NUCLEAR DATA AND MEASUREMENTS SERIES

ANL/NDM-26

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by

Donald L. Smith

December 1976

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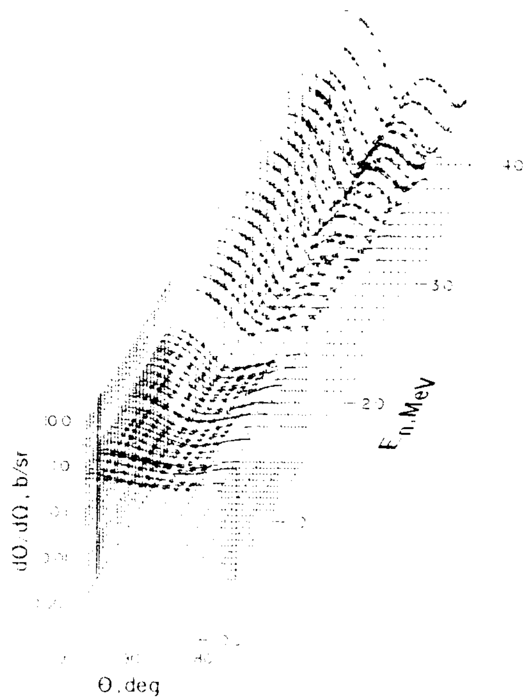

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In January 1975, the research and development functions of the former U.S. Atomic Energy Commission were incorporated into those of the U.S. Energy Research and Development Administration.

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NUCLEAR DATA AND MEASUREMENTS SERIES

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EVALUATION OF THE
IN-115(N,N')IN-115M
REACTION FOR THE ENDF/B-V
DOSIMETRY FILE*

by

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ABSTRACT

An evaluation of the $^{115}In(n,n')^{115m}In$ reaction for the ENDF/B-V Dosimetry File is presented. This evaluation is based entirely on reported experimental differential data. Several data sets were renormalized prior to the evaluation in order to take into account recent adjustments in corresponding standard cross sections and in other nuclear parameters used for derivation of cross sections. The present evaluation is compared with the corresponding ENDF/B-IV evaluation. The value of the spectrum-average cross section for the standard neutron field resulting from thermal-neutron fission of ^{235}U has been computed for this reaction using cross section values from the present evaluation. This computed cross section compares favorably with the result of a recent evaluation of integral data.

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* This work performed under the auspices of the U.S. Energy Research and Development Administration.

I. INTRODUCTION

This reaction plays a major role in reactor dosimetry applications. It is one of thirty-six reactions selected by a Task Force appointed by the Normalization and Standards Subcommittee of the Cross Section Evaluation Working Group (CSEWG) to be included in the ENDF/B-IV Dosimetry File [1]. This reaction is useful for dosimetry applications because indium is readily available and can be easily fabricated into metal foils. The reaction threshold is relatively low and the cross section is large. The half life of the isomer is convenient and the decay yields gamma radiation which can be measured easily.

Table I lists properties of indium and of the $^{115}In(n,n')^{115m}In$ reaction which are relevant to the present evaluation [2-5]. Of particular importance for dosimetry applications are the isotopic abundance of ^{115}In(95.72% [2]), the half life of the 0.336-MeV isomeric state (4.486 h [5]) and the percentage of the disintegrations of the isomer which produce the 0.336-MeV gamma ray that is usually detected (45.9% [5]). The isotopic abundance of ^{115}In in natural indium is well known and not a source of significant uncertainty in cross section measurements. The isomeric half life has been reasonably well known for some time and the minor adjustments which have been made during the last several years do not seriously affect the cross section measurements reported in the literature. However, the situation is quite different for the gammas-per-disintegration factor. The measured cross section is very sensitive to this factor, and it has been revised significantly over the years. The work of Hansen et al. [5] yielded the value which is currently accepted. Care was taken in the present evaluation to determine which values were used for this parameter in various reported experiments so that the data could be renormalized wherever necessary and feasible.

Fig. 1 illustrates the decay of ^{115m}In and compliments the information in Table I. These are other activities produced in natural indium by neutron bombardment. If a Ge(Li) detector is used to detect the 0.336-MeV gamma rays, it is easy to discriminate against interference from the other activities.

The experimental data base for this evaluation is described in Section II. Renormalization of several data sets was required in order to obtain consistency with the ^{115m}In decay data in Table I. Furthermore, several data sets were adjusted so that the values of various standard cross sections used in the measurements would conform to ENDF/B-IV [1,6]. The evaluation process is discussed in Section III. The results of the evaluation are discussed in Section IV. The numerical evaluated cross sections are presented in the Appendix in the form sent to the National Neutron Cross Section Center, Brookhaven National Laboratory.

II. EXPERIMENTAL DATA BASE FOR THE EVALUATION

CINDA-76/77 [7] and the Experimental Data File, CSISRS, from the National Neutron Cross Section Center [8] were used in a search for the available data on this reaction. The documentation for the ENDF/B-IV evaluation of this reaction was also useful in this context [9]. The available data sets are described in chronological order in the remainder of this section.

Cohen (1948)

These data are available from the CSISRS File [8]. The original paper in Nature [10] provides insufficient details to permit renormalization of the cross sections. They were included in original form for this evaluation.

Martin et al. (1954)

These data are available from the CSISRS File [8]; experimental details are reported in the literature [11]. Absolute neutron fluence was monitored with a long counter which had been calibrated using a RaBe neutron source; therefore, no adjustments to the data were required to correct the original measurement of neutron fluence. However, a value of 0.474 was used in this work for the number of 0.336-MeV gamma rays per ^{115m}In decay. Since this differs from the corresponding value in Table I, all the cross sections of Martin et al. were increased by 3.4%.

Ebel et al. (1954)

Measurements were made on this reaction near threshold by Ebel et al. [12] in order to investigate the shape of the excitation function near threshold. These data were normalized relative to the data of Martin et al. [8, 11], so they provide no new information of value for this evaluation other than to generally support the earlier work insofar as the shape is concerned.

Nagel (1966)

The work of Nagel [13] provides a single data point at 14.6 MeV which is available from the CSISRS File [8]. The measurement was made relative to the $^{56}Fe(n,p)^{56}Mn$ reaction and a value of 0.103 barn was used for this standard cross section. This compares favorably with the value of 0.104 barn from the ENDF/B-IV File [1,6] so an increase of < 1% in the cross section was required on this account. A value of 0.48 was used for the number of gamma rays per disintegration. Thus, the cross section was revised upward by 4.6% to achieve consistency with Table I. Nagel refers to earlier papers which provide some pertinent experimental details in readily available form [14,15].

Menlove et al. (1967)

These workers have measured the cross section over a wide energy range from 1 to 19.4 MeV relative to ^{235}U fission [16]. The values reported in the CSISRS File [8] were normalized using BNL-325 [17] values for the fission cross sections and therefore had to be renormalized to be compatible with ENDF/B-IV [1,6]. Also, the value of gammas per decay used was 0.499 which prompted an upward renormalization by 8.7% to achieve compatibility with the currently accepted value of 0.459 from Table I.

Grench et al. (1968)

This is a continuation of work from the same group reporting the work of Menlove et al. (1967). The later work involved a study of the threshold region (E_n < 1.02 MeV) and differs from the earlier work in that $^{197}Au(n,\gamma)^{198}Au$ was used as the standard cross section [18]. The values used for the $^{197}Au(n,\gamma)^{198}Au$ cross section were obtained from Vaughn et al. [19] and these values are consistent with ENDF/B-IV [1,6]. An upward renormalization of 8.7% was required to adjust the data so it would be compatible with the currently accepted value of

0.459 gammas per decay (as described in the preceding discussion of the work by Menlove et al. [16]). The cross section values are available from the CSISRS File [8].

Minetti et al. (1968)

The associated-particle method was utilized by these workers in measuring the cross section at 14.7 MeV [20]. Since the value 0.495 was used for the number of gammas per disintegration, it was necessary to revise the cross section upward by 7.8% for the present evaluation. The cross section information was acquired from the CSISRS File [8].

Roetzer (1968)

Roetzer [21] measured the cross section at 14.7 MeV relative to an assumed value of 0.118 barn for the $^{27}\text{A}\ell(n,\alpha)^{24}\text{Na}$ reaction. This value for the standard cross section is in agreement with ENDF/B-IV [1,6]. From the information provided in his paper, it was deduced that the value used for the number of gammas per disintegration was 0.461 which is close enough to the value from Table I that no renormalization seemed necessary. This data is available from the CSISRS File [8].

Barrall et al. (1969)

This 14.6-MeV measurement also used the $^{27}\text{A}\ell(n,\alpha)^{24}\text{Na}$ reaction as a standard [22]; the value for the standard cross section used was 0.121 barn which leads to a 2.2% reduction to comply with ENDF/B-IV [1,6]. The value 0.5 was used for the gamma rays per decay so an upward renormalization of 8.9% was required. The data is available from the CSISRS File [8].

Kimura et al. (1969)

This work was performed using the hydrogen standard via a proton-recoil detector [23]. However, upward renormalization is required since the value 0.5 was used for the gamma rays per disintegration instead of the value 0.459 from Table I. The original data was obtained from the CSISRS File [8].

Decowski et al. (1970)

These measurements were performed using the $^{64}Zn(n,2n)^{63}Zn$ reaction as a standard [24]. Although values were given for the standard cross sections used, no information on the decay of ^{115m}In(4.486 h) was given. Therefore, no attempt was made to renormalize these data. The cross section values from the CSISRS File [8] were used directly in the evaluation.

Temperley et al. (1970)

This 14.1-MeV measurement was performed using the $^{56}Fe(n,p)^{56}Mn$ reaction as a standard [25]. The value of the standard cross section used is 0.106 barn which differs from the ENDF/B-IV [1,6] value of 0.111 barn by 4.7%. The value used for the number of gamma rays per disintegration was 0.473. The cross section reported in the CSISRS File [8] was renormalized upward by 7.6% to correct for both these factors prior to evaluation.

Pazsit et al. (1972)

Although this work has been published [8,26], there is insufficient information available to enable the data to be renormalized--it was included in the evaluation as reported in the CSISRS File [8].

Kobayashi et al. (1973)

The work of Kobayashi et al. [26] is similar to that of Kimura et al. [8]. This discussion of the earlier work is applicable to this work as well. The original data were obtained from the CSISRS File [8].

Santry et al. (1975)

This work represents one of the more recent and extensive studies of this reaction which has been undertaken [8,28]. The measurements at energies $\lesssim$ 5 MeV were normalized relative to a calibrated long counter so these results have been used in the evaluation in original form. The $^{32}S(n,p)^{32}P$ reaction was used as a standard above $\sim$ 5 MeV. The authors have indicated the values used for the standard. Their results have been renormalized in this region to achieve consistency with ENDF/B-IV [1,6]. The value used for the number of gamma rays per disintegration is 0.459 which agrees with Table I.

Smith et al. (1975)

This recent work spans the region from threshold to $\sim$ 10 MeV and the measurements were performed using ^{235}U fission as a standard [8,29]. ENDF/B-IV values were used for the fission cross sections; however, the value 0.444 was used for the gamma rays per disintegration. These data have been revised downward by 3.3% to correct for this factor prior to evaluation.

III. EVALUATION PROCEDURE

The data sets mentioned in Section II were used to evaluate the cross section for this reaction. The evaluation is subjective in the sense that no fitting or quantitative weighting procedures were used.

A smooth curve was drawn amidst the data points. Wherever possible, greater credence was given data which was well documented and which could be renormalized to conform with currently accepted decay properties or standard cross sections.

Although several of the recent studies have reported on correction procedures used to treat absorption and scattering of radiation, details on this aspect of the measurements were lacking for others. No attempt was made to consider this aspect of the experimental process in the present evaluation.

The approach which was used in the present evaluation is mentioned in this section. Treatment is based on division of the range 0.339 - 20 MeV into several energy intervals—each with its own characteristics.

E_n = 0.339 to 1.25 MeV

The data of Smith et al. [29], Santry et al. [28] and Grench et al. [18] are in excellent agreement and were assumed to define the cross section for this range. The observed structure in the excitation function near threshold appears to be correlated with excited levels in ^{115}In at 0.595, 0.825, 0.858, 0.935, 1.08 and 1.14 MeV [3], though experimental resolution smears out these effects considerably. The cross sections of Martin et al. [11] and Kimura et al. [23] are larger than the values mentioned above for $E_n < 0.9$ MeV and were given less consideration in the evaluation for this range. In the region from 0.9 to 1.25 MeV, all of the above-mentioned data sets are in reasonably good agreement. The cross section is taken to be zero at all energies $E_n \leq 0.339$ MeV (the threshold energy). The smooth curve drawn amidst the experimental points was approximated by straight-line segments (on semilog paper) to arrive at a numerical representation of the evaluation for most neutron energies below 1.25 MeV. Consequently, semilog interpolation should be used in generating evaluated cross sections for energies between mesh points in this region. The detailed procedure for generating evaluated cross sections for applications is discussed in Section IV.

$\underline{E_n = 1.25 \text{ to } 3 \text{ MeV}}$

In this energy range, the cross section continues to increase with energy and finally saturates at $\sim$ 2.5 MeV. The data of Smith et al. [29] and Santry et al. [28] are in excellent agreement throughout this energy range and the evaluation is influenced strongly by these data. The data of Martin et al. [11], Menlove et al. [16], Kimura et al. [23], Kobayashi et al. [27] and Cohen [10] also cover this region, and they generally indicate larger cross sections for the region. The evaluated cross sections were deduced from a smooth curve drawn amidst the data points. Evidence of the effects of excited levels at 1.30, 1.42, 1.60 and 1.98 MeV in ^{115}In is seen in the structure of the excitation function. Linear interpolation can be used to generate evaluated cross sections for energies between the mesh points for <u>all</u> energies above 1.25 MeV.

$\underline{E_n = 3 \text{ to } 6 \text{ MeV}}$

There is noticeable disagreement in the reported cross sections for this energy range. The data of Smith et al. [29] and Santry et al. [28] are in good agreement in this energy range and the evaluation is based on these two sets. The excitation function exhibits a saddle shape--dipping by as much as $\sim$ 10% within this interval. The shape of the excitation function reported by Martin et al. [11] appears to be anomalous with predicted cross sections that are low at $\sim$ 5 MeV. The data of Cohen [10] and Kimura et al. [23] predict cross sections which are large relative to the values selected for this evaluation. The data of Kobayashi et al. [27] and Menlove et al. [16] predict larger cross sections than were chosen for this evaluation in the range 3 - 4.5 MeV; however, at $\sim$ 5 MeV these data are in reasonable agreement with those of Smith et al. [29] and Santry et al. [28]. The evaluated cross sections were read from a smooth curve drawn amidst the experimental values.

<u>E_n = 6 to 10 MeV</u>

In this range, the only available data are from Smith et al. [29], Santry et al. [28] and Menlove et al. [16]. These data are in reasonably good agreement--considering the large error assigned to the 8.06-MeV data point of Menlove et al. [16]. The evaluated cross sections were derived from a smooth curve sketched amidst the experimental data.

<u>E_n = 10 to 12.5 MeV</u>

The only data available for this energy range is that of Santry et al. [28]. The present evaluation is based on a smooth curve drawn amidst the data points.

<u>E_n = 12.5 to 15 MeV</u>

There several reported data sets which span this energy range including single-energy measurements made with neutron generators in the region of 14 to 15 MeV. The data of Santry et al. [28] span the entire range and considerable weight was attached to this data set. The data of Menlove et al. [16] are consistent with the data Santry et al. [28] within the assigned errors although the latter data set predicts generally lower cross sections. The reported details for the data of Decowski et al. [24] are limited. The point at ∿ 13 MeV is very high and appears to be erroneous; it was disregarded. The remaining data points appear to be more or less consistent with the other sets although the point at ∿ 14.5 MeV appears somewhat high. The single data points of Temperley et al. [25], Pazsit et al. [26] and Nagel [15] are reasonably consistent in the vicinity of 14-15 MeV. The data point of Roetzer [21] at 14.7 MeV seems somewhat high while that of Minetti et al. [20] is very and probably in error; it was disregarded in the evaluation. The present evaluation is derived from a smooth curve sketch amidst the data points.

E_n = 15 to 20 MeV

The data of Menlove et al. [16] and Decowski et al. [24] span this energy region. These two data sets are in reasonably good agreement, and the evaluation was derived from a straight line drawn amidst the data points.

IV. RESULTS

The renormalized experimental cross sections, the present evaluation and the ENDF/B-IV evaluation [1,6] are plotted in Figs. 2 and 3. The cross sections corresponding to the present evaluation appear in the Appendix.

In order to estimate the uncertainty in the present evaluation, bands were sketched about the evaluated curve so as to include a majority of the experimental data points considered in the evaluation. These bands are defined by the dispersion indicators listed in Table II. The present evaluation is probably considerably less uncertain than implied by these indicators--particularly above 1 MeV. Because of the qualitative nature of this analysis, the information in Table II should not be used to generate a formal error file for the present evaluation.

The significant differences between the present evaluation and the ENDF/B-IV evaluation [1,6] are that the present evaluation takes greater cognizance of structural detail in the excitation function and predicts larger cross sections in the regions from 2 to 3 and 4 to 6 MeV than does the ENDF/B-IV evaluation.

The two evaluations can also be compared by calculating the spectrum-average cross sections in the standard neutron field produced by thermal-neutron fission of ^{235}U. The spectral shape for this field can be represented quite accurately by the Maxwellian function

$$\chi_{25}(E) = C_{25}E^{\frac{1}{2}} \exp(-E/T_{av}) \tag{1}$$

where T_{av} is taken to be 1.32 MeV from the ENDF/B-IV Dosimetry File [1], and C_{25} is a normalizing constant selected so that

$$\int_0^\infty \chi_{25}(E)\, dE = 1 \tag{2}$$

The spectrum-average cross section is designated $\bar{\sigma}_{nn'}(^{115}\mathrm{In},\chi_{25})$ and is given by the formula

$$\bar{\sigma}_{nn'}(^{115}\mathrm{In},\chi_{25}) = \int_0^\infty \sigma_{nn'}(E)\, \chi_{25}(E)\, dE \tag{3}$$

where $\sigma_{nn'}(E)$ is the normalized excitation function for excitation of ^{115m}In by neutron inelastic scattering. The present evaluation only covers the range from 0 - 20 MeV, however the contribution to the integral from energies > 20 MeV is negligible. The values of the spectrum-average cross section deduced using the present evaluation and the corresponding ENDF/B-IV evaluation [1] are given in Table III. Also shown is the result from a recent evaluation by Fabry et al. [30] of integral measurements. Considering the uncertainties in the integral and differential data used in these evaluations and in the uncertainties in the parameters of the standard neutron field, the perfect agreement of the evaluated integral cross section and the value computed from the present evaluation of differential data is gratifying though rather fortuitous. The significance of the good agreement is enhanced by the fact that the normalization of differential data, upon which the present evaluation is based, is essentially independent of the integral data which formed the basis for the evaluation by Fabry et al. [30].

Each cross section value from the present evaluation carries an index in the range 1 to 99 (see Appendix). The prescription to be used in deriving cross sections at energies intermediate to the mesh points depends upon the index range. Linear interpolation is to be used for energies in the ranges defined by the indices 1 to 3 and 43 to 99. For example, given energy E, we can find the appropriate cross section σ by the formula

$$\sigma = a + b\,E \tag{4}$$

where

$$a = \frac{(E_{i+i}\ \sigma_i - E_i\ \sigma_{i+i})}{(E_{i+i} - E_i)} \qquad (5)$$

and

$$b = \frac{(\sigma_{i+i} - \sigma_i)}{(E_{i+i} - E_i)}\ , \qquad (6)$$

provided (E_i, σ_i) and (E_{i+i}, σ_{i+i}) are mesh points from the Appendix with

$$1 \leq i \leq 2 \quad \text{or} \quad 43 \leq i \leq 98 \qquad (7)$$

and

$$E_i \leq E \leq E_{i+i}\ . \qquad (8)$$

In the range defined by the indices 3 to 43, semilog interpolation is to be used. Thus

$$\ln \sigma = a' + b'\ E \qquad (9)$$

where

$$a' = \frac{(E_{i+i}\ \ln \sigma_i - E_i\ \ln \sigma_{i+i})}{E_{i+i} - E_i} \qquad (10)$$

and

$$b' = \frac{(\ln \sigma_{i+i} - \ln \sigma_i)}{(E_{i+i} - E_i)}\ , \qquad (11)$$

provided (E_i, σ_i) and (E_{i+i}, σ_{i+i}) are mesh points from the Appendix, with

$$3 \leq i \leq 42\ , \qquad (12)$$

and E satisfies Eq.(8).

APPENDIX

EVALUATED CROSS SECTIONS FOR THE IN-115(N,N')IN-115M REACTION

The cross sections comprising the present evaluation have been transmitted to the National Neutron Cross Section Center, Brookhaven National Laboratory, Upton, New York, 11973, U.S.A., in the form shown by the following listing.

```
ISOTOPE INDIUM-115                                          IN115001
INELASTIC NEUTRON EXCITATION OF 4.486 H ISOMER              IN115002
REACTION Q VALUE   -.3350E 06 EV                            IN115003
INTERPOLATION LAW BETWEEN ENERGIES                          IN115004
   INDEX       DESCRIPTION                                  IN115005
  1 TO  3    Y LINEAR IN X                                  IN115006
  3 TO 43    LN Y LINEAR IN X                               IN115007
 43 TO 99    Y LINEAR IN X                                  IN115008
DATA FORMAT (I5,2E10.4)                                     IN115009
   COLUMN      DESCRIPTION                                  IN115010
     1         INDEX                                        IN115011
     2         ENERGY, EV                                   IN115012
     3         ISOTOPIC CROSS SECTION, BARNS                IN115013
EVALUATED CROSS SECTIONS (99 POINTS)                        IN115014
    1 .0000E 00 .0000E 00                                   IN115015
    2 .3390E 06 .0000E 00                                   IN115016
    3 .3400E 06 .1000E-05                                   IN115017
    4 .3500E 06 .1000E-03                                   IN115018
    5 .3600E 06 .4900E-03                                   IN115019
    6 .3700E 06 .7550E-03                                   IN115020
    7 .3800E 06 .9510E-03                                   IN115021
    8 .3900E 06 .1190E-02                                   IN115022
    9 .4000E 06 .1390E-02                                   IN115023
   10 .4100E 06 .1550E-02                                   IN115024
   11 .4200E 06 .1720E-02                                   IN115025
   12 .4300E 06 .1860E-02                                   IN115026
   13 .4400E 06 .2010E-02                                   IN115027
   14 .4500E 06 .2120E-02                                   IN115028
   15 .5000E 06 .2520E-02                                   IN115029
   16 .5300E 06 .2770E-02                                   IN115030
   17 .5400E 06 .2970E-02                                   IN115031
   18 .5500E 06 .3210E-02                                   IN115032
   19 .5600E 06 .3560E-02                                   IN115033
   20 .6000E 06 .5940E-02                                   IN115034
   21 .6300E 06 .8910E-02                                   IN115035
   22 .6500E 06 .1130E-01                                   IN115036
   23 .6700E 06 .1340E-01                                   IN115037
   24 .7000E 06 .1610E-01                                   IN115038
   25 .7500E 06 .2120E-01                                   IN115039
   26 .8000E 06 .2730E-01                                   IN115040
   27 .8500E 06 .3700E-01                                   IN115041
   28 .9000E 06 .4800E-01                                   IN115042
   29 .9200E 06 .5200E-01                                   IN115043
   30 .9400E 06 .5600E-01                                   IN115044
   31 .9600E 06 .5900E-01                                   IN115045
   32 .1000E 07 .6250E-01                                   IN115046
   33 .1020E 07 .6400E-01                                   IN115047
   34 .1050E 07 .6300E-01                                   IN115048
   35 .1070E 07 .6500E-01                                   IN115049
   36 .1100E 07 .6800E-01                                   IN115050
   37 .1120E 07 .7500E-01                                   IN115051
   38 .1140E 07 .8250E-01                                   IN115052
   39 .1160E 07 .9100E-01                                   IN115053
   40 .1180E 07 .9900E-01                                   IN115054
   41 .1200E 07 .1040E 00                                   IN115055
   42 .1220E 07 .1060E 00                                   IN115056
   43 .1250E 07 .1120E 00                                   IN115057
   44 .1300E 07 .1200E 00                                   IN115058
   45 .1500E 07 .1710E 00                                   IN115059
```

46	.1550E 07	.1820E 00	IN115060
47	.1750E 07	.2030E 00	IN115061
48	.1800E 07	.2120E 00	IN115062
49	.1850E 07	.2210E 00	IN115063
50	.2000E 07	.2560E 00	IN115064
51	.2100E 07	.2810E 00	IN115065
52	.2150E 07	.2910E 00	IN115066
53	.2250E 07	.3080E 00	IN115067
54	.2350E 07	.3190E 00	IN115068
55	.2450E 07	.3260E 00	IN115069
56	.2600E 07	.3340E 00	IN115070
57	.2700E 07	.3370E 00	IN115071
58	.2800E 07	.3370E 00	IN115072
59	.3000E 07	.3330E 00	IN115073
60	.3100E 07	.3320E 00	IN115074
61	.3500E 07	.3310E 00	IN115075
62	.3650E 07	.3290E 00	IN115076
63	.3750E 07	.3260E 00	IN115077
64	.4000E 07	.3170E 00	IN115078
65	.4150E 07	.3120E 00	IN115079
66	.4250E 07	.3100E 00	IN115080
67	.4400E 07	.3100E 00	IN115081
68	.4600E 07	.3120E 00	IN115082
69	.5000E 07	.3200E 00	IN115083
70	.5250E 07	.3270E 00	IN115084
71	.5500E 07	.3340E 00	IN115085
72	.5750E 07	.3370E 00	IN115086
73	.6000E 07	.3390E 00	IN115087
74	.6250E 07	.3390E 00	IN115088
75	.6350E 07	.3380E 00	IN115089
76	.6500E 07	.3350E 00	IN115090
77	.7000E 07	.3070E 00	IN115091
78	.7150E 07	.3000E 00	IN115092
79	.7250E 07	.2980E 00	IN115093
80	.7350E 07	.2960E 00	IN115094
81	.7500E 07	.2950E 00	IN115095
82	.7750E 07	.2960E 00	IN115096
83	.8000E 07	.2970E 00	IN115097
84	.8250E 07	.2950E 00	IN115098
85	.8500E 07	.2920E 00	IN115099
86	.8650E 07	.2600E 00	IN115100
87	.1000E 08	.2470E 00	IN115101
88	.1020E 08	.2390E 00	IN115102
89	.1050E 08	.2230E 00	IN115103
90	.1075E 08	.2080E 00	IN115104
91	.1250E 08	.1150E 00	IN115105
92	.1310E 08	.8400E-01	IN115106
93	.1325E 08	.7900E-01	IN115107
94	.1350E 08	.7300E-01	IN115108
95	.1360E 08	.7100E-01	IN115109
96	.1375E 08	.6900E-01	IN115110
97	.1450E 08	.6200E-01	IN115111
98	.1475E 08	.6100E-01	IN115112
99	.2000E 08	.5400E-01	IN115113

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TABLE I

Properties of Indium and the $^{115}In(n,n')^{115m}In$ Reaction Which are of Concern for the Present Evaluation

Property	Value
Natural abundance of indium isotopes	^{113}In : 4.28% [a] ^{115}In : 95.72% [a,b]
Mass of indium isotopes	^{113}In : 112.90 amu [a] ^{115}In : 114.90 amu [a]
Half life of ^{115m}In	4.486 h [c]
Excitation energy for ^{115m}In	0.336 [d]
Decay modes for ^{115m}In (see Fig. 1)	IT : 94.95% [c] β_1 : 0.05% [c] β_2 : 5.00% [c]
Total internal conversion coefficient α for the 0.336-MeV IT	1.073 [c]
Number of 0.336-MeV gamma rays produced per ^{115m}In decay	0.459 [c]
Threshold for $^{115}In(n,n')^{115m}In$ reaction	0.339 MeV [a,d]

[a] Ref. 2.
[b] The ground state of ^{115}In has a half life to β^- decay of 6×10^{14}y [2] and can be considered stable for practical purposes.
[c] Ref. 5.
[d] Ref. 4.

TABLE II

Dispersion Indicators for Data Considered in the Present Evaluation[a]

Energy Range	Dispersion Indicator
Threshold to 0.5 MeV	> ± 30%
0.5 to 0.6 MeV	± 20%
0.6 to 1 MeV	± 15%
1 to 3 MeV	± 10%
3 to 6 MeV	± 8%
6 to 12 MeV	± 6%
12 to 13 MeV	± 10%
13 to 15 MeV	± 20%
15 to 20 MeV	± 15%

[a] The dispersion indicator defines a band around about the final evaluation within which the majority of the experimental data points for a given energy range can be found. It provides a qualitative method for estimating the uncertainty in the evaluation. The uncertainty in the evaluation is probably considerably smaller for each energy region than is implied by the dispersion indicators.

TABLE III

Comparison of Spectrum-Average
Cross Section Values of the $^{115}In(n,n')^{115m}In$
Reaction for $^{235}U(n_{thermal}, f)$ Neutrons

Source	$\bar{\sigma}_{nn'}$ ($^{115}In,\chi_{25}$)
Reproduced from the ENDF/B-IV Dosimetry File[a]	0.1668 barn
Computed from the present evaluation using the standard neutron spectrum defined by ENDF/B-IV parameters[b]	0.1891 barn [c]
Result from an evaluation of integral data by Fabry et al.[d]	0.189 ± 0.008 barn

[a] Ref. 1.

[b] See Section IV and Ref. 1.

[c] Based on the dispersion indicators given in Table III, it is estimated that the uncertainty in the computed average cross section is smaller than ± 9%.

[d] Ref. 30.

FIGURE CAPTIONS

Fig. 1. Decay scheme for ^{115m}In (4.486 h).

(ANL Neg. No. 116-76-404)

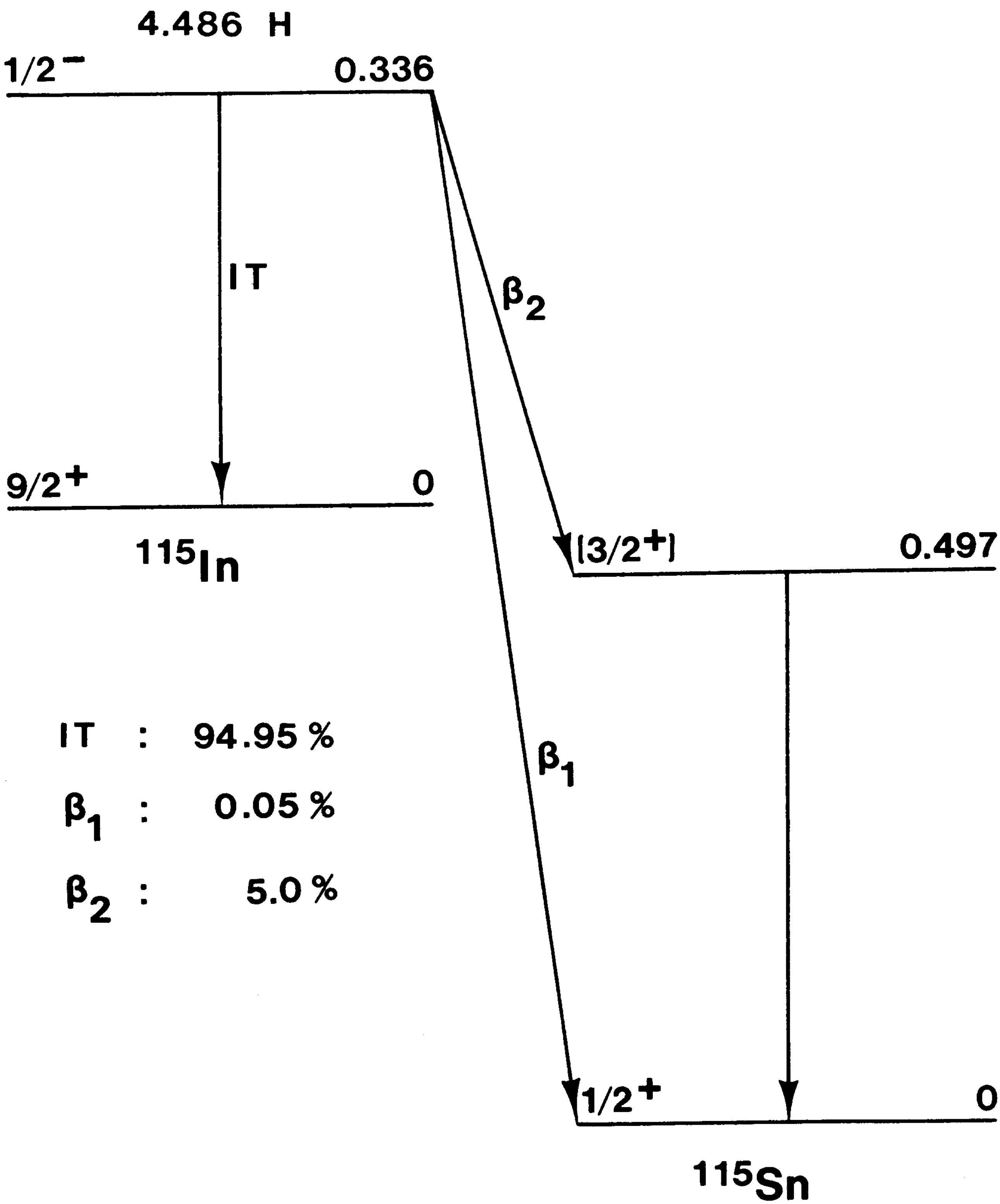
Fig. 2. Plot of experimental data and evaluations for the $^{115}In(n,n')^{115m}In$ in the energy region near threshold.

(ANL Neg. No. 116-76-402)

Fig. 3. Plot of experimental data and evaluations for the $^{115}In(n,n')^{115m}$ reactions from threshold to 20 MeV. Experimental data for energies below 1.2 MeV are not plotted.

(ANL Neg. No. 116-76-402)

Figure 1


4.486 H
1/2⁻
0.336
IT
β2
9/2+
0
115In
(3/2+)
0.497
IT : 94.95 %
β1 : 0.05 %
β2 : 5.0 %
β1
1/2+
0
115Sn

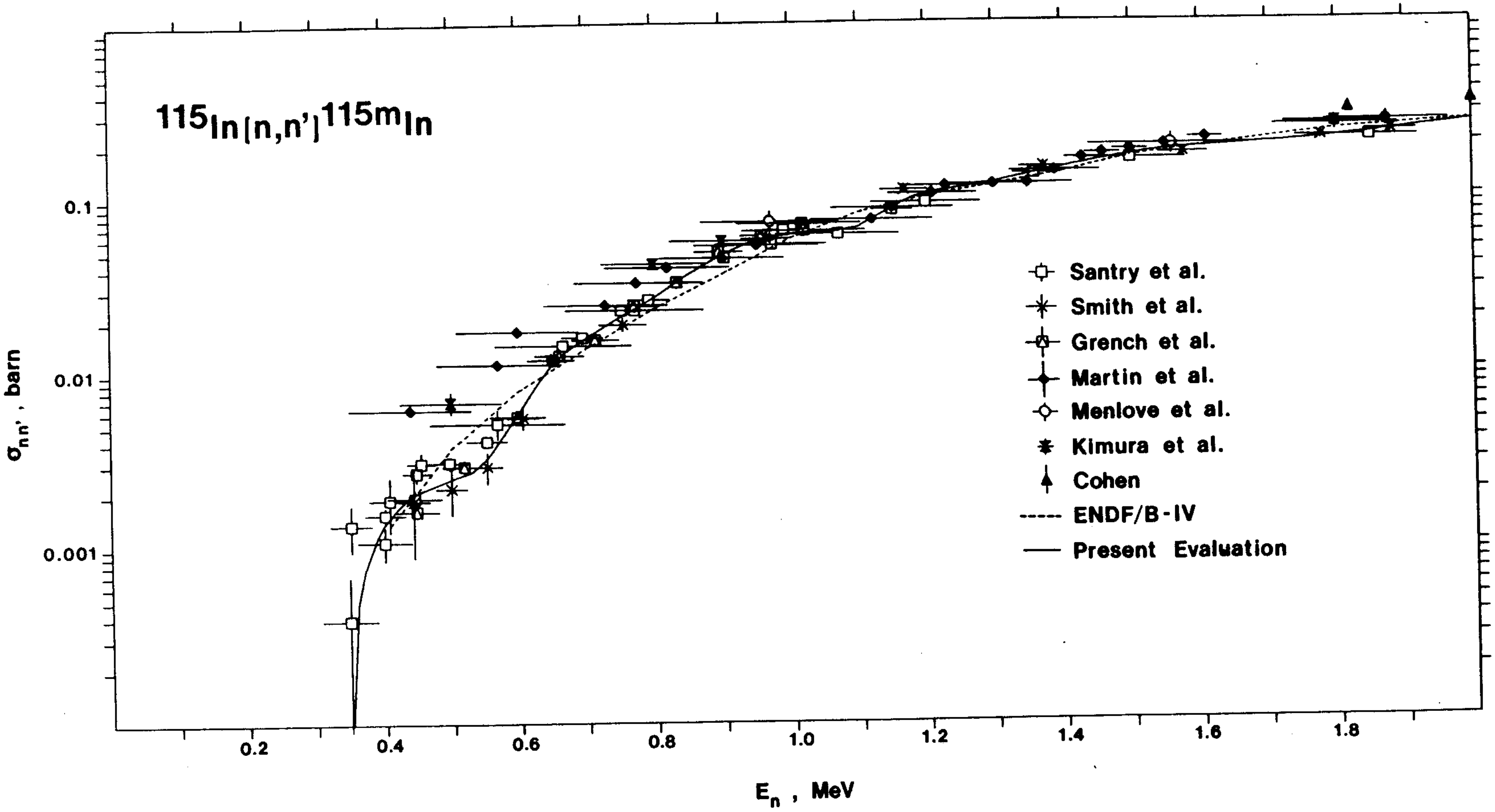

$^{115}In(n,n')^{115m}In$
$\sigma_{nn'}$, barn
0.1
0.01
0.001
0.2
0.4
0.6
0.8
1.0
1.2
1.4
1.6
1.8
E_n , MeV
Santry et al.
Smith et al.
Grench et al.
Martin et al.
Menlove et al.
Kimura et al.
Cohen
ENDF/B-IV
Present Evaluation

Figure 2

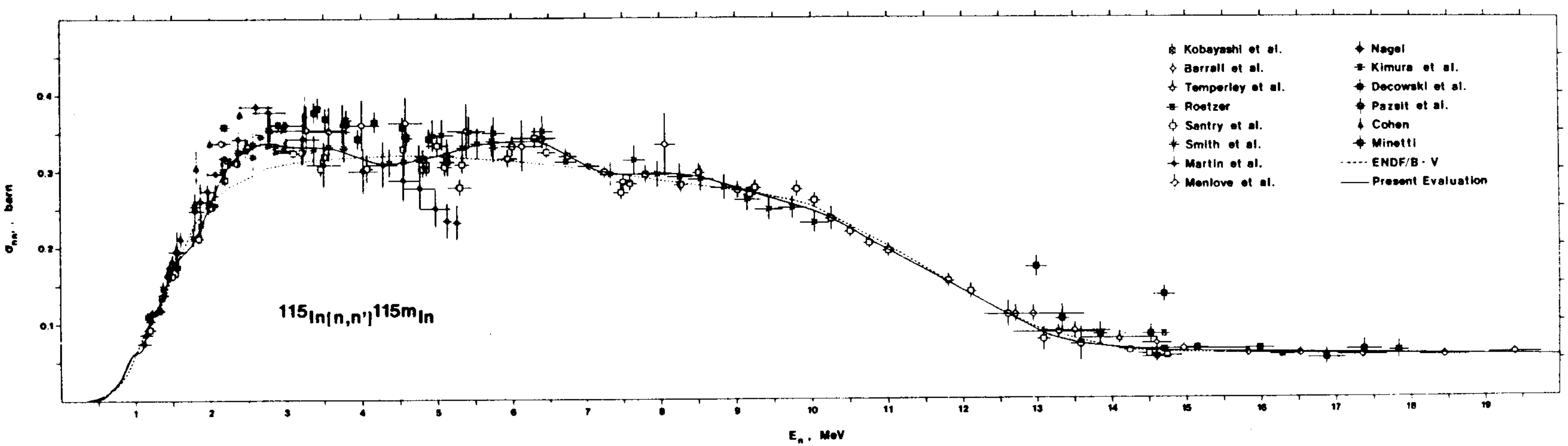

Kobayashi et al.
Barrall et al.
Temperley et al.
Roetzer
Santry et al.
Smith et al.
Martin et al.
Menlove et al.
Nagel
Kimura et al.
Decowski et al.
Pazsit et al.
Cohen
Minetti
ENDF/B-V
Present Evaluation
115In(n,n')115mIn
σn,n', barn
0.1
0.2
0.3
0.4
En, MeV
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19

Figure 3