(1)

NUCLEAR DATA AND MEASUREMENTS SERIES

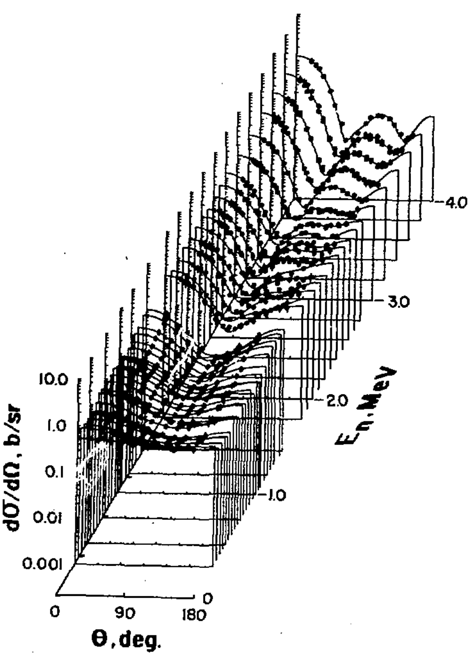
ANL/NDM-7[illegible]

EVALUATION OF THE ^{238}U NEUTRON TOTAL CROSS SECTION*

by

A. Smith**, W. Poenitz**, and R. Howerton†

December 1982



DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

ARGONNE NATIONAL LABORATORY,
ARGONNE, ILLINOIS 60439, U.S.A.

MASTER

ANL/NDM--74

DE83 007898

ANL/NDM-74

EVALUATION OF THE ^{238}U NEUTRON TOTAL CROSS SECTION*

by

A. Smith**, W. Poenitz**, and R. Howerton+

December 1982

*This work supported by the U.S. Department of Energy
**Argonne National National Laboratory
+Lawrence Livermore National Laboratory

Applied Physics Division
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439
USA

NUCLEAR DATA AND MEASUREMENTS SERIES

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

OTHER ISSUES IN THE ANL/NDM SERIES ARE:

ANL/NDM-1 Cobalt Fast Neutron Cross Sections-Measurement and Evaluation by P. T. Guenther, P. A. Moldauer, A. B. Smith, D. L. Smith and J. F. Whalen, July 1973.

ANL/NDM-2 Prompt Air-Scattering Corrections for a Fast-Neutron Fission Detector: $E_n \leq 5$ MeV by Donald L. Smith, September 1973.

ANL/NDM-3 Neutron Scattering from Titanium; Compound and Direct Effects by E. Barnard, J. deVilliers, P. Moldauer, D. Reitmann, A. Smith and J. Whalen, October 1973.

ANL/NDM-4 ^{90}Zr and ^{92}Zr; Neutron Total and Scattering Cross Sections by P. Guenther, A. Smith and J. Whalen, July 1974.

ANL/NDM-5 Delayed Neutron Data - Review and Evaluation by Samson A. Cox, April 1974.

ANL/NDM-6 Evaluated Neutronic Cross Section File for Niobium by R. Howerton, Lawrence Livermore Laboratory and A. Smith, P. Guenther and J. Whalen, Argonne National Laboratory, May 1974.

ANL/NDM-7 Neutron Total and Scattering Cross Sections of Some Even Isotopes of Molybdenum and the Optical Model by A. B. Smith, P. T. Guenther and J. F. Whalen, June 1974.

ANL/NDM-8 Fast Neutron Capture and Activation Cross Sections of Niobium Isotopes by W. P. Poenitz, May 1974.

ANL/NDM-9 Method of Neutron Activation Cross Section Measurement for E_n = 5.5-10 MeV Using the D(d,n)He-3 Reaction as a Neutron Source by D. L. Smith and J. W. Meadows, August 1974.

ANL/NDM-10 Cross Sections for (n,p) Reactions on ^{27}Al, $^{46,47,48}Ti$, $^{54,56}Fe$, ^{58}Ni, ^{59}Co and ^{64}Zn from Near Threshold to 10 MeV by Donald L. Smith and James W. Meadows, January 1975.

ANL/NDM-11 Measured and Evaluated Fast Neutron Cross Sections of Elemental Nickel by P. Guenther, A. Smith, D. Smith and J. Whalen, Argonne National Laboratory and R. Howerton, Lawrence Livermore Laboratory, July 1975.

ANL/NDM-12 A Spectrometer for the Investigation of Gamma Radiation Produced by Neutron-Induced Reactions by Donald L. Smith, April 1975.

ANL/NDM-13 Response of Several Threshold Reactions in Reference Fission Neutron Fields by Donald L. Smith and James W. Meadows, June 1975.

ANL/NDM-14 Cross Sections for the $^{66}Zn(n,p)^{66}Cu$, $^{113}In(n,n')^{113m}In$ and $^{115}In(n,n')^{115m}In$ Reactions from Near Threshold to 10 MeV by Donald L. Smith and James W. Meadows, July 1975.

ANL/NDM-15 Radiative Capture of Fast Neutrons in ^{165}Ho and ^{181}Ta by W. P. Poenitz, June 1975.

ANL/NDM-16 Fast Neutron Excitation of the Ground-State Rotational Band of ^{238}U by P. Guenther, D. Havel and A. Smith, September 1975.

ANL/NDM-17 Sample-Size Effects in Fast-Neutron Gamma-Ray Production Measurements: Solid-Cylinder Samples by Donald L. Smith, September 1975.

ANL/NDM-18 The Delayed Neutron Yield of ^{238}U and ^{241}Pu by J. W. Meadows January 1976.

ANL/NDM-19 A Remark on the Prompt-Fission-Neutron Spectrum of ^{252}Cf by P. Guenther, D. Havel, R. Sjoblom and A. Smith, March 1976.

ANL/NDM-20 Fast-Neutron-Gamma-Ray Production from Elemental Iron: $E_n \lesssim$ 2 MeV by Donald L. Smith, May 1976.

ANL/NDM-21 Note on the Experimental Determination of the Relative Fast-Neutron Sensitivity of a Hydrogenous Scintillator by A. Smith, P. Guenther and R. Sjoblom, June 1976.

ANL/NDM-22 Note on Neutron Scattering and the Optical Model Near A=208 by P. Guenther, D. Havel and A. Smith, September 1976.

ANL/NDM-23 Remarks Concerning the Accurate Measurement of Differential Cross Sections for Threshold Reactions Used in Fast-Neutron Dosimetry for Fission Reactors by Donald L. Smith, December 1976.

ANL/NDM-24 Fast Neutron Cross Sections of Vanadium and an Evaluated Neutronic File by P. Guenther, D. Havel, R. Howerton, F. Mann, D. Smith, A. Smith and J. Whalen, May 1977.

ANL/NDM-25 Determination of the Energy Scale for Neutron Cross Section Measurements Employing a Monoenergetic Accelerator by J. W. Meadows, January 1977.

ANL/NDM-26 Evaluation of the In-115(n,n')In-115m Reaction for the ENDF/B-V Dosimetry File by Donald L. Smith, December 1976.

ANL/NDM-27 Evaluated (n,p) Cross Sections of ^{46}Ti, ^{47}Ti and ^{48}Ti by C. Philis and O. Bersillon, Bruyeres-le-Chatel, France and D. Smith and A. Smith, Argonne National Laboratory, January 1977.

ANL/NDM-28 Titanium-II: An Evaluated Nuclear Data File by C. Philis, Centre d'Etudes de Bruyéres-le-Châtel, R. Howerton, Lawrence Livermore Laboratory and A. B. Smith, Argonne National Laboratory, June 1977.

ANL/NDM-29 Note on the 250 keV Resonance in the Total Neutron Cross Section of ^{6}Li by A. B. Smith, P. Guenther, D. Havel and J. F. Whalen, June 1977.

ANL/NDM-30 Analysis of the Sensitivity of Spectrum-Average Cross Sections to Individual Characteristics of Differential Excitation Functions by Donald L. Smith, March 1977.

ANL/NDM-31 Titanium-I: Fast Neutron Cross Section Measurements by P. Guenther, D. Havel, A. Smith and J. Whalen, May 1977.

ANL/NDM-32 Evaluated Fast Neutron Cross Section of Uranium-238 by W. Poenitz, E. Pennington, and A. B. Smith, Argonne National Laboratory and R. Howerton, Lawrence Livermore Laboratory, October 1977.

ANL/NDM-33 Comments on the Energy-Averaged Total Neutron Cross Sections of Structural Materials by A. B. Smith and J. F. Whalen, June 1977.

ANL/NDM-34 Graphical Representation of Neutron Differential Cross Section Data for Reactor Dosimetry Applications by Donald L. Smith, June 1977.

ANL/NDM-35 Evaluated Nuclear Data File of Th-232 by J. Meadows, W. Poenitz, A. Smith, D. Smith and J. Whalen, Argonne National Laboratory and R. Howerton, Lawrence Livermore Laboratory, February 1978.

ANL/NDM-36 Absolute Measurements of the $^{233}U(n,f)$ Cross Section Between 0.13 and 8.0 MeV by W. P. Poenitz, April 1978.

ANL/NDM-37 Neutron Inelastic Scattering Studies for Lead-204 by D. L. Smith and J. W. Meadows, December 1977.

ANL/NDM-38 The Alpha and Spontaneous Fission Half-Lives of ^{242}Pu by J. W. Meadows, December 1977.

ANL/NDM-39 The Fission Cross Section of ^{239}Pu Relative to ^{235}U from 0.1 to 10 MeV by J. W. Meadows, March 1978.

ANL/NDM-40 Statistical Theory of Neutron Nuclear Reactions by P. A. Moldauer, February 1978.

ANL/NDM-41 Energy-Averaged Neutron Cross Sections of Fast-Reactor Structural Materials by A. Smith, R. McKnight and D. Smith, February 1978.

ANL/NDM-42 Fast Neutron Radiative Capture Cross Section of ^{232}Th by W. P. Poe'itz and D. L. Smith, March 1978.

ANL/NDM-43 Neutron Scattering from ^{12}C in the Few-MeV Region by A. Smith, R. Holt and J. Whalen, September 1978.

ANL/NDM-44 The Interaction of Fast Neutrons with ^{60}Ni by A. Smith, P. Guenther, D. Smith and J. Whalen, January 1979.

ANL/NDM-45 Evaluation of $^{235}U(n,f)$ Between 100 KeV and 20 MeV by W. P. Poenitz, July 1979.

ANL/NDM-46 Fast-Neutron Total and Scattering Cross Sections of ^{107}Ag in the MeV Region by A. Smith, P. Guenther, G. Winkler and J. Whalen, January 1979.

ANL/NDM-47 Scattering of MeV Neutrons from Elemental Iron by A. Smith and P. Guenther, March 1979.

ANL/NDM-48 ^{235}U Fission Mass and Counting Comparison and Standardization by W. P. Poenitz, J. W. Meadows and R. J. Armani, May 1979.

ANL/NDM-49 Some Comments on Resolution and the Analysis and Interpretation of Experimental Results from Differential Neutron Measurements by Donald L. Smith, November 1979.

ANL/NDM-50 Prompt-Fission-Neutron Spectra of ^{233}U, ^{235}U, ^{239}Pu and ^{240}Pu Relative to that of ^{252}Cf by A. Smith, P. Guenther, G. Winkler and R. McKnight, September 1979.

ANL/NDM-51 Measured and Evaluated Neutron Cross Sections of Elemental Bismuth by A. Smith, P. Guenther, D. Smith and J. Whalen, April 1980.

ANL/NDM-52 Neutron Total and Scattering Cross Sections of ^{6}Li in the Few MeV Region by P. Guenther, A. Smith and J. Whalen, February 1980.

ANL/NDM-53 Neutron Source Investigations in Support of the Cross Section Program at the Argonne Fast-Neutron Generator by James W. Meadows and Donald L. Smith, May 1980.

ANL/NDM-54 The Nonelastic-Scattering Cross Sections of Elemental Nickel by A. B. Smith, P. T. Guenther and J. F. Whalen, June 1980.

ANL/NDM-55 Thermal Neutron Calibration of a Tritium Extraction Facility using the $^{6}Li(n,t)^{4}He/^{197}Au(n,\gamma)^{198}Au$ Cross Section Ratio for Standardization by M. M. Bretscher and D. L. Smith, August 1980.

ANL/NDM-56 Fast-Neutron Interactions with ^{182}W, ^{184}W and ^{186}W by P. T. Guenther, A. B. Smith and J. F. Whalen, December 1980.

ANL/NDM-57 The Total, Elastic- and Inelastic-Scattering Fast-Neutron Cross Sections of Natural Chromium, Peter T. Guenther, Alan B. Smith and James F. Whalen, January 1981.

ANL/NDM-58 Review of Measurement Techniques for the Neutron Capture Process by W. P. Poenitz, August 1981.

ANL/NDM-59 Review of the Importance of the Neutron Capture Process in Fission Reactors, Wolfgang P. Poenitz, July 1981.

ANL/NDM-60 Neutron Capture Activation Cross Sections of ^{94}Zr, ^{96}Zr, ^{98,100}Mo, and ^{110,114,116}Cd at Thermal and 30 keV Energy, John M. Wyrick and Wolfgang P. Poenitz, (to be published).

ANL/NDM-61 Fast-neutron Total and Scattering Cross Sections of ^{58}Ni by Carl Budtz-Jørgensen, Peter T. Guenther, Alan B. Smith and James F. Whalen, September 1981.

ANL/NDM-62 Covariance Matrices and Applications to the Field of Nuclear Data, by Donald L. Smith, November 1981.

ANL/NDM-63 On Neutron Inelastic-Scattering Cross Sections of ^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{239}U, and ^{239}Pu and ^{240}Pu by Alan B. Smith and Peter T. Guenther, January 1982.

ANL/NDM-64 The Fission Fragment Angular Distributions and Total Kinetic Energies for ^{235}U(n,f) from .18 to 8.83 MeV by James W. Meadows, and Carl Budtz-Jørgensen, January 1982.

ANL/NDM-65 Note on the Elastic Scattering of Several MeV Neutrons from Elemental Calcium by Alan B. Smith and Peter T. Guenther, March 1982.

ANL/NDM-66 Fast-neutron Scattering Cross Sections of Elemental Silver by Alan B. Smith and Peter T. Guenther, May 1982.

ANL/NDM-67 Non-evaluation Applications for Covariance Matrices by Donald L. Smith, July 1982.

ANL/NDM-68 Fast-neutron Total and Scattering Cross Sections of ^{103}Rh by Alan B. Smith, Peter T. Guenther and James F. Whalen July 1982.

ANL/NDM-69 Fast-neutron Scattering Cross Sections of Elemental Zirconium by Alan B. Smith and Peter T. Guenther (to be published).

ANL/NDM-70 Fast-neutron Total and Scattering Cross Sections of Niobium by Alan B. Smith, Peter T. Guenther and James F. Whalen July 1982.

ANL/NDM-71 Fast-neutron Total and Scattering Cross Sections of Elemental Palladium by Alan B. Smith, Peter T. Guenther and James F. Whalen, June 1982.

ANL/NDM-72 Fast-neutron Scattering from Elemental Cadmium by Alan B. Smith and Peter T. Guenther (to be published).

ANL/NDM-73 Fast-neutron Elastic Scattering Cross Sections of Elemental Tin by Alan B. Smith, Carl Budtz-Jørgensen and Peter T. Guenther, July 1982.

Table of Contents

EVALUATION OF THE ^{238}U NEUTRON TOTAL CROSS SECTION*

by

A. Smith, W. P. Poenitz, and R. J. Howerton

ABSTRACT

Experimental energy-averaged neutron total cross sections of ^{238}U were evaluated from 0.044 to 20.0 MeV using rigorous numerical methods. The evaluated results are presented together with the associated uncertainties and correlation matrix. They indicate that this energy-averaged neutron total cross section is known to better than 1% over wide energy regions. There are somewhat larger uncertainties at low energies (e.g., $\lesssim$ 0.2 MeV), near 8 MeV and above 15 MeV. The present evaluation is compared with values given in ENDF/B-V.

*This work supported by the U.S. Department of Energy.

I. INTRODUCTION

Precise knowledge of the energy-averaged neutron total cross sections of ^{238}U is important to: i) macroscopic neutron transport, where the total cross section governs the mean-free-path, ii) evaluation, where the cross section is the envelope to which the partial cross sections must conform, and iii) model development, where the total cross section forms a benchmark for testing model parameters. Here we assume the energy-averaged region extends upward in the energy from the inelastic-scattering threshold (45 keV). At lower energies, resolved and unresolved resonance treatments are more appropriate. The upper-energy limit of evaluations relevant to fission-energy systems is conventionally set at 20 MeV and is accepted in this work.

That the ^{238}U energy-averaged neutron total cross section should be well known was recognized at the recent NEANDC Specialist's Meeting on Fast-Neutron Scattering on Actinide Nuclei[1], and there it was recommended that a rigorous evaluation be undertaken in an effort to quantify the contemporary accuracies. This work is in response to that recommendation.

II. OUTLINE OF THE DATA BASE

The majority of the data base was assembled from the numerical data files of the Lawrence Livermore National Laboratory.[2] A few additional data sets were identified in CINDA[3], extracted from professional journals and/or reports, and added to the data base. Several more data sets are known to exist but they have not been formally reported and their authors did not choose to make them available through private communication. The complete experimental data base, outlined in the Appendix, consisted of 33 sets. All were the result of transmission measurements using either white- or monoenergetic-neutron sources. They span a period of approximately three decades. Only two of the sets are more recent than 1974, and only one of these extends above 0.1 MeV. Many are not well specified and their assessment is, to a considerable extent, a subjective exercise.

III. EVALUATION PROCEDURES AND RESULTS

The evaluation proceeded in five steps. First, all of the data sets were inspected using large-scale graphical displays. From this inspection a few data sets were abandoned as: i) being grossly discrepant with the body of the data base, ii) consisting of values with a large scatter widely exceeding statistical uncertainties, and iii) having such large stated uncertainties as to result in a negligible weight in the subsequent evaluation process. Several data sets systematically deviated from the majority of the data in some energy regions. Most of these cases resulted from white-source measurements, and the effect was most prominent in the lower-energy extreme of the data set where background effects could have perturbed the experimental results. In these cases the data set was truncated, deleting regions of obvious qualitative discrepancy. Some data sets were identified as superseded by later and improved work by the same authors. In these instances the earlier and less accurate data were deleted from the data base. At low energies self-shielding effects are a concern.[4] Two sets of data appeared to give attention to this matter, providing either fully corrected data[4] or sufficient information to make such corrections possible[5]. Thus, other low-energy ($\lesssim$ 0.2 MeV) data sets with results significantly below the more reliable corrected information were abandoned as probably distorted by self shielding. The above qualitative judgements are indicated in the Appendix.

Data, having passed through the above filtering process, were then examined for error specification. Only two sets of data reasonably defined both statistical and systematic uncertainties.[4,5] In these two cases the uncertainties were accepted as stated. For the majority of the data sets only the statistical uncertainties were clearly stated and they were accepted as given by the authors. Where no statistical uncertainty was given a 1-1/2% value was assumed as a conservative estimate of the statistical uncertainty of a typical transmission experiment. Estimates of systematic uncertainties are far more difficult to quantify as they can result from a number of effects; some of the more prominent of these were obviously not recognized by the experimenters. Above 0.2 MeV, in the absence of a specified systematic uncertainty, a value of 1-1/2% was assumed as representative of a measurement with reasonable attention to background, sample properties, deadtime effects, etc. Below 0.2 MeV the systematic uncertainties were assumed to increase linearly with decreasing energy to 3% at 44 keV. This increase was assumed to be an approximate estimate of unknown selfshielding perturbations. In the absence of quantitative specification, the above uncertainty estimates are obviously qualitative generalizations. These estimates may have shortcomings for any one data set but, hopefully, they generally reflect the physical situation. In the absence of more detailed information there is little alternative. One data set appeared to be of excellent relative quality (i.e., the shape was very consistent with the trend body of the data but it was systematically lower in magnitude than the assemblage of data by 2-3 times the above systematic uncertainty estimates. For this set, the systematic uncertainty estimates were increased by a factor of two.

The goal was an evaluated energy-averaged cross section. Below $\approx$ 0.1 MeV structure can be 5% or more in magnitude in a 5 keV average[5] and the structure persists into the MeV region with very much smaller magnitudes.[6] Thus the data were averaged over energy increments that would generally smooth fluctuations while maintaining the overall energy dependence. These increments were; 10 keV < 0.2 MeV, 25 keV from 0.2 to 1.0 MeV, 50 keV from 1.0 to 5.0 MeV, and 100 keV from 5.0 to 20.0 MeV. Since averaging increments generally exceeded the experimental energy resolutions, the energy-averaging increments were considered in the subsequent numerical procedures. The weighted averages and associated statistical uncertainties were constructed assuming weighting factors inversely proportional to the square of the statistical uncertainties. The systematic uncertainties were assumed to be constant over the averaging increment. Thus the systematic uncertainties of the averaged values were taken to be a simple average[5] of the individual components. The averaging procedure also reduced the data base to manageable proportions (some of the input data sets consisted of thousands of individual values).

The above averaged experimental data were evaluated using the numerical procedures as implemented in the Gauss-Markov-Aitken least-squares nuclear-data evaluation program, GMA.[7] The result is the "best" estimate of the global experimental knowledge of the ^{238}U energy-averaged neutron total cross section, inclusive of uncertainties and correlation matrix. This evaluated result is entirely founded on experiment and may reflect experimental artifacts that are not necessarily physically acceptable.

In order to remove small experimental artifacts from the evaluation and to provide a physically acceptable vehicle for interpolating between evaluated quantities, a conventional spherical optical model was chi-square fitted to the evaluated results, weighting with the respective evaluation uncertainties. Both real and imaginary potential strengths were assumed to have a linear energy dependence. The fitting procedure[8] varied concurrently the eight parameters; real and imaginary, strengths, radii, diffusenesses, and energy dependences. The resulting parameters are given in Table 1. It should be emphasized that they are a parameterization of the evaluation suitable for the assessment of general trends and for interpolation between evaluated quantities. They should not be construed as representative of a general model suitable for all aspects of the fast-neutron interaction with ^{238}U. In fact, the parameters of Table 1 are not conventional and the model gives no attention to the known deformation of the target. These shortcomings do not impair the usefulness of the model in the present context of energy interpolation.

The uncertainties associated with the experimental evaluation are shown in Fig. 1. They are generally less than 1% and increase to 2-3% only in a local region near 0.15 MeV. The optical-model results provide an excellent parameterization of the evaluated experimental results, as shown in Figs. 1 and 2. Generally, the differences between the model- and evaluation-results are smaller than the uncertainties of the latter alone. The model smooths small data and evaluation artifacts. The most prominent of these is near 8 MeV where the model results are ≈ 2.5% larger than those of the numerical evaluation. This local artifact was identified as primarily due to one highly weighted data set which appears to have low values in this local region. The model was used to interpolate the numerical-evaluated quantities to a regular energy mesh to obtain the final evaluated results and associated uncertainties (given by the numerical evaluation) given in Table 2. The corresponding correlation matrix is given in Table 3.

IV. COMPARISONS WITH ENDF/B-V

The ENDF/B-V evaluation[9] is so similar to the present results that differences are not clearly evident in graphical comparisons. The ENDF/B-V values are systematically larger than those implied by the present work but by amounts well within the respective uncertainties. Illustrative numerical comparisons are given in Table 4. From 0.2 to 15.0 MeV, the two evaluations agree to better than 1%. At the very lowest and very highest energies the ENDF/B-V values are 1-2% larger than those of the present evaluation. The differences throughout the energy range are well within the respective uncertainties. The uncertainties associated with present evaluation are 4-5 times smaller than those suggested for ENDF/B-V over much of the energy range.[10] ENDF/B-V is a subjective construction based upon graphical representations of much the same data as used in the present work. The latter procedure gave essentially the same result as the numerical methods of the present work but without the detailed uncertainty specifications.

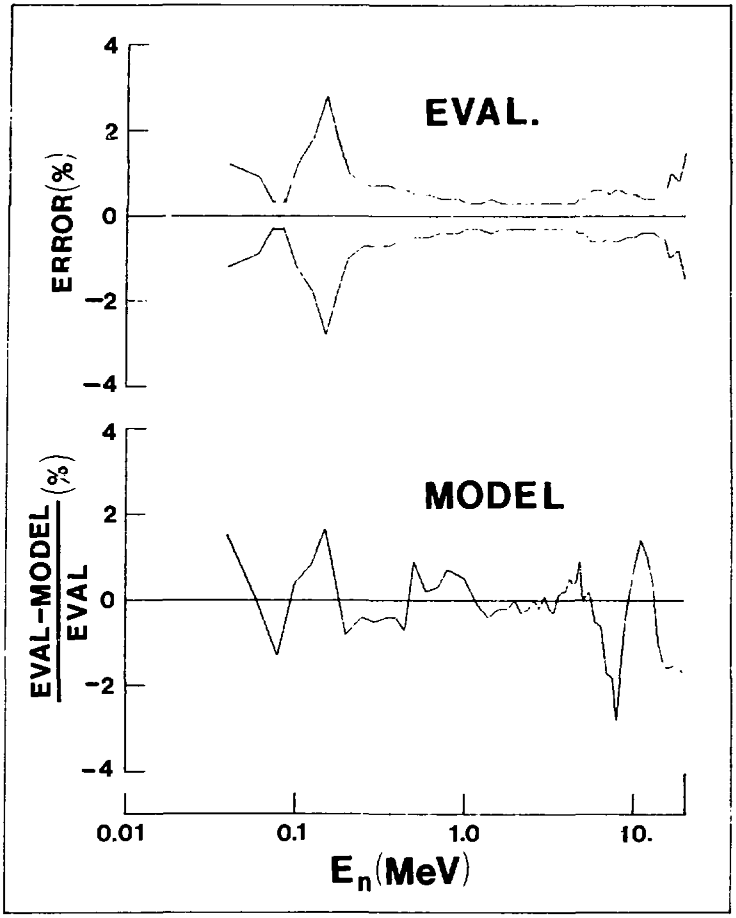


Fig. 1. Experimental-Evaluation Uncertainties (Upper) and Deviation of the Model Result from the Experimental-Evaluation (Lower).

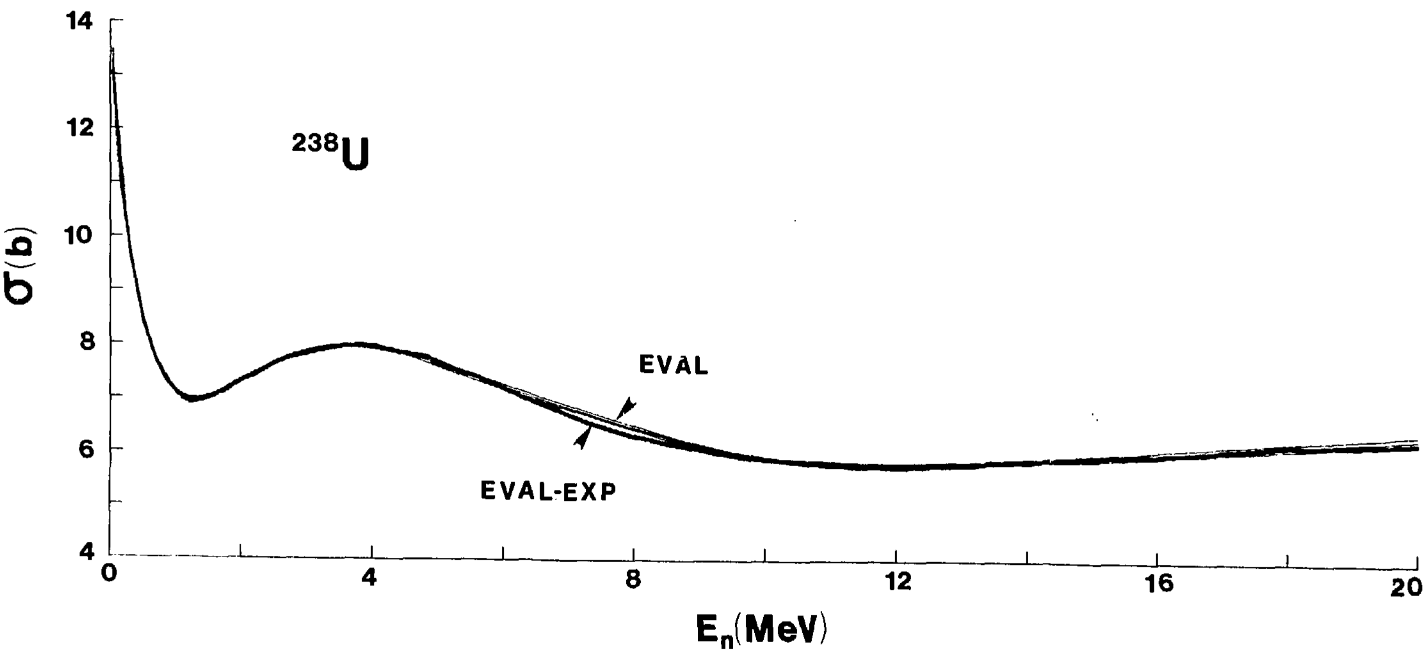


Fig. 2. Evaluated Neutron Total Cross Sections of ^{238}U. Heavy curve indicates the result of the numerical experimental evaluation, the light curves the final evaluation (± uncertainty).

V. CONCLUDING COMMENTS

The present evaluation defines the ^{238}U energy-averaged neutron total cross section to better than 1% over very wide energy ranges with only a limited area near 0.15 MeV where the uncertainties increase to 2-3%. The evaluation gives a reasonable estimate of uncertainty and the associated correlation matrix. The results imply a well known neutron total cross section of essentially a "standard" quality. A model is deduced which gives an excellent parameterization of the evaluation results and which is suitable for interpolating between the evaluated quantities. That model may not be appropriate in other contexts as it is a simple concept. In particular, it ignores collective-deformation effects. The numerical evaluation procedures are rigorous. However, underlying them are subjective judgements necessary for providing the requisite experimental data base. Much of the data base is old with poorly defined (or non-existent) specification of uncertainties, particularly of systematic uncertainties. In a number of instances it appears that very significant systematic uncertainties have escaped the knowledge of the respective experimentalists. Further improvement will require very appreciable new input with detailed attention given to both statistical and systematic uncertainties. That event seems unlikely in the near future. The present results are essentially identical to those of ENDF/B-V. They do provide a quantification of uncertainty that was heretofore lacking.

REFERENCES

1. Proc. of Meeting on Fast-Neutron Scattering on Actinide Nuclei, NEANDC-158"U", Paris (1981).

2. An Integrated System for Production of Neutronics and Photonices Calculational Constants; Neutron-Induced Interactions: Tabulated Experimental Data, M. H. MacGregor et al., Lawrence Livermore Laboratory Report, UCRL-50400, Vol. 10, Rev. 1 (1976) with the addition of later data from the LLWL-ECSIL data files.

3. CINDA, An Index to the Literature on Microscopic Neutron Data, IAEA Press (1982).

4. W. Poenitz et al., Nucl. Sci. and Eng., 78 333 (1981).

5. D. Olsen et al., Oak Ridge National Laboratory Report, ORNL/TM-5915 (1977); also private communication.

6. D. Kopsch et al., Proc. Conf. on Nucl. Data for Reactors, 2 39 (1970).

7. W. Poenitz, Proc. Conf. on Nucl. Data Evaluation Methods and Procedures, Brookhaven National Laboratory Report, BNL-NCS-51363 (1981).

8. ABAREX, a Spherical Optical-Model Code, P. Moldauer, private communication (1982).

9. Evaluated Nuclear Data File-B, Version V, Brookhaven National Laboratory Report, BNL-17541 (1979).

10. W. Poenitz et al., Argonne National Laboratory Report, ANL/NDM-32, (1977).

APPENDIX - DATA BASE

1. C. Uttley *et al.*, Conf. on Nucl. Data, Paris (1966).
   White Source, E_n < 1.0 MeV, used.

2. K. Seth *et al.*, Conf. on Neut. Phys., Antwerp (1966).
   Monoenergetic Source, E_n = 0.05 - 0.65 MeV, poor magnitude and peculiar structure, abandoned.

3. J. Whalen, private communication (1969).
   Monoenergetic Source, E_n = 0.1 - 1.5 MeV, abandoned below 0.2 MeV because of self-shielding uncertainties, used at higher energies.

4. J. Cabe, CEA-R-4524 (1973).
   Monogenergetic Source, E_n = 0.1 - 1.2 MeV, magnitudes appear in error above 0.9 MeV. Used at lower energies.

5. W. Heaton *et al.*, Nucl. Sci. and Eng., 54 322 (1974).
   White Source, E_n = 0.49 - 15.2 MeV, used.

6. L. Galloway, TID-11005 (1960).
   Monoenergetic Source, E_n = 0.5 - 1.0 MeV, few values, large scatter, abandoned.

7. C. Goulding *et al.*, Nucl. Sci. and Eng., 50 242 (1973).
   White Source, E_n = 0.7 - 20.0 MeV, used.

8. A. Smith, unpublished data (1969).
   Monoenergetic Source, E_n = 0.6 - 1.4 MeV, large scatter, preliminary data, abandoned.

9. S. Mubarakmand *et al.*, Nucl. Instr. and Methods, 115 345 (1974).
   Monoenergetic Source, E_n = 1.7 - 14.3 MeV, scattered values, bad shape above $\approx$ 5 MeV, abandoned.

10. R. Batchelor *et al.*, Nucl. Phys., <u>65</u> 236 (1965).
    Monoenergetic Source, E_n = 2 - 7 MeV, few points, used.

11. D. Foster and D. Glasgow, Phys. Rev., <u>C3</u> 604 (1971).
    White Source, E_n = 2.2 - 15.0 MeV, used.

12. C. Thibault, CEA-R-3124 (1967).
    Monoenergetic Source, E_n = 0.1 - 1.5 MeV, single very low value, abandoned.

13. A Bratenahl *et al.*, Phys. Rev., <u>110</u> 927 (1950).
    Monoenergetic Source, E_n = 7 - 14 MeV, used.

14. D. Didier, J. Phys. and Rad., <u>22</u> 651 (1961).
    Monoenergetic Source, E_n = 14.7 MeV, single value used.

15. J. Peterson *et al.*, Phys. Rev., <u>120</u> 521 (1960).
    Monoenergetic Source, E_n = 17.3 MeV, single value, used.

APPENDIX - DATA BASE (Cont'd.)

15. J. Peterson et al., Phys. Rev., 120 521 (1960).
Monoenergetic Source, E_n = 17.3 MeV, single value, used.

16. W. Poenitz et al., Nucl. Sci. and Eng., 78 333 (1981).
Monoenergetic and White Source, E_n = 0.045 - 4.8 MeV, used.

17. C. Hibdon, ANL unpublished (1953).
Monoenergetic Source, E_n = 0.04 - 0.15 MeV, large scatter, abandoned.

18. V. Filippov et al., ACC-68/17 (1968).
E_n = 0.05 - 0.3 MeV, generally very low magnitudes. Probably distorted by self shielding, abandoned.

19. H. Willard et al., Nucl. Phys., (1950).
Monoenergetic Source, E_n = 0.06 - 1.4 MeV, large scatter, abandoned.

20. R. Henkel et al., Phys. Rev., 94 141 (1954).
Monoenergetic Source, E_n = 0.1 - 20.0 MeV, used.

21. M. Divadeenam et al., Thesis Abstracts, 8 28, 3834 (1968).
Monoenergetic Source, E_n = 0.15 - 0.6 MeV, systematically low above 0.2 MeV and probably distorted by self-shielding at lower energies, abandoned.

22. D. Kopsch et al., Proc. Conf. on Nucl. Data for Reactors, 2 39 (1970)
White Source, E_n = 0.5 - 4.35 MeV, abandoned below 0.7 MeV as systematically too large, used above 0.7 MeV.

23. W. Good et al., Phys. Rev., 59 917 (1941).
Monoenergetic Source, E_n = 0.9 MeV, apparently discrepant single value, abandoned.

24. A. Smith, ANL-7110 (1965).
Monoenergetic Source, E_n = 0.8 - 1.5 MeV, scattered points - poor quality, abandoned.

25. J. Leroy et al., J. Phys., 24 826 (1963).
Monoenergetic Source, E_n = 1.6 - 14.5 MeV, scattered values, abandoned.

26. N. Neneson et al., LA-1655 (1954).
Monoenergetic Source, E_n = 2.9 - 14.1 MeV, reasonable looking data but errors are $\gtrsim$10% therefore abandoned.

27. F. Manero, Anales Fisica 66 271 (1970).
Monoenergetic Source, E_n = 3.5 - 5.1 MeV, used.

28. J. Vervier et al., Nucl. Phys., 6 260 (1958).
Monoenergetic Source, E_n = 13.6 - 14.7 MeV, appears to have both magnitude and shape error, abandoned.

APPENDIX - DATA BASE (Contd.)

29. M. Khaletski, DOK-AKAD-NAUK (USSR) 13 305 (1957).
Monoenergetic Source, E_n = 14.8 MeV, discrepant value, abandoned.

30. P. Bowen *et al.*, Nucl. Phys. 22 640 (1960).
Monoenergetic Source, E_n = 15.8 - 20 MeV, large scatter with large uncertainties, however used.

31. R. Day *et al.*, Phys. Rev., 92 358 (1958).
Monoenergetic Source, E_n = 17.9 - 19.0 MeV, used.

32. D. Syme *et al.*, Ann. Nucl. Energy, 1 305 (1974).
White Source, E_n = 0.8 - 8.9 MeV, shape appears good but normalization is low compared to body of information. Used with increased systematic error for all values (3%).

33. D. Olsen *et al.*, ORNL/TM-5915 (1977).
White Source, $E_n \lesssim 0.1$ MeV, used 5 keV average values corrected to infinitely thin samples assuming linear dependence of self-shielding on sample thickness. The correction may be slightly underestimated.

Table 1. Spherical Optical-Model Parameters for the ^{238}U Evaluation[a]

Real Potential[b]		
Strength[c]	V_o = 43.167	MeV
Radius[d]	R_v = 1.290	F
Diffuseness	a_v = 0.310	F
Imaginary Potential[c]		
Strength[f]	W = 2.313	MeV
Radius	R_w = 1.072	F
Diffuseness	a_w = 1.413	F
Spin-orbit Potential[g]		
Strength	V_s = 6.0	MeV

[a] Explicitly relevant only to the parameterization of the present evaluation.

[b] Saxon form.

[c] Assume energy dependence of form $V = V_o - 0.1046\ E(\mathrm{MeV})$.

[d] All radii expressed in the form $R = R_x A^{1/3}$.

[e] Saxon-derivative form.

[f] Energy dependence of the form $W = W_o + 0.0728\ E(\mathrm{MeV})$.

[g] Thomas form, dimensions same as real potential.

TABLE 2. EVALUATED U-238 NEUTRON TOTAL CROSS SECTIONS

EN(MEV)	XSEC(B)	DXSEC(B)	EN(MEV)	XSEC(B)	DXSEC(B)
0.44000E-01	0.13280E+02	0.19800E+00	0.50000E-01	0.12840E+02	0.13100E+00
0.60000E-01	0.12840E+02	0.11500E+00	0.70000E-01	0.12410E+02	0.11300E+00
0.80000E-01	0.12410E+02	0.11100E+00	0.90000E-01	0.12060E+02	0.12200E+00
0.10000E-00	0.12060E+02	0.14400E+00	0.12000E+00	0.11470E+02	0.21100E+00
0.14000E+00	0.11470E+02	0.32100E+00	0.16000E+00	0.10970E+02	0.32100E+00
0.18000E+00	0.10970E+02	0.19700E+00	0.20000E+00	0.10480E+02	0.10700E+00
0.22500E+00	0.10480E+02	0.89000E-01	0.25000E+00	0.10000E+02	0.71000E-01
0.27500E+00	0.10000E+02	0.69000E-01	0.30000E+00	0.95940E+01	0.68000E-01
0.32500E+00	0.95940E+01	0.67000E-01	0.35000E+00	0.92300E+01	0.66000E-01
0.37500E+00	0.92300E+01	0.60000E-01	0.40000E+00	0.89070E+01	0.54000E-01
0.42500E+00	0.89070E+01	0.53000E-01	0.45000E+00	0.86210E+01	0.52000E-01
0.47500E+00	0.86210E+01	0.47000E-01	0.50000E+00	0.82520E+01	0.42000E-01
0.55000E+00	0.82520E+01	0.41000E-01	0.60000E+00	0.78560E+01	0.40000E-01
0.65000E+00	0.78560E+01	0.36000E-01	0.70000E+00	0.75520E+01	0.31000E-01
0.75000E+00	0.75520E+01	0.30500E-01	0.80000E+00	0.73280E+01	0.30000E-01
0.85000E+00	0.73280E+01	0.29500E-01	0.90000E+00	0.71670E+01	0.29000E-01
0.95000E+00	0.71670E+01	0.25000E-01	0.10000E+01	0.70230E+01	0.21000E-01
0.11000E+01	0.70230E+01	0.21000E-01	0.12000E+01	0.69650E+01	0.21000E-01
0.13000E+01	0.69650E+01	0.24500E-01	0.14000E+01	0.70100E+01	0.28000E-01
0.15000E+01	0.70100E+01	0.24500E-01	0.16000E+01	0.71120E+01	0.21000E-01
0.17000E+01	0.71120E+01	0.21000E-01	0.18000E+01	0.72430E+01	0.22000E-01
0.19000E+01	0.72430E+01	0.22000E-01	0.20000E+01	0.74500E+01	0.22000E-01
0.22000E+01	0.74500E+01	0.22000E-01	0.24000E+01	0.76940E+01	0.22000E-01
0.26000E+01	0.76940E+01	0.23000E-01	0.28000E+01	0.78680E+01	0.23000E-01
0.30000E+01	0.78680E+01	0.23000E-01	0.32000E+01	0.79610E+01	0.23000E-01
0.34000E+01	0.79610E+01	0.23000E-01	0.36000E+01	0.79700E+01	0.23000E-01
0.38000E+01	0.79700E+01	0.23000E-01	0.40000E+01	0.79050E+01	0.23000E-01
0.42000E+01	0.79050E+01	0.23000E-01	0.44000E+01	0.77850E+01	0.23000E-01
0.46000E+01	0.77850E+01	0.31000E-01	0.48000E+01	0.76230E+01	0.31000E-01
0.50000E+01	0.76230E+01	0.30000E-01	0.55000E+01	0.72180E+01	0.44000E-01
0.60000E+01	0.72180E+01	0.43000E-01	0.65000E+01	0.68350E+01	0.42000E-01
0.70000E+01	0.68350E+01	0.34000E-01	0.75000E+01	0.64640E+01	0.36500E-01
0.80000E+01	0.64640E+01	0.39000E-01	0.85000E+01	0.61320E+01	0.35000E-01
0.90000E+01	0.61320E+01	0.31000E-01	0.95000E+01	0.59020E+01	0.30000E-01
0.10000E+02	0.59020E+01	0.29000E-01	0.11000E+02	0.57590E+01	0.23000E-01
0.12000E+02	0.57590E+01	0.23000E-01	0.13000E+02	0.58990E+01	0.23000E-01
0.14000E+02	0.58990E+01	0.29000E-01	0.15000E+02	0.60320E+01	0.29000E-01
0.16000E+02	0.60320E+01	0.61000E-01	0.17000E+02	0.62020E+01	0.62000E-01
0.18000E+02	0.62020E+01	0.63000E-01	0.19000E+02	0.63450E+01	0.63000E-01
0.20000E+02	0.63450E+01	0.80000E-01	0.00000E-01	0.00000E-01	0.00000E-01

TABLE 3. CORRELATION MATRIX (55 X 55)

EN(MEV)	MATRIX-ELEMENT VALUES													
0.044	1.00													
0.060	0.18	1.00												
0.080	0.18	0.17	1.00											
0.100	0.12	0.14	0.11	1.00										
0.125	0.03	0.05	0.03	0.08	1.00									
0.150	0.01	0.03	0.01	0.06	0.06	1.00								
0.175	0.01	0.02	0.01	0.04	0.04	0.04	1.00							
0.200	0.02	0.02	0.02	0.04	0.03	0.03	0.04	1.00						
0.250	0.02	0.03	0.02	0.04	0.04	0.03	0.05	0.11	1.00					
0.300	0.02	0.03	0.02	0.04	0.04	0.02	0.04	0.12	0.14	1.00				
0.350	0.02	0.02	0.02	0.04	0.04	0.02	0.04	0.12	0.13	0.17	1.00			
0.400	0.02	0.03	0.02	0.05	0.04	0.02	0.05	0.12	0.14	0.17	0.18	1.00		
0.450	0.02	0.03	0.02	0.05	0.05	0.02	0.05	0.13	0.15	0.18	0.19	0.20	1.00	
0.500	0.03	0.03	0.03	0.06	0.05	0.03	0.06	0.14	0.17	0.20	0.21	0.22	0.24	1.00
0.600	0.03	0.04	0.03	0.06	0.06	0.03	0.06	0.16	0.19	0.23	0.24	0.25	0.27	0.30
	1.00													
0.700	0.03	0.04	0.03	0.06	0.06	0.03	0.06	0.16	0.18	0.22	0.24	0.24	0.27	0.30
	0.34	1.00												
0.800	0.03	0.04	0.03	0.07	0.06	0.03	0.07	0.17	0.19	0.24	0.25	0.26	0.28	0.32
	0.37	0.39	1.00											
0.900	0.03	0.04	0.03	0.06	0.06	0.03	0.06	0.16	0.19	0.23	0.24	0.25	0.27	0.30
	0.35	0.36	0.41	1.00										
1.000	0.06	0.06	0.05	0.10	0.09	0.04	0.10	0.21	0.26	0.30	0.31	0.33	0.35	0.40
	0.46	0.54	0.59	0.59	1.00									
1.200	0.04	0.05	0.04	0.08	0.08	0.03	0.08	0.17	0.21	0.24	0.25	0.26	0.28	0.32
	0.36	0.37	0.42	0.41	0.75	1.00								
1.400	0.04	0.05	0.04	0.08	0.07	0.03	0.07	0.16	0.19	0.22	0.23	0.24	0.26	0.29
	0.34	0.35	0.39	0.39	0.72	0.56	1.00							
1.600	0.05	0.05	0.04	0.08	0.08	0.03	0.08	0.17	0.20	0.24	0.25	0.26	0.28	0.32
	0.37	0.37	0.42	0.41	0.76	0.59	0.57	1.00						
1.800	0.04	0.05	0.04	0.08	0.08	0.03	0.08	0.16	0.19	0.23	0.23	0.25	0.26	0.30
	0.34	0.35	0.40	0.39	0.74	0.57	0.56	0.60	1.00					
2.000	0.05	0.05	0.04	0.08	0.08	0.03	0.08	0.16	0.20	0.24	0.24	0.26	0.28	0.31
	0.36	0.37	0.41	0.41	0.74	0.58	0.56	0.61	0.60	1.00				
2.200	0.04	0.05	0.04	0.08	0.08	0.03	0.08	0.16	0.19	0.22	0.23	0.24	0.26	0.29
	0.34	0.35	0.39	0.39	0.71	0.56	0.54	0.59	0.58	0.60	1.00			
2.400	0.04	0.05	0.04	0.08	0.07	0.03	0.08	0.15	0.19	0.22	0.23	0.24	0.25	0.29
	0.33	0.34	0.39	0.38	0.70	0.55	0.54	0.58	0.57	0.59	0.57	1.00		
2.600	0.05	0.05	0.04	0.08	0.08	0.03	0.08	0.15	0.19	0.22	0.23	0.24	0.26	0.29
	0.34	0.35	0.39	0.39	0.71	0.56	0.54	0.59	0.58	0.59	0.58	0.58	1.00	
2.800	0.05	0.05	0.04	0.08	0.08	0.03	0.08	0.15	0.19	0.22	0.23	0.24	0.26	0.29
	0.34	0.35	0.39	0.39	0.70	0.55	0.54	0.58	0.57	0.59	0.57	0.58	0.59	1.00
3.000	0.05	0.05	0.05	0.09	0.08	0.03	0.08	0.17	0.20	0.24	0.25	0.26	0.28	0.31
	0.36	0.38	0.43	0.42	0.76	0.60	0.59	0.63	0.62	0.64	0.62	0.63	0.65	0.65
	1.00													
3.200	0.05	0.05	0.05	0.09	0.08	0.03	0.08	0.16	0.20	0.23	0.24	0.25	0.27	0.31
	0.35	0.37	0.42	0.41	0.74	0.59	0.57	0.62	0.61	0.62	0.61	0.61	0.63	0.63

TABLE 3. CONTINUED

EN(MEV)	MATRIX-ELEMENT VALUES													
	0.70	1.00												
3.400	0.05	0.05	0.05	0.08	0.08	0.03	0.08	0.16	0.19	0.23	0.23	0.25	0.26	0.30
	0.35	0.36	0.41	0.40	0.73	0.57	0.56	0.60	0.59	0.61	0.59	0.60	0.62	0.62
	0.68	0.69	1.00											
3.600	0.05	0.05	0.05	0.08	0.08	0.03	0.08	0.16	0.19	0.23	0.23	0.25	0.26	0.30
	0.35	0.36	0.41	0.40	0.72	0.57	0.56	0.60	0.59	0.61	0.59	0.60	0.61	0.62
	0.68	0.67	0.67	1.00										
3.800	0.04	0.05	0.04	0.08	0.08	0.03	0.08	0.15	0.18	0.21	0.22	0.23	0.25	0.28
	0.32	0.34	0.38	0.37	0.67	0.53	0.52	0.56	0.55	0.56	0.55	0.55	0.57	0.57
	0.63	0.62	0.61	0.62	1.00									
4.000	0.05	0.05	0.04	0.08	0.08	0.03	0.08	0.15	0.18	0.21	0.22	0.23	0.24	0.28
	0.32	0.34	0.38	0.37	0.67	0.53	0.52	0.55	0.55	0.56	0.55	0.55	0.57	0.57
	0.63	0.62	0.61	0.62	0.59	1.00								
4.200	0.04	0.05	0.04	0.08	0.08	0.03	0.07	0.14	0.18	0.21	0.21	0.23	0.24	0.27
	0.32	0.33	0.37	0.37	0.66	0.52	0.51	0.55	0.54	0.55	0.54	0.54	0.56	0.56
	0.62	0.61	0.60	0.61	0.58	0.58	1.00							
4.400	0.04	0.05	0.04	0.08	0.07	0.03	0.07	0.14	0.18	0.21	0.21	0.23	0.24	0.27
	0.32	0.33	0.37	0.37	0.66	0.52	0.51	0.55	0.54	0.55	0.54	0.54	0.56	0.56
	0.62	0.61	0.60	0.61	0.58	0.58	0.58	1.00						
4.600	0.04	0.04	0.04	0.07	0.07	0.03	0.07	0.13	0.16	0.19	0.20	0.21	0.22	0.25
	0.29	0.30	0.34	0.34	0.60	0.48	0.47	0.50	0.49	0.50	0.49	0.50	0.52	0.52
	0.57	0.56	0.55	0.55	0.54	0.54	0.53	0.53	1.00					
4.800	0.04	0.05	0.04	0.07	0.07	0.03	0.07	0.13	0.16	0.19	0.19	0.21	0.22	0.25
	0.29	0.30	0.34	0.34	0.59	0.47	0.46	0.49	0.49	0.50	0.49	0.49	0.51	0.51
	0.56	0.55	0.54	0.55	0.53	0.53	0.53	0.53	0.50	1.00				
5.000	0.04	0.04	0.04	0.07	0.07	0.02	0.07	0.13	0.16	0.18	0.19	0.20	0.22	0.24
	0.28	0.30	0.33	0.33	0.59	0.47	0.46	0.49	0.48	0.49	0.48	0.49	0.50	0.50
	0.55	0.54	0.54	0.55	0.53	0.53	0.53	0.53	0.50	0.51	1.00			
5.500	0.02	0.03	0.02	0.04	0.04	0.02	0.04	0.08	0.09	0.11	0.12	0.12	0.13	0.15
	0.17	0.18	0.20	0.20	0.[illegible]5	0.28	0.27	0.29	0.29	0.30	0.29	0.29	0.30	0.30
	0.33	0.33	0.32	0.32	0.31	0.31	0.30	0.31	0.29	0.28	0.28	1.00		
6.000	0.02	0.03	0.02	0.04	0.04	0.02	0.04	0.08	0.10	0.11	0.12	0.13	0.13	0.15
	0.17	0.18	0.21	0.20	0.35	0.28	0.27	0.29	0.29	0.30	0.29	0.29	0.31	0.31
	0.34	0.33	0.32	0.32	0.31	0.31	0.31	0.31	0.29	0.29	0.28	0.18	1.00	
6.500	0.02	0.02	0.02	0.04	0.04	0.01	0.04	0.07	0.09	0.11	0.11	0.12	0.13	0.14
	0.16	0.17	0.20	0.19	0.34	0.27	0.26	0.28	0.28	0.28	0.28	0.28	0.29	0.29
	0.32	0.31	0.30	0.30	0.30	0.30	0.29	0.29	0.28	0.27	0.27	0.17	0.17	1.00
7.000	0.02	0.03	0.02	0.04	0.04	0.02	0.04	0.08	0.10	0.11	0.12	0.12	0.13	0.15
	0.17	0.18	0.21	0.20	0.36	0.28	0.28	0.29	0.29	0.30	0.29	0.29	0.30	0.30
	0.33	0.33	0.32	0.32	0.31	0.31	0.31	0.31	0.29	0.28	0.28	0.18	0.18	0.17
	1.00													
7.500	0.02	0.02	0.02	0.04	0.04	0.01	0.04	0.07	0.09	0.11	0.11	0.12	0.12	0.14
	0.16	0.17	0.19	0.34	0.27	0.26	0.28	0.27	0.28	0.27	0.27	0.29	0.29	0.31
	0.31	0.30	0.30	0.29	0.29	0.29	0.29	0.27	0.27	0.26	0.17	0.17	0.16	0.16
	0.17	1.00												
8.000	0.03	0.03	0.02	0.04	0.04	0.02	0.04	0.09	0.10	0.12	0.13	0.13	0.14	0.16

TABLE 3. CONTINUED

EN(MEV)	MATHIX-ELEMENT VALUES												
	0.19	0.20	0.23	0.22	0.39	0.31	0.30	0.32	0.32	0.32	0.31	0.32	0.33	0.33
	0.37	0.36	0.35	0.35	0.35	0.34	0.34	0.34	0.32	0.32	0.31	0.20	0.20	0.19
	0.21	0.20	1.00											
9.000	0.03	0.03	0.02	0.04	0.04	0.02	0.04	0.09	0.10	0.12	0.13	0.13	0.14	0.16
	0.19	0.20	0.23	0.22	0.39	0.31	0.30	0.32	0.31	0.32	0.31	0.31	0.33	0.33
	0.36	0.36	0.35	0.35	0.34	0.34	0.33	0.34	0.32	0.31	0.31	0.19	0.20	0.19
	0.23	0.19	0.23	1.00										
10.000	0.03	0.03	0.03	0.05	0.04	0.02	0.05	0.09	0.11	0.13	0.13	0.14	0.15	0.17
	0.20	0.21	0.24	0.23	0.40	0.32	0.31	0.33	0.33	0.34	0.33	0.33	0.35	0.35
	0.38	0.38	0.37	0.37	0.36	0.36	0.35	0.35	0.34	0.33	0.32	0.20	0.20	0.20
	0.21	0.20	0.24	0.25	1.00									
11.000	0.03	0.04	0.03	0.06	0.05	0.02	0.05	0.11	0.13	0.15	0.16	0.17	0.18	0.20
	0.24	0.25	0.28	0.28	0.49	0.39	0.37	0.40	0.39	0.40	0.39	0.40	0.42	0.42
	0.46	0.45	0.44	0.44	0.44	0.43	0.43	0.43	0.41	0.40	0.39	0.25	0.25	0.24
	0.27	0.24	0.30	0.32	0.34	1.00								
12.000	0.03	0.03	0.03	0.06	0.05	0.02	0.05	0.11	0.13	0.15	0.16	0.17	0.18	0.20
	0.23	0.25	0.28	0.27	0.48	0.38	0.37	0.39	0.39	0.40	0.39	0.39	0.41	0.41
	0.45	0.45	0.43	0.43	0.43	0.43	0.42	0.42	0.40	0.39	0.38	0.24	0.24	0.23
	0.25	0.24	0.29	0.30	0.33	0.42	1.00							
13.000	0.03	0.03	0.03	0.05	0.05	0.02	0.05	0.10	0.12	0.14	0.15	0.16	0.17	0.19
	0.22	0.24	0.27	0.26	0.45	0.36	0.35	0.37	0.36	0.37	0.36	0.37	0.39	0.39
	0.42	0.42	0.41	0.41	0.40	0.40	0.39	0.39	0.38	0.37	0.36	0.23	0.23	0.22
	0.25	0.22	0.27	0.29	0.31	0.39	0.39	1.00						
14.000	0.03	0.03	0.02	0.05	0.04	0.02	0.04	0.09	0.11	0.13	0.13	0.14	0.15	0.17
	0.19	0.21	0.24	0.23	0.39	0.31	0.30	0.32	0.32	0.32	0.32	0.32	0.33	0.33
	0.37	0.36	0.35	0.35	0.35	0.35	0.34	0.34	0.32	0.32	0.31	0.20	0.20	0.19
	0.22	0.19	0.24	0.26	0.27	0.34	0.34	0.33	1.00					
15.000	0.03	0.03	0.03	0.06	0.05	0.02	0.05	0.10	0.13	0.15	0.16	0.16	0.17	0.20
	0.23	0.25	0.28	0.27	0.47	0.37	0.36	0.39	0.38	0.39	0.38	0.39	0.41	0.41
	0.45	0.45	0.43	0.43	0.43	0.42	0.42	0.41	0.39	0.39	0.38	0.23	0.23	0.22
	0.24	0.22	0.26	0.26	0.28	0.34	0.34	0.32	0.28	1.00				
16.000	0.01	0.01	0.01	0.02	0.02	0.01	0.02	0.04	0.05	0.06	0.06	0.06	0.07	0.08
	0.09	0.13	0.12	0.11	0.17	0.13	0.12	0.13	0.13	0.13	0.13	0.13	0.13	0.13
	0.14	0.14	0.13	0.13	0.13	0.13	0.13	0.13	0.12	0.12	0.11	0.07	0.07	0.07
	0.07	0.07	0.08	0.08	0.09	0.11	0.11	0.11	0.10	0.12	1.00			
18.000	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.03	0.04	0.04	0.04	0.05	0.05	0.06
	0.07	0.09	0.09	0.08	0.12	0.09	0.09	0.10	0.09	0.10	0.09	0.09	0.10	0.09
	0.10	0.10	0.10	0.10	0.09	0.09	0.09	0.09	0.09	0.09	0.08	0.05	0.05	0.05
	0.05	0.05	0.06	0.06	0.06	0.08	0.08	0.08	0.07	0.08	0.14	1.00		
20.000	0.01	0.01	0.01	0.02	0.01	0.01	0.02	0.03	0.04	0.05	0.05	0.05	0.06	0.06
	0.07	0.11	0.10	0.09	0.14	0.10	0.10	0.11	0.10	0.11	0.10	0.10	0.11	0.11
	0.12	0.11	0.11	0.11	0.10	0.10	0.10	0.10	0.10	0.09	0.09	0.06	0.06	0.06
	0.06	0.06	0.06	0.07	0.07	0.09	0.08	0.08	0.07	0.08	0.14	1.00		

TABLE 4. Illustrative Comparisons with ENDF/B-V[9]

Energy (MeV)	$\left[\frac{\text{ENDF-PE}}{\text{PE}}\right]$ in %[a]
0.044	1.5
0.1	1.2
0.2	1.0
0.4	0.6
0.8	0.4
1.0	0.3
1.5	0.3
2.0	0.3
2.5	0.3
3.0	0.3
4.0	0.3
6.0	0.6
8.0	0.6
10.0	0.5
15.0	0.5
20.0	1.3

[a]PE ≡ present evaluation.