no data from 1.5 to 4 MeV. So the fitting value was only used to check the calculations. Thus the calculated data was recommended.

The direct nonelastic scattering cross section measurement was done in the early years[26]. Thomson[27] reported for heavy nucleus the nonelastic cross section is the same as the inelastic and some data was given.Smith[17] also provided some data. These three sets of data are consistent.

5 CAPTURE CROSS SECTION

Recent six sets of experimental data were selected for Indium from a few keV to 4 MeV[14, 28~32].

Fitted cross section below 4 MeV was recommened as (n,γ) cross section.

See Fig. 3. Above 4 MeV the calculated was adopted.

6 (n,2n), (n,3n) CROSS SECTIONS

There has been no experimental data obtained by using direct neutron detection methods. The activation measurements actually give only the (n,2n) cross sections.

Using code MUP2 the (n,2n), (n,3n) cross sections were calculated. By adjusting the parameters, the calculated values are consistent with the measured data.

7 (n,x) REACTIONS

A lot of reaction channels are open up to 20 MeV. Generally, the cross sections are very small for these channels. Calculation shows that at 20 MeV the $(n,{}^{3}He)$ cross section is only pb, the (n,t), (n,d) cross sections are in microbarn region, even the (n,α), (n,p) reactions, though the cross sections are relatively large, are still in millibarn range. See Fig. 5.

8 ANGULAR DISTRIBUTION

For angular distribution of elastic scattering the reference literature have been collected. Three sets of data Vonach[23], Smith[17] and Holmqvist[25] were adopted. The calculated angular distributions at 12 energy points from 0.3 to 11 MeV are compared with the experimental values in Figs. They are consistent with each other (see Fig. 4). For the angular distribution of inelastic scattering and secondary neutron energy, no experimental data have been found and the theoretical calculated data were used.

9 DISCUSSION

Calculated cross section using code MUP–2 are consistent with the experimental values. For total cross section, scattering cross sections from 0.05 to 16 MeV, capture cross section from 0.05 to 4 MeV, the fitted experimental data were recommended. The other recommended data were taken from model theory calculation.

On the whole, this evaluation is more close to the experimental data than those of ENDF / B–4 and ENDF / B–5 files.

Recently the ENDF / B–6 was completed by Smitt in 1990. In our evaluation the total, elastic and capture cross sections were directly taken from experimental data, so for these data there are no discrepancy between ours and those of ENDF / B–6 data. The resonance parameters and angular distribution were also taken from the same data base, so our results are similary to the results of ENDF / B–6 file.

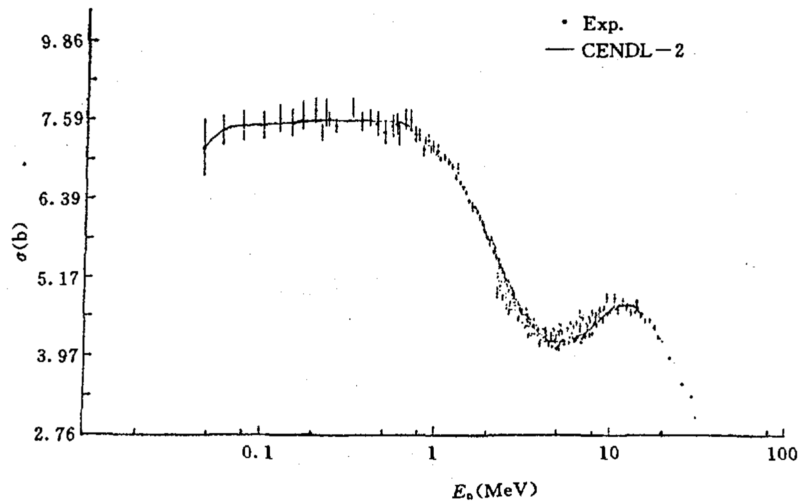

Fig. 1 The total cross section of In

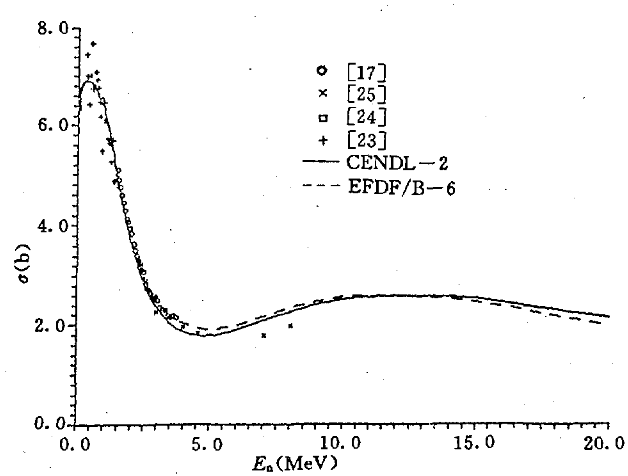

Fig. 2 The elastic scattering cross section of In

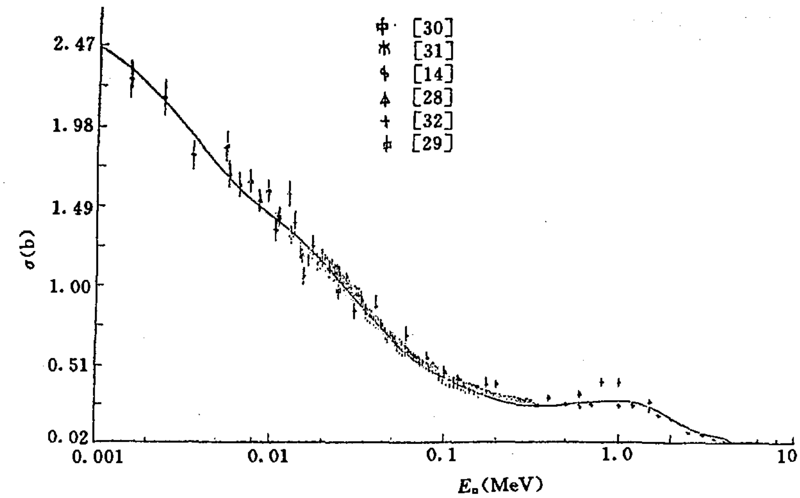

Fig. 3 The capture cross section of In

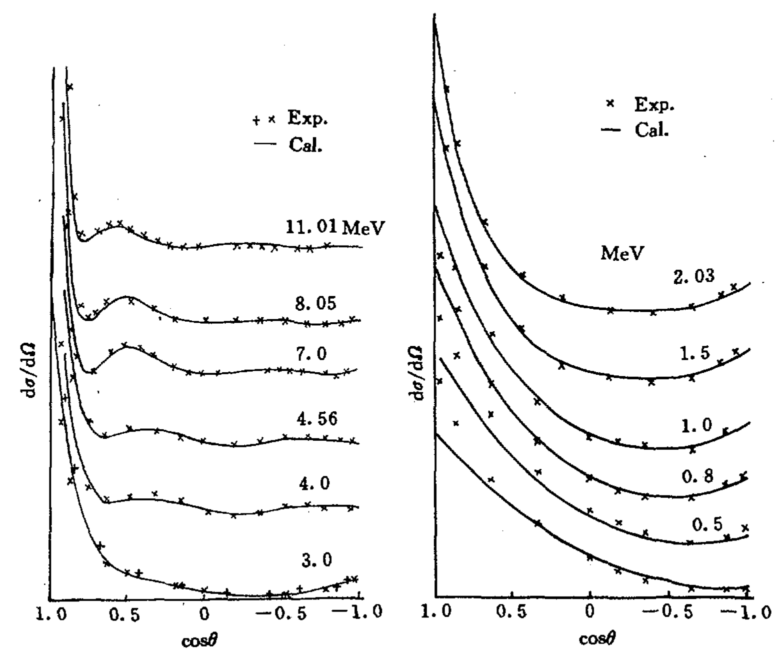

Fig. 4 The angular distribution of neutron elastic scattering on In

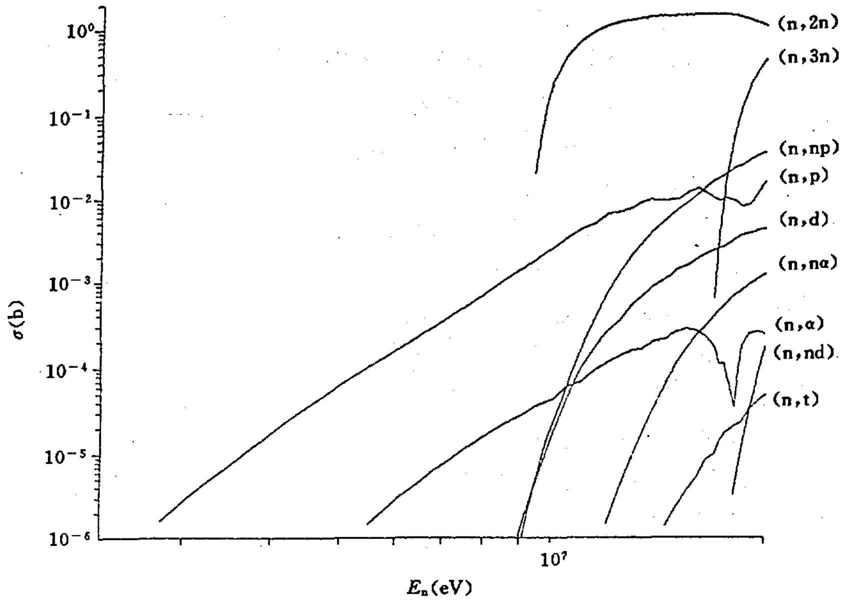

Fig. 5 Evaluated (n,x) reaction cross sections of In

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EVALUATION OF HAFNIUM NEUTRON CROSS SECTIONS

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INTRODUCTION

Neutron cross sections of the natural hafnium and hafnium isotopes (A = 176~ 180) were evaluated. Although hafnium is a good material as nuclear reactor control rod, its evaluated neutron data are scarce yet. Reynold's evaluation[1] was completed in 1967, many of his data were theoretically calculated or extrapolated from other nuclei. Drake's report[2] was completed in 1976, it was a evaluation for ENDF / B–5 file, in 1990 it was directly transfered to ENDF / B–6 file.

Our evaluation for CENDL–1 was completed in 1983, and was revised for

CENDL–2 in 1991.

1 RESONANCE PARAMETERS

The resonance parameters of Mughabghab[23] up to 100 keV were used. The resonance results are qualitatively similar to those of ENDF / B–6 and JENDL–3, because the experimental data base has not been changed greatly during these years.

2 CROSS SECTIONS

2.1 Total Cross Section

A comprehensive experimental total cross section data were collected from literatures and EXFOR file. The data[4~7] were taken as references. Three sets of data[8~10] are important in the energy region from 0.1 MeV to 15 MeV. The smooth fitted values were adopted as recommended data and was shown in Figure 1. The uncertainty from 0.1 MeV to 15 MeV was about 2 %, and above 15 MeV it could be 5 %, as there are no measureded data.

2.2 Elastic Scattering Cross Section

Sherwood[8] gave the data from 0.3 to 1.5 MeV. But the first four values are unreasonably greater than the total cross section. Some others[14, 15] are consistent with Sherwood's in the 1 MeV region. Two single point measurements [16, 17] at 7.0 MeV and 8.05 MeV were adopted. For ENDF / B–6 data, there is a peak in 8~9 MeV region (Fig. 2), which seemes unreliable.

2.3 Inelastic Scattering Cross Section

Sherwood[8] gave the inelastic excitation cross sectios of Hafnium at incident neutron energies from 0.3 to 1.5 MeV. The total inelastic cross section was the sum of all excited state values. But the sum of total inelastic cross sections and elastic scattering cross sections is unreasonably larger than total cross section. Thus the theoretic calculation data were adopted.

2.4 Capture Cross Section

Three sets of data[11~13] were selected.

2.5 (n,2n) Cross Section

Lakshmana's confirmation of the (n,2n) cross section for ^{174}Hf is 1968 ± 148 mb[18]. For ^{176}Hf, four measurements confirm the (n,2n) cross section is 2000 mb. From systematic[19], the (n,2n) cross section at 14.7 MeV for Hafnium is 2093 ± 144 mb. This value was used to adjust the pre–equilibrium emission factor in theoretical calculation.

2.6 (n,p), (n,α) Cross Sections

At 14.7 MeV region, these values are only 2 mb, which is a small value comparing to the others.

3 ANGULAR DISTRIBUTION OF SECONDARY NEUTRONS

The measured data were found only at incident neutron energies from 0.3 MeV to 1.5 MeV, and at 7.0 MeV, 8.05 MeV. The calculated values and the measured values are given in Fig. 3.

4 RECOMMENDATION

All cross sections were calculated by using optical model, H – F theory and pre–equilibrium model. It seems reasonably in agreement between experimental data and calculated results. But 5 % discrepancies for the total cross section in energy region 12 ~ 15 MeV still exists.

For total cross sections, the spline function fitting of experimental data was adopted. For elastic scattering cross sections the recommended value was obtained by subtracting nonelastic scattering cross section from the fitted total cross section. All other data were taken from calculation values (Fig. 4).

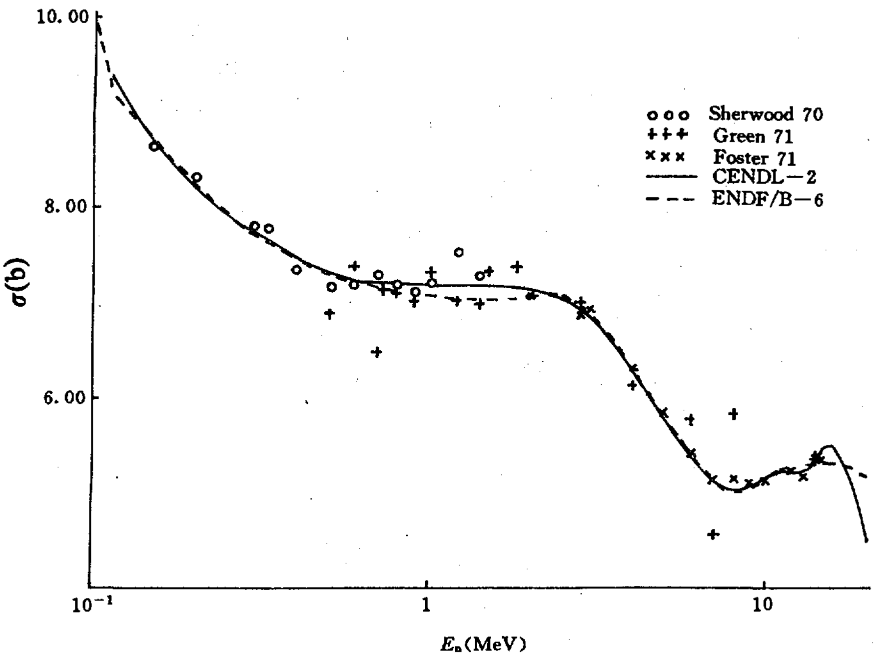

Fig. 1 The total cross section of Hf

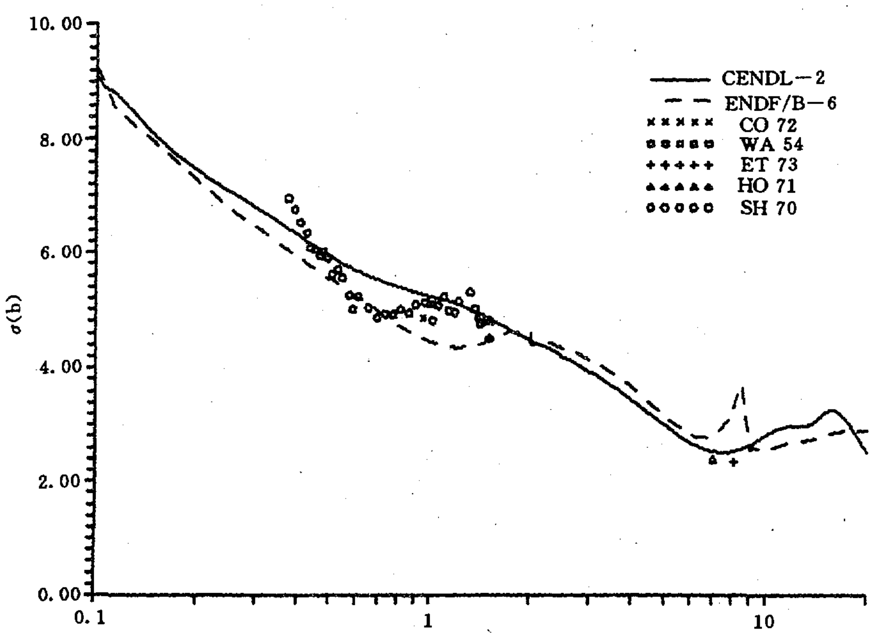

Fig. 2 The elastic scattering cross section of Hf

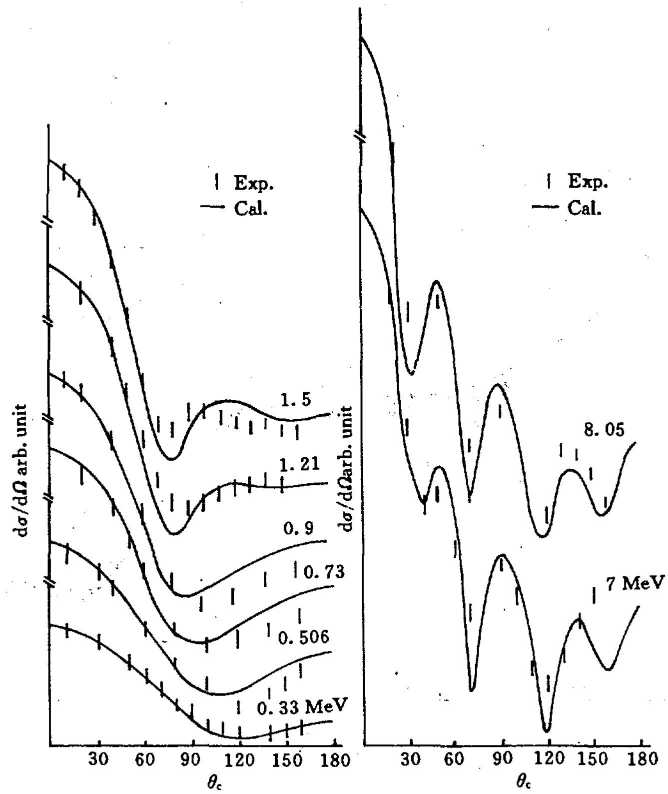

Fig. 3 The angular distribution of neutron elastic scattering cross section on Hf

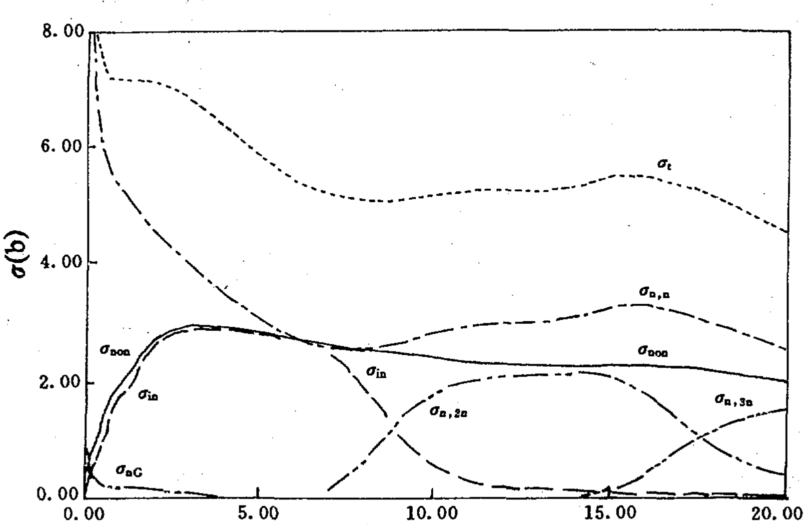

Fig. 4 The evaluated cross section of Hf

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EVALUATION OF NEUTRON NUCLEAR DATA OF ^{239}Pu FOR CENDL-2

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ABSTRACT

Based on the experimental data evaluation and theoretical calculation, the complete neutron nuclear data of ^{239}Pu have been evaluated in the incident neutron energy range from 10^{-5} eV to 20 MeV for CENDL-2. The comparison of present evaluation with ENDF / B-6, JENDL-3 and BROND-2 has been carried out.

. INTRODUCTION

^{239}Pu is one of the important fuel material in the reactors. The evaluation of neutron nuclear data of ^{239}Pu for CENDL-1 was finished by Liang Qichang et al.[1] in 1979, since then, many significant new experimental data, especially for (n,tot), (n,F) cross sections become available, so it is necessary to improve and update the previous evaluation to meet the requirement of the nuclear power development.

1 AVERAGE NUMBER OF FISSION NEUTRONS

The average number of total, delayed and prompt neutrons per fission for ^{239}Pu were taken from JENDL-3, the average number of prompt neutron is in agreement with those evaluated by Liu Zuhua[2] in 1976.

2 RESONANCE PARAMETERS

The resolved and unresolved resonance parameters were taken from JENDL−3.

3 NEUTRON CROSS SECTIONS

In the incident neutron energy range from 10^{-5} eV to 20 MeV, the main reaction cross sections are (n,tot), (n,n), (n,n′), (n,2n), (n,3n), (n,F) and (n,γ), the details of their evaluation are described as follows :

3.1 Total Cross Section

The evaluated total cross section by Tang Guoyou[3] was adopted, which was obtained by fitting the experimental data up to 1990 with the orthogonal polynomial method.

3.2 Elastic Scattering Cross Section

The elastic scattering cross section was obtained by substracting all partial cross sections from the total cross section, and was compared with experimental data[4~9], they are basically in agreement with each other.

3.3 Inelastic Scattering Cross Sections

There are only two experimental data sets (four energy points in all) [10, 11] for total inelastic scattering cross section. Therefore the theoretical calculation result was normalized to the experimental data and adopted as the recommended data.

Due to no pertinent data are available for discrete and continuous inelastic scattering cross sections, the theoretical calculation results were adopted.

3.4 (n,2n) Cross Section

In the energy range from threshold to 14 MeV, the evaluated (n,2n) cross section performed by Cai Dunjiu[12] was adopted, which was obtained by fitting the experimental data with the orthogonal polynomial method. For $E_n > 14$

MeV, the theoretical calculation result was adopted and normalized to the experimental data at low energy range.

3.5 (n,3n) and (n,4n) Cross Sections

Due to there are only one energy point experimental data of D. S. Mather[13] for (n,3n) cross section, and no experimental data are available for (n,4n) cross section, so the theoretical calculation results were adopted.

3.6 Fission Cross Section

The evaluation of fission cross section of ^{239}Pu for CENDL−1 was performed by Liu Jicai[14] in 1975. Since then, many significant new experimental data, especially in the energy region from 10 to 15 MeV, become available. The present evaluated cross section was obtained by fitting the experimental data up to 1990[15~26] with the Spline function method.

3.7 Radiative Capture Cross Section

Up to now, only few experiment was carried out for directly measuring the capture cross section of ^{239}Pu, most of $\sigma_{n,\gamma}$ was derived from the $\alpha(\sigma_{n,F} / \sigma_{n,\gamma})$ ratio values, the α ratio values of ^{239}Pu for $E_n < 1$ MeV were evaluated by Liang Qichang[27] in 1977. Since then, no significant new experimental data are available. So present evaluated (n,γ) cross section below 1 MeV was derived by combining those α ratio values with the above mentioned new evaluated $\sigma_{n,F}$. For $E_n > 1$ MeV, the theoretical calculation result was adopted and normalized to the experimental data at low energy region.

4 ANGULAR DISTRIBUTIONS OF SECONDARY NEUTRONS

The angular distributions of elastic and discrete inelastic scattering neutrons were obtained from theoretical calculation results and were given in Legendre coefficients in the C. M. System, the former was compared with the experimental data[7~9, 28], and are basically in agreement with each other.

The angular distributions for (n,2n), (n,3n), (n,4n) and fission neutrons were assumed to be isotropic in Lab. system.

5 ENERGY DISTRIBUTIONS OF SECONDARY NEUTRONS

All energy distributions of secondary neutrons emitted from various reaction channel were taken from theoretical calculation results and represented by the tabulated function.

6 THEORETICAL CALCULATION

As mentioned above, for the cross sections of no experimental data and all angular and energy distributions of secondary neutrons, the recommended data were obtained from the theoretical calculation results by using code FUP1[29], in which the optical model, F–H theory with width fluctuation correction (WFH) and pre–equilibrium evaporation model based on excitation model (PE) were included. The input parameters for FUP1 were adjusted based on the experimental data by using other codes as described below.

6.1 Compound Nuclear Reaction

The compound component was calculated with optical and statistical model, the spherical optical potential parameters adjusted with code ASOP[30] based on the experimental data of σ_{tot}, σ_{non} and $\sigma_{n,n}(\theta)$ are as follows (MeV or fermi):

$A_R = 0.5799$	$V_0 = 49.5000$
$A_S = 0.5011$	$V_1 = -0.4397$
$A_{VV} = 0.5011$	$V_2 = 0.0156$
$A_{SO} = 0.5799$	$V_3 = 24.000$
$X_R = X_{SO} = 1.2632$	$V_4 = 0.0$
$X_S = X_{VV} = 1.3475$	$W_0 = 6.405$
$U_0 = 1.48$	$W_1 = 0.1529$
$U_1 = -0.03247$	$W_2 = -12.00$
$V_{SO} = 6.2000$	

6.2 Direct Interaction

In the high neutron energy range, the direct component becomes important, it was calculated with coupled channel optical model, the deformed optical potential parameters adjusted with code CCOM[31] are as follows :

$V = 45.25 - 0.25 * E_n + 0.02 * E_n^2$	$R_{SO} = 1.12$
$W_S = 2.862 + 0.4 * E_n$	$A_R = 0.62$
$W_V = 0.0$	$A_S = A_V = 0.58$
$V_{SO} = 6.2$	$A_{SO} = 0.62$
$R_R = R_S = R_V = 1.26$	$\beta_2 = 0.21$

6.3 Fission Parameters

The fission parameters include energy level density a, pair correlation δ, fission barrier width hw, equivalent fission barrier height V_{pf} and the energy level density parameters at saddle states P_{k1}, P_{k2}. These parameters were adjusted by code ASFP[32] and are listed as follows :

	1st fission	2nd fission	3rd fission	4th fission
V_{pf}	6.1979	5.7100	5.4719	5.7100
P_{k1}	1.9545	1.3361	1.0728	1.3400
P_{k2}	0.6234	0.5439	0.0347	0.5400
δ	1.2777	0.7487	0.9539	0.4514
hw	0.7051	0.6416	0.7051	0.6416
a	28.7251	28.4035	28.9527	28.9867

7 COMPARISON

The comparisons of present evaluated data with the experimental data are shown in Figs. 1~ 2, and the comparisons among present evaluation (CENDL–2) and ENDF / B–6, JENDL–3 and BROND–2 are shown in Figs. 3~10.

ACKNOWLEDGMENTS

The authors wish to thank Prof. Shi Zhaomin for his help in the recommended data of $\sigma_{n,n}$ and $\sigma_{n,n}(\theta)$, we also thank Dr. Yu Hongwei for his help in the theoretical calculation.

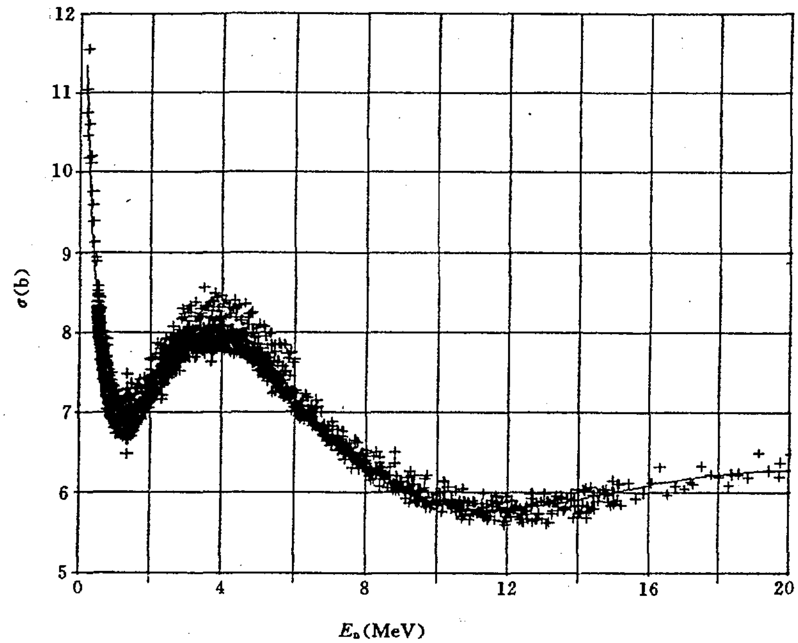

Fig. 1 Comparison of evaluated and experimental data for σ_{tot}

——— Present evaluation + Expt. data

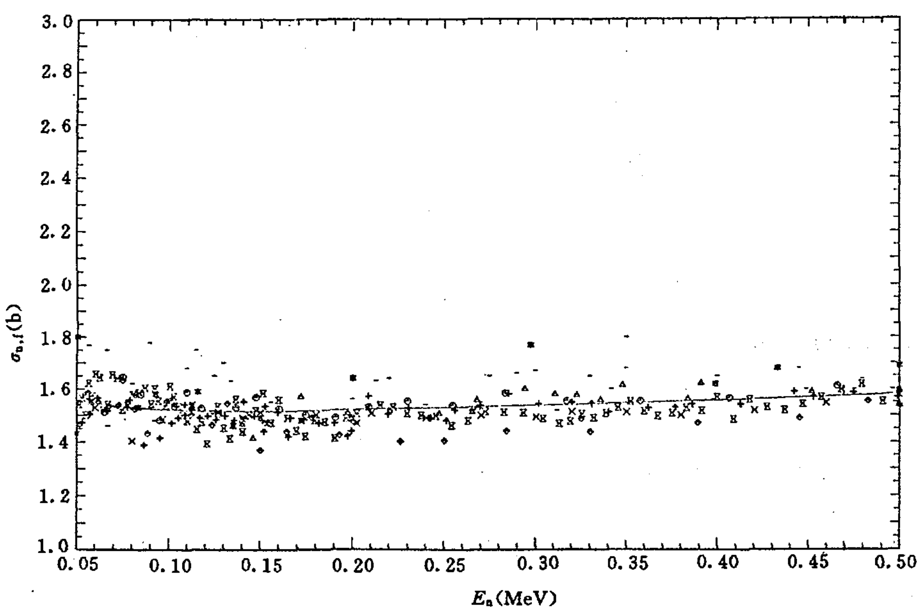

Fig. 2a Comparison of evaluated and experimental data for $\sigma_{n,f}$

——— Present evaluation Expt. data Refs. [14, 16, 17, 19, 20, 22]

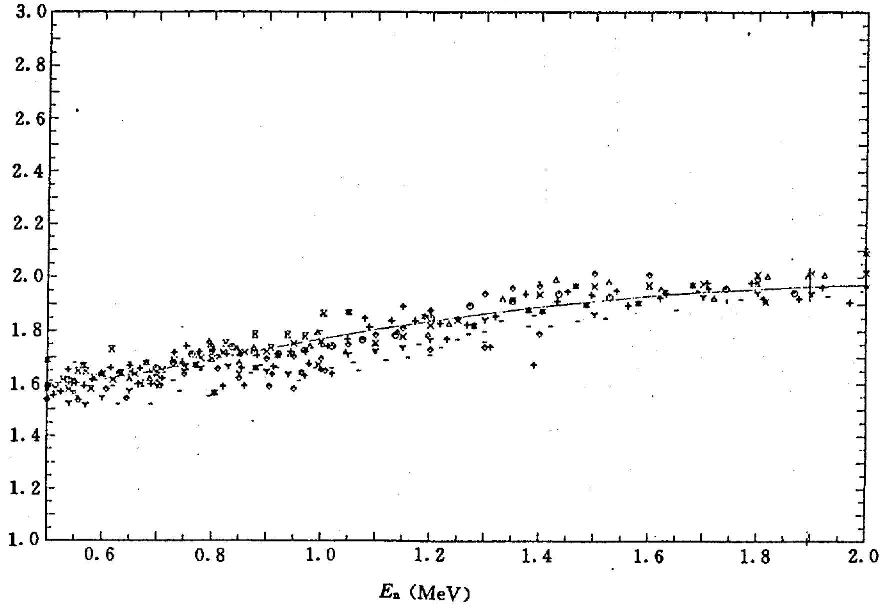

Fig. 2b Comparison of evaluated and experimental data for $\sigma_{n,f}$

——— Present evaluation Expt. data Refs. [14~23]

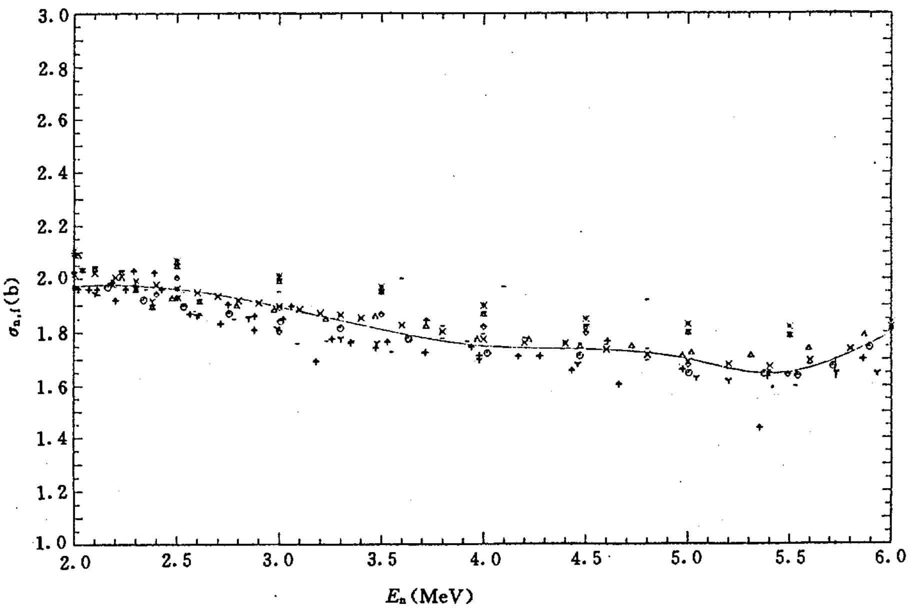

Fig. 2c Comparison of evaluated and experimental data for $\sigma_{n,f}$

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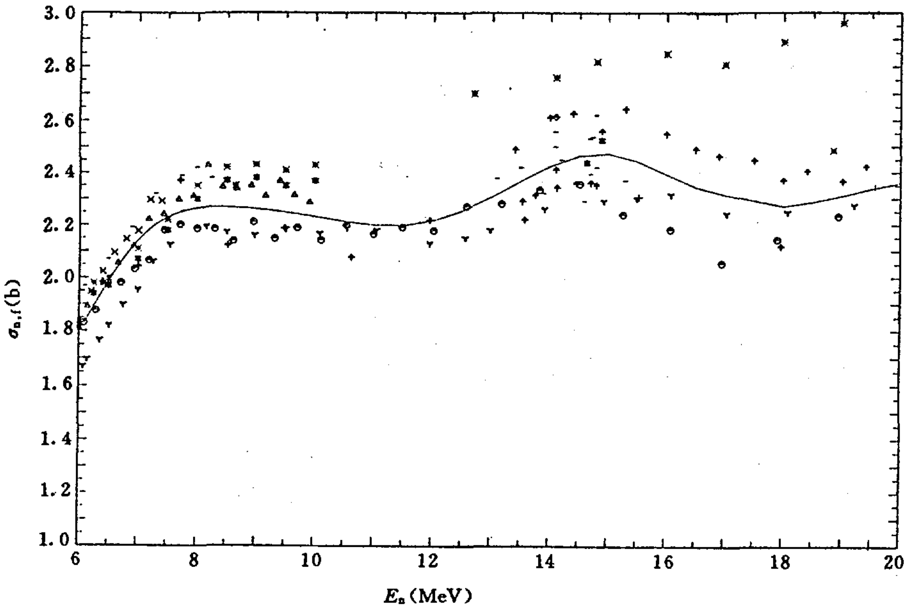

Fig. 2d Comparison of evaluated and experimental data for $\sigma_{n,f}$

——— Present evaluation Expt. data Refs. [15, 16, 19~22, 24, 25]

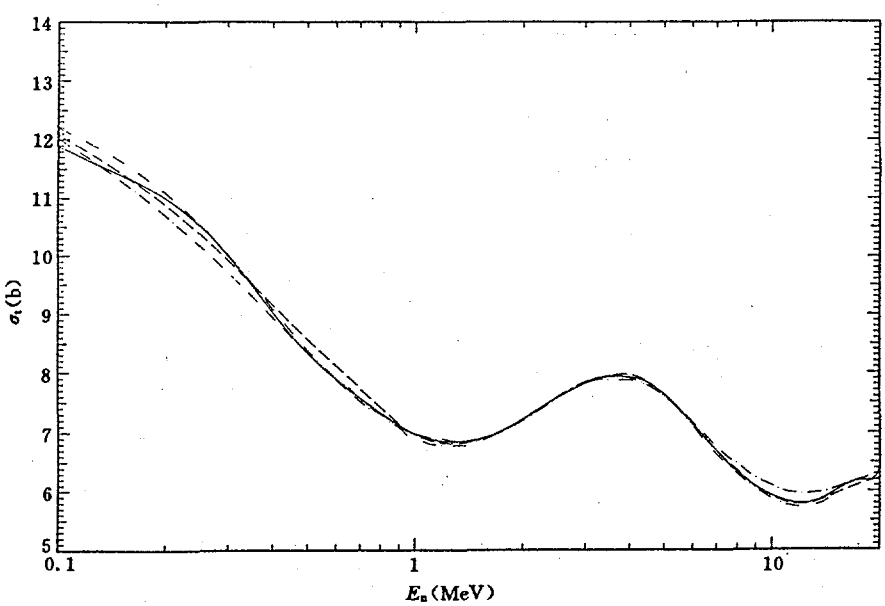

Fig. 3 Comparison of present evaluation with thoses of other libraries

——— CENDL–2, – – – ENDF / B–6, —·— JENDL–3, —· ·—BROND

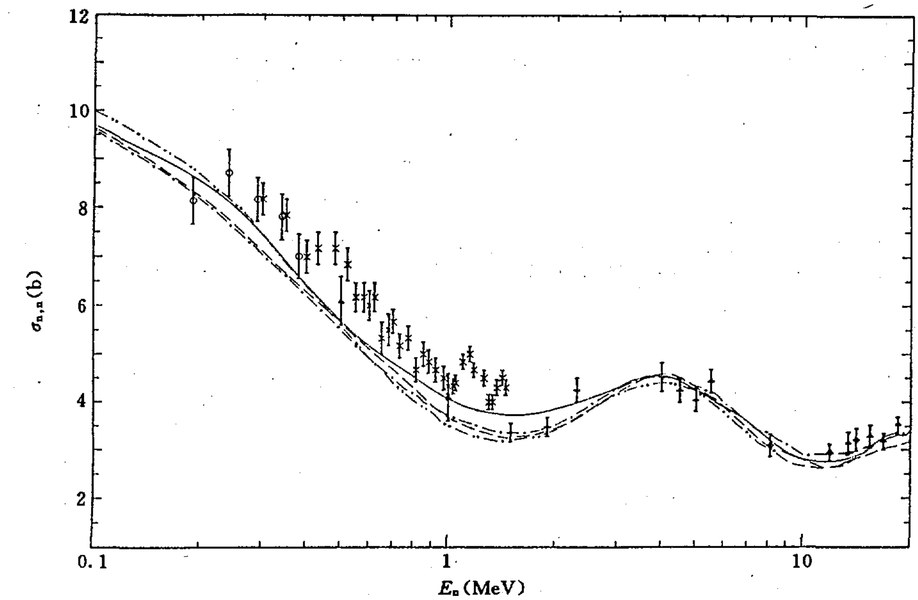

Fig. 4 Comparison of experimental data and CENDL–2, and others

▲ [3], ◇ [4], △ [5], ○ [6], + [7], ⁑ [8],

——— CENDL–2, – – – ENDF / B–6, —·— JENDL–3, —· ·— BROND

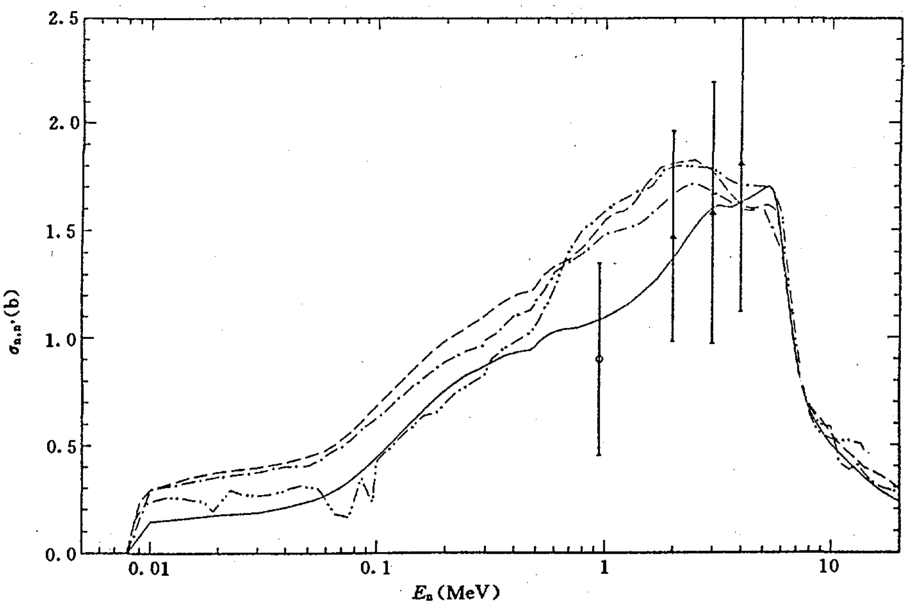

Fig. 5 Comparison of experimental data and present evaluation and others

○ [9], ▲ [10], —— Present evaluation, – – ENDF / B–6, —· JENDL–3, —· ·—BROND

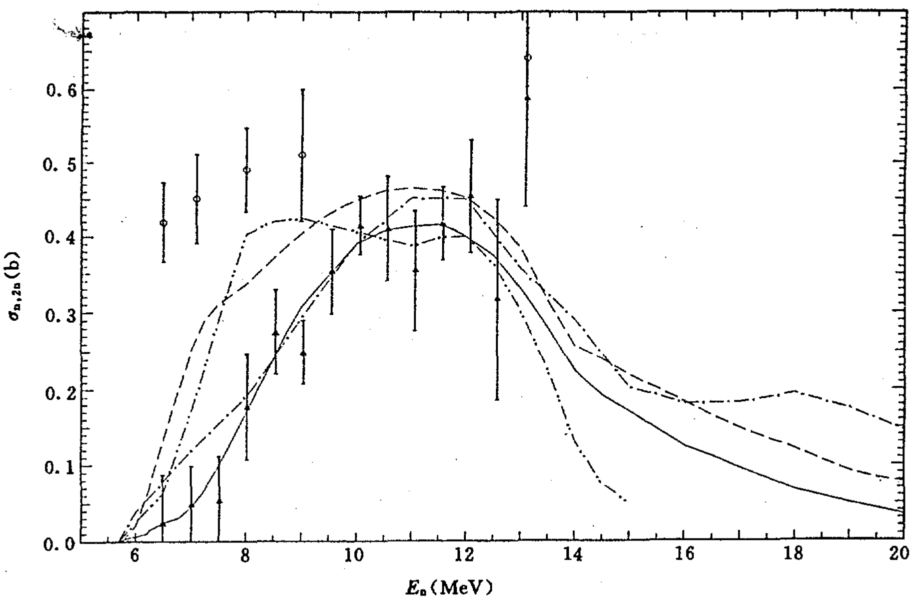

Fig. 6 Comparison of experimental data and CENDL–2, and others

▲ [11], ○ [12], —— CENDL–2, – – ENDF / B–6, —· JENDL–3, —· ·–BROND

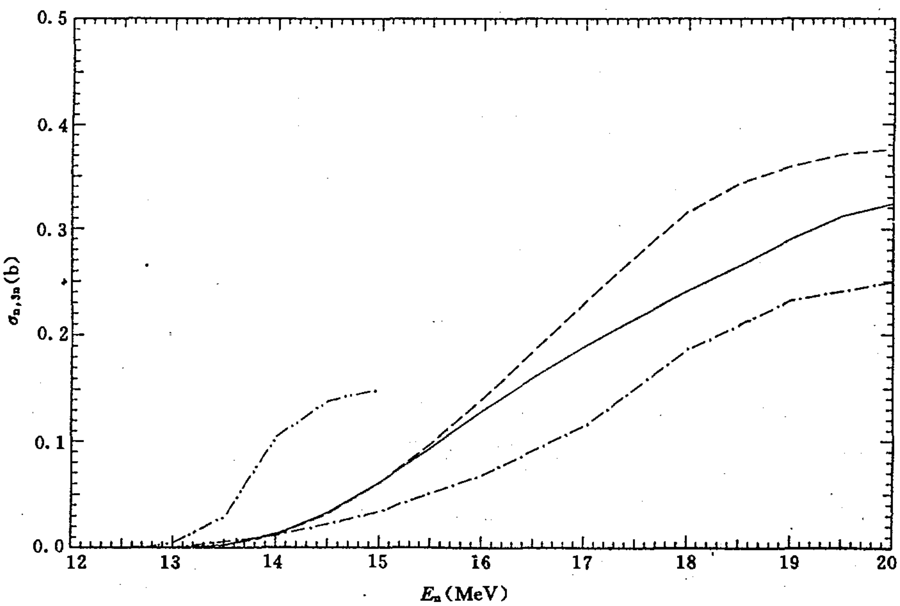

Fig. 7 Comparison of present evaluation with those of other libraries

—— CENDL–2, – – – ENDF / B–6, —·— JENDL–3, —· ·—BROND

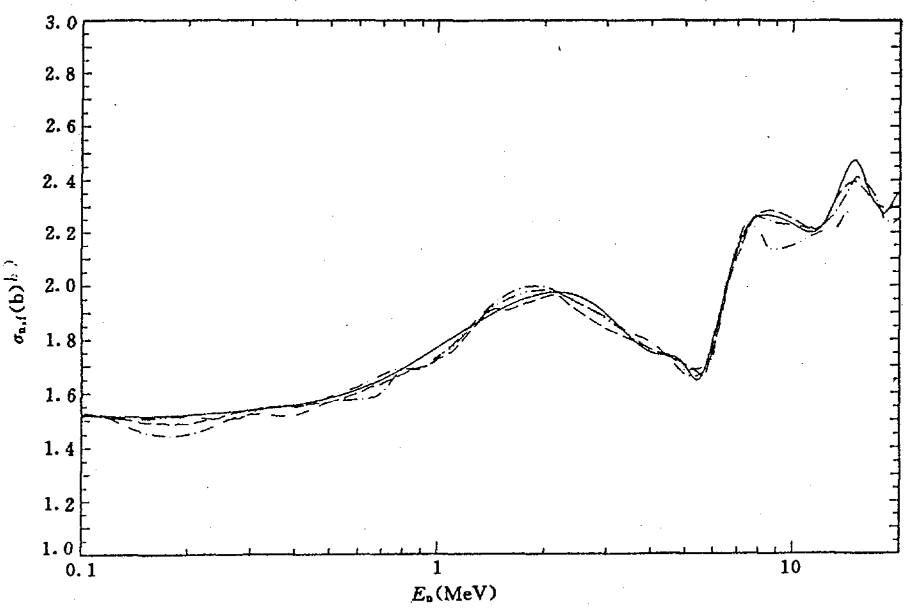

Fig. 8 Comparison of present evaluation with those of other libraries

——— CENDL–2, - - - ENDF / B–6, —·— JENDL–3, —· ·—BROND

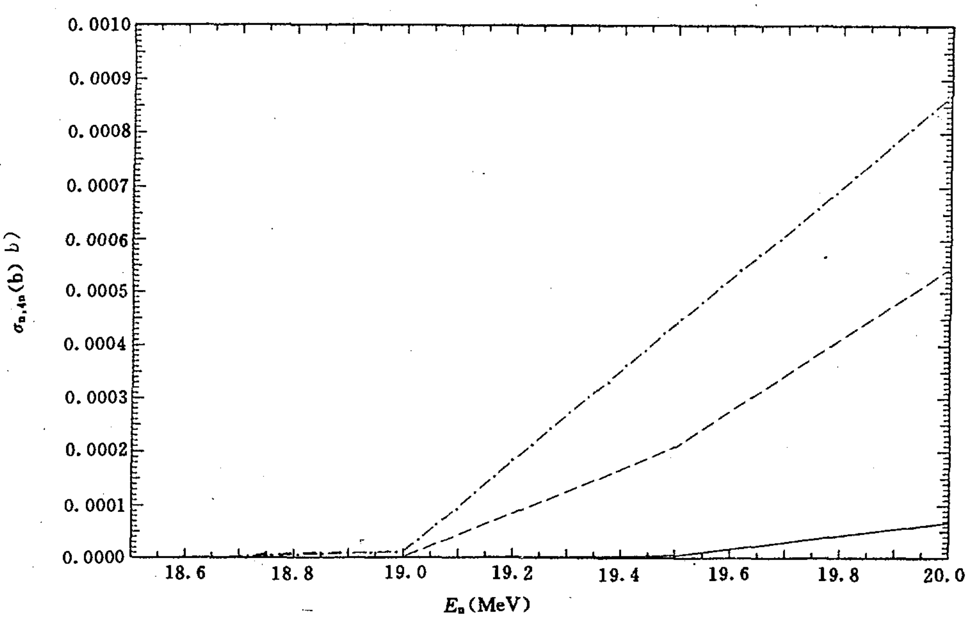

Fig. 9 Comparison of present evaluation with those of other libraries

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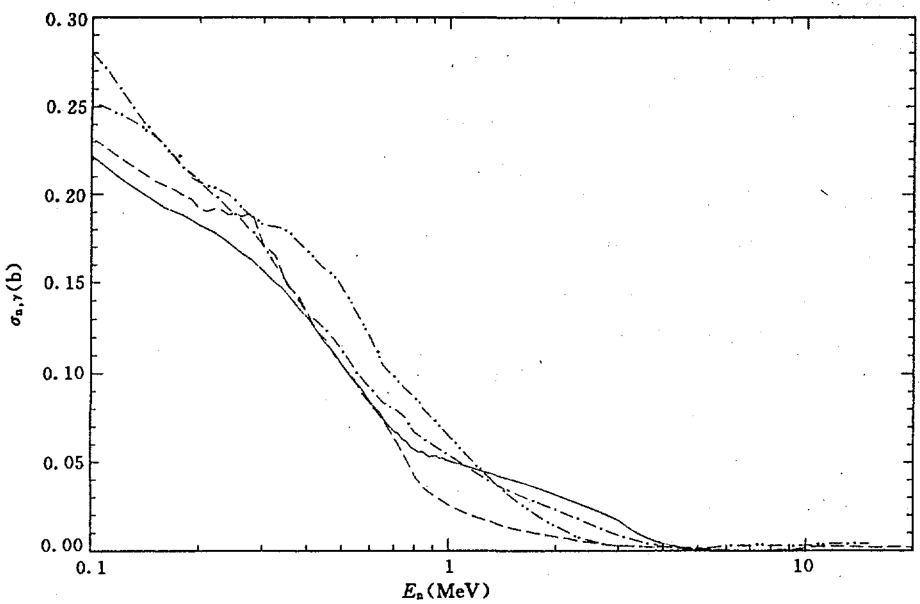

Fig. 10 Comparison of present evaluation with those of other libraries
—— CENDL–2, - - - ENDF / B–6, —·— JENDL–3, —· ·—BROND

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EVALUATION OF NEUTRON NUCLEAR DATA FOR ^{240}Pu

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ABSTRACT

Based on the experimental data evaluation and theoretical calculation, a complete set of neutron nuclear data for ^{240}Pu has been recommended in the in-

cident neutron energy range from 10^{-5} eV to 20 MeV for CENDL–2. The comparison of present evaluation with ENDF / B–6 and JENDL–3 has been carried out.

INTRODUCTION

^{240}Pu is an important nuclide in the fuel circle of nuclear reactor, particularly in context of fast breeder reactor (FBR) systems. Due to high spontaneous fission rate ($T_{1/2} = 1.32 \times 10^{11}$y) and intense α,γ radioactivities of ^{240}Pu, the experimental data for ^{240}Pu are scarce. The present evaluation work is based on CENDL–1[1], ENDF / B–6[2], recent experimental data and theoretical calculations.

(a) The experimental data on ^{240}Pu were collected up to 1991 , and measured data of σ_F and $\bar{\nu}_p$ were normalized to newly standard values (i. e. ENDF / B–6 file). Also the covariance data on $\bar{\nu}_p$, σ_T, σ_F were given.

(b) The resonance region data, γ–ray production data, fission product yield and radioactive decay data were taken from ENDF / B–6.

(c) Except items (a) and (b), all other data were theoretically calculated by FUP1 and CFUP1 codes.

The data were adjusted to make consistent in physics. The complete neutron data of ^{240}Pu were recommended in the incident neutron energy 10^{-5} eV ~ 20 MeV for CENDL–2. This evaluation was compared with ENDF / B–6 and JENDL–3.

1 THE EVALUATION OF THE EXPERIMENTAL DATA

1.1 Average Number of Neutrons Per Fission[3] – $\bar{\nu}$

1.1.1 Average Number of Prompt Neutrons Per Fission–$\bar{\nu}_p$

The experimental data on $\bar{\nu}_p$ (^{240}Pu) were collected as complete as possible, they were measured relatively to the $\bar{\nu}_p$ of ^{252}Cf spontaneous fission and based on the previous $\bar{\nu}_p^{sp}$ (^{252}Cf) values. In the evaluation on $\bar{\nu}_p$ (^{240}Pu) by Weston for ENDF / B–6[2] the older standard value $\bar{\nu}_p^{sp}$ (^{252}Cf) = 3.741

was used. Therefore, we adopted $\bar{\nu}_t^{sp}$ (^{252}Cf) new standard value 3.768 for ENDF / B–6[4], but all measured data were renormalized to $\bar{\nu}_p^{sp}$ (^{252}Cf) = 3.759, and then the experimental data of Frehaut et al.[5] were fitted by least square method and extended to 20 MeV with theoretically calculated data[6]. Finally, $\bar{\nu}_p$ (^{240}Pu) = 2.8135 + 0.1519E_n ($0 < E_n < 20$ MeV) was recommended and shown in Fig. 1.

The $\bar{\nu}_p^{sp}$ (^{252}Cf) standard value 3.759 was obtained as follows. The measured results on nubar for the spontaneous fission of ^{252}Cf using liquid scintillator, manganese bath and boron pile methods were reviewed by J. W. Boldeman et al.[7] and M. V. Blinov et al.[8] respectively. And the weighted mean values of these measured data, $\bar{\nu}^{sp}$ (^{252}Cf) = 3.766 ± 0.005 and 3.7661 ± 0.0054, were recommended by M. V. Blinov et al.[8] and E. J. Axton[9], respectively. The $\bar{\nu}_t^{sp}$ (^{252}Cf) = 3.768 ± 0.005 was recommended as "Neutron Standard Data" for ENDF / B–6 by R. W. Peelle et al.[4]. Based on S. Cox's measured data[10] $\bar{\nu}_d^{sp}$ (^{252}Cf) = 0.0086 ± 0.0010, the $\bar{\nu}_d^{sp}$ (^{252}Cf) = 0.009 ± 0.001 was recommended. Therefore, $\bar{\nu}_p^{sp}$ (^{252}Cf) = $\bar{\nu}_t^{sp} - \bar{\nu}_d^{sp}$ = 3.759.

1.1.2 Average Number of Delayed Neutrons Per Fission–$\bar{\nu}_d$ (^{240}Pu)

Based on $\bar{\nu}_d$ (^{240}Pu) values measured by G. R. Keepin et al.[11] and C. F. Masters et al.[12] in the incident energies 2.13, 14.1, 14.9 MeV and recommended by R. J. Tuttle[13] for fast and thermal neutrons, considering the energy–dependent of the values of delayed–neutron yield for ^{233,235}U(n,f) and $^{239\sim242}$Pu (n,f) measured and analysed by M. S. Krick et al.[14] and S. A. Cox et al.[15], the data set was recommended for ENDF / B–6 by T. R. England et al.[16] :

E_n, MeV	$\bar{\nu}_d$ (^{240}Pu)	
0~4	0.0090	Constant
4~7	0.0090~0.00615	Linear Extrapolation
7~20	0.00615	Constant

1.1.3 Average Number of Total Neutrons per Fission $\bar{\nu}_t$

From the $\bar{\nu}_p$, $\bar{\nu}_d$ obtained above, $\bar{\nu}_t = \bar{\nu}_p + \bar{\nu}_d$ was recommended.

1.2 Total Cross Section

There are about ten sets of measured data, which were collected and analysed. Among them three data sets of A. B. Smith et al.[17] and W. P. Poenitz et al.[18] in the incident energies 0.116~ 1.467, 0.048~ 4.807 and 1.818~ 20.91 MeV were used and fitted with orthogonal Polynomial method. The model calculated result by FUP1 code is in good agreement with the fit value. The comparison of the fit data with measured data and other evaluations is shown in Fig. 2.

1.3 Fission Cross Section[19]

About thirty sets of measured data up to 1991 were collected and analysed. The following measurements were given larger weight :

Year	Author	Lab.	E_n, MeV	Standard	Accelerator	Reference
1978, 83	J. W. Behrens+	LLL	0.006~34	^{235}U	100 MV LINAC	[20]
83, 84	L. W. Weston+	ORNL	0.005~21	^{239}Pu	ORELA	[21]
81	C. Budtz–	GEL	0.01~0.3	^{235}U	GELINA	[22]
	Jφrgensen+		0.2~9.75		7 MV VDG	
78	K. Kari+	KFK	1~21.2	H(n,p)	42 MeV Cyclotron	[23]
76	S. Cierjacks+		0.49~21.2	^{235}U		
81	J. W. Meadows	ANL	0.335~9.597	^{235}U	8 MV Dynamitron	[24]
					Tantem	
90	T. Iwasaki+	TOH	0.669~6.587	^{235}U	4.5 MV Dynamitron	[25]
79	Y. M. Kuprijanov+	IPPE	0.127~7.4	^{235}U	5 MV VDG	[26]

From these measurements, some results were obtained :

(a) These measurements were performed relative to ^{235}U and the ratio values — R_f (^{240}Pu / ^{235}U) were given. From the R_f $(E_n) \sim E_n$ curve which measured using the white neutron source (on the LINAC and Cyclotron) [20~23] and monoenergetic neutron source (on the VDG)[22, 24~26], the systematic discrepancy has not been found. These measured ratios were then fitted with the least square method and shown in Fig. 3.

(b) According to Weston's notes[2] : " The ENDF / B–5 evaluation of $^{235}U(n,f)$ cross section was used to convert the evaluated ratios to $^{240}Pu(n,f)$ cross section. If there are appreciable changes in the $^{235}U(n,f)$ cross section standard for ENDF / B–6, some revision of the present evaluation of ^{240}Pu may be necessary", and the $^{235}U(n,f)$ cross section for ENDF / B–6 standard have apparent changes compared to ENDF / B–5 Version[27, 28] , in our evaluation, the fission cross section of ^{235}U for ENDF / B–6[28] was used to derive the fission cross section of ^{240}Pu from the fitted ratios, should be more confidence.

(c) Kari's data[23] was also measured directly by the absolute technique at 42 MeV Cyclotron. The fitted values of Kari's data are in agreement with derived values from R_f(^{240}Pu / ^{235}U) within the experimental errors as mentioned above, and given in Fig. 4. Because the intensity of high energy neutrons on the KFK–Cyclotron is higher than that on the LINAC in MeV neutron energy region and the H(n,p) cross section is a fine standard cross section, Kari's data is more reliable than others. Therefore, in present evaluation the fit values of Kari's data was adopted instead of deriving $^{240}Pu(n,f)$ cross sections from fitted ratios[20~ 26] above 7 MeV; Below 0.1 MeV, the data sets of C. Budth–Jφrgensen et al.[22] and L. W. Weston et al.[21], which are in excellent agreement with each other, were used to get the average fission cross section and were normalized to the value (at 100 keV) of evaluated $^{235}U(n,f)$ cross section of ENDF / B–6. The model calculated results by FUP1 are in agreement with the present evaluation within experimental error. The comparison of the present evaluation with experimental data and the other evaluations are shown in Fig. 4 and Fig. 5.

2 THEORETICAL CALCULATION

2.1 Models and Procedures

FUP1[29], the first version of a unified program for calculating the fast neutron data (10 keV ~ 20 MeV) for fissile nuclide, is designed on the basis of MUP2[30] code which is the second version of a unified program for theoretical calculation of fast neutron data for medium–heavy nuclide by using optical model, Hauser–Feshbach theory with width fluctuation correction (WHF) and pre–equilibrium statistical theory (PE) based on exciton model. FUP1 is mainly used to calculate the following data : the total, elastic scattering, nonelastic, inelastic cross sections; (n,γ), (n,2n), (n,3n) and (n,4n) cross sections; total, (n,f), (n,n'f), (n,2nf) and (n,3nf) fission cross sections; elastic scattering angular distribution and inelastic scattering angular distribution to discrete level; secondary neutron spectra for (n,n′), (n,2n), (n,3n) and (n,4n) reactions; fission neutron (including emitting neutrons before fission) spectra for total and (n,f), (n,n′f), (n,2nf), (n,3nf) fission.

The direct reaction components of the inelastic scattering to 3 discrete levels calculated by using Coupled Channel Optical Model (CCOM) are placed as input data. These direct reaction components are added to the calculated compound nucleus cross section and angular distribution.

The optical potentials used in the calculation are Woods–Saxon form for the real part and the volume absorption imaginary part, derivative Woods–Saxon form for the surface absorption imaginary part, and Thomas form for the L–S coupling part.

With considering that the contribution from PE can′t be neglected in higher excitation energy, in principle, we use WHF in lower energy region ($E_n < 3$ MeV), and use PE in higher energy region ($E_n > 3$ MeV). The γ – ray widths are calculated based on the double peaks′ giant dipole resonance model. The fission widths used in WHF and the fission partial width used in PE are calculated based on the equivalent single fission barrier model.

CFUP1 is an extend code of FUP1 and especially used for calculating charged–particle emission cross sections ($E_n < 30$ MeV).

2.2 Determination of Parameters

The optical parameters used in this calculation were taken from Shen Qingbiao et al.[31].

The level density parameter a, the pair energy correction Δ, the fission barrier and level density on saddle states V_f, K_1, K_2, $h\omega$ and E_{psd}[29] were adjusted with ASFP program based on the experimental data of compound nucleus reactions. They are given as follows :

The fission parameters for ^{240}Pu
(for PE)

	A+1	*A*	*A*−1	*A*−2	*A*−3
V_f	5.3946	6.185	5.553	5.472	
K_1	1.96	1.95	1.009	1.07	
K_2	0.0806	0.623	0.345	0.0347	
Δ	0.749	1.278	0.749	0.950	0.45
a	28.7369	28.725	28.4	28.98	28.987
hω	0.642	0.795	0.652	0.795	

(for WHF)

V_f: 5.796 K_1: 3.5958 K_2: 0.115 E_{pad}: 0.714886

The Kalbach's constant *k*, which represents a magnitude of the residual two–body interaction in the precompound model, was deduced to 400 MeV[3] through a comparison of the experimental data with calculations.

3 FILE COMMENTS

MF = 1 General information

MT = 452 Total nubar. Sum of MT = 455 and 456;
MT = 455 Delayed neutron yields. From Cai[3] and 83England[16];
MT = 456 Prompt neutron yields. From Cai et al.[3];
MT = 458 Energy release / fission. From ENDF / B–6[2].

MF = 2 Resonance parameters

MT = 151 Resolved and unresolved resonance parameters were taken from ENDF / B–6[2].

MF = 3 Neutron cross sections

MT = 1 Total cross section

Fit the experimental data of 72 Smith[17] and 81, 83 Poenitz[18] from 0.04 to 20 MeV; from ENDF / B–6[2] below 0.1 MeV.

MT = 2 Elastic scattering cross section

Obtained by subtracting the other cross sections from total. The deduced values accord with experimental data of Smith[32]. See Fig. 6.

MT = 4 Total inelastic scattering cross section

Sum of partial inelastic scattering cross sections (MT = 51~91).

MT = 51~68, 91 Partial inelastic scattering cross sections

These data were calculated by FUP1 code. Below 3 MeV, the results of CCOM and WHF statistical calculation were adopted.

MT = 16, 17 (n,2n), (n,3n) reaction cross sections

From neutron emission cross section and branching ratio to each reaction channel calculated with FUP1.

MT = 18 Fission cross section

Taken from Cai's evaluation[19].

MT = 102 Capture cross section

Calculated with FUP1 and agreed with Weston's experimental data[33]. See Fig. 7.

MT = 103~107 (n,p), (n,d), (n,t), (n,^{3}He) and (n,α) reaction cross sections

Calculated with CFUP1 code.

MF = 4 Angular distributions of secondary neutrons

MT = 2, 51~68 Angular distributions of elastic and inelastic scattering to discrete levels calculated by using FUP1.

MT = 16~18, 91 Angular distributions of (n,2n), (n,3n), fission, and inelastic scattering (continuum part) reactions were assumed isotropic in the Center of Mass System.

MF = 5 Energy distributions of secondary neutrons

MT = 16~18, 91 Calculated with FUP1 code.

MT = 455 Delayed neutron spectra. Taken from ENDF / B–6[2]

MF = 12~15 Photon production multiplicities, photon production cross sections, photon angular distributions and photon energy spectra.

Taken from ENDF / B–6[2].

MF = 32 MT = 151 Resonance parameter covariance file.
Taken from ENDF / B–6[2].

MF = 33 Cross section covariance

MT = 1, 18 The covariance data for σ_T and σ_F were evaluated based on the uncertainties of measured data by using least square method.

MT = 102 Taken from ENDF / B–6[2].

ACKNOWLEDGMENTS

The authors would like to thank Drs. Ye Zongyuan, Meng Jiangchen, Zhuang Youxiang, Liang Qichang, Liu Tingjin, Zhou Delin, Zhao Zhixiang et al., for thier great help and discussion, during the course of this work.

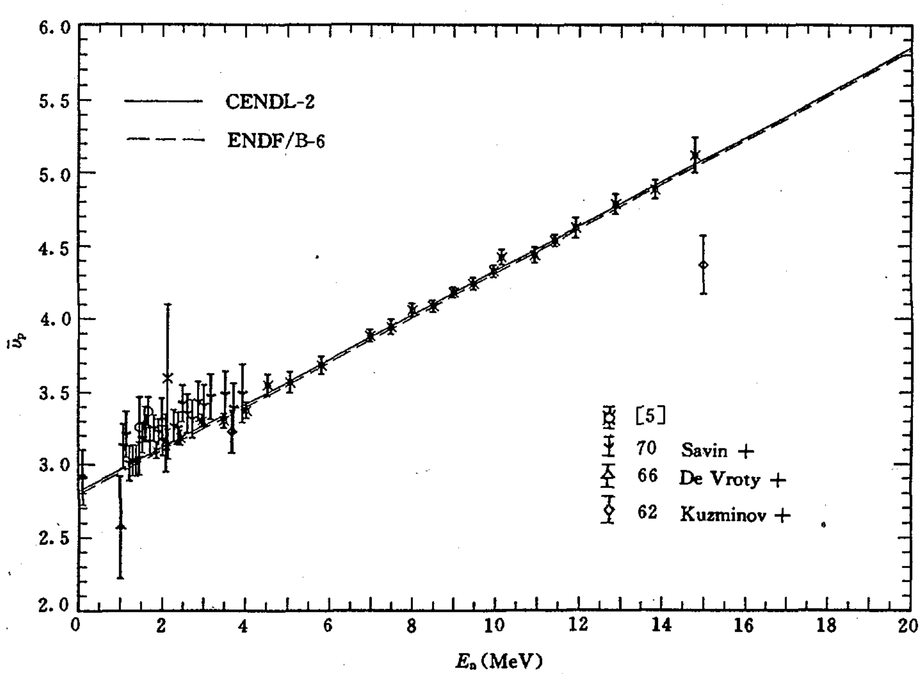

Fig. 1 Measured and evaluated $\bar{\nu}_p$ (^{240}Pu)

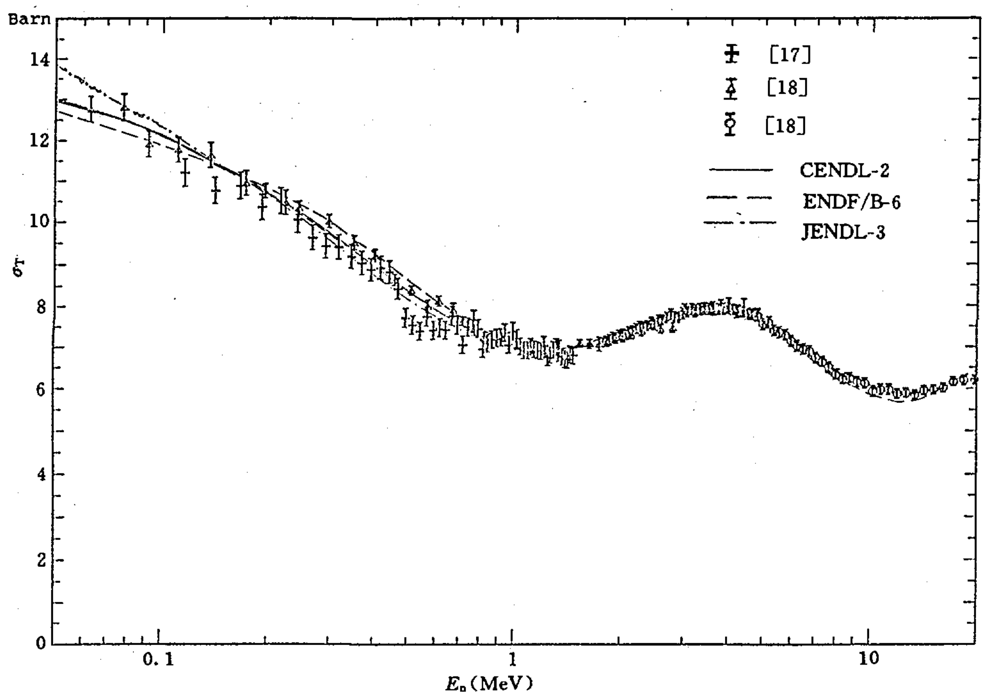

Fig. 2 Comparison of σ_T among evaluated and measured data

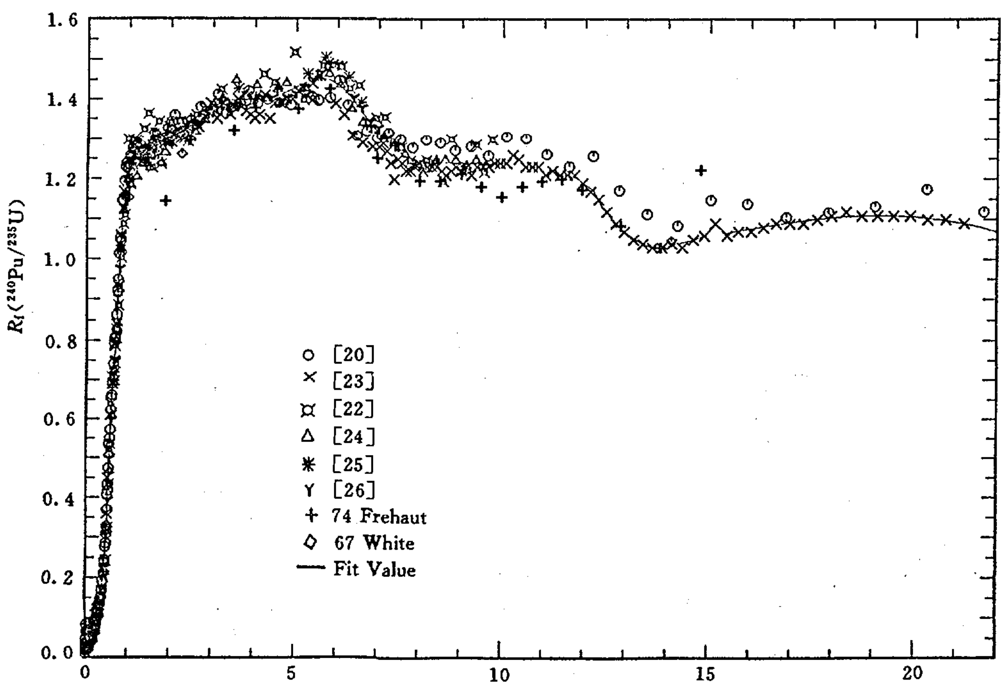

Fig. 3 The fit value of measured ratios for ^{240}Pu fission

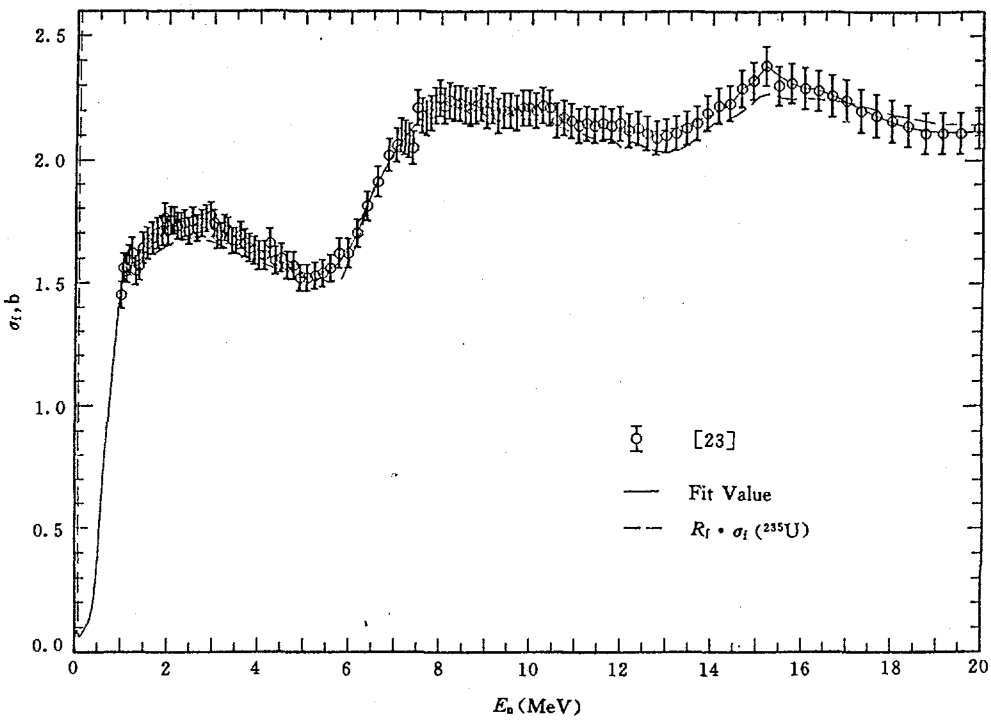

Fig. 4 The fitting value of ^{240}Pu(n,f) cross sections measured by Kari et al.

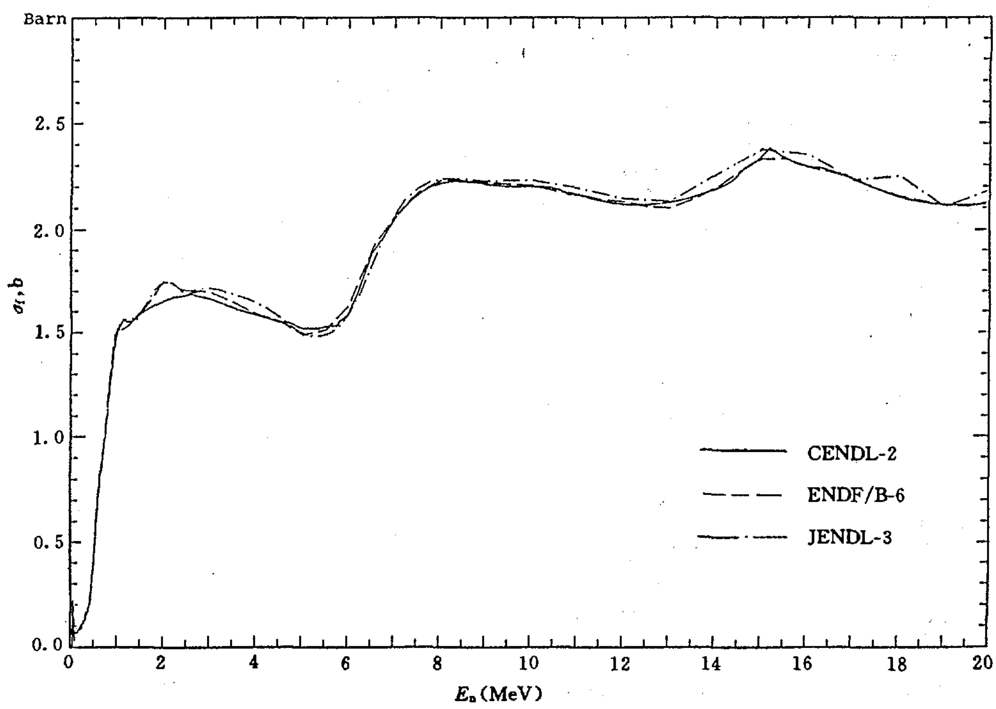

Fig. 5 Comparison of σ_f among CENDL−2, ENDF / B−6 and JENDL−3

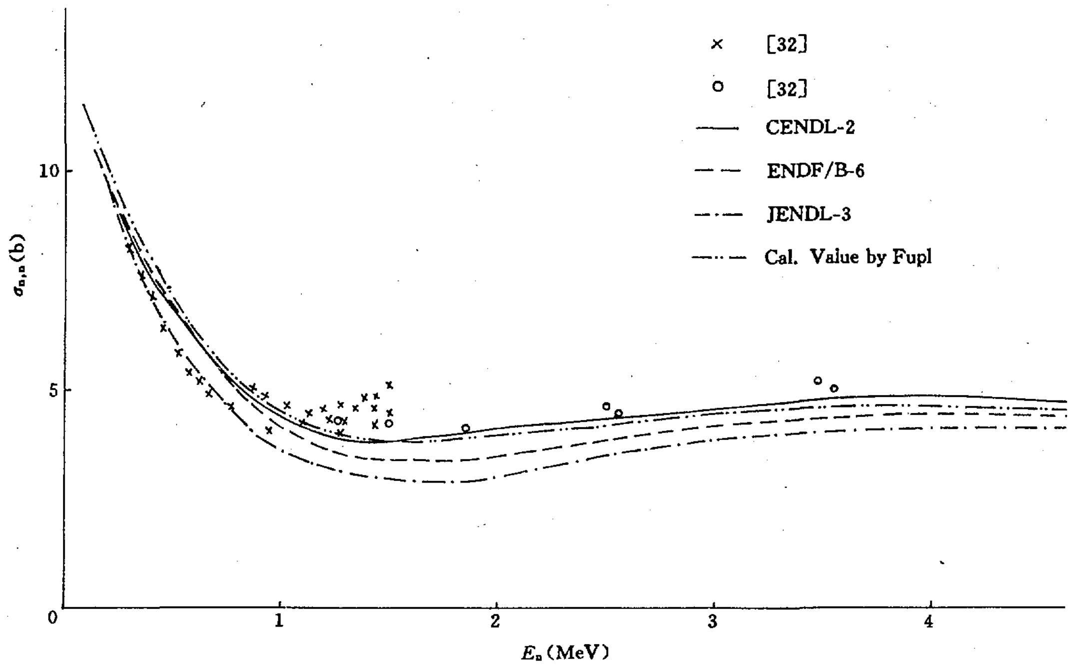

Fig. 6 Measured and evaluated ^{240}Pu(n,n) cross section

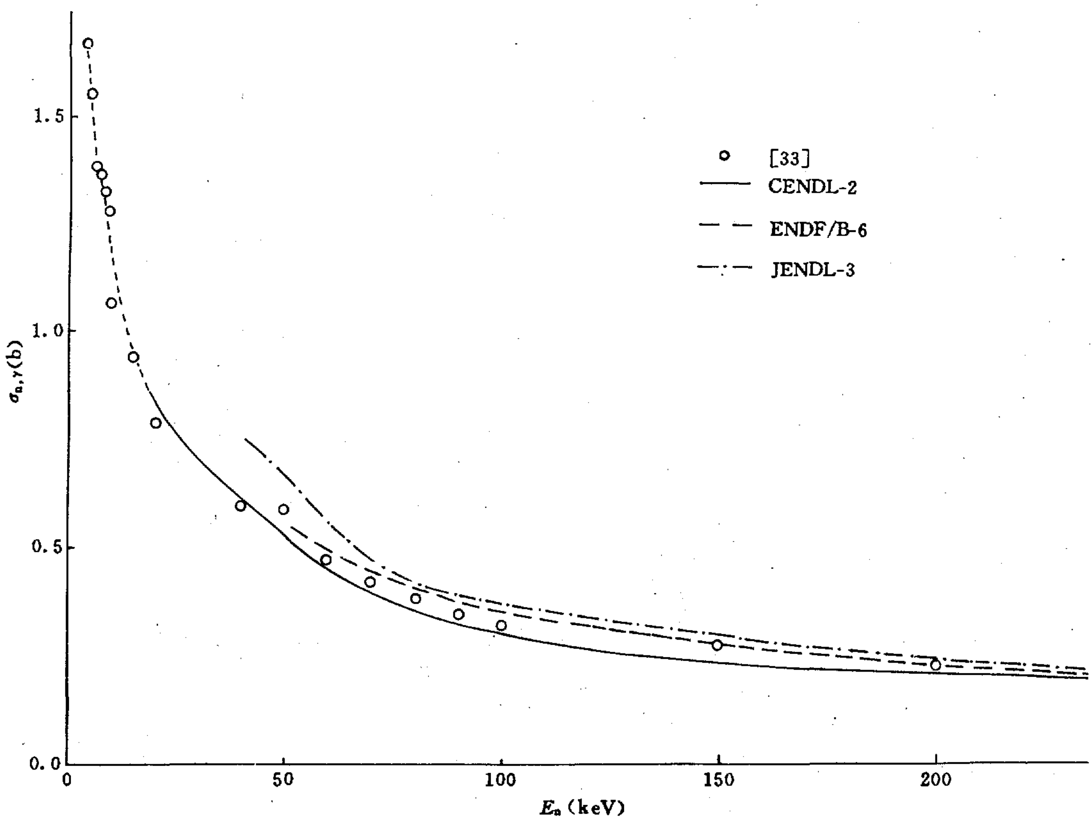

Fig. 7 Comparison of calculated ^{240}Pu(n,γ) cross section by FUP1 with experimental data

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EVALUATION OF NEUTRON NUCLEAR DATA FOR ^{241}Am

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ABSTRACT

A complet set of neutron nuclear data for ^{241}Am has been evaluated from 10^{-5} eV to 20 MeV. The numerical data are available in ENDF / B–6 format.

INTRODUCTION

The Americium isotopes play an important role in reactor production chain of the heavy actinides and the ^{241}Am is the first in the chain of transplutonium elements. Especially, of most importance is the capture process for ^{241}Am which will partially determine the ^{242}Cm (163d, α, SF) and ^{244}Cm (18y, α, SF) production. The higher production of ^{242}Cm means more severe fuel handling problem because of its relatively short half–life and its contribution to the decay heat, its spontaneous fission and α particles emission that will produce neutrons via (α,n) reactions on the fuel oxide and carbide. By α

decay, ^{242}Am forms ^{238}Pu, an interesting substance in nuclear technology applications. The other product of $^{241}Am(n,\gamma)$ reaction is ^{242m}Am which has a large absorption cross section of 7200 barns and ^{244}Cm can be produced. That is unwanted in a reactor. So the resultant path of ^{241}Am capture will be different depending on whether the ground state or isomeric state is formed.

The quantities which we have evaluated are: reaction cross sections at thermal energy; resolved and unresolved resonanse parameters; total, elastic and inelastic scattering, fission, capture, (n,2n), (n,3n) reaction cross sections; the angular and energy distribution of neutron emission; the average number of neutron emission per fission and the covariances of these data. For the reasons mentioned above, we have paid much effort to the evaluations of the (n,γ) reaction cross section and its isomeric ratio.

1 THERMAL CROSS SECTION[1]

1.1 Thermal Total Cross Section

Two transmission measurements extented to thermal energy region. One by Kalebin[2], an interpolation of 640 ± 20 b is obtained for thermal total cross section. Another older one of 600 b[3] with large uncertainty is only available as a graph. Considering the thermal capture cross section evaluation mentioned below, the Kalebin's value of 640 b was adopted in this evaluation.

1.2 Thermal Fission Cross Section

Previously evaluated value of 3.15 ± 0.1 b by Goel[4] is adopted for this evaluation for the small fission cross section. This value is supported by the newly measured one of 3.05 b by Dabbs[5].

1.3 Thermal Capture Cross Section

The thermal capture cross section is also important for the evaluation of (n,γ) reaction cross section in other energy region, since some important available (n,γ) measurements for ^{241}Am in the resonance and smooth energy region were normalized to thermal value.

The total thermal capture cross sections (the isomeric ratio at thermal energy will be described in paragraph 4. 1. 2) were obtained from several ways:

Almost all old measurements of (n,γ) reaction cross section were carried out in different reactor spectra with different Cd–cutoff energy. And what we called thermal capture cross sections were derived from the integral measurements. For ^{241}Am, the presence of the strong resonances at 0.308 and 0.576 eV makes these effective thermal cross sections meaningless and largely discrepant (some measured values even larger than total cross section) in the resulting values, as showing in Table 1, if the spectra shapes were not considered in detail. Fortunately, a careful analysis was made and it pointed out that only three of the integral measurements were carried out in well defined spectra to enable 2200 m / s values to be deduced. The first three listed in Table 1 are the values deduced from such three integral measurements[6, 7].

Information on thermal capture cross section can also be derived from transmission measurements. As mentioned above on the thermal total cross section, there are two such measurements extended to thermal energy region[2, 3]. Two values of 640 b and 600 b at 0.0253 eV are obtained respectively based on these measurements. By subtracting a potential scattering cross section and a small fission cross section, 625 b and 585 b are obtained respectively.

The sole relative differential absorption (the sum of capture and fission) measurement by Weston and Todd[8] was normalized to 582 b at 0.0253 eV. And the $2g\Gamma_n$ values of the first several resonances then obtained for this measurement are in excellent agreement with those of the transmission measurement by Derrien and Lucas[9] in the energy region from 0.8 eV to 1 keV. (see Fig. 1) Obviously, the Weston and Todd's measurement may also be considered to normalize to Derrien and Lucas's measurement at first several resonances equivalently, and a thermal value of 582 b is obtained. As the description of Weston and Todd, a correlated error of ~ 7 % existed above 0.2 eV neutron energy and these error must be added to the thermal value, since the data below and above 0.2 eV are weakly correlated (only 1 % systematic error for the data below 0.2 eV).

Indirectly, the isomeric ratio measurement by Wisshak et al.[10] gave a ratio of σ_γ ($^{241}Am \rightarrow {}^{242g}Am$) / σ_γ (Au) = 5.79 ± 0.33. From this value one can get a total capture cross section of 639 ± 37 b at 0.0253 eV in case an isomeric ratio of 0.9 ± 0.01 and a thermal capture cross section of 99.18 b for Au are supposed.

All the thermal capture cross sections mentioned above are listed in table 1. These measurements may be considered independent ones, supposing the uncertainties of the transmission measurement (the first several resonances at least) of Derrien and Lucas for normalization are negligible compared with those of capture or absorption measurements. The weighted averages are obtained as following:

619 ± 13 b (all measurements listed in Table 1 are adopted)
622 ± 14 b (the measurements of No. 1 ~ 5, 7 are adopted)
626 ± 14 b (the measurements of No. 1 ~ 3, 5, 7 are adopted)

626 ± 14 b is adopted as the thermal capture cross section evaluation of this work. For comparison, all available evaluated values are listed in Table 2.

2 RESONANCE PARAMETERS

2.1 Resolved Resonance Parameters ($E_n < 150$ eV)

Three measurements covering the resolved resonance energy region and providing resonance parameters entilely or partly are available:

Derrien and Lucas[9] trasmission (1975), E_n = 0.8 eV ~ 1 keV
Kalebin et al.[2], trasmission (1976), E_n = 0.02 eV ~ 30 eV
Weston and Todd[8] absorption, (1976), E_n = 0.01 eV ~ 370 keV

And one measurement covering the resonance energy region but no resonance parameters were given:

Dabbs et al.[5], fission (1983), E_n = 0.002 eV ~ 20 MeV

The measurements by Gayther and Thomas[12] and by Knitter and Budtz–Jorgensen[13] covering much narrow energy range have not been taking into account in the present evaluation.

Since the γ width is quite large compared with the neutron and fission width, these measurements are equivalent for neutron width information. Only point–wise data were given for Dabbs' measurement, it can be used for checking the evaluated resolved and unresolved resonance parameters based on other measurements.

One may notice that the Γ_n values of the first four resonances of Derrien et al. and Kalebin et al.'s measurements respectively are in excellent agreement with each other (see Fig. 1). And the relative measurement of Weston and Todd may be considered to normalize to the first four resonances as mentioned before. In this case, the ~ 7 % systematic error of Weston and Todd's measurement should not be included in the uncertainties of its $2g\Gamma_n$ values.

Then the resonance parameters for the full resonance energy region can be combined by averaging weighted with random error only without taking the ~ 7 % systematic error in absorption measurement into account. So that the recommended values for most resonances are averaged in equal weight except those for which the random errors of Weston et al.'s measurement or / and the errors of Kalebin et al.'s measurement are much larger than Derrien et al.'s. In such cases the averages are made with unequal weights properly.

As well known, only using the resonance parameters at positive energies, it is not able to estimate correctly (underestimated more or less, in normal case) the cross sections at thermal and / or thermal energy region. For reproducing the measured cross sections in thermal energy region or at thermal energy at least, in the evaluations of old versions, a way of applying a $1/v$ background was usually adopted to adjust the discrepancy. Currently, this way was replaced by using negtive level(s). In the present evaluation, 5 negtive levels according to Lynn et al.[7] are assumed. Hence the resolved resonance parameter set located at positive and negtive energy region are used, and the parameters of negetive levels are adjusted to reproduce the measured point–wise cross sections at thermal energy and thermal energy region. Then the resultant parameter set can be used to calculate the cross sections of resonance and thermal energy regions extended down to 10^{-5} eV.

The evaluated resolved resonance parameters have been checked further via comparing the calculated cross sections from the evaluated resonance parameters with the measured point–wise cross sections provided by Derrion et al. for σ_t, Weston et al. for (n,γ) reaction, as well as by Dabbs et al. for (n,f) reaction. The Dabbs' measurement has a medium energy resolution, the comparisons are carried out in energy averaged value properly. The calculated cross sectons at some resonances are largely discrepant to the measured ones. For these resonances the parameters have been adjusted to fit the experimental data.

2.2 Unresolved Resonance Parameters (150 eV to 30 keV)

The averaged resonance parameters $\overline{\Gamma}_n^0$, $\overline{\Gamma}_f$, $\overline{\Gamma}_\gamma$ *and* $\overline{D}$ have been estimated from the resolved resonance parameters and used as input data and adjusted to fit the present evaluated capture cross sections and the measured fission cross sections of Dabbs et al. (averaged in proper energy range), then the energy dependent unresolved resonance parameters were obtained for the present evaluation.

3 REACTION CROSS SECTIONS ABOVE RESONANCE ENERGY REGION

3.1 Total Cross Section

Only one measurement by Phillips and Howe[14] is available in the energy range from 500 keV to 25 MeV. The optical potential parameters adjusted to reproduce these measured cross sections were used for optical model calculation of total cross section as well as shape elastic scattering, which adopted in the present evaluation. The model theory calculation will be descripted in more detail in paragraph 8.

3.2 Neutron Capture Data

3.2.1 Total Capture Cross Section

There are five new measurements on total capture or absorption cross section covering this energy region:

(1) Weston and Todd[8] absorption, E_n – 0.01 eV ~ 370 keV (1976).

In case it has been considered to normalize to the first four resonances of Derrien and Lucas's measurement, the ~ 7 % systematic uncertainty above 0.2 eV might be eliminated. A normalization error of 2 % was adopted as the correlated error of this measurement.

It is interesting to point out that if Weston and Todd's measurement is considered to normalize to 626 b at thermal energy, then more better agreement with other measurements will be obtained. So that some uncertainties in shape measurement, by Weston et al. and by others, should be taken into account in the uncertainty estimations of the combinated values for the smooth energy region.

(2) Wisshak et al.[10, 15] capture, E_n = 0.01 keV ~ 350 keV (1975).

Capture cross sections have been reported for the energy region of 0.01 ~ 250 keV in Ref. [15] and also given for 0.01 ~ 350 keV in Ref. [10]. The measurements were carried out relative to the capture cross sections of gold.

(3) Gayther and Thomas[12] capture, E_n = 0.01 keV ~ 500 keV (1977).

A relative measurement normalized to Weston and Todd's measurement at 1~2 keV.

(4) Vanpraet et al.[16] capture, E_n = 0.6 ~ 200 keV (1976).

A relative measurement normalized to Derrien and Lucas's measurement at 0.576 and 1.270 eV resonances. So a 2 % normalization error is adopted as the correlated error of this measurement.

(5) Derrien and Lucas[9] transmission, $E_n = 0.8$ eV ~ 1 keV (1975).

A total cross section measurement. The capture cross sections can be obtained after deducting a constant potential scattering cross section of 11.5 b and the small fission cross sections from the total cross sections. In the higher energy region, the results are systematically lower than those of the direct measurements.

Suppose the uncertainties of the normalization are neglegible comparing the uncertainties of Wisshak et al. and Vanpraet et al.'s measurements with those of the first four resonances of Derrien and Lucas. Then all measurements mentioned above may be considered and combined as independent ones and weighted averaged in proper energy intervals, except the Gayther et al.'s measurement (normalized to Weston and Todd's measurement) which is combined with Weston and Todd's measurement in advance. The evaluated results with the measured data for the energy region of 30 eV ~ 500 keV are shown in Fig. 2.

The model theory calculations were adopted to complement the unmeasured energy region up to 20 MeV

3.2.2 Isomeric Ratio of Capture Cross Section

The capture reactions populate not only the ground state ($T_{1/2} = 16$d), but also an isomeric state ($T_{1/2} = 152$y) of ^{242}Am. So the resultant path of ^{241}Am capture will be different dependent on whether the ground state or isomeric state is formed. The isomeric ratio (IR) is defined as the relative population of the ground state to the total capture cross section. As we have mentioned before, it partially determines the relative amounts of ^{242}Cm and ^{244}Cm. The nuclei in the ground state will decay immediatelly to ^{242}Cm via beta decay. But the nuclei in the isomeric state can further transmuted into higher americium isotopes and finally to ^{244}Cm. That is unwanted. Hence the knowledge of isomeric ratio of neutron capture cross section is of considerable interest to establish its production rate in a reactor.

Before 1982, all measurements on isomeric ratio were performed by activating ^{241}Am samples in the spectra of thermal or fast reactor. These measurements are in good agreement with each other[7]. And the evaluated values[7] based on these integral measurements are shown in Table 3.

A new differential measurement of Wisshak et al.[10] determined the capture cross section for populating the ^{242g}Am by neutron capture relative to Au(n,γ) cross section, i.e., $R_1 = \sigma_\gamma(^{241}\text{Am} \rightarrow {}^{242g}\text{Am}) / \sigma_\gamma(\text{Au})$. The IR can be calculated from the measured R_1 provided the value of R_2, the ratio of the gold capture cross section and the total capture cross section of ^{241}Am, is given. It means that the *IR* values of this measurement depend on the total capture cross sections of ^{241}Am. Based on the present evaluation for total capture cross sections (see paragraph 3.1.1) at 27 and 29 keV, the *IR* for fast neutron can be obtained as shown in Table 3. The *IR* values obtained based on theoretical calculation[10] are also shown in this Table for comparison.

For thermal energy, the differential and integral measurements are in agreement within the quoted errors providing that the present evaluation of thermal capture cross section (as a standard, the thermal capture cross section for gold is well known) is used for obtaining R_2. And the value of 0.9 ± 0.01 is adopted for this evaluation for the isomeric ratio of thermal neutron capture.

Then for fast neutron, the old integral measurements are in good agreement with each other, but large discrepancy exists between the integral measurements and the new differential measurement by Wisshak et al.. The value of differential measurement for fast neutron is about 20 % lower than the integral measurements, much larger than the quoted errors. As we have mentioned before, the *IR* of differential measurement depend on total capture cross sections around 30 keV. It seems to be difficult to say that this discrepency comes from a overestimation of total capture cross section in the present evaluation based on the available measurements. The transmission measurement by Derrien and Lucas only extends to 1 keV, although it shows a tendence of supporting the lower capture cross section and higher *IR* than the present evaluation at fast neutron energy region.

Hence presently, it is reasonable to adopt the weighted mean of the value of differential measurement and the evaluated value based on integral measurements as the *IR* evaluation for fast neutron. And the Wisshak et al.'s calculations of *IR*(E_n) for the whole (except the thermal) energy region normalizing to this evaluation at fast neutron energy are adopted as the evaluation of *IR*(E_n).

3.3 Fission Cross Section

The fission cross sections of ^{241}Am were evaluated by Zhou Huimin in 1978[17]. Then a new evaluation was carried out by Gu Fuhua in 1982[18] based on the following measurements:

Bowman (LRL), 0.7~7.0 MeV (1965) 2.5 MeV data given[19],
Kazarinova (CCP), 2.5, 14.6 MeV (1960)[20],
Iyer (HAR), 14 MeV (1969) relative to ^{238}U[21],

Protopopov (CCP), 14.6 MeV (1959)[22],
Bowman (LRL), 0.5~6 MeV (1967)[23],
Fomushkin (CCP), 14.5 MeV (1967)[24],
Fomushkin (CCP), 0.45~3.6 MeV (1969) relative to $^{235,238}U$[25],
Shpak (CCP), 0.008~2.4 MeV (1969) relative to ^{239}Pu[26],
Behrens (LLL), 0.001~30 MeV (1976)[27],
Hage (KFK), 0.02~1.0 MeV (1977)[28],
Gayther (HAR), 0.001~0.01 MeV (1977)[12],
Knitter (GEL), 0.0001~5.3 MeV (1978)[13],
Wisshak (KFK), 0.01~0.25 MeV (1980) relative to ^{235}U[15].

Some data were read from graphs. All relative measurements had been renormalized to the new evaluations for $^{235,238}U$ and ^{239}Pu, respectively. Fitting the measured data with orthogonal polynomials in two energy regions (0.001 ~ 0.7 and 0.7 ~ 20 MeV) the evaluated excitation function was obtained.

After the Gu's evaluation, some new measurements are available. Among them the Dabbs et al.'s measurement[5], covering a large energy range 0,02 eV ~ 20 MeV, has been taken into account in the revision of Gu's evaluation to obtain the present evaluation of fission cross sections, as showing in Fig.3.

3.4 Other Reaction Cross Sections

The unmeasured reaction cross section including (n,2n) (only one measurement at 14 MeV is available), (n,3n) reactions, elastic, inelastic scattering cross sections and unmeasured energy region of (n,γ) reaction, etc., were obtained from model theory calculation. A more detailed discussion is given in paragraph 8.

4 ANGULAR ENERGY DISTRIBUTION OF SECONDARY NEUTRONS

4.1 Angular Distribution

No measured data on elastic and inelastic scattering are available. Optical model and Hauser–Feshbach theory calculations are adopted for this evaluation.

For (n,2n), (n,3n), and fission reactions, the angular distributions are assumed to be isotropic in center of mass system, for inelastic scattering (continuum part) reactions, assumed to be isotropic in laboratory system.

A more detailed discussion is given in paragraph 8.

4.2 Energy Distribution

For (n,2n), (n,3n) and the continuum of inelastic scattering, the secondary neutron energy spectra were calculated with Hauser–Feshbach and evaporation model taking the pre–equilibrium process into account.

For fission neutrons, Maxwellian distributions were assumed. The parameters $T(E)$ were estimated from the formulae given by Hu Jimin et al.[29].

5 AVERAGE NUMBER OF NEUTRONS EMITTED PER FISSION

According to T. R. England[30], the number of delay neutrons per fission has been estimated.

No new information is available after our previous compilation[31] or Kikuchi's evaluation[32] for the average number of prompt neutrons. The same value of Kikuchi's evaluation was adopted in the present work.

6 COVARIANCE DATA

Covariance for total, total inelastic scattering, (n,2n), (n,3n), fission and capture cross section as well as resonance parameters are estimated subjectively based on the reported uncertainties of the experimental data and the scatter of the measurements, as well as the disscusions on the uncertainties of the measured data mentioned in the above paragraphs. For the unmeasured reaction channels or energy regions, large uncertainties up to 8 % (for total cross section), or 50 ~ 100 % (for other reactions) are given for the model theory calculations.

7 MODEL THEORY CALCULATION

7.1 Optical Model Calculation

A set of optical potential parameters were obtained via adjustment of these parameters to reproduce the total cross section measured by Phillips and Howe[14] at the energy range from 500 keV to 25 MeV. The results are nearly as the same as those which have been used in the data calculations of Igarasi and Nakagawa[33] and Kikuchi[32] for ^{241}Am and other americium and curium isotopes. Then for convenience, the parameter set of Igarasi and Nakagawa was adopted in this evaluation for optical model calculation. With these parameters the total cross sections and angular distributions of shape elastic scattering are calculated.

7.2 Calculation of Hauser–Feshbach Statistical Theory with Width Fluctuation Corrections[34]

In the incident neutron energy range of 0.03 ~ 3 MeV, Hauser–Feshbach statistical theory with width fluctuation correction is used in this evaluation (Code LIANG). The transmission coefficients of neutron, fission and radiative capture channels needed in the calculations are computed from optical model by using global parameters, Bohr–Wheeler's theory on nuclear fission–double barrier model and giant dipole resonance theory respectively. All parameters occuring in concerned theories except for optical model were taken from Lynn[35] and adjusted to fit the evaluated fission and capture cross sections of this work. The inelastic scattering cross sections and angular distributions for 18 discrete levels have been calculated. The parameters of these levels are taken from the Table of Isotopes (7th edition). The levels above these 18 levels are considered to be overlapping and the secondary neutron spectra were calculated.

7.3 Calculation of Evaporation Model Taking Into Account Pre–eqilibrium Emission[36, 37]

Cross sections of various channels in the energy region of 3 MeV $< E_n <$ 20 MeV were calculated with evaporation model taking into account the pre–equilibrium emission. In the calculations (Code IPEET 103) of fission cross sections the parameters concerned were adjusted automatically to fit the evaluated data . And the the parameter T was adjusted to get the calculated capture cross sections near 3 MeV to agree with the values calculated by Hauser–Feshbach theory. In the calculation the maximum exciton number N_m = 15 and the parameter $\bar{K}$ = 367 (MeV) were adopted to fit the (n,2n) cross section of 200 mb at 14.7 MeV[38]. The total inelastic scattering cross sections were obtained by subtracting all other competing channel cross sections from total cross section. In the meantime the normalized secondary neutron energy distributions of (n,n′), (n,F), (n,2n) and (n,3n) reactions were also calculated.

8 CONCLUSION AND DISCUSSION

Generally speaking, the evaluated data have been much improved than before, not only for some newly measured data are availabe and taken into account in the present evaluation, but also the problem in the capture cross section evaluation which puzzled evaluators for years have been solved and more reasonable evaluations for thermal value then for whole energy region could be obtained without any compromise[1]. Furthermore, the higher values of the capture cross section evaluation for the present work should improve the agreement with the integral testing results.

The accuracy requested for capture cross section of ^{241}Am is 5 % in keV energy region. We should say that the present evaluation has met such request essentially. For fission cross section, the most recent measurement of Dabbs et al. has been taking into account in the present evaluation for the whole energy region. After the denying of Bombshot measurement, the Dabbs' measurement covering a wide energy range is valuable for making the present evaluation more reliable.

For reference, some typical integral results for capture and fisson cross section[39~41] are listed in table 4.

The data set for ^{241}Am, (in which the files related to γ–production data have been complemented by Howerton[42]), have been accepted by IAEA / NDS, ENDF / B–6 and CENDL–2.

ACKNOWLEDGMENT

This work was performed under contracts between IAEA / NDS and IAE.

The authors would like to give their thanks to Zhao Zhixiang, Sun Naihong, Liang Qichang and Ma Lizhen for their kind helps with this evaluation.

Table 1 Information of Thermal Capture Cross Section

1	654 ± 104	Dobchenko et al.(1971)	Data quoted from[7]
2	612 ± 35*	Harbour et al. (1973)	Data quoted from[7]
3	625 ± 35	Pomerance et al.(1955)	Data quoted from[7]
4	582 ± 50	Weston Todd (1976)	[8]
5	625 ± 20	Kalebin et al. (1976)	[2]
6	585 ± 45	Adamchuk et al. (1955)	[3]
7	639 ± 37	Wisshak et al. (1982)	[10]

* The reported value, 832 b, by Harbour et al. originally is much greater than the total thermal cross section of 640 b.

Table 2 Evaluated Values Thermal Capture Cross Section

Evaluated value (b)	Author	Reference
625 ± 20	Goel (1978) $\sigma_t - \sigma_n - \sigma_f$	[4]
600 ± 20	Lynn et al. (1980)	[7]
610 ± 19	Froehner et al.(1982)	[11]
587 ± 12	Mughabghab (1984)	BNL–325, 4th ed
626 ± 14	This work (1988)	[1]

Table 3 Isomeric Ratio

	Thermal	Fast
Eval. based on intgl meas.	0.90 ± 0.01	0.84 ± 0.03
Differential measurement	0.926 ± 0.06	0.70 ± 0.05*
Theoretical calculation	0.84	0.75
This evaluation	0.90 ± 0.01	0.80 ± 0.07

* Based on this evaluation, i.e., the total capture cross section of ^{241}Am at 27 keV: 2.47 b and at 29 keV: 2.36 b are used in the R_2 calculations.

Table 4 Integral Testing Results

No.	capture σ exp.	capture σ cal.	capture c/e	fission σ exp.	fission σ cal.	fission c/e	neutron field	mic.data used in calculat.
1	1.520	1.109	0.72	0.504	0.522	1.04	CFRMF	ENDF / B–V
2	1.05	0.90	0.86	0.70	0.52	0.74	RAPSODIE	KEDAK–3
3	1.48	1.28	0.86				ZEBRA 12 (CORE CENTER)	AERE eva.

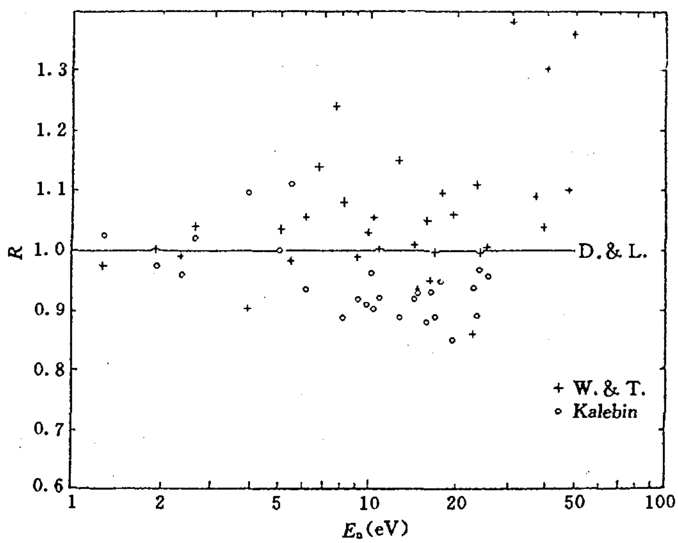
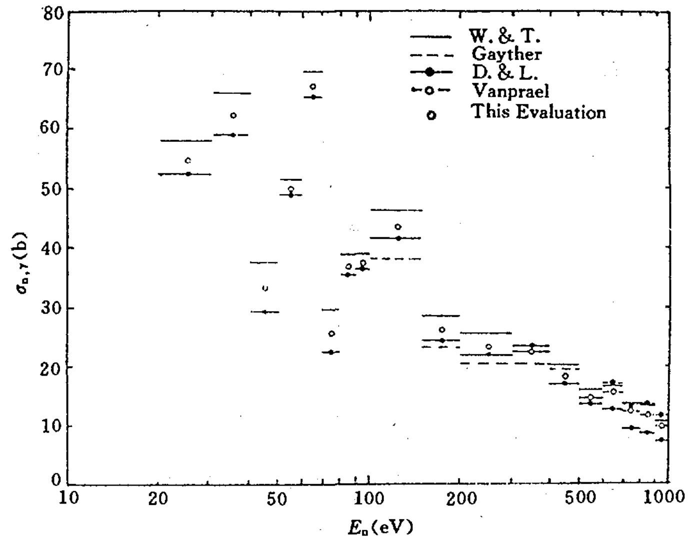


Fig. 1 Ratio of neutron width of Weston et al. and Kalebin to those of Derrien et al.



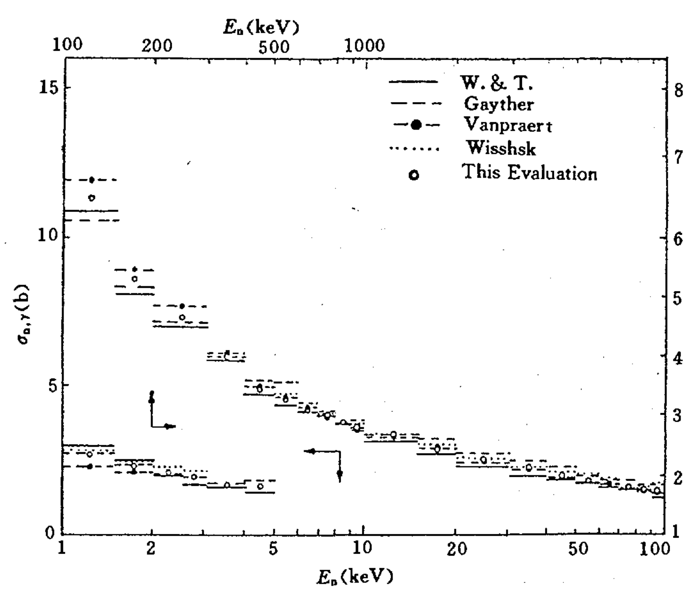


Fig. 2 Capture cross section of ^{241}Am

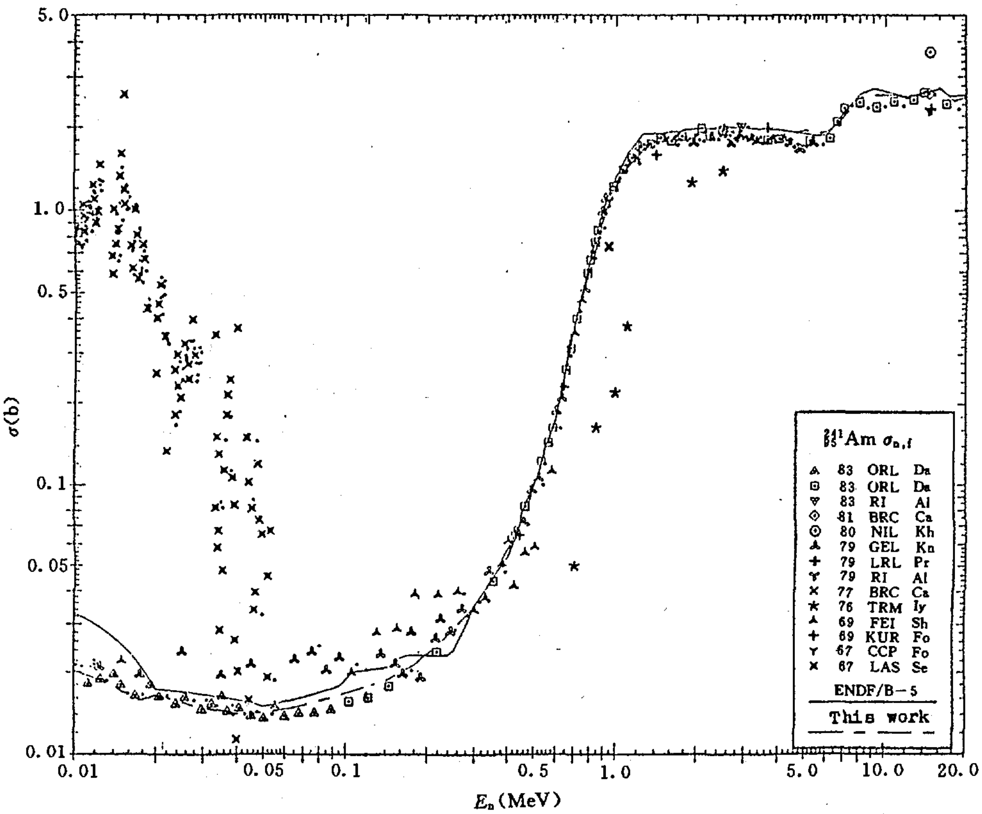


Fig. 3 Fission cross section of ^{241}Am

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EVALUATION OF NEUTRON NUCLEAR DATA FOR ^{249}Bk

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ABSTRACT

A complete set of neutron nuclear data for ^{249}Bk in ENDF / B–6 format were evaluated in the energy range from 10^{-5} eV to 20 MeV based on the measured data, systematics predictions and model theory calculations. The numerical data are available in ENDF / B–6 format.

INTRODUCTION

Neutron nuclear data of ^{249}Bk were evaluated in the energy range from 10^{-5} eV to 20 MeV. As a transplutonium isotope, its data set is useful for analyzing the down–stream problem of fuel cycle.

The quantities we have been evaluated are the total, elastic and inelastic scattering, fission, capture, (n,2n), (n,3n) reaction cross sections, the resolved and unresolved resonance parameters, the angular and energy distributions of emitted neutrons and the average number of neutrons emitted per fission.

The measured data of resonance parameters and fission cross sections are available then for these data the experimental evaluations were carried out. For the unmeasured reaction channels of (n,2n), (n,3n), (n,p) and (n,α), the excitation functions were predicted by using the systematics. Then taking these cross sections as the competing process, the other unmeasured data were calculated with model theories.

1 RESONANCE PARAMETERS AND THERMAL NEUTRON CROSS SECTIONS

MLBW resolved resonance parameters were given for the neutron energy region below 60 eV. The evaluated parameters mainly based on the measurements of Benjamin et al.[1] and Anufriev et al.[2]. Unfortunately, except the value of 710 b for (n,γ) cross section given by Benjamin, no other reliable and newly measured thermal cross sections are avalaible for this evaluation. For giving the reaction cross sections of thermal energy and thermal energy region down to 10^{-5} eV, the resonance parameters at negetive energy were given consulting the evaluation of Kikuchi et al.[3] and Mughabghab[4]. Following values of reaction cross section at thermal energy were predicted by using this parameter set:

$$\sigma_t = 756 \text{ b}$$
$$\sigma_n = 7.8 \text{ b}$$
$$\sigma_f = 4.0 \text{ b}$$
$$\sigma_\gamma = 744 \text{ b}$$

Based on the recommended resolved resonance parameters and consulting the evaluation of Kikuchi et al., the unresolved resonance parameters were obtained.

2 REACTION CROSS SECTIONS ABOVE RESONANCE ENERGY REGION

2.1 Evaluations of Reaction Cross Sections

No measured data are available for the various reaction channels except the fission reaction.

Three measurements by Sibert[5], Fomushkin et al.[6] and Vorotnikov et al.[7], respectively, are available for the fission reaction cross section. Considering that the bomb–shot measurement of Sibert is systematically higher than and denied by other measurements, only the data of Fomushkin et al. and Vorotnikov et al. were adopted in this evaluation and used as reference for model theory calculation for complementing the unmeasured energy region.

The excitation functions of the (n,2n), (n,3n) reaction cross sections (compromising with the fission cross sections), and (n,p), (n,α) reaction cross sections were calculated by using systematics based on evaporation model including pre–equilibrium effect[8, 9].

2.2 Model Theory Calculations and Parameters Concerned

Taking the fission, (n,2n), (n,3n), (n,p) and (n,α) cross sections mentioned above as competing process, all other cross sections were calculated by using model theories including optical, Hauser–Feshbach with width fluctuation correction, exciton and evaporation model, and Bohr–Wheeler's fission theory.

The optical potential parameter set which has been used in the evaluation of ^{241}Am by Kikuchi was adopted in this work. The γ– strength function was obtained from the resonance parameters evaluation. The level scheme of the discrete levels was taken from the Table of Isotopes (7th editiion).

The calculated and measured fission cross sections are shown in Fig. 1.

3 OTHER QUANTITIES

3.1 Total Number of Neutron Emission per Fission

No measured data are available for the average number of neutron emission from neutron induced fission. In this evaluation, energy dependence of the number of prompt neutron per fission was calculated by using the formula taken from Qiu Xijun et al.[10]. And the number of delay neutrons was calculated by using the formula taken from Kikuchi. Then the total number of neutrons per fission is the sum of prompt and delay neutrons.

3.2 Secondary Neutron Spectrum and Angular Distribution

The angular distributions of elastic scattering and inelastic scattering to the discrete levels were calculated by using the model theories mentioned above. The angular distributions of the neutron emission from (n,2n), (n,3n) and fission as well as the inelastic scattering from continuum were assumed to be isotropic in the laboratory system.

The energy distributions of secondary neutrons from (n,2n), (n,3n) and inelastic scattering from the continuum were calculated by using the model theories mentioned above. Maxwellian spectrum was assumed for the fission neutrons, and the temperatures were estimated by using the formula taking from Hu Jimin et al.[11].

ACKNOWLEDGEMENT

The authors would like to give their sincerely thanks to Drs. Zhao Zhixiang, Liang Qichang and Ma Lizhen for their helps during the course of this evaluation.

This work was performed under the contract between IAEA and IAE.

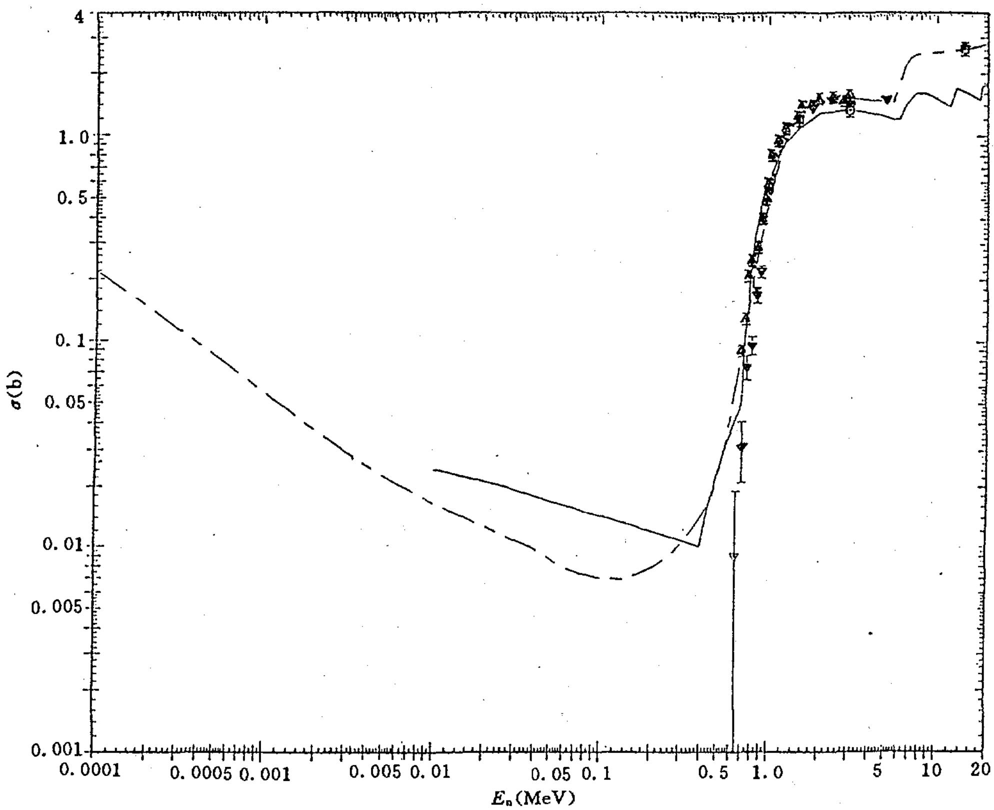

Fig. 1 Fission cross section of ^{249}Bk

—— ENDF / B–5, — • — Thins work, △ 77 LAS Silbert,

□ 71 KUR Fomshkin, ▽ 70 KUR Vorotnikov

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EVALUATION OF NEUTRON NUCLEAR DATA FOR ^{249}Cf

Zhou Delin Yu Baosheng Yuan Hanrong Liu Tong
Zhang Jin Su Zongdi Yan Shiwei Wang Cuilan Zhang Jingshang

(CHINESE NUCLEAR DATA CENTER, IAE)

ABSTRACT

A complete set of neutron nuclear data from 10^{-5} eV to 20 MeV in ENDF / B–6 format has been evaluated for ^{249}Cf based on measured data, systematics predictions and model theory calculations. The evaluated quantities are the total, elastic and inelastic scattering, fission, capture, (n,2n), (n,3n) reaction cross sections, the resolved and unresolved resonance parameters, the angular and energy distributions of emitted neutrons and the average number of neutrons emitted per fission. The numerical data are available in ENDF / B–6 format.

INTRODUCTION

The cross sections of the transplutonium nuclides constitute the basic input for predicting the production of these nuclides in reactor.

This work was performed under contracts between IAEA and CNDC for providing a complete data set of ^{249}Cf in the energy range from 10^{-5} eV to 20 MeV. The evaluated quantities are the total, elastic and inelastic scattering, fission, capture, (n,2n), (n,3n) reaction cross sections, the resolved and unresolved resonance parameters, the angular and energy distributions of emitted neutrons and the average number of neutrons emitted per fission.

The measured data of resonance parameters and fission cross sections are available. So for these cross sections the evaluations based on experimental data were carried out. For the unmeasured reaction channels of (n,2n), (n,3n), (n,p) and (n,α), the excitation functions were predicted by using systematics. Then taking these cross sections as the competing process, the other unmeasured data were calculated with model theories.

1 RESONANCE PARAMETERS AND THERMAL NEUTRON CROSS SECTIONS

MLBW resolved resonance parameters were given for the neutron energy region below 70 eV. The preliminary evaluation was performed mainly based on the measurement of Benjamin et al.[1], Anufriev et al.[2] and Silbert[3] consulting the parameters for negetive level given by Kikuchi[4] and Mughabghab[5].

Based on the evaluated resolved resonance parameters and consulting the evaluation of Kikuchi et al., the unresolved resonance parameters were obtained.

The evaluated resolved resonance parameters and the unresolved resonance parameters have been checked and adjusted if necessary to fit the point–wise data of Dabbs et al.[6] and Silbert as well as the measured thermal cross sections listed as following, then a set of recommended resonance parameter was obtained :

Total Cross Section

2000 b, Benjamin et al., 1983[1]
2400 ± 800 b, Anufriev et al., 1983[2]

Fission Cross Section

1577 ± 70 b, Metta et al., 1965[7]
1690 ± 160 b, Halperin et al., 1970[8]
1630 ± 100 b, Fomushkin et al., 1971[9]
1650 ± 50 b, Rusche et al., 1971 (integral measurement)[10]
1660 ± 50 b, Benjamin et al., 1972[11]
1670 ± 80 b, Harbour et al., 1973[12]
1610 ± 110 b, Gavrilov et al., 1976[13]

Capture Cross Section

530 ± 33 b, Benjamin et al., 1971[14]
456 ± 25 b, Gavrilov et al., 1976[13]

By using this resonance parameter set, the reaction cross sections in the energy region down to 10^{-5} eV can be calculated and the predicted thermal cross sections are :

$$\sigma_t = 2144 \text{ b}$$

$$\sigma_n = 13.7 \text{ b}$$
$$\sigma_f = 1634 \text{ b}$$
$$\sigma_\gamma = 497 \text{ b}$$

2 REACTION CROSS SECTIONS ABOVE RESONANCE ENERGY REGION

2.1 Evaluations of Reaction Cross Sections

No measured data are available for the various reaction channels except the fission reaction.

Several measurements for fission cross sections are available :

V. Kupriyanov et al., 130 keV ~ 7.4 MeV (1984)[15]
J. Dabbs et al., 6.4 MeV ~ 17 MeV (1981)[6]
E. Fomushkin et al., 250 keV ~ 5.15 MeV (1975)[16]
B. Fursov et al., 500 keV ~ 7 MeV (1974)[17]
M. Silbert, 13 eV ~ 2.9 MeV (1973)[3]
E. Fomushkin et al., 25 meV ~ 15 MeV (1971)[18]
P. Vorotnikov et al., 160 keV ~ 1.6 MeV (1972)[19]

Noticing that the Silbert's bomb–shot measurement and Vorotnikov et al.'s measurement are systematically higher than others, the results of the measurements mentioned above including Dabbs et al.'s data[6] provided by BNL / NNDC were adopted as reference for parameters adjustment of model theory calculation.

The recommended and measured fission cross sections are shown in Fig. 1.

The excitation function of the (n,2n), (n,3n) reaction cross sections (compromising with the fission cross sections) and (n,p), (n,α) reaction cross sections were calculated by using systematics based on evaporation model including pre–equilibrium emission[20, 21].

2.2 Model Theory Calculations and Parameters

Taking the fission, (n,2n), (n,3n), (n,p) and (n,α) cross sections mentioned above as competing process, all other cross sections were calculated by using model theories including optical, Hauser–Feshbach with width fluctuation correction, exciton and evaporation model, and Bohr – Wheeler's fission theory.

The optical potential parameter set which has been used in the evaluation of ^{241}Am by Kikuchi was adopted in this work. The γ– strength function was obtained from the resonance parameters evaluation. The level scheme of the discrete levels was taken from the Table of Isotopes (7th editiion).

3 OTHER QUANTITIES

3.1 Total Number of Neutron Emission per Fission

No measured data are available for the average number of neutron emission from neutron induced fission. In this evaluation, energy dependence of the number of prompt neutron per fission was calculated by using the formula taken from Qiu Xijun et al.[22]. And the number of delay neutrons was calculated by using the formula taken from Kikuchi. Then the total number of neutrons per fission is the sum of prompt and delay neutrons.

3.2 Secondary Neutron Spectrum and Angular Distribution

The angular distributions of elastic scattering and inelastic scattering from the discrete levels were calculated by using the model theory mentioned above. The angular distributions of the neutron emitted from (n,2n), (n,3n) and fission process as well as the inelastic scattering from continuum were assumed to be isotropic in the laboratory system.

The energy distributions of secondary neutrons from (n,2n), (n,3n) and inelastic scattering from the continuum were calculated by using the model theories mentioned above. Maxwellian spectrum was assumed for the fission neutrons, and the temperatures were estimated by using the formula taking from Hu Jimin et al.[23].

ACKNOWLEDGEMENT

The authors would like to give their sincerely thanks to Dr. C. Dunford (BNL / NNDC) for providing numerical data which are unavailable from EXFOR and checking through the evaluations. We also give many thanks to Drs. Zhao Zhixiang, Liang Qichang and Ma Lizhen for their kindly helps during the course of this evaluation.

This work was performed under the contracts between IAEA and IAE.

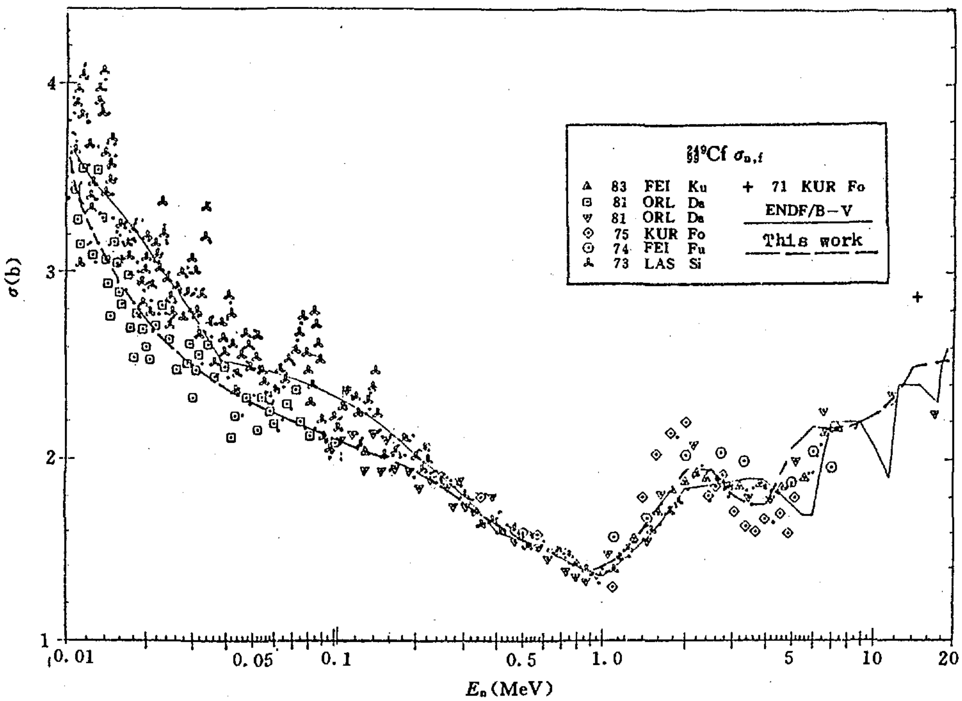

Fig. 1 Fission cross section of ^{249}Cf

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THE SUMMARY FOR RECOMMENDED DATA OF NATURAL Zn, ^{232}Th AND ^{235}U FOR CENDL–2

Liang Qichang

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For the time being the evaluations of natural Zn, ^{232}Th and ^{235}U are recommended as below. Once the new evaluations are completed, they will be replaced.

For the natural Zn, the evaluation was based on the earlier evaluation[1] contained in CENDL–1, and supplemented with those of BROND–2[2].

For the ^{232}Th, the evaluation contained in JENDL–3 was taken over to CENDL–2, because it is more newer than those contained in other libraries.

For the ^{235}U, the evaluation contained in ENDF / B–6 was taken over to CENDL–2, because it is a new evaluation and its cross section for (n,f) is an international reference standard.

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CINDA INDEX

Nuclide	Quantity	Energy (eV)		Lab	Type	Documentation			
		Min	Max			Ref	Vol	Page	Date
^{16}O	Evaluation	1.0−5	2.0+7	AEP	Eval	Jour CNDP	6 Sup.	3	Spt 92
^{23}Na	Evaluation	1.0−5	2.0+7	FUD	Eval	Jour CNDP	6 Sup.	17	Spt 92
^{31}P	Evaluation	1.0−5	2.0+7	NAN	Eval	Jour CNDP	6 Sup.	24	Spt 92
Ca	Evaluation	1.0−5	2.0+7	BJG	Eval	Jour CNDP	6 Sup.	28	Spt 92
Zn	Evaluation	1.0−5	2.0+7	NAN	Eval	Jour CNDP	6 Sup.	123	Spt 92
Mn	Evaluation	1.0−5	2.0+7	TSI	Eval	Jour CNDP	6 Sup.	36	Spt 92
^{59}Co	Evaluation	1.0−5	2.0+7	TSI	Eval	Jour CNDP	6 Sup.	41	Spt 92
Cd	Evaluation	1.0−5	2.0+7	FUD	Eval	Jour CNDP	6 Sup.	53	Spt 92
In	Evaluation	1.0−5	2.0+7	FUD	Eval	Jour CNDP	6 Sup.	60	Spt 92
Hf	Evaluation	1.0−5	2.0+7	FUD	Eval	Jour CNDP	6 Sup.	66	Spt 92
^{232}Th	Evaluation	1.0−5	2.0+7	AEP	Eval	Jour CNDP	6 Sup.	123	Spt 92
^{235}U	Evaluation	1.0−5	2.0+7	AEP	Eval	Jour CNDP	6 Sup.	123	Spt 92
^{239}Pu	Evaluation	1.0−5	2.0+7	AEP	Eval	Jour CNDP	6 Sup.	72	Spt 92
^{240}Pu	Evaluation	1.0−5	2.0+7	AEP	Eval	Jour CNDP	6 Sup.	84	Spt 92
^{241}Am	Evaluation	1.0−5	2.0+7	AEP	Eval	Jour CNDP	6 Sup.	97	Spt 92
^{249}Bk	Evaluation	1.0−5	2.0+7	AEP	Eval	Jour CNDP	6 Sup.	114	Spt 92
^{249}Cf	Evaluation	1.0−5	2.0+7	AEP	Eval	Jour CNDP	6 Sup.	118	Spt 92

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Errata

CNIC-0009 on the cover in No. 7 should be corrected to CNDC-0009.

核数据进展通讯

（第六期补充）

原子能出版社出版

（北京 2108 信箱）

中国核数据中心激光照排

核科学技术情报研究所印刷

☆

开本 787×1092 1/16・印张 7・字数 120 千字

1992 年 10 月北京第一版・1992 年 10 月北京第一次印刷

ISBN 7-5022-0827-5

TL・534

COMMUNICATION OF NUCLEAR DATA PROGRESS

No. 6 Supplement (1992)

ISBN 7-5022-0827-5
TL · 534