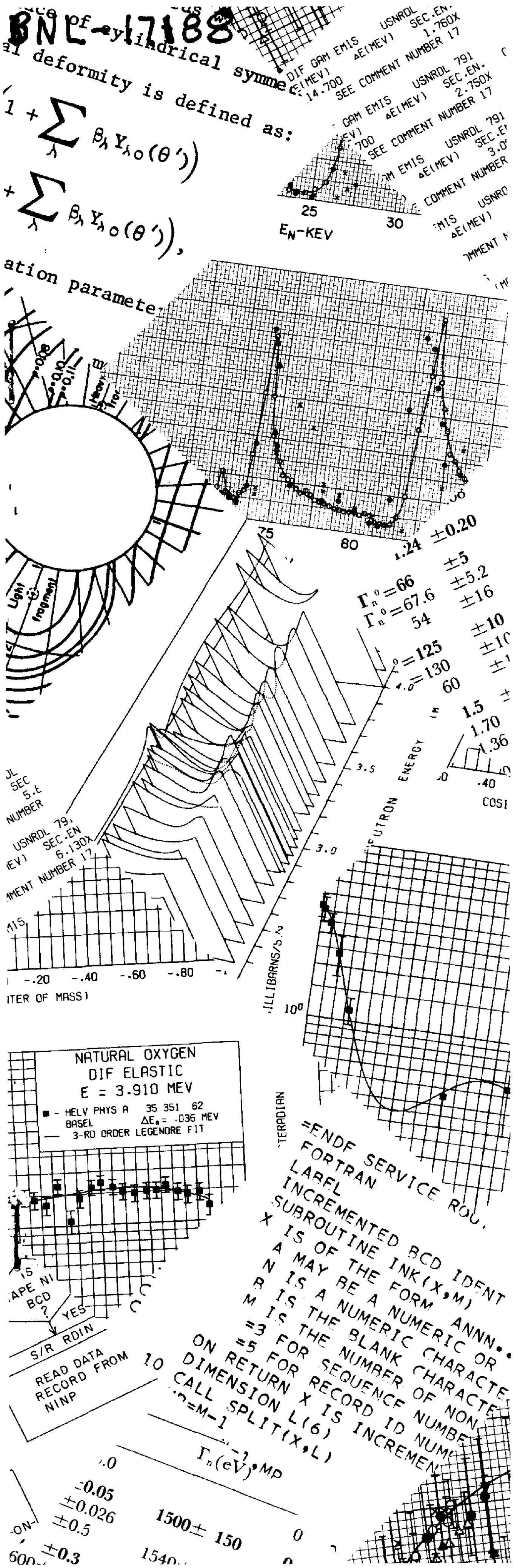


BNL 17188
(ENDF-179)

ENDF/B-III CROSS SECTION MEASUREMENT STANDARDS

M.K. DRAKE

July 1972

Contributors:

H. Alter, N.E. Holden, R. LaBauve,
B.R. Leonard, Jr., C.R. Lubitz, T.A. Pitterle,
R. Sher, J.R. Smith, J.R. Stehn, L. Stewart

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.
UPTON, NEW YORK 11973

national
neutron
Cross
Section
Center

BNL 17188
(ENDF-179)

ENDF/B-III Cross Section Measurement Standards*

M. K. Drake**

National Neutron Cross Section Center
Brookhaven National Laboratory
Upton, New York 11973

July 1972

Contributors

H. Alter, N. E. Holden, R. La Bauve,
B. R. Leonard, Jr., C. R. Lubitz, T. A. Pitterle,
R. Sher, J. R. Smith, J. R. Stehn, L. Stewart

*Research supported by the United States Atomic Energy Commission.
**Present address: Gulf Radiation Technology, San Diego, California 92112.

Contributors

Aerojet Nuclear Company
P. O. Box 1845
Idaho Falls, Idaho 83401
J. R. Smith

Atomics International
P. O. Box 309
Canoga Park, California 91304
H. Alter

Battelle-Northwest
P. O. Box 999
Richland, Washington 99352
B. R. Leonard, Jr.

Brookhaven National Laboratory
Upton, New York 11973
J. R. Stehn

Knolls Atomic Power Laboratory
P. O. Box 1072
Schenectady, New York 12301
N. E. Holden
C. R. Lubitz

Los Alamos Scientific Laboratory
P. O. Box 1663
Los Alamos, New Mexico
R. J. LaBauve
Leona Stewart

Stanford University
Stanford, California 94305
R. Sher

Westinghouse Electric Corporation
Advanced Reactor Division
P. O. Box 158
Madison, Pennsylvania 15663
T. A. Pitterle

Table of Contents

List of Tables and Figures

I. INTRODUCTION

Members of the Cross Section Evaluation Working Group (CSEWG) subcommittee on Normalization and Standards have made a study of the various cross sections that are considered to be standards for cross section measurements. Various members of the Subcommittee evaluated the best available experimental data for the standards. The Subcommittee carefully reviewed the evaluations and recommended that certain sets of evaluated data be incorporated into the ENDF/B-III Library and that these data be considered as standards for future work in the area of experimental measurements and evaluations.

This report contains brief summaries describing the various cross section standards. In certain cases only limited documentation was available. Detailed documentation, when it was available, is given in the Appendices.

The Normalization and Standards Subcommittee have made recommendations on seven standards. Not all of the standards are known to the desired accuracy. Table 1 shows the various standard cross sections which have been incorporated into the regular ENDF/B-III materials. Each material is complete in that it contains all significant partial cross sections covering entire energy ranges and all secondary angular and energy distributions.

Section II of this report contains a brief description of each standard. Some graphical displays are included which show the recommended cross sections and recent sets of experimental data. In some instances (e.g., ^{6}Li, ^{197}Au, and ^{235}U) experimental data not shown on these figures was given the greatest weight in the evaluations. Hence the evaluations, considering <u>all</u> experimental data, are far better than these figures might suggest. Figures 1 and 2 show the recommended standard cross sections.

Status of ENDF/B-III Standards

Mat No.	Material	Reaction Type	Energy Range	Accuracy Requested*	Accuracy Obtained
1148	^{1}H	σ_T	0.1 - 20 MeV	0.5	1
		$d\sigma_n/d\Omega$	3 - 20 MeV	0.5	-
1146	^{3}He	$\sigma_{n,p}$	1 keV - 3 MeV	3	
			< 10 keV		~ 3
			10 - 50 keV		~ 5
1115	^{6}Li	$\sigma_{n,\alpha}$	thr - 100 keV	2	
			0.1 - 14 MeV	3	
			.0253 eV		0.5
			< 10 keV		1 - 2
			10 - 100 keV		~ 3
1155	^{10}B	$\sigma_{n,\alpha}$	thr - 100 keV	1	
			0.1 - 1.0 MeV	4	
			< 1 keV		1
			10 keV		2
			100 keV		3
1165	^{12}C	σ_T	1 keV - 2 MeV	1	1
		$d\sigma_n/d\Omega$	1 keV - 2 MeV	1	
1166	^{197}Au	$\sigma_{n,\gamma}$	1 keV - 100 keV	2	
			10 keV - 500 keV		~ 8
1157	^{235}U	$\sigma_{n,f}$	10 keV - 14 MeV	1	> 5

*Desired accuracy as requested by the CSEWG Normalization and Standards Subcommittee and the U.S. Nuclear Data Committee (the USNDC was formerly the Nuclear Cross Section Advisory Committee (NCSAC) which was formerly the Neutron Cross Section Advisory Group (NCSAG).)

II. DESCRIPTION OF RECOMMENDED STANDARDS

A. Hydrogen Total and Differential Elastic Scattering Cross Sections

MAT = 1148 contains the recommended standards cross sections for hydrogen. This evaluation was done by Stewart, La Bauve, and Young.(1) The recommended total cross section covers the energy range from 0.1 to 20 MeV. The recommended differential elastic scattering cross section is given in File 4 and covers the energy range from 3 to 20 MeV.

The total cross section was taken to be the sum of the elastic scattering and radiative capture cross sections. The elastic scattering cross section was obtained from a theoretical analysis of measurements made by Hopkins and Breit.(2) A set of phase shifts obtained by Seamon et al.(3) was used in this analysis. The differential angular distributions were obtained from the same analysis.

Figure 3 shows a comparison between the recommended total cross section and the results from a measurement made by Davis and Barschall.(4) The uncertainty in the recommended total cross section is believed to be about one percent.

B. ^{3}He (n,p) Cross Section

MAT = 1146 contains the evaluation of ^{3}He cross sections by L. Stewart (LASL). Although this material contains all of the cross section data for ^{3}He, only the (n,p) cross section from 1.0×10^{-5} eV to 50.0 keV represents the recommended standard cross section.

Below one keV, the (n,p) cross section was taken to be 1/v and normalized to 5327 barns at 0.0253 eV. This value was taken from a measurement by Als-Nielsen and Dietrich.(5) The thermal cross section is believed to be known to within one percent.

Between 50 keV and 1 keV, the recommended standard cross section was based on the corrected results from a measurement made by Gibbons and

Macklin.[6] Below 10 keV, the recommended (n,p) is believed to be known to within 3%. Above 10 keV, the uncertainty in the recommended cross section rapidly increases beyond the desired three percent. Figure 4 shows the recommended cross section.

C. $^{6}Li\ (n,\alpha)\ ^{3}H$ Cross Section

The recommended standard cross section for $^{6}Li(n,\alpha)$ covers the energy range from 1.0×10^{-5} eV to 200 keV, and is contained in MAT = 1115. Above 200 keV the cross section is not known to the accuracy required to be considered a cross section standard. The recommended cross section was taken directly from an analysis by Uttley et al.[7] of the total, elastic scattering and (n,α) cross section data.

Up to 100 eV the cross section was calculated from the formula

$$\sigma(n,\alpha) = (149.56/\sqrt{E}) - 0.024 \text{ b}.$$

The 2200 m/sec. cross section value of 940.25 b was based on a total cross section measurement of Uttley and Diment.[8]

Between 100 eV and 500 keV, the recommended (n,α) cross section was obtained by a resonance parameter fit[8] to the best measured results for the total, elastic scattering, and (n,α) cross sections. As can be seen in Figure 5, the fitted curve is not in good agreement with the available experimental results for the $^{6}Li(n,\alpha)$ reaction.

The 2200 m/sec. value is believed to be known to within 0.5%. Between thermal and 10 keV the uncertainty is about one percent and increases to two percent at 100 keV.

Although the Uttley and Diment analysis produced a total cross section that was in excellent agreement with the available experimental data, the disagreement in the (n,α) cross section across the 247 keV resonance precludes the (n,α) cross section from being considered a standard above 100 keV at this time.

D. $^{10}B(n,\alpha)$ Cross Section

The $^{10}B(n,\alpha)$ standard cross section is part of MAT = 1155; the standard is given for the energy range from 1.0×10^{-5} eV to 100 keV. The recommended cross section was based on an analyses by Sowerby et al.[9,10] The 2200 m/sec. value of the (n,α) cross section was taken to be 3836.45 b[11] and this value is believed to be known to within one percent.

Above the thermal neutron energy range, the recommended cross section was obtained from the formula[9,10]

$$\sigma_{n,\alpha} = \frac{13.837}{\sqrt{E}} - 0.312 - 1.014 \times 10^{-2}\sqrt{E} + \frac{2.809 \times 10^{5}}{\sqrt{E}\,[(170.3 - E)^{2} + 2.243 \times 10^{4}]}$$

where σ is given in barns and E in keV.

Sowerby et al. obtained the constants for the above equation by making a least squares fit to selected data sets. The absorption cross section measured by Mooring, Monahan and Huddleston[12] was used. The absorption cross section resulting from subtracting the scattering cross section measured by Asami and Moxon[13] from the total cross section measured by Diment[14] also was used. The most important data set used was the $^{10}B(n,\alpha)$ cross section derived from the Sowerby et al.[9,10] $^{6}Li(n,\alpha)/^{10}B(n,\alpha)$ ratio measurement. In this case the $^{6}Li(n,\alpha)$ cross section was taken to be the same cross section as was used in ENDF/B-III for the standard. The resulting recommended cross sections for $^{6}Li(n,\alpha)$ and $^{10}B(n,\alpha)$ cross section contain a high degree of consistency.

The recommended standard cross section is believed to be known to within one percent for neutron energies less than one keV. This uncertainty rises from two percent at 10 keV to three percent at 100 keV.

E. ^{12}C Total Cross Section

The ^{12}C total cross section from 1.0^{-5} eV to 2.0 MeV has been designated as a cross section standard. The differential angular distribution data for elastic scattering is also a preferred standard, however, the ENDF/B data is not considered adequate as a standard at this time. The recommended cross section is part of MAT = 1165 and was taken from an analysis made by Francis et al.[15]

The recommended free atom scattering cross section is 4.729 barns. This value is believed to be known to within 0.5 percent.

Francis et al.[15] obtained the recommended cross section by analyzing several recent experimental results.[16,17,18,19] R-matrix and coupled channel model calculations were used in the analysis. The recommended cross section is believed to be accurate to within one percent and likely to within 0.5 percent.

F. $^{197}Au(n,\gamma)$ Cross Section

MAT = 1166 contains the recommended standard cross section for $^{197}Au(n,\gamma)$ which extends from 10.0 keV to 5.4 MeV. The resonance region radiative capture cross section for MAT = 1166 has not been evaluated in a manner necessary to be considered a standard cross section. The recommended cross section for the above energy range was obtained by F. J. Vaughn and H. A. Grench[20] (Lockheed Palo Alto Research Laboratory).

Vaughn and Grench analyzed the results from 15 differential cross section measurements.[21,35] In addition they used the results from

19 different monoenergetic measurements. The results from each experiment was examined and the data sets were renormalized, when necessary to take into account information which became available after publication. Renormalizations were based on new information about the standards used, in particular the ^{235}U fission cross section.

The complete energy range was divided into three overlapping energy intervals, i.e., 10-150 keV, 123-560 keV, and 400-1800 keV. A least squares polynomial fit was made for each energy interval. An iterative procedure was used to obtain a "best curve" through the experimental data. In the iterative procedure, an adjustment factor for each data set was obtained. Each point in a particular data set was multiplied by its adjustment factor. The adjusted data sets were re-fit and new adjustment factors obtained. This process was continued until the results converged. For example, the final adjustment factors for the data sets measured by Fricke et al.,(21) Poenitz,(22) and Kompe(35) were 1.01567, 1.07871, and 1.11054 respectively.

Figure 6 shows the recommended standard cross section for the energy range from 10 keV to 5 MeV. Also shown are selected sets of experimental results (unadjusted).

G. ^{235}U Fission Cross Section

The CSEWG Normalization and Standards Subcommittee has recommended that the ^{235}U fission cross section from MAT = 1157 be considered as an interim standard for the energy range from 1.0×10^{-5} eV to 15 MeV. It is difficult to estimate the uncertainty in the recommended cross section, but it is considerably greater than the preferred uncertainty of one percent.

The evaluation of the fission cross section for the thermal energy region was made by Leonard.(36) The 2200 m/sec values for the fission cross section and the "Westcott-g factor" were taken from the 1969 IAEA review,[37] i.e., σ_f = 580.2 barns and g_f = 0.9768.

The recommended single level resonance parameters were obtained by Smith and Young[38] for the neutron energy range from 1.0 eV to 82 eV. Several sets of experimental data were used in the fits, including de Saussure et al.,[39] Michaudon,[40] Blons et al.,[41] and Cao et al.[42]

In the energy range from 80 eV to several keV the fission cross section was based on the data of de Saussure et al.[39] Between several keV and 100 keV the results measured by White,[43] Perkin,[44] and Silver et al.[45] were used for absolute values. A fine structure was introduced into the recommended data. This structure was taken from a measurement by Lemley et al.[46]

Above 100 keV, the recommended data were based on measurements by White,[43] Szabo,[47] and Smith et al.[48] (as corrected by Hansen in 1968). Figures 7, 8, and 9 show the recommended fission cross section above 4 keV and selected sets of experimental results.

References

1. L. Stewart, R. J. La Bauve, and P G. Young, "Evaluated Nuclear Data for Hydrogen in the ENDF/B-II Format," LA-4574 (ENDF-141), Feb. 1971.
2. J. C. Hopkins and G. Breit, Nuclear Data, A9, 137 (1971).
3. R. E. Seamon et al., Phys. Rev. 165, 1579 (1968).
4. J. C. Davis and H. H. Barschall, Phys. Rev. C, 3, 1798 (1971).
5. J. Als-Nielsen and O. Dietrich, Phys. Rev. 133B, 925 (1964).
6. J. H. Gibbons and R. L. Macklin, Phys. Rev. 114, 571 (1959).
7. C. A. Uttley et al., "A Review of the Data on the ^{6}Li Cross Sections Below 1.7 MeV." Proc. of a Conf. on Neutron Standards and Flux Normalization, p. 80, Argonne National Laboratory, Oct. 21-23, 1970.
8. C. A. Uttley and K. M. Diment, British Reports AERE-PR/NP-14, 1968, AERE-PR/NP-15, 1969, and AERE-PR/NP-16, 1969.
9. M. G. Sowerby et al., J. Nucl. Eng. 24, 323 (1970).
10. M. G. Sowerby et al., "Recommended Values of the ^{10}B(n,α) Cross Section," Proc. Conf. on Nuetron Standards and Flux Normalization, p. 151, Argonne National Laboratory, Oct. 21-23, 1970.
11. J. Spaepen, Nuclear Data for Reactors, Vol. I, p. 241, IAEA Vienna (1967).
12. F P. Mooring, J. E. Monahan, and C. M. Huddleston, Nuc. Phys. 82, 16 (1967).
13. A. Asami and M. C. Moxon, J. Nucl. Energy 24, 85 (1970).
14. K. M. Diment, "The Average Neutron Total Cross Section of ^{10}B from 100 eV to 10 MeV and the Absorption Cross Section up to 500 keV," AERE-R-5224.
15. N. C. Francis et al., "Carbon as a Cross-Section Standard Between 0 and 2 MeV," Proc. Conf. on Neutron Standards and Flux Normalization, p. 166, Argonne National Laboratory, Oct. 21-24, 1970.
16. S. Cierjacks et al., Report KFK-1000, Supplement 2.

17. R. B. Schwartz, H. T. Heaton, II, and R. A. Schrack, Bull Am. Phys. Soc. 15, 567 (1970).

18. C. A. Uttley, and K. M. Diment, "Analysis of the Total Cross Section of Carbon from 70 eV to 1.5 MeV," Proc. Conf. on Nuclear Data for Reactors, Vol. I, 191, Helsinki, June 15-19, 1970.

19. J. W. Meadows and J. F. Whalen, Nucl. Sci. Eng., 41, 351 (1970).

20. F. J. Vaughn and H. A. Grench (Lockheed Palo Alto Research Laboratory) Private Communication, 1971.

21. M. P. Fricke et al., "Measurement of Cross Sections for the Radiative Capture of 1 keV to 1 MeV by Mo, Rh, Gd, Ta, W, Re, Au, and ^{238}U," Proc. Conf. on Nuclear Data for Reactors, Vol. II, p. 265, Helsinki June 15-19, 1970.

22. W. P. Poenitz et al., J. Nucl. Energy 22, 505 (1960).

23. J. F. Barry, J. Nucl. Energy 18, 491 (1964).

24. K. K. Harris et al., Nucl. Phys. 69, 37 (1965).

25. S. A. Cox, Phys. Rev. 122, 1280 (1961).

26. A E. Johnsrud et al., Phys. Rev. 116, 987 (1959).

27. H. A. Grench, Private Communication, 1966.

28. B. C. Diven et al., Phys. Rev. 120, 356 (1960).

29. J. A. Miskel et al., Phys. Rev. 128, 2717 (1962).

30. I. Bergqvist, Arkiv Fysik 23, 425 (1963).

31. E. G. Bilpuch, L. W. Weston, and H. W. Newson, Ann. Phys. 10, 455 (1960).

32. J. H. Gibbons et al., Phys. Rev. 122, 182 (1961).

33. M. C. Moxon and E. R. Rae, Nucl. Instr. and Meth. 24, 445 (1963).

34. L. M. Spitz, E. Barnard, and F. D. Brooks, Nucl. Phys. A121, 655 (1960).

35. D. Kompe, Nucl. Phys. A133, 513 (1969).

36. B. R. Leonard, Jr., "Thermal Cross Sections of the Fissile and Fertile Nuclei for ENDF/B-II," BNWL-1586 (ENDF-153) June 1971.

37. G. C. Hanna et al., Atomic Energy Review 7, 3 (1969).

38. J. R. Smith and R. C. Young, "^{235}U Resolved Resonance Parameters for ENDF/B, Version III," ANCR-1044 (ENDF-161) Dec. 1971.

39. G. de Saussure et al., "Simultaneous Measurements of the Neutron Fission and Capture Cross Sections for U-235 for Incident Neutron Energies from 0.4 eV to 3 keV," ORNL-TM-1804 (1967).

40. A. Michaudon, "Contribution a l'Etude par des Methodes du Temps de Vol de l'Interaction des Neutrons Lents Ave. l'U-235," CEA-R-2552 (1964).

41. J. Blon, H. Derrien, and A. Michanon, "Measurements and Analysis of the Fission Cross Section of ^{233}U and ^{235}U for Neutron Energies Below 30 keV," Proc. Conf. on Neutron Cross Sections and Technology, Vol. 2, 829, Knoxville, March 15-17, 1971.

42. M. G. Cao et al., J. Nucl. Energy 22, 211 (1968).

43. P. H. White, J. Nucl. Energy A/B 19, 325 (1965).

44. J L. Perkin et al., J. Nucl. Energy A/B 19, 423 (1965).

45. E. G. Silver et al., "Measurement of Neutron Cross Sections in ^{238}U and ^{235}U Below 100 keV at the ORELA," Proc. Conf. on Neutron Cross Section and Technology, Vol. 2, 728, Knoxville, March 15-17, 1971.

46. J. R. Lemley, G. A. Keyworth, and B. C. Diven, Nuc. Sci. Eng. 43, 281 (1971).

47. I. Szabo et al., "New Absolute Measurement of the Neutron-Induced Fission Cross Sections of ^{235}U, ^{239}Pu, ^{241}Pu from 17 keV to 1 MeV,"

Proc. Conf. on Neutron Standards and Flux Normalization, p 257 Argonne, Oct. 21-23 (1970).

48. R. K. Smith, R. L Henkel, and R. A. Nobles, Bull. Am. Phys. Soc. 2 196 (1957).

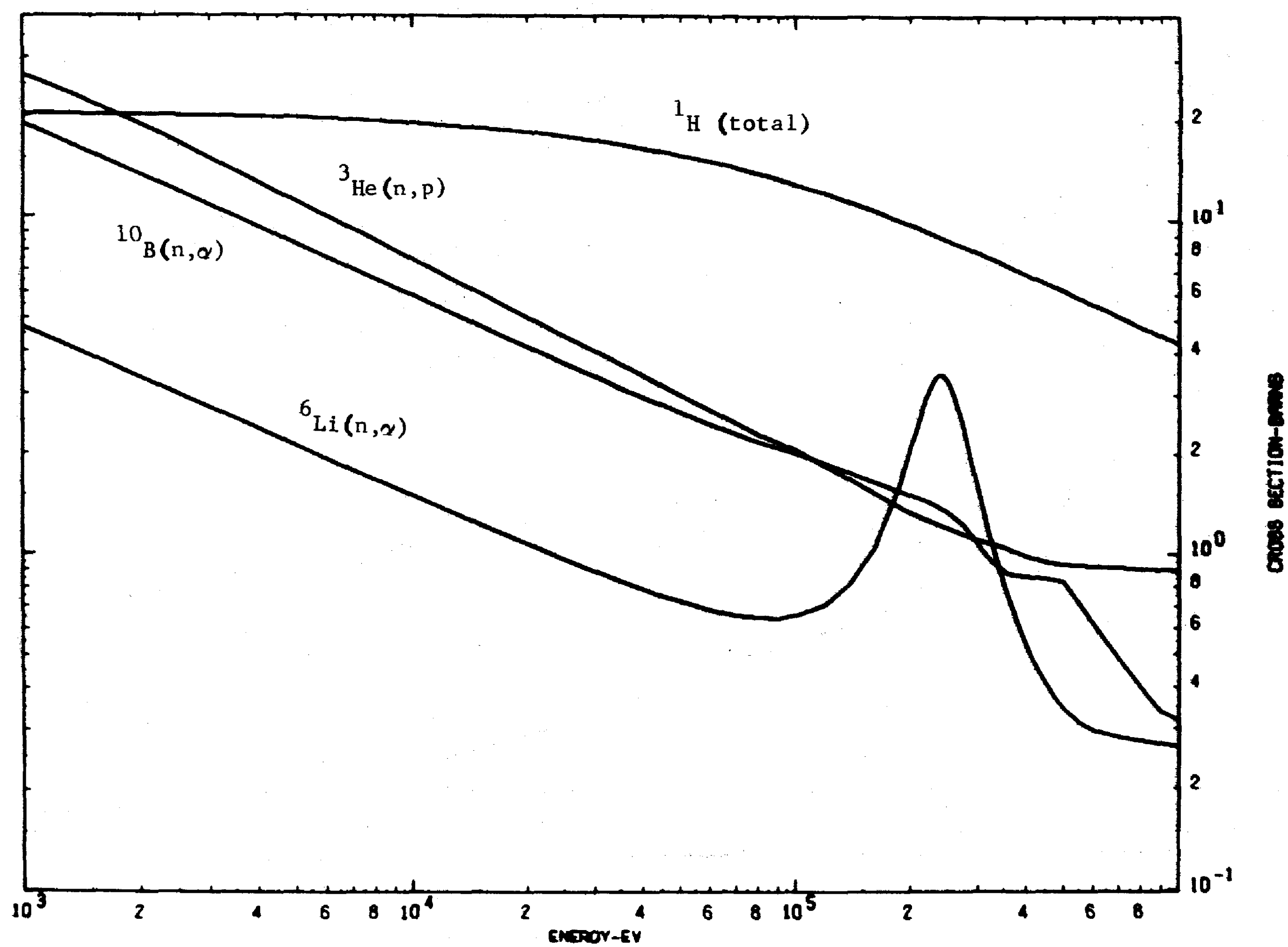


Figure 1: Cross Section Stand. rds

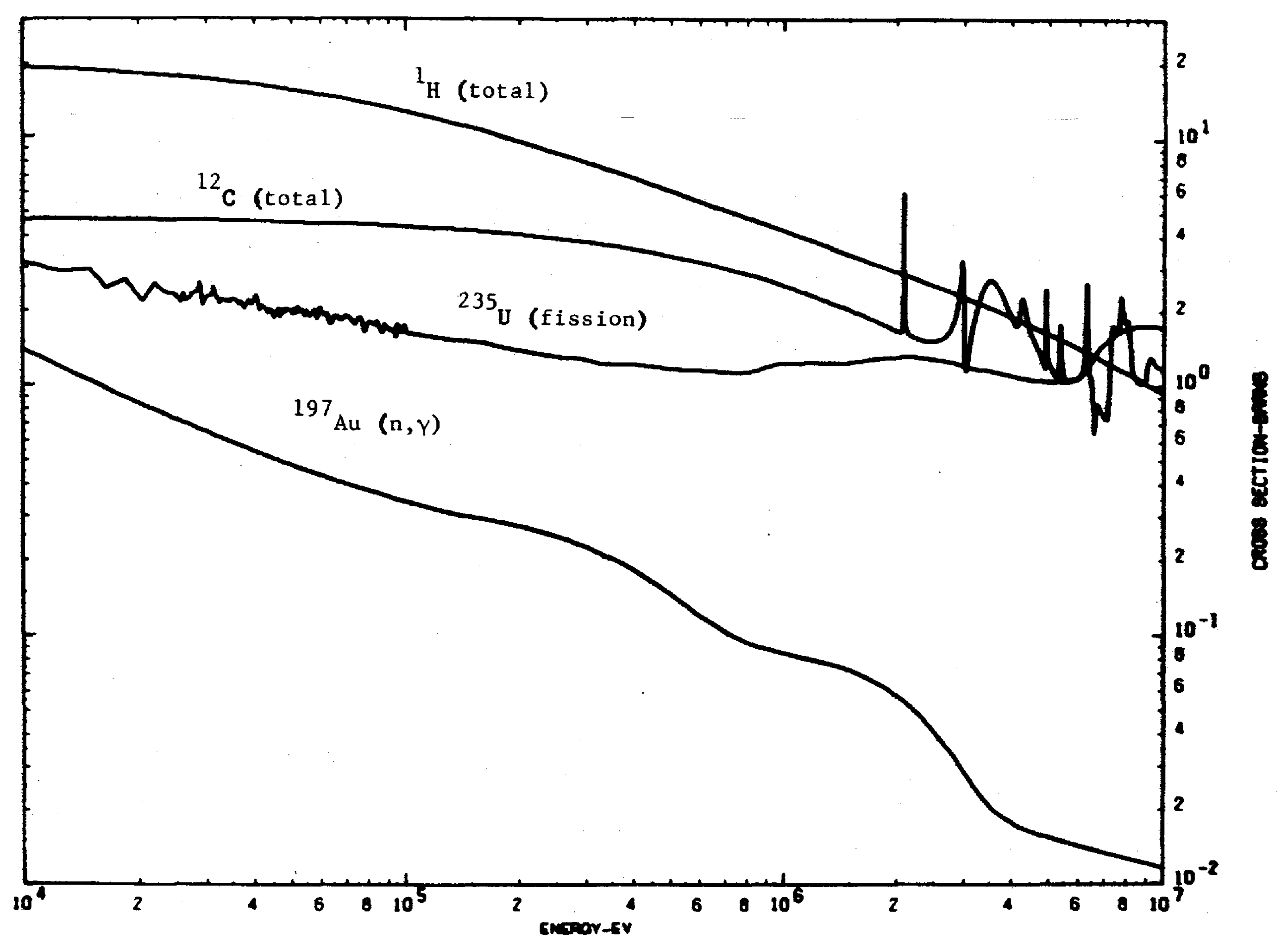


Figure 2: Cross Section Standards

Figure 3: ^{1}H TOTAL CROSS SECTION

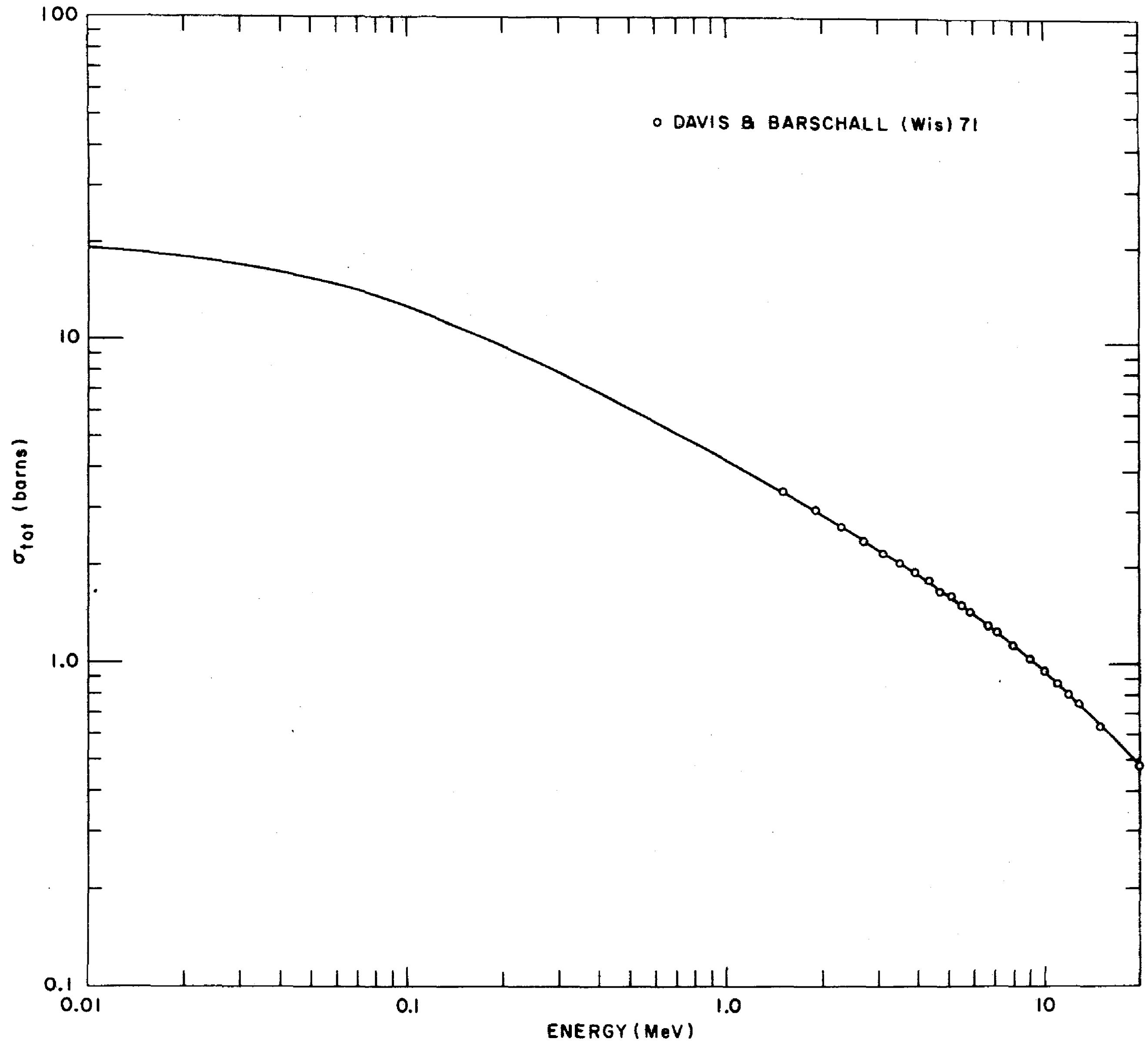


Figure 4: ^{3}He (n,p) CROSS SECTION

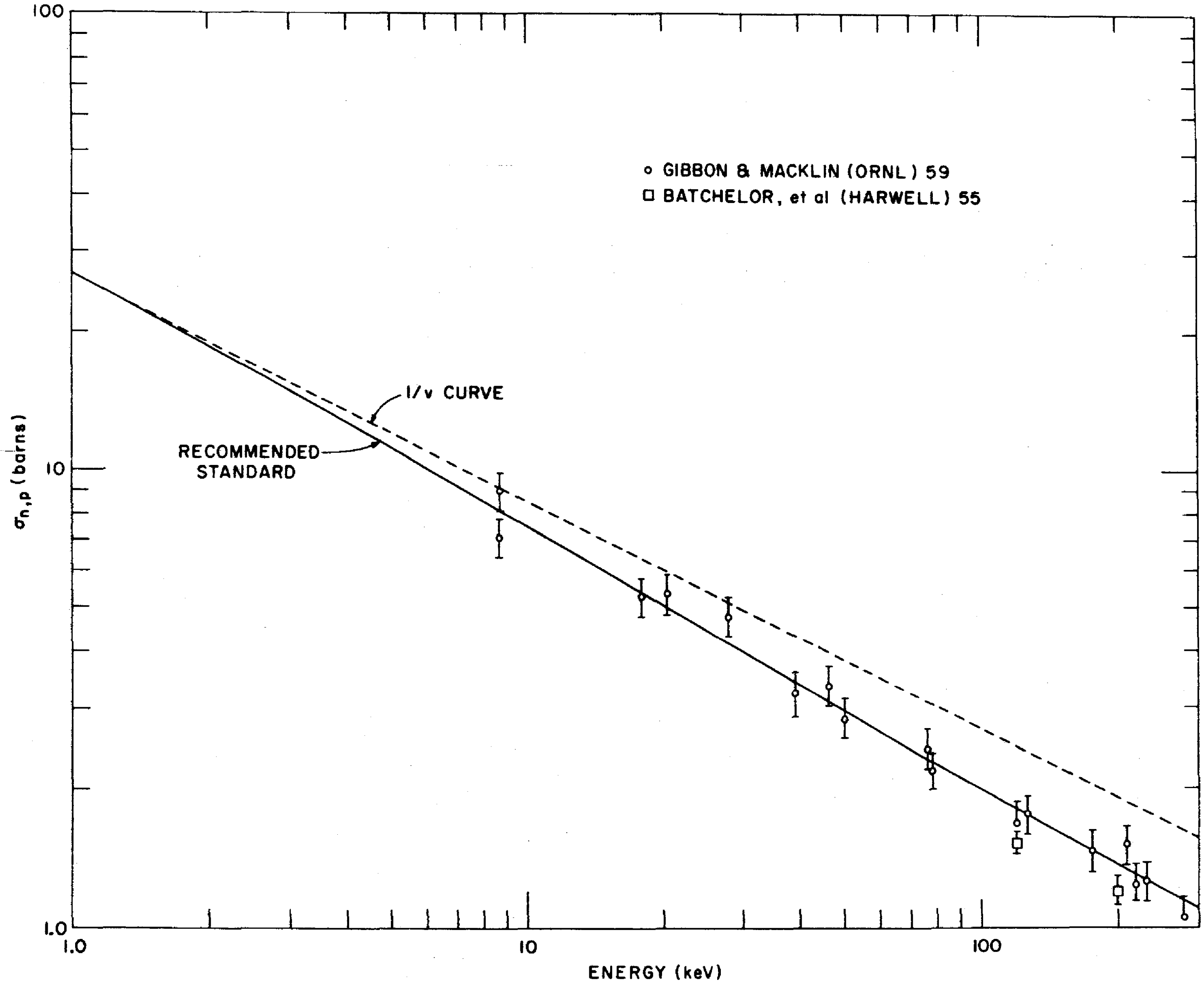
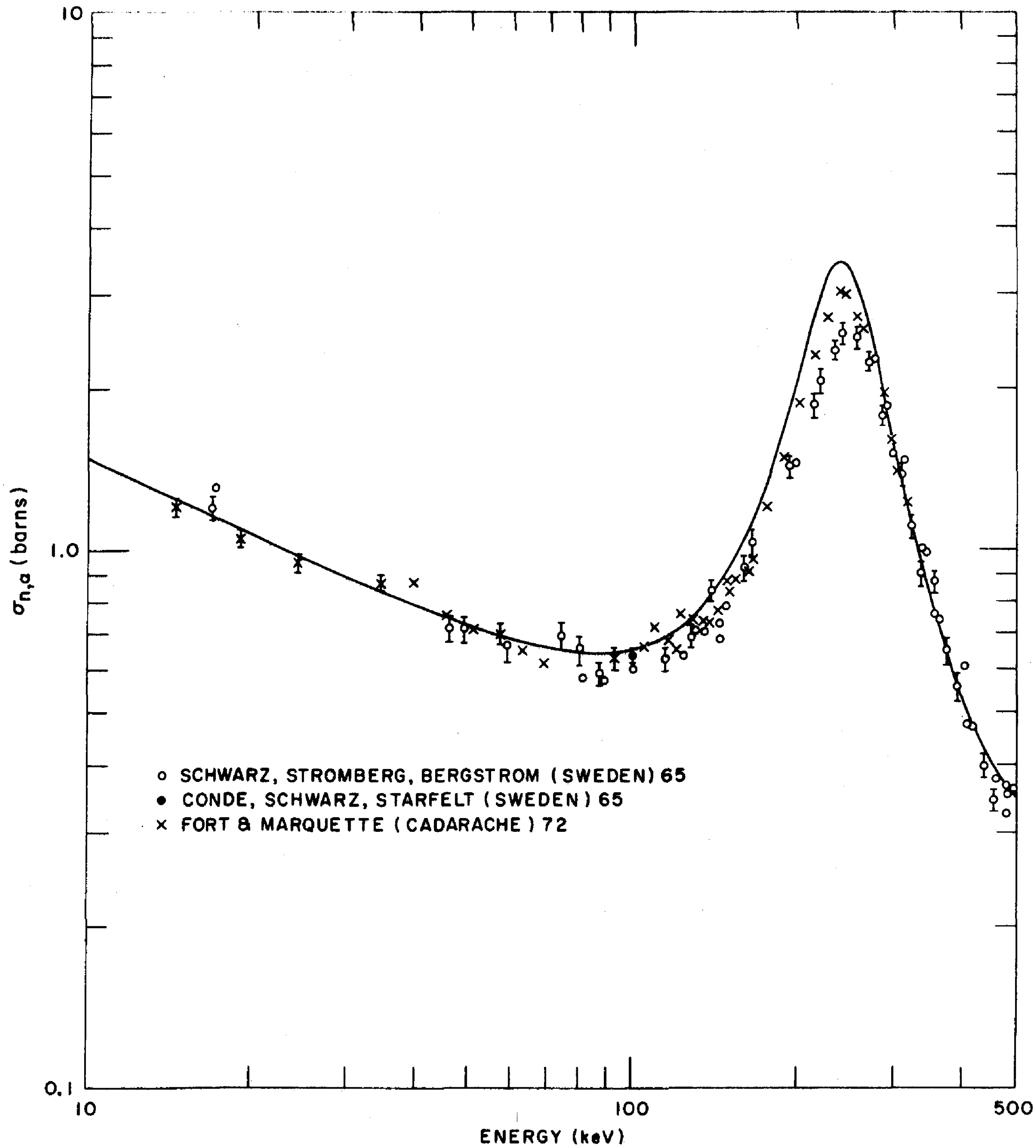


Figure 5: $^{6}Li(n,\alpha)^{3}H$ CROSS SECTION



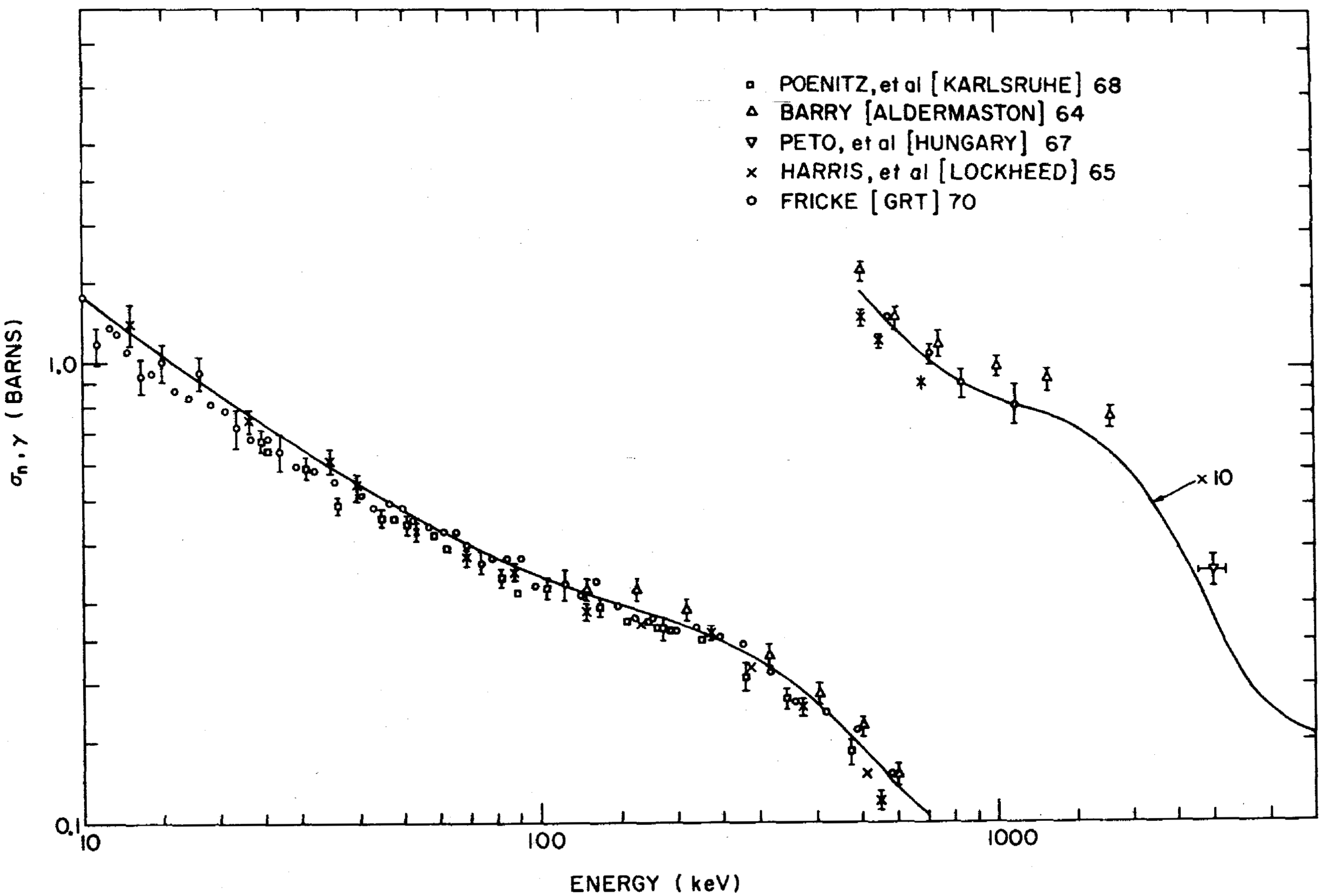


Figure 6: ^{197}Au (n,γ) Cross Section

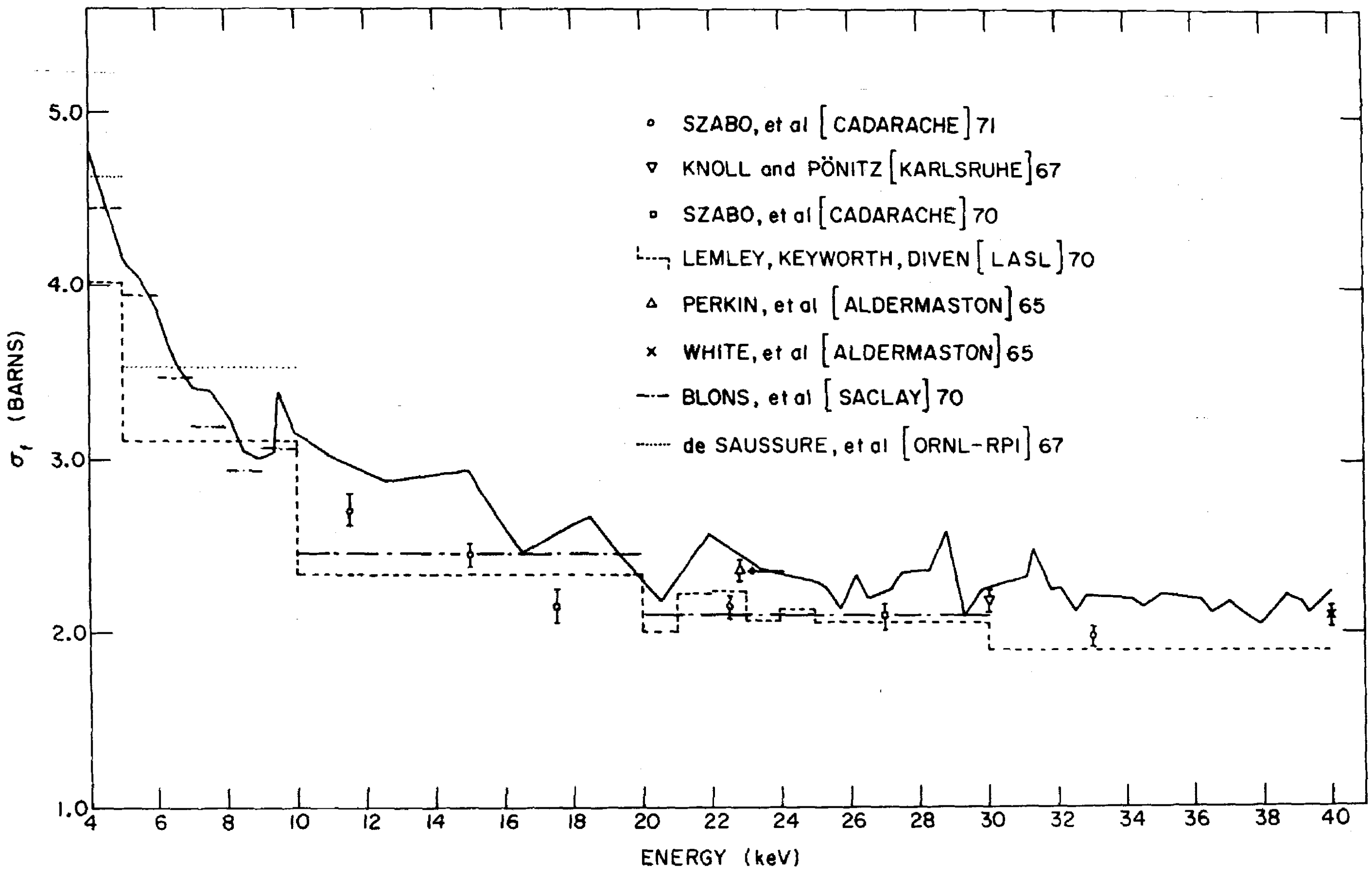


Figure 7: ^{235}U Fission Cross Section

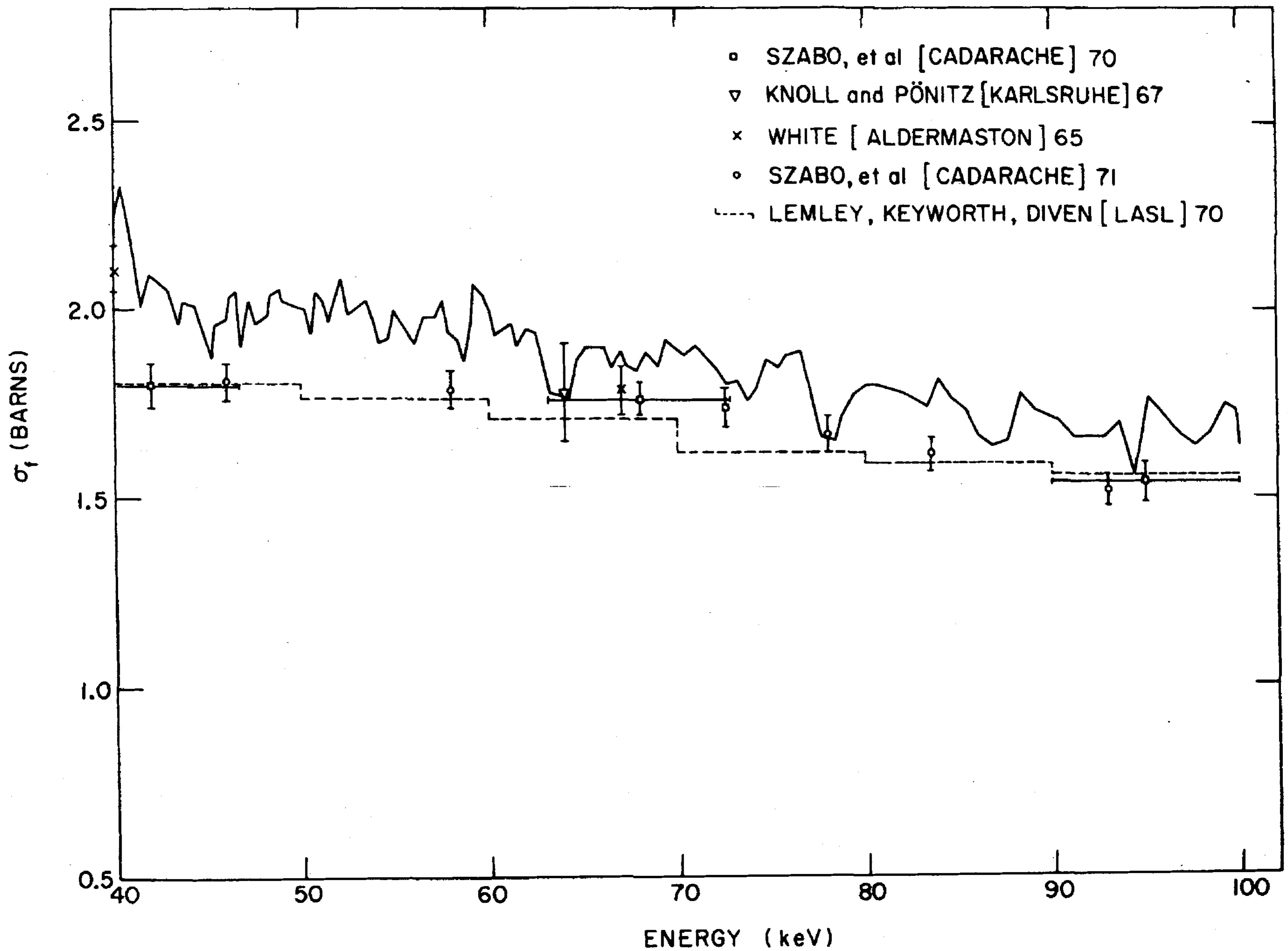


Figure 8: ^{235}U Fission Cross Section

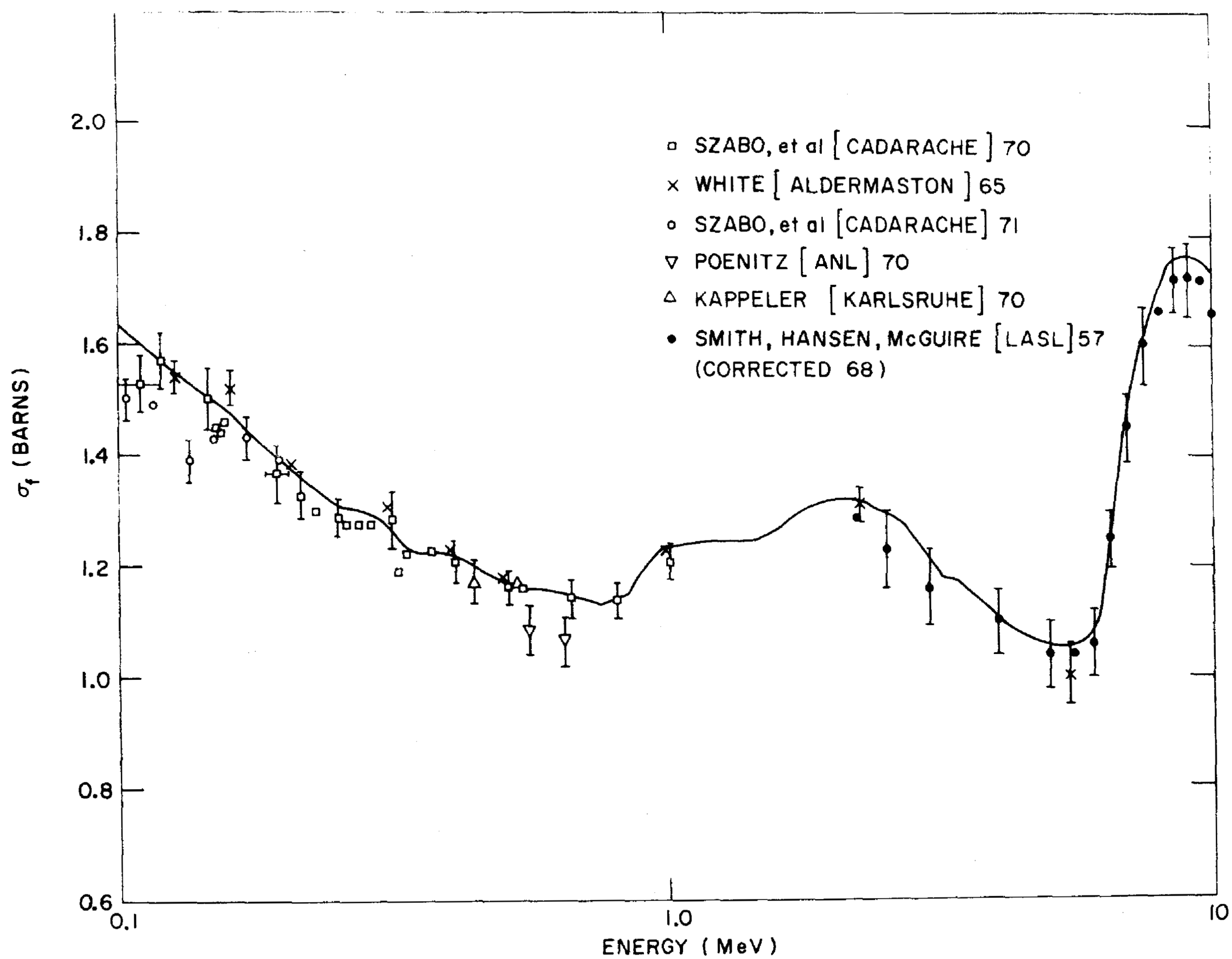


Figure 9: ^{235}U Fission Cross Section

Appendix A

Hydrogen

Written: December 1970
Distributed: February 1971

LA-4574
ENDF-141
EANDC(US)-151-A
UC-34, PHYSICS
TID-4500

LOS ALAMOS SCIENTIFIC LABORATORY
of the
University of California
LOS ALAMOS • NEW MEXICO

Evaluated Nuclear Data for Hydrogen in the ENDF/B-II Format

by

L. Stewart
R. J. LaBauve
P. G. Young

EVALUATED NUCLEAR DATA FOR HYDROGEN IN THE ENDF/B-II FORMAT

by

L. Stewart, R. J. LaBauve, and P. G. Young

ABSTRACT

The following nuclear data are given for hydrogen in the energy range from 1.0×10^{-5} eV to 20.0 MeV.

File 1. The general information file includes a brief description of the data to follow.

File 2. Values for nuclear spin and effective scattering radius are given in the resonance file.

File 3. Smooth cross-section data are given for the total cross section, the free-atom elastic scattering cross section, and the radiative capture cross section; data for $\bar{\mu}$, ξ, and γ are also included.

File 4. The angular distributions for elastic scattering are given as probability vs cosine of the scattering angle.

File 7. The free-atom-scattering cross section is the only information provided at thermal.

File 12. Secondary gamma-ray production multiplicities for capture, which are equal to one, are given in this file.

File 14. Gamma-ray angular distributions are provided for the single radiative capture gamma ray.

INTRODUCTION

This evaluation for hydrogen (MAT = 1148) differs from the previous ENDF/B evaluation (MAT = 1001) in that the elastic scattering data were taken from recent work by Hopkins and Breit[1] and the data for radiative capture were taken from recent work by Horsley.[2] Also, gamma-ray production data, not given in the MAT = 1001 evaluation, are included. A complete listing for MAT = 1148 is given in the Appendix.

FILE 1: GENERAL INFORMATION

A brief summary of the data to follow is given in File 1. The atomic mass for hydrogen was taken to be 1.007825 from the May 1969 "Chart of the Nuclides."[3]

FILE 2: RESONANCE INFORMATION

Nuclear spin and effective scattering radius are given in this file. An effective scattering radius of 1.2756×10^{-12} cm is consistent with a potential scattering cross section of 20.449 b, as determined from $4\pi a^2$. Singlet and triplet scattering radii are not included.

FILE 3: SMOOTH CROSS SECTIONS

Total cross sections (MT = 1) were obtained by adding the elastic scattering and radiative capture cross sections at all energies (1.0×10^{-5} eV to 20.0 MeV). The hydrogen total cross sections are shown in Fig. 1.

The elastic scattering cross sections (MT = 2) were taken from an extensive theoretical treatment of fast neutron measurements by Hopkins and Breit.[1] In this work, a consistent set of cross sections and angular distributions were obtained by using a set of phase shifts previously determined at Yale University.[4] Tabular values of the elastic scattering cross section are given in Ref. 1 for only a few energies, the two lowest points being 100 and 200 keV. The phase shift program and the Yale phase shifts were provided by Hopkins[1] so that many intermediate points could be calculated. At 0.1 keV, the lowest energy recommended for running this program, the scattering cross section is 20.4488 b. This value is in excellent agreement with the thermal cross section (20.442 ± 0.023 b) derived by Davis and Barschall[5] from a revised value of the effective range obtained by determining the best values of the neutron energies from many experiments below 5 MeV performed since 1950. Therefore, for this evaluation, the free-atom-scattering cross section is assumed to be constant below 100 eV and equal to the value calculated from the Yale phase shifts at 100 eV, giving a thermal cross section of 20.449 b. At higher energies, these theoretical predictions are in excellent agreement with the recent measurements of Davis[6] giving an average value of 0.84 for the square of the deviation for energies below 20.0 MeV. The elastic cross section for hydrogen from 1.0×10^{-5} eV to 20.0 MeV is shown in Fig. 2.

The cross sections for radiative capture (MT = 102) were taken from the 1966 publication of Horsley,[2] where a value of 332 mb was adopted for the thermal value. Deuteron photodisintegration cross sections were also employed in deriving radiative capture in Horsley's report. Although the Nuclear Data article by Horsley[2] was referenced for MAT = 1001, the values were taken from an early version described in AWRE 0-23/65, and these were later revised for the Nuclear Data article. The latter report (Ref. 2) has been used for this evaluation, as suggested by Horsley. The radiative capture cross section for MAT = 1148 from 1.0×10^{-5} eV to 20.0 MeV is shown in Fig. 3.

The average value of the cosine in the laboratory system ($\bar{\mu}_L$) for elastic scattering (MT = 251) was derived from the secondary angular distributions in File 4 (MT = 4). Values for $\bar{\mu}_L$ from 1.0×10^{-5} eV to 20.0 MeV are shown in Fig. 4.

Values for ξ, the average logarithmic energy change per collision (MT = 252), and for γ, the Goertzel-Greuling constant (MT = 253), are taken equal to 1 over the range 1.0×10^{-5} eV to 20.0 MeV, following the MT = 1001 evaluation.

FILE 4: SECONDARY ANGULAR DISTRIBUTIONS

Angular distributions of secondary neutrons resulting from elastic scattering are tabulated from 1.0×10^{-5} eV to 20.0 MeV. Distributions at 0.1, 5, 10, 20, and 30 MeV are provided by Ref. 1; additional and intermediate data were calculated by using the Hopkins-Breit phase shift program and the Yale phase shifts. As shown in Figs. 5 through 16, the angular distributions above 100 keV are neither isotropic below 10 MeV, nor are they symmetric about 90° at higher energies as assumed in the earlier version (MAT = 1001). At 100 keV, the angular distributions are assumed to be isotropic because the 180/0° ratio is very nearly unity (1.0011). At 500 keV, this ratio approaches 1.005; therefore, the pointwise normalized probabilities as a function of the cosine of the scattering angle are provided at 1.0×10^{-5} eV (isotropic), 100 keV (isotropic), 500 keV, and at 1-MeV intervals from 1 to 20 MeV.

FILE 5: THERMAL DATA

Free-atom cross sections specified from 1.0×10^{-5} eV to 5 eV are included in this file.

FILE 12: PHOTON PRODUCTION CROSS SECTIONS

A multiplicity representation is used to describe the single hydrogen radiative capture gamma ray from 1.0×10^{-5} eV to 20.0 MeV. The multiplicity is referred to MT = 102 in File 3 and is unity at all neutron energies. To adequately represent the gamma-ray energy for MeV-incident neutrons, the neutron energy region from 0.2 to 20 MeV is divided into 16 different energy bands, and the gamma-ray energy is tabulated for each neutron energy band as

$$\bar{E}_\gamma = 2.225 \times 10^6 + \bar{E}_n/2 \qquad \text{(eV)} ,$$

where $\bar{E}_n$ is the neutron energy at the midpoint of the band in eV. The value 2.225×10^6 eV corresponds to the deuteron binding energy; that is, the small energy change due to the nuclear recoil that accompanies gamma emission has been ignored.

FILE 14: GAMMA-RAY ANGULAR DISTRIBUTIONS

The gamma-ray angular distributions are assumed to be isotropic at all neutron energies from 1.0×10^{-5} eV to 20.0 MeV.

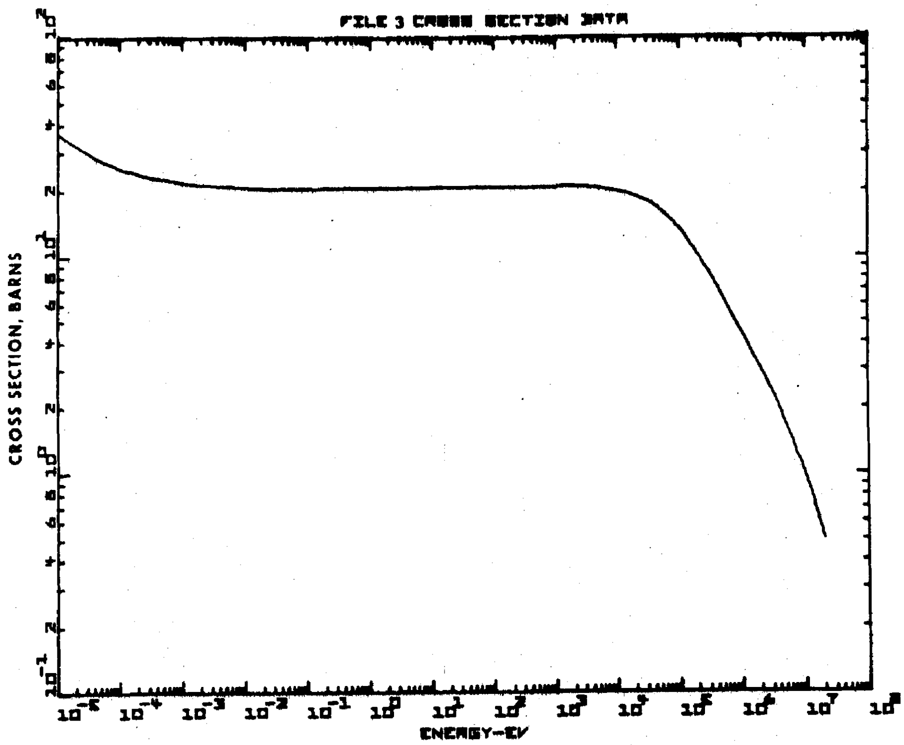


Fig. 1. Total cross section (MT = 1).

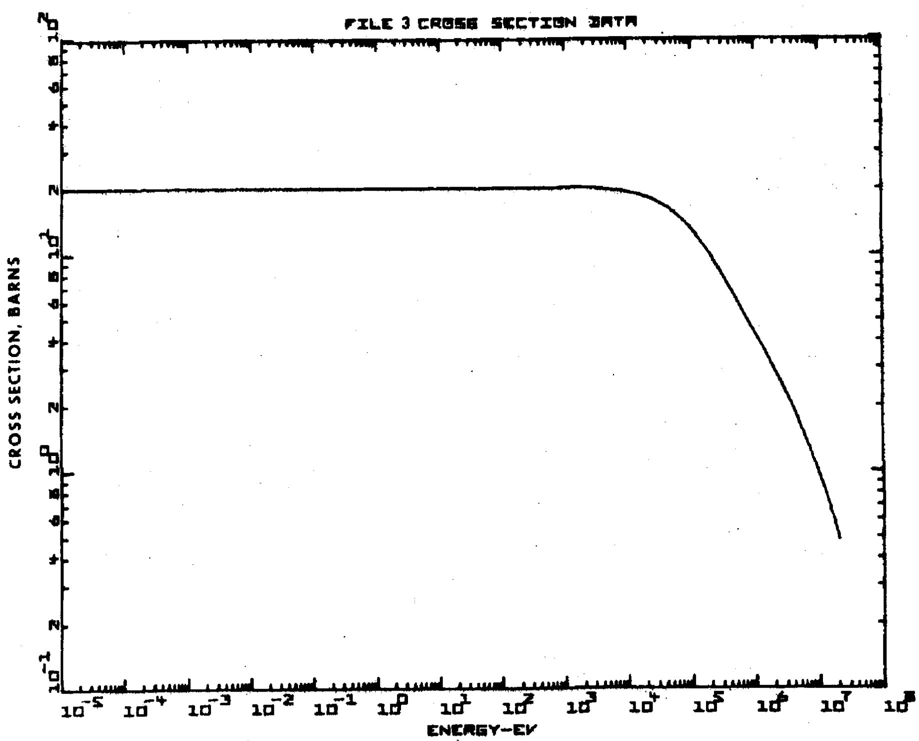


Fig. 2. Elastic scattering cross section (MT = 2).

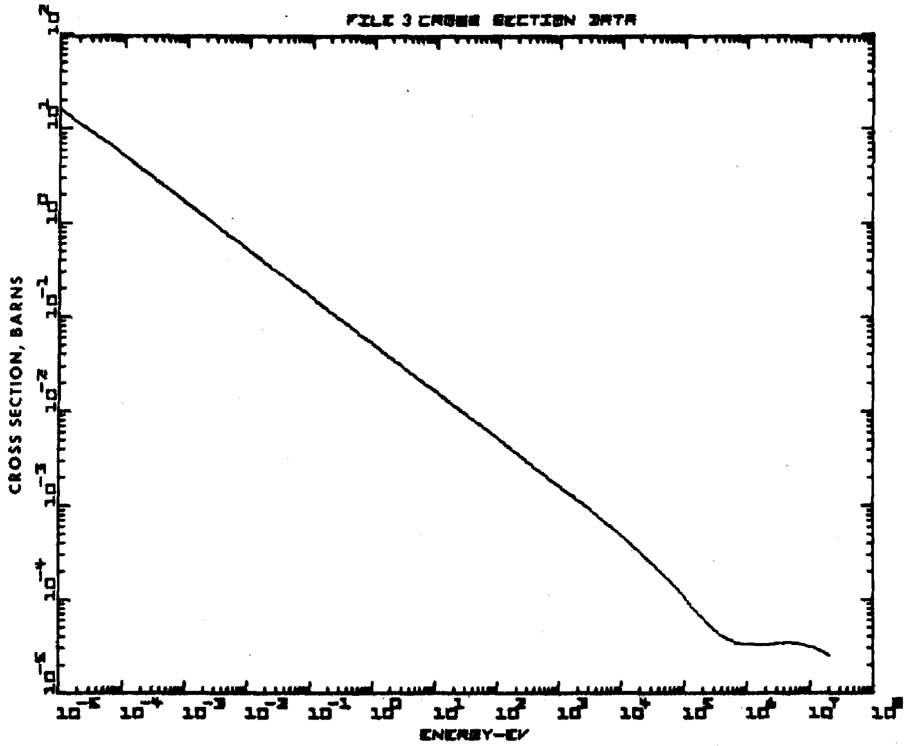


Fig. 3. Radiative capture cross section (MT = 102).

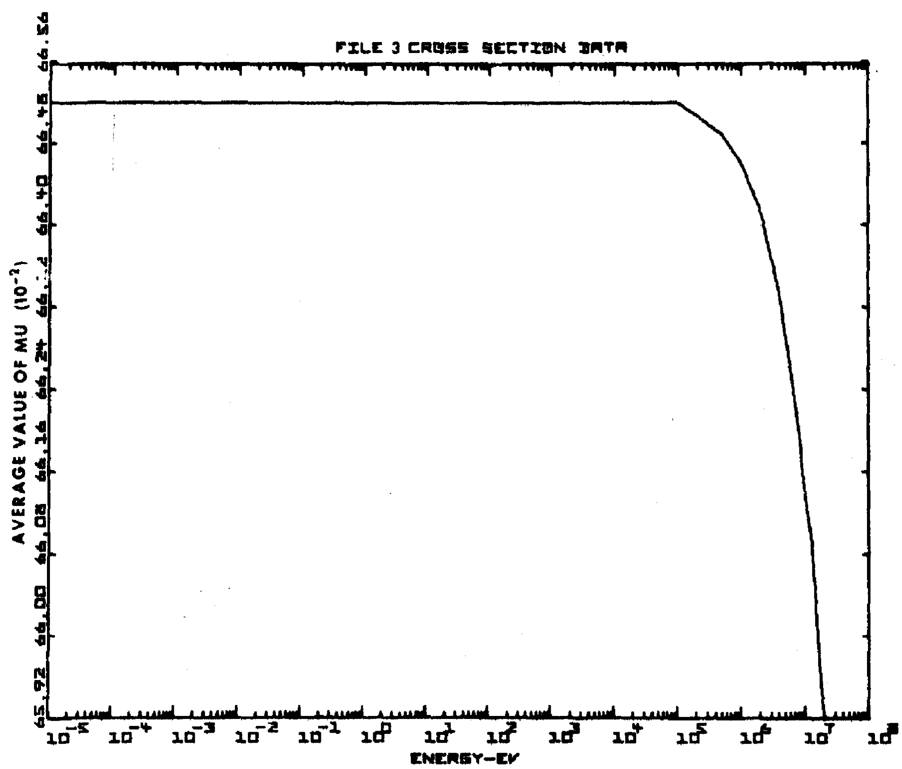


Fig. 4. Average value of cosine in laboratory system (MT = 251).

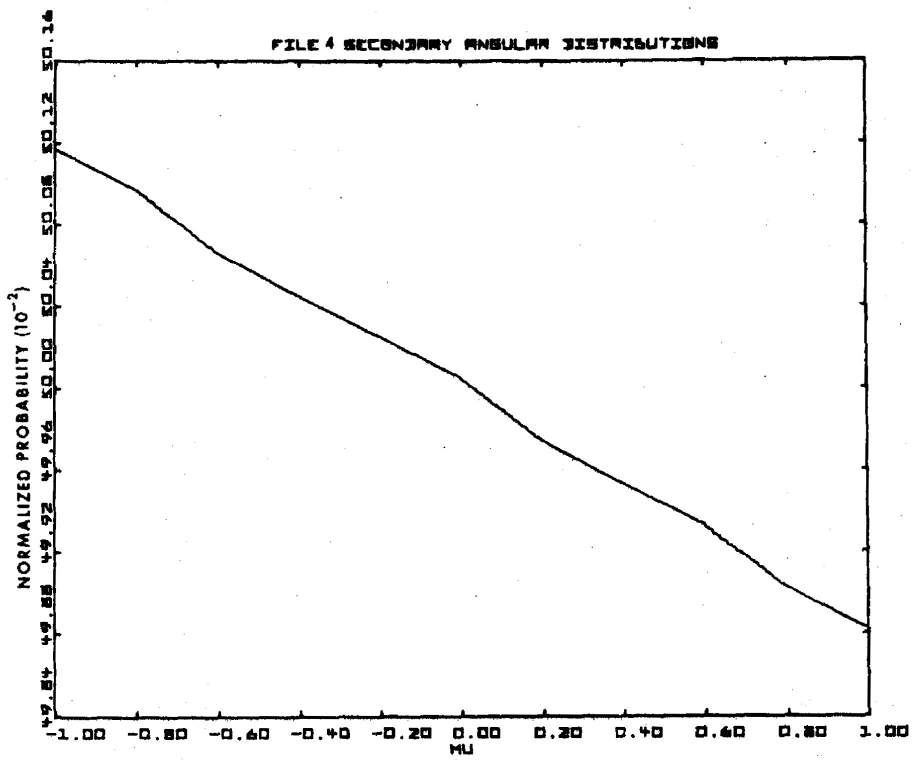


Fig. 5. Angular distribution for 0.5 MeV.

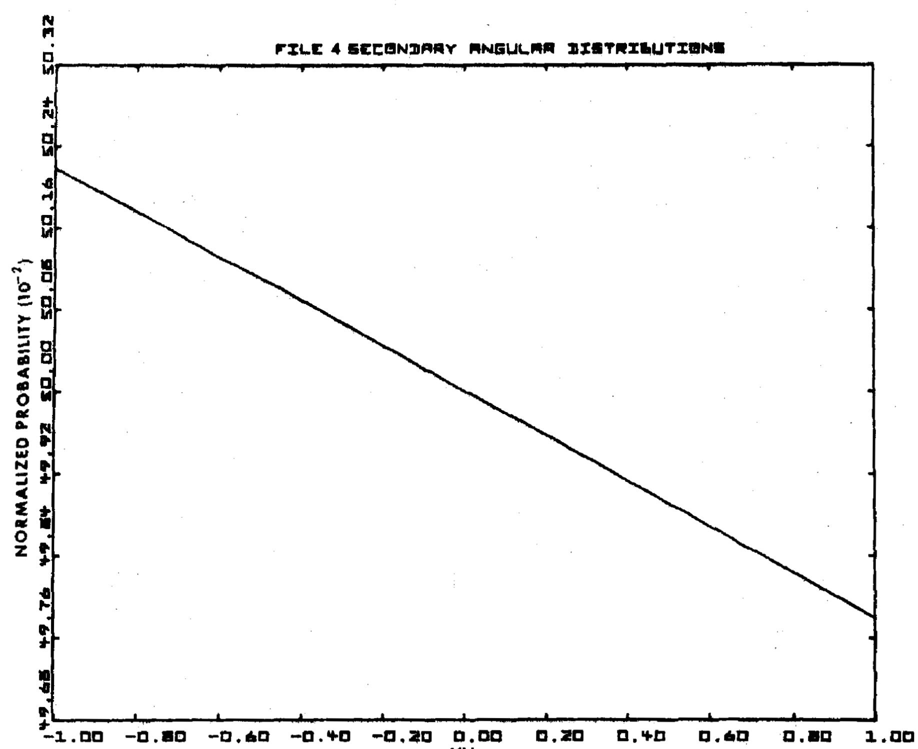


Fig. 6. Angular distribution for 1.0 MeV.

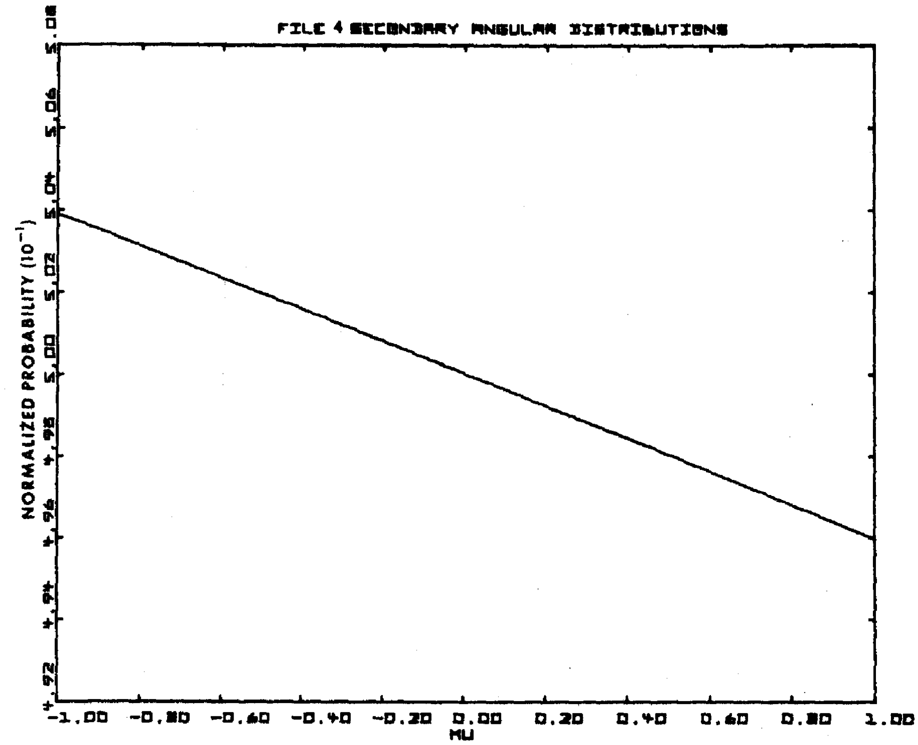


Fig. 7. Angular distribution for 2.0 MeV.

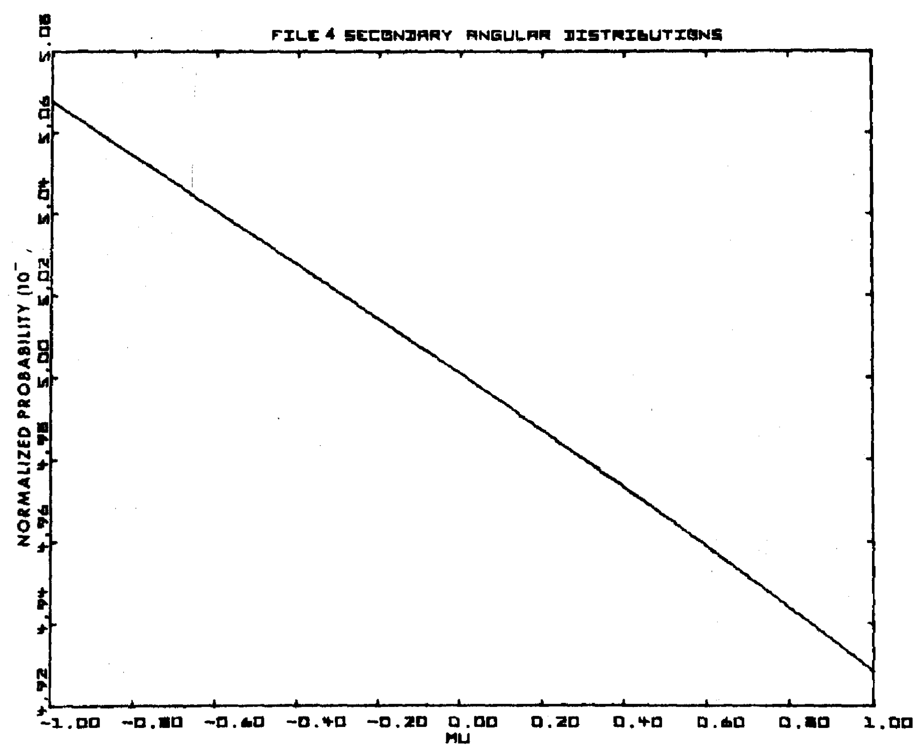


Fig. 8. Angular distribution for 4.0 MeV.

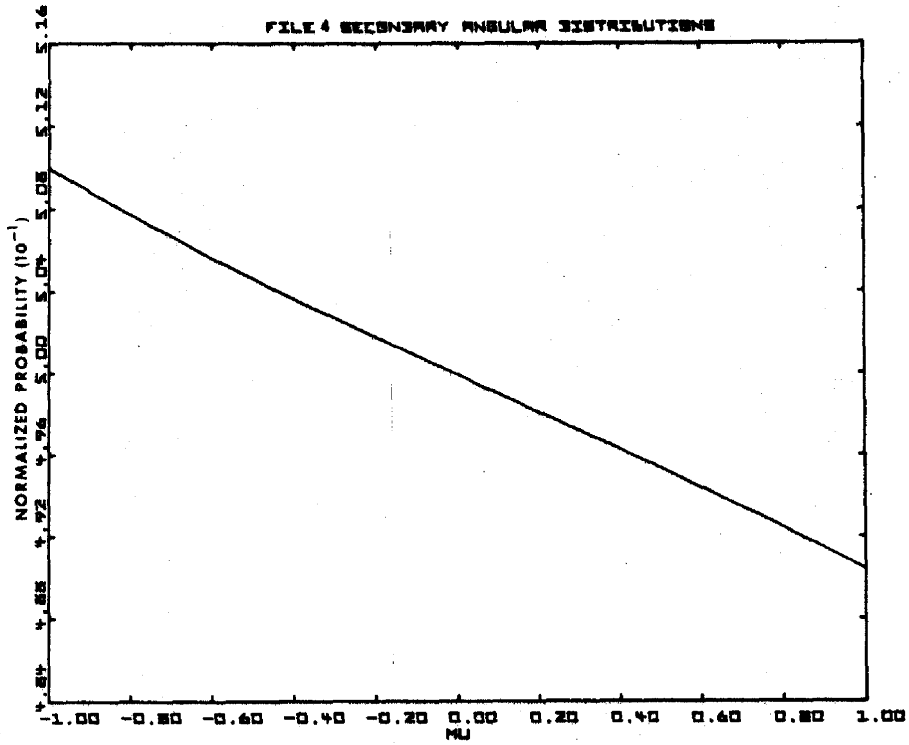


Fig. 9. Angular distribution for 6.0 MeV.

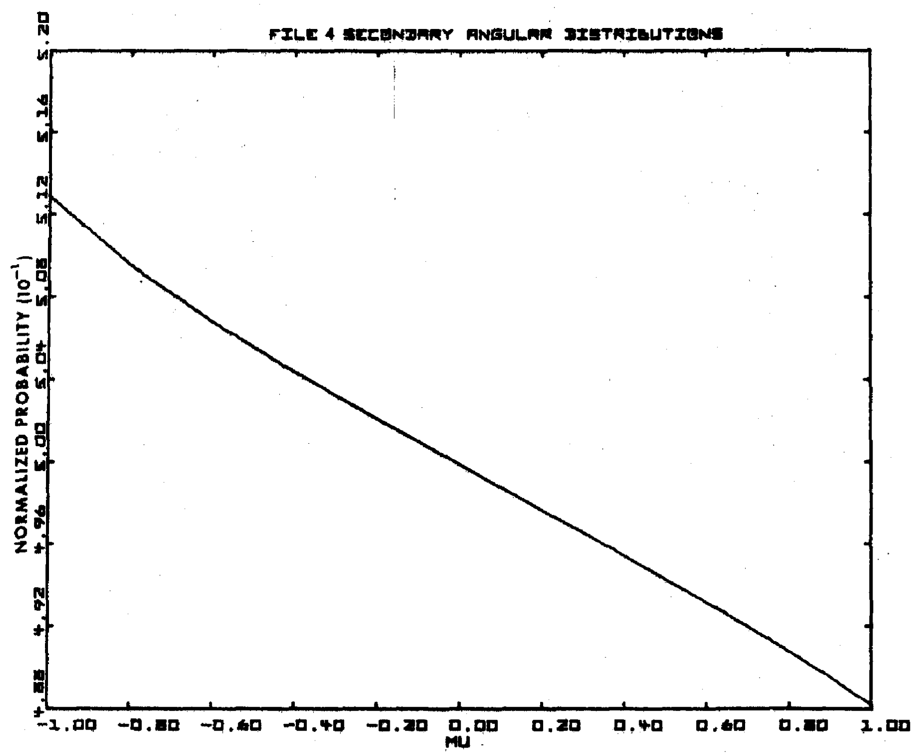


Fig. 10. Angular distribution for 8.0 MeV.

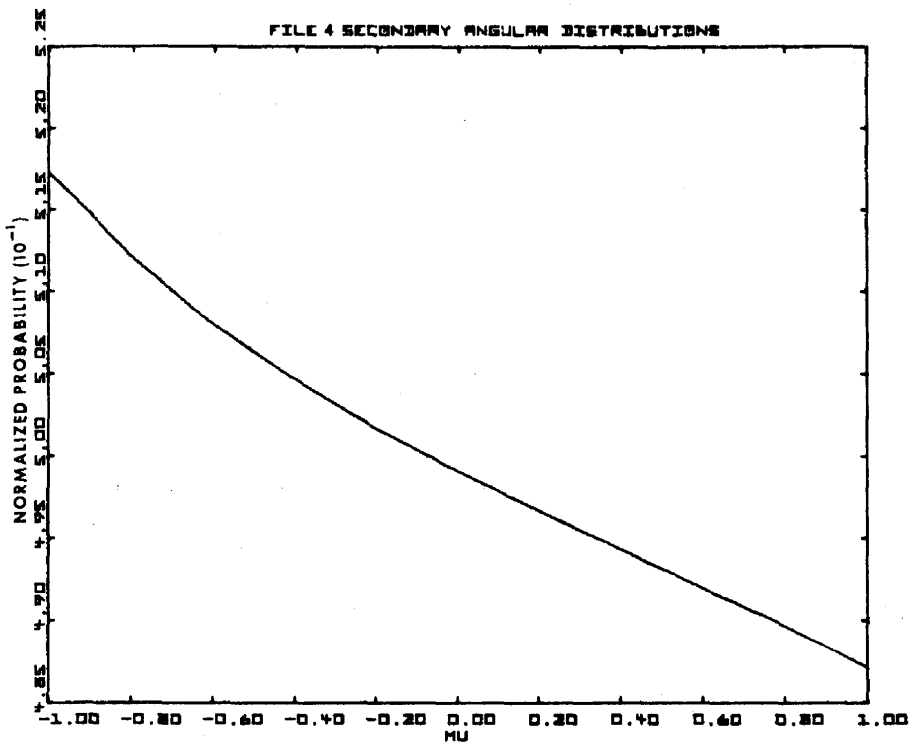


Fig. 11. Angular distribution for 10.0 MeV.

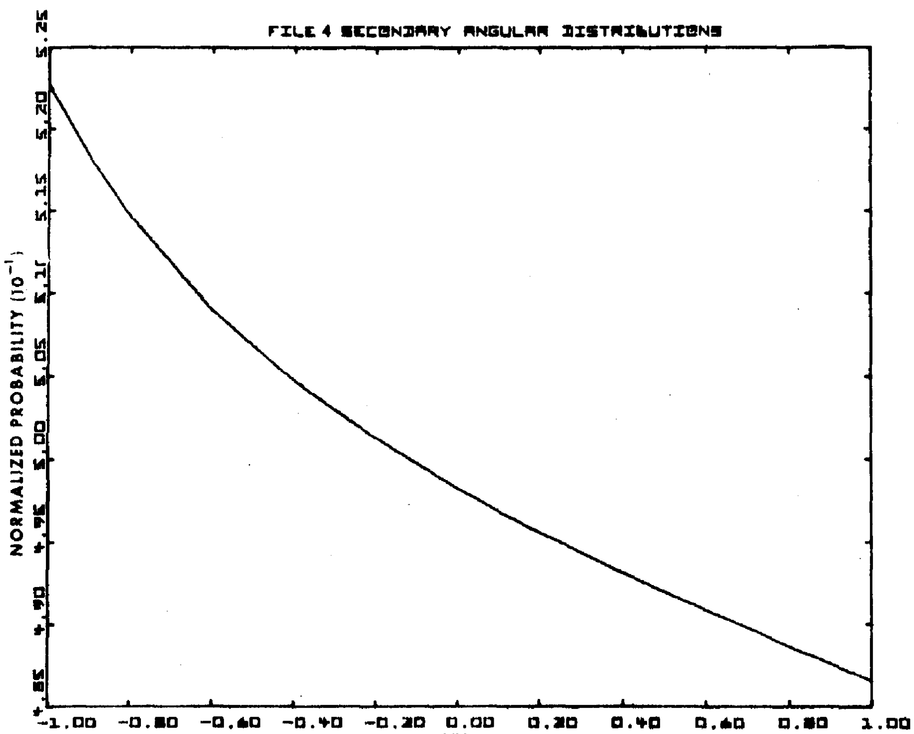


Fig. 12. Angular distribution for 12.0 MeV.

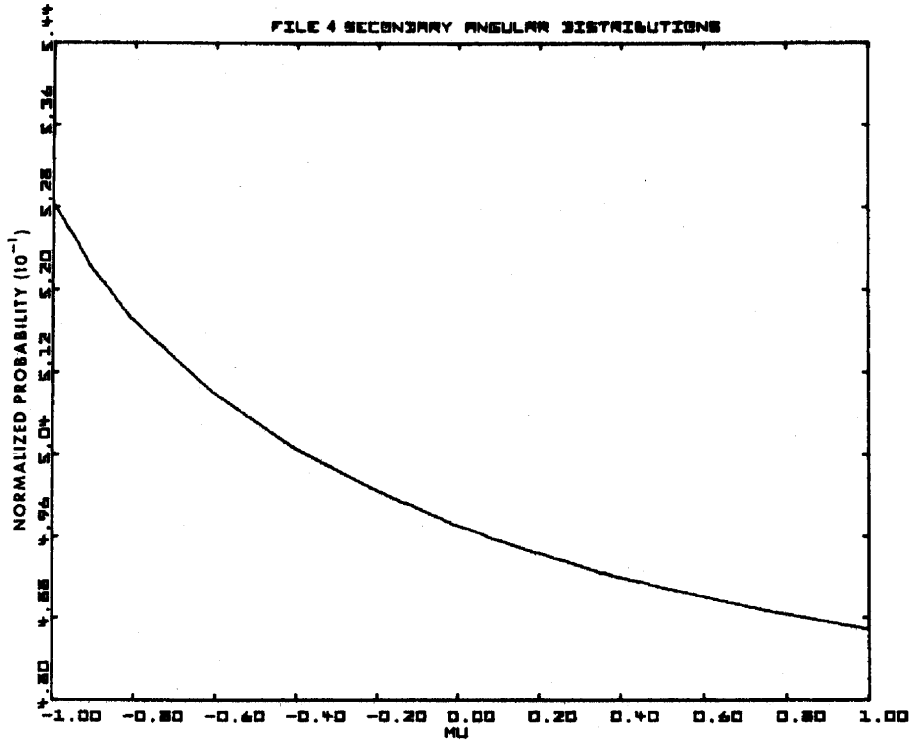


Fig. 13. Angular distribution for 14.0 MeV.

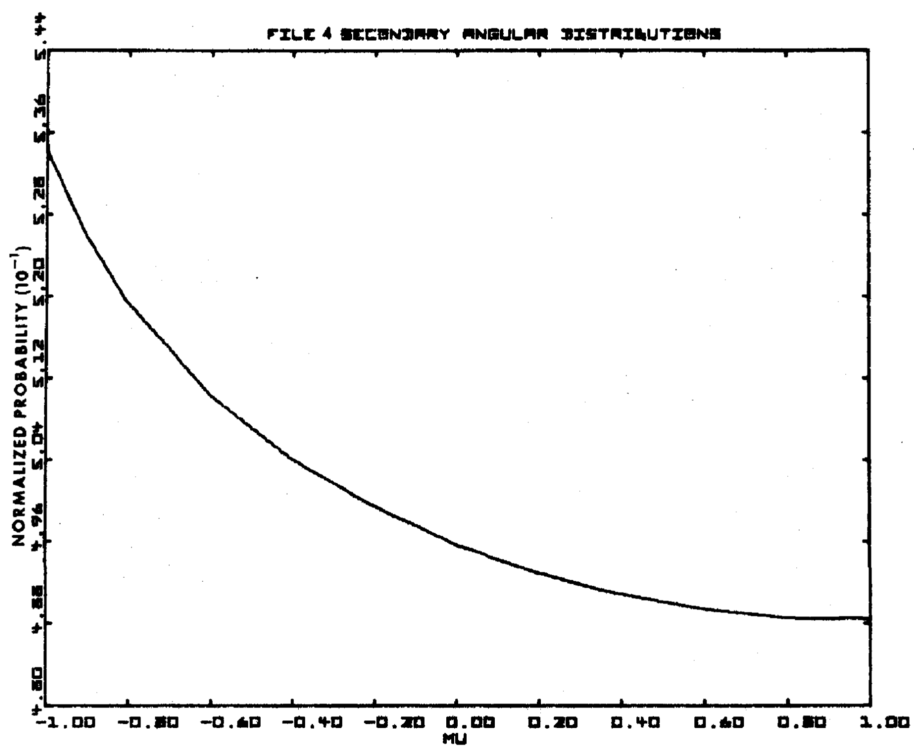


Fig. 14. Angular distribution for 16.0 MeV.

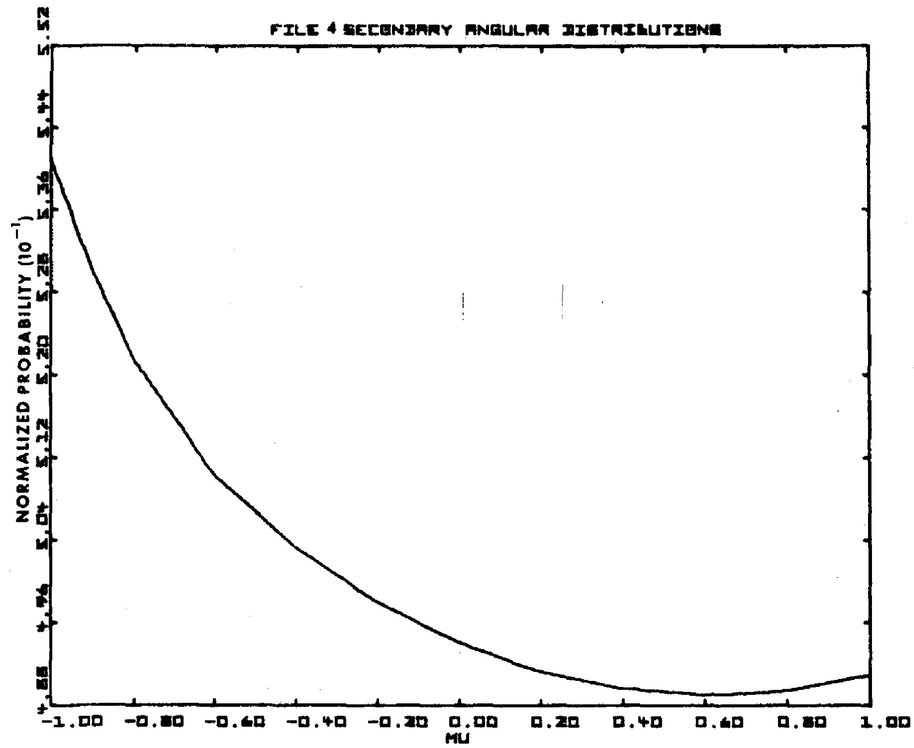


Fig. 15. Angular distribution for 18.0 MeV.

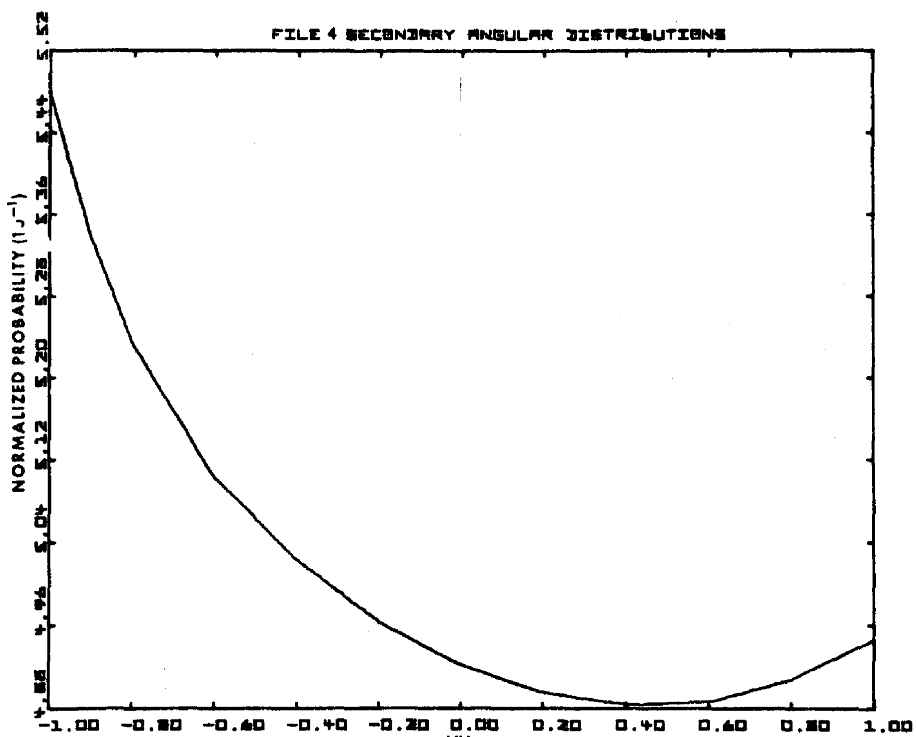


Fig. 16. Angular distribution for 20.0 MeV.

REFERENCES

1. J. C. Hopkins and G. Breit, "The H(n,n)H Scattering Observables Required for High Precision Fast-Neutron Measurements," submitted to Nuclear Data for publication; private communication prior to publication (1970).

2. A. Horsley, "Neutron Cross Sections of Hydrogen in the Energy Range 0.001 eV - 20 MeV," Nuclear Data A 2, 243 (1966) and private communication (1970).

3. "Chart of the Nuclides," Battelle Memorial Institute, May 1969.

4. R. E. Seamon, K. A. Friedman, G. Breit, R. D. Haracz, J. M. Holt, and A. Prakash, Phys. Rev. 165, 1579 (1968).

5. J. C. Davis and H. H. Barschall, "Adjustments in the n-p Singlet Effective Range," Phys. Letters 27B, 636 (1968).

6. J. C. Davis, BAPS II 15, 474 (1970) and private communication (1970).

APPENDIX

LISTING OF HYDROGEN EVALUATION

```
      HYDROGEN IN ENDF/B-II FORMAT.  MAT = 1148                          1148-0 -0     0
  1.001 E+03 9.99917E-01          0          0          0             121148 1451     1
 0.0        0.0                   0          0         67              01148 1451     2
                                                                        1148 1451     3
    HYDROGEN FREE ATOM CROSS SECTIONS                                   1148 1451     4
      ENTRY BY L. STEWART. R.J. LABAUVE. AND P.G. YOUNG                 1148 1451     5
      LOS ALAMOS SCIENTIFIC LABORATORY                                  1148 1451     6
      LOS ALAMOS.NEW MEXICO 87544                                       1148 1451     7
      OCTOBER 20.1970                                                   1148 1451     8
                                                                        1148 1451     9
  MF=1                                                                  1148 1451    10
                                                                        1148 1451    11
   MT=451. ATOMIC MASS=1.007825                                         1148 1451    12
                                                                        1148 1451    13
  MF=2                                                                  1148 1451    14
                                                                        1148 1451    15
   MT=151. SCATTERING LENGTH=1.2756E-12 CM.                             1148 1451    16
                                                                        1148 1451    17
  MF=3                                                                  1148 1451    18
                                                                        1148 1451    19
   MT=  1. TOTAL CROSS SECTIONS --- THE TOTAL CROSS SECTIONS ARE        1148 1451    20
           OBTAINED BY ADDING THE ELASTIC SCATTERING AND                1148 1451    21
           RADIATIVE CAPTURE CROSS SECTIONS AT ALL ENERGIES.            1148 1451    22
            1.0E-05 EV TO 20 MEV.                                       1148 1451    23
                                                                        1148 1451    24
   MT=  2. ELASTIC SCATTERING --- FROM AN EXTENSIVE THEORETICAL         1148 1451    25
           TREATMENT OF FAST NEUTRON MEASUREMENTS                       1148 1451    26
           BY J. C. HOPKINS(LASL) AND G. BREIT(STATE                    1148 1451    27
           UNIVERSITY OF NEW YORK).                                     1148 1451    28
            1.0E-05 EV TO 20 MEV.                                       1148 1451    29
                                                                        1148 1451    30
   MT=102. RADIATIVE CAPTURE --- THESE CROSS SECTIONS ARE TAKEN         1148 1451    31
           FROM THE 1966 PUBLICATION OF A. HORSLEY WHERE A VALUE        1148 1451    32
           OF 332 MB WAS ADOPTED FOR THE THERMAL VALUE.                 1148 1451    33
            1.0E-05 EV TO 20 MEV.                                       1148 1451    34
                                                                        1148 1451    35
   MT=251. AVERAGE VALUE OF COSINE OF SCATTERING ANGLE                  1148 1451    36
           IN LAB SYSTEM.                                               1148 1451    37
            1.0E-05 EV TO 20 MEV.                                       1148 1451    38
                                                                        1148 1451    39
   MT=252. AVERAGE LOGARITHMIC ENERGY CHANGE PER COLLISON. TAKEN        1148 1451    40
           AS 1. FROM 1.0E-05 EV TO 20 MEV.                             1148 1451    41
                                                                        1148 1451    42
   MT=253. GAMMA. TAKEN AS 1. FROM 1.0E-05 EV TO 20 MEV.                1148 1451    43
                                                                        1148 1451    44
  MF=4                                                                  1148 1451    45
                                                                        1148 1451    46
   MT=  2. NEUTRON ELASTIC SCATTERING ANGULAR DISTRIBUTIONS IN          1148 1451    47
           THE CENTER OF MASS SYSTEM--GIVEN AS NORMALIZED               1148 1451    48
           POINTWISE PROBABILITIES.                                     1148 1451    49
```

```
                                                                  1148 1451   50
                                                                  1148 1451   51
 MF=7                                                             1148 1451   52
                                                                  1148 1451   53
   MT=  4, .00001 TO 5 EV FREE GAS SIGMA=20.449 BARNS.            1148 1451   54
                                                                  1148 1451   55
 MF=12                                                            1148 1451   56
                                                                  1148 1451   57
   MT=102, GAMMA RAY MULTIPLICITIES --- MULTIPLICITY, (REFERRED   1148 1451   58
           TO MT=102, MF=3), IS UNITY AT ALL NEUTRON ENERGIES.    1148 1451   59
           SIXTEEN ENERGY BANDS ARE GIVEN FROM .2 MEV TO 20 MEV,  1148 1451   60
           AND THE AVERAGE GAMMA RAY ENERGY, EAG, IS DETERMINED   1148 1451   61
           FROM THE AVERAGE NEUTRON ENERGY, EAN, IN THE BAND BY   1148 1451   62
           EAG=2.225E+06+EAN/2., RECOIL ENERGY IGNORED.           1148 1451   63
                                                                  1148 1451   64
 MF=14                                                            1148 1451   65
                                                                  1148 1451   66
   MT=102, GAMMA RAY ANGULAR DISTRIBUTION --- ASSUMED ISOTROPIC   1148 1451   67
```

```
     HYDROGEN IN ENDF/B-II FORMAT.  MAT = 1148                    1148-0 -0    0
 1.001 E+03 9.9917E-01          0          0          0          121148 1451    1
0.0        0.0                  0          0         67           01148 1451    2
                                                                  1148 1451    3
   HYDROGEN FREE ATOM CROSS SECTIONS                              1148 1451    4
     ENTRY BY L. STEWART, R.J. LABAUVE, AND P.G. YOUNG            1148 1451    5
     LOS ALAMOS SCIENTIFIC LABORATORY                             1148 1451    6
     LOS ALAMOS,NEW MEXICO 87544                                  1148 1451    7
     OCTOBER 20,1970                                              1148 1451    8
                                                                  1148 1451    9
 MF=1                                                             1148 1451   10
                                                                  1148 1451   11
   MT=451, ATOMIC MASS=1.007825                                   1148 1451   12
                                                                  1148 1451   13
 MF=2                                                             1148 1451   14
                                                                  1148 1451   15
   MT=151, SCATTERING LENGTH=1.2756E-12 CM.                       1148 1451   16
                                                                  1148 1451   17
 MF=3                                                             1148 1451   18
                                                                  1148 1451   19
   MT=  1, TOTAL CROSS SECTIONS --- THE TOTAL CROSS SECTIONS ARE  1148 1451   20
           OBTAINED BY ADDING THE ELASTIC SCATTERING AND          1148 1451   21
           RADIATIVE CAPTURE CROSS SECTIONS AT ALL ENERGIES.      1148 1451   22
            1.0E-05 EV TO 20 MEV.                                 1148 1451   23
                                                                  1148 1451   24
   MT=  2, ELASTIC SCATTERING --- FROM AN EXTENSIVE THEORETICAL   1148 1451   25
           TREATMENT OF FAST NEUTRON MEASUREMENTS                 1148 1451   26
           BY J. C. HOPKINS(LASL) AND G. BREIT(STATE              1148 1451   27
           UNIVERSITY OF NEW YORK).                               1148 1451   28
            1.0E-05 EV TO 20 MEV.                                 1148 1451   29
                                                                  1148 1451   30
   MT=102, RADIATIVE CAPTURE --- THESE CROSS SECTIONS ARE TAKEN   1148 1451   31
           FROM THE 1966 PUBLICATION OF A. HORSLEY WHERE A VALUE  1148 1451   32
           OF 332 MB WAS ADOPTED FOR THE THERMAL VALUE.           1148 1451   33
            1.0E-05 EV TO 20 MEV.                                 1148 1451   34
                                                                  1148 1451   35
   MT=251, AVERAGE VALUE OF COSINE OF SCATTERING ANGLE            1148 1451   36
           IN LAB SYSTEM.                                         1148 1451   37
            1.0E-05 EV TO 20 MEV.                                 1148 1451   38
                                                                  1148 1451   39
   MT=252, AVERAGE LOGARITHMIC ENERGY CHANGE PER COLLISON, TAKEN  1148 1451   40
           AS 1, FROM 1.0E-05 EV TO 20 MEV.                       1148 1451   41
                                                                  1148 1451   42
   MT=253, GAMMA, TAKEN AS 1, FROM 1.0E-05 EV TO 20 MEV.          1148 1451   43
                                                                  1148 1451   44
 MF=4                                                             1148 1451   45
                                                                  1148 1451   46
   MT=  2, NEUTRON ELASTIC SCATTERING ANGULAR DISTRIBUTIONS IN    1148 1451   47
           THE CENTER OF MASS SYSTEM--GIVEN AS NORMALIZED         1148 1451   48
           POINTWISE PROBABILITIES.                               1148 1451   49
```

```
                                                                          1148 1451
                                                                          1148 1451
MF=7                                                                      1148 1451
                                                                          1148 1451
  MT=   4. .00001 TO 5 EV FREE GAS SIGMA=20.449 BARNS.                    1148 1451
                                                                          1148 1451
MF=12                                                                     1148 1451
                                                                          1148 1451
  MT=102. GAMMA RAY MULTIPLICITIES --- MULTIPLICITY. (REFERRED            1148 1451
          TO MT=102. MF=3). IS UNITY AT ALL NEUTRON ENERGIES.             1148 1451
          SIXTEEN ENERGY BANDS ARE GIVEN FROM .2 MEV TO 20 MEV.           1148 1451
          AND THE AVERAGE GAMMA RAY ENERGY. EAG. IS DETERMINED            1148 1451
          FROM THE AVERAGE NEUTRON ENERGY. EAN. IN THE BAND BY            1148 1451
          EAG=2.225E+06+EAN/2.. RECOIL ENERGY IGNORED.                    1148 1451
                                                                          1148 1451
MF=14                                                                     1148 1451
                                                                          1148 1451
  MT=102. GAMMA RAY ANGULAR DISTRIBUTION --- ASSUMED ISOTROPIC            1148 1451
          AT ALL NEUTRON ENERGIES.                                        1148 1451
                                                                          1148 1451
                                 1         451          81                1148 1451
                                 2         151           4                1148 1451
                                 3           1          48                1148 1451
                                 3           2          48                1148 1451
                                 3         102          48                1148 1451
                                 3         251           8                1148 1451
                                 3         252           4                1148 1451
                                 3         253           4                1148 1451
                                 4           2          88                1148 1451
                                 7           4           4                1148 1451
                                12         102          70                1148 1451
                                14         102           1                1148 1451
                                                                          1148 1  0
                                                                          1148 0  0
1.001  E+03 9.9917E-01           0           0           1               01148 2151
1.001  E+03 1.0000E+00           0           0           1               01148 2151
1.0    E-05 1.0    E+05          0           0           0               01148 2151
5.0    E-01 1.2756E+00           0           0           0               01148 2151
0.0         0.0                  0           0           0               01148 2  0
0.0         0.0                  0           0           0               01148 0  0
1.001  E+03 9.9917E-01           0           0           0               01148 3  1
0.0         0.0                  0           0           1            1341148 3  1
       134          5                                                     1148 3  1
1.0000E-05 3.7148E+01 2.0000E-05 3.2257E+01 5.0000E-05 2.7917E+011148 3  1
1.0000E-04 2.5730E+01 2.0000E-04 2.4183E+01 5.0000E-04 2.2811E+011148 3  1
1.0000E-03 2.2119E+01 2.0000E-03 2.1630E+01 5.0000E-03 2.1196E+011148 3  1
1.0000E-02 2.0977E+01 2.5300E-02 2.0780E+01 1.0000E+02 2.0450E+011148 3  1
1.0000E+03 2.0330E+01 2.0000E+03 2.0200E+01 3.0000E+03 2.0070E+011148 3  1
4.0000E+03 1.9940E+01 5.0000E+03 1.9820E+01 6.0000E+03 1.9690E+011148 3  1
8.0000E+03 1.9450E+01 1.0000E+04 1.9210E+01 1.5000E+04 1.8650E+011148 3  1
2.0000E+04 1.8130E+01 2.5000E+04 1.7630E+01 3.0000E+04 1.7170E+011148 3  1
3.5000E+04 1.6740E+01 4.0000E+04 1.6330E+01 4.5000E+04 1.5940E+011148 3  1
5.0000E+04 1.5580E+01 5.5000E+04 1.5230E+01 6.0000E+04 1.4900E+011148 3  1
6.5000E+04 1.4590E+01 7.0000E+04 1.4290E+01 7.5000E+04 1.4010E+011148 3  1
8.0000E+04 1.3740E+01 8.5000E+04 1.3480E+01 9.0000E+04 1.3240E+011148 3  1
9.5000E+04 1.3000E+01 1.0000E+05 1.2770E+01 1.1000E+05 1.2350E+011148 3  1
1.2000E+05 1.1960E+01 1.3000E+05 1.1610E+01 1.4000E+05 1.1280E+011148 3  1
1.5000E+05 1.0970E+01 1.6000E+05 1.0670E+01 1.7000E+05 1.0400E+011148 3  1
1.8000E+05 1.0140E+01 1.9000E+05 9.8980E+00 2.0000E+05 9.6710E+001148 3  1
2.2000E+05 9.2580E+00 2.4000E+05 8.8920E+00 2.6000E+05 8.5620E+001148 3  1
2.8000E+05 8.2620E+00 3.0000E+05 7.9870E+00 3.2000E+05 7.7340E+001148 3  1
3.4000E+05 7.5010E+00 3.6000E+05 7.2840E+00 3.8000E+05 7.0830E+001148 3  1
4.0000E+05 6.8970E+00 4.2000E+05 6.7250E+00 4.4000E+05 6.5650E+001148 3  1
4.6000E+05 6.4150E+00 4.8000E+05 6.2750E+00 5.0000E+05 6.1430E+001148 3  1
5.5000E+05 5.8450E+00 6.0000E+05 5.5840E+00 6.5000E+05 5.3540E+001148 3  1
7.0000E+05 5.1480E+00 7.5000E+05 4.9640E+00 8.0000E+05 4.7970E+001148 3  1
8.5000E+05 4.6450E+00 9.0000E+05 4.5060E+00 9.5000E+05 4.3780E+001148 3  1
1.0000E+06 4.2610E+00 1.1000E+06 4.0510E+00 1.2000E+06 3.8680E+001148 3  1
1.3000E+06 3.7060E+00 1.4000E+06 3.5610E+00 1.5000E+06 3.4290E+001148 3  1
1.6000E+06 3.3090E+00 1.7000E+06 3.1980E+00 1.8000E+06 3.0970E+001148 3  1
```

```
1.9000E+06 3.0030E+00 2.0000E+06 2.9150E+00 2.2000E+06 2.7590E+001148 3  1   120
2.4000E+06 2.6220E+00 2.6000E+06 2.5010E+00 2.8000E+06 2.3920E+001148 3  1   121
3.0000E+06 2.2930E+00 3.2000E+06 2.2030E+00 3.4000E+06 2.1200E+001148 3  1   122
3.6000E+06 2.0430E+00 3.8000E+06 1.9730E+00 4.0000E+06 1.9070E+001148 3  1   123
4.2000E+06 1.8450E+00 4.4000E+06 1.7880E+00 4.6000E+06 1.7340E+001148 3  1   124
4.8000E+06 1.6830E+00 5.0000E+06 1.6350E+00 5.2000E+06 1.5890E+001148 3  1   125
5.4000E+06 1.5470E+00 5.6000E+06 1.5060E+00 5.8000E+06 1.4670E+001148 3  1   126
6.0000E+06 1.4300E+00 6.2000E+06 1.3950E+00 6.4000E+06 1.3620E+001148 3  1   127
6.6000E+06 1.3290E+00 6.8000E+06 1.2990E+00 7.0000E+06 1.2690E+001148 3  1   128
7.5000E+06 1.2010E+00 8.0000E+06 1.1390E+00 8.5000E+06 1.0830E+001148 3  1   129
9.0000E+06 1.0320E+00 9.5000E+06 9.8590E-01 1.0000E+07 9.4320E-011148 3  1   130
1.0500E+07 9.0350E-01 1.1000E+07 8.6650E-01 1.1500E+07 8.3230E-011148 3  1   131
1.2000E+07 8.0050E-01 1.2500E+07 7.7100E-01 1.3000E+07 7.4330E-011148 3  1   132
1.3500E+07 7.1730E-01 1.4000E+07 6.9290E-01 1.4500E+07 6.6980E-011148 3  1   133
1.5000E+07 6.4800E-01 1.5500E+07 6.2740E-01 1.6000E+07 6.0780E-011148 3  1   134
1.6500E+07 5.8930E-01 1.7000E+07 5.7170E-01 1.7500E+07 5.5490E-011148 3  1   135
1.8000E+07 5.3900E-01 1.8500E+07 5.2380E-01 1.9000E+07 5.0930E-011148 3  1   136
1.9500E+07 4.9550E-01 2.0000E+07 4.8230E-01-0.         0.         1148 3  1   137
                                                                  1148 3  0   138
1.001 E+03 9.9917E-01          0          0          0          01148 3  2   139
0.         0.                  0          0          1        1341148 3  2   140
       134          5                                         1148 3  2   141
1.0000E-05 2.0449E+01 2.0000E-05 2.0449E+01 5.0000E-05 2.0449E+011148 3  2   142
1.0000E-04 2.0449E+01 2.0000E-04 2.0449E+01 5.0000E-04 2.0449E+011148 3  2   143
1.0000E-03 2.0449E+01 2.0000E-03 2.0449E+01 5.0000E-03 2.0449E+011148 3  2   144
1.0000E-02 2.0449E+01 2.5300E-02 2.0449E+01 1.0000E+02 2.0449E+011148 3  2   145
1.0000E+03 2.0329E+01 2.0000E+03 2.0198E+01 3.0000E+03 2.0068E+011148 3  2   146
4.0000E+03 1.9941E+01 5.0000E+03 1.9815E+01 6.0000E+03 1.9691E+011148 3  2   147
8.0000E+03 1.9448E+01 1.0000E+04 1.9213E+01 1.5000E+04 1.8651E+011148 3  2   148
2.0000E+04 1.8126E+01 2.5000E+04 1.7634E+01 3.0000E+04 1.7172E+011148 3  2   149
3.5000E+04 1.6737E+01 4.0000E+04 1.6327E+01 4.5000E+04 1.5941E+011148 3  2   150
5.0000E+04 1.5575E+01 5.5000E+04 1.5228E+01 6.0000E+04 1.4900E+011148 3  2   151
6.5000E+04 1.4587E+01 7.0000E+04 1.4291E+01 7.5000E+04 1.4008E+011148 3  2   152
8.0000E+04 1.3738E+01 8.5000E+04 1.3481E+01 9.0000E+04 1.3235E+011148 3  2   153
9.5000E+04 1.2999E+01 1.0000E+05 1.2774E+01 1.1000E+05 1.2351E+011148 3  2   154
1.2000E+05 1.1964E+01 1.3000E+05 1.1607E+01 1.4000E+05 1.1275E+011148 3  2   155
1.5000E+05 1.0965E+01 1.6000E+05 1.0673E+01 1.7000E+05 1.0398E+011148 3  2   156
1.8000E+05 1.0140E+01 1.9000E+05 9.8980E+00 2.0000E+05 9.6710E+001148 3  2   157
2.2000E+05 9.2580E+00 2.4000E+05 8.8920E+00 2.6000E+05 8.5620E+001148 3  2   158
2.8000E+05 8.2620E+00 3.0000E+05 7.9870E+00 3.2000E+05 7.7340E+001148 3  2   159
3.4000E+05 7.5010E+00 3.6000E+05 7.2840E+00 3.8000E+05 7.0830E+001148 3  2   160
4.0000E+05 6.8970E+00 4.2000E+05 6.7250E+00 4.4000E+05 6.5650E+001148 3  2   161
4.6000E+05 6.4150E+00 4.8000E+05 6.2750E+00 5.0000E+05 6.1430E+001148 3  2   162
5.5000E+05 5.8450E+00 6.0000E+05 5.5840E+00 6.5000E+05 5.3540E+001148 3  2   163
7.0000E+05 5.1480E+00 7.5000E+05 4.9640E+00 8.0000E+05 4.7970E+001148 3  2   164
8.5000E+05 4.6450E+00 9.0000E+05 4.5060E+00 9.5000E+05 4.3780E+001148 3  2   165
1.0000E+06 4.2610E+00 1.1000E+06 4.0510E+00 1.2000E+06 3.8680E+001148 3  2   166
1.3000E+06 3.7060E+00 1.4000E+06 3.5610E+00 1.5000E+06 3.4290E+001148 3  2   167
1.6000E+06 3.3090E+00 1.7000E+06 3.1980E+00 1.8000E+06 3.0970E+001148 3  2   168
1.9000E+06 3.0030E+00 2.0000E+06 2.9150E+00 2.2000E+06 2.7590E+001148 3  2   169
2.4000E+06 2.6220E+00 2.6000E+06 2.5010E+00 2.8000E+06 2.3920E+001148 3  2   170
3.0000E+06 2.2930E+00 3.2000E+06 2.2030E+00 3.4000E+06 2.1200E+001148 3  2   171
3.6000E+06 2.0430E+00 3.8000E+06 1.9730E+00 4.0000E+06 1.9070E+001148 3  2   172
4.2000E+06 1.8450E+00 4.4000E+06 1.7880E+00 4.6000E+06 1.7340E+001148 3  2   173
4.8000E+06 1.6830E+00 5.0000E+06 1.6350E+00 5.2000E+06 1.5890E+001148 3  2   174
5.4000E+06 1.5470E+00 5.6000E+06 1.5060E+00 5.8000E+06 1.4670E+001148 3  2   175
6.0000E+06 1.4300E+00 6.2000E+06 1.3950E+00 6.4000E+06 1.3620E+001148 3  2   176
6.6000E+06 1.3290E+00 6.8000E+06 1.2990E+00 7.0000E+06 1.2690E+001148 3  2   177
7.5000E+06 1.2010E+00 8.0000E+06 1.1390E+00 8.5000E+06 1.0830E+001148 3  2   178
9.0000E+06 1.0320E+00 9.5000E+06 9.8590E-01 1.0000E+07 9.4320E-011148 3  2   179
1.0500E+07 9.0350E-01 1.1000E+07 8.6650E-01 1.1500E+07 8.3230E-011148 3  2   180
1.2000E+07 8.0050E-01 1.2500E+07 7.7100E-01 1.3000E+07 7.4330E-011148 3  2   181
1.3500E+07 7.1730E-01 1.4000E+07 6.9290E-01 1.4500E+07 6.6980E-011148 3  2   182
1.5000E+07 6.4800E-01 1.5500E+07 6.2740E-01 1.6000E+07 6.0780E-011148 3  2   183
1.6500E+07 5.8930E-01 1.7000E+07 5.7170E-01 1.7500E+07 5.5490E-011148 3  2   184
1.8000E+07 5.3900E-01 1.8500E+07 5.2380E-01 1.9000E+07 5.0930E-011148 3  2   185
1.9500E+07 4.9550E-01 2.0000E+07 4.8230E-01-0.        -0.         1148 3  2   186
                                                                  1148 3  0   187
1.001 E+03 9.9917E-01          0          0          0          01148 3102   188
0.0          2.2247 E+06       0          0          1        1341148 3102   189
       134          5                                         1148 3102   190
```

```
1.0000E-05 1.6699E+01 2.0000E-05 1.1808E+01 5.0000E-05 7.4682E+001148 3102  191
1.0000E-04 5.2808E+00 2.0000E-04 3.7341E+00 5.0000E-04 2.3616E+001148 3102  192
1.0000E-03 1.6699E+00 2.0000E-03 1.1808E+00 5.0000E-03 7.4682E-011148 3102  193
1.0000E-02 5.2808E-01 2.5300E-02 3.3200E-01 1.0000E+02 5.2770E-031148 3102  194
1.0000E+03 1.6590E-03 2.0000E+03 1.1926E-03 3.0000E+03 9.7008E-041148 3102  195
4.0000E+03 8.3290E-04 5.0000E+03 7.3747E-04 6.0000E+03 6.6619E-041148 3102  196
8.0000E+03 5.6518E-04 1.0000E+04 4.9580E-04 1.5000E+04 3.8782E-041148 3102  197
2.0000E+04 3.2386E-04 2.5000E+04 2.8064E-04 3.0000E+04 2.4909E-041148 3102  198
3.5000E+04 2.2248E-04 4.0000E+04 2.0553E-04 4.5000E+04 1.8970E-041148 3102  199
5.0000E+04 1.7646E-04 5.5000E+04 1.6518E-04 6.0000E+04 1.5544E-041148 3102  200
6.5000E+04 1.4693E-04 7.0000E+04 1.3942E-04 7.5000E+04 1.3273E-041148 3102  201
8.0000E+04 1.2673E-04 8.5000E+04 1.2132E-04 9.0000E+04 1.1640E-041148 3102  202
9.5000E+04 1.1191E-04 1.0000E+05 1.0780E-04 1.1000E+05 1.0019E-041148 3102  203
1.2000E+05 9.3717E-05 1.3000E+05 8.8131E-05 1.4000E+05 8.3256E-051148 3102  204
1.5000E+05 7.8960E-05 1.6000E+05 7.5142E-05 1.7000E+05 7.1725E-051148 3102  205
1.8000E+05 6.8645E-05 1.9000E+05 6.5853E-05 2.0000E+05 6.3310E-051148 3102  206
2.2000E+05 5.9051E-05 2.4000E+05 5.5591E-05 2.6000E+05 5.2731E-051148 3102  207
2.8000E+05 5.0330E-05 3.0000E+05 4.8290E-05 3.2000E+05 4.6538E-051148 3102  208
3.4000E+05 4.5018E-05 3.6000E+05 4.3691E-05 3.8000E+05 4.2523E-051148 3102  209
4.0000E+05 4.1490E-05 4.2000E+05 4.0607E-05 4.4000E+05 3.9828E-051148 3102  210
4.6000E+05 3.9138E-05 4.8000E+05 3.8525E-05 5.0000E+05 3.7980E-051148 3102  211
5.5000E+05 3.7396E-05 6.0000E+05 3.6870E-05 6.5000E+05 3.6163E-051148 3102  212
7.0000E+05 3.5520E-05 7.5000E+05 3.5167E-05 8.0000E+05 3.4840E-051148 3102  213
8.5000E+05 3.4742E-05 9.0000E+05 3.4650E-05 9.5000E+05 3.4552E-051148 3102  214
1.0000E+06 3.4460E-05 1.1000E+06 3.4440E-05 1.2000E+06 3.4410E-051148 3102  215
1.3000E+06 3.4490E-05 1.4000E+06 3.4360E-05 1.5000E+06 3.4340E-051148 3102  216
1.6000E+06 3.4310E-05 1.7000E+06 3.4290E-05 1.8000E+06 3.4270E-051148 3102  217
1.9000E+06 3.4250E-05 2.0000E+06 3.4230E-05 2.2000E+06 3.4520E-051148 3102  218
2.4000E+06 3.4810E-05 2.6000E+06 3.5100E-05 2.8000E+06 3.5390E-051148 3102  219
3.0000E+06 3.5680E-05 3.2000E+06 3.5800E-05 3.4000E+06 3.5910E-051148 3102  220
3.6000E+06 3.6030E-05 3.8000E+06 3.6140E-05 4.0000E+06 3.6260E-051148 3102  221
4.2000E+06 3.6290E-05 4.4000E+06 3.6320E-05 4.6000E+06 3.6360E-051148 3102  222
4.8000E+06 3.6390E-05 5.0000E+06 3.6420E-05 5.2000E+06 3.6290E-051148 3102  223
5.4000E+06 3.6160E-05 5.6000E+06 3.6040E-05 5.8000E+06 3.5910E-051148 3102  224
6.0000E+06 3.5780E-05 6.2000E+06 3.5670E-05 6.4000E+06 3.5560E-051148 3102  225
6.6000E+06 3.5450E-05 6.8000E+06 3.5340E-05 7.0000E+06 3.5230E-051148 3102  226
7.5000E+06 3.4590E-05 8.0000E+06 3.3940E-05 8.5000E+06 3.3640E-051148 3102  227
9.0000E+06 3.3330E-05 9.5000E+06 3.2960E-05 1.0000E+07 3.2590E-051148 3102  228
1.0500E+07 3.2210E-05 1.1000E+07 3.1820E-05 1.1500E+07 3.1450E-051148 3102  229
1.2000E+07 3.1080E-05 1.2500E+07 3.0630E-05 1.3000E+07 3.0180E-051148 3102  230
1.3500E+07 3.0010E-05 1.4000E+07 2.9830E-05 1.4500E+07 2.9400E-051148 3102  231
1.5000E+07 2.8960E-05 1.5500E+07 2.8630E-05 1.6000E+07 2.8300E-051148 3102  232
1.6500E+07 2.7880E-05 1.7000E+07 2.7450E-05 1.7500E+07 2.7360E-051148 3102  233
1.8000E+07 2.7260E-05 1.8500E+07 2.6730E-05 1.9000E+07 2.6200E-051148 3102  234
1.9500E+07 2.6120E-05 2.0000E+07 2.6040E-05-0.        -0.        1148 3102  235
                                                                  1148 3  0  236
1.001 E+03 9.9917E-01          0          0          0          01148 3251  237
0.0        0.0                 0          0          1         141148 3251  238
        14          3                                             1148 3251  239
1.0    E-056.65213E-01 1.0    E+056.65213E-01 5.0    E+056.64899E-011148 3251  240
1.0    E+066.64620E-01 2.0    E+066.64149E-01 4.0    E+066.63355E-011148 3251  241
6.0    E+066.62628E-01 8.0    E+066.62018E-01 1.0    E+076.61338E-011148 3251  242
1.2    E+076.61045E-01 1.4    E+076.60449E-01 1.6    E+076.59929E-011148 3251  243
1.8    E+076.59540E-01 2.0    E+076.59196E-01                     1148 3251  244
                                                                  1148 3  0  245
1.001 E+03 9.9917E-01          0          0          0          01148 3252  246
0.0        0.0                 0          0          1          21148 3252  247
         2          2                                             1148 3252  248
1.0    E-05 1.0        2.0    E+07 1.0                            1148 3252  249
                                                                  1148 3  0  250
1.001 E+03 9.9917E-01          0          0          0          01148 3253  251
0.0        0.0                 0          0          1          21148 3253  252
         2          2                                             1148 3253  253
1.0    E-05 1.0        2.0    E+07 1.0                            1148 3253  254
                                                                  1148 3  0  255
                                                                  1148 0  0  256
1.001 E+03 9.9917E-01          0          2          0          01148 4  2  257
0.0        0.0                 0          2          0          01148 4  2  258
       0.0        0.0          0          0          1         141148 4  2  259
        14          2                                             1148 4  2  260
```

```
        0.0       1.0-05          0          0          1         111148 4   2  261
         11            2                                            1148 4   2  262
       -1.0           .5         -.8        .5        -.6          .51148 4   2  263
        -.4           .5         -.2        .5         0.          .51148 4   2  264
         .2           .5          .4        .5         .6          .51148 4   2  265
         .8           .5         1.0        .5                      1148 4   2  266
        0.0       1.0+05          0          0          1         111148 4   2  267
         11            2                                            1148 4   2  268
       -1.0           .5         -.8        .5        -.6          .51148 4   2  269
        -.4           .5         -.2        .5         0.          .51148 4   2  270
         .2           .5          .4        .5         .6          .51148 4   2  271
         .8           .5         1.0        .5                      1148 4   2  272
        0.0       5.0+05          0          0          1         111148 4   2  273
         11            2                                            1148 4   2  274
 -.10000E+01 .50117E+00-.80000E+00 .50097E+00-.60000E+00 .50066E+001148 4   2  275
 -.40000E+00 .50045E+00-.20000E+00 .50025E+000.          .50005E+001148 4   2  276
  .20000E+00 .49974E+00 .40000E+00 .49953E+00 .60000E+00 .49933E+001148 4   2  277
  .80000E+00 .49902E+00 .10000E+01 .49882E+00                       1148 4   2  278
        0.0       1.0+06          0          0          1         111148 4   2  279
         11            2                                            1148 4   2  280
 -.10000E+01 .50218E+00-.80000E+00 .50176E+00-.60000E+00 .50131E+001148 4   2  281
 -.40000E+00 .50089E+00-.20000E+00 .50044E+000.          .50000E+001148 4   2  282
  .20000E+00 .49957E+00 .40000E+00 .49913E+00 .60000E+00 .49869E+001148 4   2  283
  .80000E+00 .49823E+00 .10000E+01 .49779E+00                       1148 4   2  284
        0.0       2.0+06          0          0          1         111148 4   2  285
         11            2                                            1148 4   2  286
 -.10000E+01 .50387E+00-.80000E+00 .50311E+00-.60000E+00 .50234E+001148 4   2  287
 -.40000E+00 .50158E+00-.20000E+00 .50081E+000.          .50003E+001148 4   2  288
  .20000E+00 .49925E+00 .40000E+00 .49846E+00 .60000E+00 .49764E+001148 4   2  289
  .80000E+00 .49682E+00 .10000E+01 .49598E+00                       1148 4   2  290
        0.0       4.0+06          0          0          1         111148 4   2  291
         11            2                                            1148 4   2  292
 -.10000E+01 .50677E+00-.80000E+00 .50542E+00-.60000E+00 .50407E+001148 4   2  293
 -.40000E+00 .50275E+00-.20000E+00 .50143E+000.          .50011E+001148 4   2  294
  .20000E+00 .49873E+00 .40000E+00 .49734E+00 .60000E+00 .49589E+001148 4   2  295
  .80000E+00 .49441E+00 .10000E+01 .49283E+00                       1148 4   2  296
        0.0       6.0+06          0          0          1         111148 4   2  297
         11            2                                            1148 4   2  298
 -.10000E+01 .50999E+00-.80000E+00 .50770E+00-.60000E+00 .50560E+001148 4   2  299
 -.40000E+00 .50362E+00-.20000E+00 .50177E+000.          .49993E+001148 4   2  300
  .20000E+00 .49808E+00 .40000E+00 .49628E+00 .60000E+00 .49439E+001148 4   2  301
  .80000E+00 .49246E+00 .10000E+01 .49040E+00                       1148 4   2  302
        0.0       8.0+06          0          0          1         111148 4   2  303
         11            2                                            1148 4   2  304
 -.10000E+01 .51288E+00-.80000E+00 .50952E+00-.60000E+00 .50676E+001148 4   2  305
 -.40000E+00 .50434E+00-.20000E+00 .50207E+000.          .49987E+001148 4   2  306
  .20000E+00 .49766E+00 .40000E+00 .49546E+00 .60000E+00 .49314E+001148 4   2  307
  .80000E+00 .49077E+00 .10000E+01 .48818E+00                       1148 4   2  308
        0.0      10.0+06          0          0          1         111148 4   2  309
         11            2                                            1148 4   2  310
 -.10000E+01 .51727E+00-.80000E+00 .51201E+00-.60000E+00 .50794E+001148 4   2  311
 -.40000E+00 .50461E+00-.20000E+00 .50168E+000.          .49908E+001148 4   2  312
  .20000E+00 .49669E+00 .40000E+00 .49435E+00 .60000E+00 .49202E+001148 4   2  313
  .80000E+00 .48969E+00 .10000E+01 .48716E+00                       1148 4   2  314
        0.0      12.0+06          0          0          1         121148 4   2  315
         12            2                                            1148 4   2  316
 -.10000E+01 .52272E+00-.90000E+00 .51825E+00-.80000E+00 .51464E+001148 4   2  317
 -.60000E+00 .50898E+00-.40000E+00 .50475E+00-.20000E+00 .50129E+001148 4   2  318
 0.          .49823E+00 .20000E+00 .49556E+00 .40000E+00 .49313E+001148 4   2  319
  .60000E+00 .49085E+00 .80000E+00 .48866E+00 .10000E+01 .48654E+001148 4   2  320
        0.0      14.0+06          0          0          1         121148 4   2  321
         12            2                                            1148 4   2  322
 -.10000E+01 .52823E+00-.90000E+00 .52188E+00-.80000E+00 .51698E+001148 4   2  323
 -.60000E+00 .50982E+00-.40000E+00 .50456E+00-.20000E+00 .50048E+001148 4   2  324
 0.          .49703E+00 .20000E+00 .49431E+00 .40000E+00 .49195E+001148 4   2  325
  .60000E+00 .49005E+00 .80000E+00 .48833E+00 .10000E+01 .48688E+001148 4   2  326
        0.0      16.0+06          0          0          1         121148 4   2  327
         12            2                                            1148 4   2  328
 -.10000E+01 .53433E+00-.90000E+00 .52575E+00-.80000E+00 .51924E+001148 4   2  329
 -.60000E+00 .51024E+00-.40000E+00 .50404E+00-.20000E+00 .49939E+001148 4   2  330
```

```
0.         .49567E+00 .20000E+00 .49288E+00 .40000E+00 .49081E+001148 4  2   331
 .60000E+00 .48936E+00 .80000E+00 .48853E+00 .10000E+01 .48833E+001148 4  2   332
       0.0    18.0+06          0          0          1        121148 4  2   333
        12          2                                           1148 4  2   334
-.10000E+01 .54092E+00-.90000E+00 .52962E+00-.80000E+00 .52134E+001148 4  2   335
-.60000E+00 .51038E+00-.40000E+00 .50327E+00-.20000E+00 .49802E+001148 4  2   336
0.          .49406E+00 .20000E+00 .49126E+00 .40000E+00 .48963E+001148 4  2   337
 .60000E+00 .48905E+00 .80000E+00 .48940E+00 .10000E+01 .49091E+001148 4  2   338
       0.0    20.0+06          0          0          1        121148 4  2   339
        12          2                                           1148 4  2   340
-.10000E+01 .54807E+00-.90000E+00 .53348E+00-.80000E+00 .52332E+001148 4  2   341
-.60000E+00 .51029E+00-.40000E+00 .50221E+00-.20000E+00 .49635E+001148 4  2   342
0.          .49218E+00 .20000E+00 .48958E+00 .40000E+00 .48840E+001148 4  2   343
 .60000E+00 .48866E+00 .80000E+00 .49075E+00 .10000E+01 .49466E+001148 4  2   344
                                                                1148 4  0   345
                                                                1148 0  0   346
 1.001 E+03 9.9917E-01          0          0          0        01148 7  4   347
 0.0        0.0                 0          0         12        11148 7  4   348
 0.0        2.0    E+02 9.9917E-01 5.0        0.0        0.0    1148 7  4   349
 1.0        2.0449E+01 9.9917E-01 0.0         0.0        0.0     1148 7  4   350
                                                                1148 7  0   351
                                                                1148 0  0   352
 1.0010E+03 9.9917E-01          1          0         17       0114812102   353
 0.         0.                  0          0          1       2114812102   354
          2          2                                         114812102   355
 1.0000E-05 1.0000E+00 2.0000E+07 1.0000E+00                   114812102   356
 1.1725E+07 0.                  0          2          1       3114812102   357
          3          2                                         114812102   358
 1.8000E+07 0.         1.8001E+07 1.0000E+00 2.0000E+07 1.0000E+00114812102   359
 1.0725E+07 0.                  0          2          1       4114812102   360
          4          2                                         114812102   361
 1.6000E+07 0.         1.6001E+07 1.0000E+00 1.8000E+07 1.0000E+00114812102   362
 1.8001E+07 0.                                                 114812102   363
 9.7250E+06 0.                  0          2          1       4114812102   364
          4          2                                         114812102   365
 1.4000E+07 0.         1.4001E+07 1.0000E+00 1.6000E+07 1.0000E+00114812102   366
 1.6001E+07 0.                                                 114812102   367
 8.7250E+06 0.                  0          2          1       4114812102   368
          4          2                                         114812102   369
 1.2000E+07 0.         1.2001E+07 1.0000E+00 1.4000E+07 1.0000E+00114812102   370
 1.4001E+07 0.                                                 114812102   371
 7.7250E+06 0.                  0          2          1       4114812102   372
          4          2                                         114812102   373
 1.0000E+07 0.         1.0001E+07 1.0000E+00 1.2000E+07 1.0000E+00114812102   374
 1.2001E+07 0.                                                 114812102   375
 6.9750E+06 0.                  0          2          1       4114812102   376
          4          2                                         114812102   377
 9.0000E+06 0.         9.0001E+06 1.0000E+00 1.0000E+07 1.0000E+00114812102   378
 1.0001E+07 0.                                                 114812102   379
 6.4750E+06 0.                  0          2          1       4114812102   380
          4          2                                         114812102   381
 8.0000E+06 0.         8.0001E+06 1.0000E+00 9.0000E+06 1.0000E+00114812102   382
 9.0001E+06 0.                                                 114812102   383
 5.9750E+06 0.                  0          2          1       4114812102   384
          4          2                                         114812102   385
 7.0000E+06 0.         7.0001E+06 1.0000E+00 8.0000E+06 1.0000E+00114812102   386
 8.0001E+06 0.                                                 114812102   387
 5.4750E+06 0.                  0          2          1       4114812102   388
          4          2                                         114812102   389
 6.0000E+06 0.         6.0001E+06 1.0000E+00 7.0000E+06 1.0000E+00114812102   390
 7.0001E+06 0.                                                 114812102   391
 4.9750E+06 0.                  0          2          1       4114812102   392
          4          2                                         114812102   393
 5.0000E+06 0.         5.0001E+06 1.0000E+00 6.0000E+06 1.0000E+00114812102   394
 6.0001E+06 0.                                                 114812102   395
 4.4750E+06 0.                  0          2          1       4114812102   396
          4          2                                         114812102   397
 4.0000E+06 0.         4.0001E+06 1.0000E+00 5.0000E+06 1.0000E+00114812102   398
 5.0001E+06 0.                                                 114812102   399
 3.9750E+06 0.                  0          2          1       4114812102   400
```

```
         4           2                                                 114812102   401
3.0000E+06 0.          3.0001E+06 1.0000E+00 4.0000E+06 1.0000E+00114812102   402
4.0001E+06 0.                                                          114812102   403
3.4750E+06 0.                   0          2          1           4114812102   404
         4           2                                                 114812102   405
2.0000E+06 0.          2.0001E+06 1.0000E+00 3.0000E+06 1.0000E+00114812102   406
3.0001E+06 0.                                                          114812102   407
2.9750E+06 0.                   0          2          1           4114812102   408
         4           2                                                 114812102   409
1.0000E+06 0.          1.0001E+06 1.0000E+00 2.0000E+06 1.0000E+00114812102   410
2.0001E+06 0.                                                          114812102   411
2.6250E+06 0.                   0          2          1           4114812102   412
         4           2                                                 114812102   413
6.0000E+05 0.          6.0001E+05 1.0000E+00 1.0000E+06 1.0000E+00114812102   414
1.0001E+06 0.                                                          114812102   415
2.4250E+06 0.                   0          2          1           4114812102   416
         4           2                                                 114812102   417
2.0000E+05 0.          2.0001E+05 1.0000E+00 6.0000E+05 1.0000E+00114812102   418
6.0001E+05 0.                                                          114812102   419
2.2250E+06 0.                   0          2          1           3114812102   420
         3           2                                                 114812102   421
1.0000E-05 1.0000E+00 2.0000E+05 1.0000E+00 2.0001E+05 0.              114812102   422
                                                                       114812  0   423
                                                                       1148 0  0   424
1.0010E+03 9.9917E-01           1          0          0           0114814102   425
                                                                       114814  0   426
                                                                       1148 0  0   427
                                                                        -1         428
```

ALT/fs:1029(750)

Appendix B

Summary for ^{3}He

SUMMARY DOCUMENTATION FOR He-3*

L. Stewart and R. J. LaBauve

Los Alamos Scientific Laboratory

ABSTRACT

The following nuclear data are given for ^{3}He in the energy range from 1.0×10^{-5} eV to 20 MeV:

File 1. The general description of the data which follow

File 2. Values of the nuclear spin and effective scattering radius

File 3. Smooth point-wise data for the total, the free-atom elastic, (n,p), (n,d), and the radiative capture cross sections; data for $\bar{\mu}$, ξ, and γ are also included

File 4. The angular distributions for elastic scattering as probability vs cosine of the scattering angle in the center-of-mass system

File 7: The free-atom-scattering cross section at thermal

I. INTRODUCTION

These data were translated from an unpublished evaluation completed by L. Stewart in 1968. In 1971, the Standards Subcommittee of CSEWG reviewed the file and concluded that the (n,p) cross section was still adequately represented to be recommended as a standard cross section.

II. TOTAL CROSS SECTION

The total cross section was obtained by summing the partials up to 100 keV. From 100 keV to 20 MeV, the LASL measurements[1] were used exclusively in this evaluation.

III. ELASTIC SCATTERING

Available measurements of Seagrave, Cranberg, and Simmons[2], of Sayres, Jones, and Wu[3], and of Antolković et al.[4] were used. Also the p + T scattering was used to fill the gaps in energy where no n + ^{3}He elastic scattering measurements exist. The p + T experiments employed were those of Brolley et al.[5], Rosen and Leland[6], and of Vanetsian and Fedchenko[7]. Wick's Limit was employed at all energies to insure the nonviolation of unitarity. The angular distributions are given as probabilities versus cosine of the center-of-mass scattering angle.

---

* Work done under the auspices of the United States Atomic Energy Commission

IV. RADIATIVE CAPTURE

Gallmann, Kane, and Pixley[8] have placed upper limits on the thermal capture cross section of 100 μb and 10 μb for gamma and pair emission, respectively. Since these are upper limits and absolute measurements do not exist, no estimate is made here for radiative capture. The gamma-ray production cross sections are also assumed to be negligible and are therefore ignored.

V. (n,p) CROSS SECTION

The (n,p) cross section below 10 eV was derived solely from the measurements of Als-Nielsen and Dietrich[9] giving $5327 {}^{+10}_{-9}$ b at thermal. A 1/v extrapolation was assumed to 1.7 keV, where the slope was changed to merge with the slope of the curve given by the data of Gibbons and Macklin[10,11]. Many experiments[3,12-17] have been performed at higher energies although, all too often, the cross sections were not obtained on an absolute basis. The most extensive absolute measurements were those of Perry et al.[13] which have been heavily weighted in this evaluation.

VI. (n,d) CROSS SECTION

Only the Columbia data[3] near 7.5 MeV were available on this reaction. Bradbury and Stewart[18], however, employed detailed balance and the LASL measurements on the inverse reaction, that is, the $D(d,n)^3He$ reaction, to predict the energy dependent cross section to 15 MeV. Above 15 MeV, these data were extrapolated.

VII. THREE- AND FOUR-BODY BREAKUP

The $^3He(n,np)D$ and $^3He(n,2n2p)$ reactions have Q values of -5.494 and -7.718 MeV, respectively. Only a few measurements exist on these reactions, and these are usually limited to a search for final-state phenomena. The spectrum is usually observed at one angle very close to zero degrees. Observation of the proton spectrum at 14.4 MeV reveals no clear indication of n-d, two-nucleon, or three-nucleon final-state interaction. A strong n-p final state is evident from measurements of the deuteron spectrum at $\theta_d = 5^o$. An upper limit of 12 mb has also been set on the $^3He(n,2n2p)$ reaction. In the absence of measurements of the absolute cross sections, these break-up cross sections have been assumed small and are therefore ignored in the present evaluation.

REFERENCES

1. Los Alamos Physics and Cryogenics Groups, Nucl. Phys. 12, 29 (1959).

2. J. D. Seagrave, L. Cranberg, and J. E. Simmons, Phys. Rev. 119, 1981 (1960). (Used hydrogen scattering as standard; corrected for multiple scattering.)

3. A. R. Sayres, K. W. Jones, and C. S. Wu, Phys. Rev. 122, 1853 (1961). (Recoil proportional counter but normalized to a "total integral" obtained using an unknown extrapolation to zero degrees.)

4. B. Antolković, G. Paić, P. Thomaš, and D. Rendić, Phys. Rev. 159, 777 (1967). (Counted ^{3}He recoils using counter telescope.)

5. J. E. Brolley, Jr., T. M. Putnam, L. Rosen, and L. Stewart, Phys. Rev. 117, 1307 (1960).

6. L. Rosen and W. Leland, WASH-1079, p. 109 (1967).

7. R. A. Vanetsian and E. D. Fedchenko, Translation: Soviet Journal of Atomic Energy 2, 141 (1957).

8. A. Gallmann, J. Kane, and R. Pixley, BAPS 5, p. 19 (1960).

9. J. Als-Nielsen and O. Dietrich, Phys. Rev. 133, B925 (1964).

10. J. H. Gibbons and R. L. Macklin, Phys. Rev. 114, 571 (1959). Obtained, by reciprocity, from a measurement of the T(p,n)^{3}He reaction. These data are a corrected version of the data of R. L. Macklin and J. H. Gibbons, Phys. Rev. 109, 105 (1958); ~ 27% change due to reanalysis of target gas. Normalized to a T(p,n)^{3}He cross section of 580 572 mb ± 10% at 2.955 MeV as measured by the authors. [J. H. Gibbons, private communication (1963)].

11. R. L. Macklin and J. H. Gibbons, Proceedings of the International Conference on the Study of Nuclear Structure with Neutrons, Antwerp, 19-23 July 1965 (North-Holland Publishing Co.) p. 498. Obtained by reciprocity; normalization: based on an earlier absolute value measured by the authors of 1980 mb at 100 keV. Quote from Gibbons, ".... caution the reader that for $E_n \leq$ 10 keV, we are working into the toe of our resolution (beam plus target-thickness profile), so the data are much less trustworthy." (Private communication, 1966).

12. R. Batchelor, R. Aves, and T. H. R. Skyrme, Rev. Sci. Instr. 26, 1037 (1955). ^{3}He-filled proportional counter; resolution affected by small tritium contamination; long counter used for flux monitor.

13. J. E. Perry, Jr., E. Haddad, R. L. Henkel, G. A. Jarvis, and R. K. Smith, unpublished (1958); experiment reported by J. D. Seagrave in Proceedings of Conference on Nuclear Forces and the Few Nucleon Problem, London, 1959 (Pergamon Press). Data obtained by reciprocity from measurements of the T(p,n)^{3}He reaction; calculations and integrals by Stewart. Private communiby J. E. Perry, Jr. (1960).

14. M. D. Goldberg, J. D. Anderson, J. P. Stoering, and C. Wong, Phys. Rev. 122, 1510 (1961). Errors not given.

15. G. F. Bogdanov, N. A. Vlasov, C. P. Kalinin, B. V. Rybakov, L. N. Samoilov, and V. A. Sidorov, JETP (USSR) 36, 633 (1959).

16. W. E. Wilson, R. L. Walter, and D. B. Fossan, Nucl. Phys. 27, 421 (1961).

17. B. Antolković, G. Paić, P. Thomaš, and D. Rendić, private communication (1967).

18. J. N. Bradbury and L. Stewart, BAPS 3, 417 (1958). Obtained, by reciprocity, from selected points through the D(d,n)^{3}He data of J. E. Brolley, Jr., T. M. Putnam, and L. Rosen, Phys. Rev. 107, 820 (1957).

Appendix C

Summary for ^{6}Li

SUMMARY OF ^{6}Li DATA FOR ENDF/B-III*

by

M. E. Battat and R. J. LaBauve

Los Alamos Scientific Laboratory

INTRODUCTION

The data, in the energy range of 10^{-5} eV to 20 MeV, for ^{6}Li (MAT 1115) were submitted to the NNCSC in September 1971 for inclusion in ENDF/B-III. These data represent an extensive revision of the earlier file (MAT 1005) which was, for the most part, based on a UKAEA evaluation.[1] A major change was made in that below about 1.7 MeV the cross sections for MAT1115 reflect very strongly the detailed review of the available data by Uttley, Sowerby, Patrick and Rae.[2] In choosing the latter data, consideration was also given to the recommendation of the CSEWG Normalization and Standards Subcommittee that Uttley's (n,α) data be used in the latter energy range.[3] The data above 1.7 MeV for MAT 1115 will be discussed in later sections of this document. For this first pass reevaluation, the (n,γ) cross sections from the earlier UK evaluation were retained. Following Uttley et al., the data will be considered in the following energy intervals: (1) thermal, (2) thermal to 10 keV, (3) 10 to 500 keV, and (4) 500 keV to 1.7 MeV. Above 1.7 MeV, the data will be considered by reaction type.

THERMAL ENERGIES

The cross sections given in the file at 0.0253 eV are as follows:

$$\begin{aligned} \text{Total} &= 941.015 \text{ b} \\ \text{Elastic} &= 0.72 \\ (n,\alpha) &= 940.25 \\ (n,\gamma) &= 0.045 \end{aligned}$$

It is estimated that the (n,α) cross section is known to ± 0.5%. The choice of Uttley et al. for the (n,γ) cross section is 30 ± 8 mb.

THERMAL ENERGIES TO 10 keV

The (n,α) cross sections up to 100 eV were calculated from the formula

$$\sigma(n,\alpha) = (149.56/\sqrt{E}) - 0.024 \text{ b}.$$

Above 100 eV, the p-wave absorption contribution from the 247-keV resonance becomes increasingly important, and at 10 keV the negative s-wave absorption (-0.024b) is largely cancelled. Uttley et al. assign an uncertainty of ± 1% to the (n,α) cross section from thermal to 10 keV. Over this energy range, the data given in the file deviate from a strict 1/v dependence ($149.56/\sqrt{E}$) by a maximum of -0.4%.

---

* Work done under the auspices of the United States Atomic Energy Commission

The elastic cross section is held constant at 0.72 b up to 2 keV with a monotonic increase thereafter to 0.7221 b at 10 keV. The elastic angular distribution up to 10 keV is specified as isotropic in the CM system.

10 to 500 keV

The uncertainties estimated for the (n,α) cross section are:

± 2% at 100 keV, ± 5% from 100 to 300 keV, and
± 10 % at 500 keV.

As can be seen from Fig. 2 of Ref. 2, there is great disagreement among the (n,α) measurements in this energy range. The (n,α) curve used in this evaluation is the one recommended by Uttley et al. based on their careful review of the experimental data. In arriving at the recommended curve, the latter authors have considered the following points:

1. Errors in the energy scales of the measurements.
2. The inadequacy of those experiments in which the cross sections were measured relative to the ^{235}U fission cross section.
3. The inadequacy of those measurements in which the neutron flux was measured relative to a long counter.
4. The inconsistency of measurements made using thick lithium detectors with total and scattering cross section measurements.

Uttley et al. do, however, state that much experimental work[2] remains to be done to confirm their recommendations.

In general, the total cross sections are those reported by Uttley and Diment.[4] The scattering cross sections below and above 100 keV reflect the data of Asami and Moxon[5] and Lane, Langsdorf, Monahan, and Elwyn[6], respectively. For the elastic angular distributions, Legendre coefficients (from which the normalized probability distributions are reconstructed) were inferred by fitting the data of Lane et al.[6]

500 keV to 1.7 MeV

The (n,α) data in this energy range were obtained by subtracting the scattering cross section from the total cross section. The total cross section values were essentially those reported by Diment and Uttley.[4] For the scattering cross sections the measurements of Lane et al.[6] and Knitter and Coppola[7] were considered. The (n,α) cross section uncertainties are estimated to be:

± 10% at 500 keV, increasing to ± 15% between 700 and 1000 keV, and decreasing to ± 10% by 1.7 MeV.

TOTAL CROSS SECTION

From 2 to 15 MeV, the primary reference for the data in the file is the work of Foster and Glasgow.[8] The extrapolation to 20 MeV was based on the measurements of Peterson, Bratenahl, and Stoering.[9]

ELASTIC CROSS SECTION

Data given between 4 and 10 MeV are heavily weighted towards the Hopkins, Drake and Condé evaluation.[10] A value of 0.88 b at 14 MeV was used and data smoothly extrapolated to 20 MeV.

Legendre coefficients for the angular distributions up to 2.5 MeV were determined from the data of Lane et al.[6] Between 4.83 and 7.5 MeV coefficients were inferred from Hopkins et al. data.[10] Based on 14-MeV elastic scattering data given in BNL-400, optical model calculations (ABACUS code) were performed to infer Legendre coefficients between 10 and 20 MeV. Data for Mt 251 ($\bar{\mu}_L$), 252 (ξ), and 253 (γ) were caluclated using the elastic angular distributions given in File 4.

THE (n,2n)αp CROSS SECTION

The cross sections and angular distributions for this reaction (MT = 24) are the same as in Ref. 1 up to 15 MeV with a smooth extrapolation to 20 MeV. The secondary energy distributions given in Ref. 1 have been approximated by ENDF/B Law 9 with $\theta = 0.21\sqrt{E}$ (MeV).

THE (n,n')γ and (n,n')αd CROSS SECTIONS

The (n,n')γ cross sections tabulated under MT = 52 are those measured by Presser, Bass, and Krüger[11] up to 7 MeV, with a constant 5 mb assumed thereafter. Isotropy in the CM system is specified for the angular distribution.

The (n,n')αd data given under MT = 91 are the same as in Ref. 1 up to 3 MeV, with the data of Hopkins et al.[10] taken into account between 4 and 10 MeV. The value of 433 mb assigned at 14 MeV is higher than the 403 mb given in Ref. 1, which is higher than the nominal 330 mb reported experimentally. A more detailed evaluation of the 14-MeV cross sections will be required to resolve the latter discrepancy. The extrapolation to 20 MeV of the (n,n')αd cross section was obtained by subtracting the sum of all other partials from the total cross section. For the angular distributions, the tabulated values of Ref. 1, extrapolated to 20 MeV, were used. The secondary energy distributions of Ref. 1 were approximated using ENDF/B Law 9 with θ values obtained by linear interpolation between the following points:

$E = 1.718$ MeV, $\theta = 0.05$ MeV
$E = 4.1$, $\theta = 0.75$
$E = 20.0$, $\theta = 8.40$

THE (n,p) AND (n,α) CROSS SECTIONS

The (n,p) cross sections up to 7 MeV reflect the data of Presser et al.[11] Above 7 MeV the data of Ref. 1 were used and extrapolated to 20 MeV. For the (n,α) cross sections the data between 2 and 15 MeV are those of Ref. 1. Extrapolation to 20 MeV was based on the measurements of Kern and Kreger[12] between 15 and 18 MeV.

ACKNOWLEDGMENTS

We are indebted to Leona Stewart for providing us with the data below 2 MeV and plots of the angular distributions of Lane et al., and to P. G. Young for performing the optical model calculations.

REFERENCES

1. E. D. Pendlebury, AWRE Report O-60/64 (1964).

2. C. A. Uttley, M. G. Sowerby, B. H. Patrick, and E. R. Rae, Proceedings of A Symposium on Neutron Standards and Flux Normalization held at ANL, October 1970. AEC Symposium Series 23 (August 1971).

3. Summary of CSEWG Meeting, October 7-8, 1970. Memorandum dated November 2, 1970, from S. Pearlstein to CSEWG participants.

4. K. M. Diment and C. A. Uttley, Nuclear Physics Division Progress Reports AERE-PR/NP-14 (1968), AERE-PR/NP-15 (1969), and AERE-PR/NP-16 (1969). Also private communication from C. A. Uttley to Leona Stewart.

5. A. Asami and M. C. Moxon, Nuclear Data for Reactors, Conference Proceedings, Helsinki, 1970. Vol. I, P. 153, IAEA, Vienna, 1970.

6. R. O. Lane, A. S. Langsdorf, Jr., J. E. Monahan, and A. J. Elwyn, Ann. Phys. 12, 135 (1961).

7. H. H. Knitter and M. Coppola, Euratom Report EUR-3454e (1967).

8. D. G. Foster, Jr. and D. W. Glasgow, Phys. Rev. C3, 576 (1971).

9. J. M. Peterson, A. Bratenahl, and J. P. Stoering, Phys. Rev. 120, 521 (1960).

10. J. C. Hopkins, D. M. Drake, and H. Condé, Los Alamos Scientific Laboratory Report LA-3765 (1967).

11. G. Presser, R. Bass, and K. Krüger, Nucl. Phys. A131, 679 (1969).

12. B. D. Kern and W. E. Kreger, Phys. Rev. 112, 926 (1958).

Appendix D

Summary for ^{10}B

SUMMARY DOCUMENTATION FOR B-10*

R. J. LaBauve

Los Alamos Scientific Laboratory

ABSTRACT

The following neutron data are given for B-10 in the energy range 1.0×10^{-5} eV to 15 MeV;

File 1. General description of data which follow.

File 2. Values of nuclear spin and effective scattering radius only.

File 3. Smooth cross sections for total, free-atom elastic, total inelastic, the inelastic level at 717 keV (MT = 51), the inelastic continuum, (n,d), (n,t), and (n,α). Also data for $\bar{\mu}$, ξ, and γ are included.

File 4. Angular distributions for elastic scattering (expressed as Legendre polynomial coefficients in the center-of-mass system), the first inelastic level, and the inelastic continuum. (The inelastic distributions are given as tabulated functions with the assumption of isotropy in center-of-mass system.)

File 5. Secondary energy distribution for the inelastic continuum (law 9 - evaporation spectrum).

INTRODUCTION

This evaluation for ^{10}B (MAT 1155) is essentially the same as that given for ENDF/B, Version I in MAT 1009 which was performed in October 1967 by D. C. Irving[1] of ORNL. At the CSEWG meeting of May 19-20, 1971, the Standards Subcommittee requested LASL to modify the (n,α) cross sections (MT = 107) of ^{10}B (MAT 1009) below 100 keV to conform to a recent evaluation of Sowerby et al.[2] In carrying out this request, it was also decided to 1) extend the range of modification to 150 keV, and 2) change the elastic scattering cross sections (MT = 2) in the modified energy range, and, of course, 3) change the total cross section to be equal to (MT2 + MT107).

The MAT 1009 evaluation contained 15 datum points from 1.0×10^{-5} eV to 150 keV. As the Sowerby evaluation is based on experimental measurements of the ratio $\sigma_{n,\alpha}(^6Li)/\sigma_{n,\alpha}(^{10}B)$, the modified cross sections were put on the same mesh (30 points) currently in use in the LASL ^6Li (MAT 1115) evaluation. The energy range was also extended to 150 keV as the MAT 1009 data were easier to merge at this point than at 100 keV. Moreover, it was felt that additional points were needed between 100 keV and 150 keV to compare with the ^6Li data.

The (n,α) modified cross sections, as recommended by Sowerby et al.,[2] are given by the formula:

---

* Work done under the auspices of the United States Atomic Energy Commission

$$\sigma_{n,\alpha}(^{10}B) = \frac{13.736}{\sqrt{E}} - 0.312 - 1.014 \times 10^{-2} \sqrt{E} + \frac{2.809 \times 10^5}{\sqrt{E}\,[(170.3-E)^2 + 2.243 \times 10^4]} \qquad (1)$$

with σ in barns and E in keV.

Incidentally, the $\sigma_{n,\alpha}$ data in the old ENDF/B ^{10}B evaluation did not differ greatly from this formula. The largest deviation from formula (1) for MAT 1009 was about 4%. The most significant change was the use of a more descriptive mesh.

This investigation did reveal a need to change the elastic scattering cross section, however. The ENDF/B evaluation was based on 1966 data of Mooring;[3] whereas this modification was based on more recent (1969) data of Asami and Moxon.[4]

FILE 1: GENERAL INFORMATION

A brief summary of these data is given in File 1. The atomic mass for ^{10}B was taken as 10.0130.

FILE 2: RESONANCE INFORMATION

A nuclear spin of 3 and effective scattering radius of 0.399×10^{-12} cm is given in File 2.

FILE 3: SMOOTH CROSS SECTIONS

Below 150 keV:

$\sigma_{elastic}$ - from experimental data of Asami and Moxon[4], with a constant value of 2.2 barns below 100 eV.

$\sigma_{n,\alpha}$ - given by formula (1) (Sowerby et al.[2]).

σ_{total} - $(\sigma_{elastic} + \sigma_{n,\alpha})$.

From 150 to 500 keV:

$\sigma_{elastic}$ - from the smooth curve of Mooring.[3] This agrees with the data of Lane et al.

$\sigma_{n,\alpha}$ - from a smooth curve through the (n,α) data of Mooring[3] and Gibbons,[5] with little weight placed on the data of Cox.[6]

σ_{total} - $(\sigma_{elastic} + \sigma_{n,\alpha})$.

Above 500 keV:

$\sigma_{elastic}$ - $(\sigma_{total} - \sigma_{inelastic} - \sigma_{n,\alpha} - \sigma_{n,d} - \sigma_{(n,t)2\alpha})$.

σ_{total} - from the data displayed in BNL325.[7,8] Some minor adjustments were made to remove spurious wiggles in the elastic cross section above 4 MeV.

$\sigma_{n,\alpha}$ - from the data of Davis [9] increased by 110 mb as suggested by a comparison to the data of Gibbons[6] and Nellis.[10]

$\sigma_{(n,t)2\alpha}$ - from a smooth curve was drawn by eye through the sparse data of Frye,[11] Wyman,[12] and Perkin.[13]

$\sigma_{n,d}$ - a smooth curve was drawn through the Be(d,n) data of Bardes[14] and Siemsson[15] and detailed balance used to obtain ^{10}B(n,d) values. These were connected by a straight line to a value at 14 MeV obtained by integration of the data of Valkovic[16] which is slightly higher than that of Ribe.[17]

$\sigma_{inelastic}$ - data from Day[18] on excitation of the first level which agrees with that of Nellis[10] was used to 4.5 MeV. This was connected by a smooth curve to a value at 14 MeV obtained by subtracting (n,d), (n,α), and (n,t) from the nonelastic value of MacGregor.[19]

The (n,n'd)2α reaction was neglected since it is contained for all practical purposes in the inelastic data.

FILE 4: SECONDARY ANGULAR DISTRIBUTIONS

The experimental data for elastic scattering angular distribution is sparse, consisting of a few points between .5 and 2 MeV (BNL400)[20] and one at 14 MeV (reported in Valković[16]). By all rights an optical model should not be valid for boron. However, the calculations of Agee[21] compare beautifully with the experimental data and have been used in the evaluation. The angular distribution was taken to be isotropic below .5 MeV in agreement with BNL400 and the data of Lane et al. (see Mooring).

The secondary angular distribution for inelastic scattering was assumed isotropic in the center-of-mass system.

FILE 5: SECONDARY ENERGY DISTRIBUTIONS

Up to 4.5 MeV, inelastic scattering was assumed to proceed via the first level at .71 MeV. Above 4.5 MeV an evaporation spectrum was used with the temperature as determined by Weinberg and Wigner.[22]

REFERENCES

1. D. C. Irving, ORNL-TM-1872 (Oct. 67).

2. M. G. Sowerby, B. H. Patrick, C. A. Uttley, K. M. Diment, AERE-R6316, 1970.

3. Mooring, Monahan and Huddleston, Nucl. Phys. 82, 16 (1966).

4. A. Asami and M. C. Moxon, AERE-R5980, April 1969.

5. Gibbons, private communication, obtained from Li(α,n) and branching ratio data. Gibbons and Macklin, Phys. Rev. 140, B324 (1965). Macklin and Gibbons, Phys. Rev. 114, 571 (1969).

6. Cox, Conf. on Neut. Cross Section Technology, 701 (1966).

7. D. J. Hughes and R. B. Schwart, BNL-325, 2nd edition (1958).

8. J. Stehn et al., BNL-325, 2nd edition, supplement 2 (1964).

9. Davis, Gabbard, Bonner and Bass, Nucl. Phys. 27, 448(1961).

10. Nellis and Morgan, Wash 1064, 186.

11. Frye and Gammel, Phys. Rev. 103, 328 (1956).

12. Wyman, Fryer, and Thorpe, Phys. Rev. 112, 1264 (1958).

13. Perkin, Phys. Rev. 81, 892 (1951).

14. Bardes and Owen, Phys. Rev. 120, 1369 (1960).

15. Siemssen, Cosack, and Felst, Nucl. Phys. 69, 209 (1965).

16. Valković et al., Phys. Rev. 139, B331 (1965).

17. Ribe and Seagrave, Phys. Rev. 94, 934 (1954).

18. Day and Walt, Phys. Rev. 117, 1330 (1960).

19. MacGregor, UCRL-5229 (1958).

20. Goldberg, May, and Stehn, BNL-400, 2nd edition (1962).

21. Agee and Rosen, LA-3538-MS (1966).

22. Weinberg and Wigner, The Physical Theory of Neutron Chain Reactors, p. 104.

Appendix E

Summary for ^{235}U

ENDF/B-III SUMMARY DOCUMENTATION FOR U^{235}

Contributors:

J. R. Smith, R. C. Young: Aerojet Nuclear(1)
H. Alter: Atomics International(2)
B. R. Leonard: Battelle Northwest Laboratory(3)
T. A. Pitterle, N. C. Paik, C. Durston: Westinghouse-ARD(4)

January 1972

(1) ANCR 1044
(2) AI-AEC-12916; AI-AEC-13013
(3) BNWL-1586
(4) WARD-4210T4-1

Thermal Data

The cross section shapes in this evaluation were derived, in general, by perturbing the results of an existing evaluation in order to reproduce the desired g values. The results of existing precision experimental differential measurements were used as a guide to the nature of the perturbations.

An objective of the evaluation was to provide cross-section shapes which were smooth in the thermal region and would not produce irregularities in the behavior of thermally averaged quantities as a function, say, of neutron temperature.

Point representation of the data intended to be in the ENDF/B-I files were received from C. R. Lubitz. These files covered the energy range 10^{-3} to 5 eV. It was determined that these files contained apparently unintended irregularities as large as one percent in the entries below 4 meV. These entries were smoothed and the data files extended to 10^{-4} eV. Calculation of "g" factors showed that the value of g_f was about 0.2 percent greater than the desired value. The original evaluation of the low-energy fission cross section was reported to have been made by fitting σ_f data of LRL and Hanford. Accordingly, a new fit was made including these data and also the fission data of Safford and Melkonian. A smooth fit was obtained, the main difference from the original fit being the reduction of the rise in cross section at energies below about 0.02 eV. The fission cross section was fitted simultaneously with the capture cross section using an alpha variation deduced by Westcott. In order to achieve a smooth fit to the fission data the file was modified slightly for energies up to 0.18 eV. The capture cross section file was modified for energies up to 0.1 eV.

The cross-section shapes derived for sub-thermal neutron energies are based in large part on precision total cross section measurements of Safford, et al. and the $1+\alpha$ measurement of Safford and Melkonian. The total cross section data of Safford, et al. obtained with liquid samples is shown below compared with the ENDF/B values.

	$\sigma_T \sqrt{E}$	
Neutron Energy	Safford, et al.	ENDF/B
0.000818 eV	115.71 ± 0.35	115.56
0.00128 eV	115.79 ± 0.29	115.49

The ENDF/B scattering cross sections at these energies is 16 b and a ±3 b uncertainty in this cross section contributes about ±0.1 b - $eV^{1/2}$ to calculated $\sigma_T\sqrt{E}$ values.

A summary of the 2200 m/s cross section parameters is given below.

σ_T	694.276
σ_S	15.776
σ_f	580.2
σ_c	98.3
ν	2.423 (Delayed + Prompt)
α	0.1694
η	2.07196

Resolved Resonance Region

The resolved resonance region for ^{235}U in ENDF/B-III extends from 1 to 82 eV. The description uses single level parameters plus a smooth file. Parameters were derived from a simultaneous fit to the following sets of data:

1) Simultaneous capture and fission measurements by deSaussure, et al. The strength of this experiment is that it measured the two most important partial cross sections of U^{235} simultaneously, under the same conditions of resolution and background. Moreover, care was taken to correct for such effects as backgrounds, resonance self-shielding, and scattering in the fission chamber. These data were used principally to indicate the ratio of capture to fission for the resonances.

2) Total cross section measurements of Michaudon. These data were obtained at liquid nitrogen temperatures and fairly high resolution. They turned out to give in most cases the best indication of the total widths of the resonances. The data were available only as cross section vs. energy, with results from several samples mixed together. Total cross sections are measured from transmission of samples, and the analysis should really be performed on the transmission data for each sample.

3) Fission cross sections measured by Blons, et al. on the Saclay linear accelerator. These data were obtained at liquid nitrogen temperature, with resolution similar to that of Michaudon's total cross section measurement. The Blons data are the best resolution fission data, but below about 35 eV the normalization gets progressively more erratic because of difficulty in interpreting the backgrounds in the presence of a B-10 filter used to eliminate low energy overlap neutrons.

4) Fission cross sections measured by Cao, et al. on the linear accelerator at C. B. N. M. (Geel). These data are the highest resolution room temperature measurements of σ_f for U^{235}. They are useful for comparing with the Blons data to confirm the effectiveness of the Doppler corrections in the analysis code. They go to a lower energy than the Blons data, 6 eV vs. 17 eV. However, the Cao data are troubled by erratic background corrections in the vicinity of resonances in filters used to determine backgrounds.

Cross Section Normalization

Since this analysis covered only the resonance region above 1.0 eV, it was necessary to normalize all data to the existing ENDF/B-II low energy file. Of the principal data sets, only the deSaussure measurements extends to this low energy. His fission data were raised by 1.5% to bring their integral from 0.45 eV to 1.0 eV into better agreement with that from the ENDF/B low energy file. The difference in the capture integrals was 2.4%. Nevertheless, the capture was raised only 1.5% in order that deSaussure's α ratios might be preserved.

The Cao data were raised 7% to bring them into agreement with the renormalized deSaussure data. The Blons data, which already agreed well with the renormalized deSaussure data above 40 eV, were given an energy-dependent renormalization. The ratios to deSaussure values of a series of incremental resonance integrals were fit with a fourth degree polynomial. This polynomial was then used to normalize the Blons data. The resulting correction ranged from about 19% at 18 eV to zero at 40 eV.

Single level parameters were derived by fitting the experimental data by means of the automatic iterative fitting features of the Automated Cross Section Analysis Program (ACSAP). A value of 11.5 b was used for the potential scattering cross section.

ACSAP will upon request print and plot the differences between experimental points and the cross sections calculated from parameters. Such difference outputs were used in constructing the smooth files. In order to maintain proper α values, the deSaussure data were used as much as possible in constructing these different files. However, this ideal had to be abandoned above about 35 eV, as degraded resolution spread the intrinsic difference well away from the resonances to which they apply.

The scattering smooth file represents the difference between the single-level prediction and a multilevel calculation adjusted to minimize the effects of interference imbalance at the two ends of the resolved resonance region.

The parameters plus smooth file yield resonance integrals between 1 and 82 eV of 170.6 and 104.5 b for fission and capture, respectively. The average alpha is 0.613 in this region, compared to a value of 0.617 calculated directly from deSaussure's data.

The root mean square fractional difference between the fit and the individual data sets averages about 3.5%. This figure comes from an analysis of partial resonance integrals from the fit and from the data. Resonance energies are probably good to 0.050 eV and resonance widths to 10%. Overall error in cross sections is about 5%.

For further details see the full report, ANCR-1044.

Unresolved Resonance Region

The ENDF/B-III evaluation in the unresolved energy range is based primarily on the experimental data of deSaussure. At the time of this evaluation, detailed resolution cross sections were not available from the recent measurements of Perez, Blons or Lemley. For this evaluation, a continuous curve of the fission cross section was constructed so as to reproduce the decimal interval averages of the experimental data.

Differences between the present and ENDF/B-II evaluations are due to inclusion of new experimental data, renormalization of the experimental data to recent $^{10}B(n,\alpha)$ cross sections and to differences in methods of constructing a smooth curve. The ENDF/B-III evaluation was obtained by averaging the data of deSaussure over lethargy intervals and by passing a continuous curve through the intervals.

The unresolved resonance parameters in the ^{235}U ENDF/B-III file were modified to yield the evaluated fission cross section and to reproduce the ENDF/B-II alpha value as closely as possible. The parameters were obtained by adjusting the parameters to yield the desired fission and alpha values.

The cross sections which were fitted and the s-wave strength function and fission widths resulting from the fitting procedure are given in WARD-4210T4-1. In this evaluation, the upper energy range for the unresolved resonance parameters was cut off at 25 keV and pointwise data was used above this energy.

Data Above 25 keV

a. Fission Cross Section

Qualitatively, the experimental data in the energy range ~25 keV to 100 keV falls into two groups: the low fission values of Szabo and Lemley which use ^{6}Li as a standard and the higher fission values such as White and DeSaussure which use hydrogen and ^{10}B as a standard. However the recent data of Gwin using a ^{10}B standard supports the low fission values. The data of Blons tend to support the lower fission values while the data of Knoll are in good agreement with White's data. The use of ^{6}Li or ^{10}B as a standard does not appear to be the source of the discrepancy because the ^{10}B cross section used for normalization is partially derived from the ^{6}Li data consistent with Lemley's normalization.

No clear choice based on the differential data can be made at the present time between the low and high fission values. Integral testing against critical assemblies indicates that use of the lower fission values would require major cross section adjustments – particularly very low capture cross sections for ^{238}U in order to obtain eigenvalues as close as 1% less than unity. The latter is particularly true for soft spectrum assemblies typical of interest in LMFBR design. For this reason, the choice for the present evaluation is based on the data of Perez, White and Knoll.

The experimental data for the U^{235} fission cross section above 100 keV, as utilized in the present evaluation is based on the measurements of White, Szabo and Smith as corrected by Hansen. The data of Poenitz indicates notably lower fission values than the other measurements in this energy range. The principal new measurement since the ENDF/B-II evaluation is that of Szabo. The measurements of Kappeler were reported since the present evaluation. The Szabo measurement is the principal source of the differences between the present and ENDF/B evaluations between 0.15 and 1.0 MeV.

Between 1.0 and 10.0 MeV, the only measurements with an accuracy of better than 5% are those of White at 2.25 and 5.4 MeV with relatively poor shape determination in this energy range. In the present evaluation, the U^{235} fission cross section between 1.0 and 10.0 MeV was evaluated within existing uncertainties in order to increase the cross section by 1 to 3% primarily to enhance the U^{238} fission cross section for which the most accurate measurements are relative to U^{235} fission. Above 10 MeV, the present evaluation is evaluated to obtain a 14 MeV cross section of 2.13 barns.

The detailed structure in the U^{235} fission cross section is important as it leads to variations of up to about 3% between groups of multigroup cross sections averaged over quarter lethargy widths typical of many LMFBR calculations and it significantly influences the Pu^{239} fission cross sections derived from ratios of $^{239}Pu/^{235}U$ fission. For the present evaluation, a compromise structure was selected based on the measurements of Perez and Lemley. The evaluated structure was normalized to obtain the 10 keV evaluated average cross section. This structure could possibly be improved when pointwise data from the measurements of Perez and Lemley become available.

b. Capture Cross Section

The capture cross section above 25 keV was obtained by folding the newly evaluated fission cross section into the ENDF/B-II alpha values. Results of discussions with deSaussure (ORNL) relative to the two sets of ORNL alpha data (the lower values of Weston et al. and the higher values of deSaussure, et al.) provided no basis of establishing one data set over the other. A weighted average of the two sets was used. The alpha evaluation is summarized as follows: 15-40 keV (Schmidt evaluation); 40-60 keV (joining of new and Schmidt evaluation); 60-200 keV (5-7% higher than Schmidt evaluation); 200-400 keV (smooth joining of new and Schmidt evaluations); above 400 keV (Schmidt evaluation).

c. Total Cross Section

Above 2 MeV, the data of Glasgow and Foster was used to represent the total cross section. Below 2 MeV the total cross section of MAT 1044 was adopted.

d. Elastic Scattering Cross Section

The elastic scattering cross section was obtained by subtracting from the total cross section the sum of the remaining partial cross sections.