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An Evaluation for ENDF/B-IV of the Neutron Cross Sections for ^{235}U from 82 eV to 25 keV

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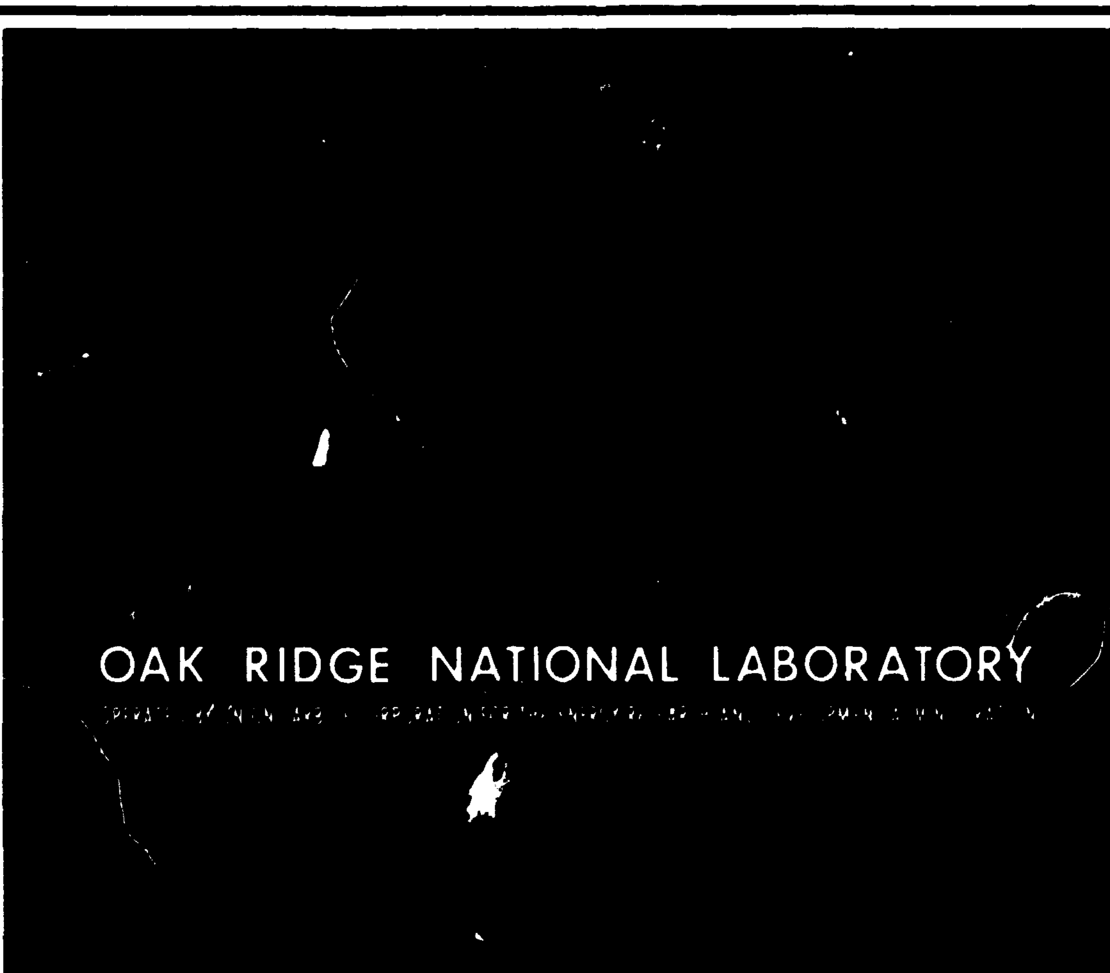

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AN EVALUATION FOR ENDF/B-IV OF THE NEUTRON CROSS SECTIONS FOR ^{235}U FROM 82 eV TO 25 keV

R. W. Peelle

MAY 1976

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CONTENTS

ABSTRACT

Capture and fission cross sections for ^{235}U in the "unresolved resonance" energy region were evaluated to permit determination of local-average resonance parameters for the ENDF/B-IV cross section file. Microscopic data were examined for infinitely dilute average fission and capture cross sections and also for intermediate structure unlikely to be reproduced by statistical fluctuations of resonance widths and spacings within known laws. Evaluated cross sections, averaged over lethargy intervals greater than 0.1, were obtained as an average over selected data sets after appropriate renormalization. Estimated uncertainties are given for these evaluated average cross sections. The "intermediate" structure fluctuations common to a few independent data sets were approximated by straight lines joining successive cross sections at 120 selected energy points; the cross sections at the vertices were adjusted to reproduce the evaluated average cross sections over the broad energy regions. Data sources and methods are reviewed, output values are tabulated, and some modified procedures are suggested for future evaluations.

Evaluated fission and capture integrals for the resolved resonance region are also tabulated. These are not in agreement with integrals based on the resonance parameters of ENDF/B versions III and IV.

I. INTRODUCTION

This report documents an evaluation for ENDF/B-IV[1] of the cross sections for interaction between ^{235}U and neutrons with energies between 82 eV and 25 keV, a region now assigned in the ENDF/B system as the region of unresolved resonances. Since the normalization for many of the measurements used is at thermal energies or in the low resonance region, it was necessary to compare fission and capture integrals through both the resolved and unresolved resonance regions.

In the ENDF/B file the cross sections in the unresolved resonance region are described in terms of an effective s-wave potential scattering radius and tabulations as a function of energy of local-average resonance parameters and spacings for each of the possible spin states of the compound nucleus.[2] Spins J = 3, 4 for s-wave neutrons and 2, 3, 4, and 5 for p-wave neutrons have been included. The p-wave resonance average behavior is poorly known and was adopted from the prior work of Pitterle et al.[3] Level spacings and constant Γ_γ values were also taken from the prior work for the ensembles of both s-wave and p-wave resonances, as were the numbers of degrees of freedom taken to represent the frequency functions for the widths. The problem then was reduced to choosing Γ_n and Γ_f values as a function of energy so that the experimental average cross sections for capture and fission are reproduced; these widths also control the scattering and total cross sections given by the data file. These parameters are normally used in processing programs which recognize the cross-section fluctuations implied by known distribution laws for widths and spacings; Γ_f and Γ_n values are entered into the file as a function of energy to allow "intermediate structure" to be represented. The evaluation problems are therefore to represent correctly both the average cross sections and that part of the structure not likely to be the result of width and spacing fluctuations. The presence of such structure in ^{235}U has been demonstrated rather strongly for fission, but only weakly for capture.[4] Furthermore, representing the structure forces the file to reproduce average cross sections close to the experimental ones even over relatively narrow intervals.

In the work presented here the local-average "structure" cross sections are given at about 120 points, and these average cross sections were subsequently used by M. Bhat[5] to define average Γ_f and Γ_n values at the 120 energies. Between these points the average cross sections are assumed to vary linearly with energy, but enough points were chosen that only a small error ($\lesssim$ 1% for an interval as large as one lethargy unit) would be made if a processing code should assume that the parameters themselves vary linearly with energy within these intervals.

In this report the terms "average cross section" and "integral cross sections" always refer to the extreme thin sample or infinite dilution case for which there is no self shielding.

II. AVERAGE CROSS SECTIONS FOR BROAD GROUPS

Data Selection and Normalization

All the energy-dependent capture and fission data of interest were obtained using linear electron accelerators. Table 1 lists each data source, the energy region on which any data renormalization was based, the renormalization factor applied, and the energy range through which

Table 1. Data Selection and Renormalization

Author	Normalization Energy	Renormalization Factor	Utilization Range (keV)
		Fission	
Deruytter[a]	2200 m/sec	1.007[b]	to .0205
ORNL-RPI[c]	2200 m/sec (indirect)	1.013[b]	to 1.0
Gwin[d]	2200 m/sec	1.007[b]	to 25.
Blons[e]	60 - 300 eV	.983	.3 to 25.
Perez[f]	60 - 300 eV	.985	.3 to 25.
		Capture	
ORNL-RPI[c]	Resonance integral[g]	1.000	up to 3
Gwin[d]	0.02 - 0.03 eV	1.000	up to 25
Perez[f]	60 - 300 eV	1.019	0.3 to 25

[a] Proc. Helsinki Conf. Nucl. Data for Reactors, vol. 1, p. 127 (1970).

[b] Based on σ_f (2200 m/sec) = 584.5 b.

[c] de Saussure et al., ORNL-TM-1804 (1967). Assumed $^{10}B(n,\alpha)$ has (1/v) shape.

[d] R. Gwin, Oak Ridge National Laboratory, letter to R. A. Dannels, NNCSC, 12-20-72. R. Gwin et al., NSE, 59, 79 (1976).

[e] Nucl. Sci. Eng. 51, 130 (1973).

[f] Perez et al., ORNL-3696 and Nucl. Sci. Eng. 52, 46 (1973).

[g] A capture resonance integral for the interval $0.45 < E \leq 1.0$ eV was used.

each data set was used. This table reflects the fact that such measurements are usually based upon a normalization at the low-energy end of the range studied, with the normalizing cross section taken from the literature. An evaluator is then free to adjust this normalization without making any challenge to the measurement itself. The energy dependence of each input data set also depends upon the shape of a reference cross section as well as the validity of the experimental techniques.

For the fission cross sections the evaluation strategy was to choose the 2200 m/sec cross section (then) believed to be the choice for ENDF/B-IV, 584.5 b, and renormalize to that value the results of de Saussure et al.,[6] Deruytter,[7] and Gwin et al.[8] [The final ENDF/B σ_f(2200 m/sec) = 585.7 ± 2.3 b[9] was not considered sufficiently different to warrant reworking the evaluation.] Earlier work was included only through its effect on the normalization of the de Saussure (ORNL-RPI) results. The results of Lemley[10] are excluded here because within the range of interest he gives an energy dependence which is inconsistent with the other data sets to the extent of about 5%. The results of Blons[11] and of Perez et al.[12] were originally given based on normalization to the results of de Saussure et al.[13] in the range 100 to 200 eV, but for the present effort these experimental results have been renormalized over the range 60 to 300 eV to the average value obtained from the three experiments first listed. Beyond the selection of data and the indicated renormalizations, the present evaluation is based on a simple average of data from the listed experiments. Within about 0.5% the fission values given in this evaluation are based on the shape vs. energy of the $^{10}B(n,\alpha)$ cross section as given by Sowerby et al.[14]

For capture cross sections the procedure was similar to that used for fission, but differed in that relatively few measurements were considered and only the results of Perez et al.[12] were renormalized as shown at the bottom of Table 1. There is no apparent incentive to alter the normalization of Gwin's values[8] since the 2200 m/sec capture cross section was not expected to change appreciably for ENDF/B-IV. If the shapes of evaluated capture and fission cross sections below 0.5 eV had been changed, the results of Gwin's normalization procedure might have been affected since two constants were fitted in this region to obtain capture cross sections. The de Saussure (ORNL-RPI)[6] normalization was based on an absorption resonance integral below 1 eV obtained from the difference between total and scattering cross sections; this chosen absorption resonance integral is not directly affected by reevaluations of the 2200 m/sec cross sections. de Saussure et al. subtracted their own fission resonance integral to obtain their capture normalization; and, since de Saussure's original fission integrals in this region match the integrals recommended in this study, it is reasonable to assume that the value of the (0.45 to 1.0 eV) fission resonance integral taken by de Saussure is correct and that his capture normalization is as valid as when it was published. Gwin notes[15] that his own fission resonance integral agrees with the ORNL-RPI work (52.4 b), but his absorption resonance integral is 2.3% higher at 59.3 b. Thus, because σ_c is much smaller than σ_f, if Gwin renormalized to the capture resonance integral chosen by de Saussure, his capture results would be lowered 17%! Such a renormalization would improve agreement between the two experiments for energies below 100 eV, but catastrophically worsen agreement in the region of importance to this evaluation. The two measurements do not have the same shape.

The capture cross sections given in this report are mainly based on the shape of the $^{10}B(n,\alpha)$ cross section given by Sowerby et al.[14] Had this shape been used consistently, the output capture cross sections would have been lowered for this purpose by ~ 0.3% at ~ 0.1 keV, ranging up to about 1% at 3 keV. Above 3 keV the change would have been a constant 0.3%.

Results

Table 2 shows experimental fission integrals over convenient intervals in the region below 100 eV, giving both raw and renormalized experimental results, the average values recommended as the evaluated output, and the ratio of the output average to ENDF/B-III integrals. Note that from 5.0 eV to 20.5 eV the renormalized Deruytter fission integral agrees with the output average to ~ 0.1%, so no renormalization is required to take his experiment into account. Table 3 gives similar information for average fission cross sections in the energy range of the evaluation. Note in both tables that the present evaluation definitely gives lower fission cross sections than ENDF/B-III. The evaluation is consistent with the cross sections chosen for energies in the range 25 - 100 keV, where evaluation task-group guidelines[16] called for cross sections 2% above those given by Gwin.

There is in Table 3 an effort to split the decimal intervals from 0.1 to 0.2 and from 0.2 to 0.3, etc., even though the formal data sources do not give these breakdowns. The divisions were made on the basis of earlier private communications and study of the structure histograms described below, together with the requirement that the average of the values for the subintervals be the evaluated result for the full decimal interval. Because of the methods used, there is perhaps twice the uncertainty in the split between the two subintervals as in their average.

For the ^{23}U(n,γ) reaction, Table 4 shows the values of capture integrals in the resonance region, and Table 5 the values of input average cross sections as renormalized and the final evaluated average cross sections. Just as for fission, the results of Perez _et al._[12] were renormalized because their original normalization was based on a capture integral in the 100- to 200-eV range which was taken from the literature (ORNL-RPI[6]).

The differences between the ORNL-RPI and Gwin capture values appear to be systematic, so one cannot be very sure about the uncertainty in averaged results such as those presented here. If the capture cross sections are crucial to any system, a more thorough evaluation of existing data through the resonance region and above is called for. Some guidance in the resonance region could also be obtained from study of total cross sections and from the average ratio of capture and fission cross sections observed in integral experiments. The results of such a study in the resonance region would be likely to also affect at least the normalization of data in the unresolved resonance region.

Table 6 summarizes the broad-group average cross sections obtained in this evaluation, along with the "average alpha" values obtained from them. Figures 1 through 4 show the evaluated average fission and capture

Table 2. ^{235}U Fission Integrals. All values in units barn-eV. The first column in each case gives the experimenter's results and the second the results following normalization per Table 1.

Energy Range (eV)	Output Average	Deruytter[a] raw	Deruytter[a] norm	ORNL-RPI[b] raw	ORNL-RPI[b] norm	Gwin[c] raw	Gwin[c] norm	Average/(ENDF-III)[d]
0.5 - 0.7	13.46	13.26	13.35	13.45	13.62	13.32	13.41	.983
0.7 - 1.0	17.05	16.66	16.78	17.09	17.31	16.94	17.06	.980
1.0 - 1.8	29.4	29.11	29.31	29.09	29.47	29.3	29.5	.991
1.8 - 5.0	50.8	49.96	50.3	50.7	51.4			(.976)
5.0 - 7.4	62.7	62.11	62.5	(62.2)	63.0			
7.4 - 10	222	220	221.5	(222)	225	217	218.5	.961
10 - 15	216	214.4	215.9	215.6	218	213	214.5	.952
15 - 20.5	318	316	318	320	324	311	313	.973
20.5 - 33	445			443	449	439	442	.957
33 - 41	495			498	504	483	486	.964
41 - 60	923			924	936	905	911	(.974)
60 - 100	963[e]			967	980	939	946	1.009
100 - 300	4134[e]			4166	4220	4021	4049	.991

[a]A. Deruytter, Proc. Conf. Nucl. Data for Reactors, vol 1, p. 127 (1970).

[b]G. de Saussure et al., ORNL-TM-1804 (1967). (The author gives only the sum of the values in parentheses.) These results assumed a (1/v) shape for the ^{10}B(n,α) reference cross section.

[c]R. Gwin, Oak Ridge National Laboratory, letter to P. A. Dannels, NNCSC, 12-20-72. R. Gwin et al., NSE, 59, 79 (1976). As given in this letter, fission integrals are normalized to a 2200 m/sec fission cross section of 580.2 b. The listed values are as given in Gwin et al., NSE, 59, 79 (1976), except for a shift of 4 barn-eV from the 41 - 60 eV interval to the 60 - 100 eV interval. These results assumed the ENDF/B-III shape for the ^{10}B(n,α) reference cross section (M. G. Sowerby et al., Ref. 14). This shape has dropped ∿ 0.7% below (1/v) by 200 eV.

[d]Based on SUPERTOG results obtained through R. Q. Wright, CSD, UCCND. In the region from 1 to 82 eV the ENDF/B-IV values, shown in parentheses, should be the same as would have been obtained using version III.

[e]If all cross sections had been based on the ^{10}B(n,α) reference shape of Sowerby et al., Ref. 14, the output fission integral from 60 to 300 eV would have been lowered about 0.3%.

Table 3. Average $^{235}U(n,f)$ Cross Sections. The first column for each experiment gives the experimenter's results and the second column the results renormalized according to Table 1. Values in [brackets] were not utilized in the averages, based on the guidelines of Table 1. All cross sections are in barns.

Energy Range (keV)	Output Average[a]	ORNL-RPI[b]		Gwin[c]		Perez[d]		Blons[e]		Average[f]/(ENDF/B-III)	Average/Gwin
		raw	norm	raw	norm	raw	norm	raw	norm		
.080 - .100	25.05	25.4	25.7	24.2	24.4	[25.6]	[25.2]	[25.6]	[25.2]		1.035
0.1 - 0.2	21.00	21.0	21.3	20.5	20.7	[21.0]	[20.7]	[21.0]	[20.6]	.992	1.024
0.10 - 0.15	22.5			21.9	22.2	[22.6]	[22.3]				1.027
0.15 - 0.20	19.5			19.0	19.2	[19.4]	[19.1]				1.026
0.2 - 0.3	20.5	20.9	21.1	19.74	19.89	[20.9]	[20.6]	[20.8]	[20.4]	1.022	1.039
0.20 - 0.25	21.5			20.7	20.9	[21.9]	[21.6]				1.039
0.25 - 0.30	19.5			18.8	18.9	[19.9]	[19.6]				1.037
0.3 - 0.4	13.12	13.16	13.33	12.75	12.84	13.34	13.13	13.42	13.19	.976	1.029
0.4 - 0.5	13.59	13.76	13.94	13.12	13.22	13.95	13.73	13.71	13.47	.988	1.036
0.5 - 0.6	15.22	15.34	15.54	14.66	14.77	15.57	15.33	15.50	15.23	1.028	1.038
0.6 - 0.7	11.50	11.59	11.74	11.11	11.19	11.71	11.53	11.72	11.52	.978	1.035
0.7 - 0.8	11.11	11.26	11.41	10.71	10.79	11.28	11.11	11.30	11.11	.994	1.037
0.8 - 0.9	8.25	8.28	8.39	7.94	8.00	8.31	8.18	8.57	8.42	.954	1.039
0.9 - 1.0	7.55	7.65	7.75	7.28	7.33	7.66	7.54	7.71	7.58	.967	1.037
1.0 - 2.0	7.32	[7.51]	[7.61]	7.08	7.13	7.52	7.40	7.55	7.42	.965	1.034
1.0 - 1.5	8.07			7.84	7.90	8.29	8.16				1.029
1.5 - 2.0	6.57			6.32	6.37	6.76	6.66				1.040
2.0 - 3.0	5.30	[5.60]	[5.67]	5.14	5.18	5.47	5.39	5.48	5.39	.945	1.035
2.0 - 2.5	5.49					5.65	5.56				
2.5 - 3.0	5.15					5.30	5.22				
3.0 - 4.0	4.75	[5.09]	[5.16]	4.58	4.61	4.86	4.78	4.94	4.86	.953	1.037
4.0 - 5.0	4.27	[4.56]	[4.62]	4.08	4.11	4.36	4.29	4.48	4.40	.955	1.047
5.0 - 6.0	3.80	[3.82]	[3.87]	3.72	3.75	3.77	3.71	4.01	3.94	.945	1.022
6.0 - 7.0	3.41	[3.66]	[3.71]	3.14	3.16	3.66	3.60	3.54	3.48	.949	1.086
7.0 - 8.0	3.15	[3.74]	[3.79]	3.05	3.07	3.22	3.17	3.28	3.22	.935	1.033
8.0 - 9.0	3.01	[2.96]	[3.00]	2.88	2.90	3.17	3.12	3.07	3.02	.973	1.045
9.0 - 10.0	3.05	[3.10]	[3.14]	3.01	3.03	3.11	3.06	3.11	3.06	.943	1.013

Table 3. Continued

Energy Range (keV)	Output[a] Average	ORNL-RPI raw	ORNL-RPI norm	Gwin[c] raw	Gwin[c] norm	Perez[d] raw	Perez[d] norm	Blons[e] raw	Blons[e] norm	Average[f]/ (ENDF/B-III)	Average/ Gwin
10.0 - 20.0	2.48			2.46	2.48	2.50	2.46	2.54	2.50	.894	1.008
10 - 15	2.65			2.63	2.65	2.67	2.63				1.008
15 - 20	2.31			2.29	2.31	2.33	2.29				1.009
20 - 30	2.14			2.11	2.13	2.16	2.13	2.20	2.16	.914	1.014
20 - 25	2.19			2.18	2.20	2.20	2.17				1.00
25 - 30	2.09[g]			2.04	2.06	2.12	2.09				1.025

[a]Proposed evaluated average cross sections.

[b]The raw values are those of Ref. 6, but have been further corrected by de Saussure to the 7.8 - 11.0 eV fission integral 240 b-eV given in Ref. 7(b) (+~ 1.5%) and to the non-1/v shape for the ^{10}B(n,α) reaction given in Ref. 14. The 1.5% renormalization was not recognized during the performance of this evaluation work; correction of this error and that indicated in footnote e of Table 2 would lower the output average values by ~ 0.7% between 80 and 300 eV, by ~ 0.5% from 0.3 to 1.0 keV, and by ~ 0.2% for energies from 1.0 to 25.0 keV.

[c]The "raw" values are from Ref. 8 and are based on the ^{10}B(n,α) cross-section shape of Ref. 14.

[d]The "raw" values are from Ref. 12 and are based on the ^{10}B(n,α) cross-section shape of Ref. 14.

[e]Based on the average values from Ref. 11, but in the 0.3- to 10-keV and 5- to 10-keV energy ranges, the values were taken from a full resolution data tape by R. Perez and adjusted a few tenths of a percent to improve agreement in the integral with the published values over these broad regions.

[f]ENDF/B-III values per private communication of O. Ozer to H. Alter, 1-9-73, giving RESEND/INTEND values for MAT 1157 from NNCSC for use of the CSEWG "Big 3" task group. Below 25 keV the Ozer fission cross sections are uniformly 1.2% lower than given by SUPERTOG through R. Q. Wright, CSD, UCCND.

[g]The value averaged from File 3 of MAT 1261, ENDF/B-IV, for this interval is 2.12 b. For this energy region the MAT 1261 values were not based on this work.

Table 4. ^{235}U Capture Integrals. All integral values in barn-eV. No renormalization of these results was performed.

Energy Range (eV)	ORNL-RPI[b]	Gwin[c]	Average	Average/ (ENDF/B-III[a])
7.4 - 10	81	95	88	1.09
10 - 15	229	276	253	1.03
15 - 20.5	190	225	208	1.08
20.5 - 33	294	344	319	1.07
33 - 41	271	317	294	1.07
41 - 60	434	502	468	1.04[d]
60 -100	477	557	517	1.22
60 -300	2525	2677	2601	1.05

[a]ENDF/B-III values obtained from MAT 1157 using SUPERTOG by R. Q. Wright, CSD, UCCND, below 60 eV.

[b]Based on R. Gwin, ORNL, private communications, 1973 and 1975. Based on summations from the data tape representing the ORNL-RPI measurements of G. de Saussure et al., as given in ORNL-TM-1804 (1967).

[c]R. Gwin, ORNL private communication (1973). Final merged values from the same experiment (R. Gwin et al., Ref. 8) are 525 b-eV for the 60 - 100 eV interval and 2641 b-eV for the 60 - 300 eV interval. Therefore, if this same method of evaluation were used again, the renormalization constant for the Perez data would be lowered by 0.7% to correct for this change.

[d]Based on the ENDF/B version 4 MAT 1261, which should not differ much from MAT 1157 in this energy region.

Table 5. Average $^{235}U(n,\gamma)$ Cross Sections. The results in brackets were not utilized in obtaining the evaluated average. The second column under Perez gives results renormalized in accord with Table 1. All cross sections are in barns.

Energy Range (keV)	Output Average	ORNL-RPI	Gwin	Perez raw	Perez norm	Average/(ENDF/B-III)[a]
0.08 - 0.10	15.7	15.0	16.4	[15.5]	[15.8]	
0.10 - 0.20	11.9	11.45	12.3	[11.45]	[11.67]	1.01
0.10 - 0.15	12.8		13.2			
0.15 - 0.20	11.0		11.4			
0.20 - 0.30	8.95	9.03	8.88	[9.02]	[9.19]	1.03
0.20 - 0.25	10.7		10.7			
0.25 - 0.30	7.1		7.1			
0.30 - 0.40	6.56	6.56	6.63	6.36	6.48	1.17
0.40 - 0.50	4.83	5.03	4.59	4.77	4.86	.93
0.50 - 0.60	4.62	5.04	4.25	4.49	4.58	.87
0.60 - 0.70	4.67	4.81	4.67	4.44	4.53	.97
0.70 - 0.80	4.91	5.07	4.82	4.75	4.84	.95
0.80 - 0.90	4.15	4.33	4.05	3.98	4.06	.94
0.90 - 1.0	5.05	5.36	4.95	4.75	4.84	1.26
1.0 - 2.0	2.98	3.26	2.97	2.67	2.72	.90
1.0 - 1.5	3.40		3.34	3.08	3.14	
1.5 - 2.0	2.56		2.60	2.26	2.30	
2.0 - 3.0	1.97	1.83	2.11	1.94	1.98	.96
2.0 - 2.5	2.20		2.36			
2.5 - 3.0	1.74		1.86			
3.0 - 4.0	1.62		1.74	1.46	1.49	.91
4.0 - 5.0	1.53		1.55	1.47	1.50	.96
5.0 - 6.0	1.42		1.42			.99
6.0 - 7.0	1.40		1.48	1.30	1.32	1.09
7.0 - 8.0	1.33		1.31	1.33	1.36	1.12
8.0 - 9.0	1.45		1.48	1.38	1.41	1.32
9.0 - 10.0	1.25		1.26	1.22	1.24	1.08
10.0 - 20.0	0.99		0.986			0.97
10 - 15	1.08		1.08			
15 - 20	0.90		0.90			
20.0 - 30.0	0.82		0.82			0.96
20 - 25	0.87		0.87			
25 - 30	0.77		0.77			

[a]Based on private communication of O. Ozer, NNCSC, BNL, to Harry Alter (1-9-73); these results were obtained using RESEND/INTEND. Up through 20 keV these version 3 capture cross sections are uniformly larger by 2.4% than those generated by R. Q. Wright, CSD, UCCND, using SUPERTOG. Thus "average alpha" values differed altogether by 3.6%. Evaluated data of MAT 1157 was used.

Table 6. A Summary of the Present ^{235}U Evaluated Average Cross Sections for ENDF/B-IV (MAT 1261)

E_{low} keV	E_{high} keV	Fission[a] Barns	Capture Barns	Alpha[b]
0.08 -	0.10[c]	25.05	15.70	0.627
0.10 -	0.15	22.50	12.80	0.569
0.15 -	0.20	19.50	11.00	0.564
0.20 -	0.25	21.50	10.70	0.498
0.25 -	0.30	19.50	7.10	0.364
0.30 -	0.40	13.12	6.56	0.500
0.40 -	0.50	13.59	4.83	0.355
0.50 -	0.60	15.22	4.62	0.304
0.60 -	0.70	11.50	4.67	0.406
0.70 -	0.80	11.11	4.91	0.442
0.80 -	0.90	8.25	4.15	0.503
0.90 -	1.00	7.55	5.05	0.669
1.00 -	1.50	8.07	3.40	0.421
1.50 -	2.00	6.57	2.56	0.390
2.00 -	2.50	5.49	2.20	0.401
2.50 -	3.00	5.15	1.74	0.338
3.00 -	4.00	4.75	1.62	0.341
4.00 -	5.00	4.27	1.53	0.358
5.00 -	6.00	3.80	1.42	0.374
6.00 -	7.00	3.41	1.40	0.411
7.00 -	8.00	3.15	1.33	0.422
8.00 -	9.00	3.01	1.45	0.482
9.00 -	10.00	3.05	1.25	0.410
10.00 -	15.00	2.65	1.08	0.408
15.00 -	20.00	2.31	0.90	0.390
20.00 -	25.00	2.19	0.87	0.397
25.00 -	30.00	2.09	0.77	0.368

[a]The fission cross-section normalization is based on the value 584.5 b at 2200 m/sec.

[b]The value given is the quotient of the tabulated capture and fission average cross sections.

[c]For the range 82 - 100 eV, $\bar{\sigma}_c$ = 16.0 b and $\bar{\sigma}_f$ = 24.6 b.

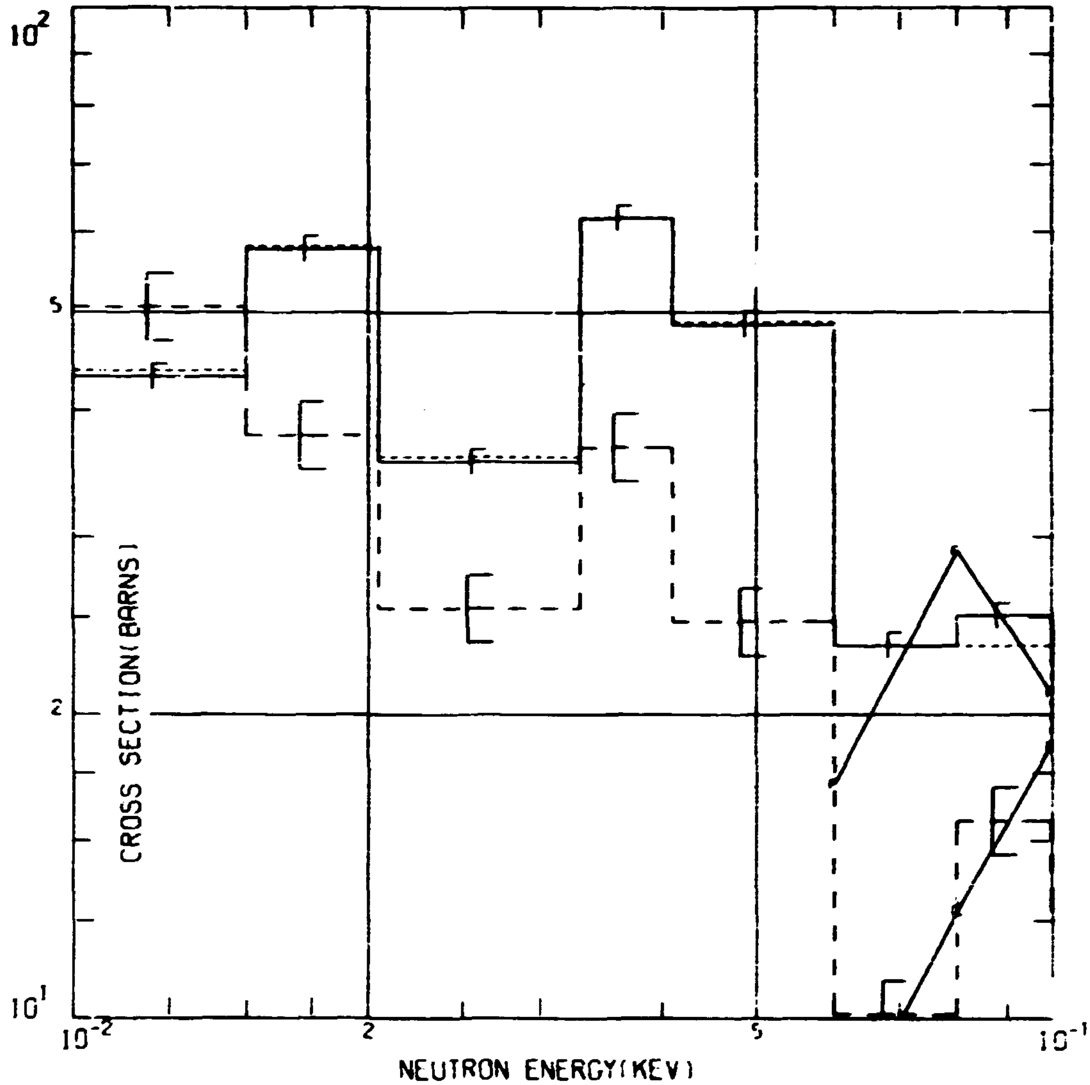

Fig. 1. A Comparison of Evaluated Average ^{235}U Fission and Capture Cross Sections with the Structure Representation for Neutron Energies Between 10 and 100 eV. The broad-dashed histogram is the evaluated average capture cross section, the solid one is the evaluated fission, and the short-dashed one illustrates an average fission cross section 1.028 times that given by Gwin (Ref. 8). The points representing the structure information for fission and capture cross sections are shown as small F and C characters joined by oblique lines to guide the eye.

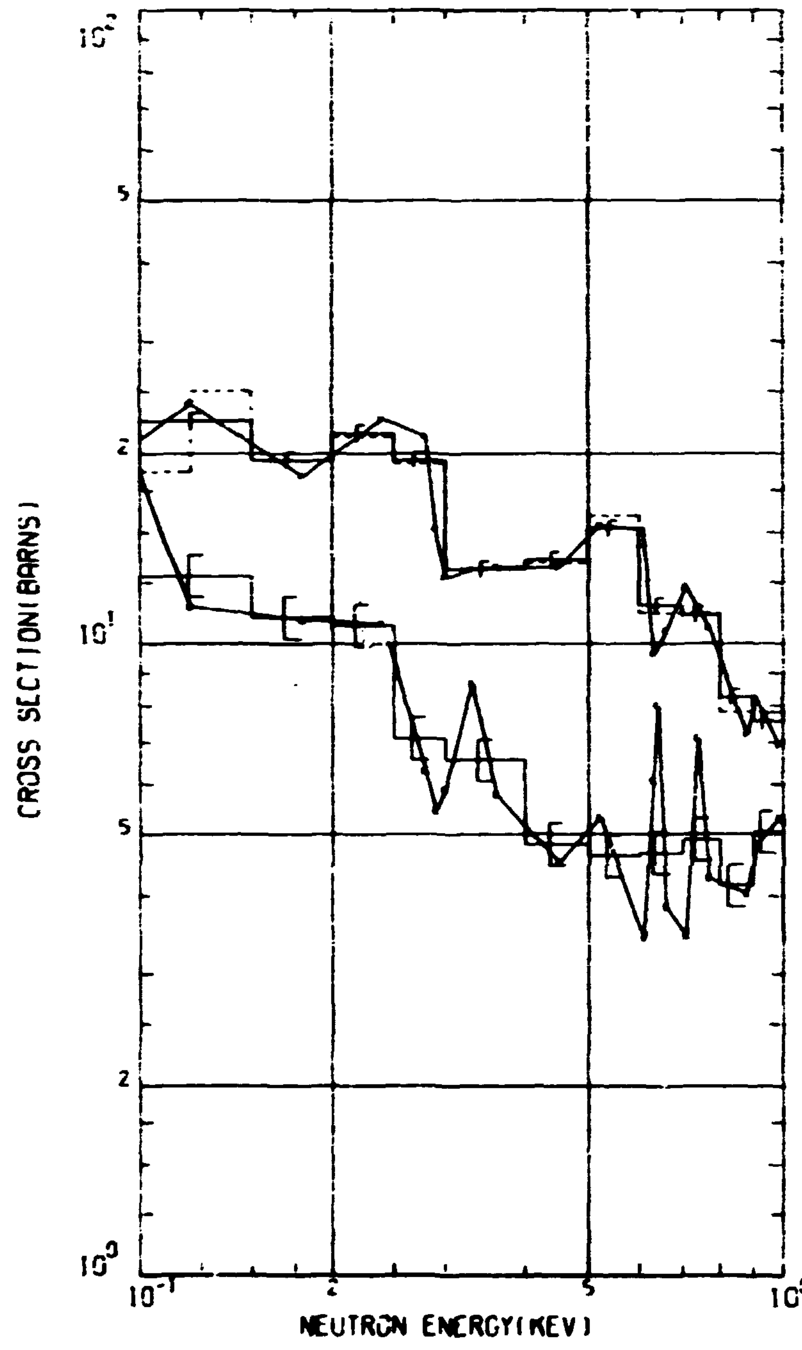

Fig. 2. A Comparison of Evaluated Average ^{235}U Fission and Capture Cross Sections with the Structure Representation for Neutron Energies Between 0.1 and 1.0 keV. The upper and lower solid-line histograms are the evaluated fission and capture average cross sections, respectively, while the dashed histogram represents an average fission cross section 1.028 times that given by Gwin (Ref. 8). The points representing the structure information are shown as small F and C characters joined by lines to guide the eye.

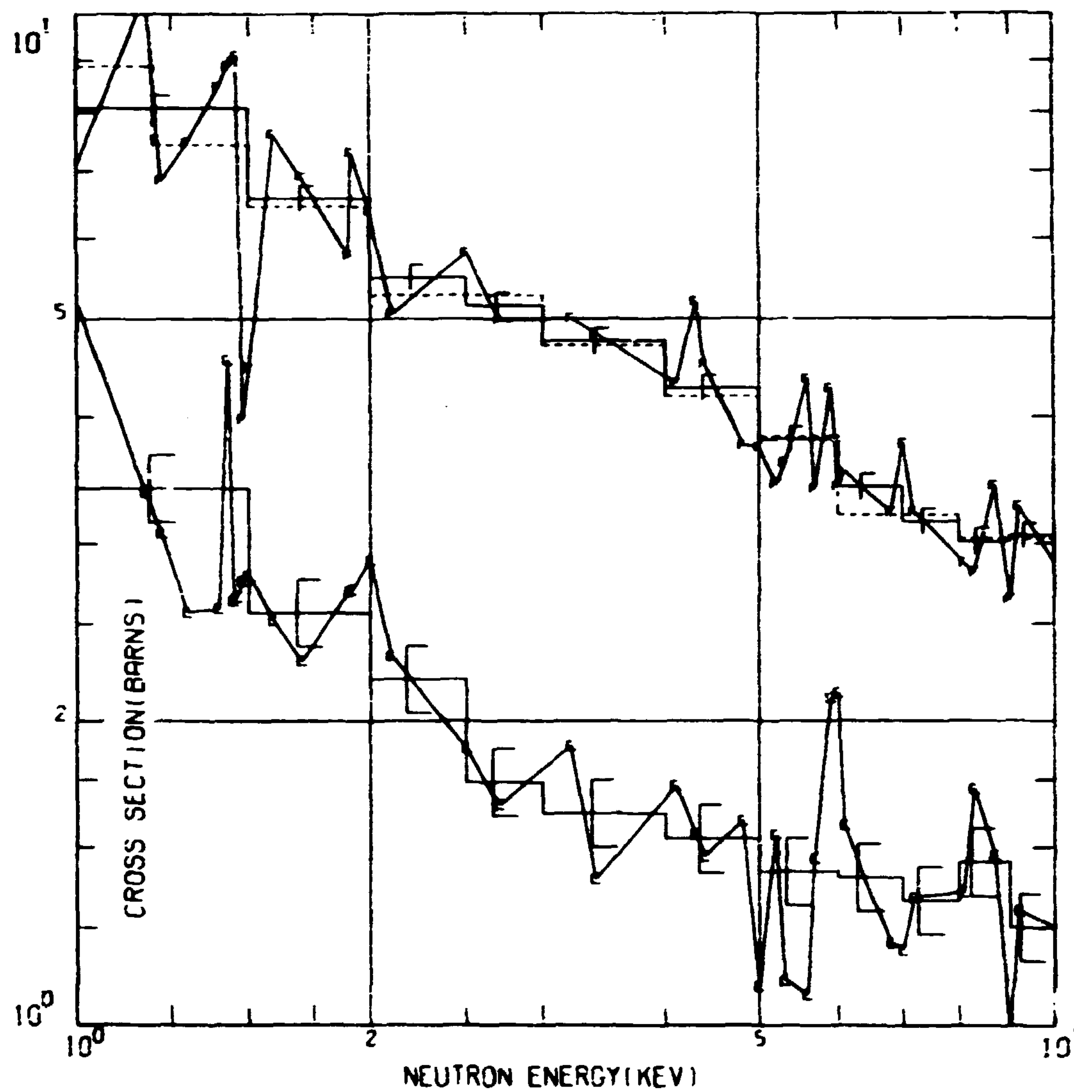

Fig. 3. A Comparison of Evaluated Average ^{235}U Fission and Capture Cross Sections with the Structure Representation for Neutron Energies Between 1.0 and 10. keV. (See Fig. 2 for an identification of the data shown.)

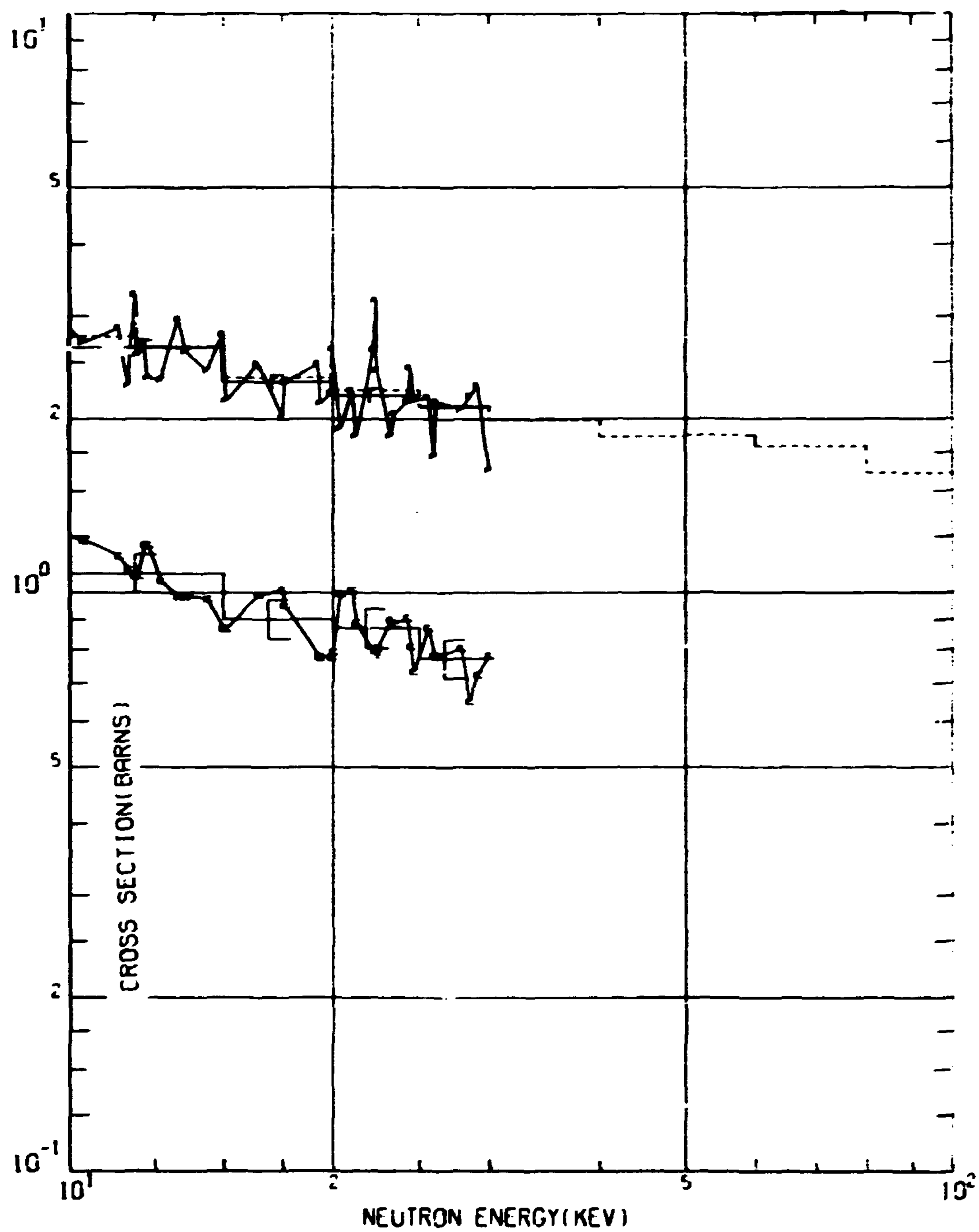

Fig. 4. A Comparison of Evaluated Average ^{235}U Fission and Capture Cross Sections with the Structure Representation for Neutron Energies Between 10. and 100. keV. (See Fig. 2 for an identification of the data shown.)

cross sections as histograms. Fission histograms obtained directly from the results of Gwin (x 1.028) are given for comparison.

Uncertainties

On the graphs of output average cross sections the uncertainties are represented as ±3% for fission and ±8% for capture. Based on scatter among the observations and some knowledge of the techniques used, more detailed cross section uncertainty estimates are made below. The uncertainty in the energy scale is inconsequential compared to the other difficulties in appropriately representing the structure.

Both of these cross sections are affected in the same way by any uncertainty in the $^{10}B(n,\alpha)$ cross section, a shape uncertainty, relative to the cross section as 1 eV, judged to be about 1% at 4 keV growing to 2% at 20 keV. A little allowance is made here for failure of the detector system accurately to reflect the reactions which occur.

The normalization of the evaluated fission cross section, counting the uncertainty in the low energy cross section, the various experimental normalizations to it, and the two or three internormalization steps often required to reach 100 eV, are estimated to have a combined uncertainty of ±2%. The corresponding normalization uncertainty on the evaluated average capture cross sections is estimated to be ±7% though, were it not for the wide discrepancy between the capture integrals of Table 4 in the region below 100 eV, a normalization uncertainty of about 4% would have been chosen. In each case these normalization uncertainties are correlated over the whole energy range.

The remaining uncertainties, generated by background subtractions and other experimental difficulties, may be partially sensed by the scatter of the average cross sections reported by the various workers, and are assumed not to be widely correlated over energy. These uncertainties are estimated to be about 2% for the evaluated average fission cross sections and about 4% for the evaluated average capture cross sections.

The rms combinations of the above uncertainty components come to about ±3% for fission and ±8% for capture as shown on the plots.

Because the concept of the structure cross sections discussed in the following sections is somewhat muddy, no effort is made to describe the uncertainty in other than the average cross sections.

III. STRUCTURE REPRESENTATION

Structure more marked than would be expected from known distributions of spacings and widths can be represented in an ENDF/B file by energy-dependent average neutron and fission widths. To make use of this option it is necessary first to determine what experimentally observed structure (averaged over ΔE much larger than the level spacing) is real, and then to estimate what part of this real structure is effectively represented by accepted spacing and width distributions. In ^{235}U it has been shown that at least the fission cross section cannot so be represented.[4]

To determine the real structure in the fission cross sections, the data sets of Gwin, Perez, de Saussure, Blons, and Lemley[6,8,10-12] were examined at full resolution to assure that they were on a common energy scale. Small changes in the flight-time zero were made to bring the others into accord with the Gwin and Lemley data sets, which agreed and were arbitrarily taken as correct. Plots were than made from each data set integrated over 10-, 100-, and 500-eV intervals within successively higher energy regions.[17] The results from the various experiments were visually compared and the major aspects of the common structure were represented with a trapezoidal approximation; below 2 keV in fission there was no disagreement even in details of the structure, and at higher energies the major features selected were fairly unambiguous. The selected energy values were then merged with those found necessary in the similar study of capture cross sections using the three available data sets, and the cross sections at the resulting ~100 energy points were adjusted to require that averages over the structure give the evaluated average cross sections of Table 6. Note that no objective criterion was utilized to indicate which fluctuations in the observed cross sections should be represented in the files. Later, points were added to assure that reconstructed average cross sections are almost independent of whether cross sections or average resonance parameters are linearly interpolated between energy points in processing codes. Table 7 gives both the fission and capture cross sections at (i.e., in the immediate neighborhood of) the 120 energy points selected to represent the structure, as well as the derived ratios of capture to fission cross section values.

To state the relationship between the evaluated average cross sections and the ENDF/B parameters more precisely, the cross sections in Table 7 reproduce the average cross sections of Table 6 using a "linear-linear" interpolation and unit weight. The parameters in the ENDF/B-IV

Table 7. Point Cross Sections for ^{235}U Fission and Capture[a]

E KEV	FISSION BARNS	CAPTURE BARNS	ALPHA	E KEV	FISSION BARNS	CAPTURE BARNS	ALPHA
0.0820	28.24	13.39	0.474	4.40	4.51	1.47	0.326
0.0865	26.43	14.72	0.556	4.80	3.75	1.59	0.426
0.0910	24.63	16.02	0.650	4.90	3.74	1.34	0.360
0.0955	22.82	17.32	0.759	5.00	3.73	1.10	0.294
0.100	21.02	18.63	0.886	5.10	3.58	1.32	0.369
0.110	22.46	15.03	0.669	5.20	3.43	1.55	0.450
0.120	23.90	11.44	0.478	5.25	3.52	1.33	0.377
0.180	18.39	10.94	0.595	5.30	3.60	1.11	0.308
0.240	22.65	10.77	0.475	5.60	4.36	1.07	0.246
0.260	21.96	8.54	0.389	5.65	3.88	1.27	0.327
0.280	21.24	6.31	0.297	5.70	3.39	1.46	0.431
0.290	15.18	5.45	0.359	5.90	4.27	2.11	0.494
0.300	12.73	5.89	0.463	6.00	3.41	2.13	0.625
0.315	12.93	7.29	0.564	6.10	3.54	1.58	0.446
0.330	13.12	8.69	0.662	6.45	3.38	1.39	0.412
0.345	13.18	7.24	0.549	6.80	3.22	1.21	0.375
0.360	13.23	5.79	0.438	7.00	3.76	1.19	0.316
0.450	13.19	4.51	0.342	7.20	3.22	1.34	0.416
0.520	15.38	5.32	0.346	8.10	2.87	1.36	0.473
0.565	15.24	4.37	0.287	8.30	2.81	1.71	0.611
0.610	15.11	3.42	0.226	8.50	3.11	1.59	0.512
0.620	12.37	4.75	0.384	8.70	3.42	1.47	0.431
0.630	9.64	6.09	0.632	9.00	2.64	0.99	0.375
0.640	9.96	8.01	0.804	9.20	3.26	1.30	0.399
0.650	10.23	5.93	0.580	10.40	2.71	1.23	0.454
0.660	10.58	3.85	0.366	11.40	2.87	1.16	0.403
0.710	12.29	3.45	0.281	11.70	2.28	1.10	0.482
0.725	11.84	5.28	0.446	11.90	3.27	1.07	0.327
0.740	11.40	7.11	0.624	12.00	2.58	1.07	0.415
0.755	11.01	5.69	0.517	12.20	2.71	1.21	0.447
0.770	10.61	4.27	0.402	12.30	2.35	1.21	0.515
0.880	7.18	4.03	0.561	12.70	2.33	1.05	0.450
0.910	8.20	4.82	0.587	13.30	2.97	0.99	0.332
0.990	6.92	5.32	0.769	13.60	2.61	0.99	0.379
1.085	8.67	4.35	0.502	14.40	2.42	0.98	0.404
1.180	10.42	3.37	0.324	15.00	2.80	0.87	0.310
1.220	6.84	3.07	0.449	15.10	2.14	0.87	0.405
1.300	7.46	2.56	0.344	16.40	2.48	0.99	0.400
1.400	8.48	2.58	0.305	17.50	2.01	1.01	0.504
1.430	8.88	4.56	0.513	17.60	2.31	0.95	0.411
1.450	8.99	2.63	0.293	18.40	2.40	0.86	0.360
1.460	9.09	2.65	0.292	19.20	2.48	0.78	0.312
1.480	3.98	2.76	0.692	19.30	2.12	0.78	0.366
1.500	4.47	2.79	0.623	19.90	2.21	0.78	0.351
1.545	6.03	2.65	0.440	20.00	2.63	0.79	0.301
1.590	7.59	2.52	0.331	20.20	2.28	0.89	0.393
1.70	6.88	2.29	0.333	20.40	1.92	1.00	0.519
1.90	5.77	2.69	0.466	21.00	2.24	1.01	0.453
1.91	7.29	2.70	0.370	21.20	1.87	0.89	0.475
2.00	6.39	2.89	0.452	21.70	2.25	0.83	0.371
2.10	5.06	2.32	0.459	22.20	2.63	0.78	0.297
2.30	5.43	2.10	0.386	22.30	3.20	0.79	0.248
2.50	5.81	1.88	0.323	22.40	2.42	0.80	0.332
2.70	4.98	1.65	0.332	22.80	2.14	0.85	0.397
3.20	5.00	1.89	0.378	23.20	1.87	0.90	0.482
3.30	4.92	1.64	0.334	23.40	2.04	0.89	0.434
3.40	4.83	1.40	0.289	24.20	2.14	0.91	0.422
3.75	4.57	1.56	0.341	24.40	2.45	0.81	0.330
4.10	4.31	1.73	0.400	24.60	2.14	0.73	0.342
4.30	5.20	1.55	0.298	25.00	2.16	0.79	0.364

[a]When these fission and capture cross sections are integrated using linear-linear interpolation between the given energies, the average cross sections of Table 6 are reproduced.

file were obtained for the energy points of Table 7 by Mulki Bhat[5] of the National Neutron Cross Section Center with the use of the code UR. This program adjusts values of $\bar{\Gamma}_n^{\circ}$ and $\bar{\Gamma}_f$ for ^{235}U for levels of spin 3 and spin 4. All other parameters were fixed without review to the values chosen earlier by Pitterlie et al.[3] and listed below, where ν_f and ν_n are the chi-square distribution parameters for fission and neutron widths.

L	J	D (eV)	ν_f	ν_n	$\bar{\Gamma}_f$ (eV)
0	3	1.0	2	1	var.
0	4	1.0	1	1	var.
1	2	1.16	2	1	.332
1	3	1.0	1	2	.127
1	4	1.0	2	2	.286
1	5	1.12	1	1	.143

The p-wave strength function was taken as 2.0 x 10^{-4} $eV^{-1/2}$, while the s-wave strength function as well as the $\bar{\Gamma}_f$ values are taken to vary from point to point. For all cases $\bar{\Gamma}_\gamma$ = 35 meV. Unlike the previous efforts, $\bar{\Gamma}_n^{\circ}$ was allowed to differ at some energy points between the J = 3 and 4 level ensembles. The previous constraint that $\bar{\Gamma}_f$ (J = 3) be twice $\bar{\Gamma}_f$ (J = 4) was also dropped at about half the energy points represented.

Figures 1 through 4 illustrate the proposed evaluated average cross sections compared to the structure representation of 100 energy points covering the range 60 eV to 30 keV. The lines which join the points representing the structure are to guide the eye; they do not follow the linear interpolation law.

Figure 5 shows for the region below 1.5 keV a consensus of experimental fission results averaged over 10-eV intervals, the proposed structure cross section from Table 7, (solid-line sawtooth), and the comparable sawtooth from ENDF/B-III, MAT 1157 (via Mulki Bhat, NNCSC). The proposed new version shows greater variation, and this trend continues to higher energies. At the right side of the figure this difference is notable in the representation of the major dip. Note that while 120 energy points are entered in the file, the structure represented there has about 30 maxima in the 25 keV interval. At the lower energies the spacings between these major fluctuations are $\sim$ 0.2 keV ($\sim$ 200 s-wave resonances) while at higher energies the typical spacings are in the neighborhood of 1 keV.

If a processing code interpolates average resonance parameters rather than cross sections between the given points, slightly discrepant group cross sections will be obtained. Such interpolation of parameters is firmly required in processing cross sections for point Monte Carlo codes.[19] A test was made for the infinitely dilute case by interpolating parameters to obtain cross sections at energies midway between those of Table 7. The average cross sections over the decimal intervals of Tables 3 and 5 using the resulting 240 points (half of them obtained by the "incorrect" interpolation procedure) were then compared with the evaluated ("correctly" interpolated) results. For the average cross sections over "decimal intervals" the differences amounted to no more than +2/3% in fission and -1.3% in capture, with maximum discrepancies in the range 600 - 700 eV. Averaged over the regions from 0.1 to 1.0 keV, 1 to 10 keV, and 10 to 25 keV, the apparent cross section deviations of the altered (240-points) averages from the input evaluated values for fission

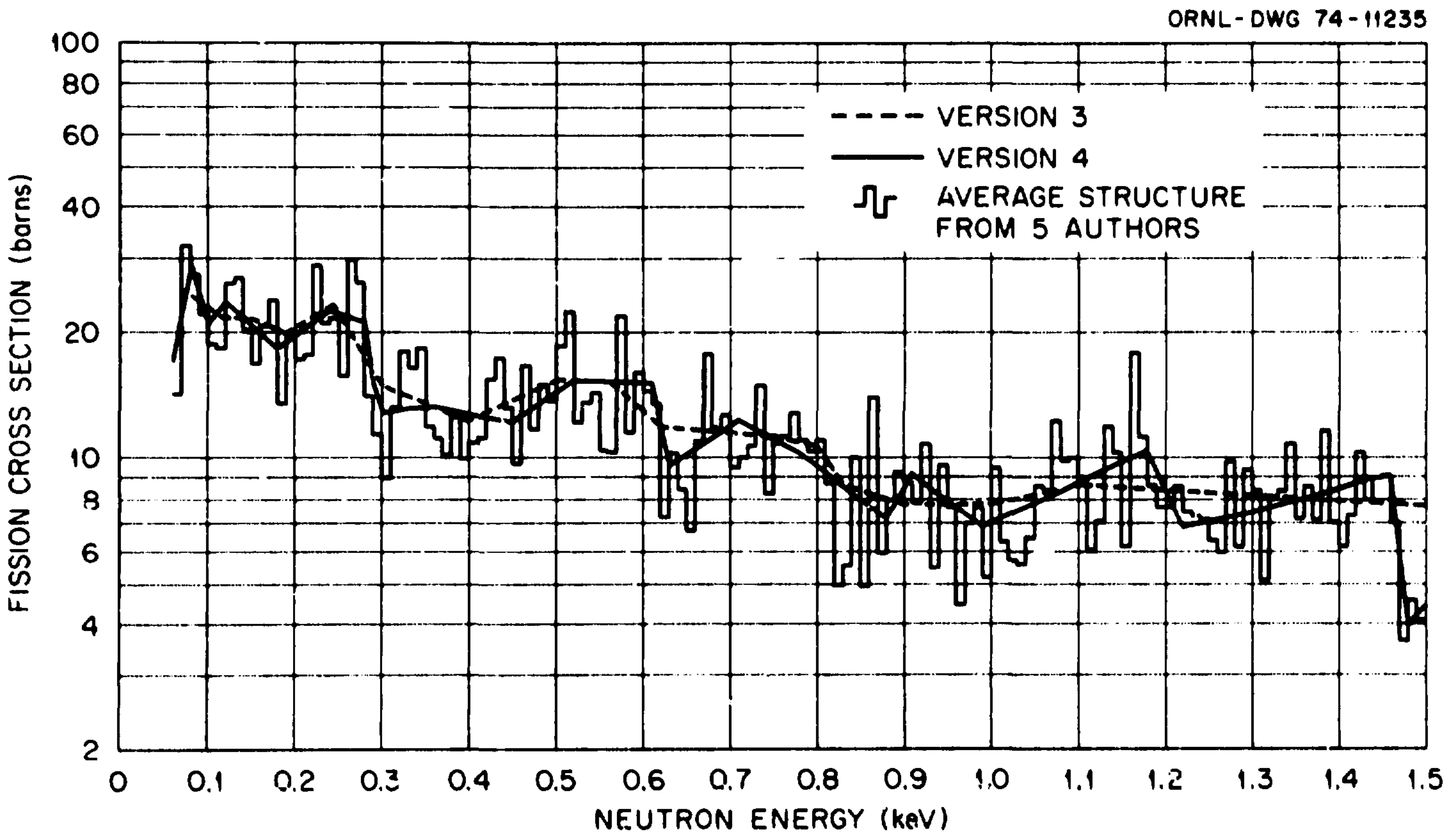

Fig. 5. Comparison of "Structure" Cross Sections with Consensus of Observations for $^{235}U(n,f)$ for Energies to 1.5 keV.

were 0.26%, 0.20%, and 0.13%; those for capture were -0.69%, -0.35%, and -0.17%. These differences are acceptable in terms of present cross-section uncertainties, although the apparent discrepancies would increase some if parameter interpolations were performed at even more energy points between those at which the parameters are given in the file.

When average resonance parameters were chosen to fit the pointwise fission and capture cross sections (by M. Bhat of NNCSC using the program UR), ENDF/B-IV total cross sections at each energy point were established once an effective scattering radius was chosen. The averages of these cross sections over broad intervals are compared in Table 8 with the experimental average total cross sections of Uttley[18] which above 1 keV are quoted to uncertainties smaller than the systematic differences between each pair of comparable values. The observations of average total cross sections were not employed in the present evaluation except to help stimulate the decision that the potential scattering cross section in the unresolved region be given the same value as in the resolved region (11.5 b) rather than the ENDF/B-III value of 10.3 b. (Uttley[18] suggested a value 11.7 b based on his experimental data.) This practice of giving little weight to observations of total cross sections seems inherently unsatisfactory, but proper consideration would require development of a fitting system to give appropriate weight to each type of data. For the case of ^{235}U, more precise (thin sample) observations of the total cross section may also be required.

Table 8. Comparison of ENDF/B-IV ^{235}U Total Cross Sections With Experimental Results of Uttley.[a]

Energy Range (keV)	Total Cross Section[b]	
	Uttley[a]	ENDF/B-IV
0.1 - 0.2	46.0 ± 1.0	45.5
0.2 - 0.3	44.7 ± 1.0	42.3
0.3 - 0.4	31.11 ± .18	32.0
0.4 - 0.5	29.73 ± .19	30.6
0.5 - 0.6	31.24 ± .21	32.2
0.6 - 0.7	27.15 ± .21	28.6
0.7 - 0.8	28.57 ± .22	28.6
0.8 - 0.9	23.69 ± .24	24.7
0.9 - 1.0	24.64 ± .25	25.2
1.0 - 1.5	24.0 ± .3[b]	23.8
1.5 - 2.0	21.6 ± .3[b]	21.2
2.0 - 3.0	19.81 ± .10	19.23
3.0 - 4.0	18.83 ± .14	18.2
4.0 - 5.0	18.09 ± .16	17.7
6.0 - 7.0	16.96 ± .18	16.6
7.0 - 8.0	16.69 ± .19	16.2
8.0 - 9.0	17.08 ± .20	16.3
9.0 - 10.0	16.47 ± .20	16.0
10.0 - 20.0	15.44 ± .10	14.9

[a] C. A. Uttley, AERE M1272 (1963).

[b] In these regions Uttley's table was collapsed assuming that the uncertainties, given for 0.1 keV intervals, were fully correlated.

IV. SUGGESTIONS FOR FUTURE EVALUATIONS

Evaluations of structure would be more straightforward using histogram interpolation of cross sections since results could be derived more defensibly by averaging experimental cross sections over selected energy intervals. So long as there remains ambiguity as to whether cross sections or average parameters will be interpolated between the energy points given in the file, the number of such energy points must be large to assure that the intended cross sections will be reproduced by processing codes. Note that when average parameters are indicated to be independent of energy, it is imperative to "interpolate" (the fixed) parameters at intermediate energies; if parameters are tabulated as a function of energy, both evaluation and processing (for multigroup codes) are simplified if cross sections are interpolated.[20] If all processing codes were able to recognize this distinction or a similar rule that parameters be interpolated whenever the ENDF/B tape lists exactly the same parameter values for successive energy points, the ambiguities could be eased. It now appears that the recommendations of Ref. 20 are generally being followed for multigroup processing.

When the ENDF/B-IV MAT 1261 file based on the present evaluation was used to give average unit-weight cross sections, it was found that the evaluated values of Table 6 were not generally reproduced within the tolerances attributable to the interpolation problem.[21] A significant portion of the discrepancies first observed seemed to depend upon the way in which a particular processing code handled the so-called fluctuation integrals which involve quantities like $\langle \Gamma_n \Gamma_x / \Gamma \rangle_{av}$ for each class of levels. If it is determined that the more important processing codes use a particular method and mesh spacing for evaluating these integrals, the same method should be used in determining the values of the average resonance parameters to be placed in the ENDF/B file.[20] Recently Hwang and Henryson have reviewed the quadrature problems and developed low-order quadrature applications which achieve good processing accuracy.[22]

It is apparent in Tables 2 and 4 that fission and capture integrals in the resonance region using ENDF/B-III parameters are inconsistent with the results of this evaluation and probably inconsistent with presently accepted 2200 m/sec values. The resolved resonance parameters for ENDF/B-IV are the same as for version 3, so version 4 integral fission cross sections in the resolved region will be about 2-4% larger than is indicated by the data considered in the present evaluation. Perhaps in the next years it will be possible to review this evaluation in the resolved resonance range taking into account all present data including experimental resonance integrals.

In performing this work the author was impressed with the arbitrariness of the procedure he found himself using to decide which portion of the observed structure should be reflected in changes in the "unresolved parameters" entered into an evaluated file. The method of inserting the structure into the file was also somewhat arbitrary. Some benchmarks in addition to the average cross section are required, and the intuitive guide (that the most dramatic structure should be reproduced if it covers an energy interval containing many resonances) seems too qualitative.

Additional relevant information is contained in thick-sample transmission and self-indication reaction data, but no effort was made to test against such data[23] for this evaluation. Future evaluations could be more authoritative if such data were taken into account. Alternatively, quantitative criteria could be set up to permit decisions based on comparison of observed cross-section fluctuations to those computed from representative "ladders" of resonances obeying the fluctuation laws assumed in the processing codes.

Beyond the details of the problems encountered in this evaluation, one should recognize that a good share of the structure in the "unresolved resonance" region _is_ resolved in a practical sense, though a resonance analysis would miss many levels which are weakly excited. For example, existing fission measurements at ORELA[24] using the 150-meter flight path resolve all the structure that would be sensed by neutrons with energy less than 5 keV in a commercial fast reactor; improved resolution is available if needed. This observation implies that evaluated resolved resonance files might contain fits to observed ^{235}U cross sections up to energies much higher than 82 eV, it being unimportant to practical applications that many small resonances are missed. For so much of the energy region as could be covered in this manner, the arbitrariness and propensity toward error connected with the "unresolved resonance region" could be avoided. About 130 resonances are now indicated for ^{235}U in ENDF/B-III or IV. With twice this number one could reproduce the data to 0.2 keV, and with a total of about 600 resonances one could reach 1 keV.

At the upper end of the interval the Doppler broadening for ^{235}U ($\sim$ 10 eV at 20 keV for ^{235}U at 830° K) is many times as large as the resonance spacing, so use of the unresolved resonance technique for this nuclide may not be required for neutron energies above $\sim$ 10 keV; any observed intermediate structure could better be represented by a "smooth" cross section interpolated between energy points spaced as closely as necessary to represent the structure. Such a conclusion is supported by the work of Bramblett and Czirr[23] but should be tested against rigorously calculated self-shielding factors before action is taken to modify the energy range now covered.

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