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R-MATRIX RESONANCE ANALYSIS AND STATISTICAL PROPERTIES OF THE RESONANCE PARAMETERS OF 233U IN THE NEUTRON ENERGY RANGE FROM THERMAL TO 600 eV

L. C. LEAL, H. DERRIEN, J. A. HARVEY, K. H. GUBER, N. M. LARSON, AND R. R. SPENCER

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ABSTRACT

The R-matrix resonance analysis of experimental neutron transmission and cross sections of ^{233}U, with the Reich-Moore Bayesian code SAMMY, was extended up to the neutron energy of 600 eV by taking advantage of new high resolution neutron transmission and fission cross section measurements performed at the Oak Ridge Electron Linear Accelerator (ORELA). The experimental data base is described. In addition to the microscopic data (time-of-flight measurements of transmission and cross sections), some experimental and evaluated integral quantities were included in the data base. Tabulated and graphical comparisons between the experimental data and the SAMMY calculated cross sections are given. The ability of the calculated cross sections to reproduce the effective multiplication factors k_{eff} for various thermal, intermediate, and fast systems was tested. The statistical properties of the resonance parameters were examined and recommended values of the average s-wave resonance parameters are given.

I- INTRODUCTION

In support of the Nuclear Criticality Safety Program (NCSP), the evaluation of the neutron resonance parameters of ^{233}U was performed in the neutron energy range up to 600 eV with the Reich-Moore Bayesian code SAMMY.[1] A previous SAMMY resonance analysis of the experimental cross sections of ^{233}U was made at the Japan Atomic Energy Research Institute (JAERI, Tokai-Mura, Japan) by one of the authors of the present paper.[2] Due to the poor experimental resolution of the data available at that time in the high energy range, the analysis was performed only up to 150 eV neutron energy; another limitation of the accuracy of the results was due to the experimental transmission data of Kolar et al.,[3] available from the Bureau Central de Mesures Nucleaires (BCMN, Geel, EURATOM, Belgium), but obtained from a sample too thin for an accurate determination of the absolute value of the total cross section. Accurate resonance parameters are needed at higher energies for the calculation of the self-shielding factors of cross sections which show strong fluctuations. To enable a SAMMY analysis at energies above 150 eV, two new high resolution measurements were performed at the Oak Ridge Electron Linear Accelerator (ORELA): (1) neutron transmission measurements of thicker samples cooled to 11K to reduce the Doppler effect, at a fight path of 80 m, by Guber et al.;[4] (2) fission cross section measurements at a flight path of 80 m by Guber et al.[5] with an experimental resolution much better than any of the previous measurements. With the results of these two new measurements in the experimental data base, the energy range of the SAMMY resonance analysis has been extended to the neutron energy of 600 eV.

The first multilevel-multichannel resonance analyses of the ^{233}U neutron cross sections were performed by Moore and Reich[6] and by Vogt[7] in the energy range below 12 eV. Bergen and Silbert[8] extended the analysis to the energy range of 20 eV to 60 eV. These analyses used both the single level and multilevel-multichannel formalism; it was shown that the variation of α (ratio of the capture cross section to the fission cross section) could not be reproduced by the single level parameters. A more extensive work was performed by Reynolds and Steiglitz[9] who used the least square fitting code MULTI[10] for the analysis of the fission and capture cross section of Weston et al.[11,12] in the energy range thermal to 60 keV. The parameters of Reynolds and Steiglitz were converted to Adler-Adler parameters by deSaussure[13] with the code POLLA[14] for the ENDF/B-V evaluated data file and are still used in the current version of ENDF/B-VI. The single level Breit-Wigner and Adler-Adler analyses were also performed by Kolar et al.,[3] Cao et al.[15] and Nizamuddin.[16] The results of the single level analysis of Nizamuddin et al. in the energy range from thermal to 100 eV were used with a large background contribution in the first version of ENDF/B-IV and JENDL-3. None of these evaluations proved to be satisfactory. The JAERI evaluation,[2] which is used in the current version of the Japanese Evaluated Nuclear Data Library (JENDL) and of the European file JEF, brought a large improvement compared to the previous evaluations by allowing accurate calculation of the cross sections over the energy range up to 150 eV. The present evaluation extends the energy range to 600 eV and improves the accuracy of the parameters, taking advantage of the excellent experimental conditions of the new ORNL neutron transmission and fission data.

In Section II of this report the general conditions and the method of the analysis are presented. The experimental data base is described in Section III. The results of the analysis and the comparison of the calculated cross sections with the experimental data and the results of other evaluations are presented in Section IV. The statistical properties of the resonance parameters and their average values are examined in Section V.

II- GENERAL CONDITIONS AND METHOD OF ANALYSIS

II-1- The ^{233}U nucleus, the spin of the resonances and the fission channels

The ^{233}U nucleus has ground state spin and parity of $5/2^+$. The states of the ^{234}U compound nucleus excited by the capture of s-wave neutrons in the ^{233}U nucleus have spin and parity of 2^+ and 3^+. When starting a Reich-Moore analysis of the experimental data in the resolved resonance region, the resonances should be partitioned into the two spin states 2^+ and 3^+. There is no direct measurement of the spin of the ^{233}U resonances. Previous evaluations have used the trial-and-error method to find a combination of spin which allowed the best fit of the interferences between resonances; the result depends strongly on the first guess of the combination of spin. Furthermore, according to the theory, the set of partial fission widths when using several fission channels is not unique unless other physical information is available.[17] The strong overlapping of the resonances due to a small average spacing and a rather large average total width render the analysis even more complicated. Some of the asymmetry appearing in the wings of the resonances could be due to unresolved small resonances or multiplets and not to the interferences between resonances. The effect of hidden small resonances in ^{233}U cross sections was examined by Koenig and Michaudon[18] and by Nizamuddin and Blons;[16] they concluded that the physical meaning of the resonance parameters, especially of the spin of the resonances, obtained from a multilevel analysis of ^{233}U cross sections could be questioned. However, it is important to reproduce the exact shape of the experimental cross section for an accurate calculation of the self-shielding factors; only a multilevel formalism can reproduce the shape of the fission interferences in the fissile nuclei resonances. The physical meaning of the resonance parameters is a separate problem which should be taken into account in the study of the statistical properties and average values of the parameters.

Some information on the 2^+ and 3^+ fission widths could be obtained from the threshold energy of the fission channels and could help to choose the spin of the resonances. By assuming that the sequence of levels of the nucleus highly deformed at fission is similar to that of the nucleus at its stable deformation, the lowest fission channel corresponds to the 2^+ state pertaining to the rotational band of the ground state. The corresponding fission threshold is believed to be at about 1.5 MeV below the neutron binding energy in the ^{234}U compound nucleus.[19,20] Some other 2^+ states exist in the low energy region of the collective band structure: gamma and double gamma vibrations, combination of bending and mass asymmetry vibrations,[21,22] giving another important contribution to the average fission width of the 2^+ resonances. The 3^+ transition states are also present in the $K=1^+$ and $K=2^+$ collective bands above the 2^+ transition states, and could be completely open for the fission. The number of open or partially open fission channels should be larger for the 2^+ states than for the 3^+ states, and, on average, the 2^+ channels are expected to lie at lower energy than the 3^+ channels. Therefore, the average fission width of the 2^+ resonances could be a few times larger than that of the 3^+ resonances.

In the SAMMY calculation only two fission channels were used for each spin state because the ENDF/B-VI format allows only two fission channels. Because the number of open or partially open fission channels is likely to be larger than 2, the fission interferences could not be taken properly into account. On average, the fission interference effects decrease when the number of channels increases. Actually, some difficulties were encountered in Ref. 2 in fitting the experimental data due to too strong interference effects with the prior parameters. Some of the difficulties were solved by using nearly orthogonal fission vectors for the neighboring strong resonances of the same spin and, in general, the final configurations were those minimizing the interferences in group of neighboring resonances. This problem was not important in the high energy region of the data where the interference effects are obscured by the experimental resolution. In the present work, the resonance parameters of Ref. 2 were taken as prior data in the energy range up to 150 eV. At higher energy, an input set of resonance parameters was obtained with the same technique as that used for ^{235}U, with a randomly assigned sign for the fission width amplitudes.[23]

II-2- The contribution of the external resonances

Before starting the analysis, the contribution of the resonances below and above the limits of the energy range analyzed was determined by using the resonance parameters of Ref. 2 with the same technique as that used in Ref. 23 for the evaluation of the resonance parameters of ^{235}U. The parameters for ten large fictitious bound levels in the energy range E<-5 eV and ten large fictitious unbound levels in the energy range E>605 eV were determined. These resonances give the energy- dependent contribution of the external levels, a contribution which is particularly important in the scattering cross section: positive at thermal energy, zero near 300 eV and negative at high energy, as it is shown on Fig. 1. The contribution from more distant levels is included in the effective scattering radius R' which is related to the distant level parameter R^∞ by $R' = a(1-R^\infty)$ where a is the channel radius. R' is a constant in the energy range analyzed and can be obtained from the fit of the effective total cross section. Five bound levels in

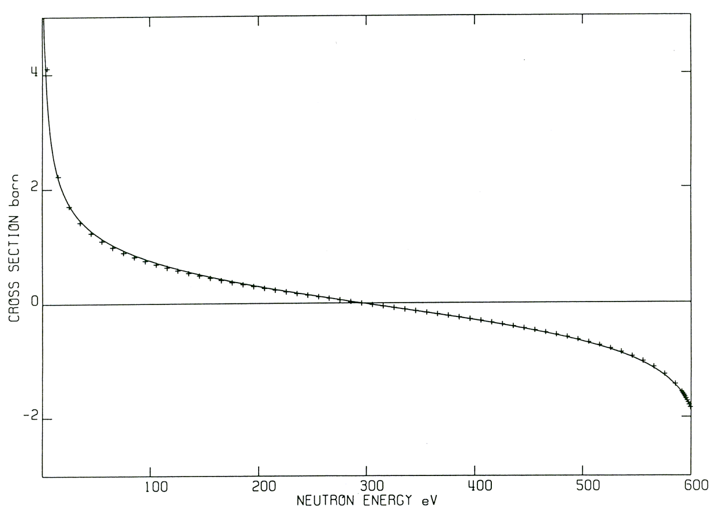

Fig. 1- ***The contribution of the external levels to the scattering cross section in the energy range 0 eV to 600 eV.*** The crosses are the cross sections calculated from an external set of resonance parameters obtained by shifting to the negative energy region and to the region just above 600 eV a preliminary set of resonances of the energy range 0 to 600 eV. The solid line represents the cross section calculated by a reduced set of 10 fictitious negative resonances and 10 fictitious resonances in the energy range above 600 eV. The parameters of these fictitious resonances were obtained from a SAMMY fit of the data represented by the crosses.

the energy range -5 eV to 0 eV contributed to the fit of the thermal range and five levels in the energy range 600 to 605 eV contributed to the fit in the energy range just below 600 eV. Note that the energy dependent contribution of the external levels could be obtained by using only two fictitious broad resonances, according to the prescription of Froehner and Bouland;[24] but using a quite large number of fictitious external resonances is more convenient for the fissile nuclei in order to minimize the spurious fission interference effects created by the fictitious external resonances. One of the arguments of Froehner and Bouland is that one should avoid increasing the number of resonances in the evaluated data file due to computer space and time; but that is not an important issue in the modern computer environment.

III- THE EXPERIMENTAL DATA BASE

Since the aim of the present work was to improve the accuracy of the ^{233}U resonance data at low energy and to extend the resolved resonance range to higher energy, new high resolution transmission and fission experiments were undertaken at ORELA. The neutron transmission measurements[4] were performed at a 79.8 m flight path with samples cooled to 11K in order to reduce the width of the resonances by decreasing the Doppler broadening by a factor of 2 compared to the experiments at room temperature; the ^{233}U sample was located 9 m from the neutron target. One series of measurements was performed with a sample of 0.00298 at/b (with Cd filter in the beam) in the energy range 0.5 eV to 80 eV, and another series was performed with a sample of 0.0119 at/b (with ^{10}B filter in the beam) in the energy range 6 eV to 300 keV. The fission cross section measurements[5] were performed at a 80 m flight path, consisting also of two series of measurements, one in the energy range 0.5 eV to 80 eV with a Cd filter, another in the energy range from 10 eV to 700 keV with a ^{10}B filter. Due to the length of the flight path and the excellent ORELA resolution, the experimental resolution was much better than any of the previous fission measurements. These new ORELA transmission and fission measurements were the primary data for the present evaluation in the energy range from 0.5 eV to 600 eV.

Most of the previous neutron cross section measurements of ^{233}U in the thermal and resolved energy range were performed before 1975. The fission data were reviewed by Deruyter and Wagemans.[25] They found large discrepancies among the data, with differences of more than 50% on the average cross sections in some energy ranges and concluded that a consistent renormalization of all the data was not possible due to unknown important sources of systematic errors. In Ref. 2, the fission data of Deruyter and Wagemans, Weston et al.[11] and Blons[26] were included in the experimental data base for the determination of the resonance parameters in the energy range up to 150 eV, and it was shown that a renormalization of the three sets of data was possible with better than 3% accuracy, at least in the energy range analyzed, by taking as reference the result of the most recent fission measurement by Wagemans et al.[27] normalized at thermal energy on the standard of Axton.[28] The same data of Wagemans et al. was used for the normalization of the new ORELA fission data. ORELA data were obtained with a relatively large quantity of fissile material coated on high-purity aluminum plates; the average cross section needed to be corrected for self-shielding and multiple scattering effects before comparison with Wagemans et al. integral value in the energy range 8.1 eV to 17.6 eV; the correction was performed by SAMMY with the current ^{233}U resonance parameters and the current evaluated neutron cross sections of Al.[29] There is no information in the publication by Blons on possible corrections for experimental effects. In Weston et al. data, the corrections for the neutrons scattered by Al were performed to the first order, which was considered sufficient by the authors of the experiment. Actually, the corrections for self-shielding and multiple scattering effects are small compared to other unknown sources of errors. The new ORELA data used in the present work were not those corrected for the experimental effects, but the corresponding uncorrected data; the corrections were directly performed by SAMMY when calculating the theoretical cross section in the fitting process.

The data of Weston et al. were the result of a simultaneous measurement of the fission and the capture cross section; the Weston capture cross section data are the only capture data available for the evaluation. The absorption cross section was normalized by Weston[11] on the absorption inferred from the total cross section of Pattenden and Harvey[30] in the energy range 1.0 eV to 2.75 eV, with an accuracy of about 3%. The evaluation of Ref. 2 has shown that the experimental capture cross sections were too small between resonances due to a possible overestimation of the background correction made by Weston et al. The same conclusion was obtained by Reynolds and Steiglitz.[9] In the present evaluation, a background of $1.5/E^{1/3}$ was added to the experimental data prior to the SAMMY calculation; this shape of background correction was suggested by Weston. Actually the statistical accuracy of the capture data was poor compared to the associated fission data. Nevertheless, accurate fits of the capture data were possible in the energy range below about 30 eV. Above this energy the capture were still included in the SAMMY experimental data base because of the presence of strong narrow resonances with enough statistical accuracy at the peaks. The ^{195}Pt resonances are seen in the original capture data; these resonances were removed prior to the SAMMY fits in the energy range up to 100 eV.

Before the new ORELA transmission experiments, the only high resolution total cross section data were those from Kolar et al.[3] taken at Geel Electron Linear Accelerator (GELINA, Geel Euratom Belgium). These data were used in the analysis of Ref. 2 and in a preliminary SAMMY calculation in the present work. The thickness of the sample used by Kolar was 0.0046 at/b, less than half the thickness of the sample in the ORELA experiment. Important normalization and background corrections were needed for consistency with the fission cross sections. Kolar data were not included in the final SAMMY runs in the present evaluation.

Several sets of experimental data are available in the thermal energy range. The total cross sections are from Moore et al.[31] Pattenden and Harvey,[30] Harvey et al.[32] The fission cross sections are from Weston et al.,[12] Deruyter and Wagemans[25] and Wagemans et al.[27] The capture cross sections from Weston et al.[12] The fission and capture data were renormalized according to Axton standard.

Some of the features of the experimental data sets selected for the SAMMY final runs are given in Table 1. The data were obtained using the time-of-flight (TOF) technique with a pulsed neutron source. The accuracy of the neutron energy depends mainly on the accuracy of the flight path length and of the timing of the neutron pulse. Therefore, the resonances could be slightly shifted when comparing independent measurements. The energy scale of the ORELA high resolution transmission was chosen as energy standard. The energy correction applied to the other data is given by the relation $aE+bE^{3/2}$, where a corresponds to the accuracy on the length of the flight path and b to the accuracy on the neutron time of flight. These parameters were obtained directly from a SAMMY adjustment or from graphical comparison of the peak energy of selected resonances.

The experimental resolution parameters and the Doppler broadening parameters are needed for an accurate description of the shape of the experimental data by SAMMY. The parameters are found in the literature or obtained directly from the authors of the measurements. The accuracy of the resolution parameters was checked in the high energy part of the data where the width of the resolution function contributes to a significant portion of the observed width of the resonance or cluster of resonances. The effective temperature of the samples, for the calculation of the Doppler broadening, was settled at 300K, with an accuracy of about 10%, for all the data taken at room temperature. For the data taken at liquid nitrogen temperature (Blons fission data) and at 11K (ORELA neutron transmission data) the effective temperatures were those recommended by the authors of the measurements, i.e. 101K and 70.6 K respectively. Since the evaluation included a large number of uncorrelated experimental data of different nature (transmission, fission and capture) it is unlikely that the errors on the temperature and resolution parameters could have a large effect on the accuracy of the resonance parameters. The most sensitive parameter is the capture width of the resonances for which one expects an accuracy of about 8% on the average value.

Table 1- Selected measurements of ^{233}U neutron transmission, fission and capture

Author	Energy Range Analyzed(eV)	Main Features
Moore et al., 1960[32]	0.020-15.0	Transmission; chopper, TOF 15.7 m Sample 0.0037 and 0.0213 at/b
Pattenden and Harvey, 1963[31]	0.080-15.0	Transmission; chopper, TOF 45 m Sample 0.00057, 0.00308, 0.01219 at/b
Weston et al., 1968[11]	1.0-600.0	Simultaneous measurement of Capture and Fission, Linac, TOF 25.2 m
Weston et al., 1970[12]	0.020-1.0	Simultaneous measurement of Capture and Fission, Linac, TOF 25.6 m
Blons, 1973[26]	4.0-600.0	Fission, Linac, TOF 50.1 m, Sample at Liquid Nitrogen Temperature
Deruyter and Wagemans, 1974[25]	0.020-15.0	Fission, Linac, TOF 8.1 m
Harvey et al., 1979[33]	0.020-1.2	Transmission, Linac, TOF 17.9 m Sample 0.00605 and 0.0031 at/b
Wagemans et al., 1988[27]	0.002-1.0	Fission, Linac, TOF 8.1 m
Guber et al., 1998[4]	1.0-80.0	Transmission, Linac, TOF 80 m
		Cd filter, Sample Temperature 11K
		Sample Thickness 0.00298 at/b
Guber et al., 1998[4]	7.0-600.0	Transmission, Linac, TOF 80 m
		^{10}B filter, Sample Temperature 11K
		Sample Thickness 0.0119at/b
Guber et al., 1999[5]	1.0-80.0	Fission, Linac, TOF 80 m
		Cd filter
Guber et al., 1999[5]	7.0-600.0	Fission, Linac, TOF 80 m
		^{10}B filter

In addition to the microscopic or differential data (from TOF transmission or cross section measurements) a variety of "integral quantities" are available within SAMMY. These integral quantities are calculated by integrating over the microscopic absorption, fission and capture cross sections, and thus are a function of the resonance parameters. The derivatives of the integral quantities with respect to the resonance parameters are also calculated by SAMMY. The Maxwellian average cross sections at thermal energies are important for the interpretation of the thermal benchmarks. They are particularly used for the calculation of the Wescott g_w factors and the K_1 parameter, defined respectively as

$$g_w = 2\sigma_x / \pi^{1/2}\sigma_{0x}$$

where w stands for fission (g_f), absorption (g_a)

and

$$K_1 = \nu\sigma_{0f}\, g_f - \sigma_{0a}\, g_a \,,$$

where σ_x and σ_{0x} are the Maxwellian-averaged cross sections and the cross sections at 0.0253 eV, respectively. Other important integral parameters are the fission and the capture resonance integrals I_c and I_f and their ratio $\alpha = I_c/I_f$. The resonance integral is defined as

$$I_x = \int_{0.5\ eV}^{20\ MeV} \sigma_x/E\,dx$$

Some evaluated ^{233}U integral data are given in Table 2.

Table 2- Evaluated integral quantities

Quantity	ENDF/B-VI Standard[33]	Axton Standard[28]	BNL[34]	Present Work
g_a	0.9996 ± 0.0011	0.9995 ± 0.0011	0.9996 ± 0.0015	1.00325
g_f	0.9955 ± 0.0014	0.9955 ± 0.0014	0.9955 ± 0.0011	1.00045
I_a			897 ± 20	917.45
I_f			760 ± 17	777.82
K_1	742.60 ± 2.40	742.25 ± 2.37	738.0	746.77
ν	2.4946 ± 0.0040	2.4950 ± 0.0040	2.493 ± 0.004	2.4946

IV- THE RESULTS OF THE SAMMY ANALYSIS

The resonance parameters file obtained from the SAMMY analysis of the experimental data in the energy range from thermal region to 600 eV neutron energy contains 769 resonances, 738 in the energy range analyzed and 31 external resonances. The resonance parameter file is given in the Appendix of this report. All the resonances were considered as induced by s-wave neutrons. Small corrections should be applied to the calculated local s-wave strength function to take into account the effect of the non-identified small p-wave neutron resonances. The average spacing of the 738 resonances is 0.81 eV which is much larger than the expected value. The meaning of the resonance parameters in the high energy range of the data, i.e., parameters of true or pseudo-resonances, will be discussed in the section on the statistical properties of the parameters.

IV-1- The thermal energy range

The cross sections in the thermal energy range must be calculated with the best accuracy possible at all energies. This accuracy was obtained in the earlier evaluation by using the cross sections evaluated directly from the experimental values since the single level or multilevel Breit-Wigner resonance parameters are not able to reproduce the shape of the experimental data where the interferences due to the fission channels are important. That is the case for ^{233}U in the energy range near the 0.17 eV resonance. The Reich-Moore formalism of SAMMY is able to reproduce with high accuracy the shape of the fission interferences. In the ^{233}U experimental data, the resonance at 0.17 eV appears as a well defined resonance in the experimental capture cross sections, but is seen in the experimental total and fission cross section as a small interference deformation; the neutron width is very small. The interference shape cannot be reproduced from only the positive energy resonances; important contributions come from the first bound levels. Some difficulties were encountered by Reynolds and Steiglitz[9] and in Ref. 2 to find a resonance configuration which could give a good description of the fission and capture cross sections. In the present work, another configuration was found which avoided the use of a resonance with an unrealistic small value of the capture width (capture width of 1 meV at 0.67 eV by Reynolds and Steiglitz and of 0.9 meV at 0.44 eV in Ref. 2). The resonance capture near 0.20 eV is considered as a doublet at energies 0.17 eV and 0.23 eV, and a very small resonance at 0.58 eV has a capture width of 25 meV. The results of the fit of the experimental data are given in Figs.2-3.

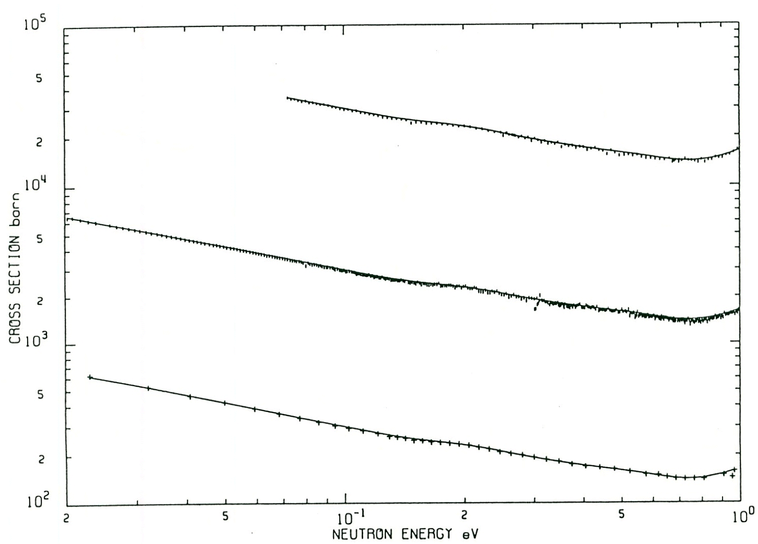

Fig. 2- ***The total cross section in the neutron energy range 0.01 to 1 eV.*** The experimental data are represented by the error bars. The solid lines are the data calculated by the resonance parameters. The experimental cross sections are, from the bottom of the figure, Harvey et al. data,[33] Moore et al. data[32] (multiplied by 10) and Pattenden et al. data[31] (multiplied by 100).

The values of the cross sections calculated from the resonance parameters at 0.0253 eV are compared to the Axton standard and to ENDF/B-VI values in Table 3. The excellent agreement between Axton and the present calculation is due to the renormalization of the experimental fission and capture data to this standard, prior to the SAMMY fit; a scattering cross section file was also created in a small energy range near 0.0253 eV according to the scattering cross section of Axton. The experimental total cross sections were not renormalized since they were the results of absolute measurements.

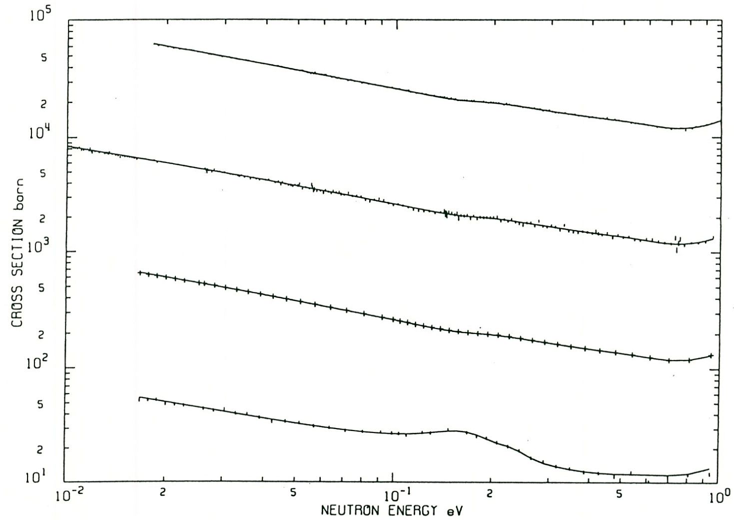

Fig. 3- ***The fission and capture cross sections in the energy range from 0.01 eV to 1 eV.*** The experimental data points are represented by the error bars. The solid lines are the data calculated from the resonance parameters. The experimental cross sections are, from the bottom of the figure, Weston et al.[12] capture, Weston et al.[12] fission, Wagemans et al.[27] fission (multiplied by 10) and Deruyter et al.[25] fission (multiplied by 100).

Table 3- The cross sections (barn) at 0.0253 eV

	Present Results SAMMY 300K	Axton Standard[28]	ENDF/B-VI Standard[34]
Fission	530.70	530.70±1.34	531.14±1.33
Capture	45.22	45.52±0.70	45.51±0.68
Scattering	12.18	12.19±0.67	12.13±0.66

The calculated cross section integrals are compared to the corresponding experimental values in Tables 4 and 5 for energy ranges below 1eV. Good agreement is evident between the experimental and calculated fission data. Good agreement is also evident between Harvey et al. experimental total cross sections and the calculated values. Note that the total cross section of Harvey et al. was the result of an absolute measurement performed with an accuracy better than 1% in order to check the accuracy of earlier data. Actually, the earlier data of Moore et al. and of Pattenden and Harvey are smaller by about 2%.

Table 4- Fission cross section integrals (b-eV) in the thermal energy range. The calculated values are those obtained at 300K from the resonance parameters. The experimental fission data were normalized at 0.0253 eV according to Axton standard. The experimental and the calculated fission integrals agree within 0.8%; the accuracy of Axton standard at 0.0253 eV is 0.25%.

Energy Range eV	Wagemans[27] b-eV	Deruyter[25] b-eV	Weston[12] b-eV	Calculated b-eV
0.01-0.020	6.92			7.01
0.020-0.050	13.89	13.92	13.95	13.91
0.050-0.400	71.43	71.69	71.79	71.07
0.400-1.000	78.46	77.58	77.81	76.93

Table 5- Total cross section integrals (b-eV) in the thermal energy range. The calculated values are those obtained at 300K from the resonance parameters.

Energy Range eV	Harvey[32] b-eV	Moore[31] b-eV	Pattenden[30] b-eV	Calculated b-eV
0.020-0.050	15.789	15.663		15.733
0.050-0.400	81.468	80.789		82.300
0.400 0.900	90.244	88.809	89.519	90.751

IV-2- The Resolved Resonance Region

Examples of SAMMY fits in some energy ranges are given in Figs. 4-8. Due to high experimental resolution, the new ORELA fission cross section and neutron transmission data were essential for the identification of the resonances in the high energy part of the analysis. A detailed comparison between the ORELA average fission cross section and the results of the SAMMY fits is given in Table 6. The experimental effects on the average fission cross sections, mainly due to the self-shielding of the U sample and the multiple scattering on the Al plates, were obtained by comparing the calculated uncorrected cross sections and the calculated corrected cross sections. The accuracy of the normalization of the experimental fission cross sections was determined by comparison with the integral value recommended by Wagemans et al.[27] in the energy range 8.1 and 17.6 eV as shown in Table 7. The agreement with Wagemans standard is better than 0.2% for Weston et al. and Deruyter et al. The ORELA value obtained by using the uncorrected data and applying the SAMMY correction is 0.5% smaller. The value calculated from the resonance parameters is 1.4% smaller. However, the Deruyter data, available with a good experimental resolution in the energy range 0.02 eV to 27 eV, normalized on Axton standard at 0.0253 eV should be considered as the best reference for the ^{233}U fission measurements up to about 30 eV. The fission integral calculated from the resonance parameters in the energy range 0.02 eV to 27 eV agrees with Deruyter value within 0.3%. The ratios of the average experimental fission cross section, capture cross section and transmission to the corresponding calculated values are given in Table 8 in 15 intervals in the neutron energy range 1 eV to 600 eV. Up to the neutron energy of about 100 eV there is a quite good agreement between the calculated cross sections and all the experimental values. Above 150 eV ORELA average fission is about 7% larger than the calculated values; in this energy range the SAMMY fit is close to the Weston data. The fit to the ORELA transmission data is very good over all the energy range of the analysis. Differences of more than 30% are seen in some energy ranges in the capture cross section; these differences, which are mainly seen in the low values, are not considered significant because of the large errors in the capture measurements.

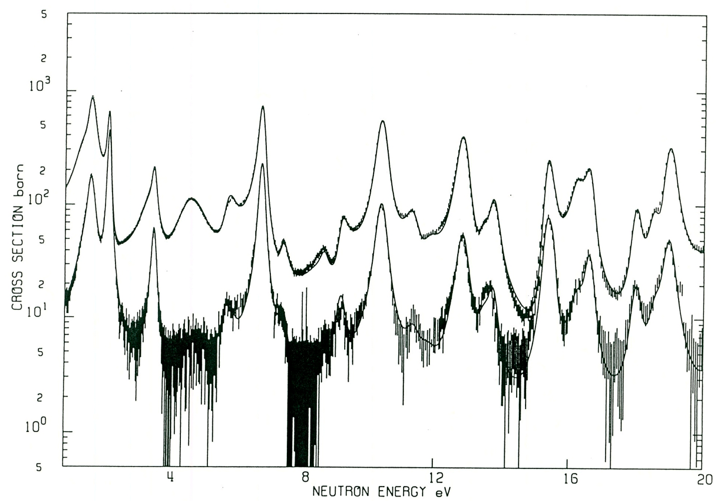

Fig. 4- ***Weston et al.[12] fission and capture cross section in the energy range 1 eV to 20 eV.*** The solid lines represent the cross sections calculated from the resonance parameters. The experimental data are represented by the statistical error bars (the upper curve is the fission cross section).

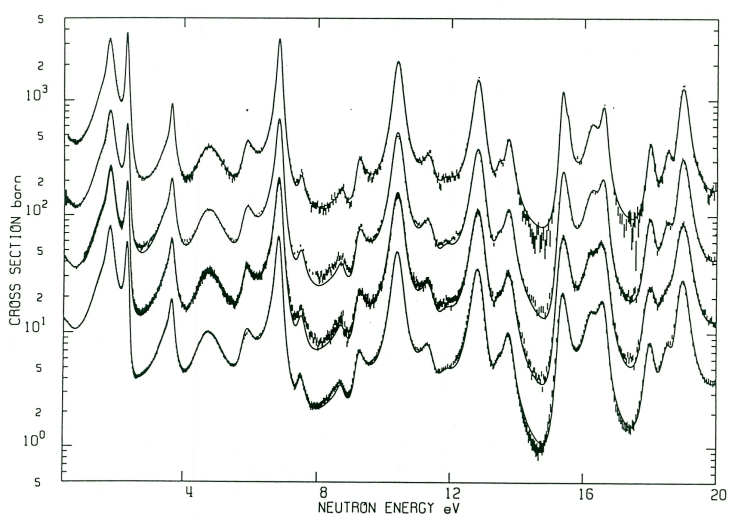

Fig. 5- ***Total and fission cross sections in the energy range 1 eV to 20 eV.*** The solid lines represent the cross sections calculated from the resonance parameters. The experimental data are, from the bottom of the figure: Weston et al.[11] fission cross section (multiplied by 0.09), Deruyter et al.[25] fission cross section (multiplied by 0.3), ORELA[5] fission cross section and ORELA[4] total cross section (multiplied by 3.0).

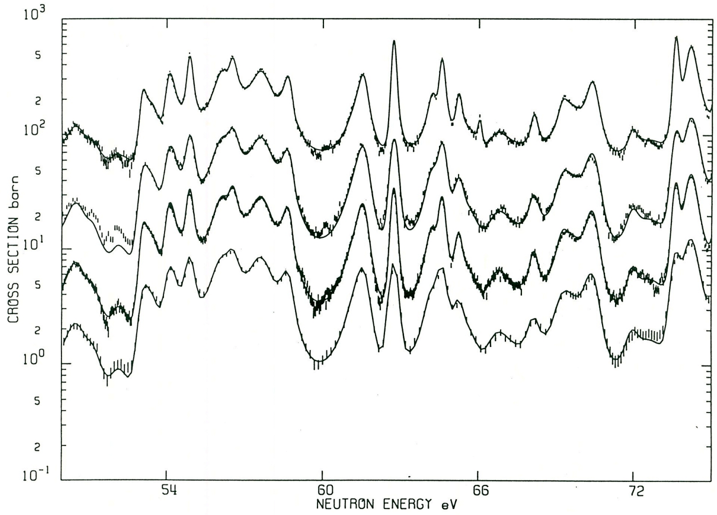

Fig. 6- ***Total and fission cross section in the energy range 50 eV to 75 eV.*** The solid lines represent the cross sections calculated from the resonance parameters. The experimental data are, from the bottom of the figure: Weston et al.[11] fission (multiplied by 0.09), Blons et al.[26] fission (multiplied by 0.30), ORELA[5] fission and ORELA[4] total cross sections (multiplied by 3.0).

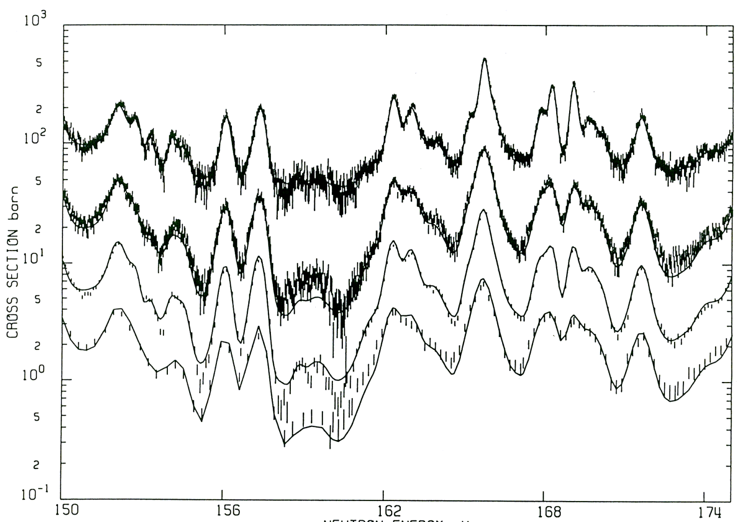

Fig. 7- ***Total and fission cross sections in the energy range 150 eV to 175 eV.*** The solid lines represent the cross sections calculated from the resonance parameters. The experimental data are, from the bottom of the figure: Weston et al.[11] fission (multiplied by 0.09), Blons et al.[26] fission (multiplied by 0.30), ORELA[5] fission and ORELA[4] total cross sections (multiplied by 3.0).

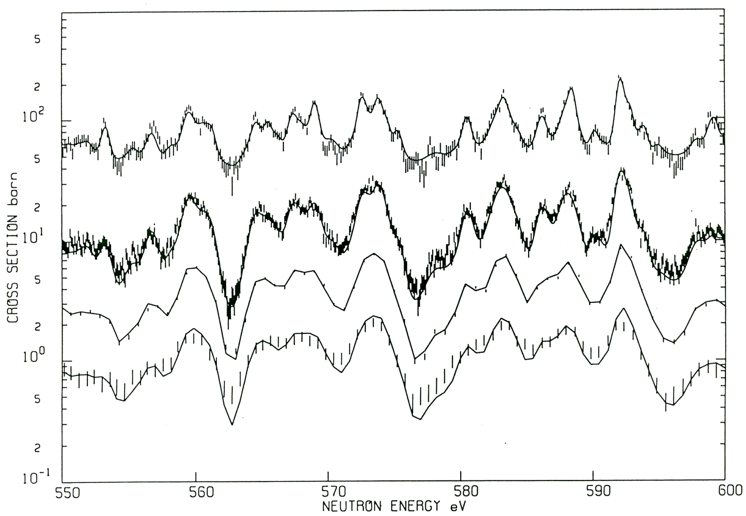

Fig. 8- ***Total and fission cross sections in the energy range 550 eV to 600 eV.*** The solid lines represent the cross sections calculated from the resonance parameters. The experimental data are, from the bottom of the figure: Weston et al.[11] (multiplied by 0.09), Blons et al.[26] (multiplied by 0.30), ORELA[5] fission cross sections and ORELA[4] total cross section (multiplied by 3.0).

Table 6- New ORELA average experimental fission cross sections and calculated cross sections. The percentage deviation between the corrected experimental and calculated data is given in the last column of the table in parenthesis.

Energy Range eV	Experimental uncorrected	Calculated uncorrected	Calculated corrected	Experimental effect %	Experimental corrected
0.40-1.00	128.16	126.46	127.48	-0.8	129.19 (1.3%)
1.00-2.10	369.21	375.57	388.92	-3.6	382.36 (1.7%)
2.10-2.75	199.01	199.48	206.73	-3.6	206.24 (0.2%)
2.75-3.00	56.82	50.08	50.24	-0.3	57.00 (14%)
3.00-15.0	103.65	102.64	104.25	-1.6	105.28 (1.0%)
15.0-30.0	93.55	94.45	94.75	-0.3	93.85 (1.0%)
30.0-50.0	41.96	40.97	40.73	0.6	41.71 (2.2%)
50.0-75.0	42.19	41.35	41.25	0.2	42.09 (1.9%)
75.0-100	37.59	37.00	36.91	0.2	37.50 (1.1%)
100-125	38.88	38.40	38.23	0.4	38.71 (1.0%)
125-150	22.73	21.33	21.09	1.1	22.47 (6.5%)
150-200	22.74	21.09	20.98	0.5	22.62 (7.8%)
200-300	24.14	23.13	23.03	0.4	24.04 (4.4%)
300-400	19.49	18.38	18.24	0.8	19.34 (6.0%)
400-500	12.17	11.13	11.01	1.1	12.04 (9.4%)
500-600	14.57	13.57	13.48	0.7	14.47 (7.3%)

Table 7- Fission cross section integral (barn-eV) in the low energy range. Weston et al. and Deruyter et al. values are those obtained after normalization on Axton standard. The ORELA value was obtained from the raw fission data corrected by SAMMY for the experimental effects.

Energy Range eV	Calculated value	ORELA[5] Exp. Cor.	Weston[11]	Deruyter[25]	Wagemans[27] recommended
8.1 - 17.6	951.49	960.19	966.31	962.94	965.2
0.02 - 27.0	3230.0			3239.8	

Table 8- Ratio of average experimental fission, capture cross sections and transmission to the corresponding calculated values in the energy range 1.0 eV to 600 eV.

Energy Range eV	Weston[11] Fission	Weston[11] Capture	Blons[26] Fission	Deruyter[25] Fission	ORELA[5] Fission	ORELA[4] Transmission
1.00 - 2.10	1.009	1.010		0.982	0.983	0.981
2.10 - 2.75	1.003	1.001		0.997	0.998	1.000
2.75 - 3.00	1.044	0.957		1.034	1.135	0.999
3.00 - 15.0	1.006	0.987		1.007	1.010	1.003
15.0 - 30.0	1.005	0.990	0.998		0.991	1.021
30.0 - 50.0	0.987	0.917	0.988		1.024	1.007
50.0 - 75.0	0.985	0.840	0.980		1.020	1.002
55.0 - 100	0.964	0.990	0.991		1.016	1.009
100 - 125	0.971	1.002	0.968		1.013	0.991
125 - 150	1.009	0.895	0.961		1.065	0.993
150 - 200	1.006	0.790	0.966		1.078	0.991
200 - 300	0.997	0.908	0.971		1.044	0.994
300 - 400	1.004	0.660	0.981		1.060	0.999
400 - 500	1.023	0.824	0.976		1.094	0.999
500 - 600	1.016	1.162	0.964		1.073	1.000
1.0 - 15.0	1.007	1.033		1.003	1.004	1.002
15.0 - 600	0.998	1.195	0.980		1.044	0.998

The experimental fission cross sections in Figs.4-8 show large discrepancies particularly in the minima of the cross sections. These discrepancies are due to experimental effects which could not be taken into account when calculating the theoretical data from the resonance parameters, because of the lack of information from the authors of the measurements. The quality of the fits could be improved by allowing background and normalization corrections in the SAMMY analyses. Attempts were made to search on the background and normalization coefficients; however, the interpretaion of the results is difficult because the background and normalization corrections are strongly correlated with each other and with the large fission widths. In this way a good fit of all the experimental data could be purely artificial. Figures 8 and 9 show the data in the upper energy range of the analysis. In Fig. 8 the theoretical cross sections were calculated with the final set of resonance parameters, i.e. after considering the results of the benchmark tests of the intermediate energy range (see **IV-3** below, with more weight on Weston et al. data). The average calculated fission cross section is about 7% smaller than the ORELA experimental value. In Fig. 9, the theoretical cross sections were calculated by the parameters obtained by fitting the

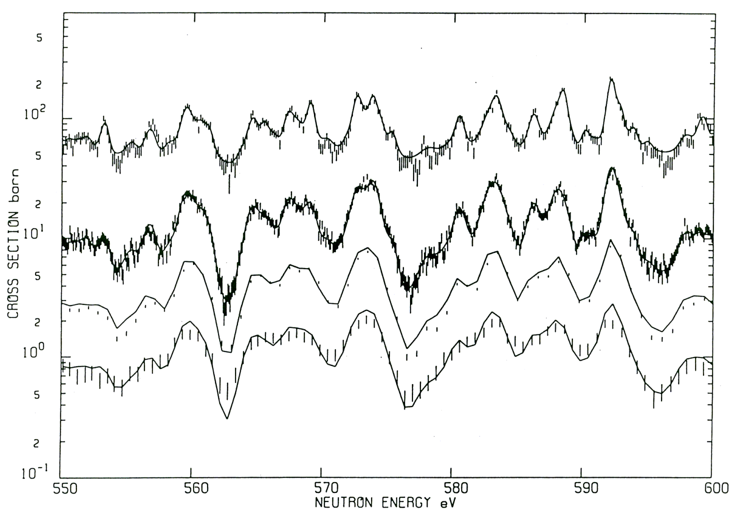

Fig. 9- ***Total and fission cross sections in the energy range 550 eV to 600 eV.*** The experimental data are the same as in Fig. 8; the calculated cross sections are from the resonance parameters obtained by fitting the microscopic cross sections only.

microscopic data only. In the sequential SAMMY fit, ORELA experimental data have the highest weight (much better experimental conditions), resulting in a good representation of the data with the resonance parameters. In this case,the average experimental fission cross section value agrees with the one calculated from the resonance parameters to within 0.5%.

The calculated average cross sections are compared to ENDF/B-VI data in Table 9. Large differences are seen above 60 eV neutron energy corresponding to an unresolved resonance region for ENDF/B-VI. Extending the resolved resonance range up to 600 eV improved the accuracy of the calculated cross section compared to the statistical calculation from inaccurate average resonance parameters.

The fission resonance integral and the capture resonance integral are 776.64 b and 139.66 b respectively, which compare to 756.92 b and 136.67 b in ENDF/B-VI, and 760±17 b and 137±6 b by Mughabghab.[34]

Table 9- Average cross sections of the present evaluation compared to ENDF/B-VI data. The average cross sections were calculated with NJOY constant flux and infinite dilution.

Energy Range eV	Total		Fission		Capture	
	Present	ENDF/B-VI	Present	ENDF/B-VI	Present	ENDF/B-VI
0.001- 0.020	1073.49	1067.81	979.24	971.62	81.98	83.
0.020- 0.050	513.60	512.74	461.63	460.02	39.81	40.21
0.050- 0.400	234.64	234.57	201.85	202.63	208.43	203.59
0.400- 1.000	151.02	147.04	127.69	126.91	11.94	9.95
1.000- 2.100	466.58	456.59	388.97	378.56	66.25	67.38
2.100- 2.750	332.56	322.95	206.67	198.02	111.58	112.00
2.750- 3.000	70.61	69.59	49.84	50.45	7.92	7.48
3.000- 15.00	133.93	130.47	104.26	101.26	17.63	17.65
15.00- 30.00	120.84	117.69	94.76	91.79	13.48	13.27
30.00- 50.00	58.61	57.33	40.73	38.86	5.65	5.46
50.00- 75.00	58.51	58.63	41.24	41.20	5.67	4.61
75.00- 100.0	58.07	50.66	36.89	33.73	8.94	4.35
100.0- 125.0	56.59	46.44	38.24	29.97	6.00	3.88
125.0- 150.0	37.04	38.21	21.09	22.12	3.71	3.54
150.0- 200.0	35.84	37.08	20.97	21.34	3.13	3.18
200.0- 300.0	38.63	35.17	23.02	19.87	3.59	2.72
300.0- 400.0	33.06	31.56	18.24	16.66	2.50	2.33
400.0- 500.0	24.27	27.79	11.01	13.17	1.42	2.08
500.0- 600.0	27.39	27.86	13.47	13.40	2.05	1.90

IV-3- The benchmark testing

An assessment of the ^{233}U resonance evaluation presented in this work regarding the calculation of critical benchmark systems was performed. The evaluations were used to calculate various benchmark systems in the thermal energy region, intermediate energy region , and high energy region. In addition to the calculations using the present ^{233}U evaluation, calculations were also done with cross section data generated with the evaluations available in the ENDF/B and JENDL libraries. The ^{233}U library were obtained by replacing the resonance region evaluation of a previous ^{233}U evaluation done at ORNL[35] with the present ^{233}U resonance evaluation. The library obtained will be referred to as the ORNL library. Benchmark calculations were performed using the ORNL, ENDF and JENDL data libraries and the multiplication factors (k_{eff}) were obtained. The selected energy group structure used in the calculations is the 199-group VITAMIN-B6 cross-section library.[36] The data libraries were processed using the NJOY code system, whereas the benchmark calculations were performed with the Monte Carlo code KENO.Va. A total of 37 benchmark calculations, namely, 16 benchmarks in the thermal region, 11 benchmarks in the intermediate energy region and 10 benchmarks in the fast energy region, were performed. Of the 16 thermal benchmarks, 6 are benchmarks available from the Cross Section Evaluation Working Group (CSEWG) and 10 are the International Criticality Safety Benchmark Experiments Program (ICSBEP). The CSEWG thermal benchmarks, the ORNL series, are unreflected spheres of ^{233}U. A general description of these thermal benchmarks is given in Table 10. The ICSBEP thermal benchmarks, the ^{233}U-SOL-THERM-003 series,[37] are paraffin reflected cylinders of ^{233}U uranyl fluoride solutions. A general description of the ^{233}U-SOL-THERM-003 is given in Table 11. The logbook experiment number of these benchmarks are given in the first column in Table 11. The 11 benchmarks in the intermediate energy region, the ^{233}U-SOL-INTER-001 series also known as *Falstaff*, are from the ICSBEP. They are aqueous solutions of ^{233}U in form of uranyl-fluoride in spheres with reflectors of Be, CH2 and BeCH2 composites. The *Falstaff* were done in the late1950's using 8 types of stainless steel spheres of inner radius varying from 7.87 cm to 12.45 cm. A general description of these intermediate benchmarks is given in Table 12. The first column in Table 12 is the sphere number as it appear in the benchmark validation documents for this system. The 10 fast critical benchmark experiments are the 9 CSEWG ^{233}U-MET-FAST in which the ^{233}U-MET-FAST-001 is known as JEZEBEL-23 and the FLATTOP-23 benchmark. A general description of these fast benchmarks is given in Table 13.

The calculated k_{eff} obtained with the KENO.Va code are shown from Table 14 to Table 17 for the ORNL, ENDF and JENDL libraries. The results of the k_{eff} for the thermal critical benchmark experiments are displayed in Table 14 and Table 15. The results shown in these tables indicate that the calculations using the 3 libraries are compatible, with the ENDF results being a little low for the ORNL thermal benchmarks. The k_{eff} results for the intermediate energy critical experiments, *Falstaff* systems, are shown in Table 16. The normalized neutron flux to one source neutron in unit of neutron per centimeter for the *Falstaff* system is shown in Fig. 10 as a function of energy for the 199-group structure. The upper limit of the ORNL resonance evaluation is 600 eV and above 600 eV the unresolved resonance representation was used. The unresolved resonance region is important to the calculation of the k_{eff} for intermediate energy benchmark as shown in Fig. 10. Work is underway to revise and reevaluate the ^{233}U unresolved resonance region. The resulting k_{eff} shown in Table 16 are very close for the three cross section libraries used, namely, ENDF, ORNL and JENDL data library. The results of the k_{eff} for the fast critical benchmark experiments are displayed in Table 17, which indicates that the k_{eff} calculated using the ORNL evaluation are greatly improved relative to both ENDF and JENDL results.

Table 10- Features of the ^{233}U thermal critical benchmark experiments

Benchmark	H/U atom ratio	wt % ^{233}U	U density (g/cm3)	Geometry	Reflector
ORNL-5	1533	97.70	-	sphere	none
ORNL-6	1470	97.70	-	sphere	none
ORNL-7	1417	97.70	-	sphere	none
ORNL-8	1369	97.70	-	sphere	none
ORNL-9	1324	97.70	-	sphere	none
ORNL-11	1986	97.70	-	sphere	none

Table 11- Features of the ^{233}U thermal critical benchmark experiments ^{233}U-SOL-THERM-003

Logbook Experiment #	H/U atom ratio	wt % ^{233}U	U density (g/cm3)	Geometry	Reflector
40	74.1	98.7	0.332	Cylinder	Paraffin
41	74.1	98.7	0.332	Cylinder	Paraffin
42	74.1	98.7	0.332	Cylinder	Paraffin
55	39.4	98.7	0.600	Cylinder	Paraffin
45	45.9	98.7	0.519	Cylinder	Paraffin
57	154	98.7	0.165	Cylinder	Paraffin
58	250	98.7	0.102	Cylinder	Paraffin
61	329	98.7	0.078	Cylinder	Paraffin
62	396	98.7	0.065	Cylinder	Paraffin
65	775	98.7	0.033	Cylinder	Paraffin

Table 12- Features of the ^{233}U intermediate critical benchmark experiments Falstaff

Sphere No.	H/U atom ratio	wt % ^{233}U	U density (g/cm3)	Geometry	Reflector
1	24.3	98.562	0.866	Sphere	Be
2	24.3	98.562	0.866	Sphere	Be
3	24.3	98.562	0.866	Sphere	Be
3	24.3	98.562	0.866	Sphere	Be + CH_2
4	24.3	98.562	0.866	Sphere	Be
4	24.3	98.562	0.866	Sphere	Be + CH_2
5	24.3	98.562	0.866	Sphere	Be
5	24.3	98.562	0.866	Sphere	CH_2
5	24.3	98.562	0.866	Sphere	Be + CH_2
6	24.3	98.562	0.866	Sphere	Be
6	24.3	98.562	0.866	Sphere	Be + CH_2

Table 13- Features of the ^{233}U fast critical benchmark experiments

Benchmark	H/U atom ratio	wt % ^{233}U	U density (g/cm3)	Geometry	Reflector
^{233}U-MET-FAST-001 (JEZEBEL-23)	-	98.2	18.424	sphere	none
^{233}U-MET-FAST-002-a	-	98.2	18.621	sphere	HEU
^{233}U-MET-FAST-002-b	-	98.2	18.644	sphere	HEU
^{233}U-MET-FAST-003-a	-	98.2	18.621	sphere	NU
^{233}U-MET-FAST-003-b	-	98.2	18.644	sphere	NU
^{233}U-MET-FAST-004-a	-	98.2	18.621	sphere	W
^{233}U-MET-FAST-004-b	-	98.2	18.644	sphere	W
^{233}U-MET-FAST-005-a	-	98.2	18.621	sphere	Be
^{233}U-MET-FAST-005-b	-	98.2	18.644	sphere	Be
FLATTOP-23	-	98.2	18.419	sphere	NU

Table 14- Test of the ^{233}U evaluation with thermal energy benchmarks using KENOV.a code

199-group (VITAMIN/B-6)			
Benchmark	JENDL	ENDF/B-6	ORNL
ORNL-5	0.9988 +/- 0.0008	0.9964 +/- 0.0008	1.0006 +/- 0.0009
ORNL-6	0.9989 +/- 0.0007	0.9962 +/- 0.0009	0.9997 +/- 0.0008
ORNL-7	0.9988 +/- 0.0008	0.9948 +/- 0.0008	0.9996 +/- 0.0008
ORNL-8	0.9990 +/- 0.0008	0.9963 +/- 0.0008	1.0000 +/- 0.0009
ORNL-9	0.9982 +/- 0.0008	0.9950 +/- 0.0008	0.9998 +/- 0.0007
ORNL-11	0.9959 +/- 0.0006	0.9951 +/- 0.0005	0.9987 +/- 0.0006

Table 15- Test of the ^{233}U evaluation with thermal energy benchmarks (^{233}U-SOL-THERM-003) using KENO.V code

199-group (VITAMIN/B-6)			
Benchmark	JENDL	ENDF/B-6	ORNL
1	0.9978 +/- 0.0003	0.9976 +/- 0.0003	0.9970 +/- 0.0003
2	1.0019 +/- 0.0003	1.0073 +/- 0.0003	1.0075 +/- 0.0003
3	0.9931 +/- 0.0003	0.9850 +/- 0.0003	0.9910 +/- 0.0003
4	0.9989 +/- 0.0003	1.0003 +/- 0.0003	0.9970 +/- 0.0003
5	1.0064 +/- 0.0003	1.0017 +/- 0.0003	1.0011 +/- 0.0003
6	1.0196 +/- 0.0003	1.0143+/- 0.0003	1.0184 +/- 0.0003
7	1.0074 +/- 0.0003	1.0085 +/- 0.0003	1.0124 +/- 0.0003
8	1.0121 +/- 0.0003	1.0040 +/- 0.0003	1.0067 +/- 0.0003
9	1.0061 +/- 0.0003	1.0013 +/- 0.0003	1.0074 +/- 0.0003
10	1.0048 +/- 0.0003	1.0006 +/- 0.0003	1.0090 +/- 0.0003

Table 16- Test of the ^{233}U evaluation with intermediate energy benchmarks using KENOV.a code

199-group (VITAMIN/B-6)			
Sphere No.	JENDL	ENDF/B-6	ORNL
1	0.9913 +/- 0.0003	0.9899 +/- 0.0003	0.9908 +/- 0.0003
2	0.9865 +/- 0.0003	0.9855 +/- 0.0003	0.9858 +/- 0.0003
3	0.9879 +/- 0.0003	0.9859 +/- 0.0003	0.9866 +/- 0.0003
3	0.9970 +/- 0.0003	0.9946 +/- 0.0003	0.9963 +/- 0.0003
4	0.9906 +/- 0.0003	0.9892 +/- 0.0003	0.9898 +/- 0.0003
4	0.9896 +/- 0.0003	0.9884 +/- 0.0003	0.9894 +/- 0.0003
5	0.9878 +/- 0.0003	0.9858 +/- 0.0003	0.9874 +/- 0.0003
5	1.0046 +/- 0.0003	1.0019 +/- 0.0003	1.0032 +/- 0.0003
5	0.9847 +/- 0.0003	0.9831 +/- 0.0003	0.9834 +/- 0.0003
6	0.9850 +/- 0.0003	0.9823 +/- 0.0003	0.9842 +/- 0.0003
6	0.9850 +/- 0.0003	0.9833 +/- 0.0003	0.9842 +/- 0.0003

Table 17- Test of the ^{233}U evaluation with fast energy benchmarks using KENO.V code

199-group (VITAMIN/B-6)			
Benchmark	JENDL	ENDF/B-6	ORNL
^{233}U-MET-FAST-001 (JEZEBEL-23)	1.0078 +/- 0.0011	0.9983 +/- 0.0010	0.9974 +/- 0.0010
^{233}U-MET-FAST-002-a	1.0129 +/- 0.0011	0.9931 +/- 0.0008	0.9997 +/- 0.0010
^{233}U-MET-FAST-002-b	1.0066 +/- 0.0010	0.9939 +/- 0.0010	0.9979 +/- 0.0009
^{233}U-MET-FAST-003-a	1.0087 +/- 0.0010	0.9956 +/- 0.0010	0.9979 +/- 0.0011
^{233}U-MET-FAST-003-b	1.0068 +/- 0.0009	0.9946 +/- 0.0010	0.9977 +/- 0.0010
^{233}U-MET-FAST-004-a	1.0075 +/- 0.0009	0.9961 +/- 0.0010	0.9999 +/- 0.0011
^{233}U-MET-FAST-004-b	1.0087 +/- 0.0009	0.9952 +/- 0.0009	0.9985 +/- 0.0009
^{233}U-MET-FAST-005-a	1.0074 +/- 0.0009	1.0022 +/- 0.0010	0.9969 +/- 0.0009
^{233}U-MET-FAST-005-b	1.0169 +/- 0.0010	1.0024 +/- 0.0010	1.0037 +/- 0.0010
FLATTOP-23	1.0162 +/- 0.0009	1.0024 +/- 0.0010	1.0048 +/- 0.0010

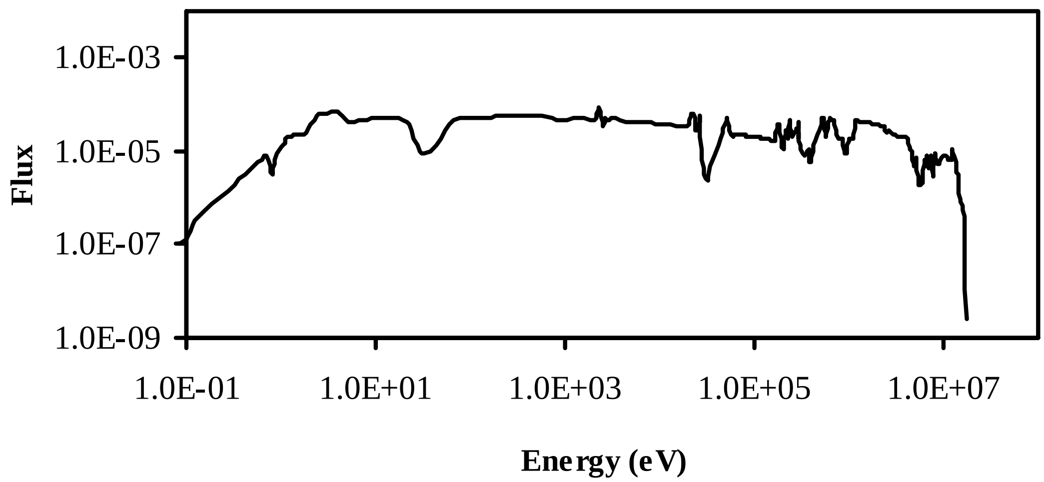

Fig. 10- ***The normalized neutron flux to one source neutron in unit of neutron per centimeter for the Falstaff system.***

A preliminary evaluation was based only on the microscopic neutron transmission and cross section data. In this evaluation the average capture width was 41 meV, and ORELA fission data were reproduced with an accuracy better than 2% in the energy range above 100 eV. However, the total cross section was significantly higher than the value obtained from ORELA transmission data. The values of k_{eff} calculated from the preliminary evaluation were improved compared to the current ENDF/B-VI file, but better results were obtained with JENDL-3 and ORNL-revised JENDL-3.[35] The fission and capture integrals of the preliminary evaluation were also large compared to the recommended BNL value.[34] Comparison of the average cross sections of the different files showed that improvement of k_{eff} could be obtained: (1) for intermediate energy spectra by decreasing the average fission cross section in the resolved resonance range above 100 eV, which means that the fit to the experimental fission cross section should be performed closer to Weston et al. data than to ORELA data in this energy range; (2) for thermal energy spectra by decreasing the capture cross section in the low energy range. In the last step of the evaluation the integral quantities of Table 2 were introduced in the experimental data base, more weight was put on Weston et al. fission data and the average radiative capture width was constrained to a value of 39 meV which is 5% lower than the value obtained in the preliminary evaluation, within an estimated experimental error of 8%. Table 18 displays the values of k_{eff} for the intermediate energy benchmarks obtained by using the preliminary evaluation and the proposed final evaluation. The discrepancies between Weston et al. fission data and the new ORELA fission data in the hundred eV energy ranges is disturbing. As a matter of fact, the ORELA fission cross section is in very good agreement with the ENDF/B-VI evaluation in the energy range above 1.5 keV. The evaluation in the unresolved energy range above 600 eV, which is in progress, will help to understand the discrepancies between Weston et al. data and the new ORELA fission data.

Table 18- Test of the ^{233}U evaluation with intermediate energy benchmarks using KENOV.a code

199-group (VITAMIN/B-6)		
Benchmark	ORNL (Guber Fission)	ORNL (Evaluation)
1	0.9871 +/- 0.0003	0.9908 +/- 0.0003
2	0.9832 +/- 0.0003	0.9858 +/- 0.0003
3	0.9831 +/- 0.0003	0.9866 +/- 0.0003
4	0.9833 +/- 0.0003	0.9963 +/- 0.0003
5	0.9876 +/- 0.0003	0.9898 +/- 0.0003
6	0.9864 +/- 0.0003	0.9894 +/- 0.0003
7	0.9840 +/- 0.0003	0.9874 +/- 0.0003
8	1.0004 +/- 0.0003	1.0032 +/- 0.0003
9	0.9811 +/- 0.0003	0.9834 +/- 0.0003
10	0.9810 +/- 0.0003	0.9842 +/- 0.0003
11	0.9810 +/- 0.0003	0.9842 +/- 0.0003

V- STATISTICAL PROPERTIES OF THE RESONANCE PARAMETERS

V-1- Generalities

The study of the statistical properties of the resonance parameters is useful for testing some aspects of the nuclear reaction theory particularly concerning the Wigner distribution of the resonance spacings, the Porter-Thomas distribution of the reaction widths, the nuclear level density theories and the multiplicity of reaction channels. The average values and the law of distribution of the resonance parameters obtained from the statistical study are also the basis for the evaluation of the cross section data in the unresolved resonance energy range just above the resolved energy range. They are also used in the high energy range for optical model or statistical model calculations. The accuracy achieved on the average values depends on the size and the quality of the resonance sample. The number of resonances used in the present evaluation for the description of the data in the energy range 0 eV to 600 eV is 738. All of the resonances of the sample should be considered as s-wave resonances since the penetrability factor of the p-wave neutrons is very small at 600 eV compared to the s-wave neutrons. The variation of the number of resonances versus energy E identified in the energy range 0 to E is shown in the staircase histogram of Fig 11. Since the level density should be constant over such a small energy range, the variation should correspond to a straight line with a slope equal to the average s-wave resonance spacing. Actually, the slope begins to deviate from its original value at about 70 eV. The observed average spacing is 0.7 eV in the energy range 0 to70 eV and increases to 0.82 eV in the energy range 550 to

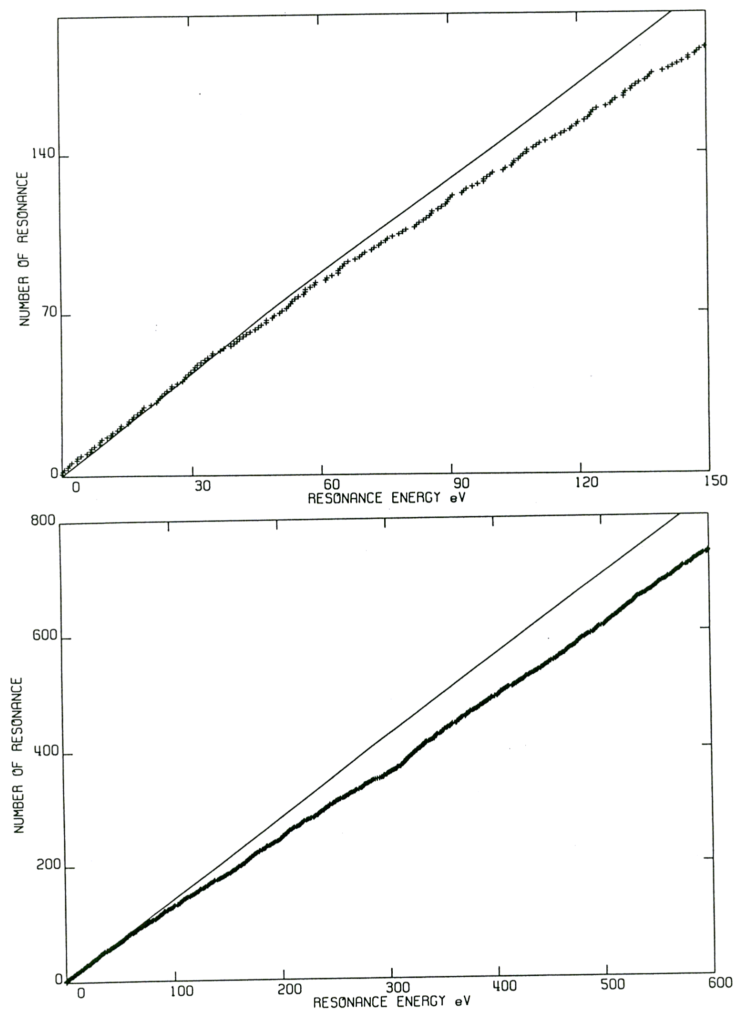

Fig. 11- ***The number of resonances identified in the energy range 0 to E versus energy E.*** The upper part of the figure shows the energy range 0 to 150 eV where most of the maxima in the experimental cross sections correspond to single resonances; the resonance parameters obtained in the SAMMY analysis of the experimental data could have a sound physical meaning. The straight line, which corresponds to a constant level density, shows an increasing loss of resonance above about 50 eV. The lower part of the figure shows the data in the energy range up to 600 eV. The SAMMY analysis above 150 eV was aimed to reproduce the shape of the experimental data with about the same level density as in the 150 eV energy region; the physical meaning of the resonance parameters is ambiguous and the resonances used to fit the experimental data are more pseudo-resonances than the real resonances.

600 eV. In the high energy part of the data the maxima in the experimental cross sections correspond to multiplets of n resonances and n could be as large as 5. The energy of the resonances were determined by means of statistical tests, mainly Δ_3 statistic of the level spacing distribution.[38] The solution is not unique and the statistical properties of those pseudo-resonances could be different from the properties of the resonances at low energy. Moreover, due to the strong overlapping of the resonances, even at low energy the identification of the small resonances is doubtful. Koenig and Michaudon[18] found that about 30% of the resonances (small resonances) were not observed in cross sections calculated with resonance parameters obtained by Monte-Carlo sampling with the expected statistical properties of the ^{233}U resonances. The same result was obtained by Nizamuddin and Blons.[16] In conclusion, the best sample for the study of the statistical properties of the parameters should be restricted to the resonances in the energy range below 70 eV where the number of missing levels is the smallest. This sample contains 95 observed resonances. Nevertheless, some information concerning the behavior of the s-wave strength function and of the total fission widths could be obtained from the entire sample of resonances.

V-2- The level spacing distribution

The distribution of the resonance spacings in the energy range up to 70 eV is given in Fig. 12. The experimental distribution is compared to the sum of two uncorrelated Wigner distributions in the ratio of the populations of the s-wave spin states, normalized to the number of observed spacings. The agreement between the experimental and theoretical distribution is better than expected since the number of missing levels could be large. The results could be compared to those obtained in the evaluation of Ref. 2 where a strong disagreement was observed. It is likely that the accuracy of the resonance energies is better in the present work due to the better experimental resolution of the ORELA transmission data and fission data.

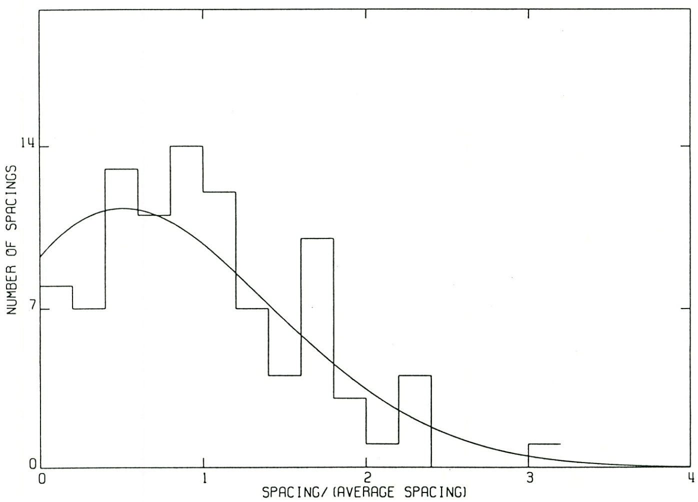

Fig. 12- ***Integral distribution of the level spacings in the energy range 0 to 70eV.*** The histogram is the experimental distribution. The solid line is the sum of two uncorrelated Wigner distributions in the ratio of the 2^+ and 3^+ level populations.

V-3- The s-wave neutron strength function

The variation of the sum of the reduced neutron widths $2g\Gamma_n^0$ versus neutron energy is shown on Fig.13 in the energy range 0 to 600 eV. On the average, the histogram has a linear behavior over the entire energy range. The s-wave neutron strength function, which is the sum of the reduced neutron widths divided by the energy interval, is given by the slope of the histogram. The value is $(0.88\pm0.13)10^{-4}$ in the energy range 0 to 70 eV and $(0.895\pm0.047)10^{-4}$ in the energy range 0 to 600 eV; the error is the sampling error $(2/N)^{½}$, N being the number of resonance in the sample. The pseudo-resonance analysis in the high energy part of the data allows the determination of an accurate value of the strength function, since accurate values of the average total cross section were obtained from the SAMMY fits. The local values of the strength function are given in Table 19. One observes a small value in the energy range 400 to 500 eV which is not statistically compatible with the other local values and with the value over the entire energy range. Statistically incompatible local values of the strength function were also observed for other heavy nuclei, such as ^{239}Pu, ^{240}Pu and ^{238}U, and have not been explained.

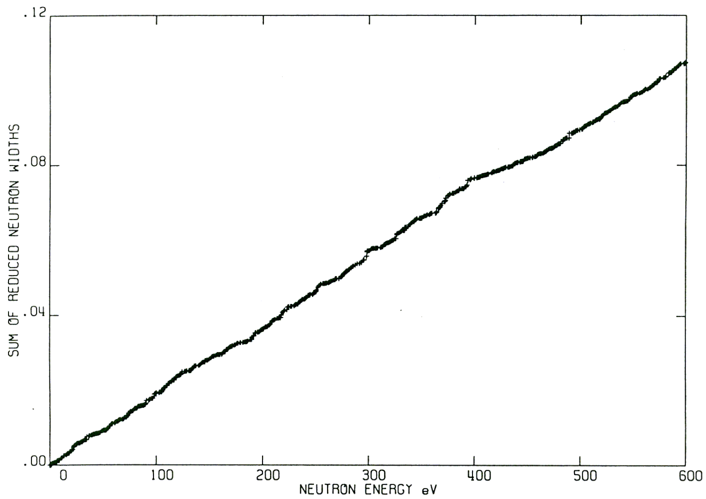

Fig. 13- ***Cumulative sum of the reduced neutron widths of the resonances in the energy range 0 to 600 eV.***

V-4- The neutron width distribution

The integral distribution of the reduced neutron widths of the resonances in the energy interval 0 to 70 eV is shown in Fig. 14. It is evident that many small resonances are missing in the experimental distribution. The missing resonances have reduced neutron width values smaller than about 25% of the observed average value of 0.130 meV. The largest values are roughly fit by a Porter-Thomas distribution normalized to a total number of 135 resonances. The sum of the reduced neutron widths of the missing resonances is about 5% of the sum of the measured values. However, the effect of the missing resonances does not need to be taken into account in the evaluation of the strength function accuracy since the parameters obtained by the SAMMY analysis allows the accurate representation of the shape of the experimental data; the measured neutron widths are representative of the total area of single resonances, multiplets or resonances containing small hidden resonances. The values of the strength functions listed in Table 19 contain little bias due to the effect of the missing resonances.

If one assumes that the number of resonances in the energy range up to 70 eV is 135, the average s-wave level spacing is 0.52 eV instead of the observed value of 0.74 eV, which is only a rough estimation from the neutron width distribution of Fig. 13. The same kind of estimation was also performed by Reynolds and Steiglitz who deduced a value of 0.56 eV from their analysis in the energy range thermal to 60 eV. These results are consistent with the conclusions of Koenig and Michaudon, and of Nizamuddin and Blons who found that about 40% of the resonances are not observed in Monte-Carlo simulated data. The errors on these kinds of estimations could be10-15%, mainly due to the error on the estimation of the number of missing levels.

Table 19- Local values of the s-wave neutron strength function.

Energy range eV	**Srength Function**
	10^{-4}
0- 70	0.880±0.128
0-100	0.949±0.117
100-200	0.862±0.113
200-300	1.048±0.139
300-400	0.981±0.122
400-500	0.648±0.094
500-600	0.909±0.113
0-600	0.895±0.047

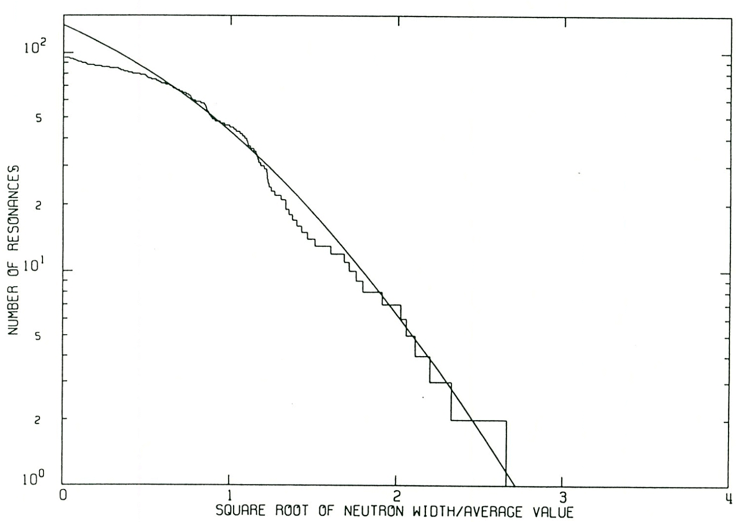

Fig. 14- ***Integral distribution of the reduced neutron widths of the resonances in the energy range 0 to 70 eV.*** The histogram is the distribution of the experimental values. The solid line is the Porter-Thomas distribution normalized to 135 values.

V-5- The fission widths distribution

The integral distribution of the total fission widths in the energy range 0 to 70 eV is given in Fig. 15. In the SAMMY analysis, the spin and parity 2^+ were assigned to 41 resonances and the spin and parity 3^+ to 54 resonances; the corresponding average fission widths are 760 meV and 296 meV, respectively. The experimental distribution of Fig. 14 is compared to a sum of two χ^2 distributions $P(\nu,x)$ with a number of degrees of freedom ν equal to 5 and 4, average fission widths of 760 meV and 296 meV, and normalized

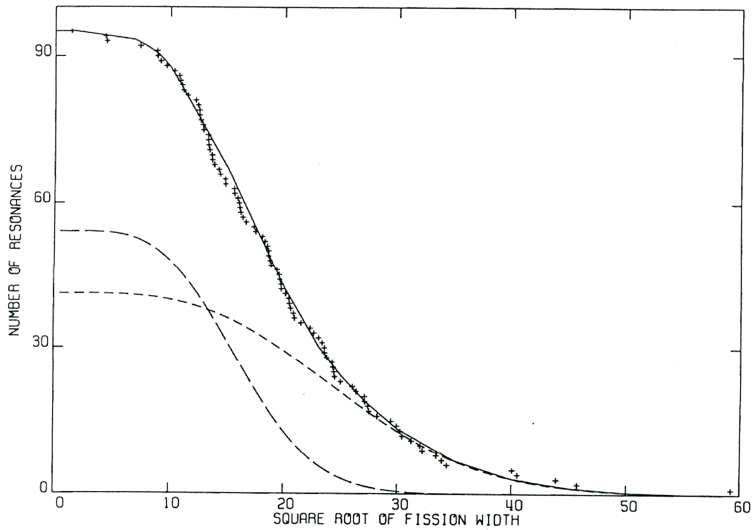

Fig. 15- ***Integral distribution of the total fission widths of the resonances of the energy range 0 to 70 eV.*** The crosses represent the experimental distribution. The dashed lines are two χ^2 distributions normalized to 51 resonances with an average value of 272 meV (long dashed), and to 44 resonances with an average value of 729 meV (short dashed), respectively. The solid line is the sum of the two distributions.

to a number of resonances of 41 and 54, respectively. The agreement between the experimental and the theoretical distribution is excellent. This agreement suggests that the way the SAMMY analysis was carried out had, on average, a sound physical meaning. The number of channels contributing to the fission in the s-wave resonances is quite large, and the 2^+ average fission width is much larger than the 3^+ average fission width, as it was suggested at the beginning of this report. However, the agreement between the experimental and the theoretical distribution could be spurious, since about 30% of the resonances are missing in the experimental sample. But one should also note that in the theory of the nuclear reactions induced by low energy neutron there is no correlation between the neutron channels and the exit channels. In the experimental sample the missing resonances have small neutron widths but not necessarily small fission widths. The missing fission widths should have the same statistical properties as the observed widths. The lack of small fission widths in the experimental distribution is a strong indication of a relatively large number of open fission channels.

Another interpretation of the results is to consider the effective number of fission channels obtained from the ratio of the average fission width to the average level spacing by the relation:

$$N_{eff} = 2\pi \frac{<\Gamma_f>}{<D>}$$

for a given fission channel spin. One finds $N_{eff} = 4.0$ for the 2^+ channels and $N_{eff} = 2.0$ for the 3^+ channels, which is consistent with the experimental distribution of the fission widths if one assumes that some of the fission channels (particularly the 3^+ channels) could be partially open with the consequence that N_{eff} is smaller than the number of degree of freedom of the distributions.

The average total fission widths in energy bins of 100 eV are given in Table 20 for the two groups of resonances used in the SAMMY calculation. Group 1 contains the assigned 2^+ resonances and group 2 the assigned 3^+ resonances. The spin assignment was made randomly. Apart from the interpretation given above of the data in the 0 to 70 eV energy range, the average values in each group have no physical meaning. Note that the average values calculated from all the resonances (mixed) do not deviate too much from the average value of the energy range 0 to 70 eV.

Table 20- Average total fission widths

Energy Range eV	Average Fission Widths meV		
	Group 1	Group 2	Mixed
0 - 70	760.0	296.0	496
0 - 100	839.2	317.5	532
100 - 200	625.3	422.8	509
200 - 300	538.1	442.0	484
300 - 400	744.2	746.5	745
400 - 500	841.7	561.2	673
500 - 600	539.2	401.5	459

V-6- The radiative capture widths

Neutron capture is a process involving a large number of channels which are the primary transitions of the compound nucleus to all the available intermediate states. The distribution of the total capture widths should be a χ^2 distribution with a large number of degrees of freedom, i.e., small fluctuations around the average value. The quite large fluctuations observed generally on the experimental values obtained from the analysis of experimental data including measured capture cross sections are mainly due to the inaccuracy of the experimental cross section data. In the present work the only capture cross section data available were those from Weston et al. The capture width was obtained for 59 resonances from the SAMMY fit of the entire experimental microscopic data base in the low energy region or by trial-and-error method in the higher energy range. The variance of the distribution is 41.3 meV^2 corresponding to a number of degree of freedom of 84. The average value is 41.6 meV with an estimated systematic error of 8% (mainly due to the errors on the resolution and Doppler broadening parameters and to the inaccuracy of the experimental capture cross sections). This value is in agreement with previous results.[2,9] In the SAMMY analysis, the capture width of the other resonances was kept at a constant value of 41 meV which was the prior estimation obtained from Refs. 2 and 9. As mentioned above, in the last step of the evaluation (which consisted of introducing the integral data and the results of benchmark testings) the average value of the capture widths decreased by about 5%, within the error bar obtained from the analysis of the microscopic data.

V-7- The recommended values of the average s-wave resonance parameters

Examination of the properties of the observed resonance parameters performed above has shown that even in the low energy part of the data the resonance sample is far from complete. From the distribution of the reduced neutron widths it is clear that most of the resonances with reduced neutron width smaller than 25% of the observed average value are missing. This is probably because they are hidden by the overlap of the observed wide resonances or misinterpreted as interference effects of the two fission channel Reich-Moore formalism used in the SAMMY analysis of the experimental data. Therefore, the interpretation of the statistical properties of the observed parameters is difficult. The shape of the distribution of the reduced neutron widths suggests that 30% of the resonances are missing and the corrected average level spacing should be (0.52 ± 0.08) eV, corresponding to (1.19 ± 0.12) eV and (0.92 ± 0.10) eV for the 2^+ and 3^+ level, respectively (with a value of 6 for the spin cut-off parameter). Assuming that there is no correlation between the neutron widths and the fission widths, the missing fission width values should have the same average properties as the observed values. These properties could be obtained by fitting the experimental distribution of the observed fission widths. The recommended average values of the s-wave resonance parameters are given in Table 21. The values are similar to those given in Ref. 2. Better accuracy is obtained on the strength function because the evaluation was done on a wider energy range.

Table 21- Recommended average values of the s-wave resonance parameters

J	Average Spacing eV	Strength Function 10^4	Fission Width meV	Neff	Capture Width meV
mixed	0.52 ± 0.08	0.895 ± 0.047	496		39.0 ± 3.0
2+	1.19 ± 0.12		760 ± 60	4.0	
3+	0.92 ± 0.10		296 ± 30	2.0	

VI- CONCLUSION

The Reich-Moore parameters of the resonances induced by the s-wave neutron on the ^{233}U nucleus in the incident neutron energy range thermal to 600 eV were obtained from the SAMMY analysis of a large experimental data base. The extension to a neutron energy range much larger than in previous similar analyses was possible because of the recent time-of-flight high resolution neutron transmission and fission cross section measurements performed at ORELA. Extending the resolved resonance range to higher energies could be more efficient than the statistical methods using the average parameters in the unresolved range for the calculation of the self-shielding factors. In fact, the resonances used in the upper region of the analysis are pseudo-resonances with properties different from those of the low energy region. Concerns have been raised as to whether the pseudo-resonance representation is adequate for the calculation of accurate self-shielding factors or for other purposes. This question was recently assessed by Guimareas et al.[39] who showed that the pseudo-resonance representation in the keV range for ^{235}U was adequate for accurate calculation of self-shielding parameters; the results could be extended to any of the fissile nuclei. Integral data and results of benchmark calculations were used in the last step of the evaluation allowing an additional adjustment of the average capture width. The ability of the calculated cross sections to reproduce the effective multiplication factors k_{eff} for various thermal, intermediate and fast systems was tested. The performances of the new evaluation are improved compared to the current ENDF/B-VI evaluation. The discrepancy between the ORELA average fission cross section and the data of Weston et al. and Blons in the energy range above 100 eV was not understood. The results of the present evaluation were kept close to the values of Weston et al. in order to obtain better agreement with the multiplication factor k_{eff} of the intermediate energy systems.

Transmission measurements of a thicker sample of ^{233}U are in progress at ORELA, in order to obtain accurate values of the total cross section in the unresolved resonance region. The set of average resonance parameters obtained in the present work will be used as prior data for a SAMMY analysis of the new total cross section and for a reevaluation of the partial cross sections in the unresolved range. It is also expected that the better accuracy achieved on the total cross section could help in understanding the discrepency among the experimental fission cross sections.

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APPENDIX. RESONANCE PARAMETERS FOR ^{233}U IN THE ENDF FORMAT

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 9.22330+ 4 2.31038+ 2          0          0          1          09222 2151    1
 9.22330+ 4 1.00000+ 0          0          1          2          09222 2151    2
 1.00000- 5 6.00000+ 2          1          3          0          09222 2151    3
 2.50000+ 0 9.62000- 1          0          0          1          19222 2151    4
 2.31038+ 2 0.00000+ 0          0          0       4620       7709222 2151    5
-1060.00000      3.000 1.524000+0 4.000000-2 2.600000-1 2.370000-19222 2151    6
-800.000000      2.000 1.315000+0 4.000000-2 3.120000-1-2.250000-19222 2151    7
-575.000000      3.000 8.079000-1 4.000000-2-2.550000-1 2.400000-19222 2151    8
-387.500000      2.000 5.888000-1 4.000000-2 2.825000-1-2.420000-19222 2151    9
-237.500000      3.000 3.885000-1 4.000000-2-2.850000-1-2.300000-19222 2151   10
-137.500000      2.000 1.954000-1 4.000000-2 3.120000-1 2.150000-19222 2151   11
-77.5000000      3.000 1.106000-1 4.000000-2-3.340000-1 2.480000-19222 2151   12
-40.0000000      2.000 4.306000-2 4.000000-2-2.530000-1-3.520000-19222 2151   13
-17.9130000      3.000 3.031000-2 4.000000-2 3.400000-1-2.040000-19222 2151   14
-7.79660000      2.000 3.010000-4 4.000000-2-1.228000-1 5.123000-29222 2151   15
-4.53180000      2.000 1.766000-5 4.000000-2-2.252000+0 1.625000-39222 2151   16
-3.32780000      2.000 2.653000-5 4.471000-2 1.755000+0 1.328000-39222 2151   17
-3.05440000      3.000 9.165000-4 6.374000-2-4.607000-1 1.818000+09222 2151   18
-2.45840000      3.000 1.907000-4 8.530000-2 1.065000-2 5.637000-59222 2151   19
-1.75650000      3.000 1.867000-4 3.763000-2 2.166000+0 8.589000-59222 2151   20
-0.99317000      2.000 1.281000-7 2.850000-2 4.316000-2-1.421000+09222 2151   21
0.165977000      3.000 1.001000-7 8.298000-2 1.881000-2-2.639000-49222 2151   22
0.232443000      2.000 3.177000-8 4.848000-2 1.397000-3 6.553000-49222 2151   23
0.576819000      2.000 6.288000-7 2.519000-2 3.424000-1 3.388000-29222 2151   24
1.451663000      2.000 2.007000-4 3.830000-2 3.934000-4-5.557000-19222 2151   25
1.767321000      3.000 2.579000-4 3.999000-2-1.455000-1 5.685000-29222 2151   26
2.303152000      3.000 1.525000-4 3.743000-2 5.437000-2 6.901000-69222 2151   27
3.509805000      2.000 1.813000-4 3.896000-2 5.875000-1 4.431000-49222 2151   28
3.633436000      3.000 5.447000-5 3.589000-2 8.323000-2 2.594000-59222 2151   29
4.457119000      2.000 3.787000-4 3.449000-2-7.006000-5-1.149000+09222 2151   30
5.805104000      3.000 1.004000-4 3.900000-2 3.519000-1-1.855000-49222 2151   31
6.638419000      2.000 3.430000-4 3.900000-2 8.468000-6 5.560000-19222 2151   32
6.825227000      3.000 6.894000-4 3.488000-2-1.051000-1-1.008000-29222 2151   33
7.475214000      3.000 3.342000-5 3.900000-2 4.163000-2 1.444000-19222 2151   34
8.702149000      2.000 1.897000-5 3.900000-2 1.788000-1 7.635000-49222 2151   35
8.771688000      2.000 3.293000-4 3.900000-2-4.463000-3-1.601000+09222 2151   36
9.171721000      3.000 6.771000-5 5.023000-2 2.078000-1 1.155000-29222 2151   37
10.37642000      3.000 1.413000-3 4.711000-2-2.568000-1 1.657000-59222 2151   38
11.28210300      3.000 2.486000-4 3.900000-2-4.186000-2-4.206000-19222 2151   39
11.69787900      2.000 9.985000-5 3.900000-2-5.314000-1-3.455000-29222 2151   40
12.78735900      3.000 1.329000-3 4.011000-2 4.803000-2-2.524000-19222 2151   41
13.44897700      3.000 1.259000-4 3.900000-2 2.523000-1 5.118000-49222 2151   42
13.67799900      2.000 3.243000-4 3.408000-2-2.134000-1 5.421000-29222 2151   43
15.32828200      3.000 6.386000-4 4.158000-2-4.198000-4-1.534000-19222 2151   44
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90. C. Mounier, CEN Saclay, DMT/SERMA/LENR, 91191 Gif Sur Yvette Cedex, France
91. M. C. Moxon, 3 Hyde Copse, Marcham, Abingdon, Oxfordshire, England
92. S. F. Mughabghab, Brookhaven National Laboratory, Advanced Technology Building 197d, Upton, NY 11973-5000
93. D. Muir, IAEA Nuclear Data Section, Wagramerstr. 5, P. 0. Box 100, A-1400, Wien, Austria
94. C. Nordborg, OECDNEA, Le Seine St-Germain 12, Boulevard Iles, 92130, Issy-les-Moulineaux, France

95. A. Nouri, OECD/NEA Data Bank, Le Seine Saint Germain 12 Bd des Iles, 92130 Issy-les-Moulineaux, France
96. S. Y. OH, Nuclear Data Evaluation Lab. Korea Atomic Energy Research Institute, P. O. Box 105, Yusung Taejon, 305-600 Korea
97. A. Popov, Frank Laboratory of Neutron Physics, Joint Institute for Nuclear Research, RU-141980 Dubna, Moscow Region, Russia
98. C. Raepsaet, CEN Saclay, DMT/SERMA/LEPP, 91191, Gif Sur Yvette Cedex, France
99. M. Salvatores, DRN/P, Batiment 707, C. E. CADARACHE, 13108, Saint Paul-lez-Durance, France
100. E. Sartori, OECDNEA, Le Seine St-Germain 12, Boulevard Iles, 92130, Issy-les-Moulineaux, France
101. O. A. Shcherbakov, Petersburg Nuclear Physics Institute, 18 8 3 5 0 Gatchina, Leningrad District, Russia
102. R. Shelley, Central Bureau for Nuclear Measurements, Steenweg op Retie, 2240 Geel, Belgium
103. K. Shibata, Nuclear Data Center, Japan Atomic Energy Research Institute, Tokai-mura Naka-gun, Ibaraki-ken 319-11, Japan
104. R. Slovacek, Gaerttner LINAC Laboratory, Rensselaer Polytechnic Institute, Department of Environmental and Energy Engineering, Troy, NY 12180-3590
105. A. B. Smith, Argonne National Laboratory, TD 362 D216, Argonne, IL 60544
106. D. L. Smith, Argonne National Laboratory, TD-360-L106, Argonne, IL 60544
107. H. Takano, Nuclear Data Center, Japan Atomic Energy Research Institute, Tokai-mura, Naka-gun, Ibaraki-ken 319-11, Japan
108. C. Wagemans, Central Bureau for Nuclear Measurements, Steenweg op Retie, 2240 Geel, Belgium
109. J. J. Wagschal, Racah Institute of Physics, The Hebrew University of Jerusalem, 91904 Jerusalem, Israel
110. H. Weigmann, Central Bureau for Nuclear Measurements, Steenweg op Retie, 2240 Geel, Belgium
111. J. P. Weinman, Lockheed Martin Corporation, P.O. Box 1072, Schenectady, NY 12301-1072
112. C. Werner, Rensselaer Polytechnic Institute, Residence Hall, 1999 Burdette Ave, Troy, NY 12180-3711
113. R. White, Lawrence Livermore National Laboratory, P. O. Box 808, Livermore, CA 94550
114. M. Williams, Louisiana State University, Nuclear Science Center, Baton Rouge, LA 70803
115. K. Yoo, Korea Atomic Energy Research Inst., Nuclear Data Evaluation Lab., P. O. Box 105, Yusung, Taejon 305-600, Korea
116. Phillip G. Young, Los Alamos National Laboratory, MS B243 T-16, Los Alamos, NM 87545