ANL/NDM-26
EVALUATION OF THE
IN-115(N,N')IN-115M
REACTION FOR THE ENDF/B-V
DOSIMETRY FILE

by

Donald. L. Smith
December 1976

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.

NOTES
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research and Development Administration, nor any of their employees, nor any of their contractors, subcontractors, or their employees, make any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights.

Applied Physics Division
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439
U.S.A.

NUCLEAR DATA AND MEASUREMENTS SERIES

The Nuclear Data and Measurements Series presents results of studies in the field of microscopic nuclear data. The primary objective is the dissemination of information in the comprehensive form required for nuclear technology applications. This Series is devoted to: a) Measured microscopic nuclear parameters, b) Experimental techniques and facilities employed in data measurements, c) The analysis, correlation and interpretation of nuclear data, and d) The evaluation of nuclear data. Contributions to this Series are reviewed to assure technical competence and, unless otherwise stated, the contents can be formally referenced. This Series does not supplant formal journal publication but it does provide the more extensive information required for technological applications (e.g., tabulated numerical data) in a timely manner.

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	3
I. INTRODUCTION	4
II. EXPERIMENTAL DATA BASE FOR THE EVALUATION.	5
III. EVALUATION PROCEDURE	10
IV. RESULTS.	14
APPENDIX: EVALUATED CROSS SECTIONS FOR THE IN-115(N,N')IN-115M REACTION.	18
REFERENCES	21
TABLES.	24
FIGURES	27

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

by

Donald L. Smith

Argonne National Laboratory, Argonne, Illinois 60439, U.S.A.

ABSTRACT

An evaluation of the $^{115}\text{In}(n,n')^{115m}\text{In}$ 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.

* 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 $^{115}\text{In}(n,n')^{115m}\text{In}$ 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 $^{56}\text{Fe}(n,p)^{56}\text{Mn}$ 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 $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ was used as the standard cross section [18]. The values used for the $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ 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 $^{27}\text{Al}(n,\alpha)^{24}\text{Na}$ 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 $^{27}\text{Al}(n,\alpha)^{24}\text{Na}$ 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 $^{64}\text{Zn}(n,2n)^{63}\text{Zn}$ reaction as a standard [24]. Although values were given for the standard cross sections used, no information on the decay of $^{115\text{m}}\text{In}$ (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 $^{56}\text{Fe}(n,p)^{56}\text{Mn}$ 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.

Pazzit 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 ~ 5 MeV were normalized relative to a calibrated long counter so these results have been used in the evaluation in original form. The $^{32}\text{S}(n,p)^{32}\text{P}$ reaction was used as a standard above ~ 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 ~ 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.

$E_n = 0.339 \text{ to } 1.25 \text{ MeV}$

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 $E_n < 0.9 \text{ MeV}$ 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 $E_n \leq 0.339 \text{ MeV}$ (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.

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

In this energy range, the cross section continues to increase with energy and finally saturates at ~ 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 all energies above 1.25 MeV.

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

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 ~ 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 ~ 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.

$E_n = 6$ to 10 MeV

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.

$E_n = 10$ to 12.5 MeV

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.

$E_n = 12.5$ to 15 MeV

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

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

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

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

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

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

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

$$\sigma = a + b E \quad (4)$$

where

$$a = \frac{(E_{i+1} \sigma_i - E_i \sigma_{i+1})}{(E_{i+1} - E_i)} \quad (5)$$

and

$$b = \frac{(\sigma_{i+1} - \sigma_i)}{(E_{i+1} - E_i)}, \quad (6)$$

provided (E_i, σ_i) and (E_{i+1}, σ_{i+1}) are mesh points from the Appendix with

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

and

$$E_i \leq E \leq E_{i+1}. \quad (8)$$

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

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

where

$$a' = \frac{(E_{i+1} \ln \sigma_i - E_i \ln \sigma_{i+1})}{E_{i+1} - E_i} \quad (10)$$

and

$$b' = \frac{(\ln \sigma_{i+1} - \ln \sigma_i)}{(E_{i+1} - E_i)}, \quad (11)$$

provided (E_i, σ_i) and (E_{i+1}, σ_{i+1}) are mesh points from the Appendix, with

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

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.

IS. TYPE 100100-113
 INELASTIC SCATTER EVALUATION OF 0.400 n ISOTOP
 REACTION Q VALUE = 0.5000 eV
 INTERPOLATION LAW BETWEEN SECTIONS

INDEX DESCRIPTION
 1 12 3 Y LIGAND IN A
 2 10 43 L Y LIGAND IN A
 3 10 99 Y LIGAND IN A
 DATA FORMAT (15,0000,4)
 COLUMN DESCRIPTION
 1 INDEX
 2 ENERGY, eV
 3 ISOTOPIC CROSS SECTION, BARS
 EVALUATED CROSS SECTIONS (44 POINTS)

1	.0000	00 .0000e+00
2	.0390	00 .0000e+00
3	.0400	00 .0100e+00
4	.0500	00 .0000e+00
5	.0600	00 .0000e+00
6	.0700	00 .0000e+00
7	.0800	00 .0000e+00
8	.0900	00 .0000e+00
9	.1000	00 .0000e+00
10	.1100	00 .0000e+00
11	.1200	00 .0000e+00
12	.1300	00 .0000e+00
13	.1400	00 .0000e+00
14	.1500	00 .0000e+00
15	.1600	00 .0000e+00
16	.1700	00 .0000e+00
17	.1800	00 .0000e+00
18	.1900	00 .0000e+00
19	.2000	00 .0000e+00
20	.2100	00 .0000e+00
21	.2200	00 .0000e+00
22	.2300	00 .0000e+00
23	.2400	00 .0000e+00
24	.2500	00 .0000e+00
25	.2600	00 .0000e+00
26	.2700	00 .0000e+00
27	.2800	00 .0000e+00
28	.2900	00 .0000e+00
29	.3000	00 .0000e+00
30	.3100	00 .0000e+00
31	.3200	00 .0000e+00
32	.3300	00 .0000e+00
33	.3400	00 .0000e+00
34	.3500	00 .0000e+00
35	.3600	00 .0000e+00
36	.3700	00 .0000e+00
37	.3800	00 .0000e+00
38	.3900	00 .0000e+00
39	.4000	00 .0000e+00
40	.4100	00 .0000e+00
41	.4200	00 .0000e+00
42	.4300	00 .0000e+00
43	.4400	00 .0000e+00
44	.4500	00 .0000e+00

IN115001
 IN115002
 IN115003
 IN115004
 IN115005
 IN115006
 IN115007
 IN115008
 IN115009
 IN115010
 IN115011
 IN115012
 IN115013
 IN115014
 IN115015
 IN115016
 IN115017
 IN115018
 IN115019
 IN115020
 IN115021
 IN115022
 IN115023
 IN115024
 IN115025
 IN115026
 IN115027
 IN115028
 IN115029
 IN115030
 IN115031
 IN115032
 IN115033
 IN115034
 IN115035
 IN115036
 IN115037
 IN115038
 IN115039
 IN115040
 IN115041
 IN115042
 IN115043
 IN115044
 IN115045
 IN115046
 IN115047
 IN115048
 IN115049
 IN115050
 IN115051
 IN115052
 IN115053
 IN115054
 IN115055
 IN115056
 IN115057
 IN115058
 IN115059

46 .1550 07 .154000 00
 47 .1750 07 .203000 00
 48 .1800 07 .212000 00
 49 .1850 07 .221000 00
 50 .2000 07 .230000 00
 51 .2100 07 .239000 00
 52 .2150 07 .248000 00
 53 .2250 07 .257000 00
 54 .2350 07 .266000 00
 55 .2450 07 .275000 00
 56 .2600 07 .284000 00
 57 .2700 07 .293000 00
 58 .2800 07 .302000 00
 59 .3000 07 .311000 00
 60 .3100 07 .320000 00
 61 .3300 07 .329000 00
 62 .3650 07 .338000 00
 63 .3750 07 .347000 00
 64 .4000 07 .356000 00
 65 .4150 07 .365000 00
 66 .4250 07 .374000 00
 67 .4400 07 .383000 00
 68 .4600 07 .392000 00
 69 .5000 07 .401000 00
 70 .5250 07 .410000 00
 71 .5500 07 .419000 00
 72 .5750 07 .428000 00
 73 .6000 07 .437000 00
 74 .6250 07 .446000 00
 75 .6350 07 .455000 00
 76 .6500 07 .464000 00
 77 .7000 07 .473000 00
 78 .7150 07 .482000 00
 79 .7250 07 .491000 00
 80 .7350 07 .500000 00
 81 .7500 07 .509000 00
 82 .7750 07 .518000 00
 83 .8000 07 .527000 00
 84 .8250 07 .536000 00
 85 .8500 07 .545000 00
 86 .8650 07 .554000 00
 87 .1000 00 .247000 00
 88 .1020 00 .239000 00
 89 .1050 00 .223000 00
 90 .1070 00 .200000 00
 91 .1250 00 .119000 00
 92 .1310 00 .840000-01
 93 .1320 00 .790000-01
 94 .1350 00 .730000-01
 95 .1360 00 .710000-01
 96 .1370 00 .690000-01
 97 .1450 00 .620000-01
 98 .1470 00 .610000-01
 99 .2000 00 .540000-01

IN115060
 IN115061
 IN115062
 IN115063
 IN115064
 IN115065
 IN115066
 IN115067
 IN115068
 IN115069
 IN115070
 IN115071
 IN115072
 IN115073
 IN115074
 IN115075
 IN115076
 IN115077
 IN115078
 IN115079
 IN115080
 IN115081
 IN115082
 IN115083
 IN115084
 IN115085
 IN115086
 IN115087
 IN115088
 IN115089
 IN115090
 IN115091
 IN115092
 IN115093
 IN115094
 IN115095
 IN115096
 IN115097
 IN115098
 IN115099
 IN115100
 IN115101
 IN115102
 IN115103
 IN115104
 IN115105
 IN115106
 IN115107
 IN115108
 IN115109
 IN115110
 IN115111
 IN115112
 IN115113

REFERENCES

1. "ENDF/B-IV Dosimetry File", ed. B. A. Magurno, BNL-NCS-50446, National Neutron Cross Section Center, Brookhaven National Laboratory (1975).
2. "Chart of the Nuclides," prepared by Norman E. Holden and William Walker, Knolls Atomic Power Laboratory, General Electric Company, Schenectady, N.Y. 12345 (1972).
3. "Nuclear Level Schemes A=45 Through A=257," ed. Nuclear Data Group, Oak Ridge National Laboratory, Academic Press, New York (1973).
4. A. Bäcklin, B. Fogelberg and S. G. Malmkog, Nucl. Phys. A96, 539 (1967).
5. H. H. Hansen, E. de Roost, W. van der Eijk and R. Vaninbroukx, Z. Physik 269, 155 (1974).
6. "Evaluated Neutron Data File, ENDF/B-IV", National Neutron Cross Section Center, Brookhaven National Laboratory, Upton, N.Y. 11973, U.S.A. (1975).
7. "CINDA-76/77, An Index to the Literature on Microscopic Neutron Data," International Atomic Energy Agency, Vienna (1976).
8. "Experimental Data File, CSISRS," National Neutron Cross Section Center, Brookhaven National Laboratory, Upton, N.Y. 11973, U.S.A. (1976).
9. R. Sher, "The $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ Reaction for ENDF/B-III," p. 150 in Ref. 1 from the present list of references.
10. S. G. Cohen, Nature 161, 475 (1948).
11. H. C. Martin, B. C. Diven and R. F. Taschek, Phys. Rev. 93, 199 (1954).
12. A. A. Ebel and C. Goodman, Phys. Rev. 93, 197 (1954).

13. W. Nagel, "Some Nuclear Reactions Induced by D+T Neutrons," Thesis submitted to the University of Amsterdam (1966).
14. I. Heertje, W. Nagel and A. H. W. Aten, Jr., *Physica* 30, 775 (1964).
15. W. Nagel and A. H. W. Aten, Jr., *Physica* 31, 1091 (1965).
16. H. O. Menlove, K. L. Coop, H. A. Grench and R. Sher, *Phys. Rev.* 163, 1308 (1967).
17. "Neutron Cross Sections, Vol. IIC, Z=61 to 87," BNL-325 2nd Ed., Suppl. No. 2, Brookhaven National Laboratory (1966).
18. H. A. Grench and H. O. Menlove, *Phys. Rev.* 165, 1298 (1968).
19. F. J. Vaughn, K. L. Coop, H. A. Grench and H. O. Menlove, *Bull. Am. Phys. Soc.* 11, 753 (1966).
20. B. Minetti and A. Pasquarelli, *Z. Physik* 217, 83 (1968).
21. H. Roetzer, *Nucl. Phys.* A109, 694 (1968); see also *Oesterr. Akad. Wiss., Math.-Naturw. Kl., Sitzungsber* 176, 289 (1968).
22. R. C. Barrall, J. A. Holmes and M. Silbergold, AFWL-TR-68-134, Air Force Weapons Laboratory, Kirtland AFB (1969).
23. I. Kimura, K. Kobayashi and T. Shibata, *J. Nucl. Sci. Tech. (Japan)* 6, 485 (1969).
24. P. Decowski, W. Grochudski, A. Marcinkowski, J. Karolyi, J. Piotrowski, E. Saad, K. Siwek-Wilezyska, I. M. Turkiewicz and Z. Wilhelmi, INR-1197, Inst. Badan Jadrowych (Nucl. Res.), Warsaw, Poland (1970).
25. J. K. Temperley and D. E. Barnes, BRL-1491, U.S. Army Ballistics Research Laboratory (1970).
26. A. Paszit and J. Csikai, *Soviet Journal of Nuclear Physics* 15, 232 (1972).

27. K. Kobayashi, I. Kimura, H. Gotoh and H. Yagi, J. Nucl. Energy 27, 741 (1973).
28. D. C. Santry and J. P. Butler, Can. J. Phys. 54, 757 (1975).
29. D. L. Smith and J. W. Meadows, ANL/NDM-14, Argonne National Laboratory (1975); see also Nucl. Sci. Eng. 60, 319 (1976).
30. A. Fabry, H. Ceulemans, P. Vandeplas, W. N. McElroy and E. P. Lippincott, "Reactor Dosimetry Integral Reaction Rate Data in LMFBF Benchmark and Standard Neutron Fields: Status, Accuracy and Implications," prepared for the First ASTM-EURATOM Symposium on Reactor Dosimetry, Petten, The Netherlands, Sept. 22-26, 1975.

TABLE I

Properties of Indium and the $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$
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 $^{115\text{m}}\text{In}$	4.486 h ^c
Excitation energy for $^{115\text{m}}\text{In}$	0.336 ^d
Decay modes for $^{115\text{m}}\text{In}$ (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 $^{115\text{m}}\text{In}$ decay	0.459 ^c
Threshold for $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ 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	> $\pm 30\%$
0.5 to 0.6 MeV	$\pm 20\%$
0.6 to 1 MeV	$\pm 15\%$
1 to 3 MeV	$\pm 10\%$
3 to 6 MeV	$\pm 8\%$
6 to 12 MeV	$\pm 6\%$
12 to 13 MeV	$\pm 10\%$
13 to 15 MeV	$\pm 20\%$
15 to 20 MeV	$\pm 15\%$

^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 $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$
Reaction for $^{235}\text{U}(n_{\text{thermal}}, f)$ Neutrons

Source	$\bar{\sigma}_{nn'}, (^{115}\text{In}, \chi_{25})$
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 ^d by Fabry et al.	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 $\pm 9\%$.

^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 $^{115}\text{In}(n,n')^{115m}\text{In}$ in the energy region near threshold.

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

Fig. 3. Plot of experimental data and evaluations for the $^{115}\text{In}(n,n')^{115m}\text{In}$ 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

Figure 2

Figure 3