Carlson

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U. S. DEPARTMENT OF COMMERCE

National Institute of Standards and Technology

Formerly the National Bureau of Standards

Annual Report to

U. S. DEPARTMENT OF ENERGY

Division of Nuclear Physics

on

Neutron Cross Section Standards and Instrumentation

July 1, 1993

DISCLAIMER

U. S. DEPARTMENT OF COMMERCE

National Institute of Standards and Technology

Formerly the National Bureau of Standards

Ionizing Radiation Division

Continuation of the Interagency Agreement
with the

U. S. DEPARTMENT OF ENERGY

Neutron Cross Section Standards and Instrumentation

Interagency Agreement No. DE-AI05-91ER40610

Jan 1, 1994 to Dec 31, 1994 with extensions for CY-95 and CY-96

Amount Requested: $485,000 (12 Months Jan 1, 1994 to Dec 31, 1994)
$1,485,000 (36 Months Jan 1, 1994 to Dec 31, 1996)

Approved:

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Department of Energy
Office of Energy Research (OER)
Face Page

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1910-1400
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TITLE OF PROPOSED RESEARCH:
Neutron Cross Section Standards and Instrumentation

PLEASE TYPE THE FOLLOWING INFORMATION:

1. CATALOG OF FEDERAL DOMESTIC ASSISTANCE NUMBER: 81.049

2. CONGRESSIONAL DISTRICT:
Applicant Organ's Dist.: Maryland Eighth
Project Site's Dist.: Maryland Eighth

3. I.R.S. ENTITY IDENTIFICATION OR SOCIAL SECURITY NUMBER:
U. S. Government

4. AREA OF RESEARCH OR ANNOUNCEMENT TITLE/NUMBER
Nuclear Sciences (Nuclear Data)

5. HAS THIS RESEARCH PROPOSAL BEEN SUBMITTED TO ANY OTHER FEDERAL AGENCY? ☐ yes X no
PLEASE LIST

6. DOE/OER PROGRAM STAFF CONTACT (if known)
Dr. Richard A. Meyer

7. TYPE OF APPLICATION: ☐ New X Continuation ☐ Supplement ☐ Renewal ☐ Revision

8. ORGANIZATION TYPE: Local Government ☐ State Government ☐ Non-Profit ☐ Hospital ☐ Indian Tribal Government Individual ☐ Other ☐ Institution of Higher Education For-Profit ☐ (Small Business ☐ Disadvantaged Business ☐ 8(a) ☐ Women-owned ☐)

9. CURRENT DOE AWARD NUMBER (IF APPLICABLE)
DE-AI05-91ER40610

10. WILL THIS RESEARCH INVOLVE:
10A. Human Subjects no X If yes,
Exemption No. (or)
IRB Approval Date
Assurance of Compliance No.
10B. Vertebrate Animals no X If yes,
IACUC Approval Date
Animal Welfare Assurance No.

11. AMOUNT REQUESTED FROM DOE FOR ENTIRE PROJECT PERIOD $ $1,485,000

12. DURATION OF ENTIRE PROJECT PERIOD
01/01/94 to 12/31/96
Mo/day/yr. Mo/day/yr.

13. REQUESTED AWARD START DATE
01/01/94 (Mo/day/yr.)

14. IS APPLICANT DELINQUENT ON ANY FEDERAL DEBT?
Yes (If "Yes," attach an explanation) X No

15. PRINCIPAL INVESTIGATOR/PROGRAM DIRECTOR NAME, TITLE, ADDRESS AND PHONE NUMBER

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SIGNATURE OF PRINCIPAL INVESTIGATOR/PROGRAM DIRECTOR R S Caswell 6/29/93
Date

PI/PD ASSURANCE I agree to accept responsibility for the scientific conduct of the project and to provide the required progress reports if an award is made as a result of this submission. Willful provision of false information is a criminal offense (U.S. Code, Title 18, Section 1001).

SIGNATURE OF ORGANIZATION'S CERTIFYING REPRESENTATIVE
Date

CERTIFICATION & ACCEPTANCE: I certify that the statements herein are true and complete to the best of my knowledge, and accept the obligation to comply with DOE terms and conditions if an award is made as the result of this submission. A willfully false certification is a criminal offense. (U.S. Code, Title 18, Section 1001).

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U. S. DEPARTMENT OF COMMERCE
National Institute of Standards and Technology

Continuation of Interagency Agreement

with the

U. S. DEPARTMENT OF ENERGY

Neutron Cross Section Standards and Instrumentation

NIST project numbers used in Calendar Year 1993 were:

8463421	Neutron Measurements
8463191	Neutron Measurements

I. OBJECTIVE

The objective of this interagency program is to provide accurate neutron interaction measurements for the U.S. Department of Energy nuclear programs which include waste disposal, fusion, safeguards, defense, fission, and personnel protection. These measurements are also useful to other energy programs which indirectly use the unique properties of the neutron for diagnostic and analytical purposes. The work includes the measurement of reference cross sections and related neutron data employing unique facilities and capabilities at NIST and other laboratories as required; leadership and participation in international intercomparisons and collaborations; the preservation of standard reference deposits and the development of improved neutron detectors and measurement methods. A related and essential element of the program is critical evaluation of neutron interaction data including international coordinations. Data testing of critical data for important applications is included. The program is jointly supported by the Department of Energy and the National Institute of Standards and Technology.

II. INTRODUCTION

This report from the National Institute of Standards and Technology contains a summary of the accomplishments of the Neutron Cross Section Standards and Instrumentation Project during the third year of this three-year interagency agreement. The proposed program and required budget for the following three years are also presented. The program continues the shifts in priority instituted in order to broaden the program base.

Recognizing the decreasing funding levels for nuclear data, we consider it appropriate to continue to reassess our measurement and evaluation activities. We respond to the 1992 report of the Nuclear Science Advisory Committee entitled "Nuclear Data Needs of the 1990's" which states that a core program element must include priority measurements and standards. "These include those measurements that are essential tie points for calculations and modeling, those that demand higher accuracy than available from calculations/models, and those critical data that are benchmarks/standards for large classes of important measurements made by users." Also the nation's requirements for nuclear data have expanded beyond the traditional areas of nuclear power and national security to include nuclear waste processing, space applications, nuclear medicine, fusion energy and industrial applications. At the same time it is important to maintain the required data base and measurement capability for a certain resurgence of the fission and fusion power nuclear data demands.

We have reevaluated the proposed measurement program to respond to the possible closing of the two remaining high intensity neutron facilities in the U.S., the WNR/LANSCE sources at the proton accelerator LAMPF at Los Alamos National Laboratory (LANL) and the Oak Ridge Electron Linear Accelerator (ORELA) at the Oak Ridge National Laboratory (ORNL). The proposed measurements, at a reduced counting rate, could be carried out at other facilities. We have established contacts with the experienced neutron measurement staffs at the Tandem Accelerators at Ohio University and the Triangle Universities Nuclear Laboratory. Measurements could also be transferred to two European facilities where we have a long standing relationship. This includes the pulsed electron accelerator at the Institute for Reference Materials and Measurements (IRMM), formerly known as the Central Bureau for Nuclear Measurements, at Geel, Belgium. Also the cyclotron at our sister laboratory, The Physikalisch Technische Bundesanstag (PTB), located in Braunschweig, Germany is available. The scientists at both of these institutions have an established reputation for quality neutron measurements and have shown an interest in our program.

The main measurement program for the current calendar year is the completion of data acquisition for crucial experiments in progress at the Oak Ridge and Los Alamos facilities. A major emphasis has been focused on the ^{10}B standard cross sections where serious discrepancies in the nuclear data base remain. In particular, there are important problems with the interpretation of the helium gas production associated with diagnostic measurements of interest in nuclear technology. The increased use of this isotope for medical treatment is also of significance. Crucial, careful measurements of neutron reaction cross sections for ^{10}B are in progress in collaboration with scientists at ORNL. The earlier measurements at higher neutron energy will be published in August of this year. Crucial experiments are in progress at the Los Alamos facility on the important fission cross sections for $^{237}Np(n,f)$, $^{239}Pu(n,f)$, and $^{233}U(n,f)$ below 1 MeV neutron energy. In addition, collaborative measurements of charged-particle production in basic biological elements (N,O) for medical applications are underway. Planning for an important measurement of the fundamental standard cross section, the angular distribution of neutron scattering from hydrogen, is in progress with collaboration with scientists from Ohio University, Duke University and LANL. NIST will assume the lead role in these measurements. Further measurements are planned or in progress in collaborations which include fission fragment energy and angular distributions, and neutron energy spectra and angular distributions from neutron induced fission. Data evaluation will shift to include a unified international effort to motivate new measurements and evaluations.

In response to the requests of the measurement community, NIST has begun the formation of a national repository for fissionable isotope mass standards. This action will preserve for future measurements the valuable and irreplaceable critical samples whose masses and composition have been carefully determined and documented over the past 30 years of the nuclear program.

III. TECHNICAL PROGRAM

A. RECENT TECHNICAL PROGRESS

1. *The ^{10}B Total Neutron Cross Section Measurement* (O.A. Wasson and A.D. Carlson, NIST; J.A. Harvey and N.W. Hill, ORNL)

One of the most important cross section measurements requested by the international Nuclear Energy Agency Nuclear Sciences Committee (NEANSC) is the neutron total cross section of ^{10}B in the neutron energy region less than 20 MeV. The evaluator of ^{10}B for the ENDF/B-VI evaluation also stated that the data for the ^{10}B cross sections was in the worst shape of all the standards. This cross section is also important for use in the R-Matrix analysis of the mass 11 system to define the ^{10}B(n,α) standard reaction.

Since a major source of error in previous measurements was the composition of the boron samples, we have carefully prepared the samples for the present measurements. Preliminary neutron experiments and chemical and isotopic analyses using B_4C samples enriched to 91% ^{10}B indicated that these samples would not be adequate for the transmission measurement. Thus, both ^{10}B and ^{11}B powder samples were purchased and analyzed at NIST. Two boron powder samples with an isotopic enrichment of 99.82% ^{10}B and 0.18% ^{11}B were contained in 3.2 cm diameter cylindrical holders with 0.05 cm thick tantalum end caps. The amount of ^{10}B in each sample (approximately 0.235 atoms per barn) was determined from the dimensions of the precisely machined holder, the chemical analysis of the boron, and the weight of the material. The uncertainty in the areal density is less than 0.1%. Similar samples of ^{11}B, enriched to 98.5% ^{11}B, were also prepared.

We have completed total cross section measurements of ^{10}B and ^{11}B at the ORELA facility. The neutron energy region extended from 0.2 to 20 MeV using the 200 m flight path and a 10 ns electron beam width. The neutron detector consisted of a 2.5 cm thick NE110 plastic scintillator viewed by two photomultiplier tubes at opposite sides. Coincidences were used to reduce background. Four different summed amplitude intervals were used to measure the sample transmission. Since the detector viewed only the tantalum part of the ORELA target using 2.3 cm diameter collimation, lower energy neutrons were not available in this measurement. Neutron total cross sections were measured with two sample thicknesses and found to agree within 0.2%. The ratio of the cross section measured with the two sample thicknesses is shown in Fig. 1. Analysis of additional measurements will be performed to improve the statistical precision of the final results.

The total cross section is shown by the histogram in Fig. 2 along with the ENDF/B-VI evaluation. The approximately 4% difference from the ENDF/B-VI evaluation in the neutron energy region below 2 MeV is demonstrated in the ratio of the cross sections shown in Fig. 3.

Since the total cross sections of H and C are well known, samples of CH_2 were also measured as a test of the accuracy of the measurement and analysis techniques. The measured H and C cross sections were in excellent agreement with the ENDF/B-VI evaluation as is shown in Fig. 4. The histogram shows the measured CH_2 cross section while the smooth curve shows the ENDF/B-VI evaluation for the same material. The ratio of the two curves is shown in Fig. 5. This agreement (within 0.2%) between the measured and the standard values demonstrates the accuracy of the measurement technique and data analysis, including dead time correction and background subtraction.

The ^{11}B total cross section measurement has also been completed, but the analysis is still in progress. Additional measurements are planned to extend the ^{10}B measurements to lower neutron energies with the neutron detector viewing the moderated (lower neutron energy spectrum) portion of the ORELA target.

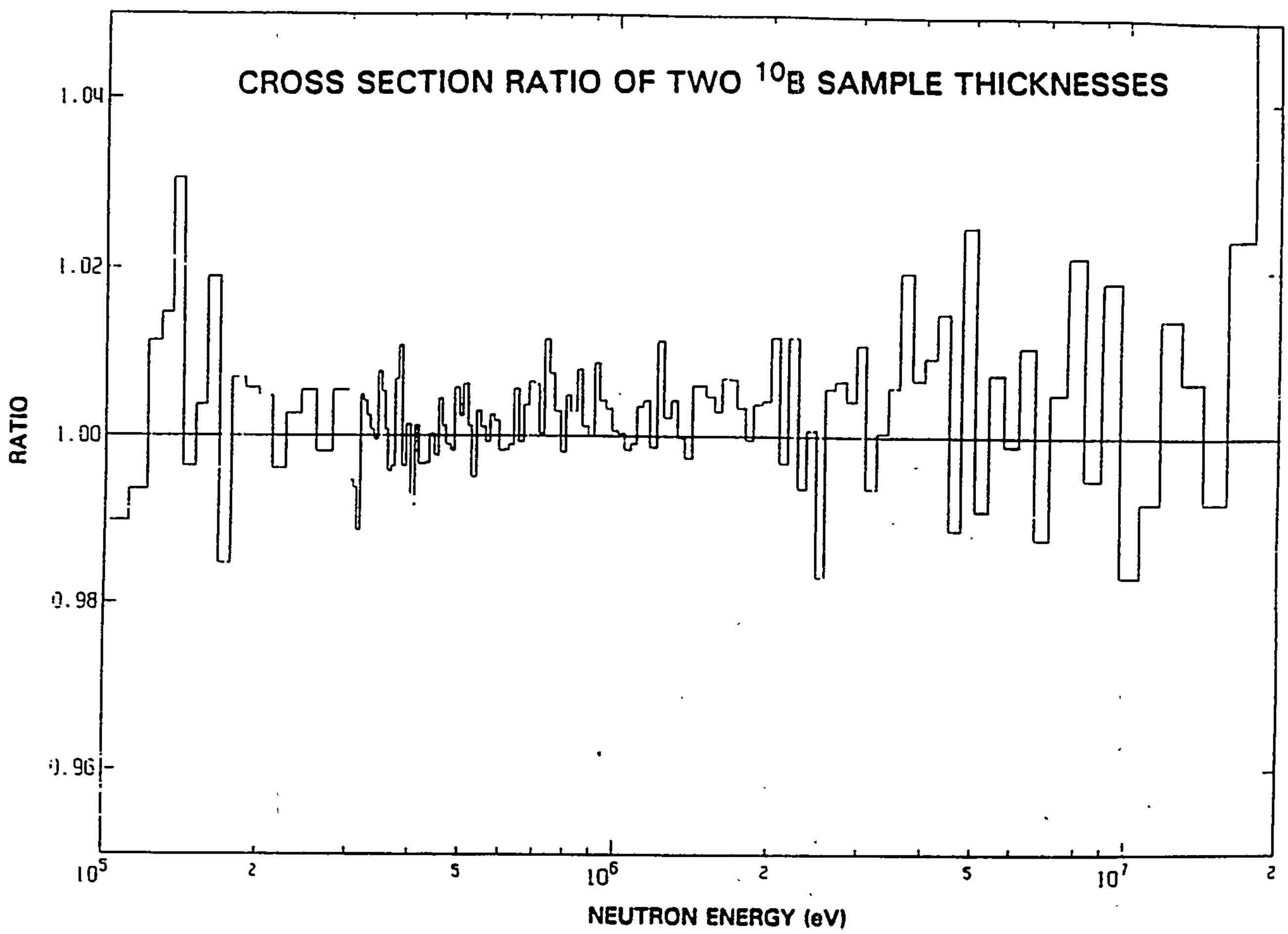

Fig. 1. Ratio of the ^{10}B total cross sections measured with 0.235 and 0.470 atoms per barn thick samples.

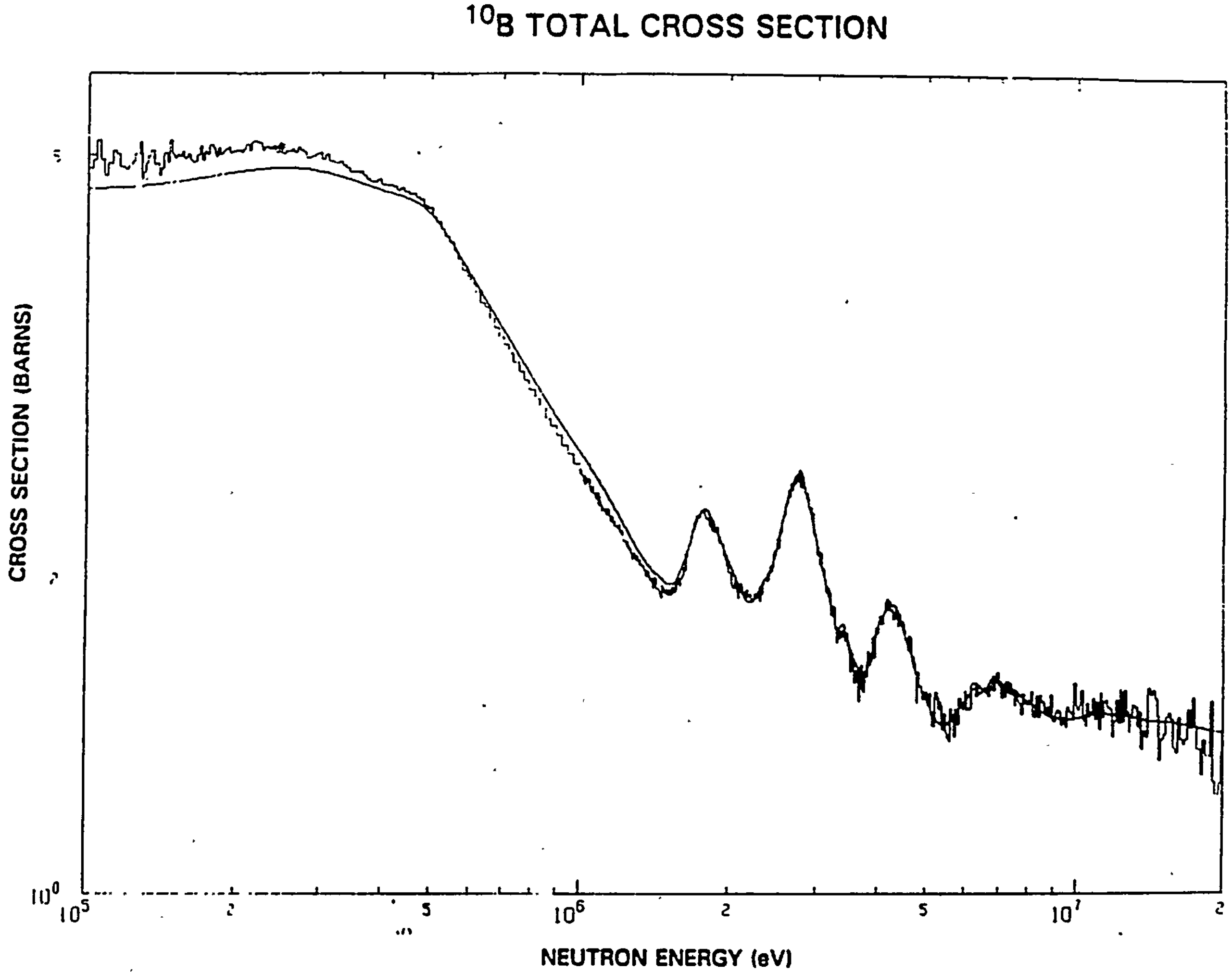

Fig. 2. The ^{10}B total cross section. The histogram shows the measurement while the smooth curve indicates the ENDF/B-VI evaluation.

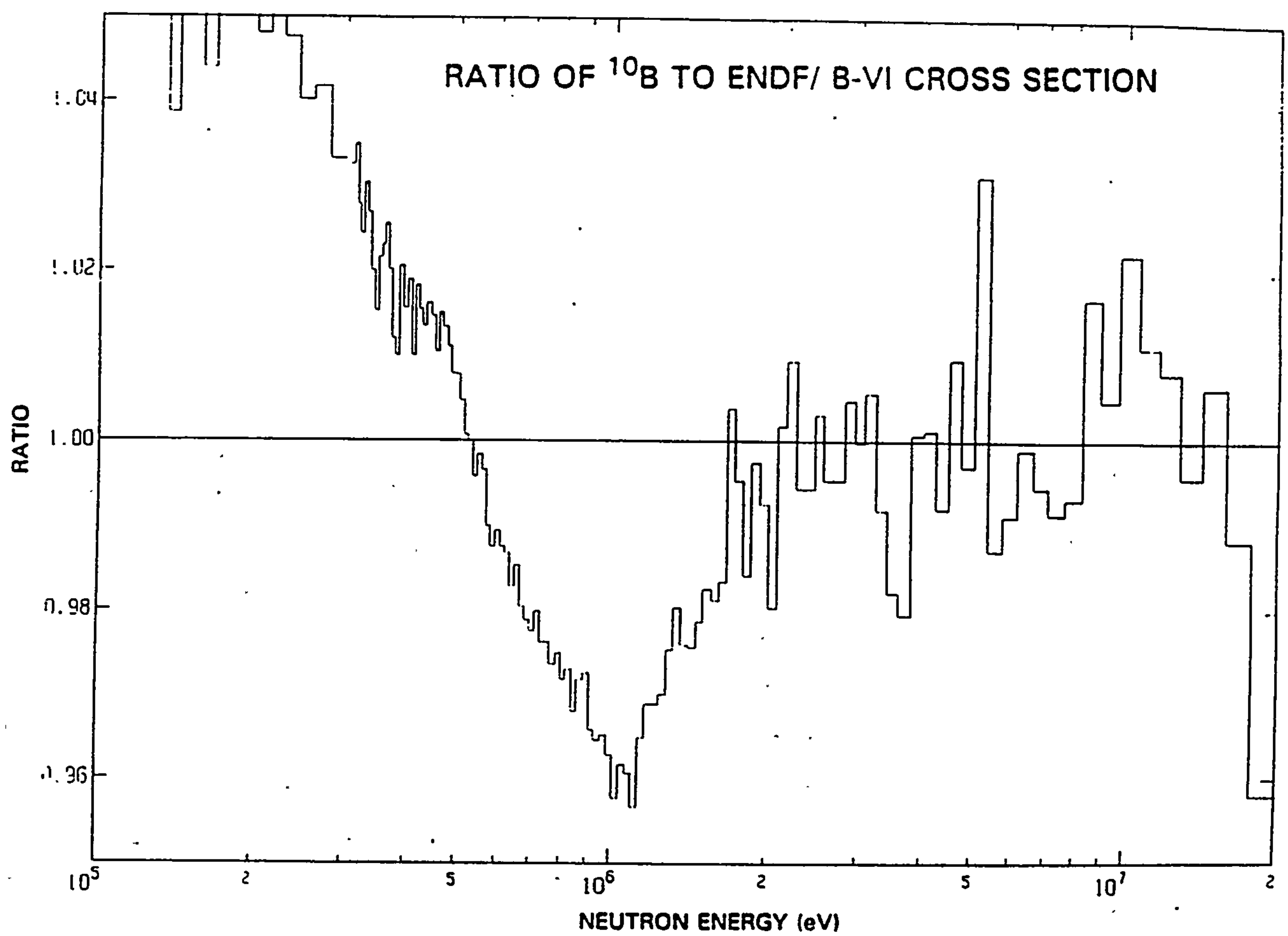

Fig. 3. Ratio of the measured ^{10}B total cross section to the ENDF/B-VI evaluation.

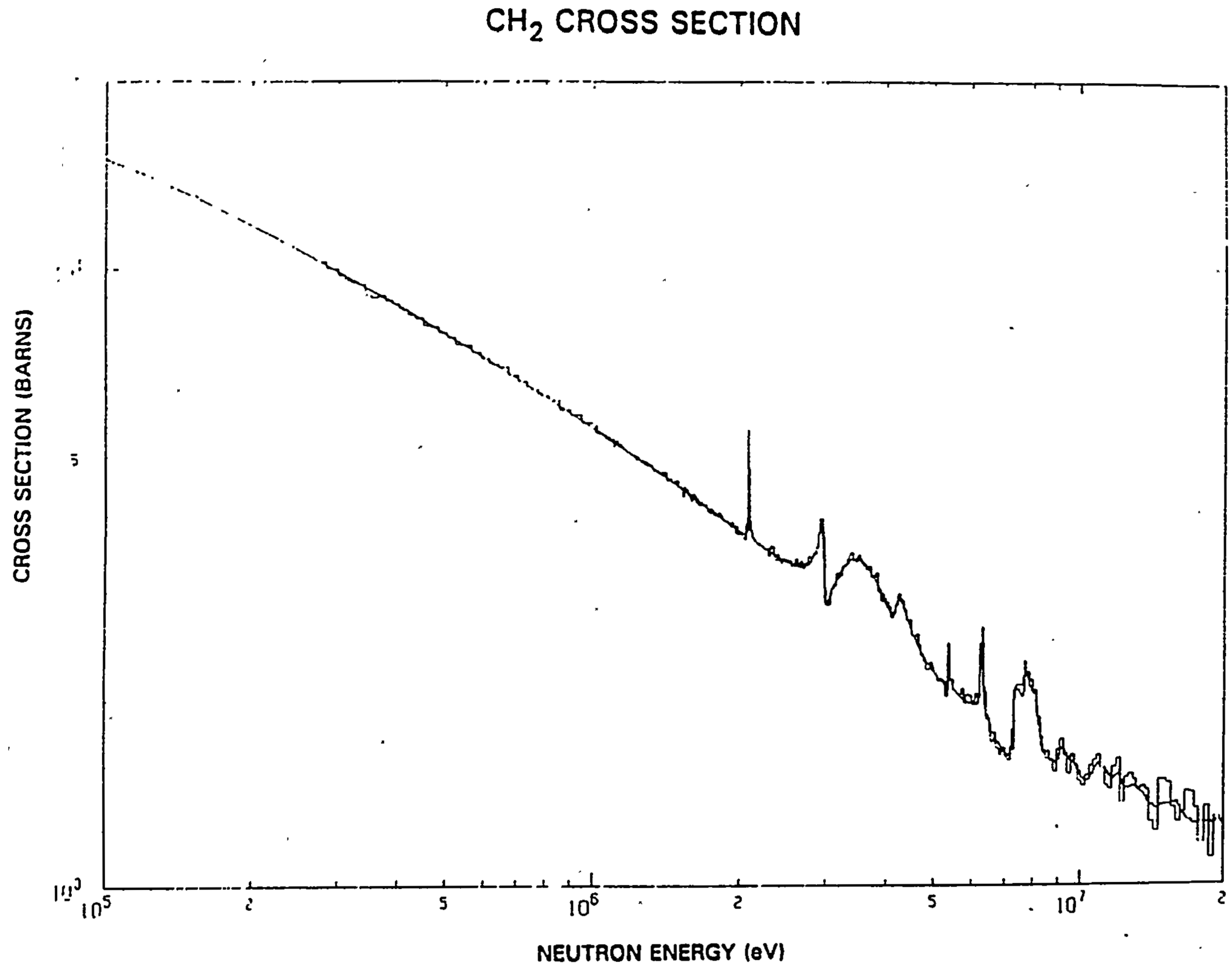

Fig. 4. The CH_2 total cross section. The histogram shows the measured values while the smooth curve indicates the ENDF/B-VI evaluation.

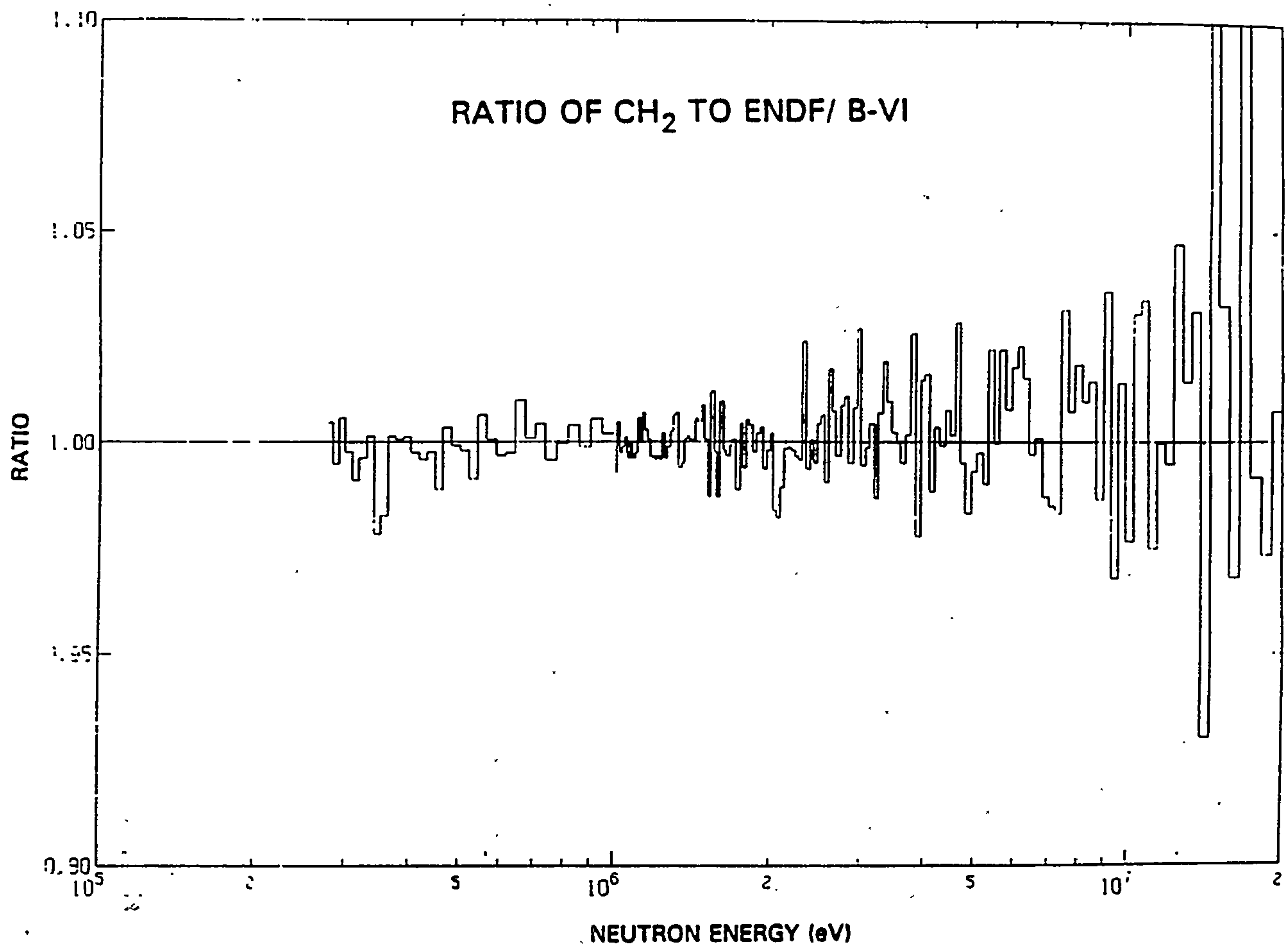

Fig. 5. The ratio of the measured CH_2 total cross section to the ENDF/B-VI evaluation.

2. *Measurement of the $^{10}B(n,\alpha_1\gamma)$ Cross Section from 10 keV to 1.0 MeV Neutron Energy* (R.A. Schrack and O.A. Wasson, NIST; D.C. Larson, J.K. Dickens, and J.H. Todd, ORNL)

Because of the large uncertainties in the determination of the ^{10}B(n,$\alpha_1\gamma$) standard cross section, an NIST-ORNL collaborative effort was undertaken to improve the accuracy of this cross section. The NIST "Black detector," a well characterized neutron fluence monitor was utilized together with a large intrinsic germanium detector to measure the reaction. Measurements from 200 keV to 4 MeV made with these detectors showed that the currently accepted values in ENDF/B-VI are probably in error by 40% above 1.5 MeV.

The results for the region from 200 keV to 4.0 MeV have been accepted for publication by Nuclear Science and Engineering and are scheduled for August publication. A copy of the page proofs[1] is included in the appendix. Although the agreement of this work with ENDF/B-VI is very good from 200 keV to 1.5 MeV, it can confirm only the shape and not the normalization of the cross section in that region. Neither other experimental results for this reaction nor the results of ^{10}B(n,α) and ^{10}B(n,α_0) cross section measurements are well enough known to unambiguously determine the normalization. Lower energy measurements are now underway employing a hydrogen gas proportional counter for the fluence measurement. This work will allow the shape measurements to be made to 10 keV where the absolute value of the cross section is well known.

Preliminary results of the lower energy experiment shown in the last report were discrepant with ENDF/B-VI and previous work done at NBS in 1978. In the past year extensive efforts have been made to improve the measurement and find possible sources of the discrepancy. Possible errors in the analysis were investigated but no problems were found.

Two changes were made in the experiment to allow the measurements to be made at the lower energies. The neutron source was changed from a block of beryllium to a water moderated tantalum assembly with an associated shadow bar for gamma flash reduction. The neutron flux detector was changed from a plastic scintillator, the "Black detector," to a hydrogen proportional counter.

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[1]R.A. Schrack, O.A. Wasson, D.C Larson, J.K. Dickens, and J.H. Todd, "The ^{10}B(n,$\alpha_1\gamma$)^{7}Li Cross Section Between 0.2 and 4.0 MeV," to be published in Nucl. Sci. Eng. (1993).

The 1 μs response time of the hydrogen proportional counter used to measure the neutron fluence is much slower than the 5 ns response time of the germanium detector system used to measure the 478 keV photon. This difference in response time causes non-statistical fluctuations in the calculated cross section because the neutron beam energy distribution has fine structure induced by the filters used to reduce the gamma-flash. Because the energy structure in the neutron beam is so much finer than the response time of the hydrogen proportional counter, it is not possible to remove the beam structure effects by smearing the data obtained by the shorter response time germanium detector. Different filter systems were tried to reduce the non-statistical structure. ^{238}U filters were replaced by lead with no improvement. A mixture of filters of thorium and uranium has provided a slight reduction in the non-statistical fluctuations.

Careful checks of backgrounds, dead-time corrections, and possible interaction of data collection electronics have been made and no problems found.

The collimators were changed to allow the hydrogen proportional counter to have the same sensitivity to the neutrons coming from the outer edges of the neutron source as the reaction detector. No change in the resultant cross section was observed as a result of the collimator change.

The shadow bar used to shield the detectors from the strong gamma flash coming from the tantalum bremsstrahlung and photoneutron source has been removed to allow both detectors to have the same sensitivity to neutrons emitted from regions of the moderator immediately adjacent to the tantalum. When the shadow bar is in place the reaction detector system is prevented from seeing neutrons emitted immediately adjacent to the tantalum which the neutron fluence monitor can see. Monte Carlo calculations show that the energy spectrum of neutrons emitted from the moderator changes rapidly as the point of emission moves away from the tantalum. Figure 6 shows the ratio of the measured $^{10}B(n,\alpha_1\gamma)$ cross section to ENDF/B-VI with the shadow bar in place. The effect observed in Fig. 6 is obtained if the neutron energy spectrum seen by the reaction detector is deficient in higher energy neutrons compared to the energy spectrum seen by the fluence monitor. The shape and magnitude of the neutron energy spectrum change inferred from the data of Fig. 6 agree well with the Monte Carlo calculations for a water moderator and the same relative geometry.

Data is now being accumulated with this new experimental arrangement. In addition detector systems are being prepared to measure the $^{10}B(n,\alpha_1\gamma)$ reaction using the $^{6}Li(n,\alpha)$ as . neutron fluence monitor to verify the results obtained with the hydrogen proportional counter.

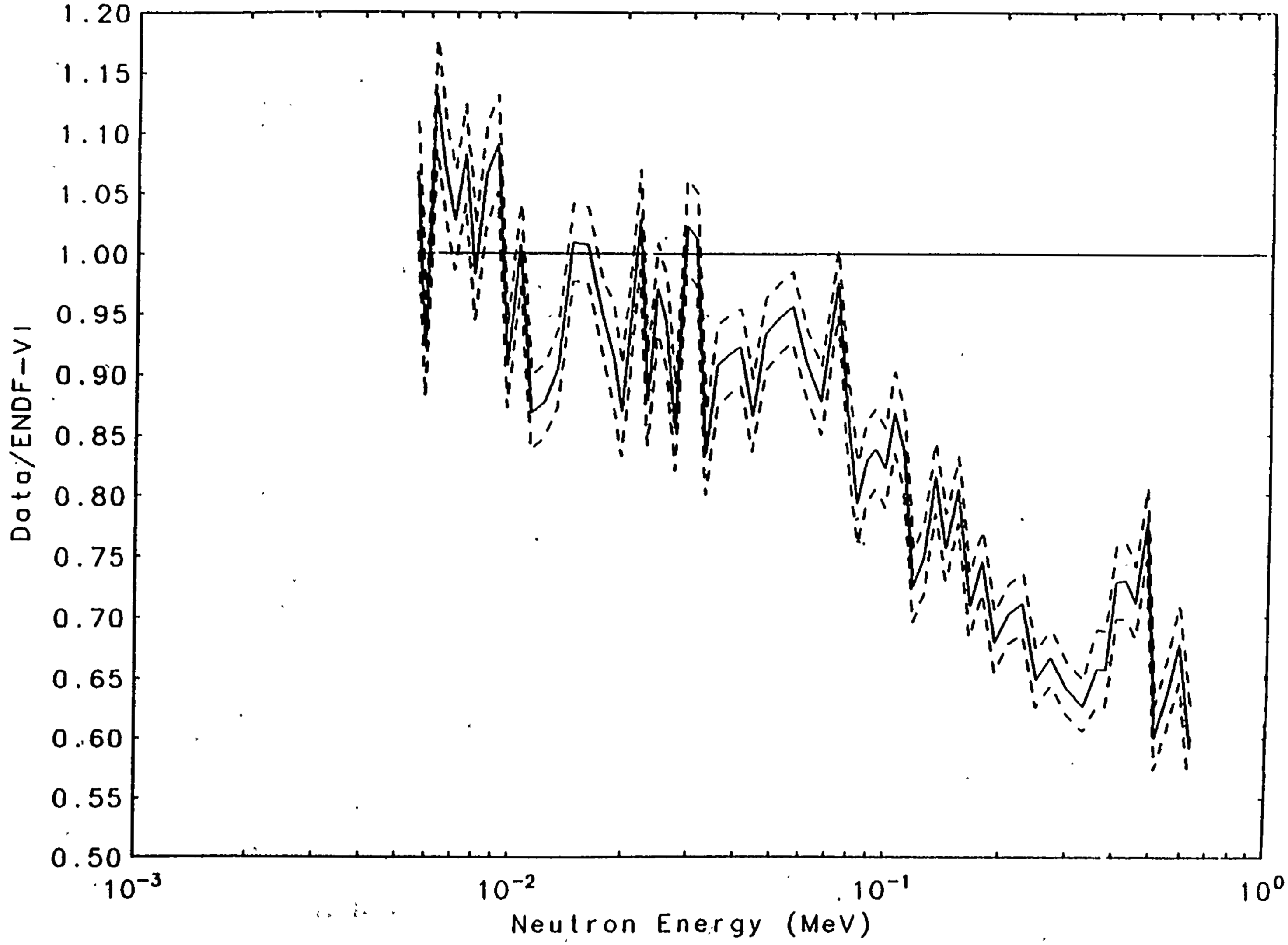

Fig. 6. Ratio of this $^{10}B(n,\alpha_1\gamma)$ measurement to ENDF/B-VI. The dashed lines indicate 1 σ statistical uncertainty. The lack of agreement of this measurement with ENDF/B-VI is attributed to a difference in the neutron beam energy distribution as seen by the reaction detector and the neutron fluence monitor. The observed ratio is in good agreement with Monte Carlo calculations for the same relative geometry.

3. *Fission Cross Section Measurements Below 1 MeV* (A.D. Carlson, NIST; W.E. Parker, P.W. Lisowski, G.L. Morgan and S.J. Balestrini, LANL; N.W. Hill, ORNL)

Measurements have been completed of a number of important fission cross sections at the Los Alamos Neutron Scattering Center (LANSCE). These time-of-flight data extend from about 1 MeV down to the resonance region and were obtained with the 60 m flight paths at the flight path 2 and flight path 5 end stations. The measurements are relative to the ^{235}U(n,f) standard cross section. In addition to the fission deposits, a thin ^{10}B deposit was placed in the ionization chamber so that the ^{10}B(n,α) standard cross section could be used to establish the shape of the neutron fluence and allow very accurately known fission resonance integrals for ^{235}U to be used for normalization. The ^{10}B(n,α) response which has a smooth energy dependence, also allows estimates of the background to be obtained with notch filters. The data were stored in event mode so that they could be analyzed after the conclusion of the data-taking under different biasing conditions.

The data for the ^{236}U(n,f) measurements which were obtained at the flight path 2 end station have been analyzed and are being submitted for publication in Nuclear Physics. This work was initiated to investigate the suitability of this cross section as a fission diagnostic and in an attempt to understand some theoretical inconsistencies with the intermediate structure of ^{236}U fission. It is expected that ^{236}U should exhibit intermediate structure which has spacings of about a keV and widths in the 1-100 meV range resulting from its double-humped fission barrier. It was anticipated that the fission width distribution would be similar to the Porter-Thomas distribution with large variations from resonance to resonance. Previous measurements[2] of the ^{236}U(n,f) cross section did not observe intermediate resonance effects. Appreciable fission widths were observed for the fine structure (class I) resonances which were relatively constant from level to level. The present results for the ^{236}U measurements in the resonance region are very revealing. The new data indicate that the fission width is about a factor of 100 smaller than the previously measured value for the lowest energy resonance at 5.45 eV. There is no evidence of fission in any of the other resonances observed in the earlier work. There is an indication that the previous measurements have been effected by a gamma ray sensitivity of the fission detectors. This would explain the higher fission widths that they observed and the relative constancy from level to level which is characteristic of the capture width. A number of resonances

[2]J.P. Theobald, J. A Wartena, H. Weigmann and F. Poortmans, Nucl. Phys. A181, 639 (1972).

were observed up to about 10 keV for which the fission widths were obtained. These resonances and their fission widths are given in Table I. More information on this work is given in a preprint[3] at the end of this report.

Table I

$E_{\lambda II}$(eV)	Γ_f(meV)
5.45	0.0013 ± 0.0001
1291.7	0.93 ± 0.11
1281.7	7.7 ± 5.0
1268.8	0.82 ± 0.3
2958.9	2.3 ± 0.7
6300.0	10.9 ± 6
10400.0	4.6 ± 2.6

Fission measurements were obtained last fall for ^{237}Np, ^{239}Pu and ^{233}U, at the flight path 5 facility. The ^{237}Np cross section measurements were motivated by the need to improve the accuracy of this important materials dosimetry standard. This cross section has been utilized in crucial detectors for investigating pressure vessel degradation problems and providing information on the lifetime of these pressure vessels. The need for additional measurements on ^{239}Pu(n,f) arose since the ENDF/B-VI values for this cross section caused some inconsistencies with the results obtained from well defined critical assemblies. This cross section is also an important dosimetry standard. The ^{233}U work was prompted by the need for improved data for the design of a system for the transmutation of nuclear waste. The move to the flight path 5 facility was required since the flight path 2 end station had become dedicated to a series of parity violation experiments. The beam tube for flight path 5 was put into place and a below ground 60-m end station was constructed last year. New collimators were designed and fabricated for this flight path. In addition to the normal collimators, special collimators with virtually complete freedom for positioning were made which could be put far into the beam pipe with high accuracy of alignment. In the summer, the detectors, electronics and associated equipment were lowered into the 60 m station which is about 12 m below ground and the collimators were aligned. In the fall, the electronics and computer were set up and measurements were begun. The

[3] W.E. Parker, J.E. Lynn, G.L. Morgan, P.W. Lisowski, A.D. Carlson, and N.W. Hill, "Intermediate Structure in the Neutron-Induced Fission Cross Section of ^{236}U," to be submitted to Nucl. Phys. (1993).

measurements were begun with diagnostic tests which had been made at the flight path 2 end station and had to be repeated at the new end station. It was found that the background is low and about the same as that measured at the flight path 2 facility which is above ground. Some dedicated time was obtained on the LANSCE facility with a repetition rate of 10 Hz instead of the normal 20 Hz so that measurements could be made down to thermal energies for a thermal normalization of the $^{239}Pu(n,f)$ and $^{233}U(n,f)$ cross sections. This should provide a result at the 1% level. The measurements made during these data runs should satisfy the requests for these data. The measurements are presently under analysis.

4. *Neutron-Induced Charged-Particle Production Measurements from Oxygen and Nitrogen for Neutron Energies from 1 to 40 MeV* (O.A. Wasson, NIST; R.C. Haight, S.M. Sterbenz and T.M. Lee, LANL; and H. Vonach, University of Vienna)

The neutron-induced charged-particle production cross sections from oxygen and nitrogen are important for the determination of kerma factors for medical therapy and radiation protection. The oxygen reactions are also required for the use of manganese baths in neutron source calibrations. Preliminary measurements of the oxygen reaction in the neutron energy interval from 2 to 40 MeV have been completed using the 90° left 10 m flight path on the target 4 neutron time of flight beam at the LAMPF accelerator facility at LANL. Particle identification and particle energy measurements were done at angles of 30°, 60°, 90°, and 135° in the laboratory system. The neutron fluence was determined from ^{238}U and ^{235}U fission chambers located downstream from the scattering chamber. The initial measurements were performed at the end of the 1991 LAMPF running cycle. The detection system was optimized for the measurement of alpha particles during this test. Analysis of the data revealed that the ZrO_2 sample used in that experiment was inadequate for the precision desired. Measurements with detectors optimized for proton detection and a more carefully studied plastic oxygen sample were made during the 1992 LAMPF beam cycle. The data analysis is in progress. The NIST participation in the analysis was reduced due to the emphasis on the ^{10}B total cross section measurement. During the 1993 measuring cycle, we plan to obtain thinner oxide samples and initiate measurements on nitrogen using melamine samples.

5. *A National Repository for Fissionable Isotope Mass Standards (FIMS).*
(D.M. Gilliam, NIST)

The present NIST collection of FIMS consists of two parts, a set of carefully assayed Reference Deposits and a much larger number of working deposits which are calibrated with respect to the Reference Deposits by fission counting and/or alpha counting. The assay of the Reference Deposits is based primarily on isotope dilution mass spectrometry and non-destructive relative comparisons. Reference deposit collections of comparable or sometimes superior assay have been built up at some other national laboratories, also. However, with parts of several DoE programs now facing termination, some program directors have appealed to us to help prevent the loss of their valuable isotopic mass standards. Two Argonne groups with quite well-assayed and well-documented deposit collections are among those in danger. It will not be surprising to see similar situations arise at other national laboratories in the future. In response to this need, the Director of the NIST Physics Laboratory has approved the expansion of the FIMS Collection to provide a National Repository for valuable isotopic mass standards. This Repository will prevent the loss of valuable isotopic samples which may otherwise have to be replaced at great expense at some future time.

The activities of the National Repository for Fissionable Isotope Mass Standards will include acquiring deposits, lending deposits, maintaining scientific and regulatory records, and making occasional intercomparisons of well-characterized deposits by fission counting and alpha counting to verify the documented isotopic and mass analyses. The pace of these various activities is expected to increase gradually over the next three years. The acquisition of selected Argonne National Laboratory foil collections and intercomparisons with NIST reference deposits are the first anticipated steps.

So far this year, the expected acquisitions from Argonne have not taken place, but we have been reassured that these transfers are still intended.

We have acquired three deposits of very pure ^{233}U on single-crystal silicon substrates as new alpha counting standards. These standards were provided by the target preparation group of the Institute of Reference Materials and Measurements (IRMM, formerly CBNM, Geel) and are being characterized by the collaborative efforts of the Radioactivity Group at IRMM and our group at NIST. The NIST alpha counting system has been upgraded by the development of a new ultra-precision copper aperture, which was fabricated on a diamond turning machine by the NIST Precision-Engineering Division. The improved alpha counting accuracy permits detection of smaller deviations of standard deposit masses. This improvement is important, not only for scientific purposes, but also for purposes of health physics, safety, and accountability.

6. *The Evaluation of The Standards for ENDF/B*. (A.D. Carlson; NIST; W.P. Poenitz, ANL; R.W. Peelle, ORNL; G.M. Hale, LANL)

A special meeting was held last October with all of the principle participants of the ENDF/B-VI standards evaluation process present. Prior to the meeting, draft copies of the documentation were sent to the attendees with the intent that at this meeting final modifications could be made to this report. A detailed outline of all changes and corrections to the documentation was established at that meeting. The implementing of these changes led to the major effort of this year which was the publication of the documentation of the ENDF/B-VI standards, ENDF-351.[4] A copy of this documentation is included in the appendix. The report contains a description of the evaluation process for the standards which involved separate R-matrix analyses for the H(n,n), ^{3}He(n,p) and C(n,n) reactions; and a combination of R-matrix and simultaneous evaluations for the remaining standards. Measurements of the ^{238}U(n,f), ^{238}U(n,γ) and ^{239}Pu(n,f) cross sections and an evaluation of the thermal constants were also included as input data for the evaluation. Also given are plots showing comparisons of the R-matrix, simultaneous evaluation and combination results; comparisons of the smoothed with the combination results; and comparisons of the output with the ENDF/B-V evaluations. Tables are given of the cross sections and uncertainties for the standards. Expanded estimated uncertainties are also given for the standards obtained from the combination of the R-matrix and simultaneous analyses. Complete sets of references showing the quantity determined are given for the data bases for the R-matrix and simultaneous evaluations used in the combination process. A detailed description of the combination process is given. For completeness the entire output listing from the cross section evaluation process, as it was when it first became available, is provided. This document also contains the non-standards cross sections produced by the evaluation process including the thermal constants, and data beyond the standards regions for the standards.

[4]A.D. Carlson, W.P. Poenitz, G.M. Hale, R.W. Peelle, D.C. Dodder, C.Y. Fu, W. Mannhart, "The ENDF/B-VI Neutron Cross Section Measurement Standards," National Institute of Standards and Technology, NISTIR-5177 (1993); also ENDF-351, Brookhaven National Laboratory.

Recently additions to the data base for the standards since the ENDF/B-VI evaluation were reviewed. Comparisons were made between these data and the standards evaluations. In most cases the new data were in agreement with the evaluations. There are a few important measurements which differ from the evaluations which will be added to the data base for the next evaluation of the standards. A paper[5] resulting from this review was presented at the Symposium on Nuclear Data Evaluation Methodology and is included in the appendix.

Concern has been expressed about possible loss of the essential materials used in the evaluation of the standards. Consideration is being given to a way to ensure that the data bases and evaluation codes used for the ENDF/B-VI standards evaluation be available for future standards evaluations. They could possibly be archived at the National Nuclear Data Center at Brookhaven National Laboratory. Most important are the data bases which represent a significant effort and in some cases contain corrections and information about experiments which are not generally known.

Possible standards for the energy region above 20 MeV neutron energy are now being investigated by the standards subcommittee of the Cross Section Evaluation Working Group (CSEWG). The Specialists Meeting on Neutron Cross Section Standards for the Energy Region Above 20 MeV which was held at Uppsala, Sweden was quite timely. This meeting resulted from a growing interest in measurements in the energy region above 20 MeV for both basic and applied research. The existence of accurate standard reference data is of utmost importance to the quality of measured data. The objective of that meeting was to identify candidates for neutron cross section standards in the intermediate energy range, to summarize the status of those data and to give recommendations for further work to improve the existing data bases. A significant portion of the meeting was devoted to experimental and theoretical work on the H(n,n) cross section which is at present the best known of the cross sections above 20 MeV. There was also a session on implementing this cross section. Other candidates which are easier to implement than the H(n,n) reaction, notably uranium and thorium fission cross sections and some which will require much more experimental work were considered. The fission cross sections are the best candidates at this time, other than the H(n,n) reaction, as a result of the comprehensive program of cross section measurements carried out in the NIST-LANL collaboration at the WNR facility at

[5]W.P. Poentiz and A.D. Carlson, "The Data Base of the Standards and Related Cross Sections after ENDF/B-VI," Symp. on Nuclear Data Evaluation Methodology, 12-16 October, 1992, Brookhaven National Laboratory, USA, p. 3-I-4.

LANL. Measurements of the $^{235}U(n,f)$ cross section from this facility differ as much as 5% with the ENDF/B-VI values from about 15-20 MeV.

All of the candidates for standards in the high energy region need additional work done to improve their data bases. This includes the H(n,n) cross section as was pointed out at the meeting. Most of the candidates have been measured relative to the H(n,n) cross section so there is very strong support for work which could lead to improved determinations of that cross section.

There is no established United States standard data file for any of the candidates for a standard in this higher energy region, even the H(n,n) cross section. The NEANDC/INDC has recently accepted the Arndt energy-dependent partial-wave representation as a standard below 400 MeV.

The standards subcommittee of the CSEWG , during this period between major re-evaluations of the ENDF/B files, is devoting more of its time to the encouragement and fostering of new measurement activities in the area of standards.

7. *Progress Report on the NEANSC International Inter-laboratory Collaboration on the* $^{10}B(n,\alpha)$ *Standard Cross Sections* (A.D. Carlson, Chairman)

A large number of cross sections have been measured relative to the $^{10}B(n,\alpha)$ standard cross sections, particularly in the low energy region. There is a need to extend the energy range over which this cross section can be used as a standard so that a smooth and easy transition to higher energy standards such as H(n,n) can be obtained. Then cross-section measurements relative to standards can be made from low energies to very high energies using two standards which have a large region of overlap in their useful ranges. The quality of the $^{10}B(n,\alpha)$ standard is significantly reduced at the higher neutron energies due to inconsistencies in the experimental measurements. There is general interest on the part of the community of cross section measurers, evaluators and users in improving the $^{10}B(n,\alpha)$ standards. These standards have received much attention lately as a result of their relatively poor data base and the problems they caused in the ENDF/B-VI standards evaluation process.

The strong interest in improving the $^{10}B(n,\alpha)$ standard cross sections has led to the formation of an Inter-Laboratory collaboration to provide a mechanism for improving these cross sections. This collaboration has representatives from the measurement, evaluation and user communities. The objective is to have many laboratories collaborate in programs to improve the data base for the cross sections of neutrons with ^{10}B.

Appreciable effort has been focused on the $^{10}B(n,\alpha)$ cross section problem since the formation of this collaboration. Work has been done on the $^{10}B(n,\alpha_1\gamma)$ cross section, the branching ratio, the total cross section and the differential cross section for the $^{10}B(n,\alpha)^7Li$ reaction.

There have been two meetings of the collaboration. Since the second meeting of this collaboration which was held during the Jülich nuclear data meeting, significant activity has occurred. The ORNL branching ratio work of Weston and Todd has been published. This work suggests a problem with the ratios of the ENDF/B-VI cross sections at the 10 to 30% level in the 100 to 600 keV energy region. Schrack *et al.*, in an NIST-ORNL collaboration have continued their investigation of the $^{10}B(n,\alpha_1\gamma)$ cross section. The first of these measurements employed a black detector to obtain the neutron fluence and data was obtained from 0.3 to 4.0 MeV. The shape of these data are consistent with that of ENDF/B-VI to about 1 MeV, but differ by up to 40% for energies above 1 MeV. New data have been obtained using a hydrogen gas proportional counter for the fluence measurement and the energy range covered was from about 5 keV to 1 MeV. These shape data were in disagreement with ENDF/B-VI. Extensive efforts have been made to determine the origin of the disagreement. New experiments are now being undertaken without the shadow bar which is used to reduce the gamma-flash. The presence of the shadow bar causes a different source energy distribution to be seen by the reaction and neutron fluence detectors since the reaction detector is located much closer to the source than the neutron fluence detector. If disagreements with ENDF/B-VI persist, additional measurements using a ^6Li glass detector will be considered.

Two separate measurements of the ^{10}B total cross section are now being made at the Geel GELINA and ORNL ORELA facilities. These measurements can be effectively utilized in helping to define the $^{10}B(n,\alpha)$ cross sections when used in an R-matrix analysis. Such an analysis can use information including the total, scattering and (n,α) cross sections to define the parameters needed to accurately calculate the (n,α) cross sections. Brusegan of Geel is now obtaining total cross section data at the

GELINA facility with a ^6Li glass detector and the thinnest of three recently fabricated $^{10}B_4C$ samples which will be used in the measurement program. Backgrounds are being evaluated with black resonance techniques and by replacing the ^6Li glass detector with a ^7Li glass detector. The measurements will extend from 100 eV to about 2 MeV. Also Harvey and Wasson, in an ORNL-NIST collaboration, recently completed taking total cross section data for ^{10}B and ^{11}B samples at the ORELA facility. The neutron energy region extended from 0.2 to 20 MeV using a 200 m flight path. The neutron detector consisted of a NE-110 plastic scintillator viewed by two photomultiplier tubes on opposite sides. Coincidences were used to reduce background. Since the detector viewed only the tantalum part of the ORELA target, lower energy neutrons were not used in this measurement. Two boron powder samples with an isotopic enrichment of 99.82% ^{10}B were used. The Total cross sections measured with the two different sample thicknesses agreed within 0.2%. Samples of CH_2 were also measured as a test of the accuracy of the measurement and analysis techniques. The cross sections measured were in excellent agreement with the ENDF/B-VI evaluation values for H and C. The ^{10}B total cross section agrees with the ENDF/B-VI evaluation for neutron energies greater than approximately 1.5 MeV, but deviates by more than 4% for the lower energies. The ^{11}B total cross section has also been measured, but the analysis is still in progress. Additional measurements are in progress to extend the measurements to lower neutron energies using the 80 m flight path with the detector viewing the moderated portion of the ORELA target.

Measurements of the differential cross section for the $^{10}B(n,\alpha)^7Li$ reaction have been obtained by Haight *et al.* The data obtained from these experiments will be analyzed soon.

Though no action has been taken, experiments have been suggested to improve the $^{10}B(n,\alpha_0)$ cross section. Also a high priority request for such a measurement has been submitted to the U.S. Compilation of Requests for Nuclear Data.

Another meeting of this collaboration is being considered during the Gatlinburg, Tennessee, Nuclear Data Conference in May 1994.

8. *Measurements of the H(n,n)H Angular Distribution* (A.D. Carlson, O.A. Wasson, NIST; S.M. Grimes, Ohio University; R.C. Haight, LANL; and C.R. Howell, W. Tornow, Duke University)

The 1971 Hopkins and Breit phase shift analysis of the hydrogen scattering cross section was the basis for the ENDF evaluations for versions II through V. It has been used for many years by the cross section community as an important neutron cross section standard. Improvements in the data base for nucleon-nucleon scattering measurements led to a new evaluation by Dodder and Hale of the hydrogen scattering cross section. This evaluation employed a charge independent R-matrix analysis making use of the large data base of n-p and p-p experimental data at energies below 30 MeV. This evaluation was adopted as the hydrogen standard for the new ENDF/B-VI library. The maximum difference between the ENDF/B-VI and ENDF/B-V evaluations is about 2% for neutron energies of about 10 MeV and a center of mass angle of 180°. This is a large difference for a standard which was thought to be very well defined. The center of mass angle of 180° corresponds to proton recoils at 0° in the laboratory system which is a very commonly used configuration for proton recoil telescopes which are often used to implement this standard. Ryves and Kolkowski[6] of the National Physical Laboratory have reported measurements of the hydrogen differential scattering cross section at 14.5 MeV made with rather poor geometry over a very limited range. The data measured were the ratio of the differential cross sections at center of mass angles of 180° and 110°, and the absolute cross section at 180°. The ratio measurement is in better agreement with ENDF/B-V than ENDF/B-VI. The absolute measurement lies about midway between the evaluations. Both the Hopkins-Breit and Dodder-Hale evaluations rely strongly on rather dated measurements and Hale has stated that the differences between the evaluations results not so much from the disagreements among the measurements as the fact that there are so few of them for the differential cross section below 15 MeV. The data which do exist are largely clustered around 14 MeV and most of these measurements have uncertainties which are large. The few measurements which have small reported uncertainties have led to the pronounced backward peaking of the cross sections in ENDF/B-VI.

[6]T.B. Ryves and P. Kolkowski, Ann. Nucl. Energy, 17 , 657 (1990).

A NIST-Ohio U.-TUNL-LANL collaborative experiment is now being designed to improve the database for the hydrogen scattering cross section. The measurements will be made at the Ohio University Tandem Accelerator Laboratory and are expected to begin next year. Investigations of corrections and experimental effects suggest that the measurements can be done at the 1% level. The experiment will employ a scattering chamber for which the differential cross section can be measured at many angles simultaneously in order to remove some problems associated with the very accurate monitoring of the neutron beam intensity which has plagued a number of previous experiments where data were taken an angle at a time. A pulsed beam of deuterons used with the D(d,n) neutron source reaction and a gas target cell will provide a pulsed neutron source. The use of pulsed beam time of flight techniques combined with ΔE-E detectors at each angle should provide a low background. The first measurements will be made at 10 MeV neutron energy where the difference between the ENDF/B-V and ENDF/B-VI evaluations are approximately maximum.

9. *Study of the Mechanism of Nuclear Fission* (C. Zoeller and M. Mutterer, Darmstadt Germany; A. Gavron and W.E. Parker, LANL; G. Petitt, Georgia State University; A.D. Carlson, NIST)

These measurements were initiated in order to provide important information on the various parameters of neutron fission. Data have been obtained on neutron emission following fission. These data are of general interest in the modelling of neutron spectra. Measurements were also been made of the distribution of fission masses and kinetic energies as a function of excitation energy. These data can be applied to corrections for the loss of fission fragments when thick films are used in a fission chamber.

This experiment was recently completed at the WNR facility at LANL using ^{238}U deposits which are thin to the fission fragments. For each event, measurements were made of the directions and kinetic energies of the two fission fragments, as well as the time-of-flight of any coincident neutrons. The fragments were detected in two semiconductor detector arrays each containing 19 PIN diodes which were located on each side of the ^{238}U. The neutrons were detected in encapsulated NE-213 liquid scintillator detectors with gamma-ray discrimination located outside the array of PIN diodes. The data involve incoming neutron energies between 2 and 200 MeV. By considering the correlation between the fission fragment direction, and the energy and angle of the emitted neutrons, the average multiplicity of neutrons preceding fission can be determined. Deviations from statistical model predictions for pre-fission neutrons signal the onset of viscosity effects and the breakdown of the Bohr-Wheeler formalism. The WNR experiment will help in the determination of the energy range over which the Bohr-Wheeler formalism is valid. The project is the PhD thesis project of C. Zoeller. The co-authors assisted in obtaining experimental equipment, setting up of the experiment and data taking but the data analysis is being done by C. Zoeller.

10. *Fission Cross Section Measurements at the LANL-WNR Facility* (A.D. Carlson and O.A. Wasson, NIST; P.W. Lisowski, J.L. Ullmann, A. Gavron and W. Parker, LANL; N.W. Hill, ORNL)

Additional work has been done on the analysis of the fission cross section work in the MeV neutron energy region which was performed at the target 4 neutron facility of the Los Alamos Meson Physics Facility. The most important contribution has been the redetermination of the masses of the fission deposits by alpha counting. The original measurements were of fission cross sections for ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np and ^{239}Pu. The fission reaction rates were measured with a fast parallel multi-plate ionization chamber containing the fission deposits.

The first group of measurements was made with an annular proton telescope for the fluence measurement. These measurements were successfully completed from 3 up to 30 MeV neutron energy. Above this energy, it became difficult to resolve the background from the foreground. The result of this work was presented at the nuclear data conference at Jülich. This contribution[7] is contained in the appendix to this report.

[7]A.D. Carlson, O.A. Wasson, P.W. Lisowski, J.L. Ullmann, and N.W. Hill, "Measurements of the $^{235}U(n,f)$ Cross Section in the 3 to 30 MeV Neutron Energy Region," in Proc. *International Conference on Nuclear Data for Science and Technology*, 13-17 May, 1991, Jülich, Fed. Rep. of Germany, S.M. Qaim, Ed., pp. 518-520.

A second group of measurements was made with the focus on obtaining cross section data above 30 MeV. For these measurements, new detectors were used to measure the neutron fluence. A proton telescope similar to the annular proton telescope positioned at an angle of 15° to the beam axis, in a vacuum container, was used for the lower energy region. A separate higher energy telescope was used but it is operated in air and located downstream from the other fluence detector. This detector is composed of a polyethylene disk and three proton recoil scintillators oriented at an angle of 15° to the beam. The detector is operated in a coincidence mode in order to reduce the background. With the use of these detectors fluence measurements could be made to above 100 MeV. A contribution to the specialists' meeting at Uppsala was given on these measurements and it is included in the appendix to this report.[8]

Of considerable interest are the measurements of the ^{235}U fission cross section standard. These two groups of measurements are in reasonable agreement and agree with the ENDF/B-VI evaluation for this cross section up to about 15 MeV. Above this energy differences as large as 5% with the evaluation are observed.

[8]P.W. Lisowski, A. Gavron, W.E. Parker, J.L. Ullmann, S.J. Balestrini, A.D. Carlson, O.A. Wasson, and N.W. Hill, "Fission Cross Sections in the Intermediate Energy Region," in Proc. *NEANDC Specialists Meeting on Neutron Cross Section Standards for the Energy Region above 20 MeV*, 21-23 May, 1991, Uppsala, Sweden, (NEANDC-305U, 1991) pp. 177-186.

B. SUMMARY OF PLANNED THREE-YEAR PROGRAM

The planned nuclear data activities for the next three years are an orderly continuation of the existing program. This program which recognizes the changing needs for nuclear data includes a selected set of data measurements and evaluations. The emphasis is focused on very accurate measurements of the H(n,n) angular distribution standard. These measurements are badly needed due to inconsistencies in previous measurements which have lead to different results with various evaluations. The work will continue on the ^{10}B cross sections where serious discrepancies in the nuclear data base remain. In particular, important problems with the interpretation of the helium gas production associated with diagnostic measurements of interest in nuclear technology still persist. The enhanced use of this isotope for medical treatment is also of significance.

The measurements made at LANL on the important fission cross sections for $^{237}Np(n,f)$, $^{239}Pu(n,f)$, and $^{233}U(n,f)$ below 1 MeV neutron energy will be analyzed. New measurements on the $^{237}Np(n,f)$ cross section at 14 MeV will be planned and initiated during this period. Collaborative work on charged-particle production in basic biological elements for medical applications which are underway at LANL will be completed. Further measurements of a more long-range nature are planned or in progress in collaborations. We plan to continue to provide a supporting role on work on the fission fragment angular and energy distributions and neutron energy spectra and angular distributions from neutron induced fission in ^{238}U. Data evaluation will include a unified international effort to motivate new measurements, evaluations and inspections of the data bases of some of the important standards. The national repository for fissionable isotope mass standards will extend and verify its valuable and irreplaceable critical samples whose masses have been carefully determined and documented over the past 30 years of the nuclear program.

A condensed table of this three year program is given on the following page while a yearly discussion of the program is presented in the following sections.

Three Year Program for Nuclear Data Measurements

Activity	1994	1995	1996
Measure critical H(n,n) angular distribution.	Initiate cooperative measurements at Ohio U at 10 MeV.	Complete cooperative measurements at Ohio U at 10 MeV and plan extension to higher energies.	Submit important H(n,n) cross section measurement for publication. Complete higher energy measurements.
Measure ^{10}B(n,α) cross sections to extend its energy range of use and resolve He gas production data.	Analyze and submit for publication ^{10}B(n,total) and ^{10}B(n,$\alpha\gamma$) cross section.	Study measurement of inverse ^{7}Li(α,n) reaction to define mass 11 system.	Initiate ^{7}Li(α,n) reaction measurement at suitable positive ion accelerator.
Measure neutron dosimetry standard cross sections.	Complete analysis and submit for publication ^{237}Np, ^{239}Pu, and ^{233}U neutron-induced fission cross sections.	Plan ^{237}Np(n,f) measurements at 14 MeV for normalization.	Complete ^{237}Np(n,f) measurements at 14 MeV and submit for publication.
Measure neutron-induced charged-particle production cross sections for C, N, and O from 5-30 MeV.	Analyze N(n,cp) measurements and prepare for publication.	Analyze O(n,cp) measurements and submit for publication.	
Establish National Repository for fissionable isotope mass standards.	Acquire ANL deposits. Compare with NIST standards.	Compare ^{6}Li and ^{10}B standards with thermal fission standards.	Extend and verify collection. Evaluate long-term stability of deposits.
Provide leadership and promote international cooperation in nuclear data evaluation.	Study procedure for unified international standards effort. Initiate review of H(n,n)H data base.	Study data bases for H(n,n)H and ^{237}Np(n,f) cross sections and ascertain need for new evaluations.	Prepare new evaluation of standard cross sections.

C. PLANS FOR CALENDAR YEAR 1994

The planned activities for this calendar year will place top priority to a measurement of neutron-proton scattering in addition to an orderly continuation of the measurements from the previous year.

Measurements of the H(n,n)H Angular Distribution

Inconsistencies appear to be present between recent measurements of Ryves and Kolkowski[6] and the ENDF/B-VI hydrogen scattering cross section standard. The recent experiments were made with rather poor geometry over a very limited range which make it difficult to ascertain if a discrepancy exists. Also a significant difference exists between the ENDF/B-V and ENDF/B-VI evaluations of this cross section. The ENDF/B-V evaluation performed by Hopkins and Breit was used for many years by the cross section community as an important neutron cross section standard. Recently an analysis of nucleon-nucleon data by Dodder and Hale led to a new evaluation of the hydrogen scattering cross section. This evaluation was adopted as the hydrogen standard for the new ENDF/B-VI library. The maximum difference between the ENDF/B-V and ENDF/B-VI evaluations is about 2% for a broad energy region centered at about 11 MeV and a center of mass angle of 180° (see Fig. 7).

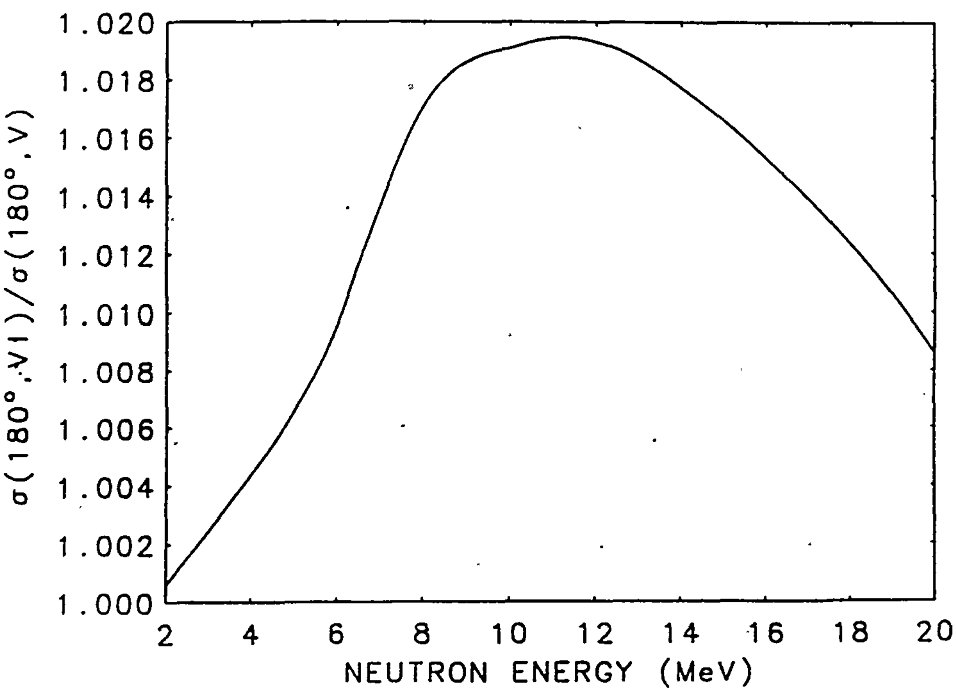

Fig. 7. The ratio of the ENDF/B-VI to ENDF/B-V hydrogen scattering cross sections at 180° in the center of mass system as a function of energy from 2 to 20 MeV.

This is a large difference for a standard which was thought to be very well defined. The center of mass angle of 180° corresponds to proton recoils at 0° in the laboratory system which is a very commonly used configuration for proton recoil telescopes which are often used to implement this standard. Both the Hopkins-Breit and Dodder-Hale evaluations rely strongly on rather dated measurements and Hale has stated that the differences between the evaluations result not so much from the disagreements among the measurements as the fact that there are so few of them for the differential cross section below 15 MeV. The data which do exist are largely clustered around 14 MeV and most of these measurements have uncertainties which are large. The few measurements which have small reported uncertainties have led to the pronounced backward peaking of the cross sections in ENDF/B-VI. The scattering of protons by neutrons is one of the most basic nuclear reactions. Accurate data are necessary for this cross section in order to refine theoretical calculations and phase shift analyses. The spread in calculated results from such analyses is effected by the limited data base. Calculations of the differential cross section at 10 MeV neutron energy using the Bonn potential, the Paris potential and the SM93 phase shift analysis of Arndt indicate differences in shape and absolute values at the several percent level. The spread in values is similar to that between the ENDF/B-V and ENDF/B-VI evaluations as shown in Fig. 8.

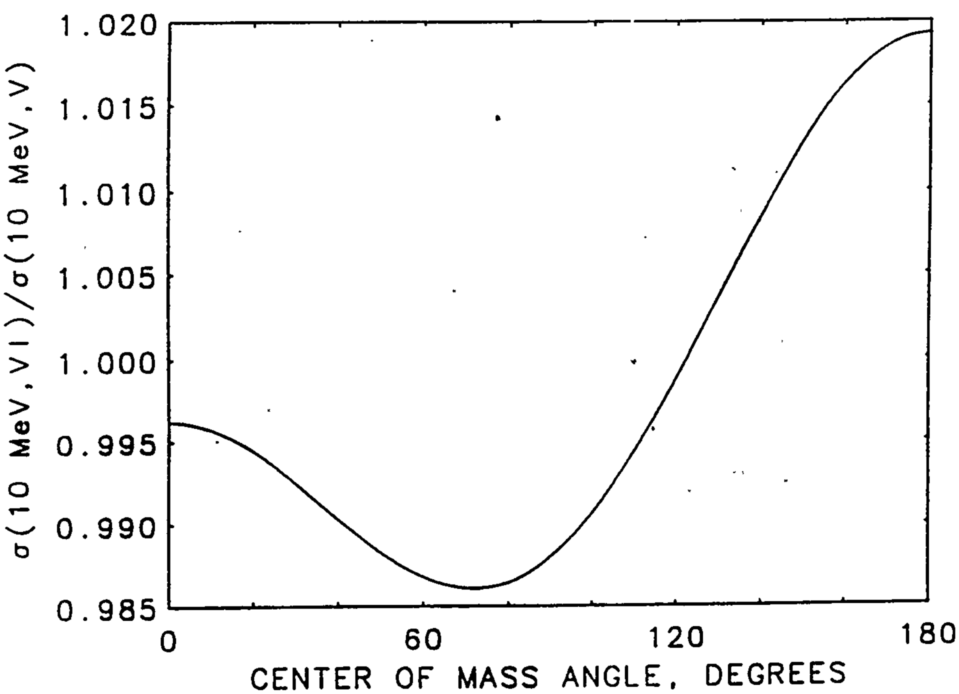

Fig. 8. The ratio of the ENDF/B-VI to ENDF/B-V hydrogen scattering cross sections at 10 MeV neutron energy as a function of center of mass angle.

A NIST-Ohio U.-TUNL-LANL collaborative experiment is now being designed to improve the database for the hydrogen scattering cross section. The measurements will be made at the Ohio University Accelerator Laboratory and are expected to begin next year. Calculations of count rate for the proposed experimental set-up are quite reasonable. Investigations of corrections and experimental effects suggest that the measurements can be done at the 1% level.

The experiment will employ a scattering chamber for which the differential cross section can be measured at 10 angles simultaneously. This will remove problems associated with the very accurate monitoring of the neutron beam intensity which was required with a number of other experiments for which data were taken an angle at a time. Measurements will be made at equivalent angles on both sides of the neutron beam axis (θ and -θ) to provide consistency checks on the results. The ratios of the solid angles for all the detectors will be made simultaneously with a rotating alpha particle source. The experiment does not require absolute measurements of the solid angles or differential cross sections since the shape of this cross section is the quantity in question. Thus the thin CH_2 films used for hydrogen samples in this experiment need only be analyzed for impurities, not the absolute hydrogen content. The very accurately known hydrogen total cross section can be used to normalize the relative differential cross section measurements if a sufficient angular range is obtained. The D(d,n) neutron source reaction using a gas target cell will be used as the neutron source. The deuteron beam will be pulsed. The use of pulsed beam time of flight techniques combined with ΔE-E detectors at each angle should provide a low background. The first measurements will be made at 10 MeV neutron energy where the difference between the ENDF/B-V and ENDF/B-VI evaluations are approximately maximum. Employing this energy will also provide new data for the hydrogen standard which can be applied to recent experiments on the n-D differential cross section at 10 MeV performed in a TUNL-Ohio University collaboration. That measurement is now limited by the uncertainty in the hydrogen scattering cross section. As can be seen in Fig. 8, measurements should extend from about 60° to 180° in the center of mass system (CMS) (60° to 0° for the proton recoil angle in the laboratory system) in order to check for the maximum effect in the ratio of the cross sections which is about 3½%. At 60° in the CMS the proton recoil energy in the laboratory system is 2.5 MeV for 10 MeV incident neutrons. Measurements at this angle will not be a problem but at angles less than 60° in the CMS (greater than 60°

in the lab system) the proton recoil energy in the laboratory system falls rapidly since it depends on the square of the cosine of the proton recoil angle in the laboratory system. Measurements at small CMS angles will depend on the low energy limit of the ΔE-E detector system. However the range from 60° to 180° in the CMS should provide the necessary information in a comparison of the ENDF/B-V and ENDF/B-VI evaluations.

Measurement of the ^{10}B Standard Cross Sections

Our primary measurements are now on the $^{10}B(n,\alpha)$ reactions where the most serious discrepancies in the nuclear data base remain. We plan to extend its region of use as a neutron fluence detector and to help resolve the discrepancies in helium gas production measurements. We plan to continue our leadership in the international (NEANSC endorsed) collaborative group which reviews the data needs and promotes required measurements for this crucial cross section. We plan to complete the work on the $^{10}B(n,\alpha_1\gamma)$ cross section at lower neutron energies (5 to 500 keV) using the hydrogen gas proportional counter as a neutron fluence monitor. This is required for more complete normalization of the higher energy measurements (0.3 to 4 MeV) which have been accepted for publication. We plan to extend the ^{10}B and ^{11}B neutron total cross section measurements to lower energies. The $^{10}B(n,\alpha_1\gamma)$ and total cross section measurements will be analyzed and submitted for publication.

Neutron Dosimetry Standard Cross Sections

The data obtained on the $^{237}Np(n,f)$, $^{239}Pu(n,f)$ and $^{233}U(n,f)$ cross sections will be analyzed. The measurements which will be analyzed were made at the LANSCE facility for neutron energies below 1 MeV. The ^{237}Np cross section, which has large uncertainties in this energy region, is used in detectors for investigating pressure degradation problems in nuclear reactors. The uncertainties in this cross section lead to costly uncertainties in the lifetime of these pressure vessels. The $^{239}Pu(n,f)$ cross section is also used as a dosimeter. The motivation for the $^{239}Pu(n,f)$ measurements is inconsistencies between measurements and calculations of well known critical assemblies using the ENDF/B-VI evaluation of this cross section. The ^{233}U

measurements were initiated due to the need for improved data for the design of a system for the transmutation of nuclear waste. It is expected that the analysis of these measurements will be completed and the work submitted for publication during this year.

Measurement of Neutron-Induced Charged-Particle Reactions

The Nuclear Interaction Measurements Project at NIST plans to continue the orderly development of measurements to improve the accuracy of important neutron cross sections as we consider the changing national nuclear data needs outlined in the recent review of the DOE nuclear data program. The group plans to continue involvement in measurements of neutron-induced charged-particle production cross sections for nitrogen, and oxygen from approximately 5- to 40-MeV neutron energy for use in Kerma calculations for medical and other applications. These measurements will be carried out in collaboration with scientists at LANL. New data will be obtained for thinner oxide samples. Measurements on the nitrogen-containing sample melamine will be made in order to further understand charged-particle production in the atmosphere. The nitrogen measurements will be analyzed and prepared for publication.

A National Repository for Fissionable Isotope Mass Standards (FIMS)

The activities of the National Repository for Fissionable Isotope Mass Standards will include acquiring deposits, lending deposits, maintaining scientific and regulatory records, and making occasional intercomparisons of well-characterized deposits by fission counting and alpha counting to verify the documented isotopic and mass analyses. The pace of these various activities is expected to increase gradually over the next three years. The acquisition of selected Argonne National Laboratory deposit collections and intercomparisons of these with NIST reference deposits are anticipated steps during this calendar year.

Leadership in Nuclear Data Evaluation

The group will maintain its leadership in the national and international standards measurement and evaluation effort. The data bases for the standards will be reviewed and necessary new measurements will be encouraged. Close investigation of the data base for hydrogen scattering will be performed. It has been noted that the large difference between the last two ENDF/B evaluations for hydrogen is largely a result of a few measurements with very small quoted uncertainties. A detailed study of the documentation for some of those measurements will be performed in order to look for subtle problems in the experiments. This process was critical for the simultaneous evaluation of the standards, which did not include hydrogen scattering, which was an important part of the process for evaluation of the standards for ENDF/B-VI. The standards subcommittee of CSEWG will study the extension of appropriate standards to higher neutron energies and encourage international cooperation in standards cross section measurements and evaluations. Efforts will be made to ensure that the data bases and evaluation codes used for the ENDF/B-VI standards evaluation will be available for future standards evaluations.

Measurement of Fission Product Properties

The measurements and analysis of the fission fragment angular and energy distributions and neutron energy spectra and angular distributions from neutron induced fission in ^{238}U should be completed. We plan to continue to provide a supporting role in this work since it provides data for applications in code modeling and corrections for detector performance.

Undirected Research

We also wish to have the flexibility to continue a modest (approximately 20%) of undirected research in the general area of neutron research in order to maintain a broad perspective within the staff on measurement methods and physical processes which influence the data program.

These measurement activities are summarized below:

Summary of Planned Activities for Calendar Year 1994

1. Begin 10 MeV measurement of the H(n,n)H angular distribution at Ohio University.

2. Complete the measurements and analyses of the ^{10}B total and $(n,\alpha_1\gamma)$ cross sections and submit them for publication.

3. Complete analysis of the dosimetry fission cross section standards in the low neutron energy region and submit the work for publication.

4. Make measurements of the O(n,cp) and N(n,cp) cross sections. Analyze the N data in preparation for publication.

5. Extend and verify the collection of samples for the National Repository for Fissionable Isotope Mass Standards.

6. Continue leadership in nuclear data standard cross section evaluations.

D. PLANS FOR CALENDAR YEAR 1995

The program this year will continue with the completion of one measurement and studying of several new measurements.

Measurement of the H(n,n)H Angular Distribution

The measurements of this important cross section at 10 MeV neutron energy at the Ohio University tandem laboratory will be completed during this year. This data will be analyzed and plans for work at higher energies will be formulated.

Measurement of ^{10}B Cross Sections

This will be an appropriate time to initiate a study of the feasibility of an improved measurement of the $^{7}Li(\alpha,n)$ reaction as the best method to determine the $^{10}B(n,\alpha_0)$ cross section at the higher neutron energies.

Measurement of the $^{237}Np(n,f)$ Dosimetry Cross Section

This will also be a suitable time to initiate a more accurate method to measure this important dosimetric reaction. The cross section at 14 MeV neutron energy currently has an uncertainty at the 7-10% level. A measurement at the 1% level at 14 MeV could be accomplished using the time-correlated associated-particle technique with the $T(d,n)^{4}He$ reaction at a low energy accelerator..

Measurement of Neutron-Induced Charged-Particle Reactions

The difficult measurements on the oxygen samples will be completed. An extensive analysis effort during this year should lead to the submission of this work for publication.

A National Repository for Fissionable Isotope Mass Standards (FIMS)

The repository will be expanded with the addition of well-characterized samples from other facilities. It is the purpose of this collection to include only verified material which has been carefully characterized and accompanied by excellent documentation. Comparisons between ^{6}Li and ^{10}B samples with ^{235}U and ^{239}Pu deposits will be made by neutron counting in a monoenergetic cold neutron beam.

Leadership in Nuclear Data Evaluation

International cooperation in the measurement and evaluation of nuclear data will continue to be encouraged. The study of the documentation of hydrogen scattering measurements with small reported uncertainties will be continued. A similar exercise will be done for important $^{237}Np(n,f)$ measurements.

E. PLANS FOR CALENDAR YEAR 1996

Measurement of the H(n,n)H Angular Distribution

The analysis of the measurements made at 10 MeV neutron energy at the Ohio University laboratory will be completed and the work will be submitted for publication. The measurements at higher neutron energies will begin.

Measurement of ^{10}B Cross Sections

This will be an appropriate time to begin measurements such as the ^{7}Li(α,n) reaction which will lead to an improved ^{10}B(n,α_0) cross section at the higher neutron energies.

Measurement of the $^{237}Np(n,f)$ Neutron Dosimetry Cross Section

The measurement of the ^{237}Np(n,f) cross section at 14 MeV neutron energy which should resolve the normalization problem for this cross section will be completed and submitted for publication.

A National Repository for Fissionable Isotope Mass Standards (FIMS)

The repository will continue to acquire critical samples. The collection will be periodically measured using alpha-particle spectrometry to verify the integrity of the material. The long-term stability of actinide deposits as a function of chemical form and substrate material will be evaluated.

Leadership in Nuclear Data Evaluation

The standards subcommittee of the CSEWG will begin to prepare for a possible new evaluation of the standard cross sections. International cooperation in the measurement and evaluation of nuclear data will continue to be encouraged.

F. RECENT PUBLICATIONS

A.D. Carlson, W.P. Poenitz, G.M. Hale, R.W. Peelle, D.C. Dodder, C.Y. Fu, W. Mannhart, "The ENDF/B-VI Neutron Cross Section Measurement Standards," National Institute of Standards and Technology, NISTIR-5177 (1993); also ENDF-351, Brookhaven National Laboratory.

D.M. Gilliam, G.P. Lamaze, M.S. Dewey, and G.L. Greene, "Mass Assay and Uniformity Tests of Boron Targets by Neutron Beam Methods," Nucl. Instr. and Meth. Phys. Res. (in press).

A.D. Liberman, P. Albats, H. Pfutzner, C. Stoller, and D.M. Gilliam, "A Method to Determine the Absolute Neutron Output of Small D-T Generators," Nucl. Instr. and Meth. Phys. Res. (in press).

R.A. Schrack, O.A. Wasson, D.C Larson, J.K. Dickens, and J.H. Todd, "The $^{10}B(n,\alpha_1\gamma)^7Li$ Cross Section Between 0.2 and 4.0 MeV," to be published in Nucl. Sci. Eng. (1993).

W.E. Parker, J.E. Lynn, G.L. Morgan, P.W. Lisowski, A.D. Carlson, and N.W. Hill, "Intermediate Structure in the Neutron-Induced Fission Cross Section of ^{236}U," to be submitted to Nucl. Phys. (1993).

C. Zoeller, A. Gavron, J.P. Lestone, M. Mutterer, J.P. Theobald, A. Carlson, and G. Petitt, "The $^{238}U(n,f)$ Reaction at Incident Neutron Energies from 1 to 200 MeV," Bull. Am. Phys. Soc. **37**, (5), 1300 (1992).

W.P. Poentiz and A.D. Carlson, "The Data Base of the Standards and Related Cross Sections after ENDF/B-VI," Symp. on Nuclear Data Evaluation Methodology, 12-16 October, 1992, Brookhaven National Laboratory, USA, p. 3-I-4.

A.D. Carlson, O.A. Wasson, P.W. Lisowski, J.L. Ullmann, and N.W. Hill, "Measurements of the $^{235}U(n,f)$ Cross Section in the 3 to 30 MeV Neutron Energy Region," in Proc. *International Conference on Nuclear Data for Science and Technology*, 13-17 May, 1991, Jülich, Fed. Rep. of Germany, S.M. Qaim, Ed., pp. 518-520.

P.W. Lisowski, A. Gavron, W.E. Parker, J.L. Ullmann, S.J. Balestrini, A.D. Carlson, O.A. Wasson, and N.W. Hill, "Fission Cross Section Ratios for ^{234,234,236}U Relative to ^{235}U from 0.5 to 400 MeV," in Proc. *International Conference on Nuclear Data for Science and Technology*, 13-17 May, 1991, Jülich, Fed. Rep. of Germany, S.M. Qaim, Ed., pp 732-733.

A.D. Carlson, O.A. Wasson, P.W. Lisowski, J.L. Ullmann, and N.W. Hill, "Measurements of the ^{235}U(n,f) Cross Section for Neutron Energies from 3 to 30 MeV," in Proc. *NEANDC Specialists Meeting on Neutron Cross Section Standards for the Energy Region above 20 MeV*, 21-23 May, 1991, Uppsala, Sweden, (NEANDC-305U, 1991) pp. 165-176.

P.W. Lisowski, A. Gavron, W.E. Parker, J.L. Ullmann, S.J. Balestrini, A.D. Carlson, O.A. Wasson, and N.W. Hill, "Fission Cross Sections in the Intermediate Energy Region," in Proc. *NEANDC Specialists Meeting on Neutron Cross Section Standards for the Energy Region above 20 MeV*, 21-23 May, 1991, Uppsala, Sweden, (NEANDC-305U, 1991) pp. 177-186.

A.D. Carlson, "Summary of the Workshop on Cross Sections Above 20 MeV Neutron Energy, Other Than Hydrogen Scattering or Activation, for Consideration as Standards," in Proc. *NEANDC Specialists Meeting on Neutron Cross Section Standards for the Energy Region above 20 MeV*, 21-23 May, 1991, Uppsala, Sweden, (NEANDC-305U, 1991) pp. 236-239.

R.A. Schrack, O.A. Wasson, D.C. Larson, J.K. Dickens, and J.H. Todd, "Measurement of the ^{10}B(n,$\alpha_1\gamma$)^{7}Li Cross Section in the 0.3 to 4 MeV Neutron Energy Interval," in Proc. *International Conference on Nuclear Data for Science and Technology*, 13-17 May 1991, Jülich, Fed. Rep. of Germany, S.M. Qaim, Ed., pp 507-509.

E.I. Sharapov, S.A. Wender, H. Postma, S.J. Seestrom, C.R. Gould, O.A. Wasson, Yu. P. Popov, and C.D. Bowman, "The Measurements of Parity Violation in Resonant Neutron-Capture Reactions," in *Capture Gamma-Ray Spectroscopy*, Pacific *Grove, CA*, 1990, ed. by Richard W. Hoff, (AIP Conference Proceedings 238, New York, 1991) pp. 756-763.

G. TECHNICAL AND PROFESSIONAL COMMITTEE PARTICIPATION AND LEADERSHIP

Allan D. Carlson

Chairman, Standards Subcommittee, Cross Section Evaluation Working Group (CSEWG), National Nuclear Data Center.

Member, Evaluation Committee of CSEWG.

Member, Data Status and Requests Subcommittee of CSEWG.

Chairman, Nuclear Energy Agency Nuclear Data Committee (NEANDC) Working Group on the $^{10}B(n,\alpha)$ Cross Section Standard.

Member, International Program Committee, International Conference on Nuclear Data for Science and Technology, to be held in Gatlinburg, TN in 1994.

David M. Gilliam

Represent NIST on Section III (Neutron Measurements) of the Comitè Consultatif pour les Ètalons (Standards) de Mesure des Rayonnements Ionisants (CCEMRI), a committee associated with the International Bureau of Weights and Measures (BIPM).

Roald A. Schrack

Department of Commerce representative to Environmental Protection Agency Interagency Working Group on Residual Radioactivity of Federal Installations

Oren A. Wasson

Member, Neutron Measurements Subcommittee of CSEWG.

IV. BUDGET REQUIREMENTS

Allocation of Funds

The budget requirements for calendar years 1994, 1995, and 1996 for the Neutron Cross Section Standards and Instrumentation Program are submitted for the period beginning January 1, 1994. NIST (formerly NBS) will continue to provide approximately one-third support for the program. The budget request from the Department of Energy for CY-94 is $485,000 which is the same Department of Energy funding as last year. It reflects the anticipated constant government salaries for fiscal year 1994. The budget requirements for CY-95 and CY-96 include the inflationary increase of approximately 2% per year. Three staff members of the Ionizing Radiation Division (Carlson, Schrack, and Wasson) will continue to work full time on this program. Additional scientific and technical staff of the Ionizing Radiation Division will provide the remaining manpower requirements.

The detailed budget request for calendar year 1994 is given in the Grant Application Budget Period Summary while the budget request for the three-year period is shown in the Grant Application Project Period Summary.

The activity covered by this agreement consists of work which requires the definition of measurement methods, material property data, and standards of basic scientific and engineering units and the application of primary standards to ensure equity and comparability in U.S. commerce, international trade, and technical activities. As such it complies with OMB Circular-A-76, revised under paragraph 5f.

ER F 4620.1
(7-85)

U.S. Department of Energy
Grant Application Budget Period Summary
(See Reverse for Definitions and Instructions)

OMB Approval
No. 1910-1400

Please Print or Type

Organization: NATIONAL INSTITUTE STANDARDS TECHNOLOGY	Period Covering: From: 1 JAN 1994	FOR DOE USE ONLY Proposal No:
Principal Investigator (P.I.)/Project Director (P.D.): O.A.WASSON (PI)/ R.S.CASWELL (PD)	To: 31 DEC 1994	Award No.:

A. SENIOR PERSONNEL PI/PD Co PIs, Faculty and Other Senior Associates (List each separately with title, A.6 show number in brackets. Attach separate sheet, if required.)	DOE Funded Person-Mos. Cal.	Acad.	Sumr	Funds Requested By Applicant $
1 Dr. A. D. Carlson	12			$86,600
2 Dr. R. A. Schrack	12			$86,600
3 Dr. O. A. Wasson (PI)	12			$86,600
4 Dr. D. M. Gilliam	3			$18,400
5 Dr. J. A. Grundl	1			$ 7,400
6 ( ) TOTAL SENIOR PERSONNEL				
B. OTHER PERSONNEL (SHOW NUMBERS IN BRACKETS)				
1 ( ) POST DOCTORAL ASSOCIATES				
2 ( ) OTHER PROFESSIONALS (TECHNICIAN, PROGRAMMER, ETC.)				
3 ( ) GRADUATE STUDENTS				
4 ( ) UNDERGRADUATE STUDENTS				
5 ( ) SECRETARIAL-CLERICAL				
6 ( ) OTHER				
TOTAL SALARIES AND WAGES (A + B)				$285,600
C. FRINGE BENEFITS (IF CHARGED AS DIRECT COSTS)				$54,400
TOTAL SALARIES, WAGES AND FRINGE BENEFITS (A + B + C)				$340,000
D. EQUIPMENT (LIST ITEM AND DOLLAR AMOUNT FOR EACH ITEM)				
TOTAL EQUIPMENT				
E. TRAVEL 1. DOMESTIC (INCL. CANADA AND U.S. POSSESSIONS)				$20,000
2. FOREIGN				0
F. OTHER DIRECT COSTS				
1. MATERIALS AND SUPPLIES				$17,000
2. PUBLICATION COSTS/PAGE CHARGES				$ 6,000
3. CONSULTANT SERVICES				0
4. COMPUTER (ADPE) SERVICES				$12,000
5. CONTRACTS AND SUBGRANTS				0
6. OTHER				$27,000
TOTAL OTHER DIRECT COSTS				$62,000
G. TOTAL DIRECT COSTS (A THROUGH F)				$422,000
H. INDIRECT COSTS (SPECIFY RATE AND BASE) TOTAL INDIRECT COSTS 0.906 of (Salaries and Benefits)				$308,000
I. TOTAL DIRECT AND INDIRECT COSTS (G & H)				$730,000
J. APPLICANT'S COST SHARING (IF ANY)				$245,000
K. TOTAL AMOUNT OF THIS REQUEST (ITEM I LESS ITEM J)				$485,000

PI/PD TYPED NAME & SIGNATURE (PD) R.S. Caswell *R S Caswell*	DATE 6/29/93
INST. REP. TYPED NAME & SIGNATURE M. A. Dewese *[illegible]*	DATE [illegible]

ER F 4620 1A
(7-85)

U.S. Department of Energy
GRANT APPLICATION
PROJECT PERIOD SUMMARY
(Must be completed for all new and renewal applications.)
Please Print or Type

OMB Approval
No 1910-1400

Categories	CY94 01 Budget Period	CY95 02 Budget Period	CY96 03 Budget Period	04 Budget Period	05 Budget Period
A. Senior Personnel Totals	$285,600	$291,000	$298,000		
B. Other Personnel Totals	0	0	0		
C. Fringe Benefit Totals	54,400	56,000	57,000		
Total of A, B & C	340,000	347,000	355,000		
D. Equipment	0	0	0		
E. Travel 1. Domestic	20,000	15,000	10,000		
2. Foreign	0	5,000	10,000		
F. Other Direct Costs	62,000	64,000	65,000		
G. Total Direct Costs	422,000	431,000	440,000		
H. Total Indirect Costs	308,000	314,000	320,000		
I. Total Direct & Indirect Costs	730,000	745,000	760,000		
J. Applicant's Cost-Sharing (If any)	245,000	250,000	255,000		
K. Total Amount of Request (Item I. Less Item J.)	(1)* $485,000	(2) $495,000	(3) $505,000	(4) –	(5) –

*This should equal Item K on Budget Period Summary (ER/F/4620.1)

ESTIMATE

TOTAL COST OF PROJECT
$ 1,485,000
(add K(1) thru (5))

Curriculum Vitae

Name **Allan D. Carlson**

Birth date, place [redacted] [redacted]

Education (College or University)

	Degree	Date
Concordia College, Moorhead, MN	B.A.	1961
University of Wisconsin, Madison	M.S.	1963
University of Wisconsin, Madison	Ph.D.	1966

Employment Affiliations

Position	Employer	Dates
Nuclear Physicist	National Institute of Standards & Technology	1972-Present
Staff Scientist	Gulf Energy and Environmental Systems	1970-1972
Staff Member	Gulf General Atomic	1967-1970
Staff Associate	General Atomic Div., General Dynamics	1966-1967
Research Assistant	Dept. of Physics, University of Wisconsin	1961-1966
Research Associate	Argonne National Laboratory (summers)	1961-1962

Professional Society Memberships

American Physical Society

Committee Activities

Member of U.S. Nuclear Cross Section Advisory Committee 1970-1972; Member of Cross Section Evaluation Work Group (CSEWG) 1972-present; Chairman of Standards Subcommittee of CSEWG 1980-present; Program Chairman, 1989 Conference on 50 Years with Nuclear Fission; Chairman of the NEANDC-endorsed Working Group on the $^{10}B(N,\alpha)$ Cross Section Standards 1989-present; Member International Program Committee for the International Conference on Nuclear Data for Science and Technology to be held in 1994.

Technical Interests

Experimental nuclear physics; Neutron cross sections; Neutron standard reference data; Neutron reaction mechanisms.

Publications

(List attached)

List of Publications

Allan D. Carlson

A.D. Carlson, W.P. Poenitz, G.M. Hale, R.W. Peelle, D.C. Dodder, C.Y. Fu, W. Mannhart, "The ENDF/B-VI Neutron Cross Section Measurement Standards," National Institute of Standards and Technology, NISTIR-5177 (1993); also ENDF-351, Brookhaven National Laboratory.

W.E. Parker, J.E. Lynn, G.L. Morgan, P.W. Lisowski, A.D. Carlson, and N.W. Hill, Intermediate Structure in the Neutron-Induced Fission Cross Section of ^{236}U, to be submitted to Nucl. Phys. (1993)

C. Zoeller, A. Gavron, J.P. Lestone, M. Mutterer, J.P. Theobald, A. Carlson, G. Petitt, The $^{238}U(n,f)$ Reaction at Incident Neutron Energies from 1 to 200 MeV, Bull. Am. Phys. Soc. 37, (5), 1300 (1992).

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A.D. Carlson, O.A. Wasson, P.W. Lisowski, J.L. Ullmann, and N.W. Hill, Measurements of the $^{235}U(n,f)$ Cross Section in the 3 to 30 MeV Neutron Energy Region, Proc. of the International Conference on Nuclear Data for Science and Technology, 13-17 May, 1991, Jülich, Fed. Rep. of Germany, S. M. Qaim, Ed., pp 518-520.

P.W. Lisowski, A. Gavron, W.E. Parker, J.L. Ullmann, S.J. Balestrini, A.D. Carlson, O.A. Wasson, and N.W. Hill, Fission Cross Section Ratios for $^{233,234,236}U$ Relative to ^{235}U from 0.5 to 400 MeV, Proc. of the International Conference on Nuclear Data for Science and Technology, 13-17 May, 1991, Jülich, Fed. Rep. of Germany, S. M. Qaim, Ed., pp 732-733.

A.D. Carlson, O.A. Wasson, P.W. Lisowski, J.L. Ullmann, and N.W. Hill, Measurements of the $^{235}U(n,f)$ Cross Section for Neutron Energies from 3 to 30 MeV, Proc. of the NEANDC Specialists Meeting on *Neutron Cross Section Standards for the Energy Region above 20 MeV*, 21-23 May, 1991, Uppsala, Sweden, NEANDC-305 U, pp. 165-176.

P.W. Lisowski, A. Gavron, W.E. Parker, J.L. Ullmann, S.J. Balestrini, A.D. Carlson, O.A. Wasson, and N.W. Hill, Fission Cross Sections in the Intermediate Energy Region, Proc. of the NEANDC Specialists Meeting on *Neutron Cross Section Standards for the Energy Region above 20 MeV*, 21-23 May, 1991, Uppsala, Sweden, NEANDC-305 U, pp. 177-186.

A.D. Carlson, Summary of the Workshop on Cross Sections Above 20 MeV Neutron Energy, Other Than Hydrogen Scattering or Activation, for Consideration as Standards, Proc. of the NEANDC Specialists Meeting on *Neutron Cross Section Standards for the Energy Region above 20 MeV*, 21-23 May, 1991, Uppsala, Sweden, NEANDC-305 U, pp. 236-238.

A.D. Carlson, Neutron Standard Cross Sections in Reactor Physics - Need and Status, *Trans. Am. Nucl. Soc.* **62**, 525 (1990).

K.H. Böckhoff, A.D. Carlson, O.A. Wasson, J.A. Harvey, and D.C. Larson, Electron Linear Accelerators for Fast Neutron Data Measurements in Support of Fusion Energy Applications, Nucl. Sci. Eng. **106**, 192 (1990).

A.D. Carlson, W.P. Poenitz, G.M. Hale, and R.W. Peelle, The Neutron Cross Section Standards Evaluations for ENDF/B-VI, *Trans. Am. Nucl. Soc.* **60**, 604 (1989).

J.W. Behrens and A.D. Carlson, Editors of the Proc. of the Conference entitled "Fifty Years with Nuclear Fission," published by the American Nuclear Society (1989).

P.W. Lisowski, J.L. Ullmann, S.J. Balestrini, A.D. Carlson, O.A. Wasson, and N.W. Hill, Neutron Induced Fission Cross Sections for ^{232}Th, ^{235,236}U, ^{237}Np and ^{239}Pu, Proc. of the Conference on Fifty Years with Nuclear Fission (1989), pp. 443-448

J.W. Behrens, O.A. Wasson, A.D. Carlson, and Hongchang Ma, Development of a ^{3}He/Xe Gas Scintillation Counter to Measure the ^{3}He(n,p)T Cross Section in the Intermediate Energy Range, Proc. of the International Conference on Nuclear Data for Science and Technology, Mito, Japan, May 30-June 3, 1988, pp. 371-374.

J.W. Behrens, Hongchang Ma, and A.D. Carlson, Development of a ^{3}He/Xe Gas Scintillation Counter for Measuring the Neutron Energy Range from Thermal to 2 MeV, *Trans. Am. Nucl. Soc.* **56**, 586 (1988).

A.D. Carlson, O.A. Wasson, P.W. Lisowski, J.L. Ullmann, and N.W. Hill, A Study of the ^{235}U(n,f) Cross Section in the 3 to 30 MeV Energy Region, Proc. of the International Conference on Nuclear Data for Science and Technology, Mito, Japan, May 30-June 3, 1988, pp. 1029-1032.

R.G. Johnson, A.D. Carlson, O.A. Wasson, K.C. Duvall, J.W. Behrens, M.M. Meier, B.D. Patrick, and M.S. Dias, Measurements of the ^{235}U(n,f) Standard Cross Section at the National Bureau of Standards, Proc. of the International Conference on Nuclear Data for Science and Technology, Mito, Japan, May 30-June 3, 1988, pp. 1037-1040.

P.W. Lisowski, J.L. Ullmann, S.J. Balestrini, A.D. Carlson, O.A. Wasson, and N.W. Hill, Neutron Induced Fission Cross Section Ratios for ^{232}Th, ^{235,238}U, ^{237}Np and ^{239}Pu from 1 to 400 MeV, Proc. of the International Conference on Nuclear Data for Science and Technology, Mito, Japan, May 30-June 3, 1988, pp. 97-99.

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P.W. Lisowski, J.L. Ullmann, S.J. Balestrini, A.D. Carlson, O.A. Wasson, and N.W. Hill, Neutron Induced Fission Cross Section Ratios for $^{235,238}U$, and ^{237}Np from 1 to 400 MeV, Bull. Am. Phys. Soc. **32**, 1035 (1987).

I. Schröder, D.M. Gilliam, A.D. Carlson, S.W. Bright, and J.M.R. Hutchinson, International ^{235}U Fission Foil Mass Intercomparison, *Trans. Am. Nucl. Soc.* **53**, 473 (1986).

A.D. Carlson, New Measurements of the Ratio of the $^{10}B(n,\alpha)$ to $^{6}Li(n,t)$ Cross Sections in the eV Energy Region, Radiation Effects, **96**, 109 (1986).

A.D. Carlson, W.P. Poenitz, G.M. Hale, and R.W. Peelle, The Neutron Cross Section Standards Evaluations for ENDF/B-VI, Radiation Effects **96**, 87 (1986).

O.A. Wasson, A.D. Carlson, and K.C. Duvall, Analysis of the Effect of Random Events on the 14 MeV $^{235}U(n,f)$ Cross Section Measurement, Radiation Effects **96**, 9 (1986).

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K.C. Duvall, A.D. Carlson, and O.A. Wasson, Calibration of the NBS Black Neutron Detector at 2.3 MeV Using the Time-Correlated Associated Particle Method. Proc. of the Eighth Conf. on the Appl. of Accel. in Research and Industry, Denton, TX, Nov. 12-14, 1984, Nucl. Instr. and Meth. **B10/11**, 931-933 (1985).

M.S. Dias, A.D. Carlson, R.G. Johnson, and O.A. Wasson, Application of the Dual Thin Scintillator Neutron Flux Monitor in a $^{235}U(n,f)$ Cross Section Measurement, Proc. of the International Atomic Energy Agency (IAEA) Advisory Group Meeting on Nuclear Standard Reference Data, Nov. 12-16, 1984, Geel, Belgium, IAEA-TEDOC-335 (1985) p. 467-470.

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A.D. Carlson, J.W. Behrens, R.G. Johnson, and G.E. Cooper, Absolute Measurements of the $^{235}U(n,f)$ Cross Section for Neutron Energies from 0.3 to 3 MeV, Proc. International Atomic Energy Agency (IAEA) Advisory Group Meeting on Nuclear Standard Reference Data, Nov. 12-16, 1984, Geel, Belgium, p. 163, IAEA-TEDOC-335 (1985).

A.D. Carlson, Standard Cross Section Data, Prog. Nucl. Energy, **13**, No. 2 (1984).

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Curriculum Vitae

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Technical Committees

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Curriculum Vitae

Name	**Roald A. Schrack**		
Birth date, place	[redacted]		[redacted]

Education (College or University)

	Degree	Date
University of California at Los Angeles	B.S.	1949
University of California at Los Angeles	M.S.	1950
University of Maryland	Ph.D.	1961

Employment Affiliations

Position	Employer	Dates
Physicist	National Institute of Standards & Technology	1949-Present

Professional Society Memberships

American Physical Society
Institute of Electronic and Electrical Engineers
Sigma Xi

Committee Activities

EPA Environmental Radioactivity Review Panel

Technical Interests

Neutron detectors; Neutron resonance radiography; Neutron cross section measurement; Computer analysis; Electronic instrumentation.

Publications
(List attached)

List of Publications

Roald. A. Schrack

Schrack, R.A., Wasson, O.A., Larson, D.C., Dickens, J.K., and Todd, J.H., "The $^{10}B(n,\alpha_1\gamma)^7Li$ Cross Section Between 0.2 and 4.0 MeV," submitted to Nuclear Science and Engineering,(1992).

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Schrack, R.A., Very Small, High Density Electron Beams from Field Emission, Symposium, Pittsburgh, PA (1954).

Schrack, R.A., Radio Frequency Power Measurements, NBS Circular 532 (1953).

Curriculum Vitae

Name	**Oren A. Wasson**	
Birth date, place	[redacted]	[redacted]

Education (College or University)

	Degree	Date
College of Wooster	B.A.	1957
Yale University	M.S.	1959
Yale University	Ph.D.	1964

Employment Affiliations

Position	Employer	Dates
Physicist	National Institute of Standards & Technology	1973-Present
Technical Advisor (IAEA)	Democritos Research Center, Greece	1975
Physicist	Brookhaven National Laboratory	1965-1973
Visiting Scientist	Oak Ridge National Laboratory	1971-1972
Post Doc Research Associate	Yale University	1963-1965
Summer Student	Los Alamos Scientific Laboratory	1957-1958

Professional Society Memberships

American Physical Society
American Association for Advancement of Science
Sigma Xi

Committee Activities

U.S. Department of Energy Nuclear Data Committee (1982-1992);
General Program Advisory Committee for International Conference on Nuclear Data for Basic and Applied Science (1983-1985);
General Chairman, 1989 Conference entitled "50 Years with Nuclear Fission."
Member Neutron Measurements Subcommittee of CSEWG (1992-present).

Technical Interests

Neutron interaction measurements; Neutron and gamma-ray detector development; Accelerator technology; Nuclear reactions; Neutron dosimetry; Applications of Rutherford backscattering spectroscopy; Nuclear data applications.

Publications

(List attached)

List of Publications

OREN A. WASSON

R. A. Schrack, O. A. Wasson, D. C. Larson, J. K. Dickens, and J. H. Todd, "The $^{10}B(n,\alpha_1\gamma)^7Li$ Cross Section Between 0.2 and 4.0 MeV," to be published in Nucl. Sci. Eng. (Aug, 1993).

T. M. Lee, S. M. Sterbenz, F. B. Bateman, R. C. Haight, O. A. Wasson, F. C. Goeckner, C. E. Brient, S. M. Grimes, and H. Vonach, "The C(n,α) Reaction from Threshold to 30 MeV," Bull. Am. Phys. Soc. **38**,1051 (1993).

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A. D. Carlson, O. A. Wasson, P. W. Lisowski, J. L. Ullmann, and N. W. Hill, "Measurements of the $^{235}U(n,f)$ Cross Section in the 3 to 30 MeV Neutron Energy Region," Proc. International Conference on Nuclear Data for Science and Technology, 13-17 May, 1991, Jülich, Fed. Rep. of Germany, S. M. Qaim, Ed., pp 518-520 (1992).

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APPENDIX

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The ENDF/B-VI Neutron Cross Section Measurement Standards

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PREFACE

The Standards Subcommittee of the U.S. Cross Section Evaluation Working Group (CSEWG) encourages, performs, coordinates, oversees and approves evaluations of cross section measurement standards for the Evaluated Nuclear Data Files (ENDF/B). This report describes the process used to obtain the standards evaluations for the sixth version of ENDF/B. New evaluations have been produced for each of the cross section standards and the spontaneous fission neutron spectrum for ^{252}Cf.

ABSTRACT

This document provides information on the neutron cross section standards placed in the ENDF/B-VI library. The H(n,n), ^{3}He(n,p), and C(n,n) cross sections were each obtained from well established R-matrix analysis techniques. The additional standards, i.e., the ^{6}Li(n,t), ^{10}B(n,α), ^{10}B(n,α_1), Au(n,γ), and ^{235}U(n,f) cross sections, were obtained with a new method. The new method involves combining the results of a simultaneous evaluation and R-matrix analyses. Contained herein is a discussion of the development of the method, a description of the evaluation process, some information on the various experiments used in the analyses, comparisons of the R-matrix, simultaneous evaluation and combination results, and comparisons to ENDF/B-V. Tables of numerical data are given for each of the cross section standards. Also the new ENDF/B-VI evaluation for the spontaneous fission neutron spectrum for ^{252}Cf is given.

KEYWORDS: Au(n,γ); ^{10}B(n,α); ^{10}B(n,α_1); C(n,n); combining procedure; ENDF/B-VI Neutron Cross Section Standards; H(n,n); ^{3}He(n,p); ^{6}Li(n,t); R-matrix; simultaneous evaluation; standard; ^{235}U(n,f); ^{252}Cf spontaneous fission neutron spectrum

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1. INTRODUCTION

The measurement of neutron cross sections can often be significantly simplified by using a cross section standard which eliminates the need for a direct measurement of the neutron fluence. A cross section measurement is then made by placing both the sample which is used for the cross section determination and the sample utilizing the cross section standard in the same neutron beam. For thin samples the ratio of counting rates is proportional to the ratio of the cross section to be measured to that of the standard. The accuracy of such a cross section measurement is limited by the uncertainty in the standard cross section relative to which it is measured. Normally such standards apply only in certain energy regions, where their cross sections are relatively smooth and well known. Improvements in the standard cause all cross sections measured relative to that standard to be improved. This is the reason for the emphasis on increasing the quality of the neutron cross section standards. This also explains why they must be evaluated prior to completion of a new version of an evaluated nuclear data file such as ENDF/B. The standards include H(n,n), ^{3}He(n,p), ^{6}Li(n,t), ^{10}B(n,α), ^{10}B(n,α_1), C(n,n), Au(n,γ), and ^{235}U(n,f).

2. EVALUATION METHODS

Measurement programs have continuously improved the data base of the standards. Evaluations have also improved with time; however, there have been significant weaknesses in earlier evaluations. In some cases evaluations have been performed by qualitatively or semi-quantitatively combining different kinds of data sets by simply drawing smooth curves through the existing data. Such evaluations are difficult to document and it is not clear how to determine meaningful uncertainties and covariance information.

In previous evaluations for ENDF/B, a hierarchical approach was followed. The lighter element cross section standards were generally considered to be better known. The H(n,n) cross section was considered the best known standard and was evaluated first and independently of the other standards. This standard is considered so well known that measurements relative to it are often called absolute measurements. The ^{6}Li(n,t) cross section evaluation was performed next. The only ^{6}Li(n,t) data which were used were absolute measurements or those measured relative to the H(n,n) standard which were converted to cross sections using the adopted hydrogen evaluation. Then the ^{10}B+n standard cross sections were evaluated. The only ^{10}B data which were used were absolute measurements and those relative to H(n,n) and ^{6}Li(n,t) which were converted using the new hydrogen and lithium evaluations. This process was continued for each of the standards. This method for using ratio measurements does not use all the information available. It does not include absolute and ratio data on the same basis as they were measured. For example, a ratio of the ^{10}B(n,α) to the ^{6}Li(n,t) cross sections would be used in the ^{10}B(n,α) cross section evaluation but not in the ^{6}Li(n,t) evaluation.

The difficulties with the hierarchical evaluation procedure and the successes already realized using comprehensive objective data combination techniques [1] led to the seeking out of a more global approach for ENDF/B-VI than had been used earlier. With such "objective" techniques one relies on least-squares or similar procedures to combine the input data consistent with experimental uncertainties. Each experiment must be evaluated in detail to represent it fairly in this process. The method should be able to handle the full information content of the data base. Thus data would be evaluated simultaneously to assure proper use of the available information. Ratio measurements of standard cross sections would have an impact on each of the cross sections in the ratio. Correlations among the experimental data would be taken into account in the simultaneous evaluation.

It was recognized that there are some absolute cross section measurements of similar quality which are not normally considered standards. For those cases where accurate ratio measurements of these cross sections to those of standards exist, the evaluation of that cross section should be performed simultaneously with the standards evaluation since it in principle will affect the values of the evaluated standards and their uncertainties. Thus the standards and other well-known cross sections would be evaluated with the same procedure. As a practical matter the addition of data from many nuclides can become a very immense problem though it can be handled. Very few cross sections would have any appreciable impact on the determination of a standard cross section except other standards. Including data on $^{238}U(n,\gamma)$, $^{238}U(n,f)$ and $^{239}Pu(n,f)$ could improve the quality of the standards evaluations since precise absolute measurements exist and many ratio measurements to the standards are available. There is of course the benefit that evaluations of these important nuclear reactor fuel cross sections will be obtained. It was also felt that the evaluation should include the use of average cross sections over selected energy intervals for appropriate heavy-element cross sections to take advantage of data sets that extend down to thermal energies.

The presence of shape measurements which extend to thermal energies in addition to absolute data imply that an evaluation of the standards will provide information on the thermal constants. In principle the thermal constants could be evaluated simultaneously with the standards. Including the results of a completed thermal constants evaluation, with the associated variance-covariance information, in the standards evaluation process is equivalent to this. Therefore either the thermal constants data base or the complete results of a thermal cross section evaluation should be used as input to this evaluation.

It was perceived that it is important to retain fits to theory in the evaluation of the light element standards. This could be implemented with R-matrix analyses. Such analyses can provide coupling to reaction theory and give a smooth meaningful analytical expression for the energy dependance of the cross sections. The accurate determination of R-matrix parameters does however require a rather large data base. Data in addition to angle integrated neutron cross sections such as differential cross sections, polarizations, and charged particle measurements involving the same compound nucleus can have a significant impact on the standard cross sections. In the R-matrix analysis, different reactions leading to the same compound nucleus are linked by unitarity to the standard cross section. This condition imposes constraints on the standard cross section which are particularly strong near resonances.

The ideal way to perform this evaluation would be to develop a single fitting program that would use all the experimental data involving these reactions. The evaluation should provide output covariance data which is consistent with a cross section evaluation that weights input data with the inverse of its variance-covariance matrix. The output for the light elements would be R-matrix parameters, while average cross sections at many energies would be output for the heavy element cross sections. It was decided that the H(n,n), $^{3}He(n,p)$ and C(n,n) cross sections would not be evaluated in this analysis. For H(n,n) the cross sections were considered very well known so that data on the other nuclides would have very little impact on it. This cross section was thus treated as absolute in the evaluation. For ^{3}He(n,p) and C(n,n) very few ratio measurements to other standards exist so little would be gained by putting them into the evaluation. Separate R-matrix evaluations were performed at a later time for each of these standards.

The single fitting program was not implemented. Instead the decision was made that the evaluation would be the result of combining a simultaneous evaluation using generalized least squares with separate R-matrix analyses. This procedure took advantage of the strengths of the two different analysis modes which can make use of separate classes of experimental information to impact on the evaluation of the standard cross sections. It should be noted that under proper conditions a global fitting procedure could be achieved by combining the output of the simultaneous and R-matrix analyses.

3. EVALUATION PROCEDURE FOR H(n,n), ^{3}He(n,p), AND C(n,n)

For ENDF/B-VI, the hierarchical approach was retained for H(n,n) to the extent that measurements relative to it were treated as absolute. A nucleon-nucleon cross section evaluation by Dodder and Hale [2] was performed. This charge independent R-matrix evaluation, which was adopted for ENDF/B-VI, made use of a large data base of n-p and p-p experimental data at energies below 30 MeV. A summary of the channel configuration and data fitting characteristics of the analysis is given below.

Channel	Maximum Angular Momentum ℓ_{max}	Channel Radius a_c (fm)
n-p	3	3.26
p-p	3	3.26

Reaction	Number of Observable Types	Number of Data Points	χ^2
n-p scattering	3	448	407
p-p scattering	4	388	399
Totals:	7	836	806
Number of parameters = 33 $\Rightarrow$ χ^2 per degree of freedom = 1.004*			

*Including recent corrections [3] to the 16.9 MeV n-p analyzing power data of Tornow *et al.* [4] reduces the overall χ^2 per degree of freedom of the fit to 0.9988.

This data base includes measurements not used in the Hopkins-Breit [5] phase-shift analysis which was the basis for the hydrogen evaluation for versions II, III, IV, and V of ENDF/B. This evaluation gives a representation of the n-p and p-p experimental data in the 0-30 MeV energy range that is comparable to or better than that of other recent work [6],[7]. The new analysis also gives reasonable results for newly measured quantities, such as the polarization transfer data from Karlsruhe [8].

The charge independent model used takes the isospin-1 reduced-width amplitudes in the R-matrix describing n-p scattering to be identically the same as those describing p-p scattering. The energy eigenvalues in the two systems are taken to differ only by an overall constant Coulomb energy shift. This simple model allows the p-p scattering data to influence the n-p fit. Where measurements of the cross section and analyzing powers for the p-p and n-p reactions are compared, the data are quite different at the same energy. These differences, coming primarily from Coulomb terms and symmetrization properties of the two systems, are well reproduced by the charge-independent calculation. The new data, including a coherent scattering length evaluation by Holden [9] leads to changes in the shapes of the angular distribution compared to those of ENDF/B-V.

Two quantities often used to characterize the center of mass n-p angular distribution near 14 MeV are the back-angle cross section, $\sigma(180°)$, and the asymmetry ratio $R = \sigma(180°)/\sigma(90°)$. The ENDF/B-VI evaluation gives for these quantities at $E_n = 14.1$ MeV:

$$\sigma(180°) = 58.89 \pm 0.60 \text{ mb} \quad , \quad R = 1.093 \pm 0.010 \quad .$$

The value of R is in agreement with most previous measurements. However, it disagrees with a measurement made after the completion of the evaluation by Ryves and Kolkowski [10], of $R = 1.053 \pm 0.015$, which is consistent with the ENDF/B-V value. The measurement of [10] was actually a determination at 14.47 MeV of the ratio of the 180° to 110° cross sections from which a value of R was derived. The ENDF/B-VI values of the back-angle cross section and asymmetry ratio, on the other hand, are in excellent agreement with an evaluation of the 14.1 MeV experimental data that was done in 1982 by Vincour, Bém, and Presperin [11].

The ENDF/B-VI total elastic cross section, its uncertainty and the Legendre coefficients for atomic hydrogen are given in tables 1 and 2. Use is confined to energies sufficiently high so that molecular binding effects do not alter the cross sections. Complete covariance files are not presently available for the H(n,n) cross section. In figure 1, measurements and the ENDF/B-V evaluation [5],[12] are compared with the ENDF/B-VI results for the angular distribution for neutron scattering at 14.1 MeV neutron energy. The difference in the cross sections at 180 degrees in the center of mass system between ENDF/B-VI and ENDF/B-V is significant. This angle corresponds to proton recoils at zero degrees in the laboratory system, which is commonly used for proton recoil detectors. In figure 2, precise hydrogen total cross section measurements and the ENDF/B-VI evaluation are compared with the ENDF/B-V evaluation. The new evaluation is in somewhat better agreement with the measurements than the ENDF/B-V results. There is also a reduction in the uncertainty of the total cross section in the new evaluation, compared with that of ENDF/B-V.

The evaluation of the $^3He(n,p)$ cross section which is in ENDF/B-V is really quite dated since it was originally performed in 1968. A new evaluation was recently done by Hale [13] using all possible two body reactions in the ^4He system. As can be seen in figure 3, the new evaluation is in much better agreement with the newer measurements than ENDF/B-V. The measurements shown in figure 3 were not included in the ENDF/B-VI evaluation. This cross section is tabulated in table 3 along with its uncertainty. Complete covariance files are not presently available for the $^3He(n,p)$ cross section.

The carbon standard is the scattering cross section for natural carbon for energies less than 1.8 MeV. In ENDF/B-V, the evaluation was based on an R-matrix analysis for ^{12}C using natural carbon data. In the use of this standard, one had to note that two resonances in ^{13}C could cause problems since they are not included in the evaluation. A revision of the ENDF/B-V evaluation by Fu [14] was made to include the effects of these two resonances. The revision was also an R-matrix analysis based on the available data. This evaluation was accepted for use in ENDF/B-VI. In figure 4, the ENDF/B-VI carbon total cross section evaluation is shown compared with the data of Heaton [15] and the ENDF/B-V evaluation near the 152.9 keV resonance in ^{13}C. The Heaton data were obtained with a natural carbon sample. In figure 5, the ENDF/B-VI evaluation is compared with that of ENDF/B-V near the 1.736 MeV resonance. Though this structure is often referred to as one resonance, it is actually composed of a narrow d-wave resonance superimposed almost directly over a broader s-wave minimum. The carbon total elastic cross section, its uncertainty and the Legendre coefficients are given in tables 4 and 5. The changes between ENDF/B-V and ENDF/B-VI are small enough so that the covariance files from version V will be used for version VI.

4. GLOBAL EVALUATION PROCEDURE

Most of the standards and other important cross sections were evaluated by combining the results of a simultaneous evaluation and R-matrix analyses. An energy grid was defined for the evaluation which is the same for all cross sections involved in the evaluation and the fitting parameters are the values of the cross sections for these grid points.

It was decided that the generalized least squares program GMA, which is described in more detail in reference [1] would be used for the simultaneous evaluation. A related type of analysis had been used successfully for the evaluation of the $^{235}U(n,f)$ cross section for ENDF/B-V. For the GMA evaluation, the cross sections which were evaluated are $^{6}Li(n,t)$, $^{6}Li(n,n)$, $^{10}B(n,\alpha_0)$, $^{10}B(n,\alpha_1)$, $^{10}B(n,n)$, Au(n,γ), $^{235}U(n,f)$, $^{238}U(n,f)$, $^{238}U(n,\gamma)$, and $^{239}Pu(n,f)$. The input data for this evaluation was composed of two independent subsets. One of these subsets is a large data base of pointwise measurements assembled at Argonne National Laboratory. The references for this data base are given in appendix A. This data base includes both shape and absolute cross section measurements and their ratios. Also total cross section measurements for ^{6}Li and ^{10}B are contained in the data base since the scattering and reaction data are interrelated in these measurements. Measurements of the ^{235}U and ^{239}Pu fission cross sections in the ^{252}Cf spontaneous fission neutron spectrum were also included in the data base. These data can be obtained with high accuracy and are only weakly dependent on the uncertainties in the ^{252}Cf spontaneous neutron fission spectrum. These data can have an important effect on the normalization of the evaluated cross sections. A considerable effort was directed at examining the various experiments looking for corrections, etc. which were not fully documented in the published papers. Ratio measurements other than those to the hydrogen standard which have been converted to cross section values were reinstated to the originally measured quantities. Measurements relative to hydrogen were converted using the ENDF/B-VI values for the total elastic cross section. Perhaps the most difficult part of this work was the determination for each experiment of the uncertainties and correlations in that experiment and correlations with other experiments. This information was used to form covariance matrices for the measurements so that a full covariance analysis could be performed for the evaluation. Rather than include the entire data base for the thermal constants, the results of the recent evaluation by Axton [16], with the associated variance-covariance data were used as the second independent data input subset to the GMA analysis. It should be noted that the Au(n,γ) and $^{10}B(n,\alpha)$ cross sections at thermal were treated as constants in the Axton evaluation though they are parameters in the present evaluation procedure.

The evaluations of the ^{6}Li and ^{10}B cross sections for both versions IV and V of ENDF/B have been produced with the R-matrix coupled channel program EDA [17]. It was decided that this program would be a suitable R-matrix code for the present evaluation process if all experiments which are correlated and all ratio measurements (except those to the hydrogen standard) were put into the data base used for the simultaneous evaluation. The R-matrix fits were done at Los Alamos National Laboratory. In these analyses the experimental data are used as measured with weighting normally based on the quoted uncertainties. It is assumed that no correlations other than the overall normalization are present among the data from a particular experiment. The code uses automated search routines to minimize chi-square of the fits to the input data. In addition to the R-matrix parameters, derivatives of fitted cross sections with respect to these parameters and the covariance matrix are available as output. Following the fitting process, the cross sections were calculated for the same energy grid as is used for the simultaneous evaluation to permit the combination of the results. The parameters deduced from these analyses provide neutron cross sections well beyond the standards region. The ^{6}Li+n and ^{10}B+n analyses were each done separately with this code. For the ^{7}Li system the data base includes ^{6}Li total, ^{6}Li(n,n) integrated, ^{6}Li(n,n) differential, ^{6}Li(n,n) polarization, ^{6}Li(n,t) integrated, ^{6}Li(n,t) differential, ^{4}He(t,t) differential, and ^{4}He(t,t) polarization data. For the ^{11}B system the data base includes ^{10}B total, ^{10}B(n,n) integrated, ^{10}B(n,n) differential, ^{10}B(n,n) polarization, ^{10}B(n,α_0) integrated, ^{10}B(n,α_0) differential, ^{10}B(n,α_1) integrated, ^{10}B(n,α_1) differential, ^{7}Li(α,α_0) differential, ^{7}Li(α,α_1) differential, and ^{7}Li(α,n) differential data. The references for the ^{7}Li and ^{11}B data bases are given in appendix B.

Altogether more than 10,000 data points were fit with 109 R-matrix parameters and 935 pointwise cross sections and parameters. The 935 pointwise cross sections and parameters include 406 pointwise cross sections for reactions other than with boron and lithium, 370 pointwise cross sections for boron and lithium reactions, 22 thermal constants, and 137 normalization values for shape data.

A procedure for combining the simultaneous and R-matrix evaluations was determined. It is described in appendix C. It is based on the observation that the individual fitting processes described above include computation of sums that can be combined to produce the same overall output parameters as would have been obtained from a global least squares fit of all the input data in terms of R-matrix parameters for the ^{6}Li+n and ^{10}B+n systems and pointwise values for the other cross sections. A program for performing the combination was written at Oak Ridge National Laboratory. Methods were studied to handle correlations between data sets used in the R-matrix and simultaneous evaluations. It was concluded that it would not be difficult to handle correlations if only a small amount of common data were present, but it would become increasingly difficult as more common data are present in the two evaluation processes. For simplicity then, the boron and lithium experimental data were separated into two uncorrelated groups, one to be used in the R-matrix analyses, called segment 1, and the other in the simultaneous analysis, called segment 2. All ratio measurements other than those to the hydrogen standard were used in the simultaneous evaluation. Experiments which are correlated were put into the segment 2 data base. The combining procedure makes use of the variance-covariance matrices from the separate fits as well as the derivatives with respect to the evaluation parameters of the fitted values corresponding to the input data elements. The input data sets are thus taken into account in a consistent manner. The output is adjusted R-matrix parameters for the ^{6}Li+n and ^{10}B+n systems and final point cross sections for the remaining reactions. The adjusted R-matrix parameters were used to calculate the ^{6}Li+n and ^{10}B+n cross sections for ENDF/B-VI.

For the combination process, the fitting variables in the least-squares minimization were small changes in the R-matrix parameters (e_{LR} and e_{BR}) for the lithium and boron reactions and small relative cross section changes (e_p) for the heavy elements. For simplicity in writing the equations below, the Axton set is not explicitly recognized and only the lithium R-matrix fit is represented. The least-squares equation [18] can be written for this case in terms of submatrices:

$$\left[\begin{bmatrix} Q_{1L} & 0 \\ 0 & 0 \end{bmatrix} + S^t Q_2 S\right] \begin{bmatrix} \varepsilon_{LR} \\ \varepsilon_P \end{bmatrix} = S^t R_2 \quad . \tag{1}$$

The matrix Q_{1L} is the inverse of the R-matrix parameter covariance matrix for the Segment 1 lithium data.

$$R_2 = \begin{bmatrix} G_{2L}^t \\ G_{2P}^t \end{bmatrix} V_2^{-1} \eta_2 \quad , \quad S = \begin{bmatrix} S_L & 0 \\ 0 & 1 \end{bmatrix} \quad ,$$

and

$$Q_2 = \begin{bmatrix} G_{2L}^t \\ G_{2P}^t \end{bmatrix} V_2^{-1} (G_{2L}, G_{2P}) \quad .$$

The matrix V_2 is the variance-covariance matrix of the Segment 2 input data. The elements of G_{2L} and G_{2P} are the partial derivatives of the approximation equations, corresponding to each Segment 2 reduced input datum, with respect to relative changes in the pointwise cross section parameters. The reduced data vector η_2 contains the differences between the experimental data, reduced to a fixed energy grid, and the initial estimates derived from the zero-order parameter values. Finally, the elements of S_L are the logarithmic derivatives of the pointwise interpolated cross sections for the ^{7}Li system with respect to its R-matrix parameters; these were obtained from the R-matrix equations at the Segment 1 solution point.

Equation (1) was solved for the 674 elements of e. The output covariance matrix propagating the input data uncertainties and correlations is the inverse of the matrix on the left side. The equivalents of the matrices Q_{1L}, S, Q_2, and R_2 in eq (1) were obtained from the EDA and GMA programs for their respective data segments as assumed in the formulation above. The lithium and boron results from the Segment 1 fits and the Axton output thermal parameters were used as initial estimates in the final Segment 2 iteration.

Since the R-matrix formulations were quite nonlinear for some of the parameter refinements, the final parameters for lithium and boron were obtained from R-matrix fits to the cross sections obtained from the combination output.

Due to the difficulties associated with the transfer of large data files among the participants of the evaluation process, an effort was made to select initial estimates of the output variables which were sufficiently close to the output values so that a single iteration would suffice. In a trial analysis, a non-optimum partitioning of the boron data base between the R-matrix and simultaneous analyses was performed [19]. There were significant differences between the R-matrix and simultaneous evaluations. Though the results of the combination process should be independent of the partitioning of the input data, the desire to run only one iteration led to a more favorable partitioning.

In one case, the combination of the R-matrix and simultaneous pointwise evaluations of the independent data segments was iterated. This was done for an early data set where a second iteration was performed by obtaining the R-matrix parameter covariance matrix for the parameters corresponding to the output of the first iteration. This matrix had one negative eigenvalue, but the iteration could proceed formally. The resulting cross section changes were substantially smaller than those in the first iteration, but the output data covariance matrix had many negative eigenvalues and the results were considered unusable. The underlying problem was that the R-matrix fits were quite nonlinear in some parameters over intervals comparable to the iteration increments, though the development of the combination equations and the tentatively quoted output uncertainties assume linearity. The one-pass combination results were accepted regardless of this inconsistency because they seemed to represent the input data.

5. UNCERTAINTIES

Few experimenters document their known experimental uncertainties with enough detail to allow an evaluator to fully determine the required input data covariance matrix [20],[21]. Also since some experimental uncertainties are unrecognized or underestimated, inconsistencies among input data commonly occur. Under these circumstances, there may be inconsistencies in the output from the evaluation process. Also the output uncertainties may be too small. To account in some sense for unknown systematic errors, separate factors of the square root of χ^2/(degree of freedom) were determined for the simultaneous evaluation, for the R-matrix evaluation of lithium and for the R-matrix evaluation of boron. Each of these factors were applied, to the analyses where they were determined, as a scale factor to increase the output uncertainties. Very unusual results can be obtained with discrepant correlated data. To remove problems associated with these discrepancies, data greater than three standard deviations away from the output results were down weighted in the GMA analysis. This had the effect of reducing the χ^2/(degree of freedom) quantity referred to above to essentially 1. This process was not performed for the R-matrix evaluations where this quantity was 4.00 for the lithium analysis and 1.25 for the boron analysis. The parameter covariance matrices from the EDA analyses were scaled by the χ^2/(degree of freedom) factors. Even after this increase in the uncertainties, they still appear small to the reviewers of the evaluation [22]. Concerns have been expressed that users would not use these uncertainties but instead would arbitrarily increase them to what they considered a more acceptable level. A strong statement has been made that the Standards Subcommittee should provide such expanded uncertainties since they have had the closest contact with the data base and could make better estimates of more "acceptable" values. Such expanded uncertainties were provided. These uncertainties [23] are qualitative estimates such that if a modern day experiment were performed today on a given standard using the best techniques, most of these results would be expected to fall within these expanded uncertainties. They were intended to take into account data inconsistencies and concerns about R-matrix parameters. These estimated (expanded) uncertainties are given in table 6. It is not assumed that these uncertainties are totally correlated within the energy ranges given. These expanded uncertainties will be put in file 1 for each standard. Complete covariance files for the combination output are available but are very large. Based on the number of experimental data points it is clear that the covariance matrix is much larger than necessary. Work is now being done to collapse the matrix. Preliminary results for ^{6}Li(n,t), ^{10}B(n,α), Au(n,γ), ^{235}U(n,f), ^{238}U(n,γ), ^{238}U(n,f), and ^{239}Pu(n,f) are available as part of the International Reactor Dosimetry File [24].

6. COMPARISONS OF THE R-MATRIX, SIMULTANEOUS EVALUATION, AND COMBINATION RESULTS

In figures 6-20, the combination output is compared with the Segment 1 and Segment 2 results for each of the cross sections in the evaluation process. In figures 6-11, the combination output and Segment 2 results are compared to the Segment 1 (R-matrix) results. For these cross sections the partitionings of the data bases were done to provide the highest quality data for the R-matrix analyses so that convergence could be more easily obtained for that work. Thus the Segment 2 (simultaneous evaluation) results are poorly defined, have large uncertainties and have a much smaller effect on the combination output than the Segment 1 results. In figures 12-20, the combination output and Segment 2 results are compared with the initial estimates for the final iteration of the Segment 2 data in GMA. The Segment 1 results impact on the combination results for these cases only through ratio measurements to the ^{6}Li and ^{10}B standard cross sections. Including the Segment 1 data reduces the uncertainties and causes some changes in the combination results. In figures 12-20, the results shown are those prior to smoothing.

7. SMOOTHING/FITTING OF OUTPUT DATA

The lithium and boron cross sections which are obtained from this evaluation process are smooth since they are calculated from R-matrix parameters. However the results obtained for the heavy element standards in some cases showed fluctuations that seemed unreasonable based on expectations from the theory of average cross sections. Significant fluctuations can occur, for example, if not all the output points in a neighborhood reflect the same input data sets and if unrecognized uncertainties existed. Possible methods for fitting the capture and fission cross sections were considered. However instead such methods were used only to provide insight on how to do the smoothing. In figures 21-25, the differences between the smoothed results and the original combination output are shown. The uncertainties shown are the output values from the combination procedure.

8. RESULTS

The cross sections and uncertainties for the standards which were obtained from the combination of the simultaneous and R-matrix analyses are given here in tables 7-11. The results are given only for the energy regions over which these cross sections are recommended as standards by the CSEWG standards subcommittee, but the user must decide where to use the standard based on the uncertainty information available. There may be situations where a standard can be used over a much larger energy interval. In some cases, data points have been added to the original evaluated output to improve the definition of the cross section shape and to ease in interpolation of the data. For the ^{10}B and ^{6}Li cross sections shown in tables 7-9, the results are point values at all energies. Though in some cases these cross sections were obtained in part with measurements having moderate neutron energy resolution, its effect on the evaluation is probably not significant. For the Au(n,γ) and ^{235}U(n,f) cross sections shown in tables 10 and 11, the results are low resolution smooth point values.

In tables 7-11 not all of the cross section results from the evaluation process are given. Non-standard cross sections, data beyond the standards region for a standard and most of the thermal constants are not shown. For completeness the entire output listing from the cross section evaluation process, as it was when it first became available, is given in appendix D. The 9.4 eV value for the ^{235}U fission cross section is the integral cross section over the range from 7.8 to 11 eV. For the heavy elements the cross sections shown for energies from 0.15 through 15 keV represent decimal interval average values labeled at the center energies for intervals starting at 0.1 to 0.2 keV and ending with the interval 10 to 20 keV.

The standard cross sections are shown in figures 26-30 as ratios of the ENDF/B-VI to ENDF/B-V values. For the $^{6}Li(n,t)$ cross section, shown in figure 26, the changes are relatively small below 100 keV which was the recommended maximum energy for use of this cross section as a standard for ENDF/B-V. The changes are however larger than expected based on the uncertainties of the two evaluations and the number of data sets in common for the evaluations. The inclusion of the simultaneous evaluation results in the ENDF/B-VI evaluation has only a small effect in this lower energy region. The structure near 250 keV may be largely due to effects associated with inconsistencies in both the energy scales and the width of that resonance. This cross section is recommended for use as a standard below 1 MeV. The comparisons of the ENDF/B-V and ENDF/B-VI boron cross sections are shown in figures 27 and 28. The standards are the $^{10}B(n,\alpha_1)$ and the $^{10}B(n,\alpha)$ cross sections and they are recommended as standards below 250 keV. Similar results are seen for these cross sections as for the $^{6}Li(n,t)$ reaction in the low energy region. The changes are relatively small but larger than anticipated based on the uncertainties of the evaluations. At the higher neutron energies, significant differences are observed. The $^{10}B(n,\alpha_0)$ measurements of Olson and Kavanagh [25] and Sealock [26] have had important effects on the ENDF/B-VI evaluation at these energies. Recent measurements of the branching ratio [27] are 10 to 30% lower than ENDF/B-VI in the 100 to 600 keV energy region. However new determinations [28] of the $^{10}B(n,\alpha_1)$ cross section agree well with ENDF/B-VI to above 1 MeV. The ENDF/B-VI Au(n,γ) cross section which is shown in figure 29 is similar to that of ENDF/B-V. However it is somewhat lower at the lower and higher neutron energies. Below about 300 keV, there is structure in the present evaluation which is a result of including measurements [29],[30] taken with high enough resolution to see effects due to competition with inelastic scattering and fluctuations in the neutron widths and spacings of the compound nucleus levels. This cross section is recommended as a standard from 200 keV to 2.5 MeV. The $^{235}U(n,f)$ cross section is shown in figure 30. This cross section is a standard from 150 keV to 20 MeV. The ENDF/B-VI cross section is lower than that of ENDF/B-V by 1-2% below 3 MeV. New measurements [31] and the inclusion of the ^{252}Cf spectrum averaged $^{235}U(n,f)$ data, particularly [32] and [33], had a significant impact on lowering these values in the ENDF/B-VI evaluation. Above about 15 MeV the data base is rather poor so appreciable differences occur with changes in evaluation method. Recent measurements [34]-[35] indicate differences as large as 5% with ENDF/B-V and ENDF/B-VI.

In figures 31-33, the ratios of the results of this evaluation process to those of ENDF/B-V [37] are given for the $^{238}U(n,f)$, $^{238}U(n,\gamma)$, and $^{239}Pu(n,f)$ reactions.

A new evaluation of the ^{252}Cf spontaneous fission neutron spectrum has been completed by Mannhart [38]. This evaluation made use of recent time-of-flight measurements of this spectrum. From the information that was available, a complete covariance matrix was generated for each experiment. The data were combined by generalized least-squares techniques. The evaluation was carried out with 70 energy grid points between 25 keV and 19.8 MeV. The individual experimental data were extrapolated to these grid points by using the shape of a Maxwellian distribution appropriate for each experiment. The evaluation gave a value of χ^2 per degree of freedom of approximately unity and indicated no incompatibility between the experiments. The resulting relative uncertainty of the evaluation is smaller than 2% between 180 keV and 9.3 MeV. A weighted spline interpolation between the discrete data points was used to generate a continuous shape of the evaluated neutron spectrum. The result of the evaluation has been compared with available theoretical descriptions of the ^{252}Cf neutron spectrum. None of the existing theories is compatible with the evaluation over the whole energy range. The values at the energy points chosen for the ENDF/B-VI evaluation are given in Table 12. The uncertainties which were obtained from the evaluation process are shown in Table 13. The complete covariance matrix is available but large so it is not shown in this document.

9. CONCLUSIONS AND RECOMMENDATIONS

The work described in this report has produced improved cross section standards which are self consistent. The methods employed are the most sophisticated that have been used since the ENDF/B system was introduced. There are however still a number of improvements to the evaluation process which should be considered before the next evaluation of the standards so that delays can be avoided.

There are some inconsistencies in the present work which should be handled properly. In the evaluation [16] of the thermal constants the Au(n,γ) and ^{10}B(n,α) cross sections which are standards for these measurements were treated as constants. The thermal constants from this evaluation were imported into the present evaluation where these standards were treated as variables. The effect of this transgression is small since the uncertainties on these cross sections at thermal are small.

The H(n,n) cross section was assumed to be so well known that the remaining standards would have little impact on it in this evaluation. Concerns about the hydrogen standard angular distribution for energies above a few MeV suggest that it may be valuable to include this cross section in the comprehensive combining process. Also the ^{3}He(n,p) and C(n,n) cross sections were not used in the combining process since few ratio measurements to these standards exist. For completeness, inclusion of the ^{3}He(n,p) data should be considered in the next evaluation.

Because of the success in using nuclear models in the evaluation of the ^{238}U(n,γ) cross sections [39], including such models in future standards evaluations of capture cross sections should be encouraged. More work should be done on models which could be used to fit fission data. This could possibly improve the quality of the evaluation and reduce or eliminate the need for smoothing of the output results.

Approaches should be considered for doing the evaluation so that the entire process could be done on one computer system. For the present formalism, the R-matrix and simultaneous evaluations and the procedure for combining them could then be iterated more easily.

An important problem encountered with the present evaluation was the uncertainties in the standard cross sections. Improved methods to increase the uncertainties based on the spread in the input values should be investigated. This effort could be eliminated if discrepancies in the data base could be resolved.

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Table 1. H(n,n) Cross Section
ENDF/B-VI, MAT 0125

Recommended as a standard from about 10 eV to 20 MeV

Log-log interpolation

Cross Section Values

E(ev)	σ(b)	E(eV)	σ(b)	E(ev)	σ(b)
1.000000-5	2.047800+1	2.530000-2	2.047800+1	1.000000+2	2.046400+1
1.000000+3	2.034400+1	5.000000+3	1.982700+1	1.000000+4	1.922200+1
2.000000+4	1.813000+1	3.000000+4	1.717300+1	4.000000+4	1.632500+1
5.000000+4	1.557100+1	6.000000+4	1.489400+1	7.000000+4	1.428300+1
8.000000+4	1.373000+1	9.000000+4	1.322600+1	1.000000+5	1.276500+1
1.200000+5	1.195100+1	1.400000+5	1.125500+1	1.600000+5	1.065200+1
1.800000+5	1.012500+1	2.000000+5	9.660900+0	2.200000+5	9.247800+0
2.400000+5	8.878000+0	2.600000+5	8.544700+0	2.800000+5	8.242800+0
3.000000+5	7.967800+0	3.200000+5	7.716200+0	3.400000+5	7.485000+0
3.600000+5	7.271800+0	3.800000+5	7.074500+0	4.000000+5	6.891200+0
4.200000+5	6.720500+0	4.400000+5	6.561100+0	4.600000+5	6.411700+0
4.800000+5	6.271500+0	5.000000+5	6.139600+0	5.500000+5	5.841200+0
6.000000+5	5.580500+0	6.500000+5	5.350400+0	7.000000+5	5.145300+0
7.500000+5	4.961100+0	8.000000+5	4.794500+0	8.500000+5	4.642900+0
9.000000+5	4.504200+0	9.500000+5	4.376500+0	1.000000+6	4.258600+0
1.100000+6	4.047100+0	1.200000+6	3.862500+0	1.300000+6	3.699200+0
1.400000+6	3.553300+0	1.500000+6	3.421900+0	1.600000+6	3.302500+0
1.700000+6	3.193400+0	1.800000+6	3.093100+0	1.900000+6	3.000300+0
2.000000+6	2.914200+0	2.100000+6	2.833900+0	2.200000+6	2.758800+0
2.300000+6	2.688300+0	2.400000+6	2.621900+0	2.500000+6	2.559200+0
2.600000+6	2.499900+0	2.700000+6	2.443600+0	2.800000+6	2.390200+0
2.900000+6	2.339300+0	3.000000+6	2.290700+0	3.200000+6	2.199900+0
3.400000+6	2.116600+0	3.600000+6	2.039800+0	3.800000+6	1.968600+0
4.000000+6	1.902400+0	4.200000+6	1.840600+0	4.400000+6	1.782800+0
4.600000+6	1.728600+0	4.800000+6	1.677500+0	5.000000+6	1.629400+0
5.500000+6	1.520200+0	6.000000+6	1.424400+0	6.500000+6	1.339600+0
7.000000+6	1.263900+0	7.500000+6	1.195900+0	8.000000+6	1.134500+0
8.500000+6	1.078600+0	9.000000+6	1.027700+0	9.500000+6	9.810600-1
1.000000+7	9.381600-1	1.050000+7	8.985700-1	1.100000+7	8.619400-1
1.150000+7	8.279300-1	1.200000+7	7.962800-1	1.250000+7	7.667600-1
1.300000+7	7.391500-1	1.350000+7	7.132900-1	1.400000+7	6.890000-1
1.450000+7	6.661500-1	1.500000+7	6.446200-1	1.550000+7	6.242900-1
1.600000+7	6.050700-1	1.700000+7	5.696100-1	1.800000+7	5.376400-1
1.900000+7	5.086600-1	2.000000+7	4.822700-1		

H(n,n) Total Cross Section Uncertainty
(at each point in the energy range)

Energy Range (keV)	Uncertainty (%)
(1.E-08) - 20000	0.2

Table 2. H(n,n) Center of Mass Legendre Coefficients

$$\text{DSIGMA} = \frac{\text{SIGMA}}{4\pi} \sum_{\ell=0}^{N} (2\ell+1) A_\ell P_\ell \quad , \quad A_0 = 1$$

Linear-linear interpolation

Legendre Coefficients

E(ev)	A_1	A_2	A_3	A_4	A_5	A_6
1.000000+3	-1.919300-6	2.09480-11	6.46600-16	1.75260-15	1.59600-15	6.75230-16
5.000000+3	-9.565900-6	5.32530-10	-2.28790-14	2.18920-15	2.91700-15	4.33020-16
1.000000+4	-1.905700-5	2.174000-9	-2.03730-13	1.04840-15	2.98240-15	2.67990-16
2.000000+4	-3.781800-5	9.040700-9	-1.63080-12	-7.69510-16	3.35790-15	3.78840-16
3.000000+4	-5.629700-5	2.109700-8	-5.44050-12	-1.55260-15	5.40630-15	-1.19990-15
4.000000+4	-7.450300-5	3.881400-8	-1.27190-11	-3.98810-16	7.24070-15	-1.26220-15
5.000000+4	-9.244800-5	6.264000-8	-2.44990-11	2.53870-15	5.79960-15	-3.97000-15
6.000000+4	-1.101400-4	9.300100-8	-4.17500-11	1.06790-14	7.25530-15	-4.26570-15
7.000000+4	-1.276000-4	1.303000-7	-6.53900-11	2.52870-14	9.05720-15	-3.84690-15
8.000000+4	-1.448200-4	1.749200-7	-9.62850-11	4.81870-14	9.60700-15	-4.50220-15
9.000000+4	-1.618200-4	2.272300-7	-1.35250-10	8.48110-14	1.11240-14	-5.19310-15
1.000000+5	-1.786000-4	2.875800-7	-1.83080-10	1.38780-13	1.13670-14	-6.18790-15
1.200000+5	-2.115600-4	4.336800-7	-3.08190-10	3.20310-13	1.40520-14	-6.60940-15
1.400000+5	-2.437500-4	6.156800-7	-4.77060-10	6.38410-13	1.68590-14	-8.54390-15
1.600000+5	-2.752400-4	8.358100-7	-6.94620-10	1.15700-12	1.70030-14	-9.83310-15
1.800000+5	-3.060700-4	1.096100-6	-9.65340-10	1.95720-12	1.78870-14	-1.34820-14
2.000000+5	-3.362900-4	1.398300-6	-1.293300-9	3.13180-12	2.07690-14	-1.23530-14
2.200000+5	-3.659500-4	1.744200-6	-1.682200-9	4.79110-12	2.32330-14	-1.40190-14
2.400000+5	-3.950900-4	2.135200-6	-2.135600-9	7.06500-12	2.61440-14	-1.49890-14
2.600000+5	-4.237300-4	2.572800-6	-2.656600-9	1.01010-11	2.87370-14	-1.63780-14
2.800000+5	-4.519300-4	3.058100-6	-3.248300-9	1.40680-11	3.29280-14	-1.76030-14
3.000000+5	-4.797000-4	3.592200-6	-3.913500-9	1.91510-11	3.93490-14	-1.76720-14
3.200000+5	-5.070800-4	4.176300-6	-4.655000-9	2.55590-11	4.33610-14	-1.99170-14
3.400000+5	-5.341100-4	4.811200-6	-5.475300-9	3.35230-11	4.96140-14	-2.11050-14
3.600000+5	-5.607900-4	5.497700-6	-6.377000-9	4.32970-11	5.70690-14	-2.19600-14
3.800000+5	-5.871600-4	6.236700-6	-7.362500-9	5.51500-11	6.51150-14	-2.37860-14
4.000000+5	-6.132400-4	7.028700-6	-8.434300-9	6.93850-11	7.69690-14	-2.27990-14
4.200000+5	-6.390500-4	7.874500-6	-9.594700-9	8.63200-11	9.00210-14	-2.41450-14
4.400000+5	-6.646000-4	8.774600-6	-1.084600-8	1.06300-10	1.06280-13	-2.59090-14
4.600000+5	-6.899300-4	9.729400-6	-1.219100-8	1.29700-10	1.25220-13	-2.65130-14
4.800000+5	-7.150300-4	1.073900-5	-1.363200-8	1.56910-10	1.47950-13	-2.69690-14
5.000000+5	-7.399400-4	1.180500-5	-1.517200-8	1.88350-10	1.74600-13	-2.81080-14
5.500000+5	-8.014300-4	1.471500-5	-1.946700-8	2.88460-10	2.63420-13	-3.07190-14
6.000000+5	-8.619700-4	1.797800-5	-2.443500-8	4.25570-10	3.95680-13	-3.07690-14
6.500000+5	-9.217400-4	2.159900-5	-3.011900-8	6.08420-10	5.84140-13	-3.09700-14
7.000000+5	-9.808800-4	2.557900-5	-3.656800-8	8.46840-10	8.46730-13	-2.92840-14
7.500000+5	-1.039500-3	2.991800-5	-4.383300-8	1.151800-9	1.20650-12	-2.44010-14
8.000000+5	-1.097800-3	3.461700-5	-5.197100-8	1.535300-9	1.68400-12	-1.79070-14
8.500000+5	-1.155800-3	3.967600-5	-6.104600-8	2.010600-9	2.31400-12	-4.71540-15
9.000000+5	-1.213600-3	4.509400-5	-7.112300-8	2.592200-9	3.12430-12	1.19130-14
9.500000+5	-1.271300-3	5.086800-5	-8.227600-8	3.295700-9	4.15800-12	3.94310-14
1.000000+6	-1.328900-3	5.699700-5	-9.458400-8	4.138000-9	5.45600-12	7.24780-14
1.100000+6	-1.444200-3	7.031000-5	-1.230100-7	6.312500-9	9.04730-12	1.80950-13
1.200000+6	-1.559900-3	8.501200-5	-1.571400-7	9.275900-9	1.43630-11	3.60870-13
1.300000+6	-1.676200-3	1.010700-4	-1.978300-7	1.320900-8	2.19700-11	6.50090-13
1.400000+6	-1.793400-3	1.184700-4	-2.460200-7	1.831400-8	3.25500-11	1.09330-12

Table 2 (Continued)

E(ev)	A_1	A_2	A_3	A_4	A_5	A_6
1.500000+6	-1.911600-3	1.371700-4	-3.027600-7	2.481500-8	4.69040-11	1.75150-12
1.600000+6	-2.030900-3	1.571400-4	-3.691900-7	3.295700-8	6.59660-11	2.70220-12
1.700000+6	-2.151500-3	1.783500-4	-4.465600-7	4.300700-8	9.08120-11	4.04220-12
1.800000+6	-2.273300-3	2.007700-4	-5.362000-7	5.525600-8	1.22660-10	5.89340-12
1.900000+6	-2.396400-3	2.243500-4	-6.395600-7	7.001700-8	1.62880-10	8.40060-12
2.000000+6	-2.520800-3	2.490800-4	-7.581600-7	8.762500-8	2.13020-10	1.17450-11
2.100000+6	-2.646500-3	2.749000-4	-8.936500-7	1.084400-7	2.74770-10	1.61410-11
2.200000+6	-2.773600-3	3.017900-4	-1.047700-6	1.328400-7	3.50000-10	2.18450-11
2.300000+6	-2.902000-3	3.297100-4	-1.222300-6	1.612300-7	4.40750-10	2.91530-11
2.400000+6	-3.031600-3	3.586300-4	-1.419100-6	1.940300-7	5.49240-10	3.84210-11
2.500000+6	-3.162500-3	3.885100-4	-1.640200-6	2.317100-7	6.77840-10	5.00540-11
2.600000+6	-3.294600-3	4.193200-4	-1.887700-6	2.747200-7	8.29130-10	6.45250-11
2.700000+6	-3.427900-3	4.510300-4	-2.163800-6	3.235600-7	1.005800-9	8.23760-11
2.800000+6	-3.562300-3	4.835900-4	-2.470700-6	3.787600-7	1.210800-9	1.04220-10
2.900000+6	-3.697800-3	5.169900-4	-2.810600-6	4.408400-7	1.447200-9	1.30760-10
3.000000+6	-3.834300-3	5.511800-4	-3.186100-6	5.103800-7	1.718200-9	1.62790-10
3.200000+6	-4.110200-3	6.218100-4	-4.053500-6	6.741500-7	2.377500-9	2.46990-10
3.400000+6	-4.389700-3	6.952500-4	-5.093600-6	8.749900-7	3.217400-9	3.65310-10
3.600000+6	-4.672300-3	7.712300-4	-6.327900-6	1.118100-6	4.268100-9	5.28250-10
3.800000+6	-4.957600-3	8.495200-4	-7.779400-6	1.409100-6	5.561500-9	7.48650-10
4.000000+6	-5.245200-3	9.299000-4	-9.471700-6	1.753900-6	7.129800-9	1.042000-9
4.200000+6	-5.534600-3	1.012100-3	-1.142900-5	2.158400-6	9.005600-9	1.426900-9
4.400000+6	-5.825500-3	1.096000-3	-1.367700-5	2.629300-6	1.122000-8	1.925200-9
4.600000+6	-6.117400-3	1.181300-3	-1.624200-5	3.173000-6	1.380400-8	2.562700-9
4.800000+6	-6.409900-3	1.267800-3	-1.915000-5	3.796500-6	1.678300-8	3.369400-9
5.000000+6	-6.702800-3	1.355300-3	-2.242900-5	4.506800-6	2.017900-8	4.380000-9
5.500000+6	-7.433800-3	1.577500-3	-3.242800-5	6.710700-6	3.059300-8	8.074200-9
6.000000+6	-8.158900-3	1.802300-3	-4.535600-5	9.620600-6	4.373800-8	1.409400-8
6.500000+6	-8.873400-3	2.027300-3	-6.165800-5	1.335800-5	5.919400-8	2.349800-8
7.000000+6	-9.573000-3	2.250400-3	-8.177900-5	1.804400-5	7.587300-8	3.766600-8
7.500000+6	-1.025400-2	2.469800-3	-1.061500-4	2.380200-5	9.178600-8	5.835400-8
8.000000+6	-1.091300-2	2.683900-3	-1.352000-4	3.075000-5	1.037700-7	8.775100-8
8.500000+6	-1.154700-2	2.891400-3	-1.693000-4	3.899900-5	1.071600-7	1.285300-7
9.000000+6	-1.215400-2	3.091200-3	-2.088200-4	4.865400-5	9.550800-8	1.838900-7
9.500000+6	-1.273100-2	3.282500-3	-2.540700-4	5.980900-5	6.016700-8	2.576000-7
1.000000+7	-1.327700-2	3.464700-3	-3.053300-4	7.254700-5	-1.005200-8	3.540700-7
1.050000+7	-1.379100-2	3.637400-3	-3.628100-4	8.693800-5	-1.293000-7	4.782900-7
1.100000+7	-1.427100-2	3.800500-3	-4.266800-4	1.030300-4	-3.150700-7	6.359500-7
1.150000+7	-1.471700-2	3.953900-3	-4.970100-4	1.208800-4	-5.886800-7	8.333400-7
1.200000+7	-1.512700-2	4.098000-3	-5.738300-4	1.404900-4	-9.756100-7	1.077400-6
1.250000+7	-1.550200-2	4.233300-3	-6.570900-4	1.618900-4	-1.506000-6	1.375600-6
1.300000+7	-1.584100-2	4.360400-3	-7.466400-4	1.850500-4	-2.215000-6	1.735800-6
1.350000+7	-1.614500-2	4.480300-3	-8.422500-4	2.099600-4	-3.143100-6	2.166600-6
1.400000+7	-1.641200-2	4.594000-3	-9.436200-4	2.365900-4	-4.336800-6	2.676400-6
1.450000+7	-1.664400-2	4.703100-3	-1.050300-3	2.648900-4	-5.848400-6	3.273900-6
1.500000+7	-1.684100-2	4.809100-3	-1.161800-3	2.948100-4	-7.736700-6	3.967600-6
1.550000+7	-1.700200-2	4.913900-3	-1.277500-3	3.262900-4	-1.006700-5	4.765600-6
1.600000+7	-1.713000-2	5.019600-3	-1.396600-3	3.592600-4	-1.291100-5	5.675100-6
1.700000+7	-1.728200-2	5.244000-3	-1.641500-3	4.295000-4	-2.046100-5	7.851600-6
1.800000+7	-1.730400-2	5.505700-3	-1.888000-3	5.052000-4	-3.109100-5	1.052300-5
1.900000+7	-1.719900-2	5.833900-3	-2.125200-3	5.864200-4	-4.559100-5	1.367200-5
2.000000+7	-1.697400-2	6.264600-3	-2.339400-3	6.737900-4	-6.481300-5	1.721100-5

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

ENDF/B-VI, MAT 0225

Recommended as a standard below 50 keV

Log-log interpolation

Cross Section Values

E (eV)	σ(b)
1.0000E-05	2.6745E+05
2.5300E-02	5.3160E+03
1.0000E-01	2.6733E+03
1.0000E+00	8.4453E+02
1.0000E+01	2.6622E+02
1.0000E+02	8.3357E+01
2.0000E+02	5.8590E+01
4.0000E+02	4.1081E+01
6.0000E+02	3.3326E+01
8.0000E+02	2.8705E+01
1.0000E+03	2.5552E+01
2.0000E+03	1.7733E+01
4.0000E+03	1.2216E+01
6.0000E+03	9.7779E+00
8.0000E+03	8.3288E+00
1.0000E+04	7.3426E+00
1.5000E+04	5.8163E+00
2.0000E+04	4.9128E+00
2.5000E+04	4.3006E+00
3.0000E+04	3.8520E+00
3.5000E+04	3.5058E+00
4.0000E+04	3.2288E+00
4.5000E+04	3.0012E+00
5.0000E+04	2.8100E+00

$^{3}He(n,p)$ Cross Section Uncertainty
(at each point in the energy range)

Energy Range (keV)	Uncertainty (%)
(1.E-08) - 0.1	0.3
0.1 - 1	0.7
1 - 10	2.0
10 - 50	5.0

Table 4. C(n,n) Cross Section
ENDF/B-VI, MAT 600

Recommended as a standard below 1.8 MeV

Linear-linear interpolation

Cross Section Values

E(ev)	σ(b)	E(eV)	σ(b)	E(ev)	σ(b)
1.000000-5	4.739200+0	2.530000-2	4.739200+0	5.000000+3	4.716100+0
1.000000+4	4.699100+0	1.500000+4	4.682100+0	2.000000+4	4.665300+0
2.500000+4	4.648600+0	3.000000+4	4.631900+0	3.500000+4	4.615400+0
4.000000+4	4.598900+0	4.500000+4	4.582500+0	5.000000+4	4.566200+0
7.500000+4	4.486200+0	1.000000+5	4.408400+0	1.250000+5	4.330100+0
1.300000+5	4.314800+0	1.350000+5	4.301200+0	1.400000+5	4.289100+0
1.425000+5	4.284800+0	1.450000+5	4.283800+0	1.475000+5	4.292400+0
1.487500+5	4.307000+0	1.500000+5	4.340000+0	1.510000+5	4.393700+0
1.520000+5	4.476500+0	1.529000+5	4.520600+0	1.540000+5	4.454500+0
1.550000+5	4.373100+0	1.560000+5	4.321000+0	1.580000+5	4.271000+0
1.600000+5	4.248400+0	1.612500+5	4.239300+0	1.625000+5	4.232000+0
1.650000+5	4.220400+0	1.700000+5	4.202100+0	1.750000+5	4.186100+0
1.800000+5	4.171200+0	2.000000+5	4.115900+0	2.250000+5	4.048400+0
2.500000+5	3.982800+0	2.750000+5	3.918200+0	3.000000+5	3.855100+0
3.250000+5	3.793700+0	3.500000+5	3.733800+0	3.750000+5	3.675300+0
4.000000+5	3.618200+0	4.250000+5	3.562600+0	4.500000+5	3.508200+0
4.750000+5	3.455100+0	5.000000+5	3.403300+0	5.250000+5	3.352700+0
5.500000+5	3.303200+0	5.750000+5	3.254900+0	6.000000+5	3.207600+0
6.250000+5	3.161500+0	6.500000+5	3.116300+0	6.750000+5	3.072200+0
7.000000+5	3.029000+0	7.250000+5	2.986800+0	7.500000+5	2.945400+0
7.750000+5	2.905000+0	8.000000+5	2.865400+0	8.250000+5	2.826700+0
8.500000+5	2.788800+0	8.750000+5	2.751700+0	9.000000+5	2.715400+0
9.250000+5	2.679800+0	9.500000+5	2.645000+0	9.750000+5	2.610800+0
1.000000+6	2.577400+0	1.025000+6	2.544600+0	1.050000+6	2.512500+0
1.075000+6	2.481100+0	1.100000+6	2.450300+0	1.125000+6	2.420100+0
1.150000+6	2.390500+0	1.175000+6	2.361500+0	1.200000+6	2.333100+0
1.225000+6	2.305200+0	1.250000+6	2.277900+0	1.275000+6	2.251100+0
1.300000+6	2.224900+0	1.325000+6	2.199100+0	1.350000+6	2.173900+0
1.375000+6	2.149200+0	1.400000+6	2.125000+0	1.425000+6	2.101200+0
1.450000+6	2.078000+0	1.475000+6	2.055200+0	1.500000+6	2.032800+0
1.525000+6	2.010500+0	1.550000+6	1.988800+0	1.575000+6	1.967400+0
1.600000+6	1.946500+0	1.625000+6	1.925900+0	1.650000+6	1.905900+0
1.675000+6	1.885800+0	1.680000+6	1.881900+0	1.700000+6	1.865500+0
1.710000+6	1.857400+0	1.715000+6	1.853700+0	1.720000+6	1.850700+0
1.725000+6	1.850100+0	1.730000+6	1.854700+0	1.732000+6	1.858300+0
1.734000+6	1.861600+0	1.736000+6	1.862500+0	1.738000+6	1.859800+0
1.740000+6	1.854400+0	1.745000+6	1.840800+0	1.750000+6	1.832700+0
1.755000+6	1.828000+0	1.760000+6	1.824400+0	1.770000+6	1.818000+0
1.775000+6	1.814700+0	1.780000+6	1.811500+0	1.790000+6	1.804800+0
1.800000+6	1.798000+0				

C(n,n) Total Elastic Scattering Cross Section Uncertainties
(at each point in the energy range)

Energy (keV)	Uncertainty (%)
1 - 500	0.46
500 - 1500	0.53
1500 - 1800	0.60

Table 5. C(n,n) Center of Mass Legendre Coefficients

$$DSIGMA = \frac{SIGMA}{4\pi} \sum_{\ell=0}^{N} (2\ell+1) A_\ell P_\ell \quad , \quad A_0 = 1$$

Linear-linear interpolation

Legendre Coefficients

E(ev)	A_1	A_2	A_3	A_4	A_5
1.000000-5	0.000000+0				
1.000000+3	1.401100-4				
5.000000+3	6.982000-4				
1.000000+4	1.372000-3				
5.000000+4	6.603000-3	7.423100-5			
1.000000+5	1.245700-2	2.801300-4			
1.100000+5	1.339600-2	3.524300-4			
1.200000+5	1.424700-2	4.297600-4			
1.300000+5	1.491100-2	5.240000-4			
1.400000+5	1.512000-2	7.038700-4			
1.450000+5	1.499300-2	9.579300-4			
1.475000+5	1.516000-2	1.225300-3			
1.500000+5	1.627200-2	1.635400-3			
1.529000+5	1.967700-2	1.983700-3			
1.550000+5	2.200800-2	1.845300-3			
1.575000+5	2.305600-2	1.505600-3			
1.600000+5	2.311400-2	1.250600-3			
1.650000+5	2.283700-2	1.013300-3			
1.700000+5	2.278700-2	9.371100-4			
1.750000+5	2.294100-2	9.177400-4			
1.800000+5	2.322000-2	9.221200-4			
1.900000+5	2.397600-2	9.601900-4			
2.000000+5	2.486800-2	1.014300-3			
3.000000+5	3.389700-2	1.946500-3	6.334400-5		
4.000000+5	4.148600-2	3.011600-3	1.281000-4		
5.000000+5	4.780200-2	4.132600-3	2.230800-4		
6.000000+5	5.314500-2	5.274000-3	3.476700-4		
7.000000+5	5.744800-2	6.394900-3	4.974000-4	-5.641100-5	
8.000000+5	6.078900-2	7.486100-3	6.650900-4	-1.013500-4	
9.000000+5	6.322600-2	8.553600-3	8.401900-4	-1.700600-4	
1.000000+6	6.479700-2	9.618100-3	1.008100-3	-2.704900-4	
1.100000+6	6.550000-2	1.070900-2	1.146800-3	-4.118800-4	
1.200000+6	6.536000-2	1.188100-2	1.234700-3	-6.050800-4	
1.300000+6	6.436300-2	1.319900-2	1.240500-3	-8.620300-4	
1.400000+6	6.247500-2	1.475200-2	1.122800-3	-1.195200-3	6.114500-5
1.500000+6	5.963500-2	1.664300-2	8.259800-4	-1.615700-3	8.828800-5
1.600000+6	5.578100-2	1.897000-2	2.492400-4	-2.126300-3	1.247900-4
1.650000+6	5.328500-2	2.039500-2	-2.552000-4	-2.416100-3	1.488000-4

Table 5 (Continued)

E(ev)	A_1	A_2	A_3	A_4	A_5
1.680000+6	5.180600-2	2.117000-2	-5.907000-4	-2.578500-3	1.632100-4
1.700000+6	5.076500-2	2.158000-2	-8.678800-4	-2.658500-3	1.728200-4
1.710000+6	4.995100-2	2.184900-2	-1.114800-3	-2.661400-3	1.789500-4
1.720000+6	4.878600-2	2.243800-2	-1.386900-3	-2.581800-3	1.850800-4
1.730000+6	4.760800-2	2.426300-2	-1.410200-3	-2.435300-3	1.912100-4
1.736000+6	4.763000-2	2.514000-2	-1.194900-3	-2.453600-3	1.948900-4
1.740000+6	4.770900-2	2.508700-2	-1.051500-3	-2.525400-3	1.973400-4
1.745000+6	4.747100-2	2.454800-2	-9.698700-4	-2.660800-3	2.004000-4
1.750000+6	4.693000-2	2.410800-2	-1.024200-3	-2.804400-3	2.034700-4
1.755000+6	4.639400-2	2.402300-2	-1.150300-3	-2.917200-3	2.065300-4
1.760000+6	4.596400-2	2.415400-2	-1.286800-3	-2.997300-3	2.096000-4
1.770000+6	4.529100-2	2.458900-2	-1.529300-3	-3.104400-3	2.157200-4
1.780000+6	4.468800-2	2.503500-2	-1.736800-3	-3.183100-3	2.218500-4
1.790000+6	4.408700-2	2.545600-2	-1.925600-3	-3.252000-3	2.279800-4
1.800000+6	4.347500-2	2.586000-2	-2.104400-3	-3.316800-3	2.341100-4

Table 6. Estimated (Expanded) Uncertainties Compared with those Obtained from the Evaluation Process

(Note the two types of uncertainties are defined differently)

$^{6}Li(n,t)$ Cross Section

Energy Range (keV)	Estimated Uncertainty (%)	Comb. Result (%)
(1.E-08) - 0.1	0.3	(0.14)
0.1 - 1	0.5	(0.14)
1 - 10	0.7	(0.14)
10 - 50	0.9	(0.16)
50 - 90	1.1	(0.25)
90 - 150	1.5	(0.33)
150 - 450	2.0	(0.29)
450 - 650	5.0	(0.32)
650 - 800	2.0	(0.36)
800 - 1000	5.0	(0.39)

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

Energy Range(keV)	Estimated Uncertainty (%)	Comb. Result (%)
(1.E-08) - 0.1	0.2	(0.16)
0.1 - 5	0.4	(0.17)
5 - 30	0.6	(0.20)
30 - 90	1.0	(0.34)
90 - 150	1.6	(0.46)
150 - 200	2.1	(0.57)
200 - 250	2.7	(0.60)

$^{10}B(n,\alpha_1)$ Cross Section

Energy Range(keV)	Estimated Uncertainty (%)	Comb. Result (%)
(1.E-08) - 0.1	0.2	(0.16)
0.1 - 5	0.4	(0.17)
5 - 30	0.6	(0.20)
30 - 90	1.0	(0.34)
90 - 150	1.5	(0.48)
150 - 200	2.0	(0.58)
200 - 250	2.5	(0.62)

Table 6 (Continued)

Au(n,γ) Cross Section

Energy (keV)	Estimated Uncertainty (%)	Comb. Result (%)
2.53E-05	0.14	(0.14)
200 - 500	3.0	(1.31)
500 - 1000	3.5	(2.1)
1000 - 2500	4.5	(2.0)

$^{235}U(n,f)$ Cross Section

Energy (keV)	Estimated Uncertainty (%)	Comb. Result (%)
2.53E-05	0.2	(0.19)
150 - 600	1.5	(0.67)
600 - 1000	1.6	(0.60)
1000 - 3000	1.8	(0.55)
3000 - 6000	2.3	(0.69)
6000 - 10000	2.2	(0.94)
10000 - 12000	1.8	(1.14)
12000 - 14000	1.2	(0.88)
14000 - 14500	0.8	(0.55)
14500 - 15000	1.5	(0.68)
15000 - 16000	2.0	(0.97)
16000 - 17000	2.5	(1.18)
17000 - 19000	3.0	(1.26)
19000 - 20000	4.0	(1.39)

Table 7. ^{6}Li(n,t) Cross Section

ENDF/B-VI MAT 325

Recommended as a standard below 1 MeV
Log-log interpolation up to 500 keV
Linear-linear interpolation above 500 keV

Cross Section Values

E_n (MeV)	σ(b)	Uncertainty(%)	E_n (MeV)	σ(b)	Uncertainty(%)
0.2530E-07	940.9827	0.14	0.2360E+00	3.2416	0.24
0.9400E-05	48.7928	0.14	0.2370E+00	3.2495	0.24
0.1500E-03	12.1957	0.14	0.2380E+00	3.2546	0.24
0.2500E-03	9.4428	0.14	0.2390E+00	3.2571	0.24
0.3500E-03	7.9777	0.14	0.2400E+00	3.2568	0.23
0.4500E-03	7.0337	0.14	0.2410E+00	3.2539	0.23
0.5500E-03	6.3613	0.14	0.2420E+00	3.2485	0.23
0.6500E-03	5.8502	0.14	0.2430E+00	3.2403	0.23
0.7500E-03	5.4454	0.14	0.2440E+00	3.2297	0.23
0.8500E-03	5.1137	0.14	0.2450E+00	3.2164	0.23
0.9500E-03	4.8371	0.14	0.2500E+00	3.1145	0.24
0.1500E-02	3.8470	0.14	0.2600E+00	2.7896	0.26
0.2500E-02	2.9791	0.14	0.2700E+00	2.4026	0.28
0.3500E-02	2.5181	0.14	0.2800E+00	2.0366	0.29
0.4500E-02	2.2214	0.14	0.3000E+00	1.4743	0.31
0.5500E-02	2.0110	0.14	0.3100E+00	1.2746	0.32
0.6500E-02	1.8516	0.14	0.3250E+00	1.0498	0.32
0.7500E-02	1.7243	0.14	0.3350E+00	0.9365	0.32
0.8500E-02	1.6221	0.14	0.3500E+00	0.8056	0.32
0.9500E-02	1.5358	0.14	0.3600E+00	0.7376	0.32
0.1500E-01	1.2301	0.14	0.3750E+00	0.6565	0.31
0.2000E-01	1.0737	0.15	0.3850E+00	0.6128	0.31
0.2400E-01	0.9867	0.15	0.4000E+00	0.5591	0.31
0.3000E-01	0.8930	0.16	0.4100E+00	0.5294	0.31
0.3500E-01	0.8359	0.17	0.4250E+00	0.4919	0.31
0.4500E-01	0.7562	0.18	0.4350E+00	0.4706	0.31
0.5500E-01	0.7052	0.20	0.4500E+00	0.4431	0.31
0.6500E-01	0.6725	0.22	0.4600E+00	0.4273	0.31
0.7500E-01	0.6532	0.24	0.4750E+00	0.4065	0.31
0.8500E-01	0.6450	0.26	0.4850E+00	0.3944	0.31
0.9500E-01	0.6468	0.29	0.5000E+00	0.3782	0.31
0.1000E+00	0.6515	0.30	0.5200E+00	0.3597	0.31
0.1100E+00	0.6687	0.32	0.5400E+00	0.3441	0.32
0.1200E+00	0.6976	0.33	0.5700E+00	0.3249	0.32
0.1300E+00	0.7406	0.34	0.6000E+00	0.3093	0.33
0.1400E+00	0.8013	0.35	0.6500E+00	0.2891	0.34
0.1500E+00	0.8843	0.36	0.7000E+00	0.2740	0.35
0.1600E+00	0.9968	0.36	0.7500E+00	0.2623	0.36
0.1700E+00	1.1485	0.35	0.8000E+00	0.2529	0.37
0.1800E+00	1.3522	0.35	0.8500E+00	0.2454	0.38
0.1900E+00	1.6207	0.34	0.9000E+00	0.2391	0.39
0.2000E+00	1.9653	0.32	0.9400E+00	0.2349	0.39
0.2100E+00	2.3763	0.30	0.9600E+00	0.2331	0.40
0.2200E+00	2.8047	0.27	0.9800E+00	0.2313	0.40
0.2300E+00	3.1393	0.25	0.1000E+01	0.2297	0.41
0.2350E+00	3.2311	0.24			

Table 8. $^{10}B(n,\alpha)$ Cross Section

ENDF/B-VI MAT 525

This cross section is the sum of the $^{10}B(n,\alpha_0)$ and $^{10}B(n,\alpha_1)$ cross sections and is recommended as a standard below 250 keV.

Log-log interpolation

Cross Section Values

E_n (MeV)	σ(b)	Uncertainty(%)
0.2530E-07	3839.4960	0.16
0.9400E-05	198.9352	0.16
0.1500E-03	49.6172	0.16
0.2500E-03	38.3776	0.16
0.3500E-03	32.4006	0.16
0.4500E-03	28.5444	0.16
0.5500E-03	25.7955	0.16
0.6500E-03	23.7176	0.16
0.7500E-03	22.0661	0.16
0.8500E-03	20.7130	0.16
0.9500E-03	19.5736	0.16
0.1500E-02	15.5351	0.17
0.2500E-02	11.9871	0.17
0.3500E-02	10.1050	0.17
0.4500E-02	8.8924	0.17
0.5500E-02	8.0282	0.18
0.6500E-02	7.3740	0.18
0.7500E-02	6.8551	0.18
0.8500E-02	6.4321	0.18
0.9500E-02	6.0785	0.18
0.1500E-01	4.8211	0.20
0.2000E-01	4.1703	0.21
0.2400E-01	3.8072	0.23
0.3000E-01	3.4107	0.25
0.4500E-01	2.8081	0.30
0.5500E-01	2.5605	0.32
0.6500E-01	2.3775	0.34
0.7500E-01	2.2354	0.36
0.8500E-01	2.1212	0.39
0.9500E-01	2.0275	0.41
0.1000E+00	1.9851	0.43
0.1200E+00	1.8443	0.48
0.1500E+00	1.6747	0.53
0.1700E+00	1.5729	0.55
0.1800E+00	1.5248	0.56
0.1900E+00	1.4774	0.57
0.2000E+00	1.4314	0.58
0.2100E+00	1.3856	0.59
0.2200E+00	1.3411	0.59
0.2300E+00	1.2982	0.60
0.2350E+00	1.2776	0.60
0.2400E+00	1.2565	0.60
0.2450E+00	1.2365	0.60
0.2500E+00	1.2168	0.60

Table 9. $^{10}B(n,\alpha_1)$ Cross Section

ENDF/B-VI MAT 525

Recommended as a standard cross section below 250 keV

Log-log interpolation

Cross Section Values

E_n (MeV)	σ(b)	Uncertainty(%)
0.2530E-07	3598.2280	0.16
0.9400E-05	186.4350	0.16
0.1500E-03	46.5003	0.16
0.2500E-03	35.9671	0.16
0.3500E-03	30.3659	0.16
0.4500E-03	26.7516	0.16
0.5500E-03	24.1756	0.16
0.6500E-03	22.2287	0.16
0.7500E-03	20.6811	0.16
0.8500E-03	19.4130	0.17
0.9500E-03	18.3446	0.17
0.1500E-02	14.5603	0.17
0.2500E-02	11.2352	0.17
0.3500E-02	9.4717	0.17
0.4500E-02	8.3353	0.18
0.5500E-02	7.5254	0.18
0.6500E-02	6.9123	0.18
0.7500E-02	6.4260	0.18
0.8500E-02	6.0296	0.18
0.9500E-02	5.6981	0.18
0.1500E-01	4.5194	0.20
0.2000E-01	3.9090	0.21
0.2400E-01	3.5682	0.23
0.3000E-01	3.1959	0.25
0.4500E-01	2.6285	0.30
0.5500E-01	2.3943	0.33
0.6500E-01	2.2203	0.35
0.7500E-01	2.0844	0.38
0.8500E-01	1.9745	0.40
0.9500E-01	1.8835	0.43
0.1000E+00	1.8420	0.44
0.1200E+00	1.7031	0.49
0.1500E+00	1.5330	0.55
0.1700E+00	1.4303	0.57
0.1800E+00	1.3817	0.58
0.1900E+00	1.3340	0.59
0.2000E+00	1.2876	0.60
0.2100E+00	1.2417	0.61
0.2200E+00	1.1972	0.62
0.2300E+00	1.1543	0.62
0.2350E+00	1.1338	0.62
0.2400E+00	1.1127	0.62
0.2450E+00	1.0928	0.63
0.2500E+00	1.0732	0.63

Table 10. Au(n,γ) Cross Section

ENDF/B-VI MAT 7925

Recommended as a standard cross section from 200 keV to 2.5 MeV

Linear-linear interpolation

Cross Section Values

E_n (MeV)	σ(b)	Uncertainty(%)
0.2000E+00	0.2502	1.24
0.2100E+00	0.2469	1.43
0.2200E+00	0.2445	1.35
0.2300E+00	0.2415	1.35
0.2350E+00	0.2403	1.33
0.2400E+00	0.2388	1.67
0.2450E+00	0.2374	1.23
0.2500E+00	0.2360	1.36
0.2600E+00	0.2331	1.38
0.2700E+00	0.2299	1.49
0.2800E+00	0.2141	1.31
0.3000E+00	0.1999	1.30
0.3250E+00	0.1877	1.23
0.3500E+00	0.1778	1.25
0.3750E+00	0.1689	1.25
0.4000E+00	0.1614	1.16
0.4250E+00	0.1538	1.21
0.4500E+00	0.1462	1.17
0.4750E+00	0.1389	1.27
0.5000E+00	0.1324	1.15
0.5200E+00	0.1270	1.22
0.5400E+00	0.1236	1.33
0.5700E+00	0.1186	1.64
0.6000E+00	0.1084	1.37
0.6500E+00	0.1002	1.73
0.7000E+00	0.0964	1.42
0.7500E+00	0.0928	1.66
0.8000E+00	0.0897	1.57
0.8500E+00	0.0869	2.10
0.9000E+00	0.0843	3.10
0.9400E+00	0.0825	3.10
0.9600E+00	0.0818	3.70
0.9800E+00	0.0810	4.00
0.1000E+01	0.0803	2.50
0.1100E+01	0.0772	1.80
0.1250E+01	0.0729	1.48
0.1400E+01	0.0694	1.75
0.1600E+01	0.0665	1.70
0.1800E+01	0.0596	1.78
0.2000E+01	0.0534	2.05
0.2200E+01	0.0433	2.01
0.2400E+01	0.0360	3.02
0.2600E+01	0.0311	2.40

At thermal energy, E = 0.0253 eV, the ENDF/B-VI value is 98.71 b (the result of this evaluation process is 98.69 b with an uncertainty of 0.14%).

Table 11. $^{235}U(n,f)$ Cross Section

ENDF/B-VI MAT 9228

Recommended as a standard above 150 keV
Linear-linear interpolation

Cross Section Values

E_n (MeV)	σ(b)	Uncertainty(%)	E_n (MeV)	σ(b)	Uncertainty(%)
0.1500E+00	1.4203	0.58	0.1400E+01	1.2200	0.54
0.1700E+00	1.3967	0.72	0.1600E+01	1.2435	0.52
0.1800E+00	1.3800	0.70	0.1800E+01	1.2619	0.54
0.1900E+00	1.3647	0.82	0.2000E+01	1.2714	0.51
0.2000E+00	1.3510	0.79	0.2200E+01	1.2699	0.53
0.2100E+00	1.3370	0.71	0.2400E+01	1.2561	0.55
0.2200E+00	1.3265	0.71	0.2600E+01	1.2442	0.59
0.2300E+00	1.3130	0.63	0.2800E+01	1.2220	0.62
0.2350E+00	1.3100	0.67	0.3000E+01	1.2010	0.58
0.2400E+00	1.3070	0.68	0.3600E+01	1.1473	0.62
0.2450E+00	1.3030	0.64	0.4000E+01	1.1295	0.63
0.2500E+00	1.2930	0.58	0.4500E+01	1.1011	0.64
0.2600E+00	1.2690	0.63	0.4700E+01	1.0923	0.70
0.2700E+00	1.2500	0.62	0.5000E+01	1.0617	0.70
0.2800E+00	1.2350	0.65	0.5300E+01	1.0502	0.72
0.3000E+00	1.2300	0.60	0.5500E+01	1.0388	0.72
0.3250E+00	1.2300	0.70	0.5800E+01	1.0408	0.78
0.3500E+00	1.2230	0.64	0.6000E+01	1.0985	0.85
0.3750E+00	1.2130	0.70	0.6200E+01	1.1817	0.87
0.4000E+00	1.2020	0.66	0.6500E+01	1.3481	0.87
0.4250E+00	1.1900	0.74	0.7000E+01	1.5467	0.89
0.4500E+00	1.1662	0.70	0.7500E+01	1.6964	0.94
0.4750E+00	1.1510	0.73	0.7750E+01	1.7300	1.05
0.5000E+00	1.1410	0.59	0.8000E+01	1.7606	1.01
0.5200E+00	1.1365	0.72	0.8500E+01	1.7800	0.89
0.5400E+00	1.1300	0.62	0.9000E+01	1.7700	0.99
0.5700E+00	1.1220	0.67	0.1000E+02	1.7415	1.06
0.6000E+00	1.1185	0.62	0.1100E+02	1.7219	1.11
0.6500E+00	1.1182	0.58	0.1150E+02	1.7170	1.26
0.7000E+00	1.1135	0.59	0.1200E+02	1.7347	1.14
0.7500E+00	1.1120	0.58	0.1300E+02	1.9002	0.90
0.8000E+00	1.1100	0.56	0.1400E+02	2.0600	0.59
0.8500E+00	1.1135	0.59	0.1450E+02	2.0800	0.51
0.9000E+00	1.1372	0.57	0.1500E+02	2.0890	0.84
0.9400E+00	1.1691	0.60	0.1600E+02	2.0890	1.10
0.9600E+00	1.1876	0.64	0.1700E+02	2.0413	1.27
0.9800E+00	1.1992	0.72	0.1800E+02	1.9748	1.26
0.1000E+01	1.1969	0.52	0.1900E+02	1.9325	1.25
0.1100E+01	1.1938	0.54	0.2000E+02	1.9343	1.52
0.1250E+01	1.2020	0.51			

At thermal energy, E = 0.0253 eV, the ENDF/B-VI value [40] is 583.98 at 300 °K (the result of this evaluation process is 584.25 b with an uncertainty of 0.19%).

Table 12. Prompt Neutron Spectrum from the Spontaneous Fission of ^{252}Cf

Linear-linear interpolation

E (eV)	N(E) (1/eV)	E (eV)	N(E) (1/eV)	E (eV)	N(E) (1/eV)
1.00000-5	2.03300-12	2.00000-5	2.87500-12	5.00000-5	4.54600-12
1.00000-4	6.43000-12	2.00000-4	9.09300-12	5.00000-4	1.43800-11
1.00000-3	2.03300-11	2.00000-3	2.87500-11	5.00000-3	4.54600-11
1.00000-2	6.43000-11	2.00000-2	9.09300-11	5.00000-2	1.43800-10
1.00000-1	2.03300-10	2.00000-1	2.87500-10	5.00000-1	4.54600-10
1.00000+0	6.43000-10	2.00000+0	9.09300-10	5.00000+0	1.43800-09
1.00000+1	2.03300-09	2.00000+1	2.87500-09	5.00000+1	4.54600-09
1.00000+2	6.42900-09	2.00000+2	9.09200-09	5.00000+2	1.43700-08
1.00000+3	2.03200-08	2.00000+3	2.87100-08	5.00000+3	4.53000-08
1.00000+4	6.38500-08	2.50000+4	9.98900-08	3.00000+4	1.09000-07
3.50000+4	1.17400-07	4.50000+4	1.32200-07	5.50000+4	1.45200-07
7.00000+4	1.62200-07	8.50000+4	1.76900-07	1.00000+5	1.90000-07
1.30000+5	2.12300-07	1.60000+5	2.30800-07	2.00000+5	2.51200-07
2.50000+5	2.71600-07	3.00000+5	2.87700-07	3.70000+5	3.04900-07
4.60000+5	3.20100-07	5.00000+5	3.24800-07	6.00000+5	3.32900-07
7.00000+5	3.36300-07	8.50000+5	3.35200-07	1.00000+6	3.28900-07
1.20000+6	3.15200-07	1.50000+6	2.87500-07	2.00000+6	2.35200-07
2.40000+6	1.95000-07	2.70000+6	1.67700-07	3.00000+6	1.43100-07
3.30000+6	1.21300-07	3.50000+6	1.08200-07	3.70000+6	9.63600-08
3.90000+6	8.55900-08	4.10000+6	7.58800-08	4.30000+6	6.71600-08
4.50000+6	5.93700-08	4.70000+6	5.24200-08	4.90000+6	4.62500-08
5.10000+6	4.07800-08	5.30000+6	3.59200-08	5.50000+6	3.16300-08
5.70000+6	2.78200-08	5.90000+6	2.44600-08	6.10000+6	2.14900-08
6.30000+6	1.88700-08	6.50000+6	1.65500-08	6.70000+6	1.45100-08
6.90000+6	1.27200-08	7.10000+6	1.11400-08	7.30000+6	9.75500-09
7.50000+6	8.54000-09	7.70000+6	7.47500-09	7.90000+6	6.54100-09
8.10000+6	5.72500-09	8.30000+6	5.01100-09	8.50000+6	4.38500-09
8.70000+6	3.83800-09	8.90000+6	3.36000-09	9.10000+6	2.94000-09
9.30000+6	2.57300-09	9.50000+6	2.25200-09	9.70000+6	1.97000-09
9.90000+6	1.72300-09	1.01000+7	1.50600-09	1.03000+7	1.31700-09
1.05000+7	1.15100-09	1.07000+7	1.00700-09	1.09000+7	8.80200-10
1.11000+7	7.69700-10	1.13000+7	6.72900-10	1.15000+7	5.88300-10
1.17000+7	5.14200-10	1.19000+7	4.49400-10	1.21000+7	3.92700-10
1.23000+7	3.43000-10	1.25000+7	2.99600-10	1.27000+7	2.61700-10
1.29000+7	2.28400-10	1.31000+7	1.99400-10	1.32000+7	1.86200-10
1.34000+7	1.62500-10	1.35000+7	1.51700-10	1.37000+7	1.32300-10
1.39000+7	1.15400-10	1.41000+7	1.00700-10	1.42000+7	9.39900-11
1.44000+7	8.19500-11	1.46000+7	7.14500-11	1.53000+7	4.42000-11
1.60000+7	2.73300-11	1.67000+7	1.68800-11	1.74000+7	1.04200-11
1.81000+7	6.42200-12	1.88000+7	3.95000-12	1.95000+7	2.42700-12
2.00000+7	1.71300-12				

Table 13. Uncertainty in the Prompt Neutron Spectrum from the Spontaneous Fission of ^{252}Cf

Energy Range (eV)	Uncertainty (%)	Energy Range (eV)	Uncertainty (%)
0.000E+0 - 1.500E+4	30.00	2.150E+6 - 2.350E+6	1.14
1.500E+4 - 3.500E+4	10.35	2.350E+6 - 2.550E+6	1.14
3.500E+4 - 5.500E+4	4.68	2.550E+6 - 2.750E+6	1.22
5.500E+4 - 7.500E+4	4.28	2.750E+6 - 2.950E+6	1.19
7.500E+4 - 9.500E+4	4.78	2.950E+6 - 3.250E+6	1.17
9.500E+4 - 1.150E+5	3.41	3.250E+6 - 3.550E+6	1.19
1.150E+5 - 1.350E+5	3.34	3.550E+6 - 3.850E+6	1.18
1.350E+5 - 1.650E+5	2.65	3.850E+6 - 4.150E+6	1.19
1.650E+5 - 1.950E+5	2.24	4.150E+6 - 4.450E+6	1.32
1.950E+5 - 2.250E+5	2.25	4.450E+6 - 4.750E+6	1.37
2.250E+5 - 2.550E+5	1.85	4.750E+6 - 5.050E+6	1.35
2.550E+5 - 3.050E+5	1.84	5.050E+6 - 5.550E+6	1.50
3.050E+5 - 3.550E+5	1.69	5.550E+6 - 6.050E+6	1.47
3.550E+5 - 4.050E+5	1.73	6.050E+6 - 6.550E+6	1.58
4.050E+5 - 4.550E+5	1.66	6.550E+6 - 7.050E+6	1.65
4.550E+5 - 5.050E+5	1.65	7.050E+6 - 7.550E+6	1.79
5.050E+5 - 5.550E+5	1.80	7.550E+6 - 8.050E+6	1.86
5.550E+5 - 6.050E+5	1.75	8.050E+6 - 8.550E+6	2.07
6.050E+5 - 6.550E+5	1.62	8.550E+6 - 9.050E+6	2.16
6.550E+5 - 7.050E+5	1.82	9.050E+6 - 9.550E+6	2.25
7.050E+5 - 7.550E+5	1.87	9.550E+6 - 1.005E+7	2.47
7.550E+5 - 8.050E+5	1.83	1.005E+7 - 1.055E+7	2.69
8.050E+5 - 8.550E+5	1.75	1.055E+7 - 1.105E+7	2.90
8.550E+5 - 9.050E+5	1.93	1.105E+7 - 1.155E+7	2.93
9.050E+5 - 9.550E+5	1.74	1.155E+7 - 1.205E+7	3.44
9.550E+5 - 1.050E+6	1.56	1.205E+7 - 1.255E+7	3.59
1.050E+6 - 1.150E+6	1.22	1.255E+7 - 1.305E+7	5.04
1.150E+6 - 1.250E+6	1.21	1.305E+7 - 1.355E+7	4.47
1.250E+6 - 1.350E+6	1.24	1.355E+7 - 1.405E+7	6.27
1.350E+6 - 1.450E+6	1.24	1.405E+7 - 1.460E+7	9.61
1.450E+6 - 1.550E+6	1.23	1.460E+7 - 1.590E+7	12.41
1.550E+6 - 1.650E+6	1.23	1.590E+7 - 1.690E+7	14.70
1.650E+6 - 1.750E+6	1.25	1.690E+7 - 1.790E+7	19.44
1.750E+6 - 1.850E+6	1.20	1.790E+7 - 1.910E+7	32.31
1.850E+6 - 1.950E+6	1.23	1.910E+7 - 2.000E+7	76.95
1.950E+6 - 2.150E+6	1.15		

H(n,n)

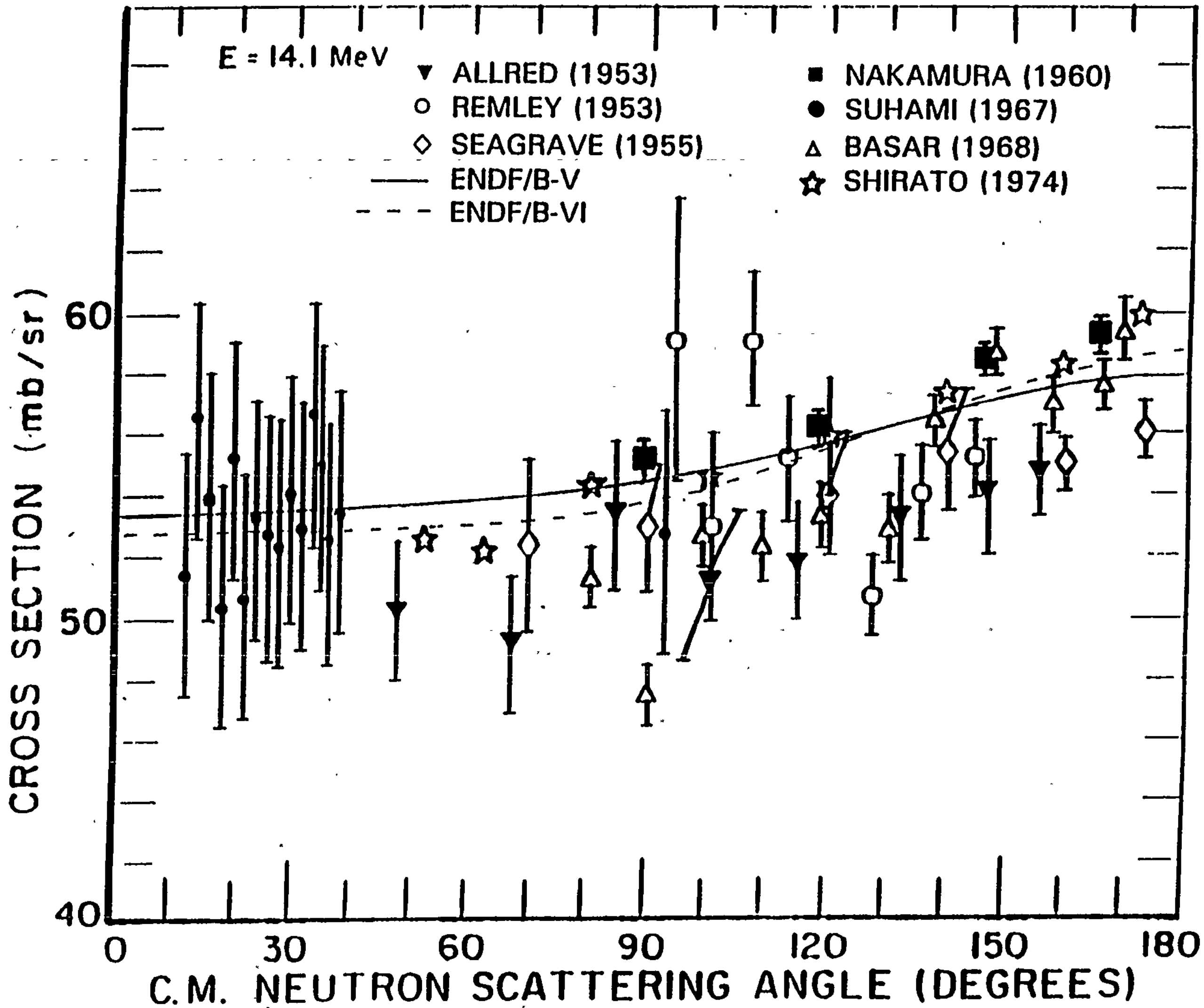

Fig. 1 Measurements of the hydrogen scattering cross section at 14.1 MeV compared with the ENDF/B-V and ENDF/B-VI evaluations. The references for the experimental data are given in reference [41].

H(n,n)

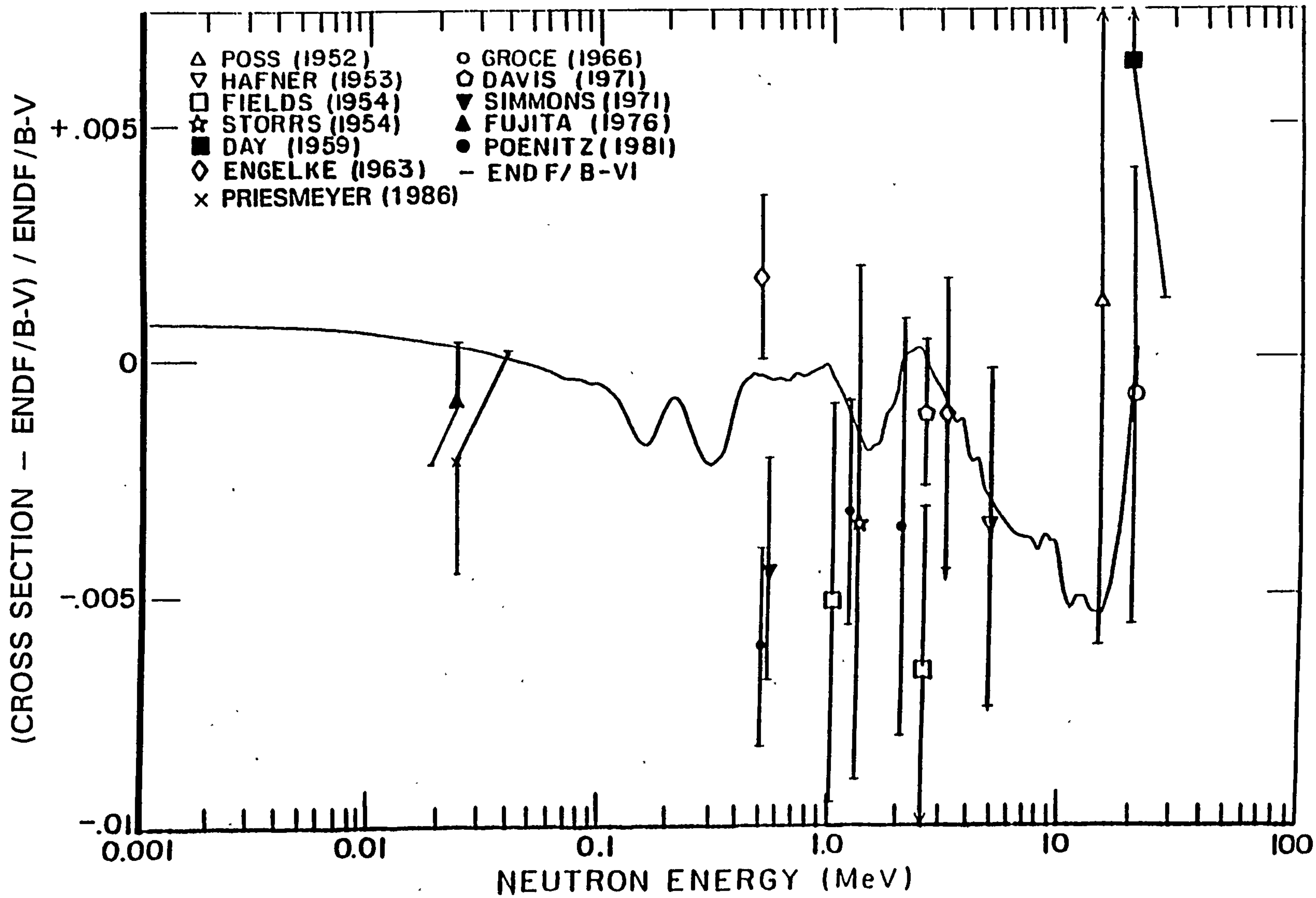

Fig. 2 Comparison of high accuracy measurements of the hydrogen total neutron cross section and the ENDF/B-V and ENDF/B-VI evaluations. The references for the experimental data except Priesmeyer, *et al.* [42] are given in reference [41].

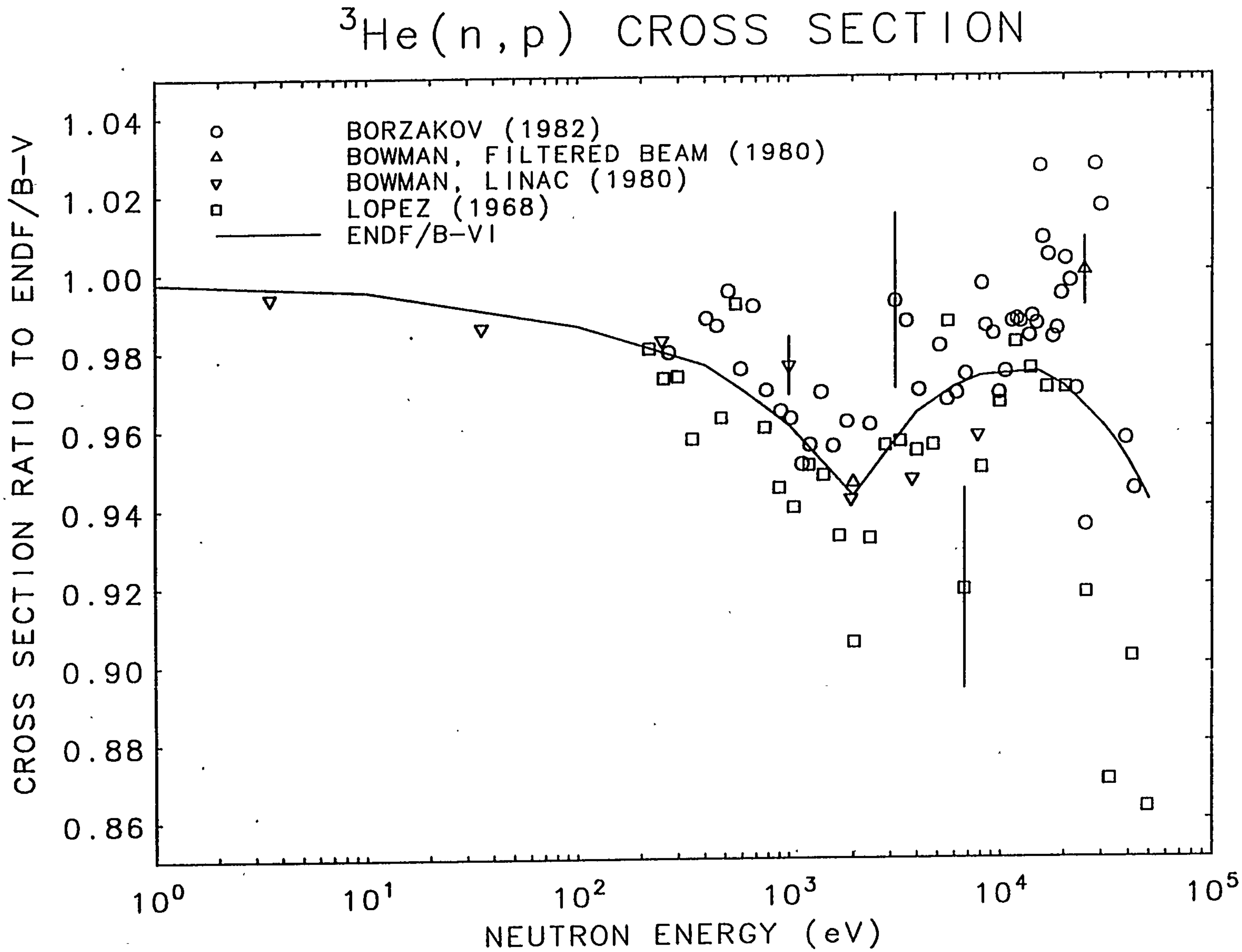

Fig. 3 Measurements of the ^{3}He(n,p) cross section compared with the ENDF/B-V and ENDF/B-VI evaluations. The references for the experimental data except Borzakov [43] are given in reference [41].

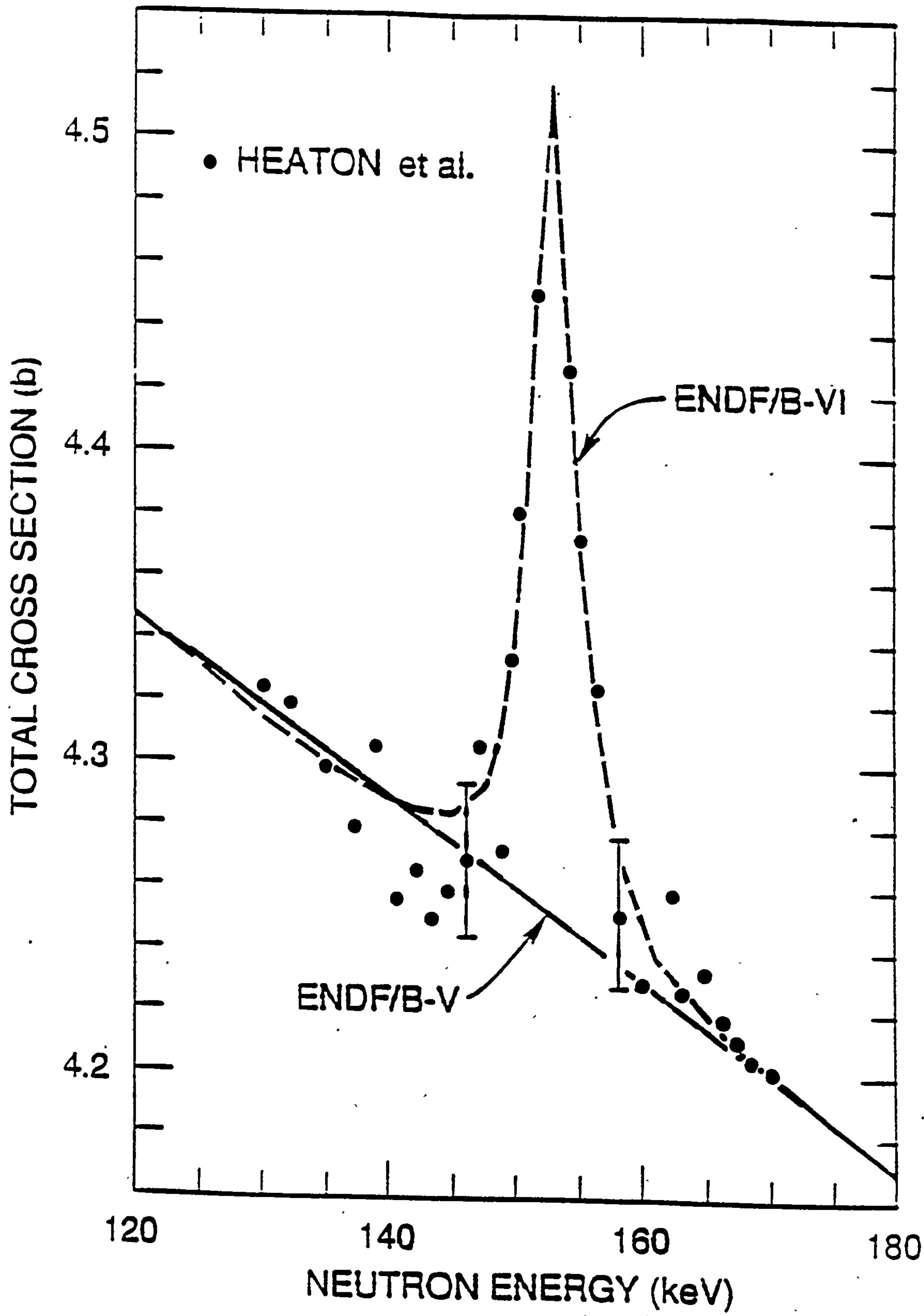

Fig. 4 Comparison of the measurements of Heaton *et al.* [32], of the carbon total cross section near the 152.9 keV resonance in ^{13}C with the ENDF/B-V and ENDF/B-VI evaluations.

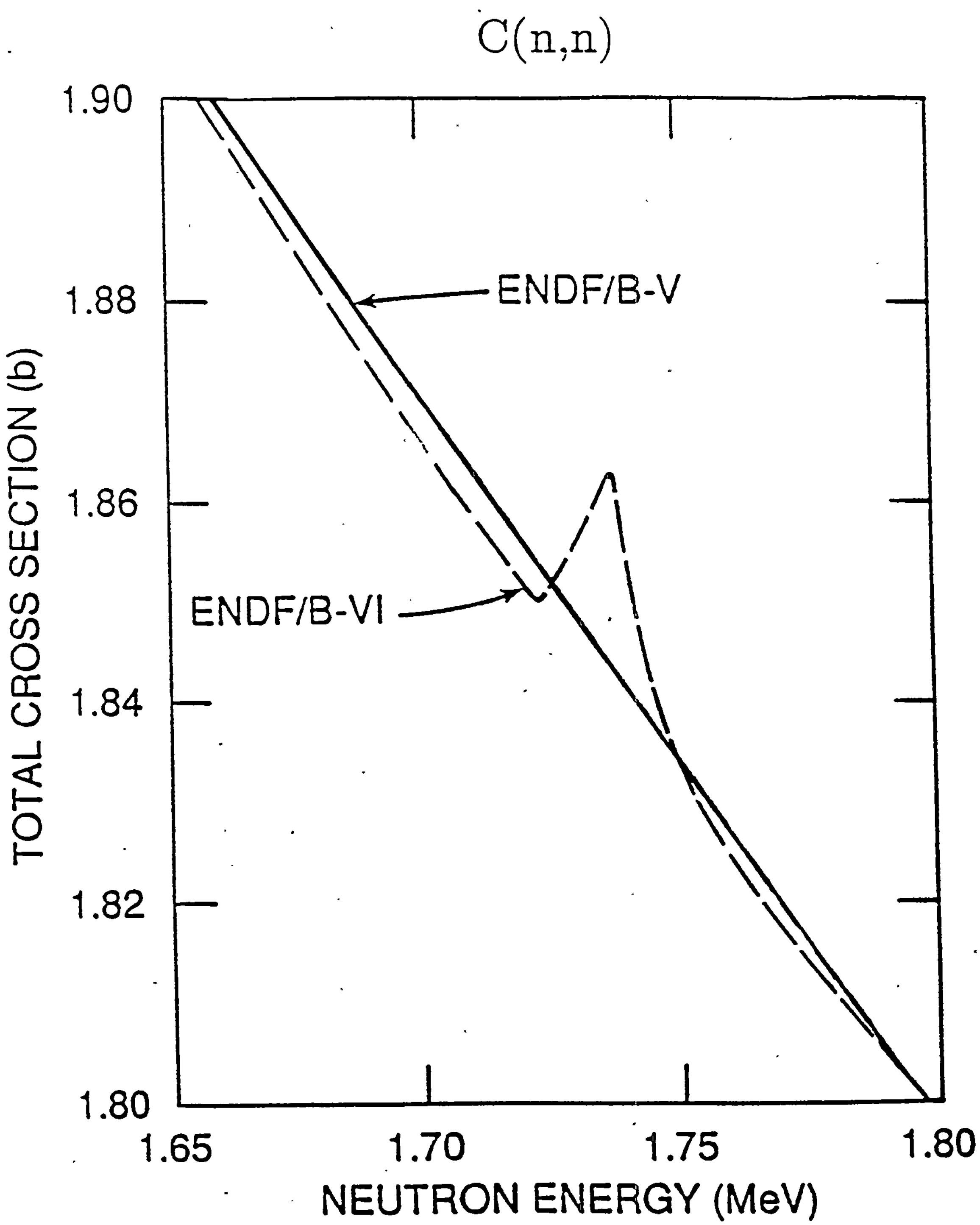

Fig. 5 Comparison of the ENDF/B-V and ENDF/B-VI carbon total cross sections near the 1.736 MeV resonance in ^{13}C.

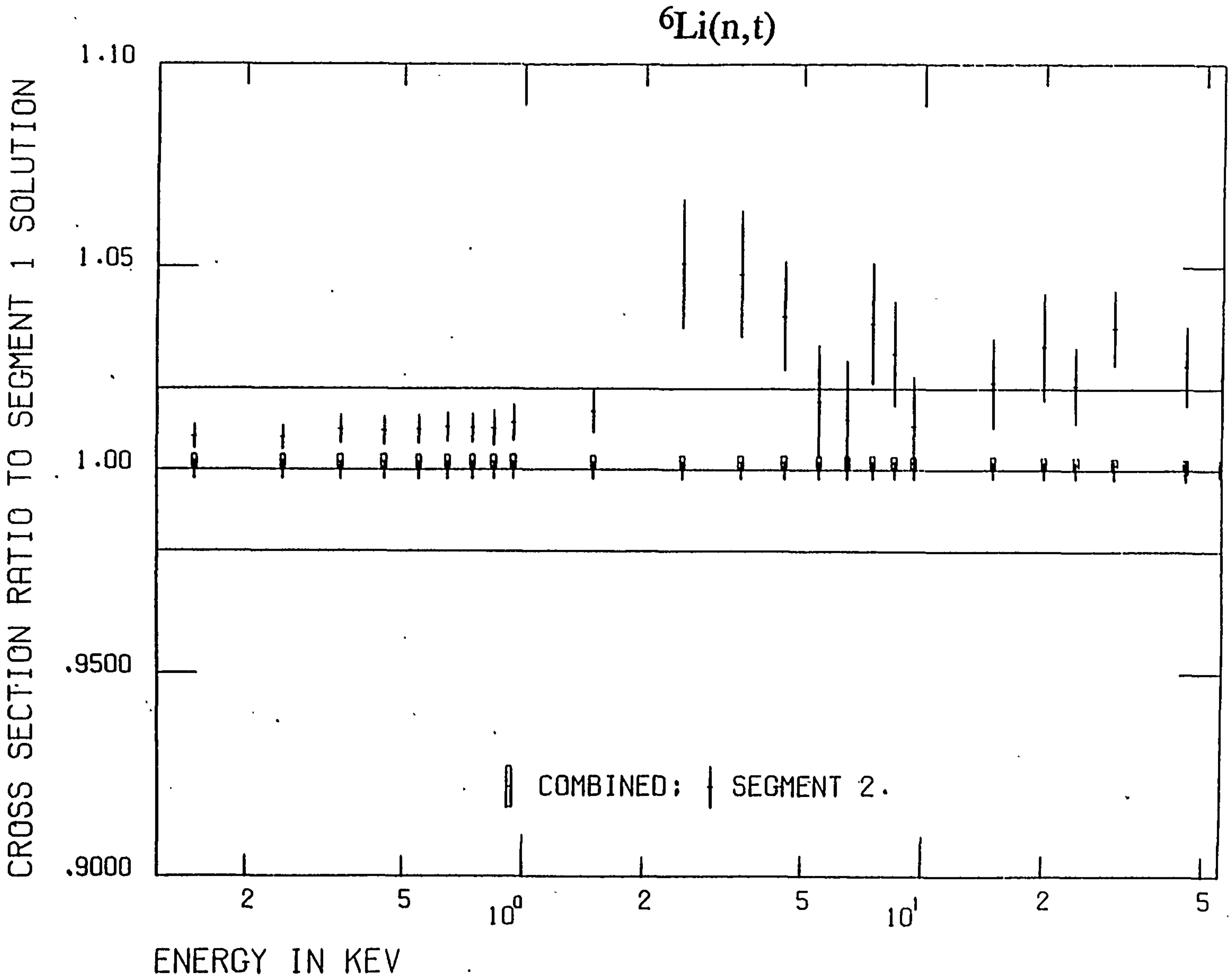

Fig. 6 Cross section ratios to the Segment 1 solution for the ^{6}Li(n,t) reaction from about 0.2 to 50 keV. The rectangles refer to the ratio of the combination output to the R-matrix fit of the Segment 1 data. The $+$'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the R-matrix fit of the Segment 1 data. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the R-matrix fit of the Segment 1 data. The lines at ratios of 0.98 and 1.02 are guides to the eye.

$^6Li(n,t)$

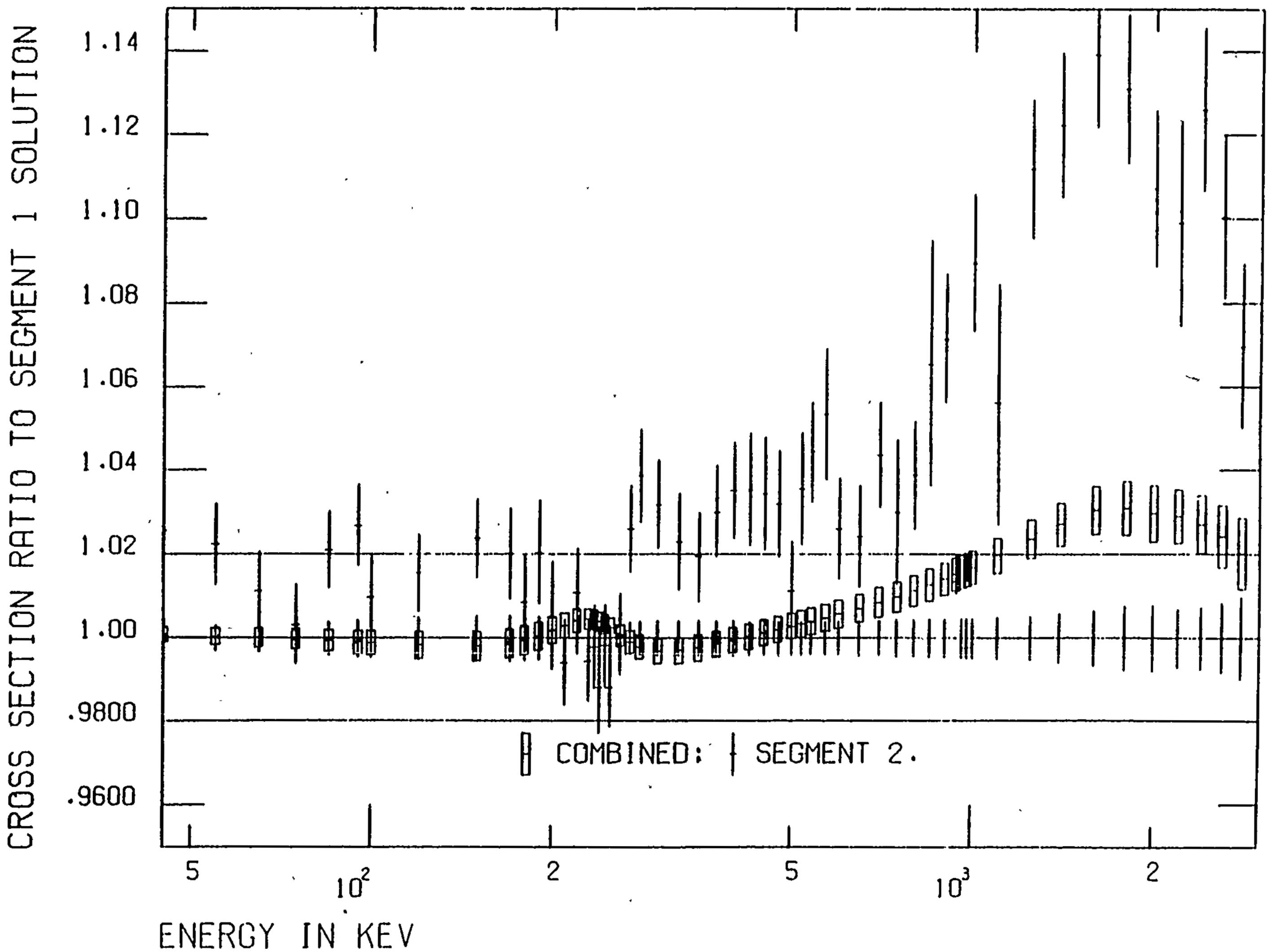

Fig. 7 Cross section ratios to the segment 1 solution for the $^6Li(n,t)$ reaction from about 50 keV to 2 MeV. The rectangles refer to the ratio of the combination output to the R-matrix fit of the segment 1 data. The ┼'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the R-matrix fit of the segment 1 data. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the R-matrix fit of the segment 1 data. The lines at ratios of 0.98 and 1.02 are guides to the eye.

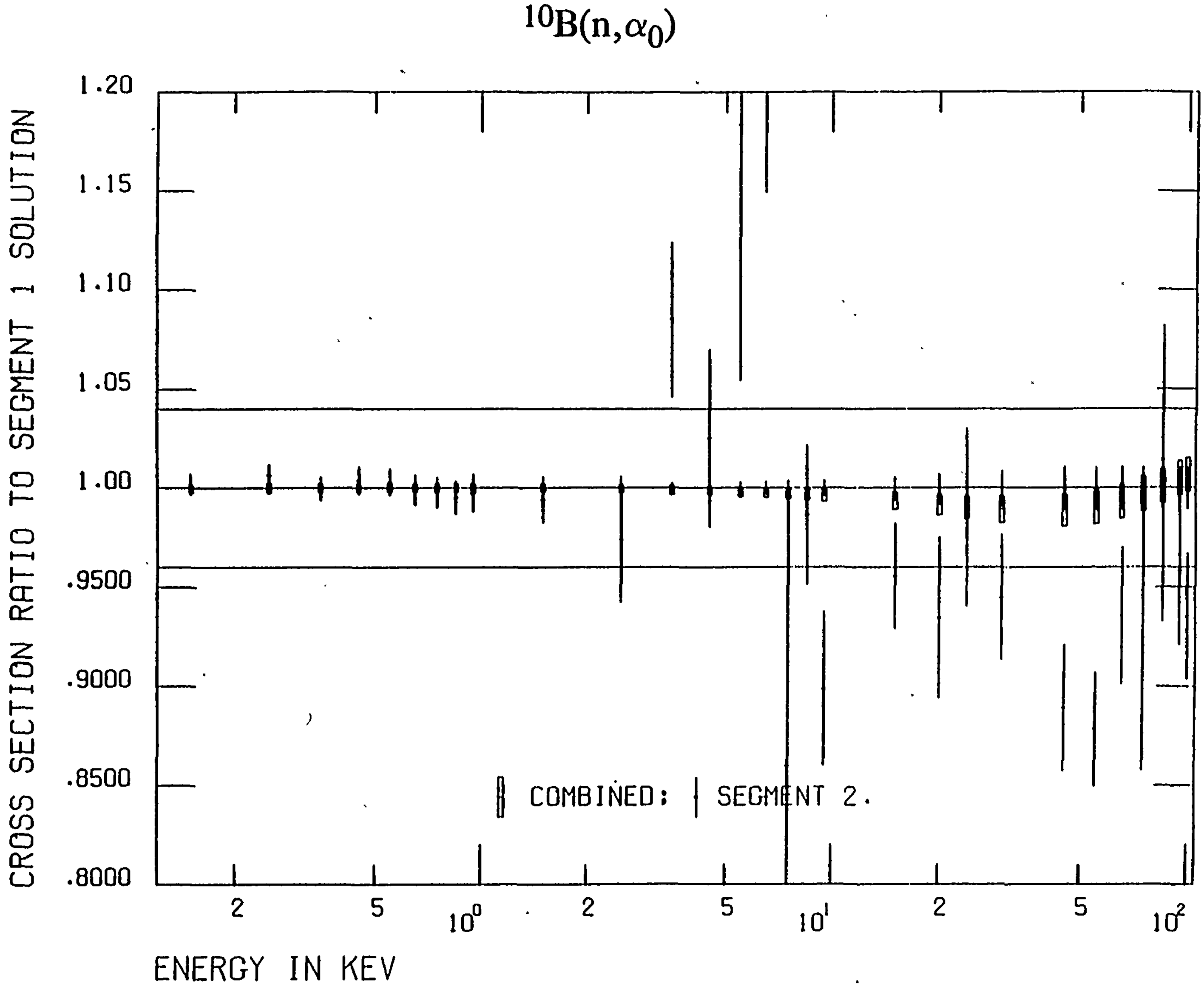

Fig. 8 Cross section ratios to the segment 1 solution for the $^{10}B(n,\alpha_0)$ reaction from about 0.2 to 100 keV. The rectangles refer to the ratio of the combination output to the R-matrix fit of the segment 1 data. The ┼'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the R-matrix fit of the segment 1 data. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the R-matrix fit of the segment 1 data. The lines at ratios of 0.96 and 1.04 are guides to the eye.

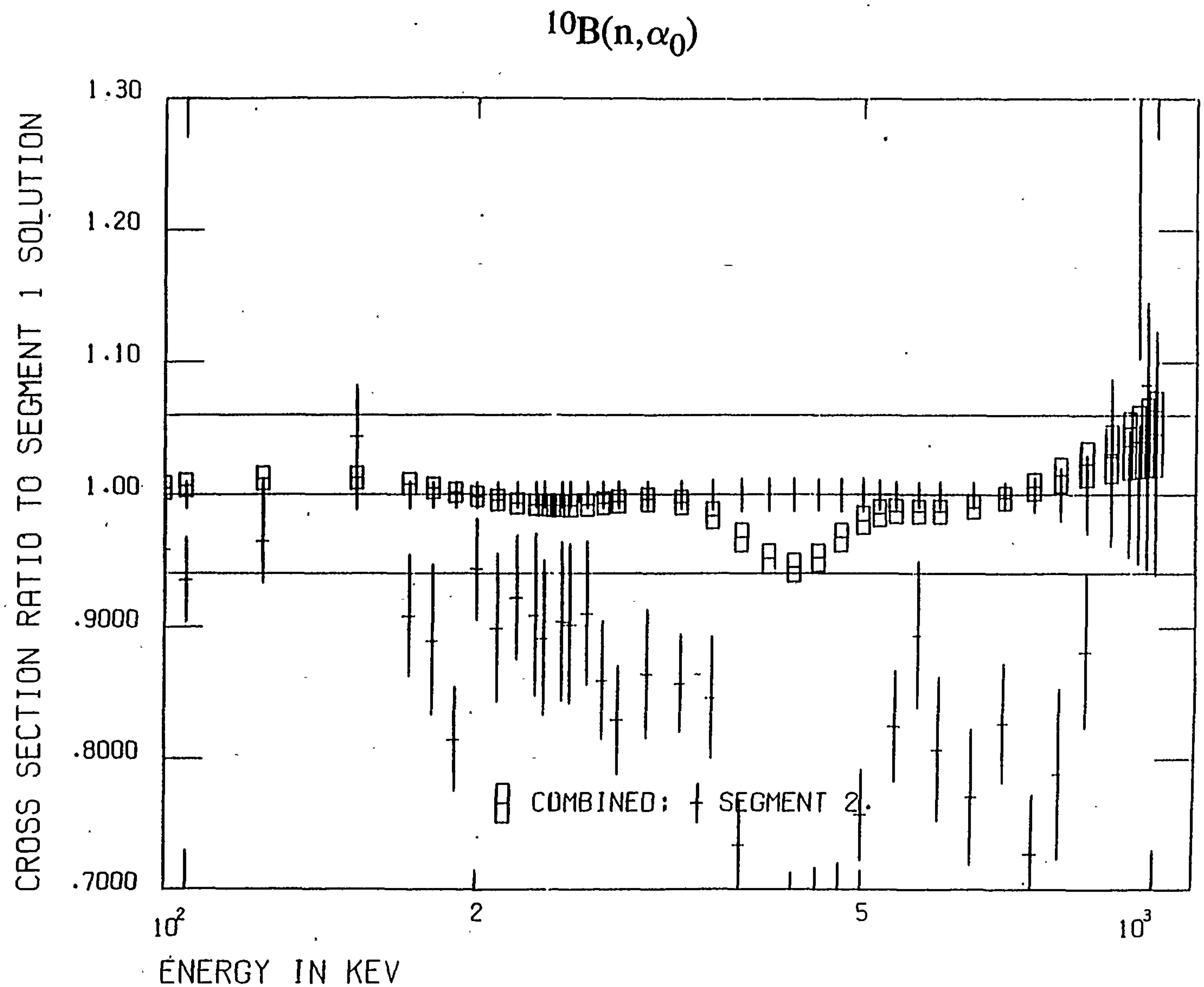

Fig. 9 Cross section ratios to the segment 1 solution for the $^{10}B(n,\alpha_0)$ reaction from about 100 keV to 1 MeV. The rectangles refer to the ratio of the combination output to the R-matrix fit of the segment 1 data. The ┼'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the R-matrix fit of the segment 1 data. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the R-matrix fit of the segment 1 data. The lines at ratios of 0.93 and 1.07 are guides to the eye.

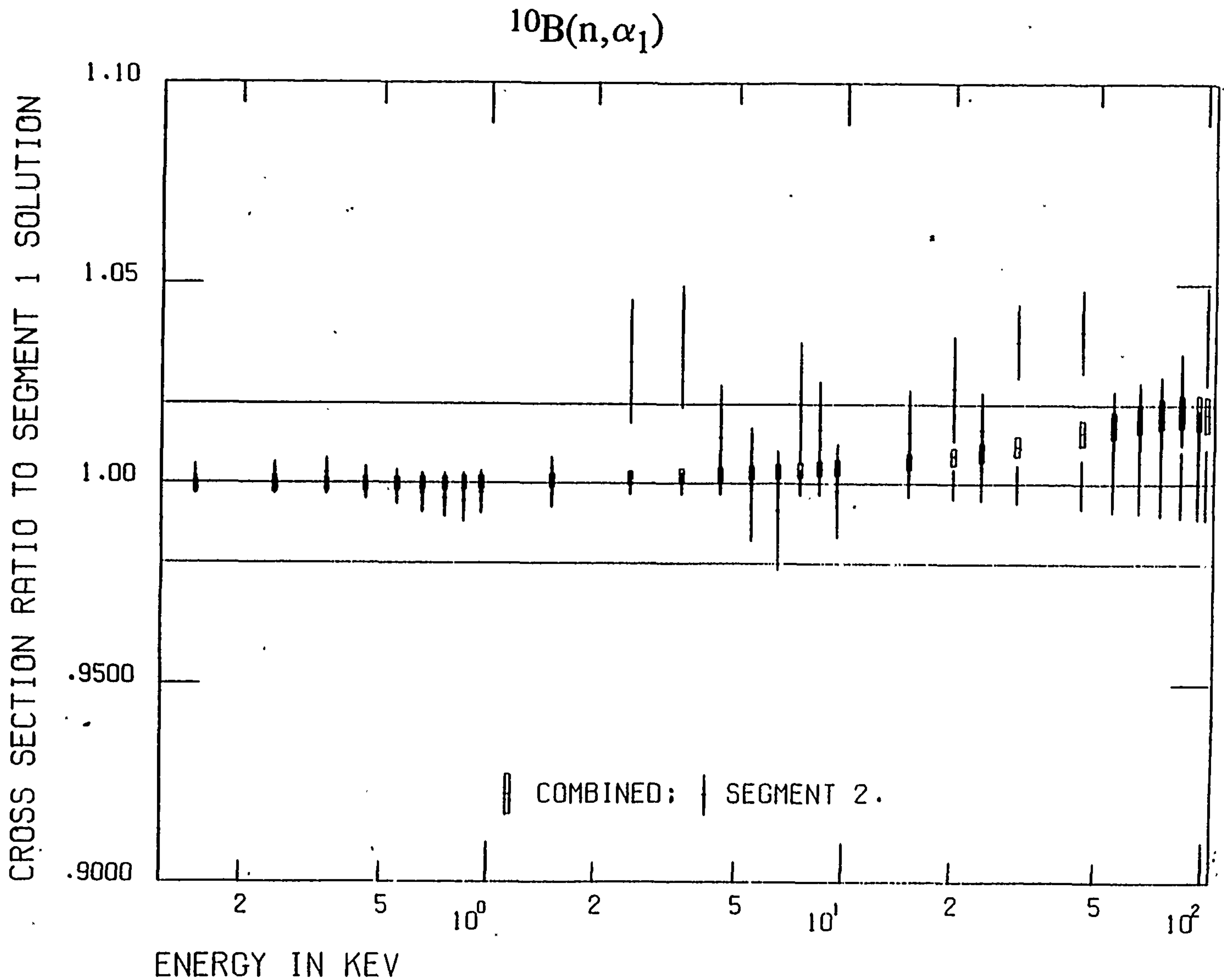

Fig. 10 Cross section ratios to the segment 1 solution for the $^{10}B(n,\alpha_1)$ reaction from about 0.2 keV to 100 keV. The rectangles refer to the ratio of the combination output to the R-matrix fit of the segment 1 data. The $\dagger$'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the R-matrix fit of the segment 1 data. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the R-matrix fit of the segment 1 data. The lines at ratios of 0.98 and 1.02 are guides to the eye.

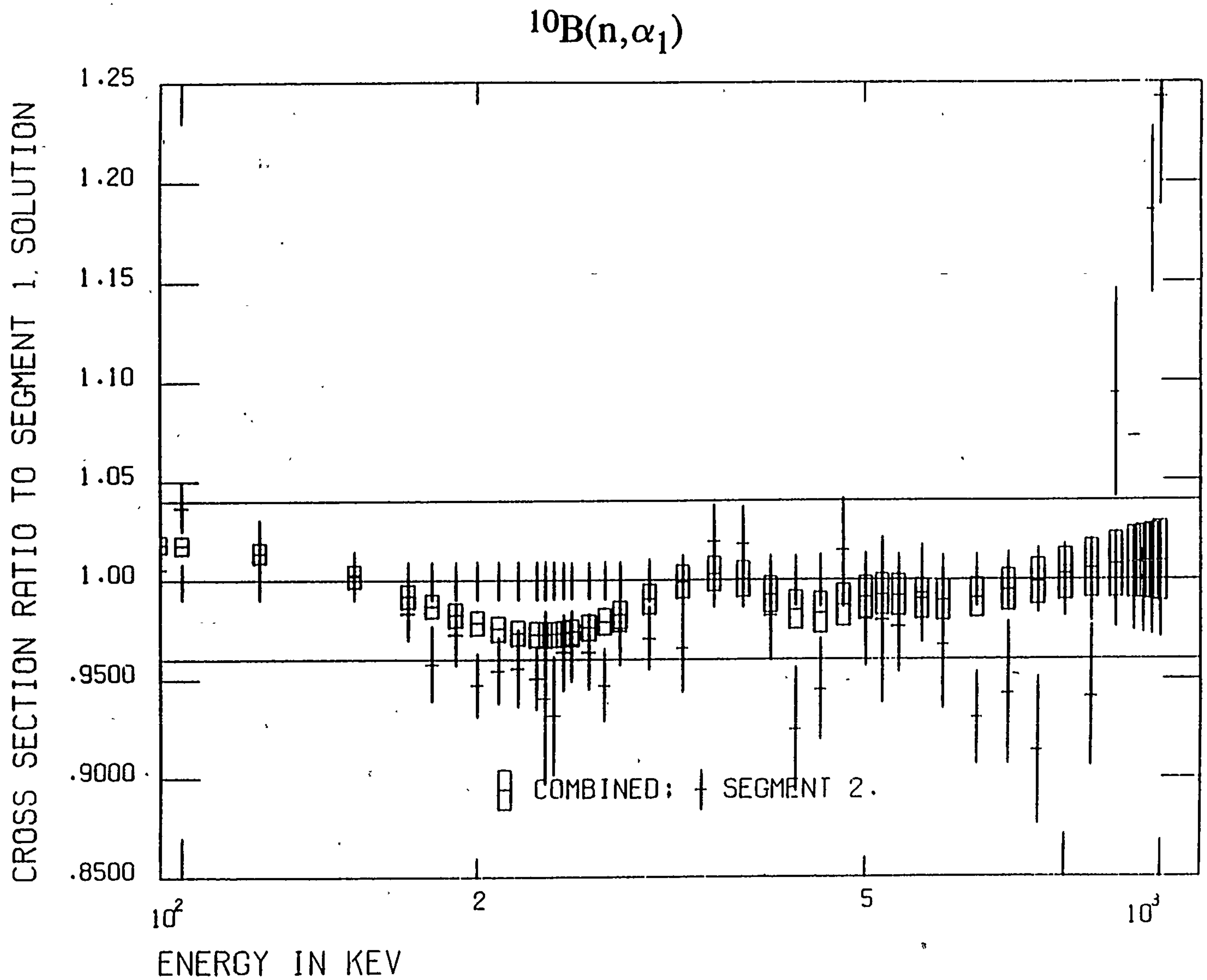

Fig. 11 Cross section ratios to the segment 1 solution for the $^{10}B(n,\alpha_1)$ reaction from about 100 keV to 1 MeV. The rectangles refer to the ratio of the combination output to the R-matrix fit of the segment 1 data. The $+$'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the R-matrix fit of the segment 1 data. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the R-matrix fit of the segment 1 data. The lines at ratios of 0.96 and 1.04 are guides to the eye.

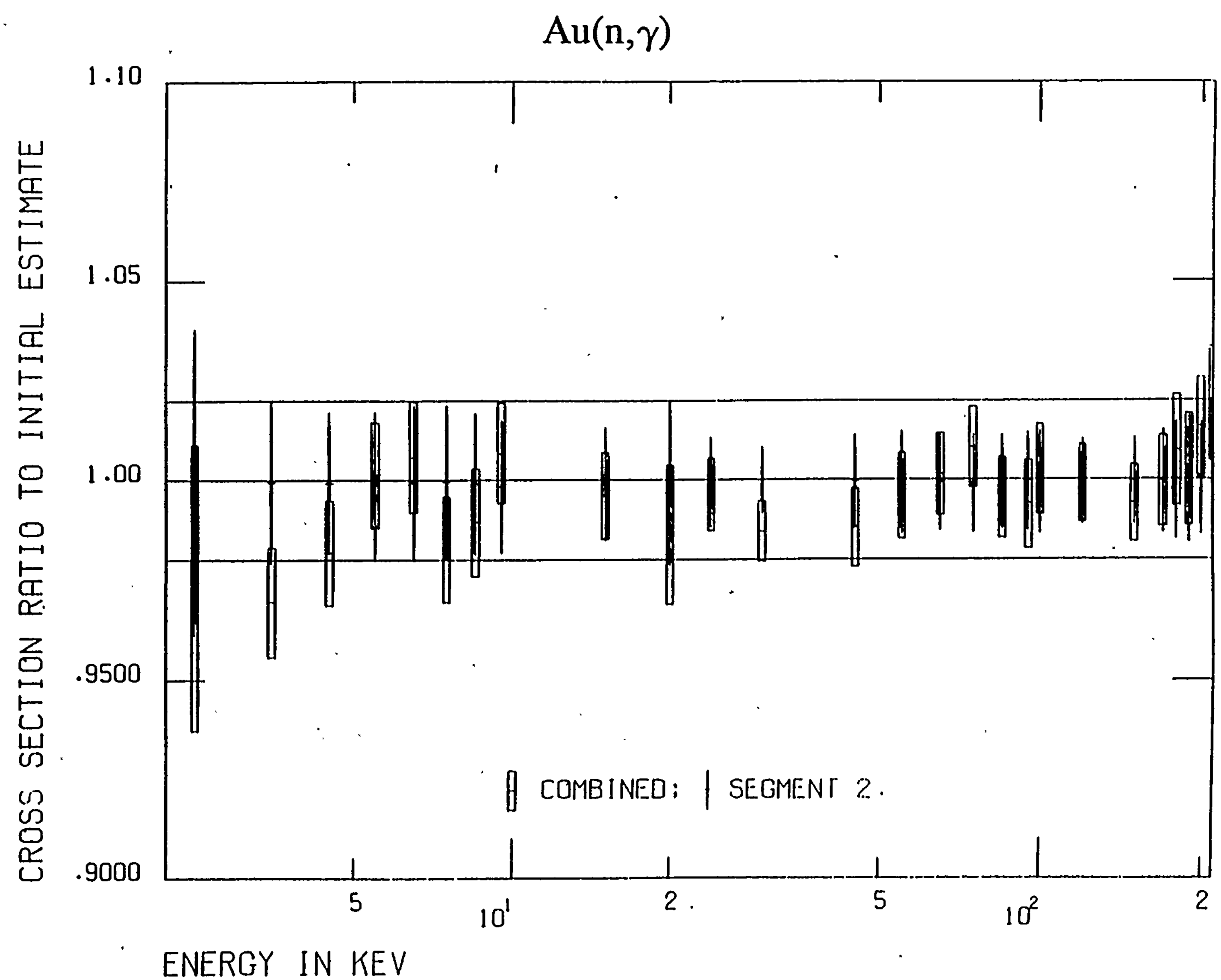

Fig. 12 Cross section ratios for the Au(n,γ) reaction from about 2 keV to 200 keV. Shown are the cross section ratios to the initial estimates for the final iteration of the simultaneous evaluation of the segment 2 data. The rectangles refer to the ratio of the combination output to the initial estimates. The ┼'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the initial estimates. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the simultaneous evaluation of the segment 2 data. The lines at ratios of 0.98 and 1.02 are guides to the eye.

Au(n,γ)

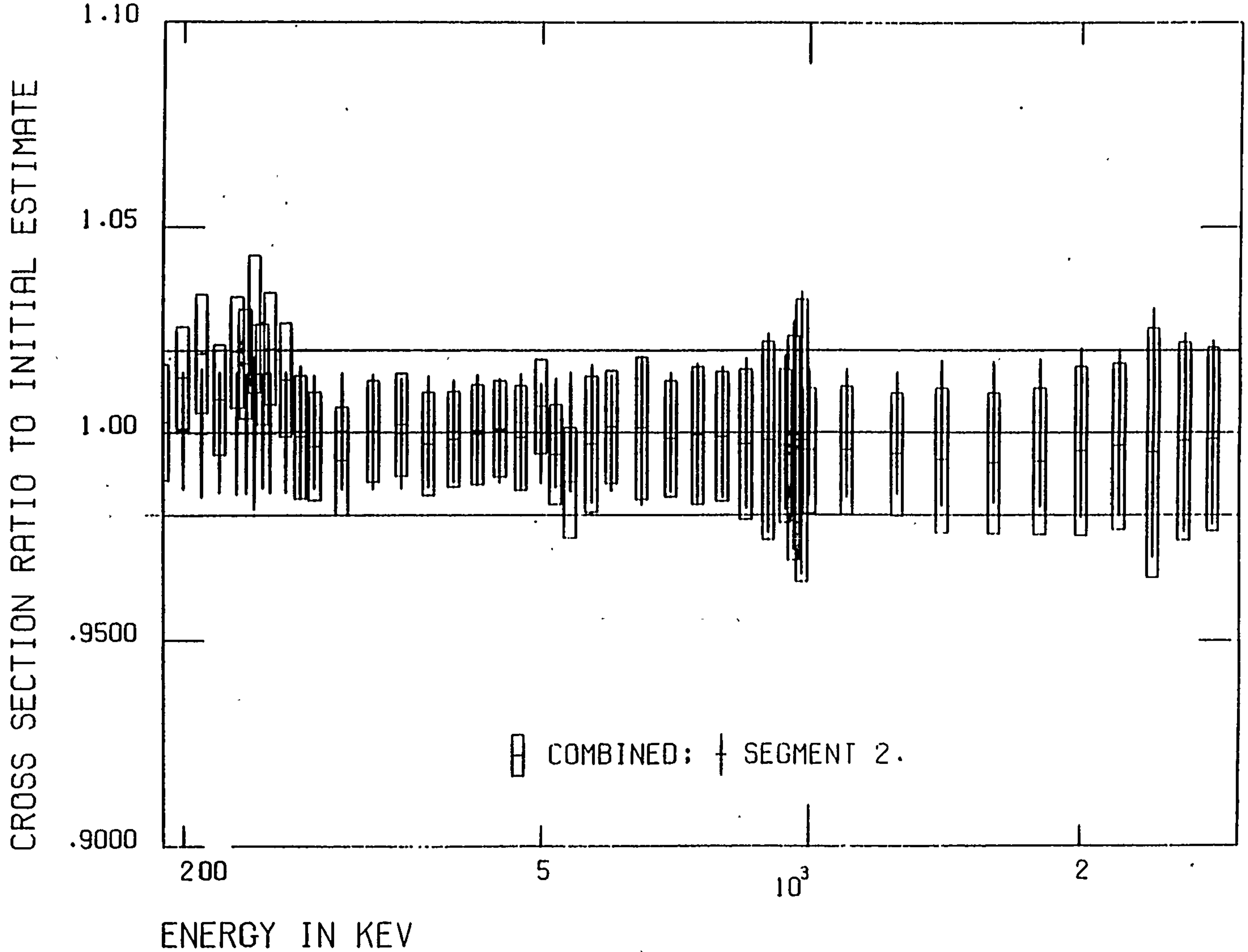

Fig. 13 Cross section ratios for the Au(n,γ) reaction from about 200 keV to 2.5 MeV. Shown are the cross section ratios to the initial estimates for the final iteration of the simultaneous evaluation of the segment 2 data. The rectangles refer to the ratio of the combination output to the initial estimates. The ┼'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the initial estimates. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the simultaneous evaluation of the segment 2 data. The lines at ratios of 0.98 and 1.02 are guides to the eye.

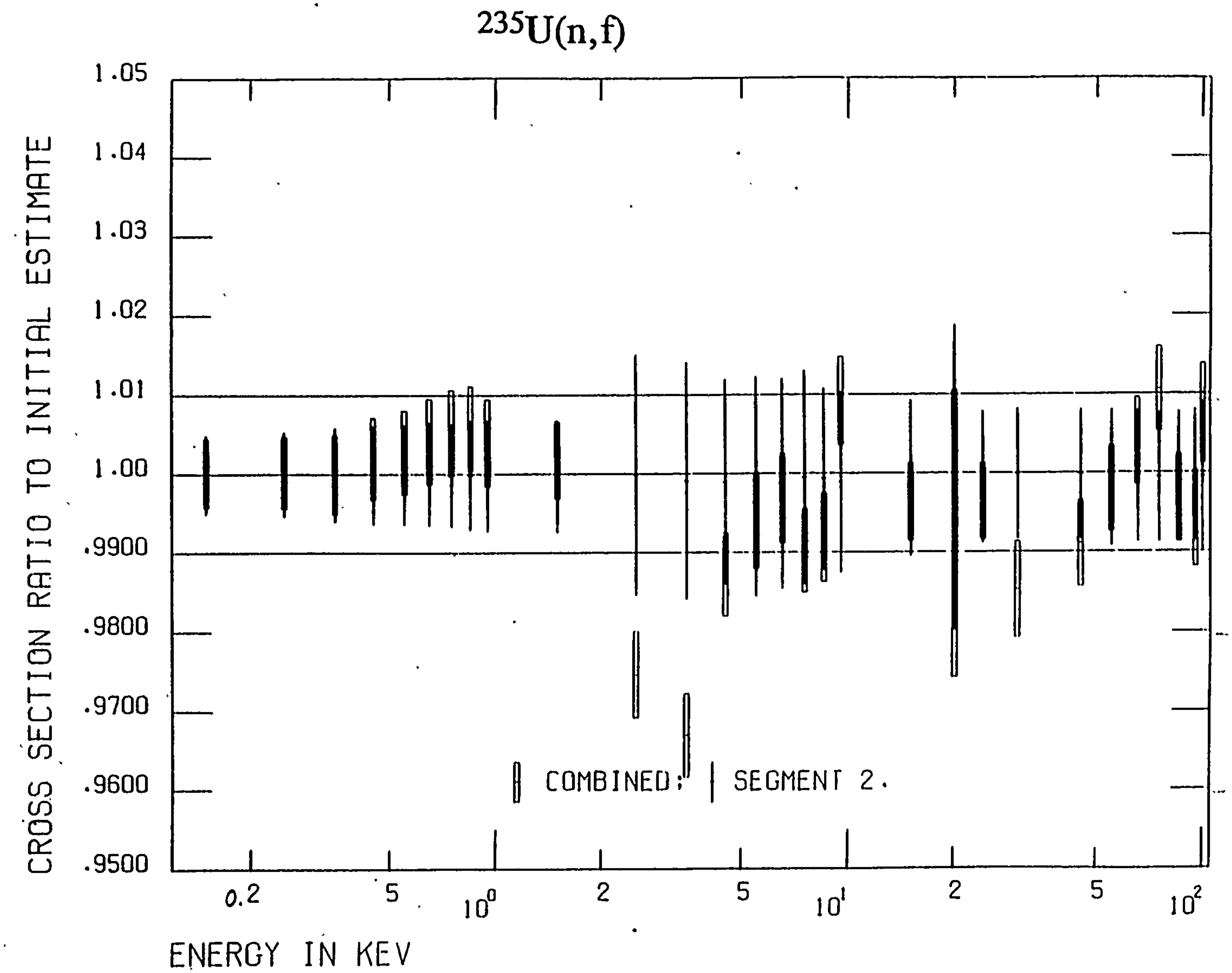

Fig. 14 Cross section ratios for the $^{235}U(n,f)$ reaction from about 0.2 keV to 100 keV. Shown are the cross section ratios to the initial estimates for the final iteration of the simultaneous evaluation of the segment 2 data. The rectangles refer to the ratio of the combination output to the initial estimates. The ┼'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the initial estimates. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the simultaneous evaluation of the segment 2 data. The lines at ratios of 0.99 and 1.01 are guides to the eye.

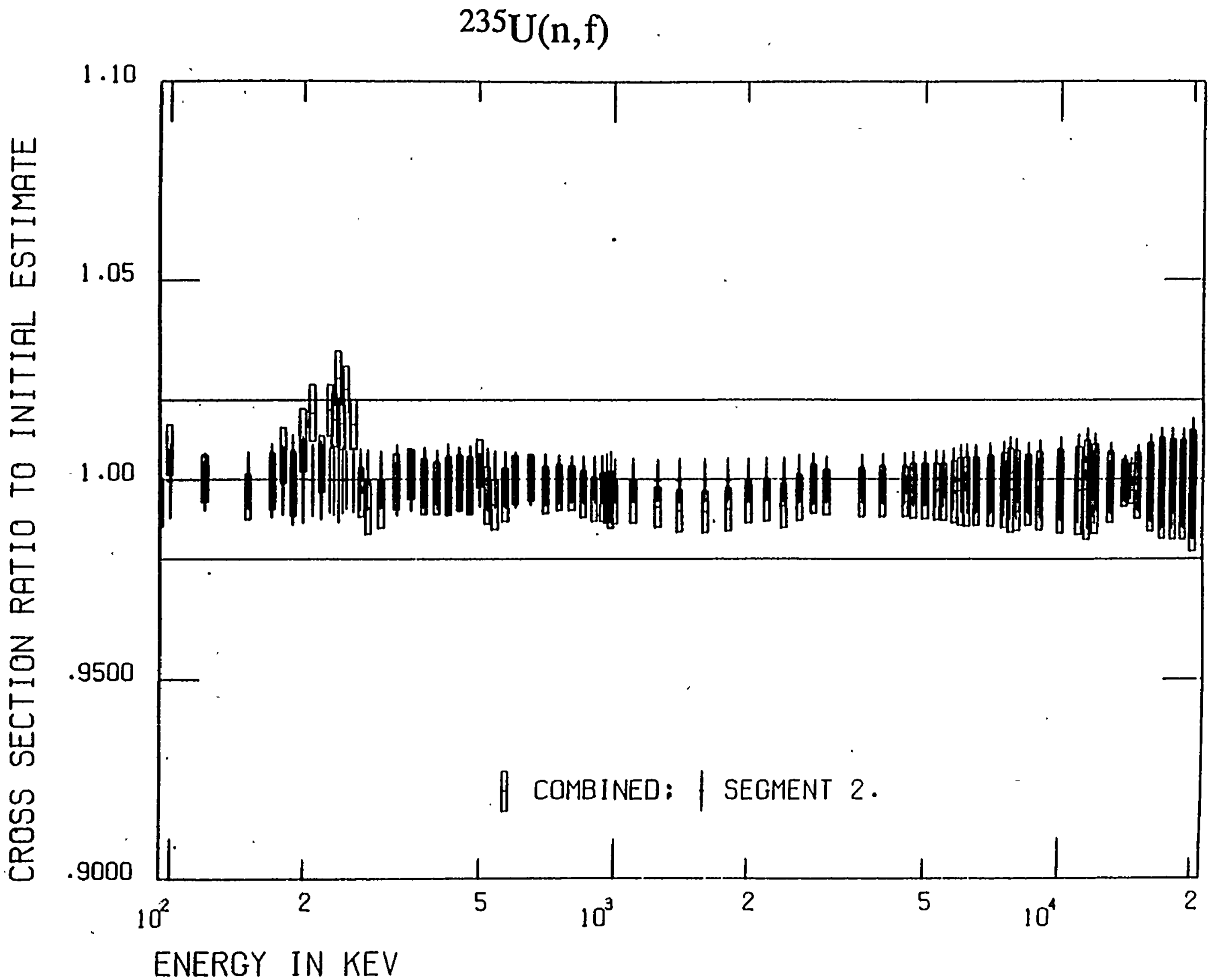

Fig. 15 Cross section ratios for the $^{235}U(n,f)$ reaction from about 0.1 MeV to 20 MeV. Shown are the cross section ratios to the initial estimates for the final iteration of the simultaneous evaluation of the segment 2 data. The rectangles refer to the ratio of the combination output to the initial estimates. The $\dagger$'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the initial estimates. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the simultaneous evaluation of the segment 2 data. The lines at ratios of 0.98 and 1.02 are guides to the eye.

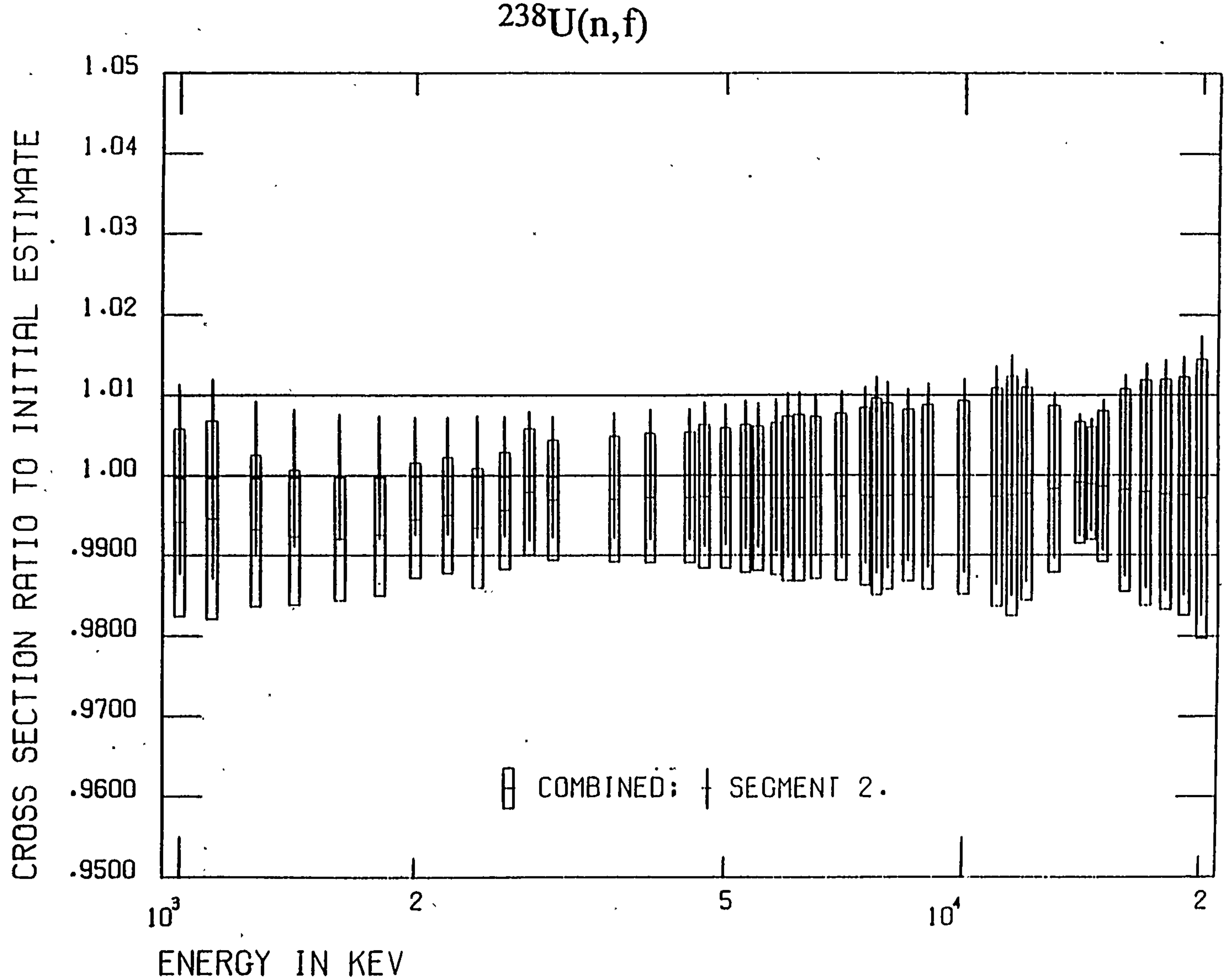

Fig. 16 Cross section ratios for the $^{238}U(n,f)$ reaction from about 1 MeV to 20 MeV. Shown are the cross section ratios to the initial estimates for the final iteration of the simultaneous evaluation of the segment 2 data. The rectangles refer to the ratio of the combination output to the initial estimates. The ┼'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the initial estimates. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the simultaneous evaluation of the segment 2 data. The lines at ratios of 0.99 and 1.01 are guides to the eye.

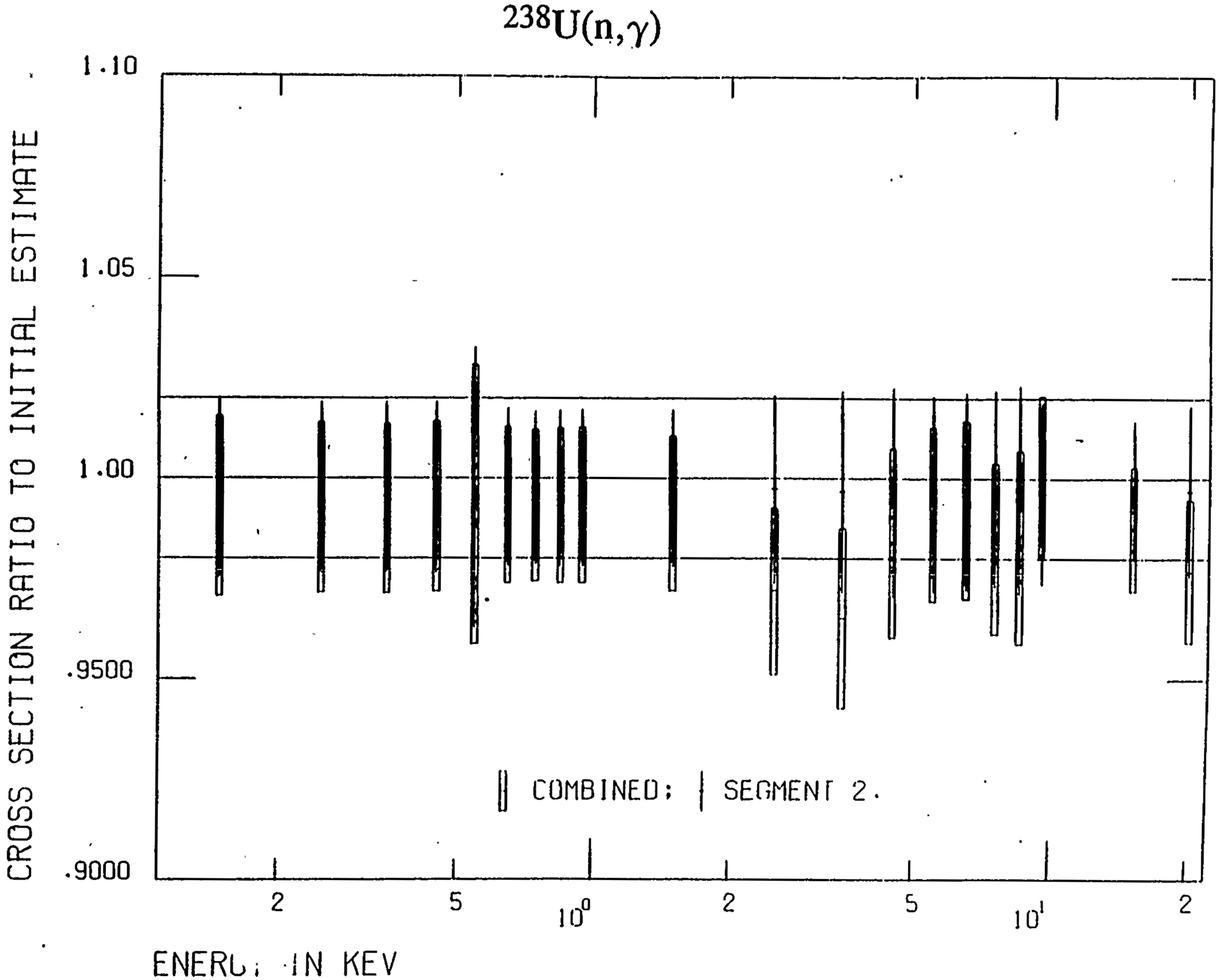

Fig. 17 Cross section ratios for the $^{238}U(n,\gamma)$ reaction from about 0.2 keV to 20 keV. Shown are the cross section ratios to the initial estimates for the final iteration of the simultaneous evaluation of the segment 2 data. The rectangles refer to the ratio of the combination output to the initial estimates. The ┼'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the initial estimates. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the simultaneous evaluation of the segment 2 data. The lines at ratios of 0.98 and 1.02 are guides to the eye.

$^{238}U(n,\gamma)$

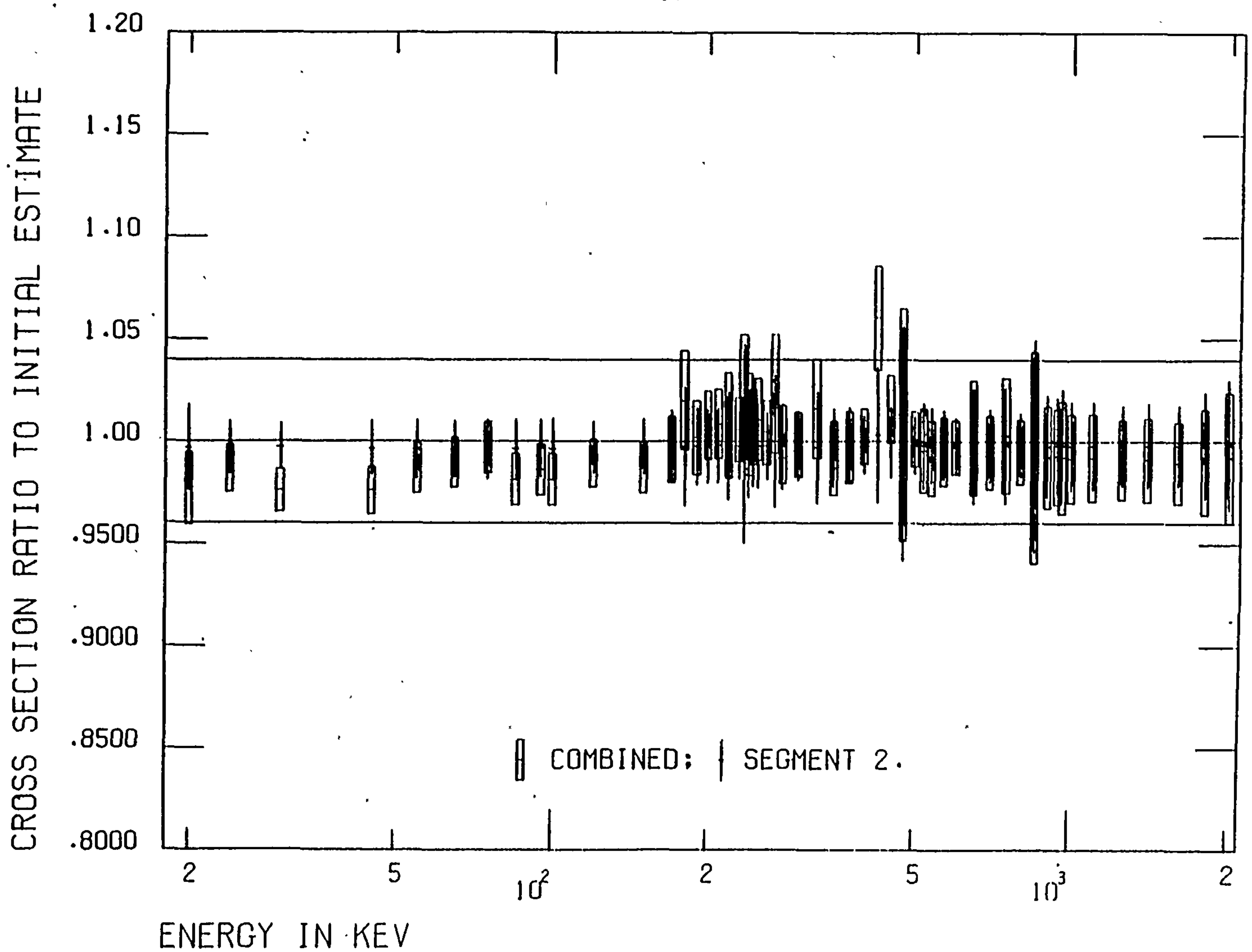

Fig. 18 Cross section ratios for the $^{238}U(n,\gamma)$ reaction from about 20 keV to 2 MeV. Shown are the cross section ratios to the initial estimates for the final iteration of the simultaneous evaluation of the segment 2 data. The rectangles refer to the ratio of the combination output to the initial estimates. The ┼'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the initial estimates. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the simultaneous evaluation of the segment 2 data. The lines at ratios of 0.96 and 1.04 are guides to the eye.

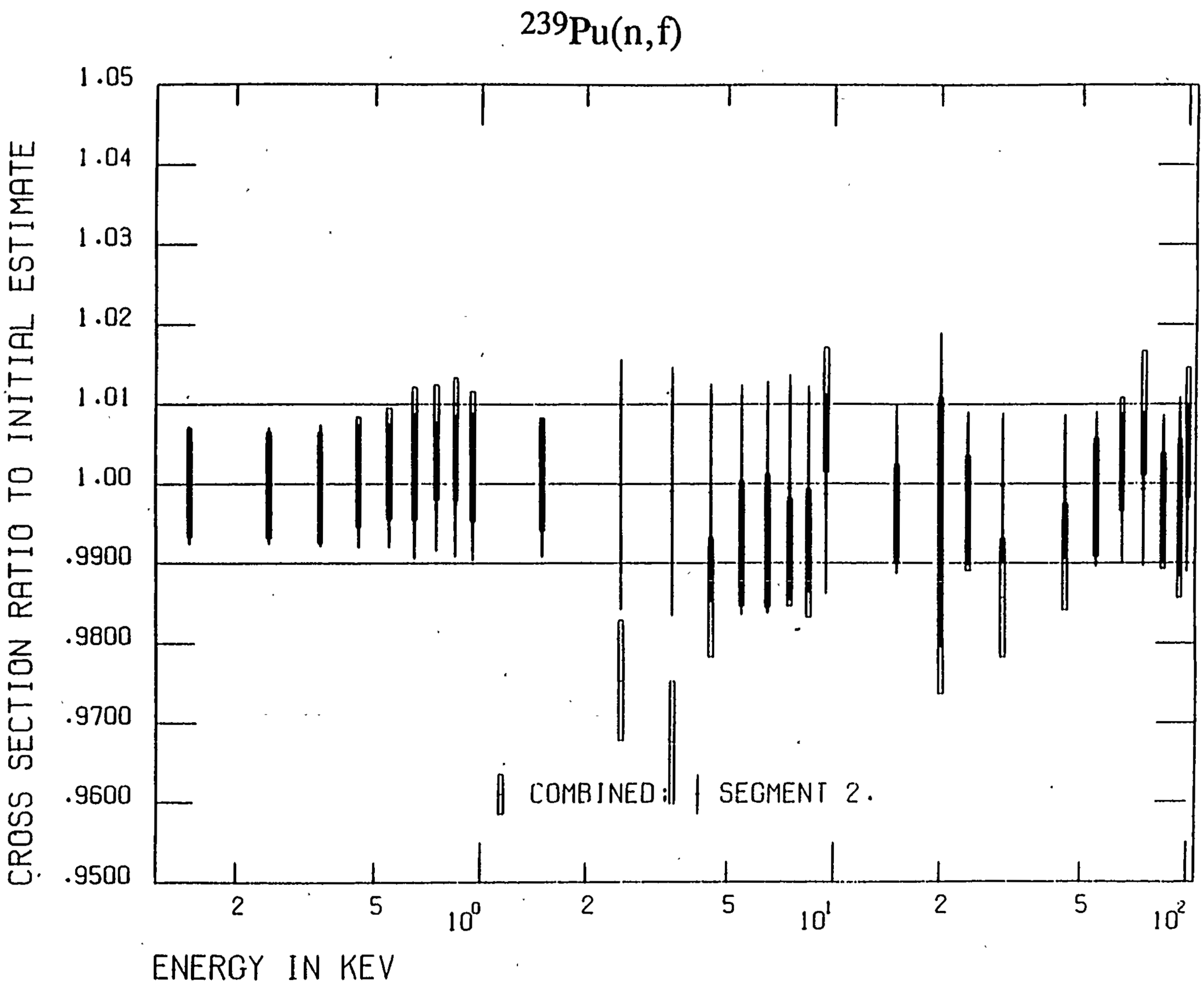

Fig. 19 Cross section ratios for the ^{239}Pu(n,f) reaction from about 0.2 keV to 100 keV. Shown are the cross section ratios to the initial estimates for the final iteration of the simultaneous evaluation of the segment 2 data. The rectangles refer to the ratio of the combination output to the initial estimates. The ┼'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the initial estimates. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the simultaneous evaluation of the segment 2 data. The lines at ratios of 0.99 and 1.01 are guides to the eye.

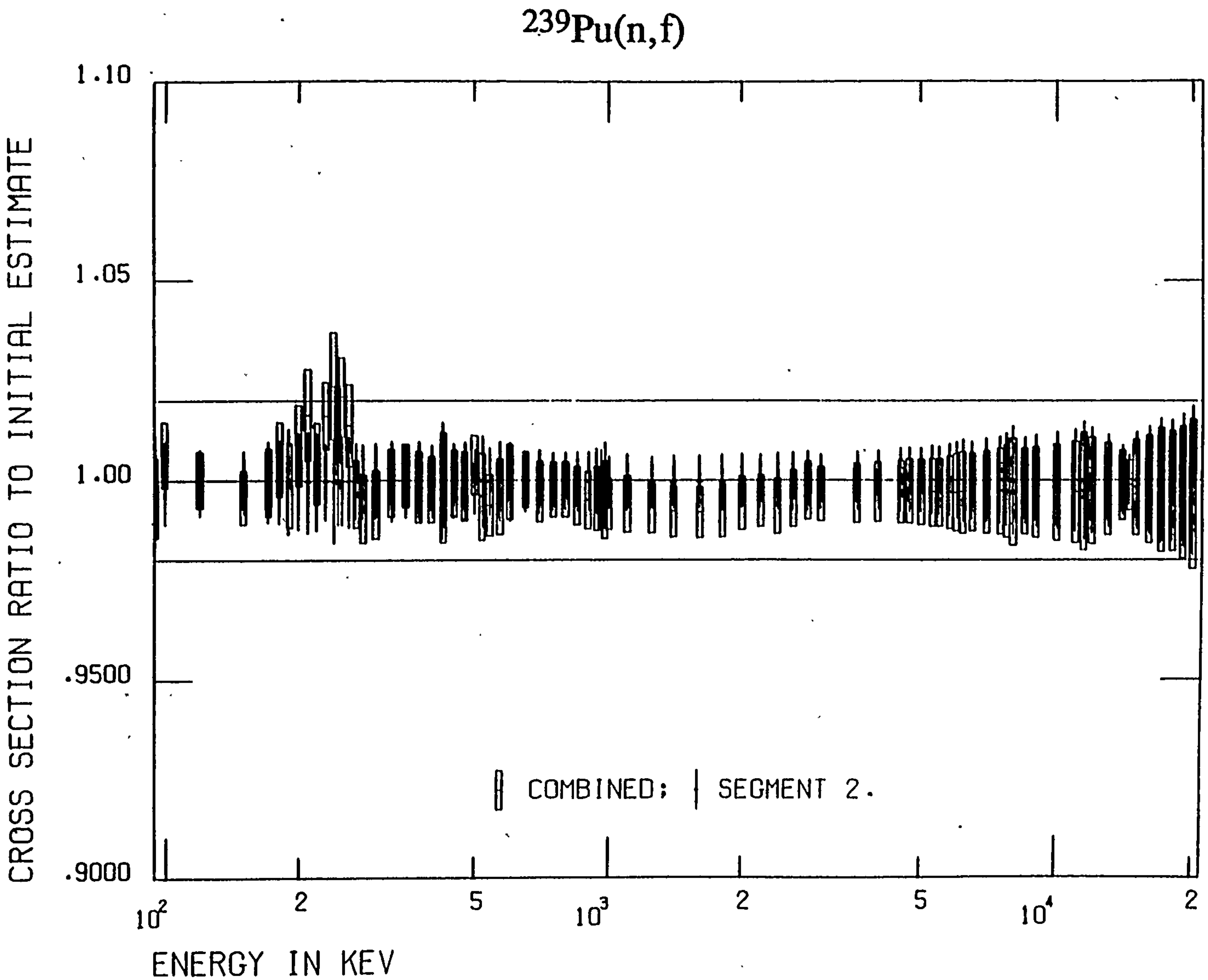

Fig. 20 Cross section ratios for the ^{239}Pu(n,f) reaction from about 100 keV to 20 MeV. Shown are the cross section ratios to the initial estimates for the final iteration of the simultaneous evaluation of the segment 2 data. The rectangles refer to the ratio of the combination output to the initial estimates. The $\dagger$'s refer to the ratio of the simultaneous evaluation of the segment 2 data to the initial estimates. The error bars indicate the uncertainties for the fits. The error bars on the unit ratio line are the uncertainties in the simultaneous evaluation of the segment 2 data. The lines at ratios of 0.98 and 1.02 are guides to the eye.

Au(n,γ)

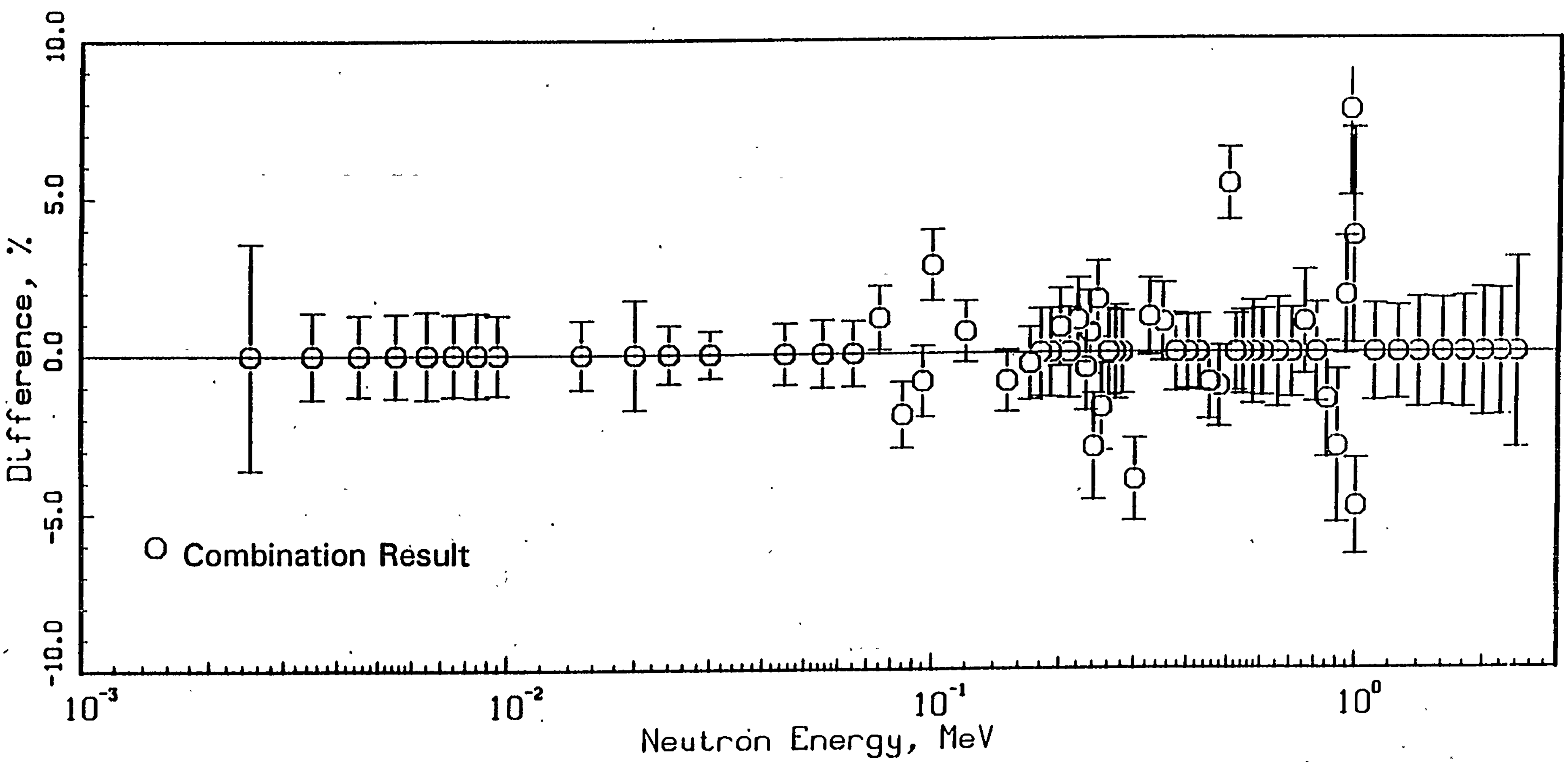

Fig. 21 Percentage difference between the combination result and the smoothed cross section for Au(n,γ) from about 1 keV to 3 MeV.

$^{235}U(n,f)$

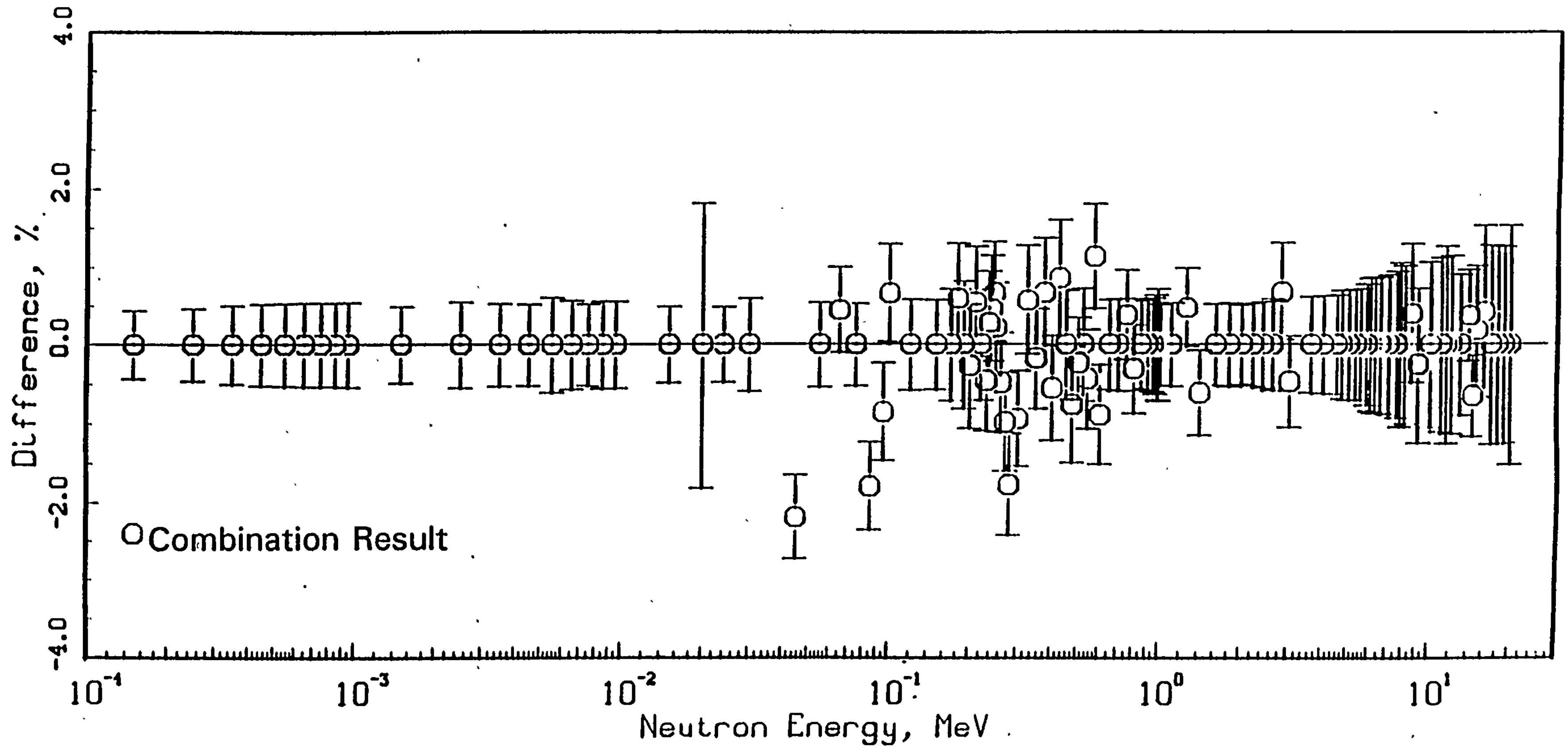

Fig. 22 Percentage difference between the combination result with reference to the smoothed cross section for $^{235}U(n,f)$ from about 0.1 keV to 20 MeV.

$^{238}U(n,f)$

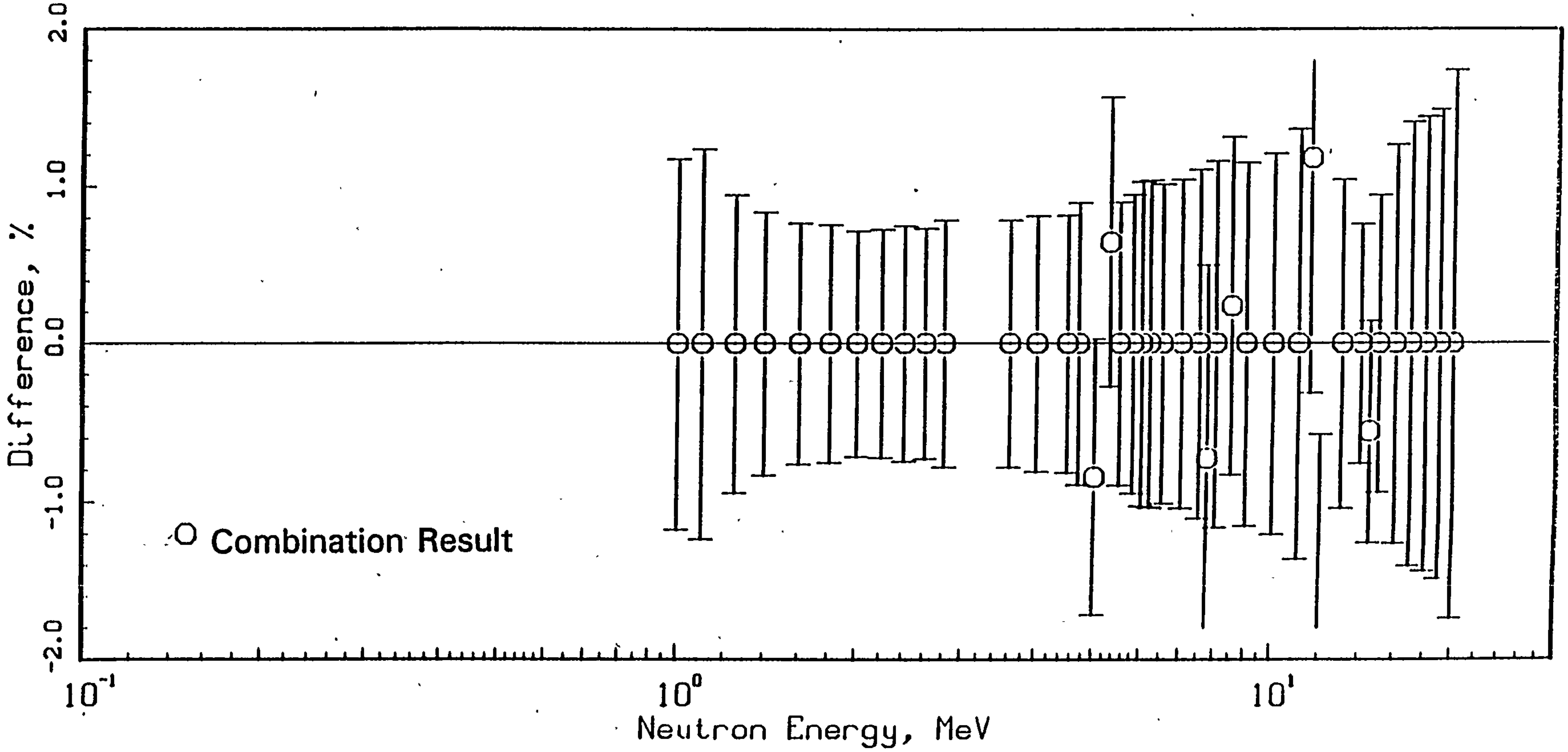

Fig. 23 Percentage difference between the combination result and the smoothed cross section for $^{238}U(n,f)$ from about 1 MeV to 20 MeV.

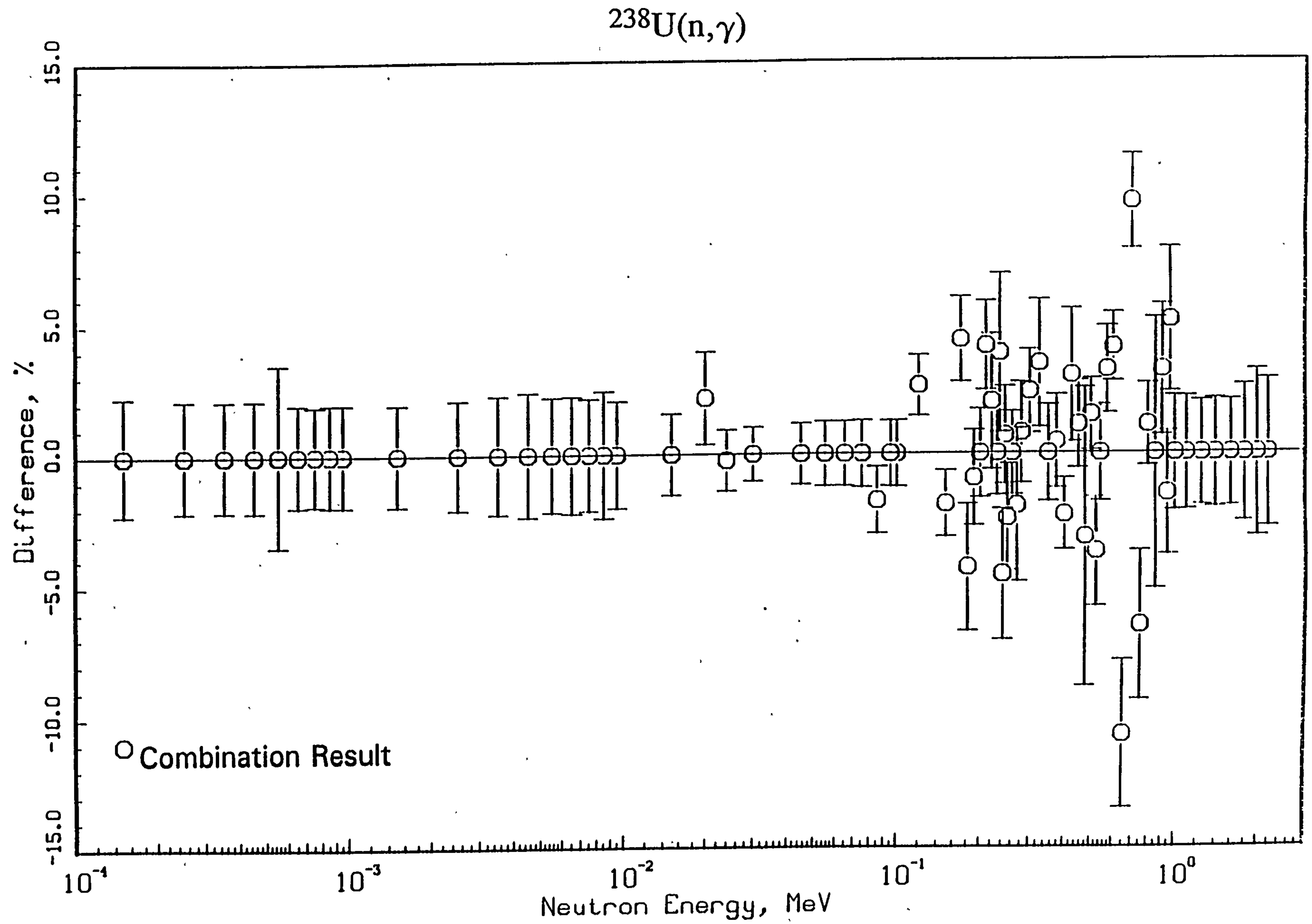

Fig. 24 Percentage difference between the combination result and the smoothed cross section for $^{238}U(n,\gamma)$ from about 0.1 keV to 3 MeV.

$^{239}Pu(n,f)$

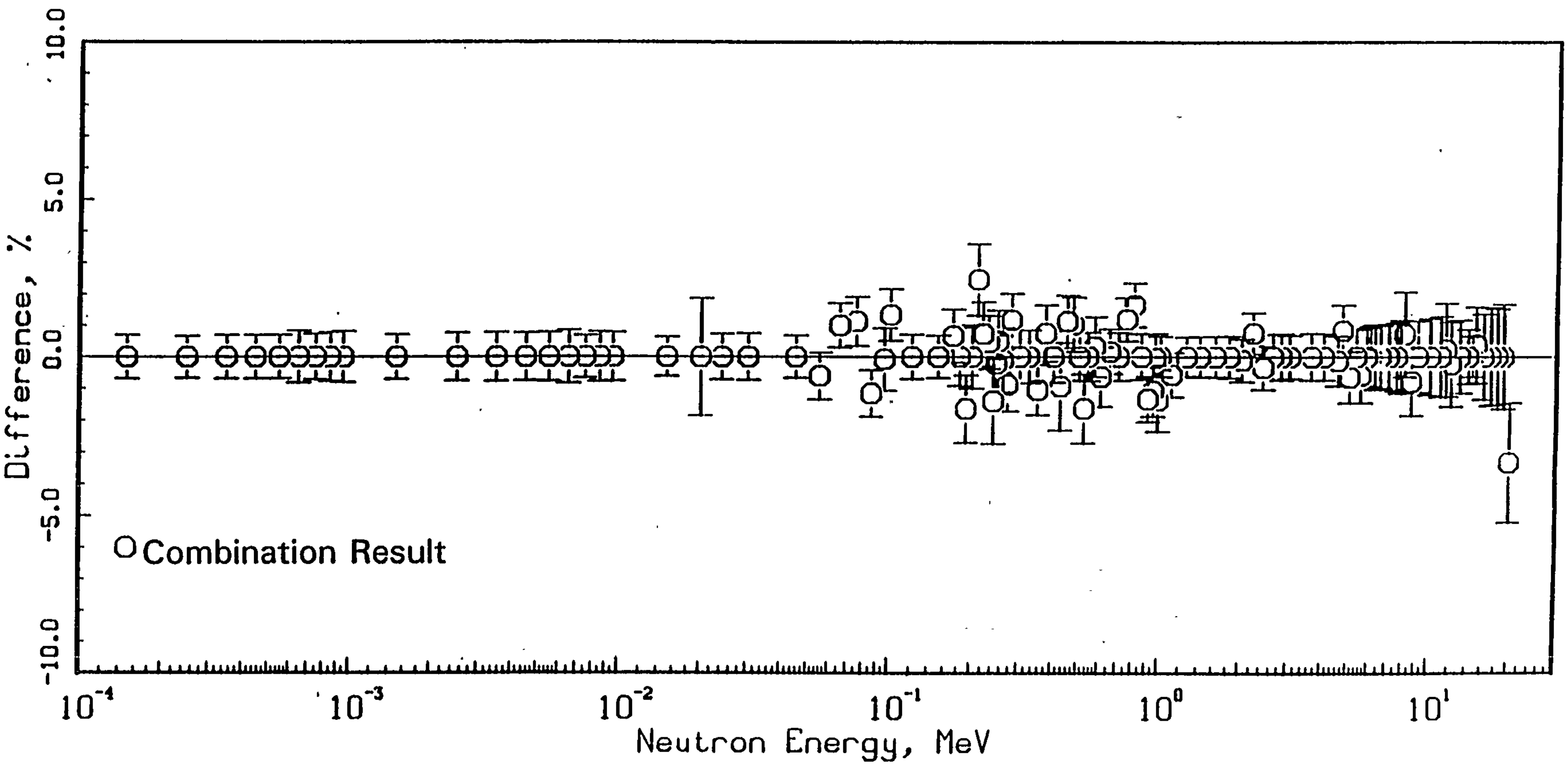

Fig. 25 Percentage difference between the combination result and the smoothed cross section for $^{239}Pu(n,f)$ from about 0.1 keV to 20 MeV.

$^{6}Li(n,t)$

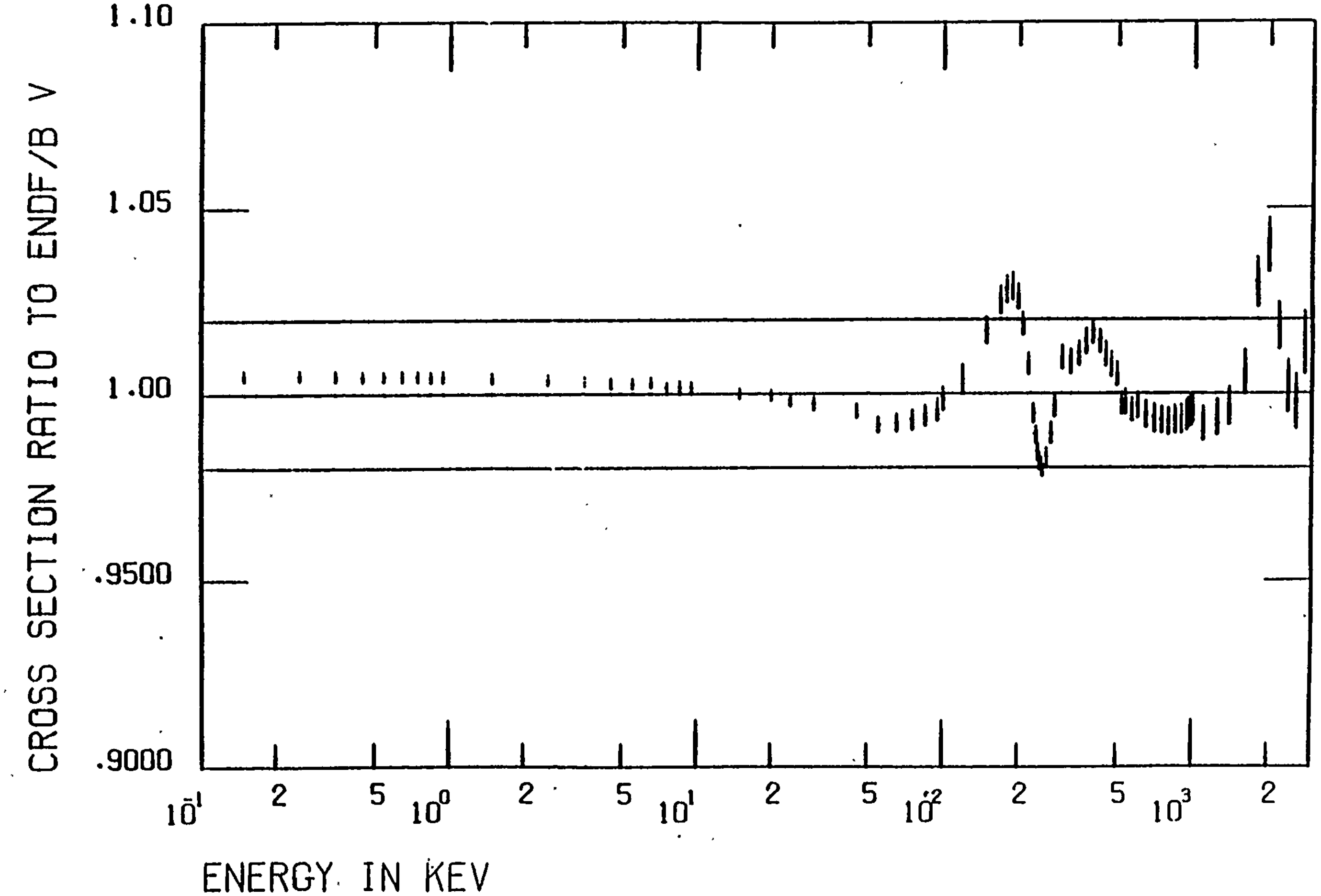

Fig. 26 Ratio of the result of this evaluation process to that of ENDF/B-V for $^{6}Li(n,t)$ from about 0.1 keV to 3 MeV. The uncertainties shown are those obtained from the combination output. The lines at ratios of 0.98 and 1.02 are guides to the eye.

$^{10}B(n,\alpha)$

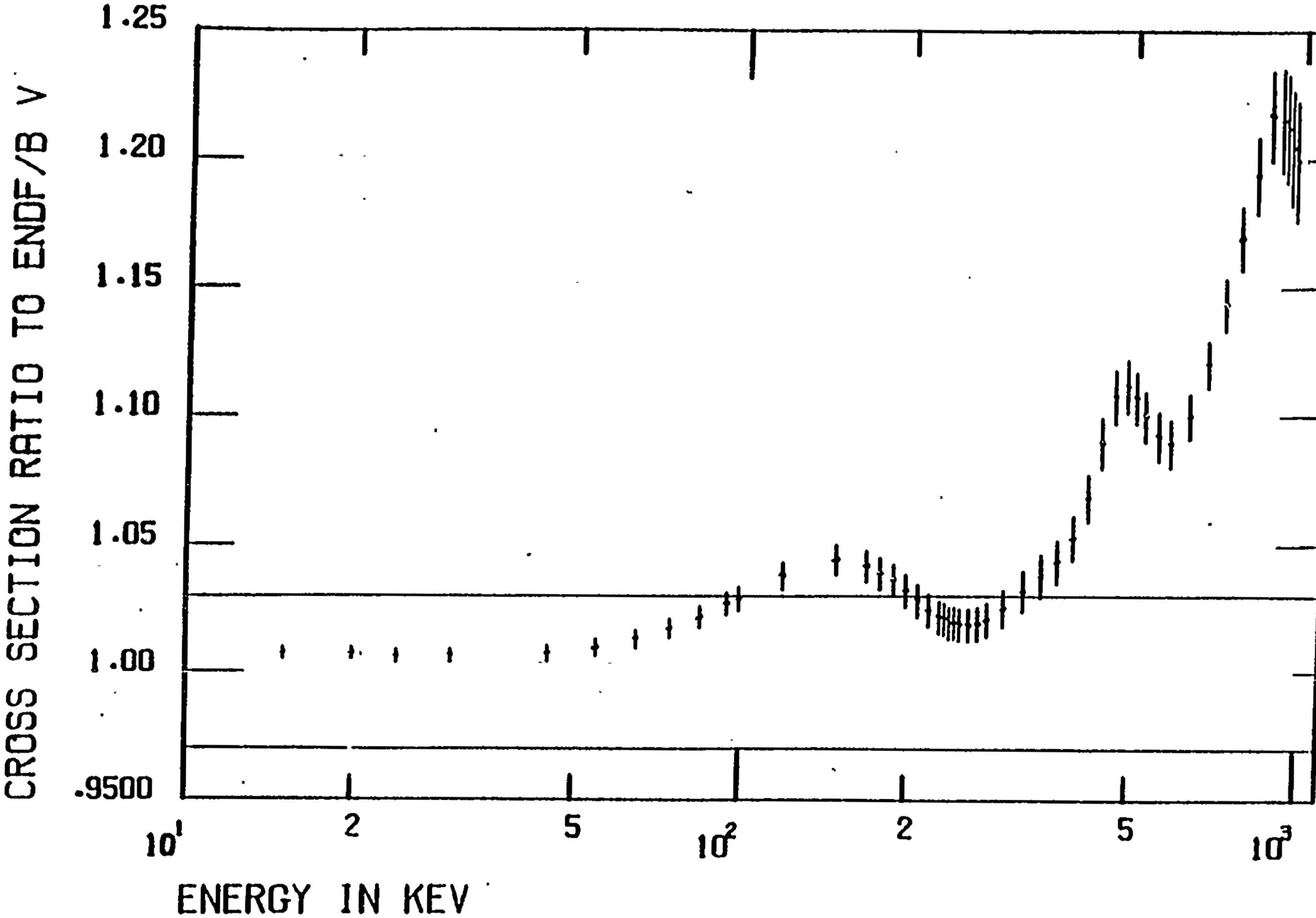

Fig. 27 Ratio of the result of this evaluation process to that of ENDF/B-V for the $^{10}B(n,\alpha)$ reaction from about 10 keV to 1 MeV. The lines at ratios of 0.97 and 1.03 are guides to the eye.

$^{10}B(n,\alpha_1)$

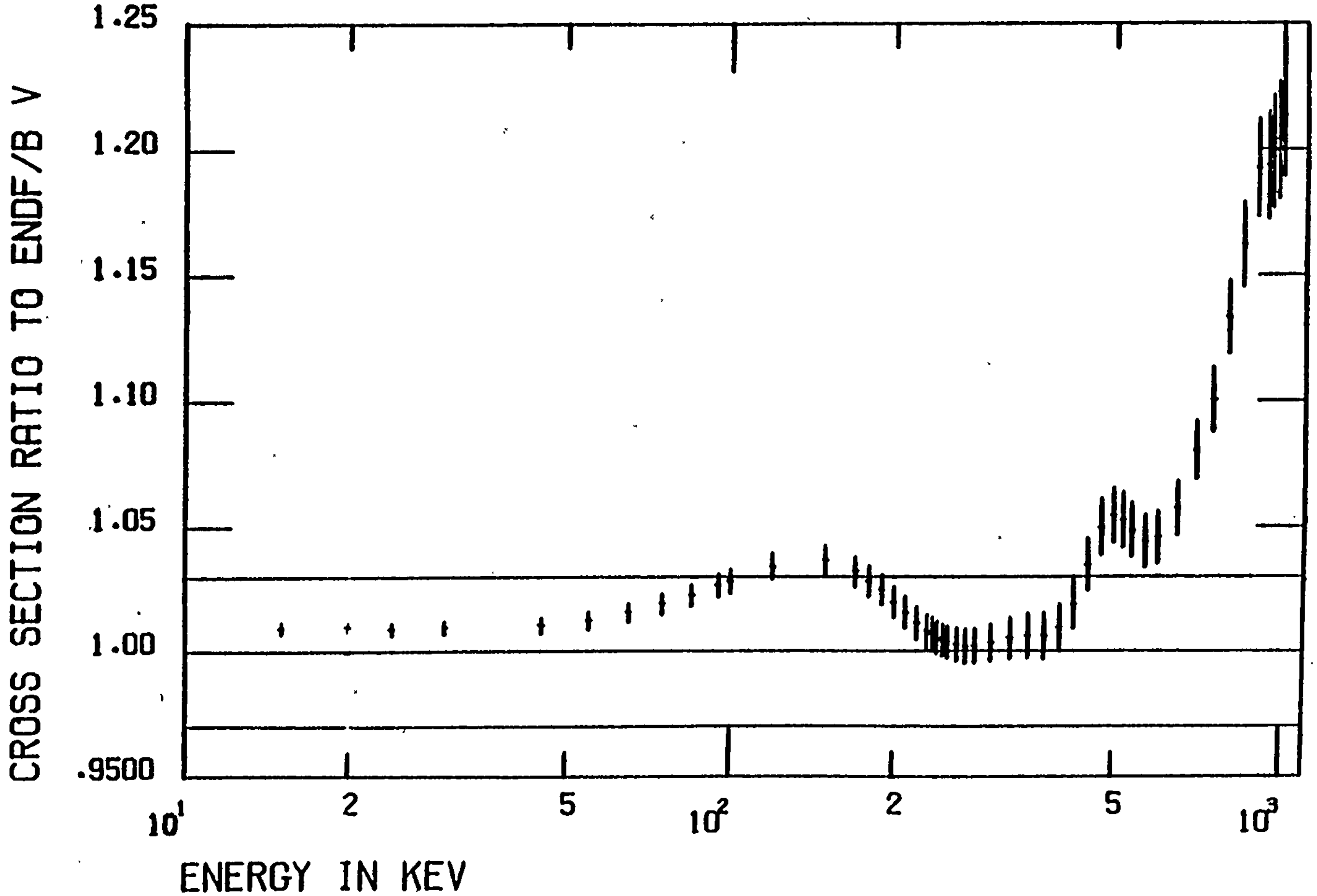

Fig. 28 Ratio of the result of this evaluation process to that of ENDF/B-V for the $^{10}B(n,\alpha_1)$ reaction from about 10 keV to 1 MeV. The lines at ratios of 0.97 and 1.03 are guides to the eye.

Au(n,γ)

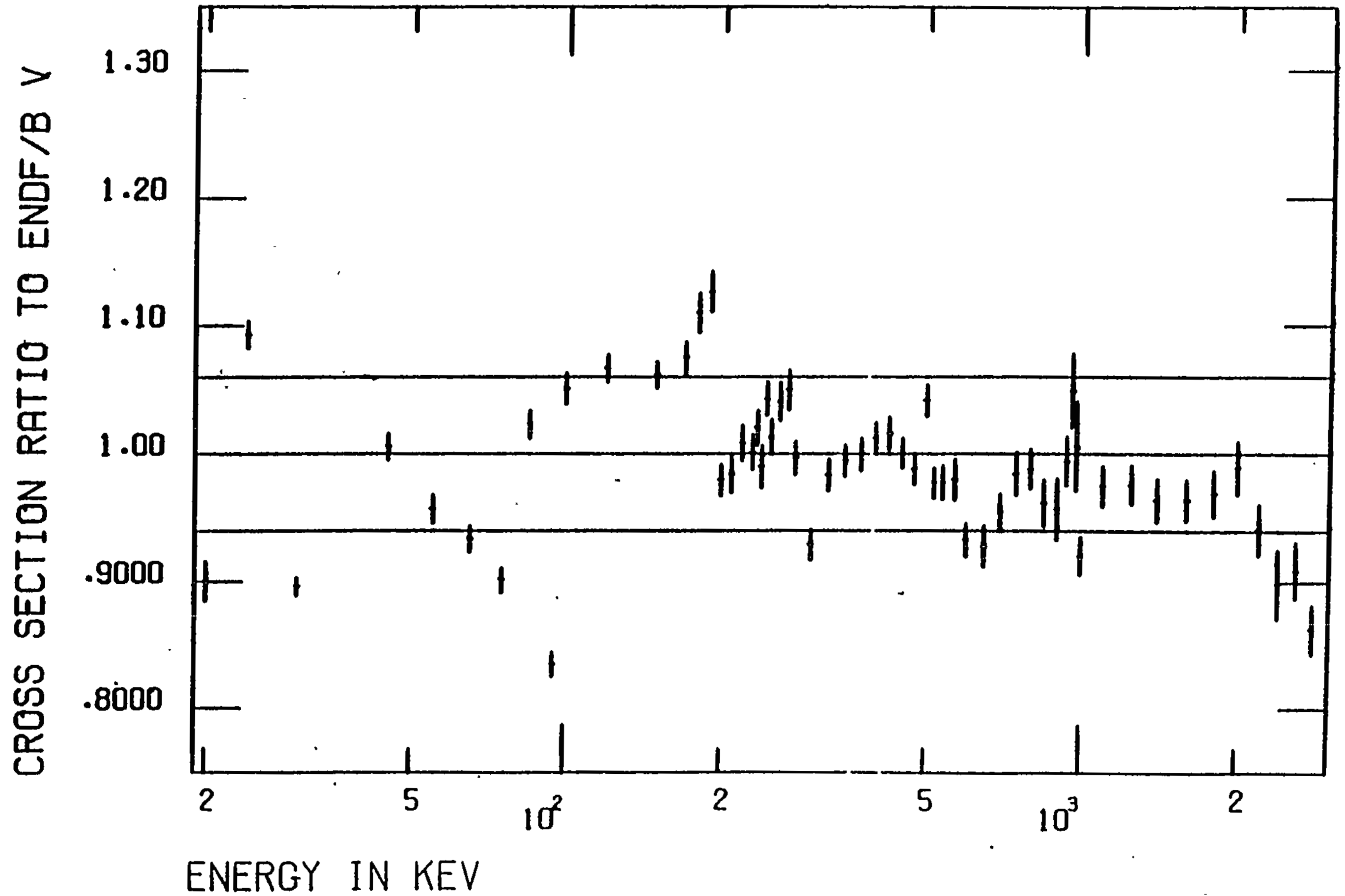

Fig. 29 Ratio of the unsmoothed result of this evaluation process to that of ENDF/B-V for the Au(n,γ) reaction from about 20 keV to 3 MeV. The lines at ratios of 0.94 and 1.06 are guides to the eye.

$^{235}U(n,f)$

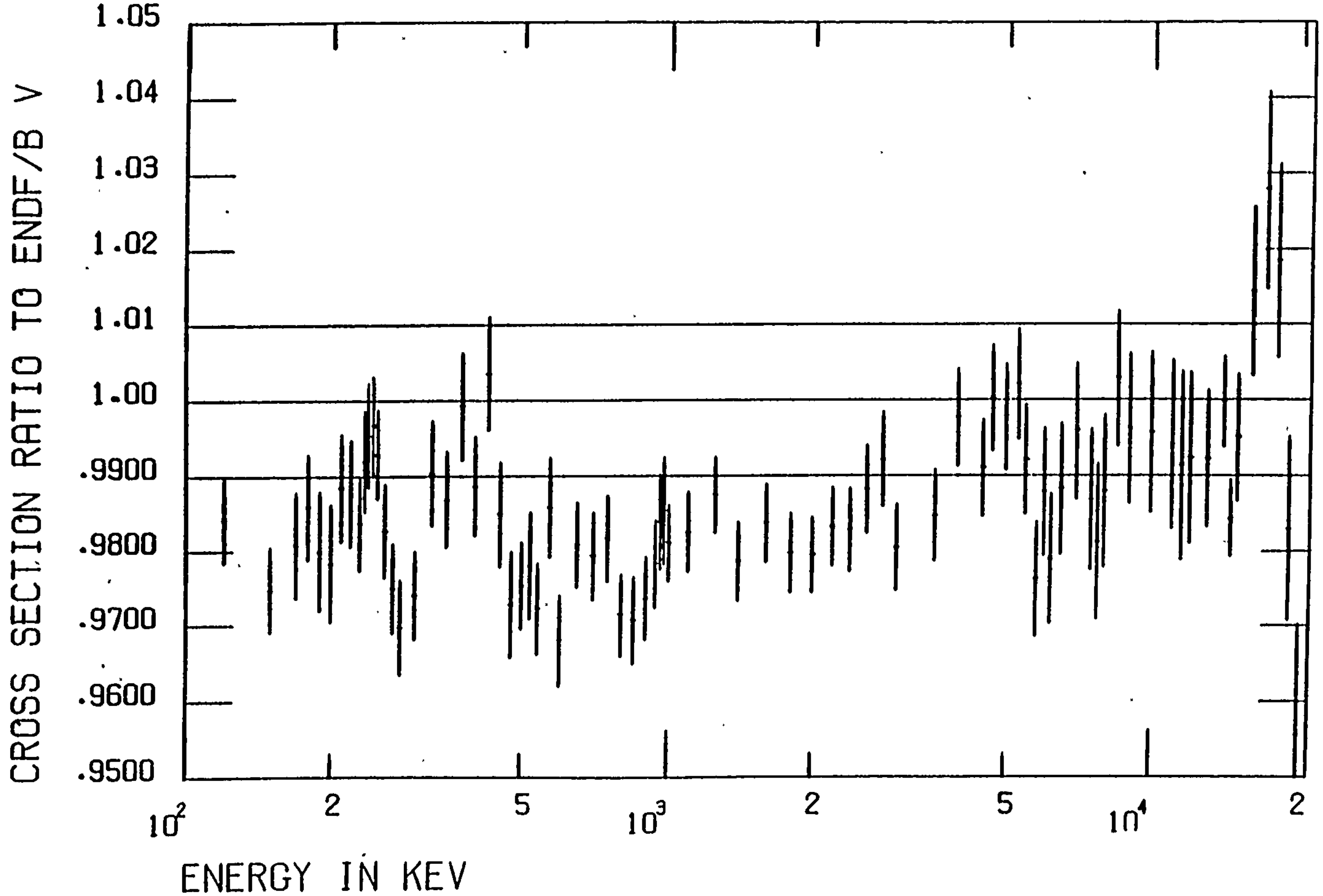

Fig. 30 Ratio of the unsmoothed result of this evaluation process to that of ENDF/B-V for the $^{235}U(n,f)$ reaction from about 100 keV to 20 MeV. The lines at ratios of 0.99 and 1.01 are guides to the eye.

$^{238}U(n,f)$

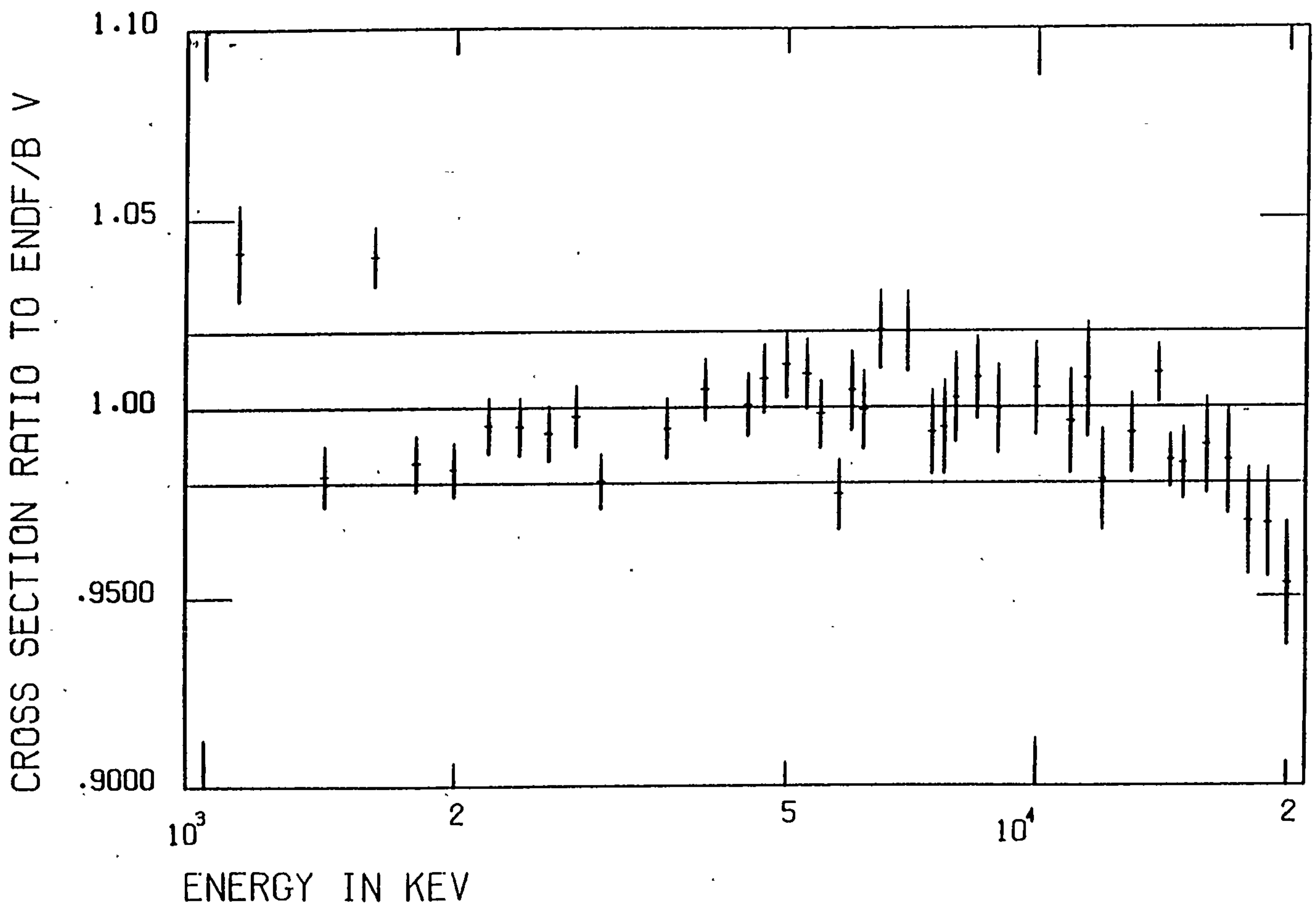

Fig. 31 Ratio of the unsmoothed result of this evaluation process to that of ENDF/B-V for the $^{238}U(n,f)$ reaction from about 1 MeV to 20 MeV. The lines at ratios of 0.98 and 1.02 are guides to the eye.

$^{238}U(n,\gamma)$

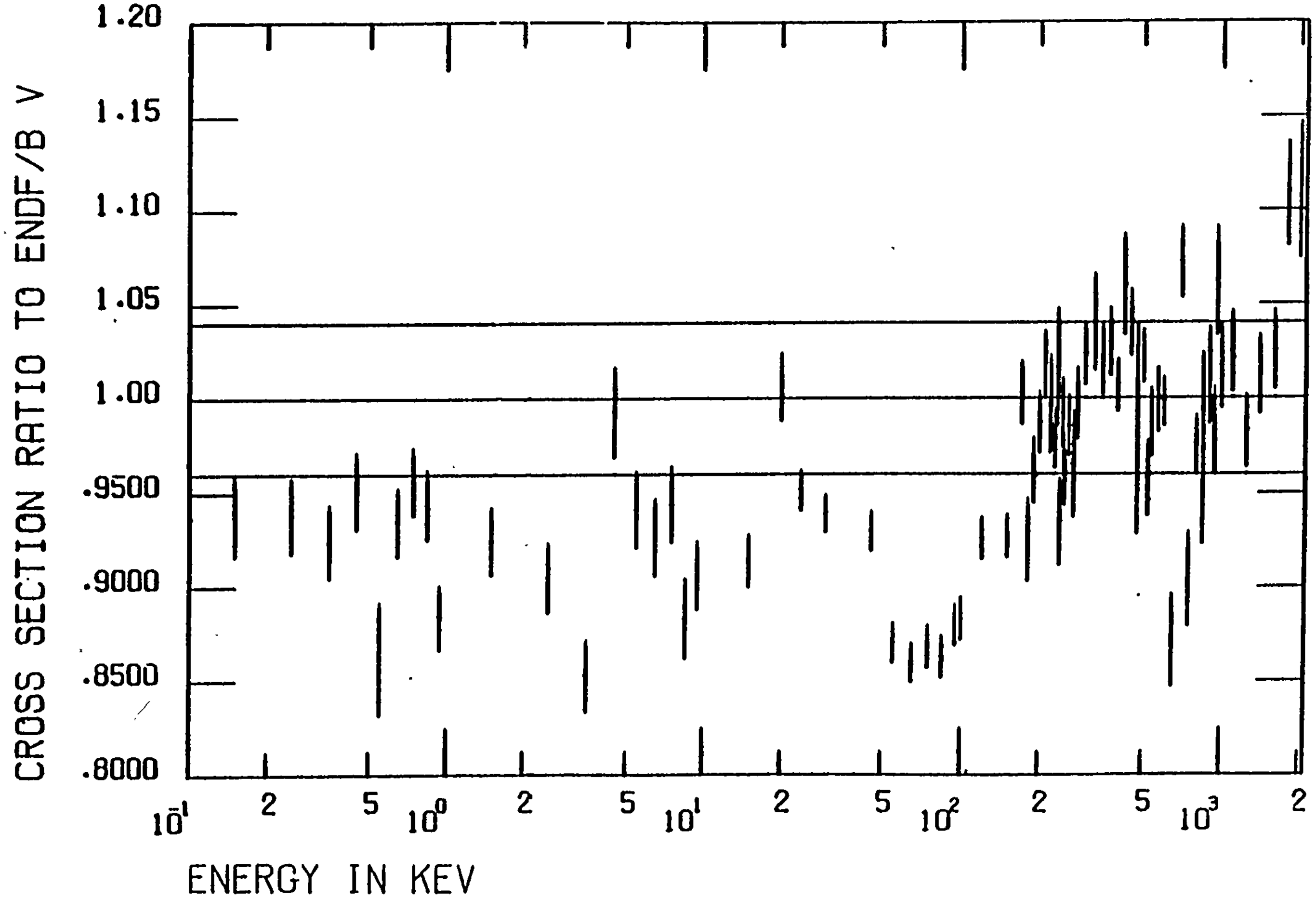

Fig. 32 Ratio of the unsmoothed result of this evaluation process to that of ENDF/B-V for the $^{238}U(n,\gamma)$ reaction from about 0.1 keV to 2 MeV. The lines at ratios of 0.96 and 1.04 are guides to the eye.

$^{239}Pu(n,f)$

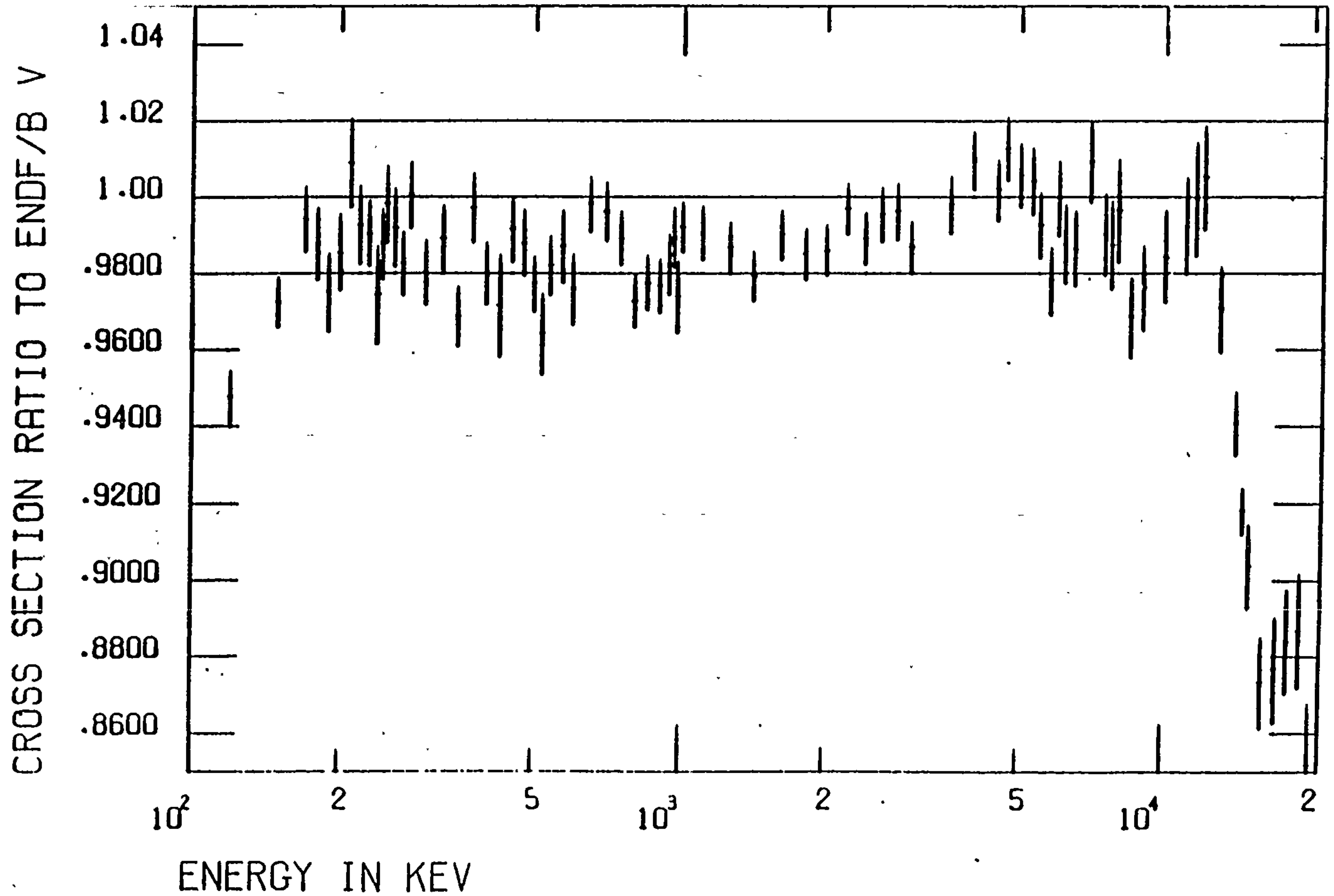

Fig. 33 Ratio of the unsmoothed result of this evaluation process to that of ENDF/B-V for the $^{239}Pu(n,f)$ reaction from about 100 keV to 20 MeV. The lines at ratios of 0.98 and 1.02 are guides to the eye.

APPENDIX A. References for the Data Base Used for the Simultaneous Evaluation

Adamchuk, Yu V., *et al.* (1977) Conf. on Neutron Physics, Kiev, **2**, 192; ^{238}U(n,γ)/^{10}B(n,α_1) [445]

Adamov, V.M., *et al.* (1983) Int'l Nucl Data Comm. INDC(USSR)-180L; ^{235}U(n,f) Cf-AV [575]; also Adamov, V.M., *et al.* (1979) Radium Inst. Report RI-52; transl. in INDC(CCP)-180L; ^{235}U(n,f) Cf-AV [575]

Adams. B., *et al.* (1961) J. Nucl. Energy A/B **14**, 85; ^{235}U(n,f)/^{238}U(n,f), shape [836]; ^{238}U(n,f), shape [835]; ^{239}Pu(n,f)/^{238}U(n,f), shape [837]

Alfimenkov, V.P., *et al.* (1982) Nuclear Data for Science and Technology, Antwerp, 353; also Published in Yad. Fiz. **36**, 1089 (1982); transl: Sov. J. Nucl. Phys. **36**, 637; ^{6}Li(n,n) [210]

Allen, W.D. and Ferguson, A.T.G. (1957) Proc. Phys. Soc. **LXX**, 573; ^{239}Pu(n,f), shape [671]; ^{235}U(n,f) [735]; ^{235}U(n,f), shape [568]; ^{239}Pu(n,f) [672]

Andersson, P., *et al.* (1985) Nucl. Phys. **A443**, 404; ^{197}Au(n,γ)/^{235}U(n,f) [363]: {These measurements were relative to indium capture. They were converted using measurements made by D.L. Smith and J.W. Meadows ANL-NDM-14}.

Arlt, R., *et al.* (1984) Nuclear Standard Reference Data, Geel, IAEA-TECDOC-335; 174; ^{235}U(n,f) [590]

Asami, A. and Moxon, M.C. (1969) Harwell Report AERE-R-5980; also J. Nucl. Energy **24**, 85 (1970); ^{10}B(n,n) [178]
Asami, A. and Moxon, M.C. (1970) Nuclear Data for Reactors, Helsinki, STI/PUB/259, I, 153; ^{6}Li(n,n) [208]

Auchampaugh, G.F., *et al.* (1979) Nucl. Sci. Eng. **69**, 30; ^{10}B(tot) [180]

Axton, E.J. (1985) private communication; g_a ^{233}U [910]; g_f ^{233}U [911]; ^{233}U(n,n) [912]; ^{233}U(n,f) [913]; ^{233}U(n,γ) [914]; ν ^{233}U [915]; g_a ^{235}U [916]; g_f ^{235}U [917]; ^{235}U(n,n) [918]; ^{235}U(n,f) [919]; ^{235}U(n,γ) [920]; ν ^{235}U [921]; g_a ^{239}Pu [922]; g_f ^{239}Pu [923]; ^{239}Pu(n,n) [924]; ^{239}Pu(n,f) [925]; ^{239}Pu(n,γ) [926]; ν ^{239}Pu [927]; g_a ^{241}Pu [928]; g_f ^{241}Pu [929]; ^{241}Pu(n,n) [930]; ^{241}Pu(n,f) [931]; ^{241}Pu(n,γ) [932]; ν ^{241}Pu [933]; ν ^{252}Cf [934]

Barry, J.F., *et al.* (1964) J. Nucl. Energy A/B **18**, 481; ^{238}U(n,γ)/^{235}U(n,f) [415]
Barry, J.F. (1966) Neutron Cross Section Technology, Washington DC, AEC Report Conf-660303, **2**, 763; ^{6}Li(n,α)/^{235}U(n,f), shape [288]

Bartle, C.M. (1979) Nucl. Phys. **A330**, 1; ^{6}Li(n,α) [238]

Barton, D.M., *et al.* (1976) Nucl. Sci. Eng. **60**, 369; ^{235}U(n,f) [580]

Bastian, C. and Riemenschneider, H. (1984) Nuclear Standard Reference Data, Geel, IAEA-TECDOC-335, 118; ^{6}Li(n,α)/^{10}B(n,α), shape [297]

Becker, R.L. and Barschall, H.H. (1956) Phys. Rev. **102**, 1384; ^{10}B(tot) [185]

Beer, H. and Spencer, R.R. (1979) Nucl. Sci. Eng. **70**, 98; ^{10}B(tot) [192#,193#]

Behrens, J.W. and Carlson, G.W. (1977) Nucl. Sci. Eng. **63**, 250; ^{238}U(n,f)/^{235}U(n,f) [805]

Belanova, T.S., *et al.* (1966) J. Nucl. Energy A/B **20**, 411; ^{238}U(n,γ) [435]: {with reinterpretation by Miller, L.B. and Poenitz, W.P. (1969) Nucl. Sci. Eng. **35**, 295, and Dietze, K. (1977) in this Appendix}.

Bergman, A.A., *et al.* (1957) Zh. Eksp. Teor. Fiz. **33**, 9; transl. in Sov. Phys. JETP **6**, 6; ^{6}Li(n,α)/^{10}B(n,α), shape [160]

Bergman, A.A., *et al.* (1980) Conf. on Neutron Physics, Kiev, **3**, 49; ^{235}U(n,f)/^{10}B(n,α), shape [550,552]; ^{239}Pu(n,f)/^{10}B(n,α), shape [551]

Berezin, A.A., *et al.* (1958) At. Energ. **5**, 659; transl. in Sov. At. En. **5**, 1604; ^{235}U(n,f)/^{238}U(n,f) [870]

Bichsel, H. and Bonner, T.W. (1957) Phys. Rev. **108**, 1025; ^{10}B(n,$\alpha_0+\alpha_1$), shape [124]

Block, R.C., *et al.* (1972) New Developments in Reactor Physics and Shielding, Kiamesha Lake, New York, CONF-720901, **2**, 1107; ^{238}U(n,γ)/^{197}Au(n,γ) [470]

Blons, J. (1973) Nucl. Sci. Eng. **51**, 130; ^{235}U(n,f)/^{10}B(n,α), shape [718]; ^{239}Pu(n,f)/^{10}B(n,α), shape [719]

Blons, J., *et al.* (1977) Private Communication, ^{238}U(n,f)/^{235}U(n,f), shape [828]

Bockelman, C.K., *et al.* (1951) Phys. Rev. **84**, 69; ^{10}B(tot) [182]

Bogart, D. and Nichols, L.L. (1969) Nucl. Phys. **A125**, 463; ^{10}B(n,$\alpha_0+\alpha_1$), shape [130]

Bollinger, L., *et al.* (1958) Peaceful uses of Atomic Energy, Geneva, **15**, 127; ^{239}Pu(n,f)/^{10}B(n,α), shape [678]

Bowman, C.D., *et al.* (1963) Phys. Rev. **130**, 1482; ^{235}U(n,f)/^{10}B(n,α), shape [732]

Cancé, M. and Grenier, G. (1976) Fast Neutron Fission Cross Sections of ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu, Argonne Nat'l Lab., ANL-76-90, 141; ^{238}U(n,f)/^{235}U(n,f) [832]

Cancé, M. and Grenier, G. (1978) Nucl. Sci. Eng. **68**, 197; ^{235}U(n,f) [596]; ^{239}Pu(n,f) [612]; ^{238}U(n,f) [812]

Cancé, M. and Grenier, G. (1981) Saclay Report CEA-N-2194; ^{235}U(n,f) [597]

Cancé, M. and Grenier, G. (1983) private communication; ^{235}U(n,f) [598]

Carlson, A.D. and Patrick, B.H. (1978) Neutron Physics and Nuclear Data for Reactors and other Applied Purposes, Harwell, 880; ^{235}U(n,f), shape [508-509]

Carlson, A.D., *et al.* (1984) Nuclear Standard Reference Data, Geel, IAEA-TECDOC-335, 162; ^{235}U(n,f) [523]

Carlson, G.W. and Behrens, J.W. (1978) Nucl. Sci. Eng. **66**, 205; ^{239}Pu(n,f)/^{235}U(n,f) [600]

Chen Ying, *et al.* (1982) Nuclear Data for Science and Technology, Antwerp, 462; also Chinese J. Nucl. Phys. 3, 52 (1981); $^{197}Au(n,\gamma)$ [370]; $^{197}Au(n,\gamma)$, shape [371]

Cierjacks, S., *et al.* (1976) Fast Neutron Fission Cross Sections of ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu,, ANL-76-90, 94; $^{238}U(n,f)/^{235}U(n,f)$, shape [824]

Clements, P.J. and Rickard, I.C. (1972) Harwell Report AERE-R7075; $^{6}Li(n,\alpha)$, shape [280,281]; $^{6}Li(n,\alpha)/^{238}U(n,f)$, shape [280,281]

Coates, M.S., *et al.* (1972) Neutron Standard Ref. Data, Vienna, STI/PUB/371, 129; $^{10}B(n,\alpha_1)$, shape [128]

Coates, M.S., *et al.* (1972) Neutron Standard Ref. Data, Vienna, STI/PUB/371, 105; $^{6}Li(n,\alpha)$, shape [205]

Coates, M.S., *et al.* (1975) Nuclear Cross Sections and Technology, Washington, DC, NBS Spec. Publ. 425, II, 568; also private communication; $^{238}U(n,f)/^{235}U(n,f)$, shape [826]

Condé, H., *et al.* (1964) Arkiv Fysik 29, 45; $^{6}Li(n,\alpha)$ [198]

Condé, H., *et al.* (1982) Nuclear Data for Science and Technology, Antwerp, 447; $^{6}Li(n,\alpha)$, shape [226]

Coon, J.H., *et al.* (1952) Phys. Rev. 88, 562; $^{10}B(tot)$ [183]

Corvi, F. (1983) Priv. Comm., $^{235}U(n,f)/^{6}Li(n,\alpha)$, shape [530,531]

Cox, S.A. and Pontet, F.R. (1966) J. Nucl. Energy 21, 271; $^{10}B(n,\alpha_0+\alpha_1)$ [115]

Czirr, J.B. and Carlson, A.D. (1980) Nuclear Cross Sections for Technology, Knoxville, NBS Spec. Publ. 594, 84; $^{6}Li(n,\alpha)/^{10}B(n,\alpha)$, shape [120]

Czirr, J.B. and Carlson, G.W. (1977) Nucl. Sci. Eng. 64, 892; $^{235}U(n,f)/^{6}Li(n,\alpha)$, shape [271,272]

Czirr, J.B. and Sidhu, G.S. (1975) Nucl. Sci. Eng. 57, 18; $^{235}U(n,f)$, shape [510]

Czirr, J.B. and Sidhu, G.S. (1975) Nucl. Sci. Eng. 58, 371; $^{235}U(n,f)$, shape [511]

Czirr, J.B. and Sidhu, G.S. (1976) Nucl. Sci. Eng. 60, 383; $^{6}Li(n,\alpha)/^{235}U(n,f)$, shape [270]

Czirr, J.B. and Stelts, M.L. (1973) Nucl. Sci. Eng. 52, 299; $^{197}Au(n,\gamma)/^{235}U(n,f)$, shape [378]

Davis, E.A., *et al.* (1961) Nucl. Phys. 27, 448; $^{10}B(n,\alpha)$, abs [121]; $^{10}B(n,\alpha_0)/^{10}B(n,\alpha_1)$ [122]

Davis, M.C., *et al.* (1978) Annals Nucl. Energy 5, 569; $^{235}U(n,f)$ [564]; $^{239}Pu(n,f)$ [640]

Davis, M.C. and Knoll, G.F. (1978) Annals Nucl. Energy 5, 583; $^{235}U(n,f)$ Cf-AV [565]; $^{239}Pu(n,f)$ Cf-AV [641]

Davletshin, A.N., *et al.* (1980) At. Energ. 48, 87; transl. in Sov. J. At. En. 48, 97; $^{197}Au(n,\gamma)$ [350]; $^{238}U(n,\gamma)$ [436]

Debus, G.H. and De Bievre, P.J. (1967) J. Nucl. Energy 21 373; $^{10}B(n,\alpha)$ [708#]

Deruytter, A.J. and Wagemans, C. (1971) J. Nucl. Energy 25, 263; $^{235}U(n,f)/^{10}B(n,\alpha)$, shape [731]

De Saussure, G. and Weston, L.W. (1963) Oak Ridge Report ORNL-3360, 51; $^{238}U(n,\gamma)/^{235}U(n,f)$ [478]

De Saussure, G., *et al.* (1966) Nuclear Data for Reactors, Paris, **II**, 233; ^{235}U(n,f)/^{10}B(n,α), shape [578,730]
De Saussure, G., *et al.* (1973) Nucl. Sci. Eng., **51**, 385; ^{238}U(n,γ)/^{10}B(n,α) [408]
De Saussure, G., *et al.* (1978) Oak Ridge National Lab, ORNL/TM-6152; ^{238}U(n,γ) evaluation [480]

Dietze, K. (1977) Zentralinst. f. Kernforsch., Rossendorf, Report ZFK-341; ^{238}U(n,γ) [432]

Difilippo, F.C., *et al.* (1978) Nucl. Sci. Eng. **68**, 43; ^{238}U(n,f)/^{235}U(n,f) [808]

Diment, K.M. (1967) Harwell Report AERE-R-5224; ^{10}B(tot) [194#,195#,196#]

Diven, B.C. (1957) Phys. Rev. **105**, 1350; ^{235}U(n,f), shape [572]; ^{235}U(n,f) [573]

Dushin, V.N., *et al.* (1983) Proc. IAEA Consultants' Meeting, The ^{235}U Fast-Neutron Fission Cross Section, and the ^{252}Cf Fission Neutron Spectrum, Smolenice, July 1983, INDC(NDS)-146 53; ^{235}U(n,f) [591-595]; ^{238}U(n,f) [811]; ^{239}Pu(n,f) [611]

Ferguson, A.T.G. and Paul, E.B. (1959) J. Nucl. Eng. A **10**, 19; ^{197}Au(n,γ) [355]

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Fort, E. (1970) Nuclear Data for Reactors, Helsinki, STI/PUB/259; ^{6}Li(n,α) [294]
Fort, E. and Marquette, J.P. (1972) European-American Nucl. Data Committee Doc. EANDC(E) 148"U"; ^{6}Li(n,α) [290-292]
Fort, E. and Rigoleur, C.Le (1975) Nuclear Cross Sections and Technology, Washington, DC, NBS Spec. Publ. 425, 957; ^{197}Au(n,γ) [345]

Fricke, M.P., *et al.* (1970) Nuclear Data for Reactors, Helsinki June 1970, STI/PUB/259, **2**, 265; ^{197}Au(n,γ)/^{10}B(n,α) [300]; ^{197}Au(n,γ), shape [301]
Fricke, M.P., *et al.* (1971) Neutron Cross Section and Technology, Knoxville, CONF-710301, **1**, 252; ^{238}U(n,γ)/^{10}B(n,α) [400]; ^{238}U(n,γ), shape [401]

Friesenhahn, S.J., *et al.* (1974) Intelcom Radiation Technology Report INTEL-RT 7011-001; ^{10}B(n,$\alpha_0+\alpha_1$), shape [100]; ^{10}B(n,α_1), shape [103]; ^{6}Li(n,α), shape [246]

Fursov, B.I., *et al.* (1977) At. Energ. **43**, 261; transl. in Sov. J. At. En. **43**, 891; ^{239}Pu(n,f)/^{235}U(n,f) [653]; ^{239}Pu(n,f)/^{235}U(n,f) shape [654]
Fursov, B.I., *et al.* (1977) At. Energ. **43**, 181; transl. in Sov. J. At. En. **43**, 808; ^{238}U(n,f)/^{235}U(n,f) [844]; ^{238}U(n,f)/^{235}U(n,f) shape [845]

Garlea, I., *et al.* (1983) Int'l Nucl. Data Comm. Rep. INDC(ROM)-15, Rev. Roumaine Phys. **26**, 643; ^{239}Pu(n,f)/^{235}U(n,f) [633]; ^{238}U(n,f)/^{235}U(n,f) [863]

Gayther, D.B. (1975) Neutron Cross Sections and Technology, Washington, DC, NBS Spec. Publ. 425, **2**, 564; ^{235}U(n,f), shape [588]; ^{239}Pu(n,f), shape [589]
Gayther, D.B. (1977) Annals Nucl. Energy **4**, 515; ^{6}Li(n,α)/^{235}U, shape [261]

Gibbons, J.G. and Macklin, R.L. (1959) Phys. Rev. **114**, 571; $^{10}B(n,\alpha_0)$ [114]

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Pankratov, V.M., *et al.* (1964) At. Energ. **14**, 177; transl. in Sov. J. At. En. **14**, 167; $^{235}U(n,f)$, shape [722]; $^{238}U(n,f)$, shape [874]

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Weston, L.W. and Todd, J.H. (1984) Nucl. Sci. Eng. **88**, 567; $^{235}U(n,f)/^{10}B(n,\alpha)$, shape [532]; $^{235}U(n,f)/^{6}Li(n,\alpha)$, shape [533]; $^{239}Pu(n,f)/^{10}B(n,\alpha)$, shape [534]; $^{239}Pu(n,f)/^{6}Li(n,\alpha)$, shape [535]

White, P.H. (1965) J. Nucl. Energy A/B **19**, 325; $^{235}U(n,f)$ [499-502]

White, P.H., *et al.* (1965) Physics and Chemistry of Fission, Salzburg, **I**, 219; $^{239}Pu(n,f)/^{235}U(n,f)$ [608]

White, P.H. and Warner, G.P. (1967) J. Nucl. Energy **21**, 671; $^{239}Pu(n,f)/^{235}U(n,f)$ [609]; $^{238}U(n,f)/^{235}U(n,f)$ [815]

Willard, H.B., *et al.* (1955) Phys. Rev. **98**, 669; $^{10}B(n,n)$ [175]

Wisshak, K. and Käppeler, F. (1978) Nucl. Sci. Eng. **66**, 363; $^{238}U(n,\gamma)/^{197}Au(n,\gamma)$ [430,431]

Wu Jingxia, *et al.* (1983) Chinese J. Nucl. Phys. **5**, 158; $^{238}U(n,f)$ [850]

Yamamuro, N., *et al.* (1978) J. Nucl. Sci. Tech. (Japan) **15**, 637; $^{238}U(n,\gamma)/^{10}B(n,\alpha)$ [423]

Yamamuro, N., *et al.* (1980) J. Nucl. Sci. Tech. (Japan) **17**, 582; $^{238}U(n,\gamma)/^{10}B(n,\alpha_1)$, shape [422]

Yamamuro, N., *et al.* (1983) J. Nucl. Sci. Tech. (Japan) **20**, 797; $^{197}Au(n,\gamma)/^{10}B(n,\alpha_1)$ [340]; $^{197}Au(n,\gamma)/^{10}B(n,\alpha_1)$, shape [341]

Yan Wuguang, *et al.* (1975) At. Energ. Sci. Tech. (1975)#2, 1; $^{235}U(n,f)$ [738]

Yoshida, K., *et al.* (1983) NETU-44 (Tohoku), 30; $^{235}U(n,f)$ [528]; $^{238}U(n,f)$ [528]

Zhagrov, E.A., *et al.* (1980) Conf. on Neutron Physics, Kiev, 3, 45; $^{235}U(n,f)$ [525]

Zhuravlev, K.D., *et al.* (1977) At. Energ. **42**, 56; transl. in Sov. J. At. En. **42**, 62; $^{235}U(n,f)/^{10}B(n,\alpha)$, shape [515]; $^{239}Pu(n,f)/^{10}B(n,\alpha)$, shape [630]; $^{239}Pu(n,f)/^{10}B(n,\alpha)$ [631]

These data sets were excluded from the simultaneous evaluation so they would be available for use in the R-matrix analyses.

APPENDIX B. References for the Data Base Used for the R-matrix Evaluations

^{7}Li System Data

Bartle, C. M. (1980) Nucl. Instr. Meth. **176**, 503; ^{6}Li(n,t)^{4}He, $\sigma(\theta)$, E_n=2.2-3.9 MeV.

Brown, R. E., *et al.* (1977) Phys. Rev. C **16**, 513; ^{6}Li(n,t)^{4}He, $\sigma(\theta)$, E_n=87-398 keV.

Condé, H., *et al.* (1982) Proc. Antwerp Conf. on Nuclear Data for Science and Technology, p. 447; ^{6}Li(n,t)^{4}He, $\sigma(\theta)$, E_n=1.3-3.5 MeV.

Drigo, L. and Tornielli, G. (1982) Nuovo Cimento **70A**, 402; ^{6}Li(n,n)^{6}Li, $A_y(\theta)$, E_n=1.5-4.0 MeV.

Drosg, M., *et al*, (1982) Los Alamos report LA-9129-MS; ^{4}He(t,n)^{6}Li, $\sigma(\theta)$, E_t=8.5-12.9 MeV.

Drosg, M., *et al*, (1982) Los Alamos report LA-9129-MS; ^{4}He(t,n_1)^{6}Li*, $\sigma(\theta)$, E_t=12.9 MeV.

Hardekopf, R. A., *et al.* (1977) Los Alamos Report LA-6188; ^{4}He(t,t)^{4}He, $\sigma(\theta)$, $A_y(\theta)$, E_t=7-14 MeV.

Harvey, J. A. and Hill, N. W. (1975) Proc. Conf. on Nuclear Cross Sections & Technology, NBS Spec. Pub. **425**, 244, and private communication from J. Harvey; ^{6}Li(n,n)^{6}Li, σ_T(E), E_n=10 eV-4 MeV.

Ivanovich, M., Young, P. G., and Ohlsen, G. G. (1968) Nucl. Phys. **A110**, 441; ^{4}He(t,t)^{4}He, $\sigma(\theta)$, E_t=1-7 MeV.

Jarmie, N., *et al.* (Nov. 1980) Los Alamos Report LA-8492; ^{4}He(t,t)^{4}He, $\sigma(\theta)$, $A_y(\theta)$, E_t=6-17 MeV.

Knitter, H. H., *et al.* (1983) Nucl. Sci. Eng. **83**, 229; ^{6}Li(n,t)^{4}He, $\sigma(\theta)$, E_n=0.035-325 keV (relative data).

Knox, H. D., *et al.* (1982) Bull. Am. Phys. Soc. **27**, 723, and private communication from H. Knox (1985); ^{6}Li(n,t)^{4}He, $\sigma(\theta)$, E_n=2-3.5 MeV.

Lane, R. O. (1961) Ann. Phys. **12**, 135; ^{6}Li(n,n)^{6}Li, $\sigma(\theta)$, E_n=0.05-1.44 MeV.

Lane, R. O. (1964) Phys. Rev. **136**, B1710; ^{6}Li(n,n)^{6}Li, $A_y(\theta)$, E_n=0.2-1.7 MeV.

Meadows, J. W. (1971) *Neutron Standards and Flux Normalizations* (AEC **23**), 129; ^{6}Li(n,t)^{4}He, $\sigma(E_{therm})$.

Overley, J. C., *et al.* (1974) Nucl. Phys. **A221**, 573; ^{6}Li(n,t)^{4}He, $\sigma(\theta)$, E_n=0.1-1.9 MeV.

Renner, C., *et al.* (1978) Bull. Am. Phys. Soc. **23**, 526, and private communication from J. Harvey; ^{6}Li(n,t)^{4}He, σ(E), E_n=82-467 keV (renormalized -5%).

Smith, A. B., *et al.* (1982) Nucl. Phys. **A373**, 305; ^{6}Li(n,n)^{6}Li, $\sigma(\theta)$, E_n=1.5-3.7 MeV.

Spiger, R. J. and Tombrello, T. A. (1967) Phys. Rev. **163**, 964; ^{4}He(t,t)^{4}He, σ_T(E), E_α=12-18 MeV.

Stelts, M. L., *et al.* (1979) Phys. Rev. **C19**, 1159; ^{6}Li(n,t)^{4}He relative $\sigma(\theta)$, E_n=2, 24 keV.

^{11}B System Data

Bockelman, C. K., *et al.* (1951) Phys. Rev. **84**, 69; $^{10}B(n,n)^{10}B$, $\sigma_T(E)$, E_n=0.02-1.01 MeV.

Cusson, R. Y. (1966) Nucl. Phys. **86**, 481; also Thesis, Cal. Tech. (1965); $^{7}Li(\alpha,\alpha)^{7}Li$, $\sigma(\theta)$, E_α=3-6 MeV.

Cusson, R. Y. (1966) Nucl. Phys. **86**, 481; also Thesis, Cal. Tech. (1965); $^{7}Li(\alpha,\alpha_1)^{7}Li^*$, $\sigma(\theta)$ $\sigma(E)$, E_α=3-6 MeV.

Diment, K. M. (1967) Harwell Report AERE-R-5224; $\sigma_T(E)$, E_n=0.076 keV−1 MeV.

Kavanagh, R. W. and Marcley, R. G. (1987) Phys. Rev. C **36**, 1194; $^{10}B(n,t)2\alpha$, $\sigma(E_{therm})$.

Lane, R. O., *et al.* (1971) Phys. Rev. C **4**, 380; $^{10}B(n,n)^{10}B$, $\sigma(\theta)$, $A_y(\theta)$, E_n=0.1-1.0 MeV.

Olson, M. D. and Kavanagh, R. W. (1984) Phys. Rev. C **30**, 1375; $^{7}Li(\alpha,n)^{10}B$, $\sigma(E)$ E_α=4.4-5.5 MeV.

Sealock, R.M. and Overley, J. C. (1976) Phys. Rev. C **13**, 2149; $^{10}B(n,\alpha_1)^{7}Li^*$, $\sigma(\theta)$ E_n=0.2-1.0 MeV.

Spencer, R. R., *et al.* (1973) Report EANDC(E) **147**, A1; $^{10}B(n,n)^{10}B$, $\sigma_T(E)$, E_n=94-411 keV.

Van der Zwaan, L. and Geiger, K. W. (1972) Nucl. Phys. **A180**, 615; $^{10}B(n,\alpha)^{7}Li$, $\sigma(\theta)$ E_n=0.28-0.77 MeV.

Viesti, G. and Liskien, H. (1979) Annals Nucl. Energy **6**, 13; $^{10}B(n,\alpha_1)$, shape, E_n=0.1-2.2 MeV.

APPENDIX C. Memorandum to the Cross Section Evaluation Working Group

May 19, 1986

A TECHNIQUE FOR OBTAINING A GLOBAL FIT COMBINING ADVANTAGEOUS CHARACTERISTIC OF GMA AND EDA

R. W. PEELLE

SUMMARY

Equations are developed that are planned to be used in the combination of the outputs of the EDA analysis of the ^{7}Li and ^{11}B composite nucleus reaction systems at LANL and the GMA simultaneous evaluation at ANL of ^{6}Li(n,α), ^{6}Li(n,n), ^{10}B(n,α_0), ^{10}B(n,α_1), ^{10}B(n,n), ^{197}Au(n,γ), ^{238}U(n,γ), ^{235}U(n,f), ^{239}Pu(n,f), and ^{238}U(n,f) pointwise cross sections.

1. INTRODUCTION

In an earlier letter to CSEWG[1] and in papers by Carlson *et al.*, at Geel and Santa Fe Conferences,[2] a plan was described to combine results from LANL EDA R-matrix analyses[3] and ANL GMA simultaneous evaluations.[4] Since the earlier letter is somewhat vague on details and since the general concept has now been well documented in Refs. 2 and 3, the present note attempts primarily to clarify the development of the equation to be used.

In addition, there are some preliminary comments on the proper inclusion of (a) shape information for the fission cross sections in the thermal energy range, and (b) the thermal fissile constants.

2. THE PROBLEM AND ITS "IDEAL" SOLUTION

Our problem is to employ properly the upgraded data and evaluation techniques now available to produce an excellent evaluation of neutron standard and other important cross sections that we can fully defend. This goal implies the need to (1) evaluate in a consistent way all the standards and other cross sections for which absolute cross sections have been well measured, (2) retain the advantages of use of the R-matrix evaluation tool for the light elements, and (3) obtain the output covariance information corresponding to the data combination method employed.

The problem statement presumes the conclusions that through ratio measurements (1) knowledge of the ^{6}Li cross section affects the ^{235}U(n,f) standard, (2) vice versa, (3) absolute measurements on non-standards such as ^{239}Pu(n,f) are sufficiently precise to affect the standards to some degree. The Standards Evaluation Subcommittee of CSEWG has sought an evaluation strategy that treats these possibilities in a consistent way.

Neglect "practicality" for the moment and consider an ideal solution scheme for the problem posed. Since we do not know all the uses of the evaluated data well enough to know if loss to the sponsors would be more severe for cross section values too high or too low, it is reasonable to seek a solution vector e such that $z^t e$ has minimum variance for any vector z. Subject to general conditions, a least-squares solution meets this criterion provided the input data are weighted with their inverse variance-covariance matrix.[5] Here, the evaluation is parameterized in the vector (e_α) by whatever means are best; R-matrix parameters are components of e along with whatever parameters represent the $^{235}U(n,f)$ and other cross sections as a function of energy. Some parameters may represent the normalizing constants for cross section "shape" measurements. The parameter vector change in one iteration toward this solution is given by the solution of the least-squares matrix equation.

$$G^t V^{-1} G \, \delta e = G^t V^{-1} \delta \bar{y} \tag{1}$$

or equivalent forms, where the $(\bar{y}_i)$ vector stacks up all the significant absolute and ratio data and $\delta\bar{y}$ represents differences between $\bar{y}_i$ and $y_i^0(e^0)$, the calculated value at the same point based on the current best-estimate parameters e_α^0. (V_{ij}) is the variance-covariance matrix of these experimental data including effects of energy uncertainty, and the

$$G_{i\alpha} = \frac{dy_i}{de_\alpha}\Big|_{e=e^0}$$

are the derivatives with respect to the evaluation parameters of the interpolated values for the cross sections at the energies for which these are input data elements. The derivatives are evaluated at the best known parameter set e_α^0. For the approach to be valid, we must demand (1) that the equations relating y to e can be linearized near the solution point, and (2) that any formula from which the derivatives are obtained does represent the observed physical phenomena.

Since the problem is nonlinear, the convenient variables are the indicated relative or absolute differences in e and y from sequential estimates refined in the course of iteration. The corresponding covariance matrix of the output parameters is $C = (G^t V^{-1} G)^{-1}$, a choice that can be defended when the input data scatter is consistent with the data covariance matrix V. The point is that the needed evaluation outputs would be available from such a global least-squares effort if (1) input data covariance matrices are available and sufficiently well-drawn that they can be inverted, (2) the derivatives $G_{i\alpha}$ are available both for R-matrix parameters and pointwise cross section parameters such as are used in the ANL code GMA, (3) the evaluators could devise means to deal with any discrepancies in the data base, and (4) the numerical problem could be put on a machine and solved.

Condition (1) requires considerable evaluator effort and codification of the results if the consequent evaluated cross sections are to be fully credible. W. Poenitz has completed much or all of this effort.[6] Derivatives $G_{i\alpha}$ are now largely available, but not all in the same computer program. Handling discrepancies in the data base must occur in any case; global evaluation may point up discrepancies more sharply. Getting the global least-squares problem on one machine is not incredible for the future; however, as seen in the next section, a global fitting program should not be needed.

Fortunately, many of the measured data sets required for input to the hypothetical least-squares problem are not intercorrelated. This fact greatly increases the tractability of the data variance-covariance matrix and its inverse, and leads to consequent simplification of the numerical problem.[4] The first step of the analysis can proceed separately for each subset or "segment" of the overall data for which the experimental results are not correlated with the data in the other subsets. The complexity of the resulting formulae depends on the degree to which the fitting parameters in this first step can be the same as those for the final evaluation.

The idea of the present approach is to obtain the numerical values of the sums on the right side and the term $G^t V^{-1} G$ on the left of the global least-squares Eq. (1) above by combining values from the existing separate fitting programs set up at LANL and ANL, using "segment 1" and "segment 2" data sets, respectively. It is assumed that we will know at the outset how to estimate an overall parameter vector (e_α^0) very close to the final global solution, so that the ψ^2 minimization can be linearized.

3. SEPARATION OF THE DATA AND PARAMETER VECTORS

First one separates the overall relevant data base into segments not correlated with each other. For our example, segment 1 consists of all the charged-particle data in the ^{7}Li and ^{11}B reaction systems, plus a portion of the neutron cross section data on ^{6}Li and ^{10}B strictly uncorrelated with the second segment. The segment 1 of the data base is further separated into subsegments 1L for the ^{7}Li system and 1B for the ^{11}B system. It is assumed that no experimental value in segment 1A is correlated with any value in segment 1B. The segment 2 data base consists of all the neutron cross section and ratio data for ^{235,238}U(n,f), ^{239}Pu(n,f), ^{6}Li(n,α), etc., from measurements uncorrelated with those in segment 1. Suitable divisions of the global data base have been shown to be possible in practice.[6]

The trial parameter vector for the global fit can be separated into a stack of subvectors as follows:

$$e^0 = \begin{bmatrix} e^0_{LR} \\ e^0_{BR} \\ e^0_P \end{bmatrix} , \tag{2}$$

where e^0_{LR} and e^0_{BR} are respectively the best estimate of the final R-matrix parameters for the lithium and boron reaction systems, and the e^0_P (the σ of Ref. 4) are the best pointwise* cross section parameter estimates of Poenitz, for targets other than ^{6}Li and ^{10}B, together with various experimental data set normalization parameters. [Later we employ e_L and e_B. The vector $e^0_L(e^0_{LR})$ is the set of pointwise estimates for the lithium system that is obtained from the R-matrix parameters e^0_{LR}. The vector e^0_B of pointwise estimates is similarly obtained from e^0_{BR}. However, since for Li and B the R-matrix parameters are sought, e^0_L and e^0_B are derived quantities.]

Note that the R-matrix parameterization generates neutron cross sections covering a wider energy range than that for which it has been applied in the ENDF/B system. Moreover, the ratio and other data in use with GMA for ^{6}Li+n and ^{10}B+n, which we wish to include in the R-matrix evaluation, does not cover the whole range of data types already included in the R-matrix evaluation.

The parameter vector δe of Eq. (1) corresponds to a change from the base parameter vector e^0.

$$\delta e \equiv \varepsilon = \begin{bmatrix} \varepsilon_{LR} \\ \varepsilon_{BR} \\ \varepsilon_P \end{bmatrix} , \tag{3}$$

where $\varepsilon_{LR} = e_{LR} - e^0_{LR}$, $\varepsilon_{BR} = e_{BR} - e^0_{BR}$, and ε_P ($\equiv \delta p$ of Ref. 4) has as components the relative differences $(e_{P,i} - e^0_{P,i})/e^0_{P,i}$ utilized by the GMA code of Poenitz,[4] in the present case only for variables other than the lithium and boron cross sections. The output parameters from an iteration of the grand system are the vector segments e_{LR}, e_{BR}, and e_P.

* "Pointwise" is a modest misnomer; since the GMA system now utilizes in the energy region below 100 keV some parameter definitions corresponding to cross section averages or integrals.

The (δy_i) vector of eq (1) is a stacked difference vector including both segment 1 and segment 2 of the data base.

$$\delta y = \eta = \begin{bmatrix} \eta_{1L} \\ \eta_{1B} \\ \eta_2 \end{bmatrix} . \tag{4}$$

The vector $\eta_{1L} = \bar{y}_{1L} - y_{1L}(e^0_{LR})$ the difference between the segment 1L measurement vector and the calculated vector from the best-guess R-matrix parameters for lithium. The vector segment η_{1B} is defined similarly for the segment 1B boron data and R-matrix computed values. The vector segment η_2 has components

$$\eta_{2i} = \bar{y}_{2i} - y_{2i}(e^0) \quad . \tag{5}$$

The element $y_{2i}(e^0)$ is the value comparable to $\bar{y}_{2i}$ but calculated from the base parameter vector eq (2). Recall that the information of ^{7}Li and ^{11}B reaction systems are contained in R-matrix parameters. The η_{2i} are M_i of Ref. 4 multiplied by the standard deviations of the corresponding observations.

4. SEGMENTING THE DERIVATIVE AND VARIANCE-COVARIANCE MATRICES

The derivative matrix $G = (\partial y_i/\partial \varepsilon_\alpha)|_{\varepsilon=0}$ can be expanded as follows:

$$G = \begin{bmatrix} G_{1LR} & 0 & 0 \\ 0 & G_{1BR} & 0 \\ G_{2L}S_L & G_{2B}S_B & G_{2P} \end{bmatrix} . \tag{6}$$

This matrix has as many columns as the number of output parameters e_α, and as many rows as the elements of the global data base. It is evaluated at the best known parameter set e^0. There are large null submatrices because the interpolated cross sections from the R-matrix fit for the lithium system are not explicit functions of the R-matrix parameters of the boron system or the pointwise cross section parameters for ^{239}Pu, ^{238}U, etc.

In eq (6) the matrix G_{LR} elements are obtained from the R-matrix program

$$G_{1LR,i\alpha} = \frac{\partial y_{1L,i}}{\partial e_{LR,\alpha}}\bigg|_{e_{LR}=e^0_{LR}} , \tag{7}$$

and the elements of G_{BR} are similarly obtained. The matrices G_{1LR} and G_{1BR} are generated in the R-matrix analysis of the segment 1 data.

The elements of G_{2L}, G_{2B}, and G_{2P} take into account the relative nature of the fitting variables of Ref. 4 and correspond to the derivatives of Ref. 4 for the portions of the pointwise data base for ^{6}Li, ^{10}B, and all others, respectively. For example, going back to the definition in Eq. (5)

$$G_{2P,i\alpha} = e_{P\alpha} \frac{\partial y_{2i}(e)}{\partial e_{P\alpha}}\bigg|_{e=e^0} , \tag{8}$$

and

$$G_{2B,i\alpha} = e_{B\alpha} \frac{\partial y_{2i}(e)}{\partial e_{B\alpha}}\bigg|_{e=e^0} . \tag{9}$$

Recall that e_B is a set of pointwise parameters for the ^{11}B reaction system, values of which are computed from the R-matrix parameters for the boron system. In the case of interest to CSEWG, G_{2L}, G_{2B}, and G_{2P} are inherent in the ANL analysis of the segment 2 data. Indeed, the matrix (G_{2L}, G_{2B}, G_{2P}) is the matrix A of Ref. 4 with each element multiplied by the standard deviation of the corresponding observed value, when only segment 2 data is included in the computation of the matrix elements.

Finally, the elements of the matrices S_L and S_B in Eq. (6) are the derivatives of the pointwise relative cross section parameters ε_P with respect to the corresponding R-matrix parameters. For example,

$$S_{L,\alpha\beta} = \frac{1}{e^0_{L,\alpha}} \left(\frac{de_{L,\alpha}}{de_{LR,\beta}}\right)\bigg|_{e=e^0} . \tag{10}$$

The e_L and e_B are just interpolated values from the R-matrix analysis, and so the elements given by Eq. (10) can be computed in a modification of that program. Note from Eqs. (9) and (10) that the product matrices $G_{2L}S_L$ and $G_{2B}S_B$ just yield the derivatives like $\partial y_{2i}/\partial e_{LR,\beta}$ required for Eq. (6).

Because the data segments 1L, 1B, and 2 are each free of observations correlated with observations in either of the other two segments, the data covariance matrix V of Eq. (1) is "block diagonal" and one can directly write V^{-1} in terms of the inverses of submatrices that are the variance-covariance matrices of each of the data segments.

$$V^{-1} = \begin{bmatrix} V_{1L}^{-1} & 0 & 0 \\ 0 & V_{1B}^{-1} & 0 \\ 0 & 0 & V_2^{-1} \end{bmatrix} . \tag{11}$$

5. THE GLOBAL LEAST-SQUARES PROBLEM EXPRESSED IN SEGMENTED FORM

Equation (1) can now be expanded in terms of the segmented vectors and matrices defined above.

$$\begin{bmatrix} G_{1LR}^t & 0 & S_L^t G_{2L}^t \\ 0 & G_{1BR}^t & S_B^t G_{2B}^t \\ 0 & 0 & G_{2P}^t \end{bmatrix} \begin{bmatrix} V_{1L}^{-1} & 0 & 0 \\ 0 & V_{1B}^{-1} & 0 \\ 0 & 0 & V_2^{-1} \end{bmatrix} \begin{bmatrix} G_{1LR} & 0 & 0 \\ 0 & G_{1BR} & 0 \\ G_{2L}S_L & G_{2B}S_B & G_{2P} \end{bmatrix} \begin{bmatrix} \varepsilon_{LR} \\ \varepsilon_{BR} \\ \varepsilon_P \end{bmatrix}$$
$$= \begin{bmatrix} G_{1LR}^t & 0 & S_L^t G_{2L}^t \\ 0 & G_{1BR}^t & S_B^t G_{2B}^t \\ 0 & 0 & G_{2P}^t \end{bmatrix} \begin{bmatrix} V_{1L}^{-1} & 0 & 0 \\ 0 & V_{1B}^{-1} & 0 \\ 0 & 0 & V_2^{-1} \end{bmatrix} \begin{bmatrix} \eta_{1L} \\ \eta_{1B} \\ \eta_2 \end{bmatrix} . \tag{12}$$

Equation (12) may be rewritten using the following matrix definition

$$W_{1L} = G_{1LR}^t V_{1L}^{-1} G_{1LR} ,$$
$$W_{1B} = G_{1BR}^t V_{1B}^{-1} G_{1BR} ,$$
$$W_2 = \begin{bmatrix} G_{2L}^t \\ G_{2B}^t \\ G_{2P}^t \end{bmatrix} V_2^{-1} (G_{2L}, G_{2B}, G_{2P}) , \text{ and} \tag{13}$$
$$S = \begin{bmatrix} S_L & 0 & 0 \\ 0 & S_B & 0 \\ 0 & 0 & 1 \end{bmatrix} .$$

Similarly define the following vectors:

$$R_{1L} = G_{LR}^{t} V_{1L}^{-1} \eta_{1L} \quad ,$$

$$R_{1B} = G_{BR}^{t} V_{1B}^{-1} \eta_{1B} \quad , \quad and$$

$$R_2 = \begin{bmatrix} G_L^t \\ G_B^t \\ G_P^t \end{bmatrix} V_2^{-1} \eta_2 \quad . \tag{14}$$

Based on all the definitions of Eqs. (13) and (14), the global least-squares equation becomes

$$\left[\begin{bmatrix} W_{1L} & 0 & 0 \\ 0 & W_{1B} & 0 \\ 0 & 0 & 0 \end{bmatrix} + S^t W_2 S \right] \begin{bmatrix} \varepsilon_{LR} \\ \varepsilon_{BR} \\ \varepsilon_P \end{bmatrix} = \begin{bmatrix} R_{1L} \\ R_{1B} \\ 0 \end{bmatrix} + S^t R_2 \quad . \tag{15}$$

This may be solved for ε, and the decision then made whether additional iterations are required.

6. OBTAINING THE VECTORS AND MATRICES REQUIRED FOR THE GLOBAL LEAST-SQUARES FIT

It is believed that the vectors and sums in Eq. (15) can be obtained from relatively simple modifications to the programs that separately perform the R-matrix fits at LANL and the simultaneous evaluation fits at ANL. This might be accomplished without actually performing iterative fits to the separate segments of the data base.

The S-matrix contains derivatives with respect to R-matrix parameters of interpolated values from the R-matrix formula in EDA. That program has been trained to output the needed values.

The matrix W_2 is the matrix C_δ^{-1} in Eq. (20) of Ref. 4, when the latter is obtained using the estimated parameter set e^0 and only segment 2 of the input data base. (Note that the matrix inversion to obtain C_δ is not required.) R_2 is the vector $A^t C^{-1} M$ of Ref. 4 when obtained under the same circumstances. Therefore, W_2 and R_2 are readily available.

If the EDA program were based on the usual general least-squares approach, W_{1L} and R_{1L} would be the least-squares matrix and right-hand side of the equation expressing an iteration starting with the parameter set e^0_{LR} and using only segment 1L of the data base. However, this is not the basis of the search in EDA, which is based on the following relation[6] for the change in the parameter vector δp:

$$\delta p = -G^{-1} g \ ,$$

where

$$g_i = \frac{\partial \chi^2}{\partial p_i}$$

and

$$G_{ij} = \frac{\partial^2 \chi^2}{\partial p_i \partial p_j} \ .$$

The gradient vector $g_L = 2R_{1L}$ under the conditions indicated above; it is readily available. However, G^{-1} is directly obtained in EDA as part of the iterative process and in the EDA program it is not defined except at the solution point.

At least three options remain if all the matrices of Eq. (15) are to be available for the global least-squares approach.

(a) One may use for e^0_{LR} and e^0_{BR} the iterated EDA solutions obtained from the segment 1 data set. This choice of e^0 is not apt to be our actual best estimate. This option would permit only one full iteration of the global problem. In this case one is using the G matrices relating R_{1L} and R_{1B} to δp of Eq. (16) (ε of earlier sections here) from EDA rather than from least-squares theory. G. Hale has shown that for this problem the matrices do differ, and has indicated that G^{-1} is a more general approximation to the parameter covariance matrix.

(b) One could modify the EDA program to give W_{1L} (and W_{1B}) just for this application, computed at any selected parameter vector e^0_{LR} (and e^0_{BR}). This approach negates any advantage from the EDA formulation of the equations, but does permit (rather clumsy multilab) iteration.

(c) Again for a single iteration, one could choose for e^0_{LR} and e^0_{BR} the parameters from the R-matrix analyses of unsegmented data sets, as is likely to be favored. One could then compute the W_i just for that portion of the neutron data being "given up" in the data base segmentation, and subtract this from the G computed via EDA. This approach assumes that the contribution to G from this part of the neutron data is the same as would be obtained in least-squares theory. While this approach would be relatively simple, this author believes that numerical problems are apt to ensue.

The author believes that approach (b) is the most promising because it gives the most options for the completion of the global analysis. The initial 1986 results have been obtained using option (a).

7. INCLUSION OF THE THERMAL FISSILE CONSTANTS

At present the CSEWG Standards Committee has access to the results of the thermal fissile standards.[7,8] These overlap the information in the pointwise file. The situation is:

(a) The output ^{239}Pu and ^{235}U "thermal" fission cross sections from Ref. 7 are included as input data to GMA.

(b) The thermal parameter measurements utilized in Ref. 7, for ^{6}Li, ^{10}B, ^{197}Au, are also employed as input to GMA.

(c) The output values from GMA may differ from inputs in the cases of ^{6}Li, ^{10}B, ^{235}U, and ^{239}Pu.

In addition to these problems, the output cross sections and *g*-factors from Ref. 7 may be inconsistent with R-matrix fits to the resonance regions of the fissile nuclides.

Recent R-matrix fits to the fissile nuclide cross sections do not exist now, but may well exist by the time ENDF/B-VI is being assembled. It is recommended that for ENDF/B-VI such R-matrix fits be employed in file construction if the 2.2 km/s value lies within ~ 1/2 standard deviation of the thermal parameters recommended by the Standards Subcommittee. Otherwise, the Standards Subcommittee should review any proposed resolution of the conflict.

Whenever the *g* factors derived from fissile resonance parameterizations differ from the values assumed in the derivation of the fissile thermal parameters, the Standards Subcommittee should consider the impact of such changes on the thermal parameter outputs. This would be facilitated by earlier documentation of the sensitivity to such inputs.

The overlap between the GMA analysis and the traditional thermal fissile constant fit has been handled in the May 1986 results by including all the traditional thermal constants as variables in GMA along with the covariance data obtained by Axton in his analysis of the thermal data.[8]

Appendix C (Continued)

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MEMO of August 31, 1987
Some differences may exist
compared to ENDF/B-VI

APPENDIX D. Complete Results of the Combination Process

August 31, 1987

MEMORANDUM To: C. Dunford

From: A. Carlson, G. Hale, R. Peelle, W. Poenitz

Subject: Results of the ENDF/B-VI Standards Evaluation

The following are the complete output listings from the standards evaluation process, plus an extension to low energies of the ^{238}U fission cross section by the evaluator, W. Poenitz, for completeness. Note that cross sections such as the integral scattering cross sections for the light elements are included though they are not useful as standards.

This is the smoothed output of the combination of the generalized least squares (using segment 2 data) and R-matrix (using segment 1 data) evaluations. The uncertainties are not yet final. In some cases, points have been added to improve the definition of the cross section shape and to ease interpolation thereby. The comments before each data set are important and should be distributed with the data.

For ^{10}B and ^{6}Li at all energies the cross sections are point values. Though in some cases they were obtained in part with data having moderate neutron energy resolution, its effect on the evaluation is probably not significant. Except for the range up to 20 keV, for the heavy elements, the cross sections are low resolution smooth point values. The 9.4 eV value for the ^{235}U fission cross section is the integral cross section over the range from 7.8 to 11 eV. The cross sections listed for the heavy elements with tags from 0.15 keV through 15 keV represent decimal interval average values labeled at the center energies for intervals starting at 0.1 to 0.2 keV and ending with the interval 10 to 20 keV.

It should be noted that data inconsistencies exist which can cause false cross section structure. For example, the energy grid chosen in some energy ranges because of local structure in one cross section causes few experimental values to be available for other cross sections and consequently leads to lack of correlation restraint. Therefore the combination output has been smoothed to remove such structure if evidence indicates that it is very improbable that the fluctuations are real. In some cases, e.g., $^{235}U(n,f)$, structure has been left in since there is a reasonable probability that it is remnant of real structure.

The data have been finalized only in certain energy regions in order to give the assigned ENDF/B file evaluators for these nuclides sufficient freedom in the evaluation process. For the heavy elements the evaluators may superimpose structure on the cross sections below the "finalized" energy limits given here. In these cases the committee recommends that evaluators try to impose structure that maintains the average values given to about 0.5 standard deviations. It is also recommended that the values of the thermal constants used in the ^{233}U, ^{235}U, ^{239}Pu and ^{241}Pu evaluations agree within 0.5 standard deviations of the values listed here. For the light element standards, the evaluator G. Hale may need to revise values above the upper limit given in order to take into account information required for extending the evaluation to 20 MeV.

MEMO of August 31, 1987
Some differences may exist
compared to ENDF/B-VI

^{6}Li(n,t)

Proposed final cross sections below 1 MeV
uncertainties are not final
Log-log interpolation up to 500 keV
Linear-linear interpolation above 500 keV
The values with *'s are points added for ease in interpolation

E_n(MeV)	σ(b)	Uncertainty(%)
0.2530E-07	940.9827	0.14
0.9400E-05	48.7928	0.14
0.1500E-03	12.1957	0.14
0.2500E-03	9.4428	0.14
0.3500E-03	7.9777	0.14
0.4500E-03	7.0337	0.14
0.5500E-03	6.3613	0.14
0.6500E-03	5.8502	0.14
0.7500E-03	5.4454	0.14
0.8500E-03	5.1137	0.14
0.9500E-03	4.8371	0.14
0.1500E-02	3.8470	0.14
0.2500E-02	2.9791	0.14
0.3500E-02	2.5181	0.14
0.4500E-02	2.2214	0.14
0.5500E-02	2.0110	0.14
0.6500E-02	1.8516	0.14
0.7500E-02	1.7243	0.14
0.8500E-02	1.6221	0.14
0.9500E-02	1.5358	0.14
0.1500E-01	1.2301	0.14
0.2000E-01	1.0737	0.15
0.2400E-01	0.9867	0.15
0.3000E-01	0.8930	0.16
0.3500E-01	0.8359*	0.17
0.4500E-01	0.7562	0.18
0.5500E-01	0.7052	0.20
0.6500E-01	0.6725	0.22
0.7500E-01	0.6532	0.24
0.8500E-01	0.6450	0.26
0.9500E-01	0.6468	0.29
0.1000E+00	0.6515	0.30
0.1100E+00	0.6687*	0.32
0.1200E+00	0.6976	0.33
0.1300E+00	0.7406*	0.34
0.1400E+00	0.8013*	0.35
0.1500E+00	0.8843	0.36
0.1600E+00	0.9968*	0.36
0.1700E+00	1.1485	0.35
0.1800E+00	1.3522	0.35
0.1900E+00	1.6207	0.34
0.2000E+00	1.9653	0.32
0.2100E+00	2.3763	0.30
0.2200E+00	2.8047	0.27
0.2300E+00	3.1393	0.25
0.2350E+00	3.2311	0.24
0.2360E+00	3.2416*	0.24
0.2370E+00	3.2495*	0.24
0.2380E+00	3.2546*	0.24
0.2390E+00	3.2571*	0.24
0.2400E+00	3.2568	0.23
0.2410E+00	3.2539*	0.23
0.2420E+00	3.2485*	0.23
0.2430E+00	3.2403*	0.23
0.2440E+00	3.2297*	0.23
0.2450E+00	3.2164	0.23
0.2500E+00	3.1145	0.24
0.2600E+00	2.7896	0.26
0.2700E+00	2.4026	0.28
0.2800E+00	2.0366	0.29
0.3000E+00	1.4743	0.31
0.3100E+00	1.2746*	0.32
0.3250E+00	1.0498	0.32
0.3350E+00	0.9365*	0.32
0.3500E+00	0.8056	0.32
0.3600E+00	0.7376*	0.32
0.3750E+00	0.6565	0.31
0.3850E+00	0.6128*	0.31
0.4000E+00	0.5591	0.31
0.4100E+00	0.5294*	0.31
0.4250E+00	0.4919	0.31
0.4350E+00	0.4706*	0.31
0.4500E+00	0.4431	0.31
0.4600E+00	0.4273*	0.31
0.4750E+00	0.4065	0.31
0.4850E+00	0.3944*	0.31
0.5000E+00	0.3782	0.31
0.5200E+00	0.3597	0.31
0.5400E+00	0.3441	0.32
0.5700E+00	0.3249	0.32
0.6000E+00	0.3093	0.33
0.6500E+00	0.2891	0.34
0.7000E+00	0.2740	0.35
0.7500E+00	0.2623	0.36
0.8000E+00	0.2529	0.37
0.8500E+00	0.2454	0.38
0.9000E+00	0.2391	0.39
0.9400E+00	0.2349	0.39
0.9600E+00	0.2331	0.40
0.9800E+00	0.2313	0.40
0.1000E+01	0.2297	0.41
0.1100E+01	0.2232	0.43
0.1250E+01	0.2171	0.47
0.1400E+01	0.2143	0.51
0.1600E+01	0.2143	0.56
0.1800E+01	0.2184	0.64
0.2000E+01	0.2184	0.67
0.2200E+01	0.2108	0.63
0.2400E+01	0.2006	0.68
0.2600E+01	0.1896	0.75
0.2800E+01	0.1774	0.85

MEMO of August 31, 1987
Some differences may exist
compared to ENDF/B-VI

$^{6}Li(n,n)$

Cross sections are not final
Uncertainties are not final
Linear-linear interpolation

E_n(MeV)	σ(b)	Uncertainty(%)	E_n(MeV)	σ(b)	Uncertainty(%)
0.2530E-07	0.6717	1.27	0.2350E+00	7.7537	0.35
0.9400E-05	0.6713	1.27	0.2400E+00	8.0351	0.32
0.1500E-03	0.6701	1.27	0.2450E+00	8.1469	0.30
0.2500E-03	0.6696	1.27	0.2500E+00	8.0949	0.29
0.3500E-03	0.6692	1.27	0.2600E+00	7.6053	0.30
0.4500E-03	0.6689	1.27	0.2700E+00	6.8402	0.32
0.5500E-03	0.6686	1.27	0.2800E+00	6.0288	0.34
0.6500E-03	0.6683	1.27	0.3000E+00	4.6619	0.38
0.7500E-03	0.6681	1.27	0.3250E+00	3.5376	0.39
0.8500E-03	0.6679	1.27	0.3500E+00	2.8498	0.40
0.9500E-03	0.6676	1.27	0.3750E+00	2.4095	0.41
0.1500E-02	0.6666	1.27	0.4000E+00	2.1125	0.41
0.2500E-02	0.6651	1.27	0.4250E+00	1.9013	0.41
0.3500E-02	0.6639	1.27	0.4500E+00	1.7443	0.42
0.4500E-02	0.6629	1.27	0.4750E+00	1.6258	0.43
0.5500E-02	0.6619	1.27	0.5000E+00	1.5319	0.43
0.6500E-02	0.6611	1.27	0.5200E+00	1.4706	0.43
0.7500E-02	0.6604	1.27	0.5400E+00	1.4191	0.44
0.8500E-02	0.6597	1.27	0.5700E+00	1.3536	0.44
0.9500E-02	0.6590	1.27	0.6000E+00	1.3020	0.44
0.1500E-01	0.6561	1.26	0.6500E+00	1.2332	0.45
0.2000E-01	0.6545	1.26	0.7000E+00	1.1821	0.45
0.2400E-01	0.6537	1.25	0.7500E+00	1.1428	0.45
0.3000E-01	0.6531	1.24	0.8000E+00	1.1123	0.45
0.4500E-01	0.6565	1.21	0.8500E+00	1.0876	0.45
0.5500E-01	0.6628	1.19	0.9000E+00	1.0688	0.45
0.6500E-01	0.6732	1.15	0.9400E+00	1.0561	0.45
0.7500E-01	0.6889	1.11	0.9600E+00	1.0497	0.45
0.8500E-01	0.7114	1.06	0.9800E+00	1.0454	0.45
0.9500E-01	0.7425	1.01	0.1000E+01	1.0410	0.45
0.1000E+00	0.7622	0.98	0.1100E+01	1.0239	0.45
0.1200E+00	0.8800	0.83	0.1250E+01	1.0141	0.46
0.1500E+00	1.2710	0.62	0.1400E+01	1.0194	0.49
0.1700E+00	1.8420	0.54	0.1600E+01	1.0464	0.51
0.1800E+00	2.3028	0.53	0.1800E+01	1.0972	0.54
0.1900E+00	2.9411	0.52	0.2000E+01	1.1729	0.56
0.2000E+00	3.8025	0.50	0.2200E+01	1.2564	0.57
0.2100E+00	4.9038	0.48	0.2400E+01	1.3437	0.57
0.2200E+00	6.1593	0.43	0.2600E+01	1.4398	0.57
0.2300E+00	7.3212	0.38	0.2800E+01	1.5072	0.59

MEMO of August 31, 1987
Some differences may exist
compared to ENDF/B-VI

$^{10}B(n,\alpha_0)$

Proposed final cross sections below 250 keV
Uncertainties are not final
Log-log interpolation up to 500 keV
Linear-linear interpolation above 500 keV

E_n(MeV)	σ(b)	Uncertainty(%)	E_n(MeV)	σ(b)	Uncertainty(%)
0.2530E-07	241.2677	0.21	0.1900E+00	0.1434	0.76
0.9400E-05	12.5002	0.21	0.2000E+00	0.1438	0.76
0.1500E-03	3.1169	0.21	0.2100E+00	0.1439	0.77
0.2500E-03	2.4105	0.21	0.2200E+00	0.1439	0.78
0.3500E-03	2.0347	0.21	0.2300E+00	0.1439	0.79
0.4500E-03	1.7928	0.21	0.2350E+00	0.1438	0.80
0.5500E-03	1.6199	0.21	0.2400E+00	0.1438	0.81
0.6500E-03	1.4889	0.21	0.2450E+00	0.1437	0.82
0.7500E-03	1.3850	0.21	0.2500E+00	0.1436	0.82
0.8500E-03	1.3000	0.21	0.2600E+00	0.1435	0.83
0.9500E-03	1.2290	0.21	0.2700E+00	0.1432	0.84
0.1500E-02	0.9748	0.21	0.2800E+00	0.1431	0.84
0.2500E-02	0.7519	0.21	0.3000E+00	0.1431	0.86
0.3500E-02	0.6333	0.22	0.3250E+00	0.1442	0.91
0.4500E-02	0.5571	0.22	0.3500E+00	0.1471	0.99
0.5500E-02	0.5028	0.23	0.3750E+00	0.1533	1.08
0.6500E-02	0.4617	0.25	0.4000E+00	0.1638	1.10
0.7500E-02	0.4291	0.26	0.4250E+00	0.1787	1.07
0.8500E-02	0.4025	0.28	0.4500E+00	0.1951	1.05
0.9500E-02	0.3804	0.30	0.4750E+00	0.2062	1.08
0.1500E-01	0.3017	0.40	0.5000E+00	0.2072	1.05
0.2000E-01	0.2613	0.50	0.5200E+00	0.2011	0.98
0.2400E-01	0.2390	0.57	0.5400E+00	0.1913	0.91
0.3000E-01	0.2148	0.66	0.5700E+00	0.1747	0.87
0.4500E-01	0.1796	0.81	0.6000E+00	0.1592	0.87
0.5500E-01	0.1662	0.86	0.6500E+00	0.1396	0.86
0.6500E-01	0.1572	0.87	0.7000E+00	0.1261	0.87
0.7500E-01	0.1510	0.87	0.7500E+00	0.1170	1.01
0.8500E-01	0.1467	0.87	0.8000E+00	0.1106	1.30
0.9500E-01	0.1440	0.87	0.8500E+00	0.1062	1.70
0.1000E+00	0.1431	0.86	0.9000E+00	0.1031	2.16
0.1200E+00	0.1412	0.87	0.9400E+00	0.1014	2.56
0.1500E+00	0.1417	0.84	0.9600E+00	0.1008	2.77
0.1700E+00	0.1426	0.80	0.9800E+00	0.1003	2.99
0.1800E+00	0.1431	0.78	0.1000E+01	0.0999	3.21

MEMO of August 31, 1987
Some differences may exist
compared to ENDF/B-VI

Appendix D (Continued)

$^{10}B(n,\alpha_1)$

Proposed final cross sections below 250 keV
Uncertainties are not final
Log-log interpolation up to 500 keV
Linear-linear interpolation above 500 keV

E_n(MeV)	σ(b)	Uncertainty(%)	E_n(MeV)	σ(b)	Uncertainty(%)
0.2530E-07	3598.2280	0.16	0.1900E+00	1.3340	0.59
0.9400E-05	186.4350	0.16	0.2000E+00	1.2876	0.60
0.1500E-03	46.5003	0.16	0.2100E+00	1.2417	0.61
0.2500E-03	35.9671	0.16	0.2200E+00	1.1972	0.62
0.3500E-03	30.3659	0.16	0.2300E+00	1.1543	0.62
0.4500E-03	26.7516	0.16	0.2350E+00	1.1338	0.62
0.5500E-03	24.1756	0.16	0.2400E+00	1.1127	0.62
0.6500E-03	22.2287	0.16	0.2450E+00	1.0928	0.63
0.7500E-03	20.6811	0.16	0.2500E+00	1.0732	0.63
0.8500E-03	19.4130	0.17	0.2600E+00	1.0346	0.64
0.9500E-03	18.3446	0.17	0.2700E+00	0.9976	0.65
0.1500E-02	14.5603	0.17	0.2800E+00	0.9623	0.67
0.2500E-02	11.2352	0.17	0.3000E+00	0.8967	0.73
0.3500E-02	9.4717	0.17	0.3250E+00	0.8243	0.82
0.4500E-02	8.3353	0.18	0.3500E+00	0.7636	0.87
0.5500E-02	7.5254	0.18	0.3750E+00	0.7163	0.89
0.6500E-02	6.9123	0.18	0.4000E+00	0.6833	0.91
0.7500E-02	6.4260	0.18	0.4250E+00	0.6634	0.96
0.8500E-02	6.0296	0.18	0.4500E+00	0.6494	1.02
0.9500E-02	5.6981	0.18	0.4750E+00	0.6282	1.04
0.1500E-01	4.5194	0.20	0.5000E+00	0.5910	1.04
0.2000E-01	3.9090	0.21	0.5200E+00	0.5515	1.03
0.2400E-01	3.5682	0.23	0.5400E+00	0.5082	1.02
0.3000E-01	3.1959	0.25	0.5700E+00	0.4452	1.01
0.4500E-01	2.6285	0.30	0.6000E+00	0.3901	1.00
0.5500E-01	2.3943	0.33	0.6500E+00	0.3182	1.00
0.6500E-01	2.2203	0.35	0.7000E+00	0.2659	1.04
0.7500E-01	2.0844	0.38	0.7500E+00	0.2267	1.14
0.8500E-01	1.9745	0.40	0.8000E+00	0.1965	1.28
0.9500E-01	1.8835	0.43	0.8500E+00	0.1727	1.44
0.1000E+00	1.8420	0.44	0.9000E+00	0.1535	1.62
0.1200E+00	1.7031	0.49	0.9400E+00	0.1405	1.76
0.1500E+00	1.5330	0.55	0.9600E+00	0.1348	1.84
0.1700E+00	1.4303	0.57	0.9800E+00	0.1294	1.91
0.1800E+00	1.3817	0.58	0.1000E+01	0.1244	1.99

MEMO of August 31, 1987
Some differences may exist
compared to ENDF/B-VI

Appendix D (Continued)

$^{10}B(n,n)$

Cross sections are not final
Uncertainties are not final
Linear-linear interpolation

E_n(MeV)	σ(b)	Uncertainty(%)	E_n(MeV)	σ(b)	Uncertainty(%)
0.2530E-07	2.1413	1.42	0.1900E+00	3.3303	0.38
0.9400E-05	2.1386	1.42	0.2000E+00	3.3935	0.37
0.1500E-03	2.1312	1.42	0.2100E+00	3.4529	0.36
0.2500E-03	2.1283	1.42	0.2200E+00	3.5076	0.36
0.3500E-03	2.1265	1.42	0.2300E+00	3.5578	0.36
0.4500E-03	2.1245	1.41	0.2350E+00	3.5813	0.35
0.5500E-03	2.1236	1.41	0.2400E+00	3.6027	0.35
0.6500E-03	2.1216	1.41	0.2450E+00	3.6241	0.35
0.7500E-03	2.1206	1.41	0.2500E+00	3.6444	0.35
0.8500E-03	2.1197	1.40	0.2600E+00	3.6812	0.35
0.9500E-03	2.1187	1.40	0.2700E+00	3.7130	0.35
0.1500E-02	2.1156	1.39	0.2800E+00	3.7401	0.35
0.2500E-02	2.1122	1.36	0.3000E+00	3.7821	0.35
0.3500E-02	2.1107	1.34	0.3250E+00	3.8119	0.35
0.4500E-02	2.1101	1.31	0.3500E+00	3.8191	0.35
0.5500E-02	2.1114	1.29	0.3750E+00	3.8083	0.36
0.6500E-02	2.1117	1.26	0.4000E+00	3.7833	0.37
0.7500E-02	2.1141	1.24	0.4250E+00	3.7475	0.39
0.8500E-02	2.1154	1.22	0.4500E+00	3.7065	0.41
0.9500E-02	2.1187	1.19	0.4750E+00	3.6590	0.42
0.1500E-01	2.1349	1.08	0.5000E+00	3.6044	0.42
0.2000E-01	2.1548	0.99	0.5200E+00	3.5538	0.42
0.2400E-01	2.1725	0.93	0.5400E+00	3.4994	0.42
0.3000E-01	2.2025	0.85	0.5700E+00	3.4121	0.43
0.4500E-01	2.2854	0.71	0.6000E+00	3.3228	0.44
0.5500E-01	2.3478	0.66	0.6500E+00	3.1765	0.46
0.6500E-01	2.4135	0.62	0.7000E+00	3.0380	0.49
0.7500E-01	2.4826	0.59	0.7500E+00	2.9093	0.52
0.8500E-01	2.5539	0.57	0.8000E+00	2.7902	0.55
0.9500E-01	2.6272	0.55	0.8500E+00	2.6819	0.58
0.1000E+00	2.6652	0.54	0.9000E+00	2.5813	0.62
0.1200E+00	2.8178	0.50	0.9400E+00	2.5069	0.64
0.1500E+00	3.0476	0.43	0.9600E+00	2.4722	0.66
0.1700E+00	3.1951	0.40	0.9800E+00	2.4374	0.67
0.1800E+00	3.2641	0.39	0.1000E+01	2.4046	0.68

MEMO of August 31, 1987
Some differences may exist
compared to ENDF/B-VI

Au(n,γ)

Proposed final cross sections from 200 keV to 2.5 MeV
Structure may be imposed below 200 keV

Average cross sections are given
from the interval (2.0-3.0 keV), labelled 0.2500E-02 MeV;
to the interval (10-20 keV), labelled 0.1500E-01 MeV

Uncertainties are not final
Linear-linear interpolation

E_n(MeV)	σ(b)	Uncertainty(%)	E_n(MeV)	σ(b)	Uncertainty(%)
0.2530E-07	98.6862*	0.14	0.2800E+00	0.2141	1.31
0.2500E-02	2.4597	3.59	0.3000E+00	0.1999	1.30
0.3500E-02	2.6251	1.37	0.3250E+00	0.1877	1.23
0.4500E-02	2.2280	1.30	0.3500E+00	0.1778	1.25
0.5500E-02	1.9593	1.33	0.3750E+00	0.1689	1.25
0.6500E-02	1.7731	1.40	0.4000E+00	0.1614	1.16
0.7500E-02	1.5641	1.30	0.4250E+00	0.1538	1.21
0.8500E-02	1.3459	1.34	0.4500E+00	0.1462	1.17
0.9500E-02	1.2001	1.27	0.4750E+00	0.1389	1.27
0.1500E-01	0.8918	1.09	0.5000E+00	0.1324	1.15
0.2000E-01	0.6963	1.74	0.5200E+00	0.1270	1.22
0.2400E-01	0.6396	0.92	0.5400E+00	0.1236	1.33
0.3000E-01	0.5863	0.75	0.5700E+00	0.1186	1.64
0.4500E-01	0.4570	0.98	0.6000E+00	0.1084	1.37
0.5500E-01	0.4131	1.08	0.6500E+00	0.1002	1.73
0.6500E-01	0.3796	1.04	0.7000E+00	0.0964	1.42
0.7500E-01	0.3535	1.03	0.7500E+00	0.0928	1.66
0.8500E-01	0.3300	1.03	0.8000E+00	0.0897	1.57
0.9500E-01	0.3150	1.11	0.8500E+00	0.0869	2.10
0.1000E+00	0.3099	1.14	0.9000E+00	0.0843	3.10
0.1200E+00	0.2921	0.98	0.9400E+00	0.0825	3.10
0.1500E+00	0.2719	0.97	0.9600E+00	0.0818	3.70
0.1700E+00	0.2617	1.15	0.9800E+00	0.0810	4.00
0.1800E+00	0.2567	1.40	0.1000E+01	0.0803	2.50
0.1900E+00	0.2534	1.41	0.1100E+01	0.0772	1.80
0.2000E+00	0.2502	1.24	0.1250E+01	0.0729	1.48
0.2100E+00	0.2469	1.43	0.1400E+01	0.0694	1.75
0.2200E+00	0.2445	1.35	0.1600E+01	0.0665	1.70
0.2300E+00	0.2415	1.35	0.1800E+01	0.0596	1.78
0.2350E+00	0.2403	1.33	0.2000E+01	0.0534	2.05
0.2400E+00	0.2388	1.67	0.2200E+01	0.0433	2.01
0.2450E+00	0.2374	1.23	0.2400E+01	0.0360	3.02
0.2500E+00	0.2360	1.36	0.2600E+01	0.0311	2.40
0.2600E+00	0.2331	1.38	0.2800E+01	0.0255	2.23
0.2700E+00	0.2299	1.49			

*This value differs slightly from that of ENDF/B-VI.

MEMO of August 31, 1987
Some differences may exist
compared to ENDF/B-VI

Appendix D (Continued)

$^{235}U(n,f)$

Proposed final cross sections above 150 keV
Structure may be imposed below 150 keV
The 9.4 eV value is the integral cross section from 7.8 to 11 eV

Average cross sections are given
from the interval (0.1-0.2 keV), labelled 0.1500E-03 MeV;
to the interval (10-20 keV), labelled 0.1500E-01 MeV

Uncertainties are not final
Linear-linear interpolation

E_n(MeV)	σ(b)	Uncertainty(%)	E_n(MeV)	σ(b)	Uncertainty(%)
0.2530E-07	584.2522*	0.19	0.1700E+00	1.3967	0.72
0.9400E-05	246.4970#	0.42	0.1800E+00	1.3800	0.70
0.1500E-03	21.1374	0.44	0.1900E+00	1.3647	0.82
0.2500E-03	20.6721	0.46	0.2000E+00	1.3510	0.79
0.3500E-03	13.1380	0.51	0.2100E+00	1.3370	0.71
0.4500E-03	13.7862	0.52	0.2200E+00	1.3265	0.71
0.5500E-03	15.1890	0.53	0.2300E+00	1.3130	0.63
0.6500E-03	11.4739	0.54	0.2350E+00	1.3100	0.67
0.7500E-03	11.1436	0.54	0.2400E+00	1.3070	0.68
0.8500E-03	8.2481	0.54	0.2450E+00	1.3030	0.64
0.9500E-03	7.5274	0.55	0.2500E+00	1.2930	0.58
0.1500E-02	7.3465	0.49	0.2600E+00	1.2690	0.63
0.2500E-02	5.3900	0.55	0.2700E+00	1.2500	0.62
0.3500E-02	4.7822	0.53	0.2800E+00	1.2350	0.65
0.4500E-02	4.2729	0.53	0.3000E+00	1.2300	0.60
0.5500E-02	3.8012	0.61	0.3250E+00	1.2300	0.70
0.6500E-02	3.2824	0.57	0.3500E+00	1.2230	0.64
0.7500E-02	3.2558	0.52	0.3750E+00	1.2130	0.70
0.8500E-02	3.0166	0.56	0.4000E+00	1.2020	0.66
0.9500E-02	3.1130	0.56	0.4250E+00	1.1900	0.74
0.1500E-01	2.4975	0.49	0.4500E+00	1.1662	0.70
0.2000E-01	2.3932	1.81	0.4750E+00	1.1510	0.73
0.2400E-01	2.1627	0.49	0.5000E+00	1.1410	0.59
0.3000E-01	2.0908	0.60	0.5200E+00	1.1365	0.72
0.4500E-01	1.8486	0.54	0.5400E+00	1.1300	0.62
0.5500E-01	1.8029	0.54	0.5700E+00	1.1220	0.67
0.6500E-01	1.7437	0.55	0.6000E+00	1.1185	0.62
0.7500E-01	1.6766	0.53	0.6500E+00	1.1182	0.58
0.8500E-01	1.5909	0.56	0.7000E+00	1.1135	0.59
0.9500E-01	1.5665	0.61	0.7500E+00	1.1120	0.58
0.1000E+00	1.5724	0.63	0.8000E+00	1.1100	0.56
0.1200E+00	1.4961	0.58	0.8500E+00	1.1135	0.59
0.1500E+00	1.4203	0.58	0.9000E+00	1.1372	0.57

*This value differs slightly from that of ENDF/B-VI.
#units of barn-eV

$^{235}U(n,f)$ (Continued)

E_n(MeV)	σ(b)	Uncertainty(%)	E_n(MeV)	σ(b)	Uncertainty(%)
0.9400E+00	1.1691	0.60	0.6000E+01	1.0985	0.85
0.9600E+00	1.1876	0.64	0.6200E+01	1.1817	0.87
0.9800E+00	1.1992	0.72	0.6500E+01	1.3481	0.87
0.1000E+01	1.1969	0.52	0.7000E+01	1.5467	0.89
0.1100E+01	1.1938	0.54	0.7500E+01	1.6964	0.94
0.1250E+01	1.2020	0.51	0.7750E+01	1.7300	1.05
0.1400E+01	1.2200	0.54	0.8000E+01	1.7606	1.01
0.1600E+01	1.2435	0.52	0.8500E+01	1.7800	0.89
0.1800E+01	1.2619	0.54	0.9000E+01	1.7700	0.99
0.2000E+01	1.2714	0.51	0.1000E+02	1.7415	1.06
0.2200E+01	1.2699	0.53	0.1100E+02	1.7219	1.11
0.2400E+01	1.2561	0.55	0.1150E+02	1.7170	1.26
0.2600E+01	1.2442	0.59	0.1200E+02	1.7347	1.14
0.2800E+01	1.2220	0.62	0.1300E+02	1.9002	0.90
0.3000E+01	1.2010	0.58	0.1400E+02	2.0600	0.59
0.3600E+01	1.1473	0.62	0.1450E+02	2.0800	0.51
0.4000E+01	1.1295	0.63	0.1500E+02	2.0890	0.84
0.4500E+01	1.1011	0.64	0.1600E+02	2.0890	1.10
0.4700E+01	1.0923	0.70	0.1700E+02	2.0413	1.27
0.5000E+01	1.0617	0.70	0.1800E+02	1.9748	1.26
0.5300E+01	1.0502	0.72	0.1900E+02	1.9325	1.25
0.5500E+01	1.0388	0.72	0.2000E+02	1.9343	1.52
0.5800E+01	1.0408	0.78			

Appendix D (Continued)

$^{238}U(n,f)$

Average cross sections proposed as final in the full energy range
Structure may be imposed below 4 MeV
Uncertainties are not final
Linear-linear interpolation

The values with †'s were supplied by the ENDF evaluator (W. Poenitz) but are not part of the present evaluation process

The values with *'s are points added for ease in interpolation

E_n(MeV)	σ(b)	Uncertainty(%)	E_n(MeV)	σ(b)	Uncertainty(%)
0.3000E+00	0.0001133†	10.00	0.1100E+01	0.03004	1.41
0.3500E+00	0.0001852†	9.39	0.1130E+01	0.03600†*	1.38
0.3800E+00	0.0002329†	9.03	0.1140E+01	0.03810†*	1.37
0.4000E+00	0.0002513†	8.78	0.1150E+01	0.03920†*	1.36
0.4200E+00	0.0002758†	8.54	0.1170E+01	0.04025†*	1.34
0.4300E+00	0.0002758†	8.42	0.1200E+01	0.04209†*	1.31
0.4400E+00	0.0002830†	8.30	0.1230E+01	0.04030†*	1.28
0.4500E+00	0.0002849†	8.18	0.1240E+01	0.04000†*	1.28
0.4600E+00	0.0002908†	8.06	0.1250E+01	0.03912	1.27
0.4700E+00	0.0003037†	7.93	0.1280E+01	0.05020†*	1.24
0.5000E+00	0.0003764†	7.57	0.1300E+01	0.06500†*	1.22
0.5500E+00	0.0006295†	6.96	0.1350E+01	0.1119†*	1.17
0.5800E+00	0.0006908†	6.60	0.1400E+01	0.1855	1.13
0.5900E+00	0.0007588†	6.48	0.1450E+01	0.2822†*	1.08
0.6000E+00	0.0008226†	6.36	0.1480E+01	0.3310†*	1.05
0.6200E+00	0.0009277†	6.11	0.1500E+01	0.3560†*	1.03
0.6400E+00	0.001128†	5.87	0.1525E+01	0.3805†*	1.01
0.6500E+00	0.001239†	5.75	0.1550E+01	0.3990†*	0.98
0.6600E+00	0.001294†	5.63	0.1575E+01	0.4125†*	0.96
0.6800E+00	0.001574†	5.39	0.1600E+01	0.4226	0.94
0.7000E+00	0.001717†	5.14	0.1700E+01	0.4550†*	0.84
0.7500E+00	0.002574†	4.54	0.1800E+01	0.4820	0.75
0.7800E+00	0.003578†	4.17	0.1900E+01	0.5070†*	0.75
0.8000E+00	0.004470†	3.93	0.2000E+01	0.5250	0.72
0.8500E+00	0.007168†	3.32	0.2100E+01	0.5355†*	0.73
0.8800E+00	0.01077†	2.96	0.2200E+01	0.5391	0.73
0.9000E+00	0.01362†	2.71	0.2400E+01	0.5373	0.75
0.9300E+00	0.01549†	2.35	0.2600E+01	0.5328	0.73
0.9500E+00	0.01654†	2.11	0.2800E+01	0.5270	0.79
0.9700E+00	0.01582†	1.86	0.3000E+01	0.5160	0.75
0.1000E+01	0.01398	1.50	0.3200E+01	0.5210†*	0.75
0.1020E+01	0.01570†*	1.48	0.3600E+01	0.5354	0.78
0.1030E+01	0.01693†*	1.47	0.4000E+01	0.5483	0.81
0.1050E+01	0.02000†*	1.45	0.4500E+01	0.5496	0.82
0.1080E+01	0.02700†*	1.43	0.4700E+01	0.5470	0.89

Appendix D (Continued)

$^{238}U(n,f)$ (Continued)

E_n(MeV)	σ(b)	Uncertainty(%)	E_n(MeV)	σ(b)	Uncertainty(%)
0.5000E+01	0.5404	0.87	0.9000E+01	0.9979	1.15
0.5300E+01	0.5430	0.92	0.1000E+02	0.9868	1.21
0.5500E+01	0.5500	0.90	0.1100E+02	0.9830	1.36
0.5800E+01	0.5731	0.95	0.1150E+02	0.9830	1.50
0.6000E+01	0.6153	1.03	0.1200E+02	0.9850	1.33
0.6200E+01	0.6859	1.04	0.1300E+02	1.0130	1.04
0.6500E+01	0.8257	1.01	0.1400E+02	1.1300	0.76
0.6700E+01	0.8860*	1.02	0.1450E+02	1.1555	0.70
0.7000E+01	0.9403	1.04	0.1500E+02	1.1980	0.94
0.7300E+01	0.9680*	1.08	0.1600E+02	1.2593	1.26
0.7500E+01	0.9807	1.11	0.1700E+02	1.2561	1.40
0.7750E+01	0.9910	1.23	0.1800E+02	1.2493	1.44
0.8000E+01	0.9935	1.16	0.1900E+02	1.2954	1.49
0.8500E+01	1.0000	1.07	0.2000E+02	1.3521	1.74

MEMO of August 31, 1987
Some differences may exist
compared to ENDF/B-VI

Appendix D (Continued)

$^{238}U(n,\gamma)$

Proposed final from 150 keV to 2 MeV
Structure may be imposed below 150 keV

Average cross sections are given
from the interval (0.1-0.2 keV), labelled 0.1500E-03 MeV;
to the interval (10-20 keV), labelled 0.1500E-01 MeV

Uncertainties are not final
Interpolation rules to be determined

E_n(MeV)	σ(b)	Uncertainty(%)
0.2530E-07	2.7081	0.35
0.1500E-03	16.1260	2.24
0.2500E-03	8.1332	2.12
0.3500E-03	2.7346	2.11
0.4500E-03	2.5204	2.12
0.5500E-03	4.2754	3.48
0.6500E-03	3.3746	1.94
0.7500E-03	1.6737	1.88
0.8500E-03	2.7942	1.94
0.9500E-03	3.7471	1.95
0.1500E-02	1.7843	1.92
0.2500E-02	1.3305	2.07
0.3500E-02	1.0917	2.24
0.4500E-02	0.8671	2.36
0.5500E-02	0.8445	2.17
0.6500E-02	0.7639	2.22
0.7500E-02	0.7246	2.13
0.8500E-02	0.6240	2.41
0.9500E-02	0.6430	2.03
0.1500E-01	0.5656	1.54
0.2000E-01	0.5035	1.77
0.2400E-01	0.4686	1.16
0.3000E-01	0.4264	1.03
0.4500E-01	0.3494	1.15
0.5500E-01	0.2802	1.22
0.6500E-01	0.2411	1.22
0.7500E-01	0.2116	1.27
0.8500E-01	0.1924	1.27
0.9500E-01	0.1796	1.25
0.1000E+00	0.1746	1.27
0.1200E+00	0.1585	1.16
0.1500E+00	0.1415	1.26
0.1700E+00	0.1336	1.63
0.1800E+00	0.1313	2.42
0.1900E+00	0.1285	1.81
0.2000E+00	0.1274	1.67
0.2100E+00	0.1244	1.71
0.2200E+00	0.1226	2.58

E_n(MeV)	σ(b)	Uncertainty(%)
0.2300E+00	0.1210	1.59
0.2350E+00	0.1201	3.03
0.2400E+00	0.1194	2.49
0.2450E+00	0.1192	1.85
0.2500E+00	0.1184	2.02
0.2600E+00	0.1173	1.58
0.2700E+00	0.1162	2.90
0.2800E+00	0.1158	1.92
0.3000E+00	0.1150	1.59
0.3250E+00	0.1140	2.44
0.3500E+00	0.1138	1.82
0.3750E+00	0.1128	1.77
0.4000E+00	0.1119	1.36
0.4250E+00	0.1112	2.55
0.4500E+00	0.1106	1.66
0.4750E+00	0.1098	5.66
0.5000E+00	0.1093	1.37
0.5200E+00	0.1095	2.08
0.5400E+00	0.1101	1.83
0.5700E+00	0.1102	1.90
0.6000E+00	0.1113	2.40
0.6500E+00	0.1160	5.70
0.7000E+00	0.1189	6.30
0.7500E+00	0.1192	4.10
0.8000E+00	0.1205	1.90
0.8500E+00	0.1217	5.16
0.9000E+00	0.1228	2.51
0.9400E+00	0.1248	2.35
0.9600E+00	0.1260	2.76
0.1000E+01	0.1251	2.17
0.1100E+01	0.1143	2.14
0.1250E+01	0.0927	1.98
0.1400E+01	0.0782	2.06
0.1600E+01	0.0640	2.01
0.1800E+01	0.0560	2.58
0.2000E+01	0.0461	3.17
0.2200E+01	0.0376	2.81

Appendix D (Continued)

$^{239}Pu(n,f)$

Proposed final cross sections above 150 keV
Structure may be imposed below 150 keV

Average cross sections are given
from the interval (0.1-0.2 keV), labelled 0.1500E-03 MeV;
to the interval (10-20 keV), labelled 0.1500E-01 MeV

Uncertainties not final
Interpolation rules to be determined

E_n(MeV)	σ(b)	Uncertainty(%)	E_n(MeV)	σ(b)	Uncertainty(%)
0.2530E-07	747.9861*	0.25	0.2000E+00	1.4766	1.01
0.1500E-03	18.6664	0.69	0.2100E+00	1.4820	1.14
0.2500E-03	17.8794	0.67	0.2200E+00	1.4880	1.02
0.3500E-03	8.4301	0.70	0.2300E+00	1.4920	0.84
0.4500E-03	9.5697	0.69	0.2400E+00	1.4950	1.33
0.5500E-03	15.5569	0.70	0.2450E+00	1.4980	0.92
0.6500E-03	4.4593	0.83	0.2500E+00	1.5040	0.99
0.7500E-03	5.6301	0.72	0.2600E+00	1.5090	1.02
0.8500E-03	4.9794	0.77	0.2700E+00	1.5150	0.84
0.9500E-03	8.2969	0.81	0.2800E+00	1.5200	0.85
0.1500E-02	4.4659	0.71	0.3000E+00	1.5255	0.85
0.2500E-02	3.3041	0.76	0.3250E+00	1.5348	0.84
0.3500E-02	3.0000	0.77	0.3500E+00	1.5380	0.79
0.4500E-02	2.3830	0.75	0.3750E+00	1.5400	0.90
0.5500E-02	2.3007	0.78	0.4000E+00	1.5405	0.83
0.6500E-02	2.0083	0.84	0.4250E+00	1.5450	1.39
0.7500E-02	2.0536	0.68	0.4500E+00	1.5500	0.84
0.8500E-02	2.2164	0.79	0.4750E+00	1.5520	0.87
0.9500E-02	1.8641	0.78	0.5000E+00	1.5558	0.76
0.1500E-01	1.7637	0.63	0.5200E+00	1.5630	1.09
0.2000E-01	1.7683	1.86	0.5400E+00	1.5670	0.77
0.2400E-01	1.5947	0.72	0.5700E+00	1.5800	0.95
0.3000E-01	1.6587	0.74	0.6000E+00	1.5880	0.94
0.4500E-01	1.4900	0.67	0.6500E+00	1.6100	0.71
0.5500E-01	1.5350	0.74	0.7000E+00	1.6243	0.76
0.6500E-01	1.5300	0.71	0.7500E+00	1.6420	0.69
0.7500E-01	1.5280	0.78	0.8000E+00	1.6580	0.70
0.8500E-01	1.5220	0.73	0.8500E+00	1.6752	0.73
0.9500E-01	1.5150	1.00	0.9000E+00	1.6850	0.72
0.1000E+00	1.5150	0.83	0.9400E+00	1.6900	0.81
0.1200E+00	1.5058	0.71	0.9600E+00	1.6940	0.78
0.1500E+00	1.4820	0.67	0.9800E+00	1.7020	0.98
0.1700E+00	1.4800	0.85	0.1000E+01	1.7128	0.64
0.1800E+00	1.4747	0.93	0.1100E+01	1.7420	0.70
0.1900E+00	1.4780	1.06	0.1250E+01	1.8241	0.62

*This value differs slightly from that of ENDF/B-VI.

Appendix D (Continued)

$^{239}Pu(n,f)$ (Continued)

E_n(MeV)	σ(b)	Uncertainty(%)	E_n(MeV)	σ(b)	Uncertainty(%)
0.1400E+01	1.8894	0.64	0.7000E+01	2.0813	1.04
0.1600E+01	1.9137	0.64	0.7500E+01	2.1691	1.08
0.1800E+01	1.9262	0.67	0.7750E+01	2.1974	1.16
0.2000E+01	1.9400	0.67	0.8000E+01	2.2110	1.36
0.2200E+01	1.9350	0.65	0.8500E+01	2.2030	1.07
0.2400E+01	1.9030	0.69	0.9000E+01	2.1917	1.13
0.2600E+01	1.8762	0.72	0.1000E+02	2.1778	1.21
0.2800E+01	1.8567	0.74	0.1100E+02	2.1742	1.28
0.3000E+01	1.8204	0.68	0.1150E+02	2.1850	1.48
0.3600E+01	1.7802	0.74	0.1200E+02	2.1980	1.34
0.4000E+01	1.7517	0.74	0.1300E+02	2.2674	1.16
0.4500E+01	1.7210	0.79	0.1400E+02	2.3877	0.86
0.4700E+01	1.7100	0.82	0.1450E+02	2.3753	0.64
0.5000E+01	1.6920	0.81	0.1500E+02	2.3600	1.21
0.5300E+01	1.6722	0.86	0.1600E+02	2.3038	1.35
0.5500E+01	1.6699	0.86	0.1700E+02	2.2443	1.56
0.5800E+01	1.6860	0.91	0.1800E+02	2.1855	1.51
0.6000E+01	1.7776	0.97	0.1900E+02	2.1808	1.67
0.6200E+01	1.8409	1.03	0.2000E+02	2.1850	1.88
0.6500E+01	1.9497	0.98			

Appendix D (Continued)

Thermal (2200 m/s) Constants with Uncertainties in Percent

Quantity	^{233}U	^{235}U	^{239}Pu	^{241}Pu
σ_{nf}	531.1396 ± 0.25%	584.2522 ± 0.19%	747.9861 ± 0.25%	1012.6840 ± 0.65%
$\sigma_{n\gamma}$	45.5095 ± 1.50%	98.9649 ± 0.75%	271.4265 ± 0.79%	361.290 ± 1.37%
σ_{nn}	12.1309 ± 5.48%	15.4557 ± 6.87%	7.8835 ±12.30%	12.1738 ±21.50%
g_f	0.9955 ± 0.14%	0.9771 ± 0.08%	1.0563 ± 0.21%	1.0450 ± 0.53%
g_a	0.9996 ± 0.11%	0.9790 ± 0.08%	1.0782 ± 0.22%	1.0440 ± 0.19%
$\bar{\nu}$	2.4946 ± 0.16%	2.4320 ± 0.15%	2.8815 ± 0.18%	2.9453 ± 0.20%

^{252}Cf $\bar{\nu}$ 3.7676 ± 0.13%

The evaluation by Axton (CBNM Report GE/PH/01/86) was used as input