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A NEW RESONANCE-REGION EVALUATION OF NEUTRON CROSS SECTIONS FOR U-235

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This paper describes a multilevel resonance analysis of the ^{235}U neutron cross sections in the resolved resonance energy region.[1]

The ENDF/B-V representation of the ^{235}U neutron cross sections in the resolved resonance region is unsatisfactory: below 1 eV the cross sections are given by "smooth files" (ENDF file 3) rather than being described by resonance parameters. Above 1 eV the single-level formalism used in ENDF/B-V is inadequate to represent the level-level interferences in the fission channels, and must be supplemented by a file 3 contribution consisting of more than 1300 energy points; information on level-spins has not been included in the file, and the resolved resonance region extends only up to 81 eV. Indeed, the ENDF/B-V ^{235}U resonance parameters are based on an analysis done in 1970 for ENDF/B-III,[2] which of course could not consider the results of several high quality measurements done in the past 18 years.

In 1981 an IAEA consultant group on the uranium and plutonium resonance parameters recommended a new evaluation of the ^{235}U resonance parameters.[3]

The resonance analysis code SAMMY[4] was used to perform an R-matrix multilevel analysis of the ^{235}U neutron cross sections. SAMMY allows for the Reich-Moore formalism which leads to a physically sound representation of the neutron cross sections. It uses the Bayesian approach which allows the successive incorporation of new data in a consistent manner. The option to search not only for resonance parameters but also for experimental parameters such as sample thickness, sample effective temperature, backgrounds, normalization, and the parameter of the instrumental resolution, all consistent with predetermined uncertainty limits, lead to realistic parameter uncertainties and covariance matrices. SAMMY is now running on a fast Floating Point System Computer and is

continually updated to increase the speed of data fitting and to adapt to new requirements of the evaluators. Recent modifications now allow a consistent fit of the spin-separated fission data.

Most of the ^{235}U neutron cross section data collected in the CSISRS library[5] were examined but a few recent measurements listed in Table 1 were given most of the weight.

The data sets listed in Table 1 were selected either because they provide unique information, or because they originate from recent high quality measurements about which much information on experimental conditions and possible sources of error were available to the authors.

The fission cross-section data were renormalized to yield a value 246.497 ± 1.035 b·eV for the fission integral from 7.8 to 11.0 eV. The capture data were renormalized to correspond to a value 98.9649 ± 0.7922 b at 0.0253 eV. These are values proposed by the ENDF/B-VI standards evaluation group.[6] Fission and capture data which did not include the normalization range were renormalized in the region of overlap with other sets.

The observed level spacing in ^{235}U is of the order of 0.4 eV so that above 100 eV the Doppler and resolution broadened width of the resonances is comparable with the level spacing and levels cannot be well resolved. The availability of measurements done at liquid nitrogen temperature and of spin separated data is of great help in extending the resolved region to higher energies, nevertheless above about 110 eV a large fraction of the levels is probably undetected, nevertheless in that partly resolved region it is possible to describe the cross sections with reasonable accuracy, with resonance parameters. As new and better data become available, such a representation can be improved and made consistent with the new information. Even where the resonances cannot be resolved the parameterization of the available data provides a more accurate representation of the cross sections than does a statistical treatment such as that used by ENDF/B-V in the unresolved region.

Because the ^{235}U level spacing is of the order of 0.4 eV, about 500 levels are needed to describe an interval of 200 eV. Because dealing with large numbers of levels is awkward both for data analysis and data processing, it was decided to divide the range from 0 to 500 eV into three regions: 0 to 110 eV, 110 to 300 eV, and 300 to 500 eV; in each region the cross sections are described with a different set of resonance parameters. Below 110 eV the resonances are mostly resolved and the resonance analysis is considered physically meaningful. Above 110 eV the resonances become more and more unresolved and the analysis must be considered a parameterization which describes the available data fairly well but which should not be used to infer average properties of the resonances.

Examples of results of the analysis are shown in Figs. 1 to 3. In Fig. 1 the computed and measured transmission of neutrons through two samples of ^{235}U are compared in the range 60 to 100 eV. Figure 2 shows a comparison of computed and measured fission and capture cross sections, in the same energy range. Finally, Fig. 3 shows the total and fission cross sections in the energy range from 450 to 500 eV. The measured total cross section was obtained from the transmission data of Harvey et al. In Table 2 the fission and capture resonance integrals over selected energy region, obtained from our resonance parameters, are compared with the values obtained from ENDF/B-V.

The analysis described here improves on ENDF/B-V in several respects:

1. It utilizes a physically meaningful formalism to represent available data as well or better than ENDF/B-V, without introducing artificial file 3 contributions.

2. It provides spin assignments for each level and is consistent with the spin separated fission data.[7,8]

3. It extends the resolved resonance region down to 10^{-5} eV and up to 110 eV; from 110 to 500 eV resonance parameters also are provided that represent well the measured data.

4. The cross sections in the thermal region are consistent with the values currently proposed for ENDF/B-VI by the standard evaluation committee.

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Table 1. Selected measurements of the ^{235}U neutron cross sections

Reference	Energy range used in the analysis	Measurement characteristics
Gwin et al.[9] (1984)	0.01 to 30 eV	Fission at 22 m
Weston and Todd[10] (1984)	14 to 500 eV	Fission at 18.9 m
Blons[11] (1973)	17 to 500 eV	Fission at 50 m; sample cooled to 77K
de Saussure et al.[12] (1967)	0.4 to 100 eV	Fission and capture at 25.5 m
Perez et al.[13] (1973)	8 to 100 eV	Fission and capture at 40 m
Spencer et al.[14] (1984)	0 to 1 eV	Transmission at 18 m. Samples of 0.001468 atom/b
Harvey et al.[15] (1986)	0.4 to 68 eV	Transmission at 18 m. Samples to 97K. Samples of 0.002343, 0.032808 atom/b
Harvey et al.[15] (1986)	4 to 500 eV	Transmission at 80 m. Samples cooled at 97K. Sample of 0.002343 atom/b
Moore et al.[7,8] (1978)	1.6 to 500 eV	Separated spin states fission data

Table 2. Comparison of resonance integrals with ENDF/B-V[a]

Energy Interval (eV)	Fission Resonance Integral (b)		Capture Resonance Integral (b)	
	Present Evaluation	ENDF/B-V	Present Evaluation	ENDF/B-V
0.5 - 1.0	42.76	43.78	5.20	5.24
1.0 - 10.0	75.71	77.75	43.40	43.39
10.0 - 100.0	95.88	99.53	65.30	65.28
100.0 - 200.0	14.33	14.35	8.05	8.08
200.0 - 300.0	8.20	8.22	3.80	3.70
300.0 - 400.0	3.74	3.73	1.571	1.88
400.0 - 500.0	3.01	3.00	0.712	1.06

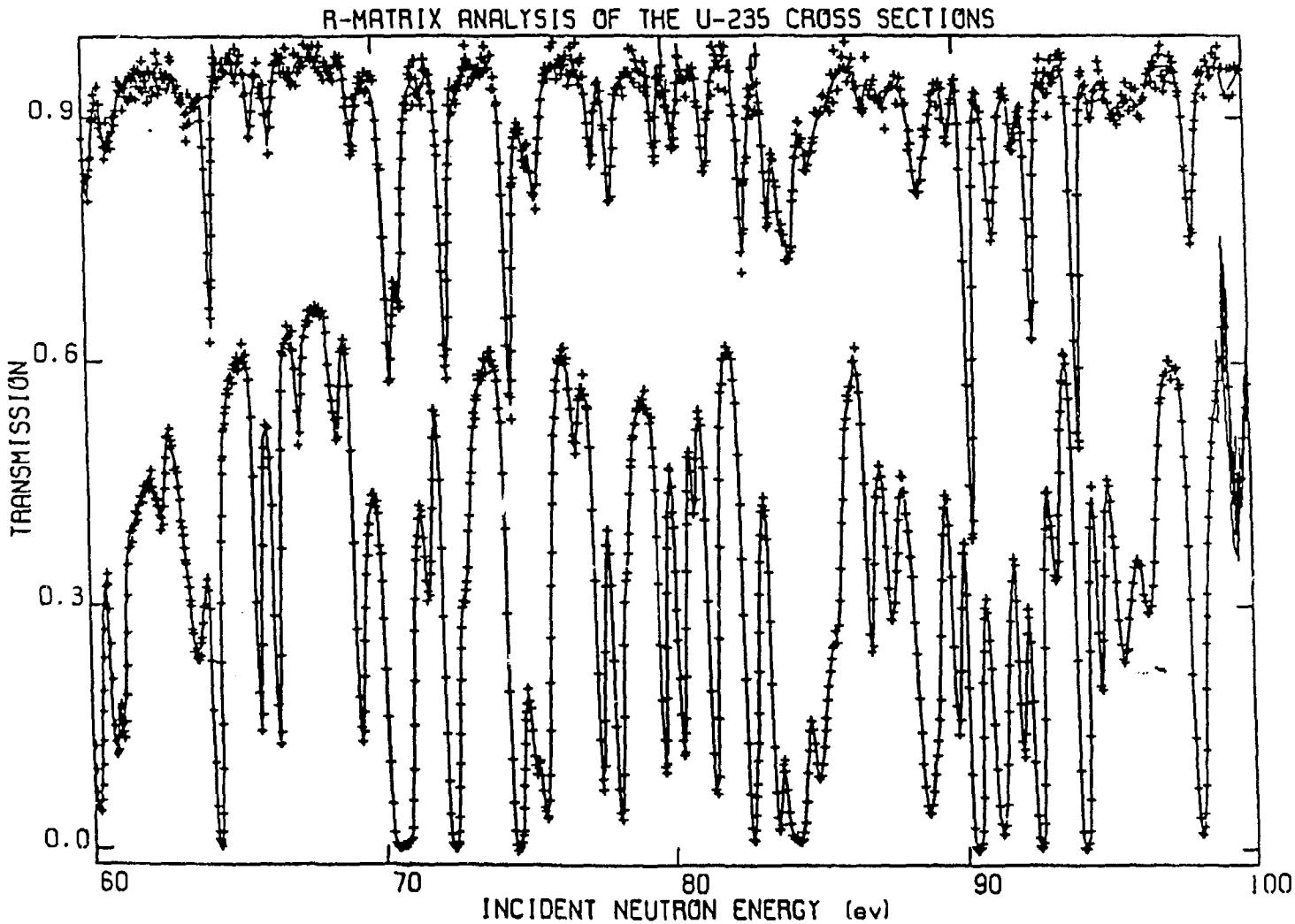
[a]These figures may change in the final evaluation.

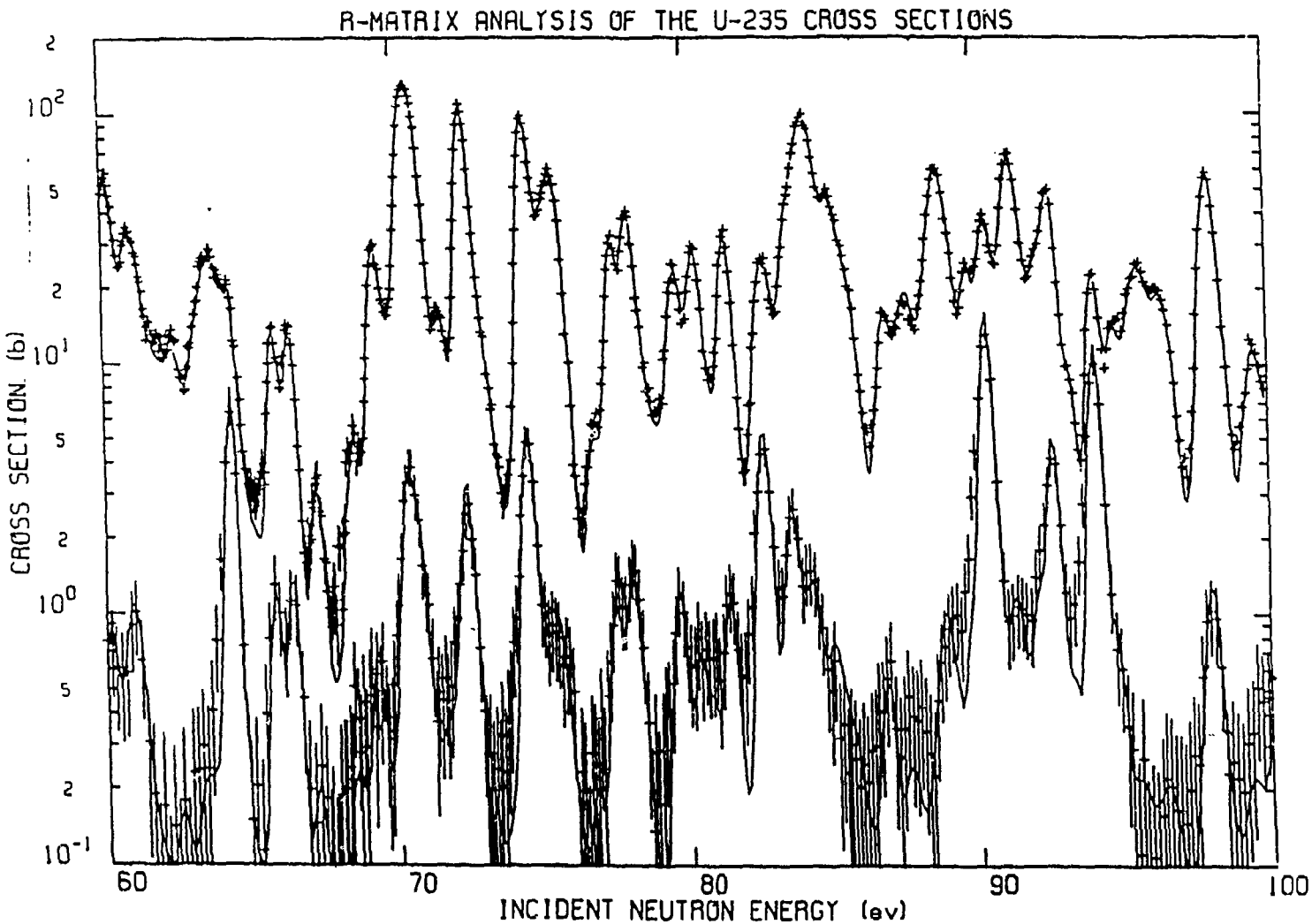
FIGURE CAPTIONS

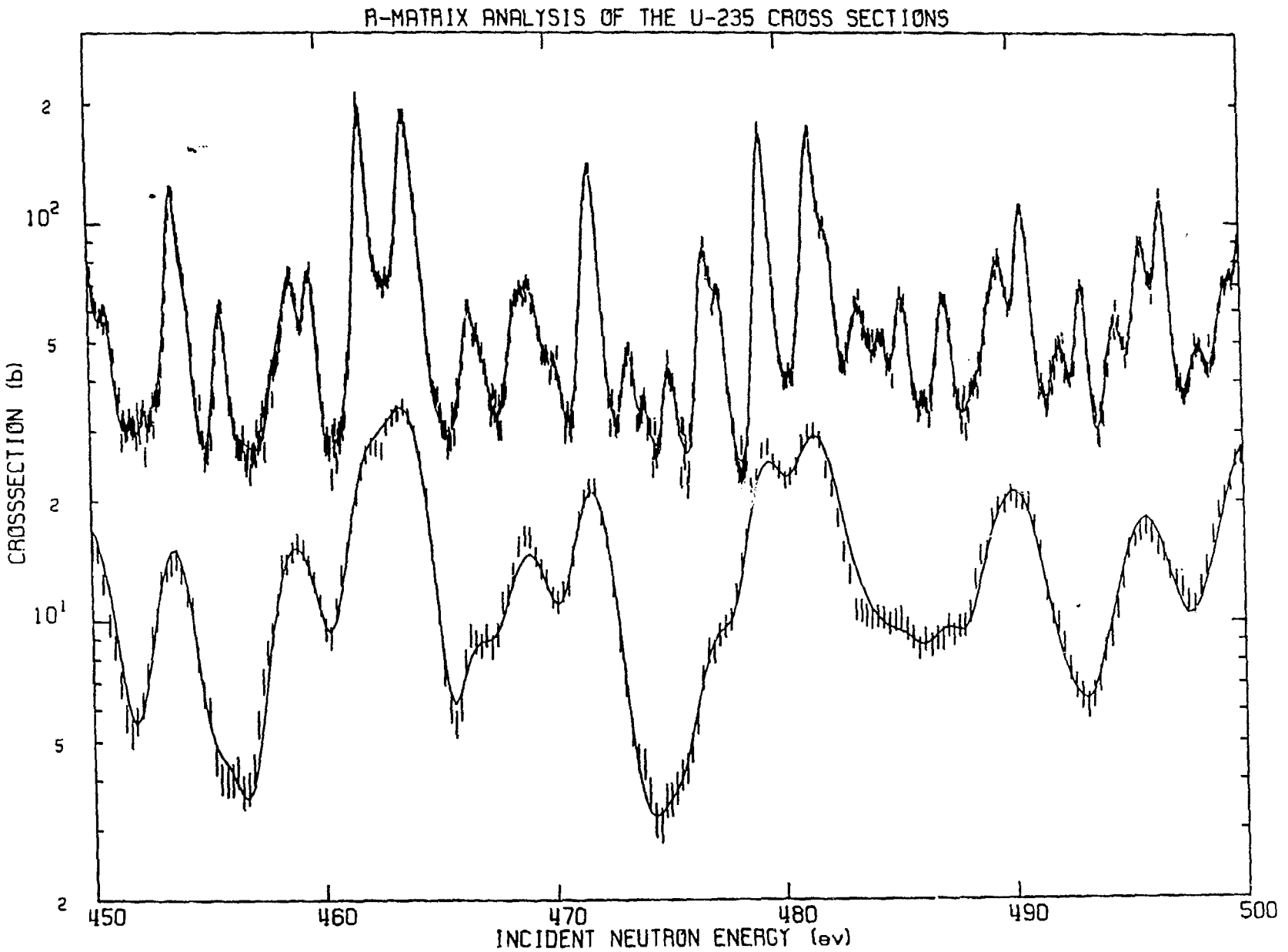
Figure 1. Transmission through samples of ^{235}U of 0.002335 at/b (upper curve) and 0.03269 at/b (lower curve) between 60 and 100 eV. Solid lines were computed with resonance parameters of our evaluation. Data from Harvey et al.[15]

Figure 2. Fission and capture cross sections of ^{235}U between 60 and 100 eV. The solid lines wee computed with resonance parametes of our evaluation. Fission data from Weston and Todd;[10] capture data from de Saussure et al.[12] The capture cross sections are displaced by one decade for clarity.

Figure 3. Total and fission cross sections of ^{235}U between 450 and 500 eV. The solid lines were computed with resonance parameters from our evaluation. Fission data from Weston and Todd;[10] total cross section data derived from the transmission measurement of Harvey et al.[15] The total cross sections are multiplied by two for clarity of display.


R-MATRIX ANALYSIS OF THE U-235 CROSS SECTIONS
TRANSMISSION
0.9
0.6
0.3
0.0
60
70
80
90
100
INCIDENT NEUTRON ENERGY (ev)


R-MATRIX ANALYSIS OF THE U-235 CROSS SECTIONS
CROSS SECTION (b)
2
10^2
5
2
10^1
5
2
10^0
5
2
10^{-1}
60
70
80
90
100
INCIDENT NEUTRON ENERGY (ev)



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