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COMPARISON OF EVALUATIONS FOR ^{235}U, ^{239}Pu, ^{240}Pu, ^{241}Pu AND ^{242}Pu WITH INTEGRAL MEASUREMENTS*

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* Work performed with the support of the International Atomic Energy Agency, Vienna, under research contract 3691/R3/RB.

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Comparison of Evaluations for ^{235}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu With Integral Measurements

Andrej Trkov

August 31, 1988

Abstract

The evaluations for ^{235}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu are considered. Intercomparison is made of the neutron cross section data from INDL/A, ENDL-84, ENDF/B-5 and ENDF/B-6 (where applicable). Integral measurements of the spectrum averaged cross sections are compared to the values derived from evaluated data libraries.

Note: The work presented herein expresses the views of the author, primarily as a user of evaluated data files. As such it is intended to point out the difficulties encountered in processing the data with the final aim to contribute to the further improvement of evaluated data and their reliability.

1 Introduction

Construction of a multigroup constants library for some particular application is a long and tedious task of processing evaluated data libraries and it usually ends in cross section adjustment to be able to reproduce in calculations some observable quantities from integral measurements.

In the last few years a number of new accurate measurements have been made and several evaluations performed. The most widely cited (but *restricted*) is the ENDF/B-5 library [1]. The European - Japanese JEF library is also not generally available. A user without access to the thoroughly tested and well documented libraries has to make his own choice from the evaluations available. One of the selection criteria can be the date of release of the library but it does not guarantee that the evaluation for the material of interest is very recent. Furthermore even recent evaluations do not always consider all the most recent data for all reaction types.

In the past few years an effort has been initiated by the International Atomic Energy Agency (IAEA) to collect some recent evaluations and perform checking which would ease the problem of selecting a suitable data set. The INDL/A evaluated data library of the Actinides [2] is the result of such effort and the present work is aimed at testing the newly introduced Soviet evaluations for ^{235}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu [3,4] which resently became available, in comparison with the ENDL-84 [5] and the available sections of the ENDF/B-5 [6,7,8] and ENDF/B-6 [9] libraries.

The scope of analysis in this work is limited to the spectrum averaged cross sections, particularly the ^{252}Cf spontaneous fission spectrum, the thermal neutron induced ^{235}U fission spectrum, the resonance integrals and the $2200ms^{-1}$ cross sections. Of special interest is the integral measurement data availability, accuracy and comparison to the values derived from evaluated libraries.

An analysis complementary to the one presented in this work was published by R.P.Corcuera et.al. [10]. It includes the JENDL-2 and the ENDF/B-5 Actinides libraries, but not the ENDF/B-5 Dosimetry and Standards libraries which contain some data on the major actinides.

2 Californium-252 Spontaneous Fission Spectrum

Until recently the commonly adopted Californium-252 spontaneous fission spectrum was the one which was recommended by BNL and tabulated in the $IRDF-85$ file [12]. Since then a new evaluation has been completed by W.Mannhart and documented in the IAEA report [13] and references therein. The two spectra can be compared on Figure 1.

In the original documentation the IAEA spectrum is tabulated only above 1 keV. The data were extrapolated to lower energies according to the formula given by Mannhart on page 163 of ref.[11]

$$\chi(E) = A\sqrt{E}\exp\left(\frac{-E}{1.42}\right) \tag{1}$$

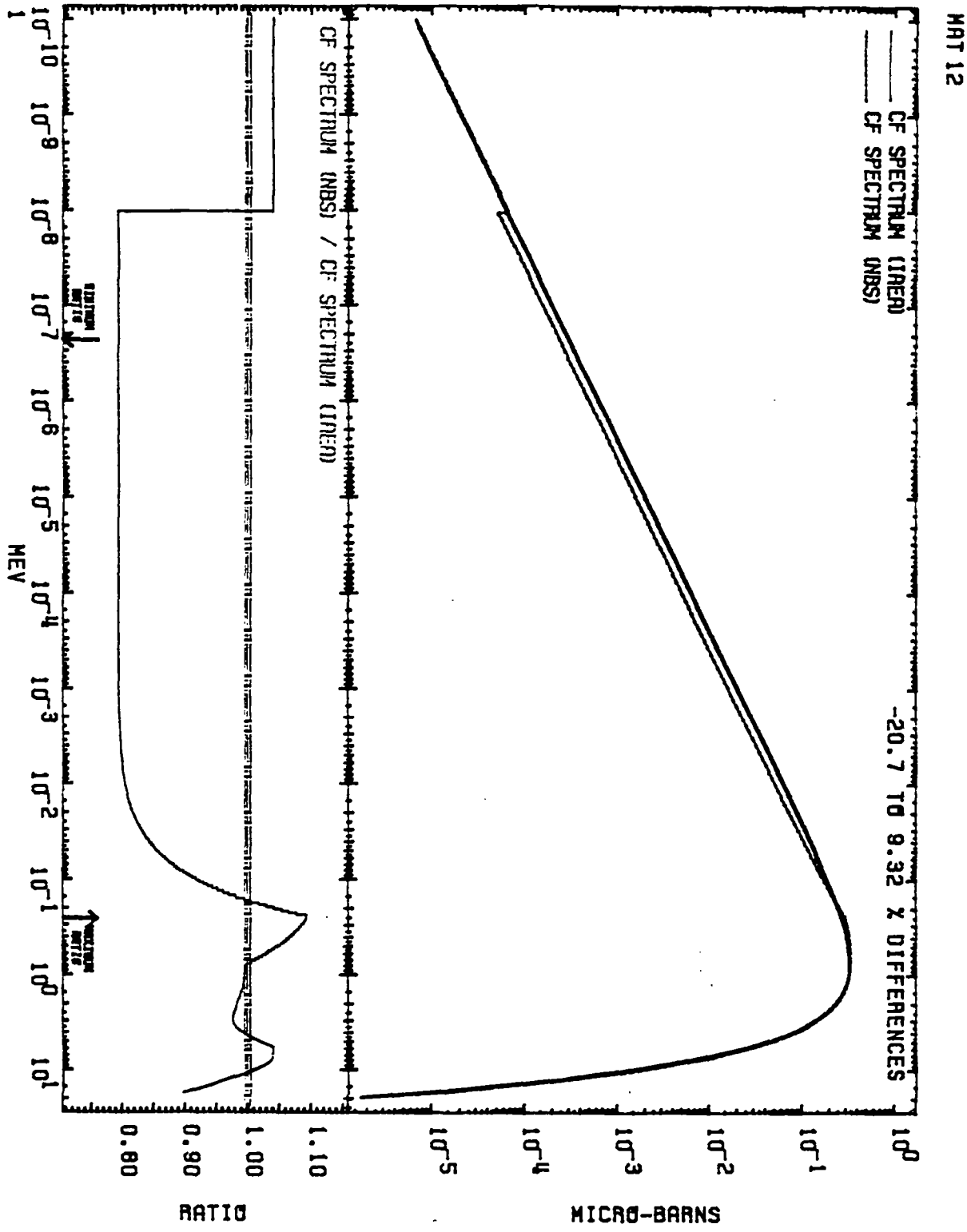

Figure 1: Comparison of the NBS and the IAEA ^{252}Cf spontaneous fisson spectrum

The constant $A = 0.642078$ was chosen to achieve continuity. The so obtained spectrum was then renormalized such that

$$\int_0^{20MeV} \chi(E)\, dE = 1 \tag{2}$$

The NBS spectrum is given only up to 18 MeV. The fraction of the neutrons born above 18 MeV is of the order 10^{-5} and it does not affect the spectrum averaged cross sections which are of interest for integral measurements. It is the different shapes of the spectra which cause the differences in the calculated spectrum averaged cross sections. Such differences are observed in the threshold reactions (for example in the $(n, 2n)$ reaction typically about 3%) while the effect on the fission and the capture cross section is normally small (much less than 1%). The derived spectrum averaged cross sections from the two spectra for the ENDL-84 library are compared in Appendix F.

3 Analysis of the ^{235}U data

3.1 Intercomparison of Evaluated Data

The emphasis is placed on the newly available evaluations by Konshin et.al. [2,3,4] which are contained in the Supplement 86/5 to the INDL/A-83 evaluated data file (in the following text referred to as the INDL/A evaluations). This file was processed with the ENDF pre-processing codes LINEAR-87/1, RECENT-87/1, GROUPIE-86/2 [14].

In the resonance region some negative cross sections were encountered (see Figure 2). They occur because during the evaluation process only a limited number of resonances were considered on each side of the resonance being processed [15]. The ENDF procedures require that at each point the contribution of all resonances must be included. In the documentation [2] the appearance of negative cross sections is not evident (compare Figure 2 with the one in the reference), what suggests that the pointwise data were reconstructed using a code tuned to the data and not strictly following the ENDF rules. An effort to remove the inconsistent representation of the evaluated data in the resonance region is currently under way [15].

A processing message was issued that the competitive widths are not given but that the sum of the partial widths does not sum up to the total in the resonances at 11.666 and 48.729eV. The larger difference is 0.05%.

Comparing evaluated data in pointwise representation would obscure rather than clarify the differences due an excessive amount of information in the resonance region (see Figure 2). For this reason it was decided to compare the flat spectrum averaged data in the 640 extended $SAND - II$ group structure.

The differences the cross sections between the INDL/A and the ENDF/B-5 Standards Mod.2 evaluations can be seen in appendix A on Figures 5 - 11. Comparison does not include inelastic scattering for discrete levels. The differences between the INDL/A and the ENDL-84 evaluations can be observed on Figures 12 - 17. As expected, discrepancies are found in the resonance region, but apart from that attention is brought to the following:

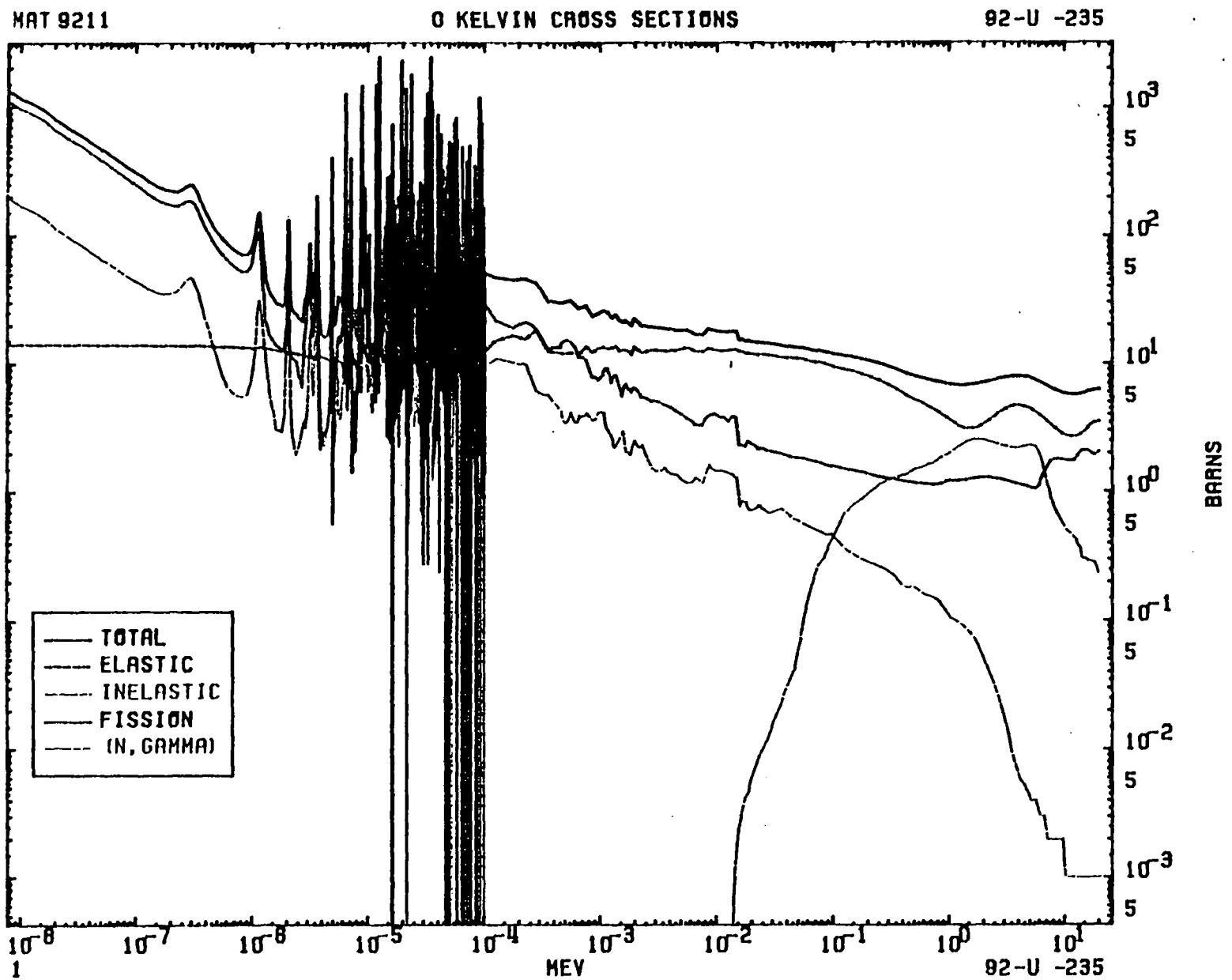

Figure 2: Cross sections from INDL/A-83 for ^{235}U, MAT=9211

- the threshold for the inelastic scattering in the ENDL-84 library differs considerably,
- the radiative capture cross section above 12 MeV rapidly decreases in the INDL/A evaluation and causes differences of more than a factor of four,
- above the resonance region the fission cross section in the INDL/A evaluation was taken over from ENDF/B-5. The differences between the original evaluation by Konshin taken from reference [4] (not included in the evaluated file), the ENDF/B-5 and the ENDF/B-6 evaluations are evident from Figure 4,
- the ratios of the evaluated cross sections (fission for example) in the thermal range exhibit some ripples of nearly 20% in magnitude. They are typical for incorrectly interpolated data and in fact it seems that the energy grid at very low energies in the INDL/A evaluation should be refined.

3.2 Evaluation of the measured ^{252}Cf fission spectrum averaged cross sections

The experimental measurements up to March 1988 were obtained from an EXFOR retrieval [16]. Their main characteristics are summarized in Table 1 and Figure 3. Some ambiguities in the published data are evident, particularly in the four publications by Adamov et.al. The EXFOR entry 40296 [21] is assigned "Preliminary" by the EXFOR evaluator. Entries 40465 [22] and 40547 [23] seem to be independent but the dates of publication, the list of authors and the identical measured values suggest that they refer to the same experiment and analysis. Neither of the two entries have the STATUS description in the EXFOR file. Furthermore a different value is reported by practically the same group of authors [24], cited in [25,26]. In the paper it is stated that it is the result of a more sophisticated analysis using more recent auxilliary data. Also, the EXFOR entry 12953 by Schroder has not been author-proofed.

In the present analysis the measurement by Schroder et.al. [20] is assumed final and the latest measurement by Adamov et.al. [24] is assumed to superseed previous ones. Five measurements remain which are consistent. Their weighted average and standard deviation (neglecting any correlation of the errors) are:

$$\overline{\sigma}_f^{Cf252} = 1.227 \pm 0.009$$

This should be compared to the evaluated *recommended* value by Mannhart on p.421 of ref. [11] which is a result of a much broader analysis including measurements for several isotopes and the covariance data.

$$\overline{\sigma}_f^{Cf252} = 1.210 \pm 0.014$$

The data base for the ^{235}U cross sections is practically the same but it excludes the latest values by Adamov [24] and Schroder [20] which favour a somewhat higher cross section. Neglecting the covariance data in the averaging process results in an underestimation of the error.

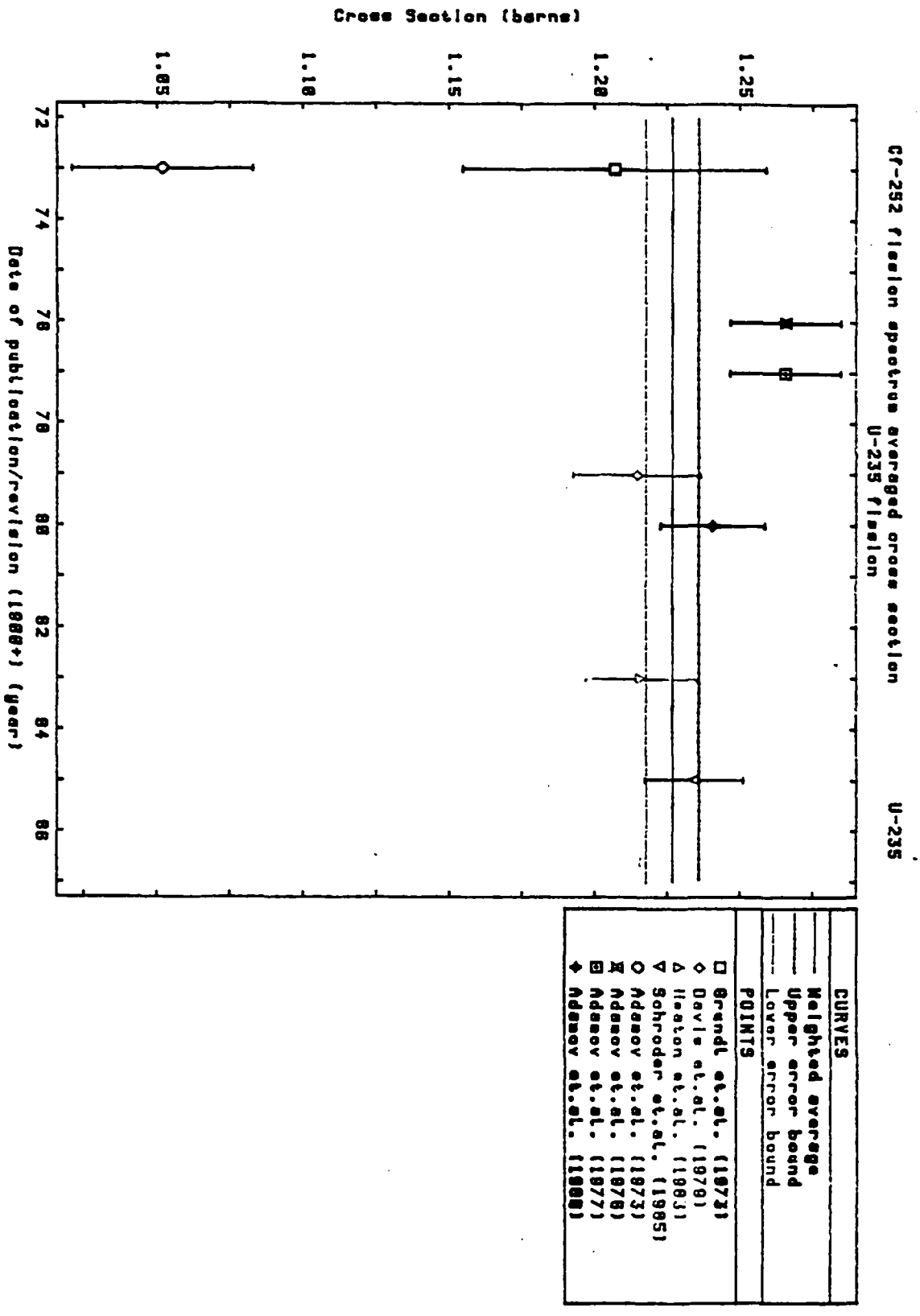

Figure 3: Experimentally measured ^{252}Cf fission spectrum averaged fission cross section of ^{235}U

Table 1: Experimental measurements of the ^{252}Cf fission spectrum averaged fission cross section of ^{235}U.

EXFOR entry	Ref.[a] (year)	Rev.[b] (year)	Status	Author	$\overline{\sigma}_f^{Cf252}$ (barns)
10304	1973	-	Approved	Grundl et.al. [17]	1.207 ± 0.052
10698	1978	1979	Approved	Davis, Knoll [18]	1.215 ± 0.022
10809	1976	1983	Approved	Heaton et.al. [19]	1.216 ± 0.019
12953	1985	-	No reply	Schroder et.al. [20]	1.234 ± 0.017
40296	1973	-	Prelim.	Adamov et.al. [21]	1.052 ± 0.031
40465	1976	-	-	Adamov et.al. [22]	1.266 ± 0.019
40547	1977	-	-	Adamov et.al. [23]	1.266 ± 0.019
-	1980	-	-	Adamov et.al. [24]	1.241 ± 0.018

(a) Year of the main reference publication,

(b) Year of the latest data revision,

Required Actions and Conclusions: Before a reliable estimate of the ^{252}Cf spontaneous fission spectrum averaged fission cross section of ^{235}U can be obtained the following actions are required:

- the EXFOR entry 12953 [20] should be author proofed,
- the status of the EXFOR entries 40296, 40465, 40547 and their relation to the later published value [24] should be clarified with the authors.
- a full analysis such as performed by Mannhart [11] should be repeated including the new data.

If the above ambiguities could favourably be resolved the averaged experimental measurements would be of sufficient quality to serve as a benchmark for evaluated data.

3.3 The Thermal Neutron Induced ^{235}U Fission Spectrum Averaged Cross Sections

In the EXFOR file only two measurements of the fission cross section exist as shown in Table 2, measured relative to ^{10}B and ^{115}In, respectively. Because of the large error they have no practical significance. On the other hand there exist ratio measurements relative to ^{238}U which are of sufficient quality. Three measurements are found in the EXFOR file and six measurements are cited by Garakani and Darbandi [29]. Two of them coincide with the EXFOR entries. They are presented in Table 3. The weighted average of the data gives:

$$\overline{\sigma}_{f(U235)}^{U235} \,/\, \overline{\sigma}_{f(U238)}^{U235} = 3.873 \pm 0.060$$

what is of sufficient quality to be used as a benchmark for evaluated data testing. However, its impact on ^{235}U data is limited because it would require a very accurate

Table 2: Experimental measurements of the thermal neutron induced ^{235}U fission spectrum averaged cross section of ^{235}U.

EXFOR entry	Ref. (year)	Author	$\overline{\sigma}_f^{U235}$ (barns)
20076	1955	Raisic [27]	1.6 ± 0.9
20264	1968	Fabry et.al. [28]	1.34 ± 0.13

Table 3: Experimental measurements of the thermal neutron induced ^{235}U fission spectrum averaged cross section ratios $\overline{\sigma}^{U235}_{f(U235)}/\overline{\sigma}^{U235}_{f(U238)}$.

EXFOR entry	Ref. (year)	Author	$\overline{\sigma}^{U235}_{f(U235)}/\overline{\sigma}^{U235}_{f(U238)}$ (barns)
12193	1968	Grundl [30]	3.85 ± 0.23
20946	1975	Fabry et.al. [31]	3.94 ± 0.08
20947	1978	Fabry et.al. [32]	3.94 ± 0.14
-	1970	Fabry et.al. [33]	3.78 ± 0.18
-	1972	Grundl [34]	3.71 ± 0.17
-	1975	McElroy, Kellog [35]	3.82 ± 0.24
-	1983	Garakani, Darbandi [29]	3.83 ± 0.25

estimate of the average fission cross section of ^{238}U. Evaluation of the ^{238}U average cross section measurements is beyond the scope of this work.

The thermal neutron induced ^{235}U fission spectrum averaged cross sections are are very similar to the ^{252}Cf spontaneous fission spectrum averaged ones, with regard to the energy range as well as the shape of the spectrum. Furthermore there are fewer uncertainties in obtaining the ^{252}Cf spectrum averaged cross sections so they usually take preference over the ^{235}U spectrum averaged cross sections.

3.4 Resonance Integrals and Thermal Cross Sections

A recent and very extensive compilation of resonance integrals and thermal cross sections was compiled by Gryntakis et.al. [11]. All available experimental data up to January 1985 were included. No new measurements were found in the EXFOR file therefore the resonance integrals and the thermal cross sections were taken over from their compilation:

$$R.I._f = 276.3 \pm 2.8$$

$$R.I._\gamma = 141.8 \pm 4.2$$

$$\sigma_f^{th} = 582.2 \pm 1.3$$

$$\sigma_\gamma^{th} = 98.6 \pm 1.5$$

3.5 Comparison of Data from Evaluated Libraries with Integral Measurements

Absolute measurements of the spectrum averaged fission cross section of ^{235}U are very rare, in fact they exist only for the ^{252}Cf spontaneous fission spectrum. The rest are ratio measurements in the thermal neutron induced ^{235}U fission spectrum and some reactor spectra. They do not help directly to improve the ^{235}U data because the absolute spectrum averaged cross sections of the complementary materials are not known with sufficient accuracy.

The ^{252}Cf spectrum averaged fission cross section, the resonance integrals (fission and capture) and the $2200ms^{-1}$ cross sections (fission and capture) derived from evaluated libraries are compared to the average of the measured values (this work) and the recommended values from ref. [11] in Table 4.

Table 4: Comparison of the ^{235}U cross sections from evaluated libraries with the measured values

	$\bar{\sigma}_f^{Cf252}$	$R.I._f$	$R.I._\gamma$	σ_f^{th}	σ_γ^{th}
Average	1.227±0.009	-	-	-	-
Recommended	1.210±0.014	276.0±2.8	141.8±4.2	582.2±1.3	98.6±1.5
INDL/$A_{Sup.86}$	1.238	276.8	142.8	582.6	98.3
ENDF/B-5	1.237	281.7	139.2	583.5±1.7	98.4
ENDF/B-6 [a]	1.218	-	-	584.2±1.1[b]	98.96±0.74
ENDL-84	1.234	283.9	139.6	602.0	100.7

(a) ENDF/B-5 below 150 eV; ENDF/B-6 below 150 keV not final.

(b) Quoted value for 300 K.

The fission cross section above the unresolved resonance region in INDLA is taken over from ENDF/B-5 and hence the exact agreement between the ^{252}Cf fission spectrum averaged fission cross sections. The ENDF/B-5 evaluation of the fission cross section at thermal and above $150keV$ energies was an international standard which has recently been superseeded by the ENDF/B-6 evaluation. The new evaluation shows good agreement with integral measurements in the fast neutron energy range (see also the comments in section 3.2). At the thermal energy the "cold" cross section from the file was not available. The quoted value was taken from the comment section and it is stated for $300K$. It is interesting to note that doppler broadening of the INDL/A fission cross sections to $300K$ produces an increase of 0.3% at $0.0253eV$ producing a value 584.1 *barns*. The effect of doppler broadening on the Maxwellian spectrum averaged cross sections is insignificant.

3.6 Conclusions

Regarding experimental measurements of the ^{252}Cf fission spectrum averaged fission cross sections the actions suggested in section 3.2 should be considered. The latest ENDF/B-6 evaluation is consistent with the *average* and the *recommended* values. The other evaluations could be slightly too high.

Generally the agreement between the INDL/A evaluation and integral measurements is quite good but some improvement is possible on the following:

- resolved resonance parameter should be given so that negative cross sections would not occur,
- since the new standard for the fission cross section above 150 keV is available it could be included in the file instead of the older ENDF/B-5 evaluation.

The new INDL/A evaluation for ^{235}U offers an attractive alternative to the ENDF/B-5 evaluation which is also available.

4 Analysis of the ^{239}Pu data

4.1 Intercomparison of Evaluated Data

The evaluated files were analysed in the same way as for ^{235}U. A processing message was issued that in 61 resonances the sum of the partial widths does not mach the total. The lagest discrepancy is 0.005% so it may be considered unimportant. It was also noted that for all reactions the cross section value for the last energy point at $20MeV$ is zero. No other problems were observed. The INDL/A evaluation can be compared in Appendix B with the ENDF/B-5 Dosimetry Mod.2 evaluation on Figure 18, and with the ENDL-84 evaluation on Figures 19 – 24.

4.2 Comparison of Data from Evaluated Libraries with Integral Measurements

The *recommended* ^{252}Cf fission spectrum averaged fission cross section, the resonance integrals and the thermal cross section values are taken from reference [11], p.421 and p.199. The experimental measurements extracted from the EXFOR retrieval are listed in Table 5. For data averaging purposes the same comments apply as for ^{235}U since they refer to the same EXFOR entries. The datum from [23] was therefore not included in the analysis.

The integral quantities derved from evaluated libraries of interest are compared to the measured values (*averaged* – this work and *recommended* – ref. [11]) in Table 6.

4.3 Conclusions

The ^{252}Cf spontaneous fission spectrum averaged fission cross sections derived from evaluated libraries are consistent with the *recommended* value. The new measurements and the *average* suggest a slightly higher value. Of the evaluated libraries

Table 5: Experimental measurements of the ^{252}Cf fission spectrum averaged fission cross section of ^{239}Pu.

EXFOR entry	Ref.[a] (year)	Rev.[b] (year)	Status	Author	$\overline{\sigma}_f^{Cf252}$ (barns)
10698	1978	1979	Approved	Davis, Knoll [18]	1.790 ± 0.041
10809	1976	1983	Approved	Heaton et.al. [19]	1.824 ± 0.035
12953	1985	-	No reply	Schroder et.al. [20]	1.844 ± 0.024
20868	1970	-	from ref.	Pauw et.al. [36]	1.800 ± 0.060
40547	1977	-	-	Adamov et.al. [23]	1.861 ± 0.030
-	1980	-	-	Adamov et.al. [24]	1.831 ± 0.027

(a) Year of the main reference publication,

(b) Year of the latest data revision,

Table 6: Comparison of the ^{239}Pu cross sections from evaluated libraries with the measured values

	$\overline{\sigma}_f^{Cf252}$	$R.I._f$	$R.I._\gamma$	σ_f^{th}	σ_γ^{th}
Average	1.828±0.014	-	-	-	-
Recommended	1.811±0.025	312.2±8.2	191±16	744.4±1.7	268.8±3.0
INDL/A$_{Sup.86}$	1.803	306.3	186	748.1	269.3
ENDF/B-5	1.794	304.0	-	732.5	-
ENDF/B-6	-	-	-	748.0±1.9	271.4±2.1
ENDL-84	1.782	307.0	206	785.7	278.0

considered the differences in integral quantities derived from them are very small and the overall agreement with the integral measurements is very good. The INDL/A evaluation is perhaps marginally better compared to other evaluations.

5 Analysis of the ^{240}Pu data

5.1 Intercomparison of Evaluated Data

The evaluated files were analysed in the same way as for ^{235}U. Like in the case of ^{239}Pu it was noted that for all reactions the cross section value for the last energy point at $20MeV$ is zero. No other problems were observed. The INDL/A evaluation can be compared in Appendix C with the ENDL-84 evaluation on Figures 25 – 30. No ENDF/B-5 data are available.

5.2 Comparison of Data from Evaluated Libraries with Integral Measurements

The *recommended* resonance integrals and the thermal cross section values are taken from reference [11]. The *recommended* ^{252}Cffission spectrum averaged fission cross section of ^{240}Pu does not exist but the experimental measurements extracted from the EXFOR retrieval are listed in Table 7.

Table 7: Experimental measurements of the ^{252}Cf fission spectrum averaged fission cross section of ^{240}Pu.

EXFOR entry	Ref. (year)	Status	Author	$\overline{\sigma}_f^{Cf252}$ (barns)
12821	1983	No reply	Grundl et.al. [37]	1.337 ± 0.032
40841	1983	-	Adamov et.al. [38]	1.310 ± 0.037

The integral quantities derived from evaluated libraries of interest are compared to the measured values (*averaged* – this work and *recommended* – ref. [11]) in Table 8.

Table 8: Comparison of the ^{240}Pu cross sections from evaluated libraries with the measured values

	$\overline{\sigma}_f^{Cf252}$	$R.I._f$	$R.I._\gamma$	σ_f^{th}	σ_γ^{th}
Average	1.325±0.024	-	-	-	-
Recommended	-	5	8460±305	0.035±0.045	289.5±1.4
INDL/A$_{Sup.86}$	1.345	9.7	8420	0.059	287.7
ENDL-84	1.421	10.	9328	0.060	301.5

5.3 Conclusions

Only two measurements for the ^{252}Cf spontaneous fission spectrum averaged fission cross section of ^{240}Pu exist. The value derived from INDL/A is in agreement with the measurements while the value from the ENDL-84 library seems be overpredicted. Similarly the capture resonance integral derived from INDL/A is in good agreement with the *recommended* value while the one derived from the ENDL-84 library is underpredicted. The opposite sign of the deviations in the ENDL data may have an effect on the capture to fission ratio. The fission resonance integral is too rough to serve as a criterion for the subthreshold fission cross section checking.

The ENDF/B-5 data are not available.

6 Analysis of the ^{241}Pu data

6.1 Intercomparison of Evaluated Data

The evaluated files were analysed in the same way as for ^{235}U. In the resolved resonance region the Adler-Adler parameters are used to represent the cross sections. On reconstructing the pointwise cross sections a lot of warnings were issued about negative cross sections, particularly in the elastic cross section. The problem was examined in more detail by Holubar [39]. Although the problem could not be resolved completely due to some ambiguity in the definition of one of the parameters in the evaluation description [4], it could be concluded that the data do not obey the ENDF rules, therefore the standard ENDF processing codes can not be used with certainty on the INDL/A data for ^{241}Pu.

The INDL/A evaluation can be compared in Appendix D with the ENDL-84 evaluation on Figures 31 - 36. No ENDF/B-5 data are available.

6.2 Comparison of Data from Evaluated Libraries with Integral Measurements

The resonance integrals and the thermal cross section values are taken from reference [11]. The *recommended* ^{252}Cffission spectrum averaged fission cross section of ^{241}Pu does not exist but the experimental measurements extracted from the EXFOR retrieval are listed in Table 9.

Table 9: Experimental measurements of the ^{252}Cf fission spectrum averaged fission cross section of ^{241}Pu.

EXFOR entry	Ref. (year)	Status	Author	$\overline{\sigma}_f^{Cf252}$ (barns)
12821	1983	No reply	Grundl et.al. [37]	1.616 ± 0.080
40841	1983	-	Adamov et.al. [38]	1.744 ± 0.054

The integral quantities derived from evaluated libraries of interest are compared to the measured values (*averaged* – this work and *recommended* – ref. [11]) in Table 10.

Table 10: Comparison of the ^{241}Pu cross sections from evaluated libraries with the measured values

	$\overline{\sigma}_f^{Cf252}$	$R.I._f$	$R.I._\gamma$	σ_f^{th}	σ_γ^{th}
Average	±	-	-	-	-
Recommended	-	558±18	161±13	1009±8	368±10
INDL/A$_{Sup.86}$	1.621	564	140	1011	358
ENDF/B-6	-	-	-	1012.7±6.6	361.3±4.9
ENDL-84	1.595	581	226	997	400

6.3 Conclusions

Only two measurements for the ^{252}Cf spontaneous fission spectrum averaged fission cross section of ^{241}Pu exist. The values derived from the evaluated libraries are in agreement with the *average*. The fission resonance integral from INDL/A agrees well with the *recommended* value while the capture resonance integral is slightly too low, but the results have to be taken with caution due to the inconsistent use of the Adler-Adler parameters. Both resonance integrals derived from the ENDL-84 are overpredicted compared to the *recommended* value.

The ENDF/B-5 data are not available.

7 Analysis of the ^{242}Pu data

7.1 Intercomparison of Evaluated Data

The evaluated files were analysed in the same way as for ^{235}U. Like in the case of ^{239}Pu it was noted that for all reactions the cross section value for the last energy point at $20MeV$ is zero. No other problems were observed. The INDL/A evaluation can be compared in Appendix E with ENDF/B-5 Actinides Mod.2 evaluation on Figures 37 – 43 (comparison does not include inelastic scattering for discrete levels), and with the ENDL-84 evaluation on Figures 44 – 49.

7.2 Comparison of Data from Evaluated Libraries with Integral Measurements

The resonance integrals and the thermal cross section values are taken from reference [11]. No measurements of the ^{252}Cffission spectrum averaged fission cross section of ^{242}Pu were found.

The integral quantities derived from evaluated libraries of interest are compared to the *recommended* values form ref. [11] in Table 11.

Table 11: Comparison of the ^{242}Pu cross sections from evaluated libraries with the measured values

	$\bar{\sigma}_f^{Cf252}$	$R.I._f$	$R.I._\gamma$	σ_f^{th}	σ_γ^{th}
Recommended	-	5	1131±57	<0.2	18.5±0.4
INDL/A$_{Sup.86}$	1.194	23	1180	0.001	18.6
ENDF/B-5	1.136	5.3	1288	0.001	19.2
ENDL-84	1.133	40.	1311	0.0009	18.8

7.3 Conclusions

No measurements for the ^{252}Cf spontaneous fission spectrum averaged fission cross section of ^{242}Pu exist. The capture resonance integral derived from INDL/A is in agreement with the *recommended* value while the values derived from the ENDF/B-5 and the ENDL-84 libraries are overpredicted. The subthreshold fission resonance integral is too rough to serve as a criterion for the fission cross section checking.

8 Relation of INDL/A to the BROND File

Very recently the BROND – USSR Evaluated Neutron Data Library became available [40]. It contains some evaluations from other files and a number of completely new evaluations. A preliminary analysis showed the following:

- the cross section data for ^{235}U are the same as in INDL/A,
- the cross section data for ^{242}Pu are the same except that the cross section values at $20MeV$ are non-zero,
- a format error was detected in the inelastic cross section data of ^{239}Pu which caused a fatal error so comparison could not be performed (a fault on magnetic tape is not excluded),
- the pointwise cross section data for ^{240}Pu and ^{241}Pu are tabulated at room temperature and resonance parameters are given assuming $0K$ temperature. There exists some ambiguity on how to treat the data so processing was terminated.

Although the analysis of the BROND library is very limited it shows that it contains evaluations which are practically the same as those in the INDL/A library. The differences seem to be only in some trivial corrections.

9 Summary of Conclusions

Within the present scope of analysis the agreement of the integral quantities derived from the INDL/A library with measured values is very good. A similar analysis should be performed for the ^{238}U data with the emphasis on the BROND evaluation.

A complete set of evaluations for the major actinides would then be obtained. In spite of some minor problems revealed during processing the INDL/A evaluations considered in this work seem worth analysing further in some more sophisticated benchmark such as criticality and spectral indices measurements.

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APPENDIX

A – Comparison of ^{235}U fine group cross sections form INDL/A with ENDF/B-5 and ENDL-84 data

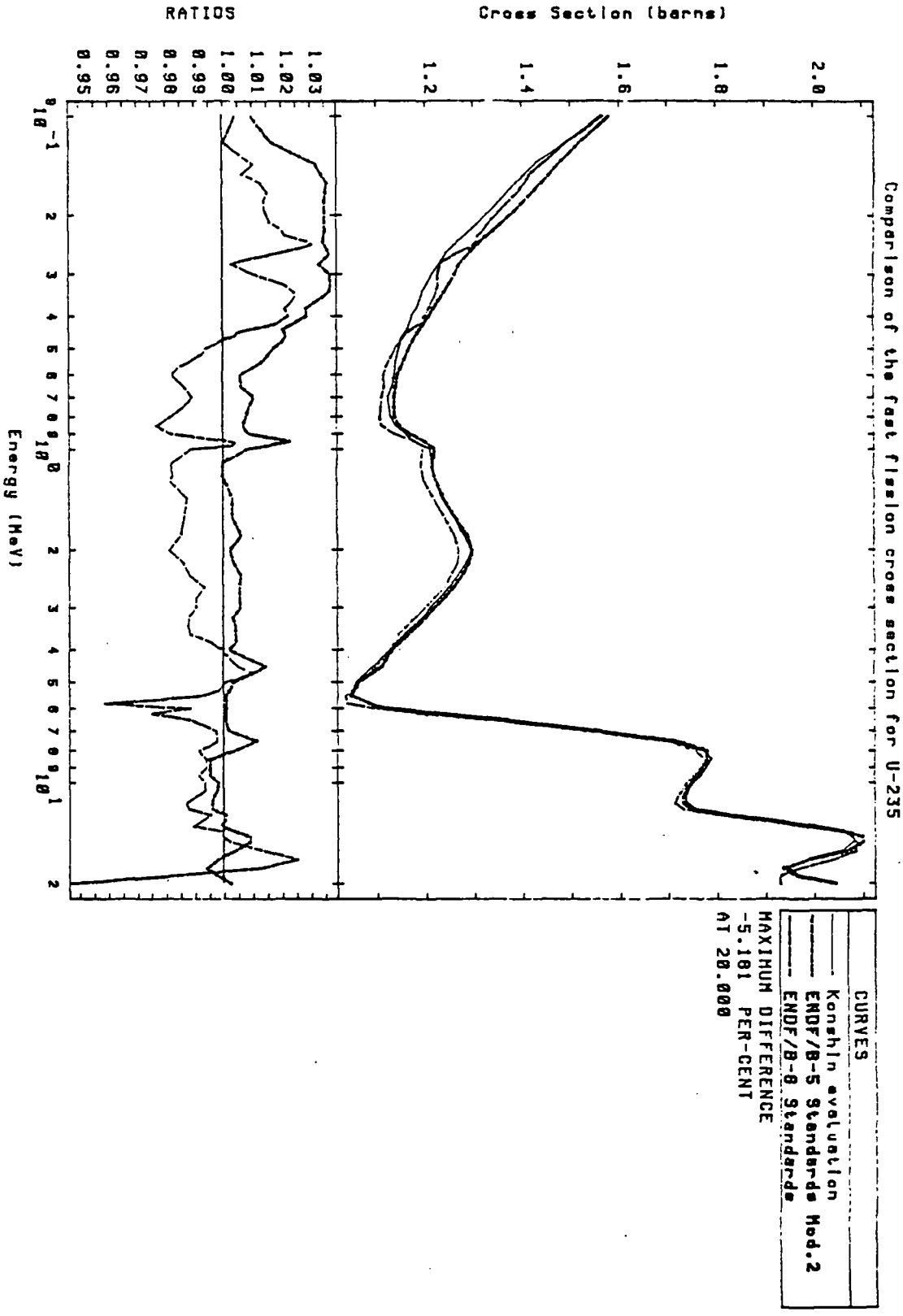


Figure 4: ^{235}U fission cross section comparison between the Konshin evaluation, the ENDF/B-5 and the ENDF/B-6 evaluations

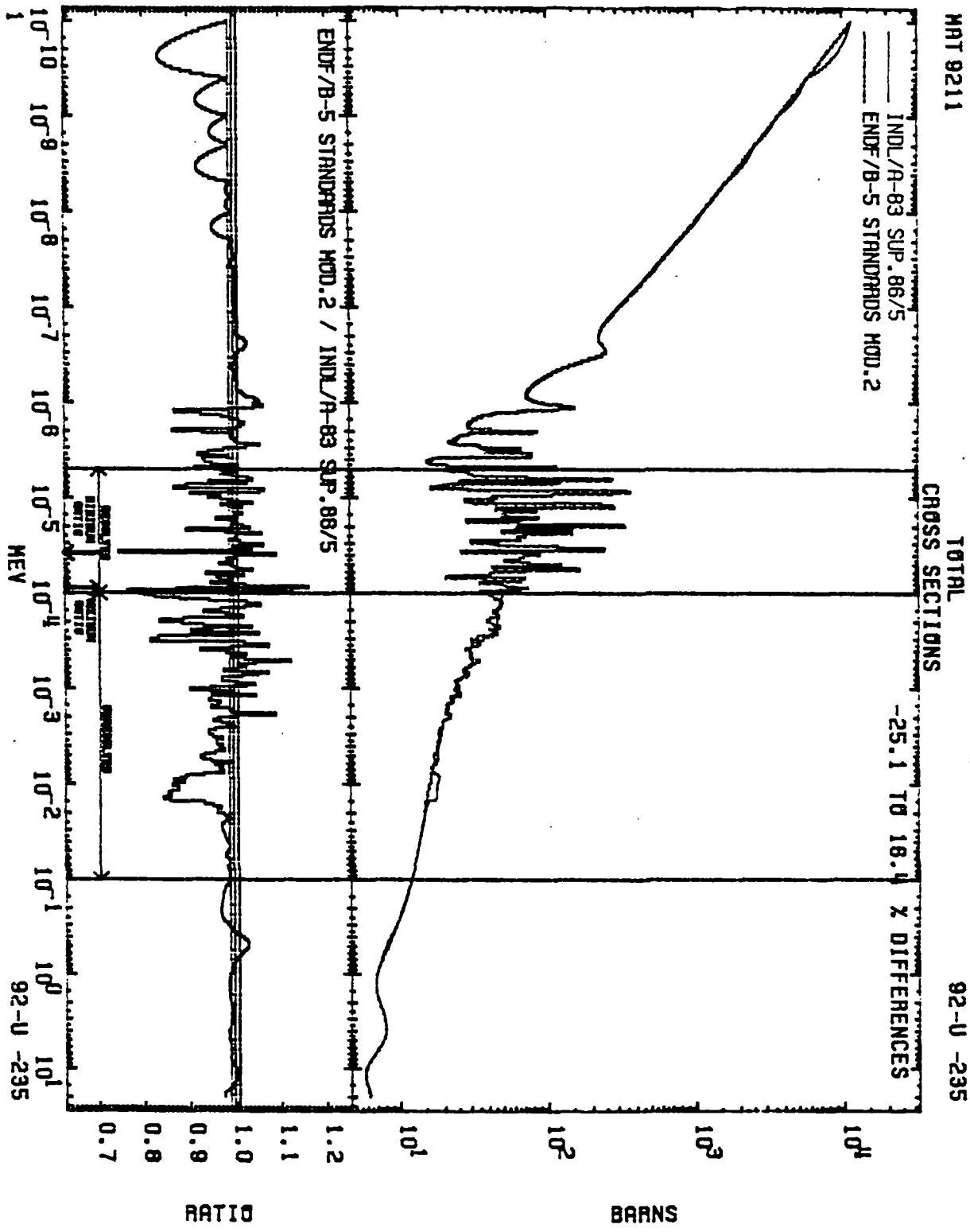


Figure 5: ^{235}U cross section comparison between INDL/A and ENDF/B-5 evaluations

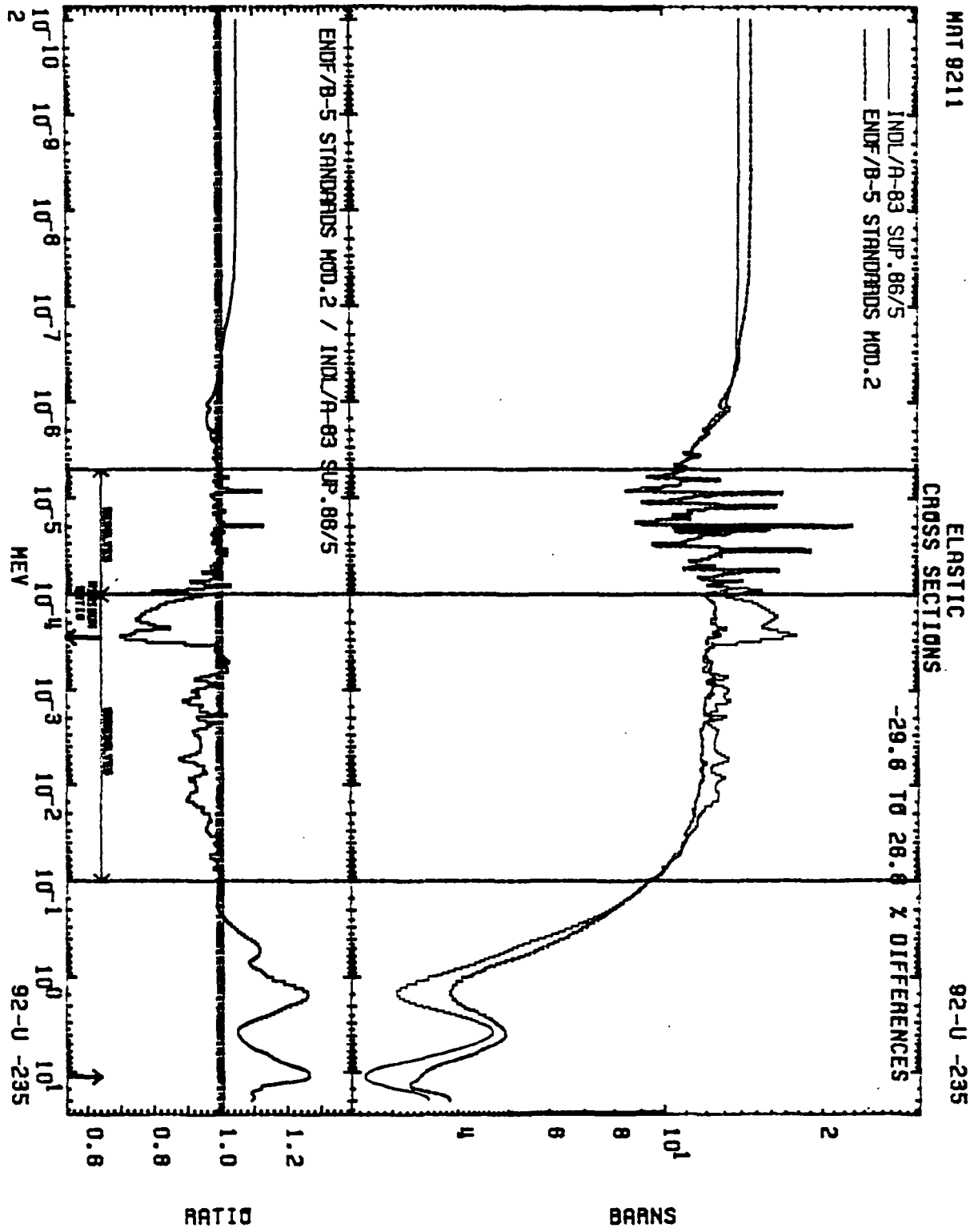


Figure 6: ^{235}U cross section comparison between INDL/A and ENDF/B-5 evaluations

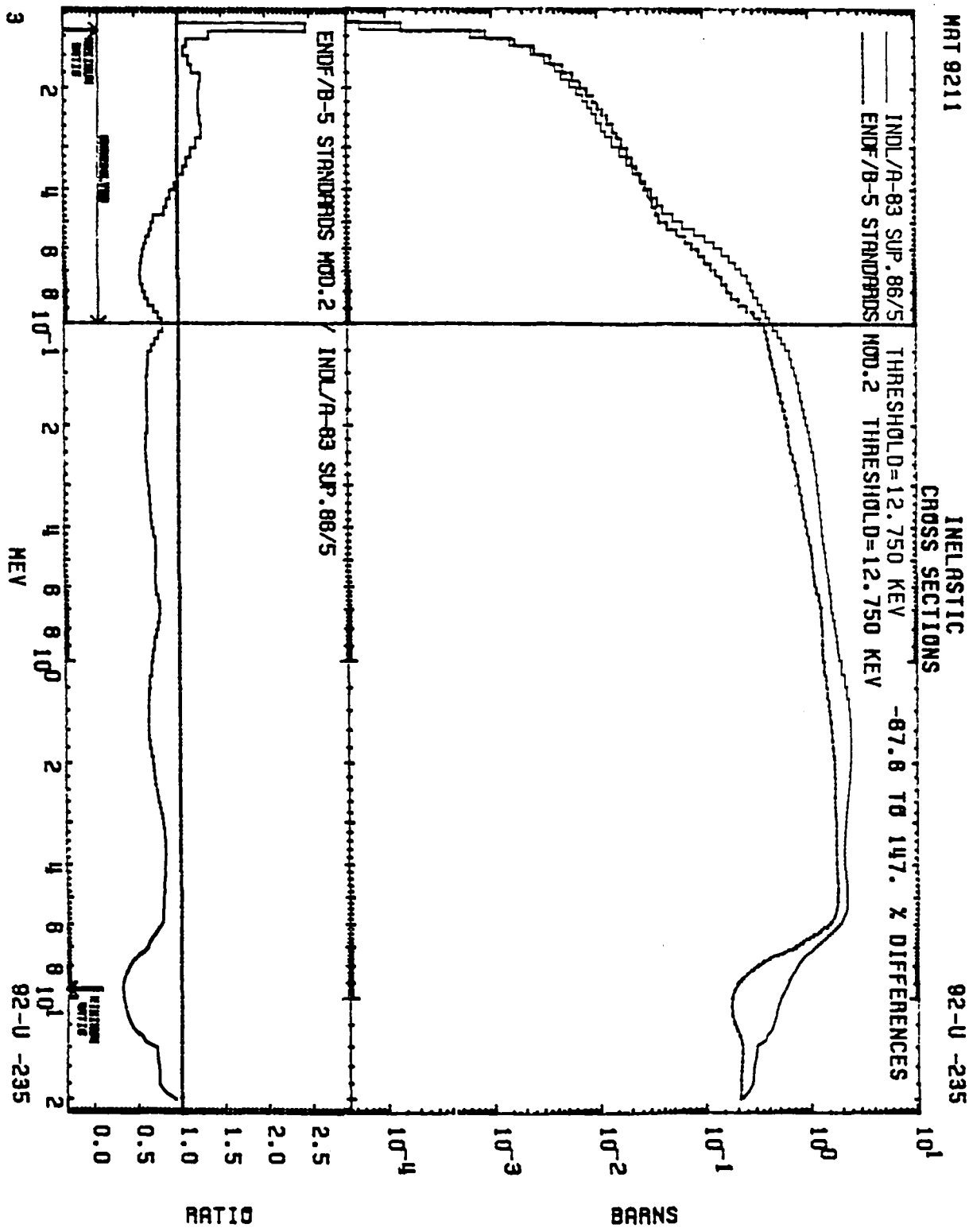


Figure 7: ^{235}U cross section comparison between INDL/A and ENDF/B-5 evaluations

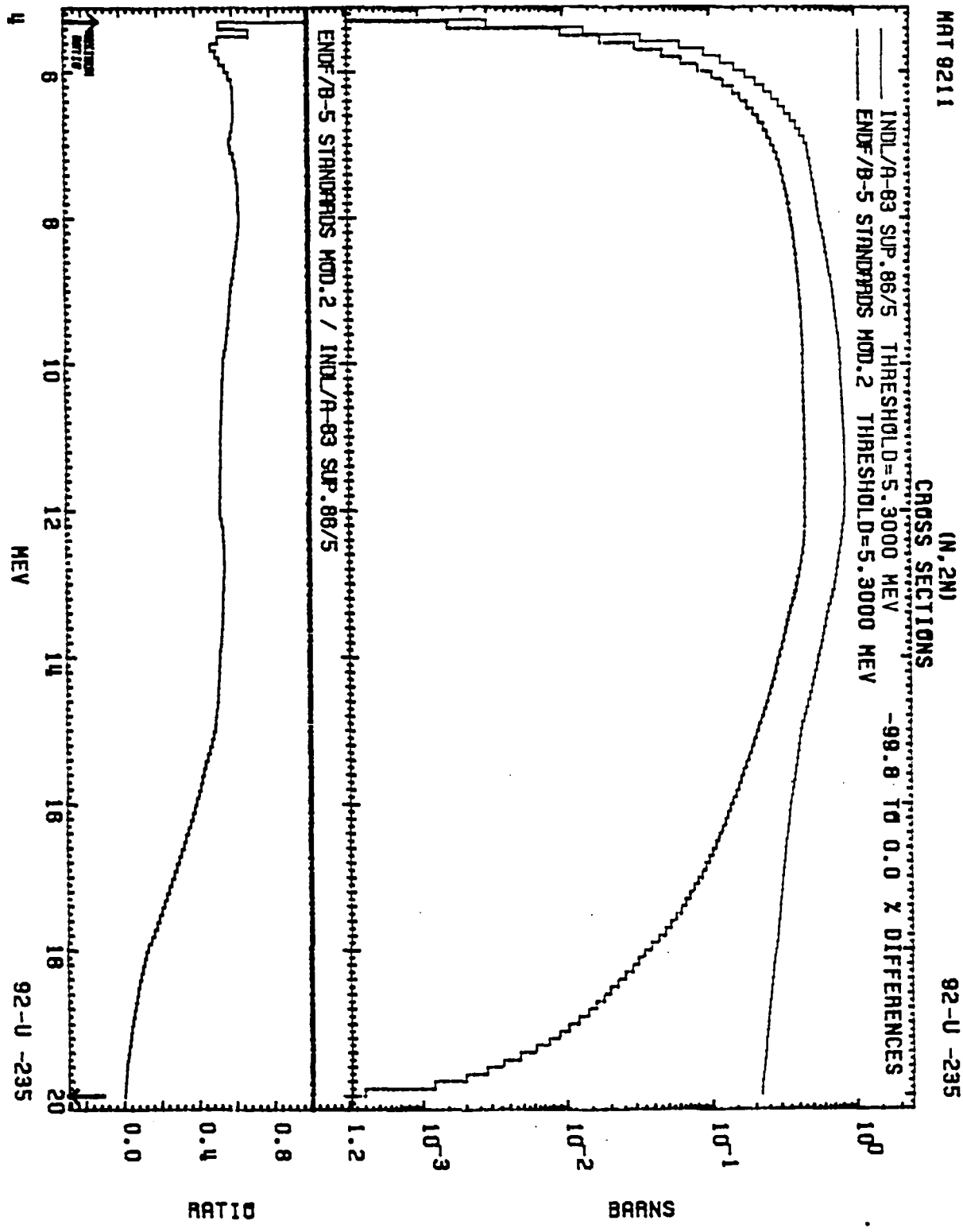


Figure 8: ^{235}U cross section comparison between INDL/A and ENDF/B-5 evaluations

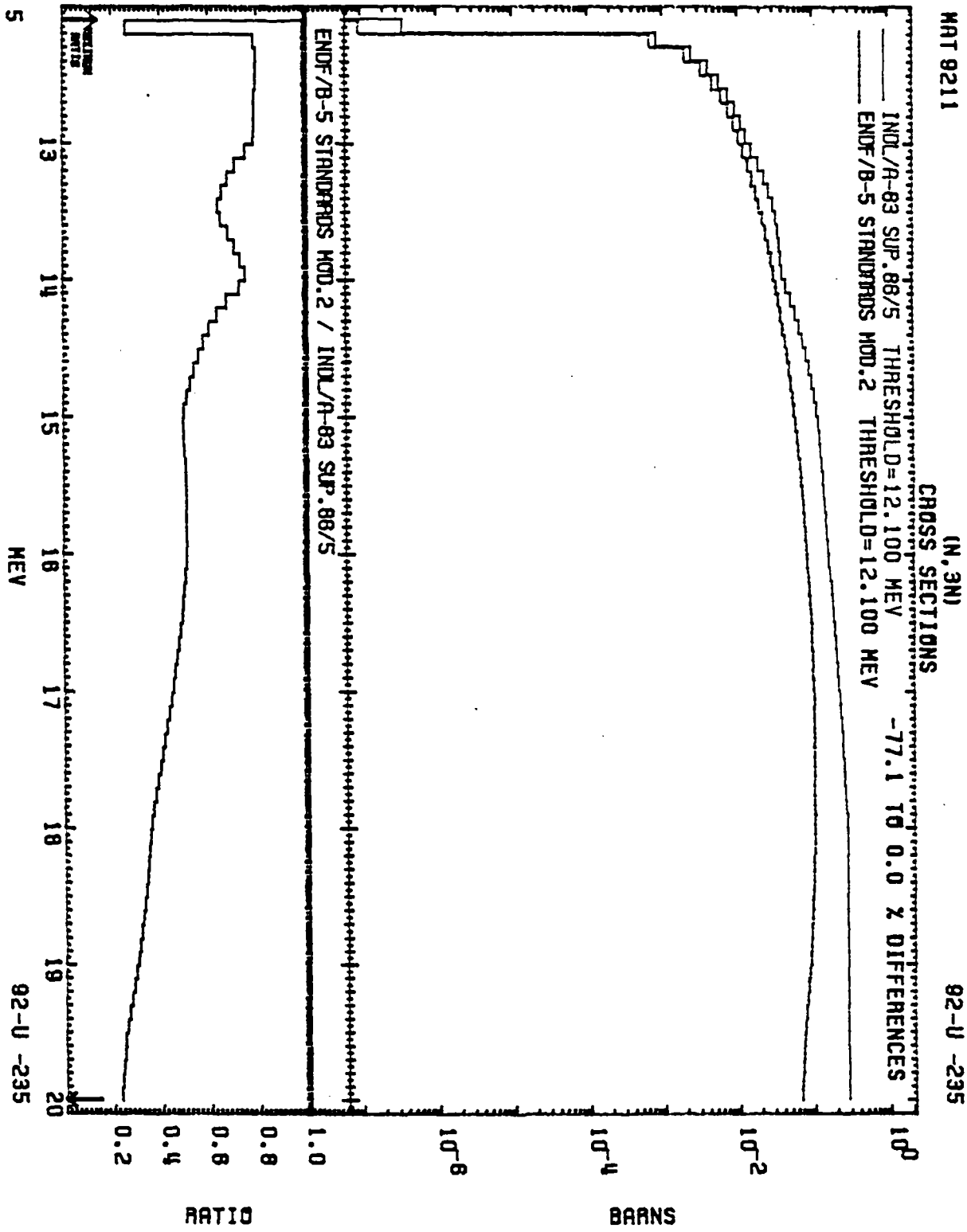


Figure 9: ^{235}U cross section comparison between INDL/A and ENDF/B-5 evaluations

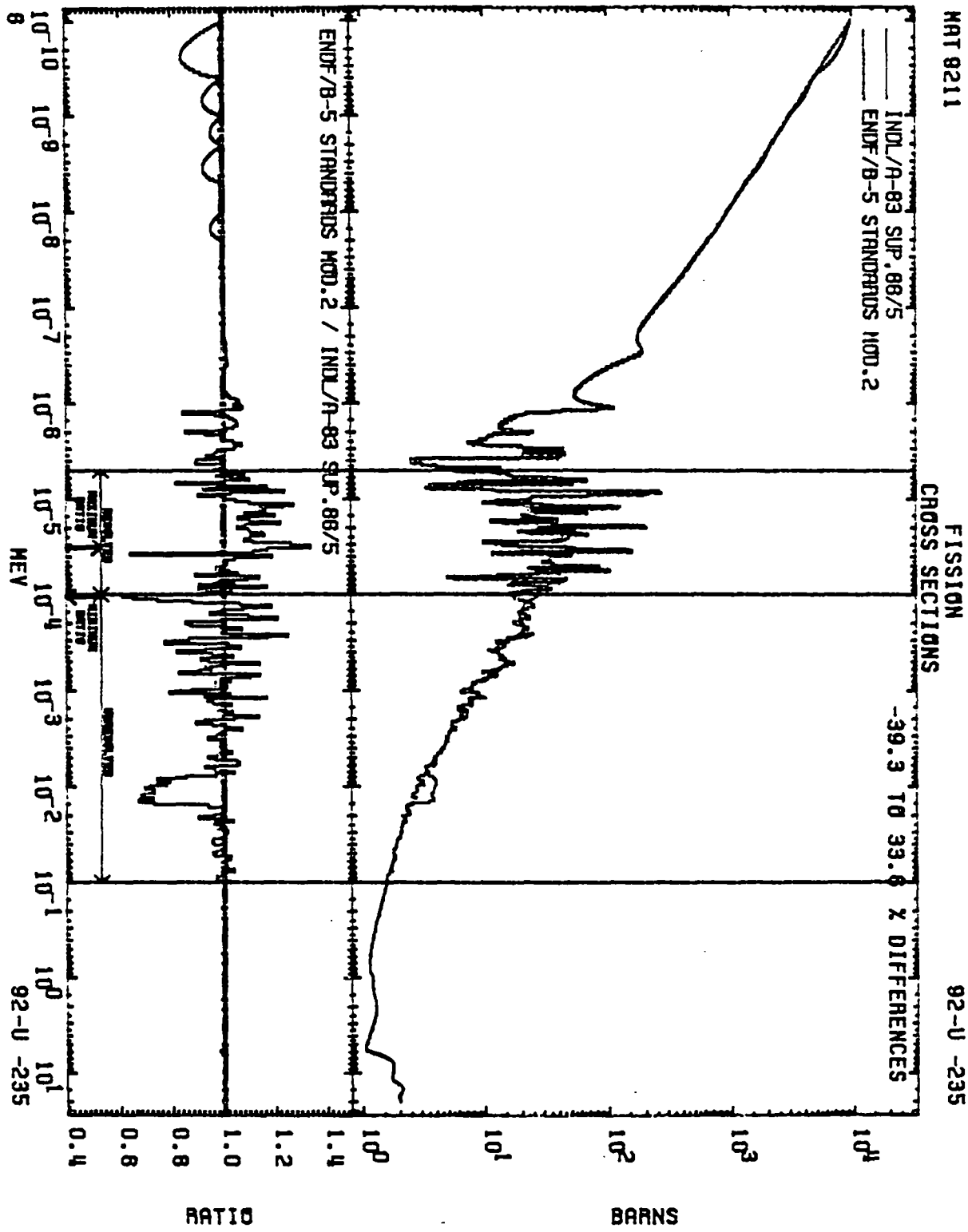


Figure 10: ^{235}U cross section comparison between INDL/A and ENDF/B-5 evaluations

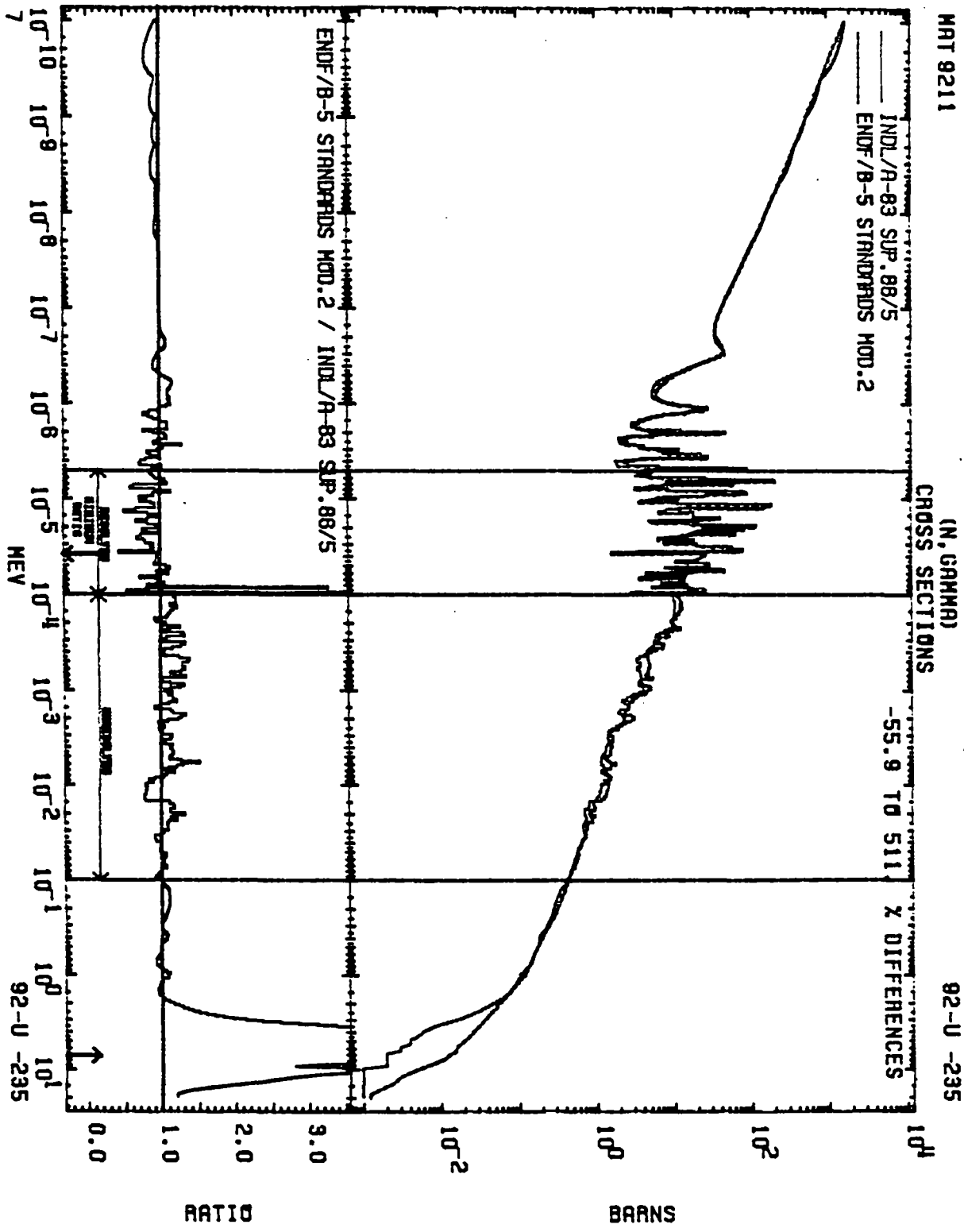


Figure 11: ^{235}U cross section comparison between INDL/A and ENDF/B-5 evaluations

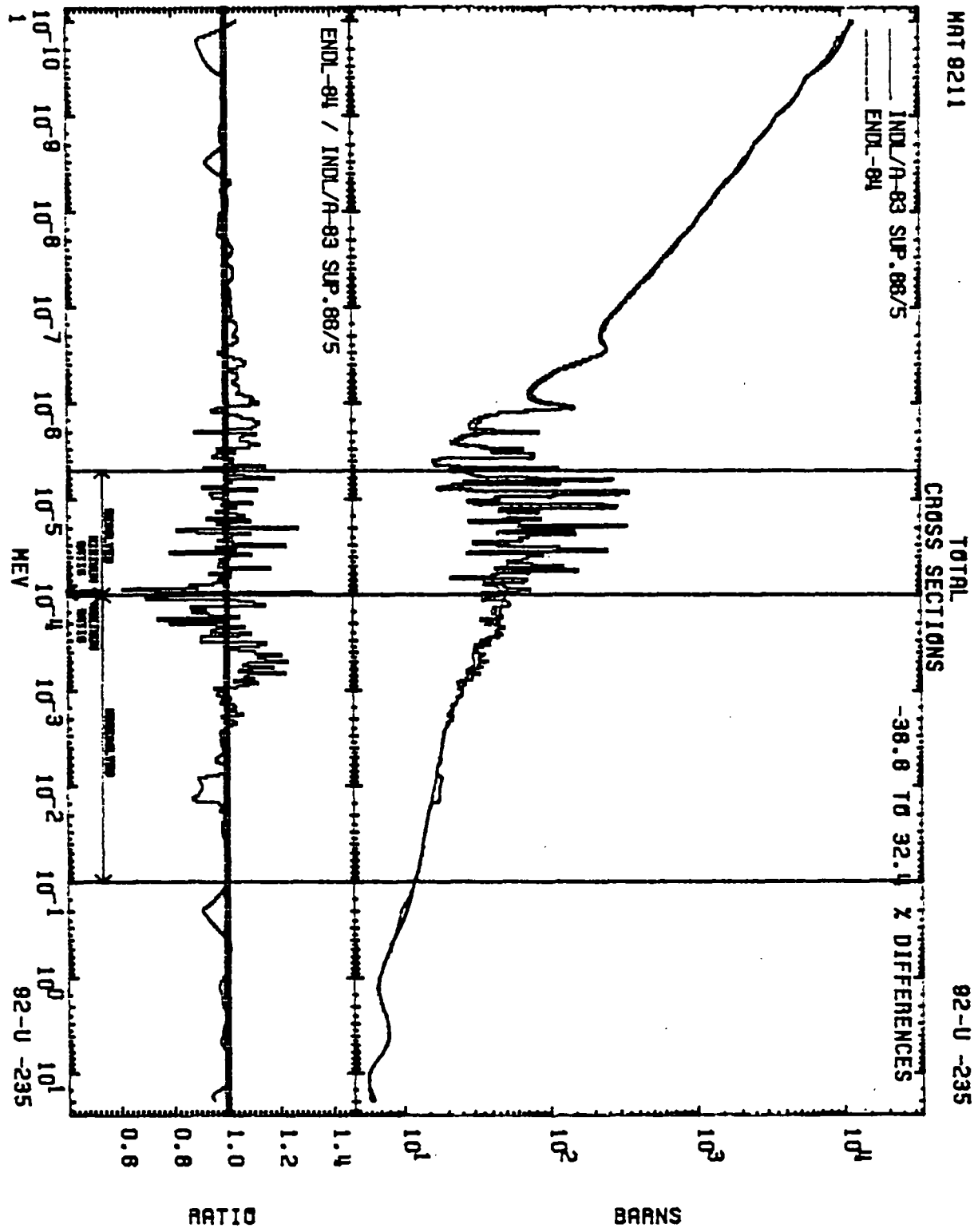


Figure 12: ^{235}U cross section comparison between INDL/A and ENDL-84 evaluations

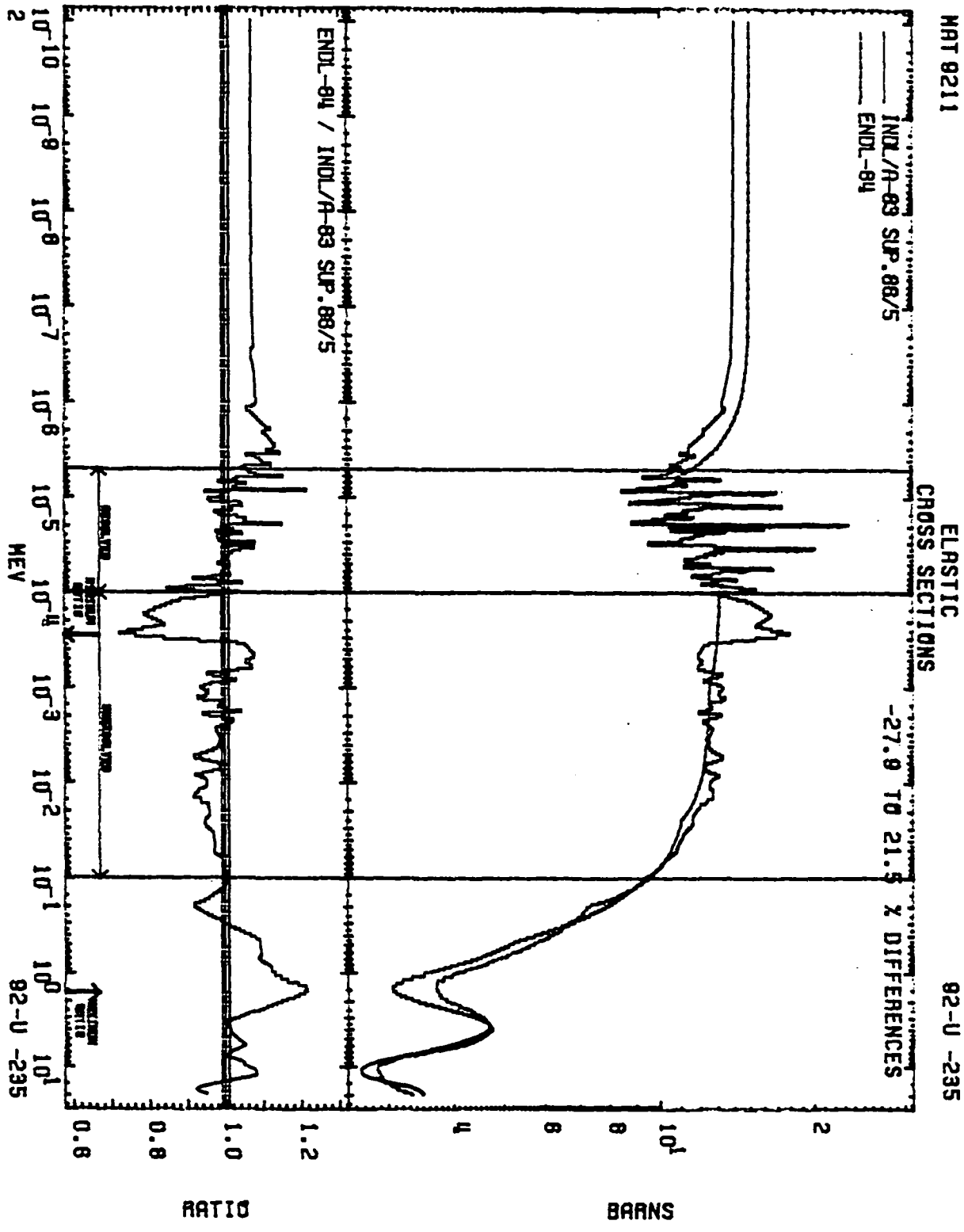


Figure 13: ^{235}U cross section comparison between INDL/A and ENDL-84 evaluations

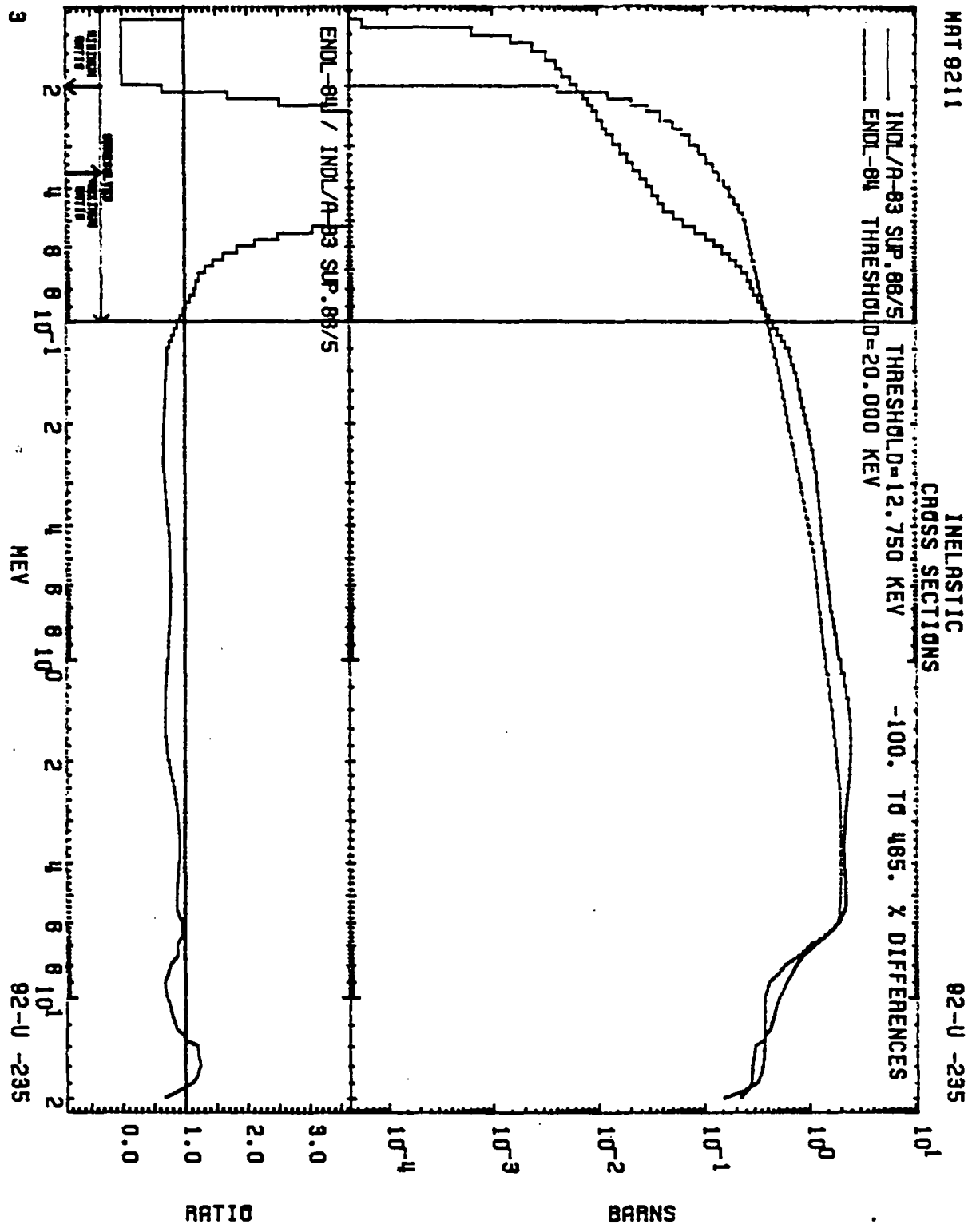


Figure 14: ^{235}U cross section comparison between INDL/A and ENDL-84 evaluations

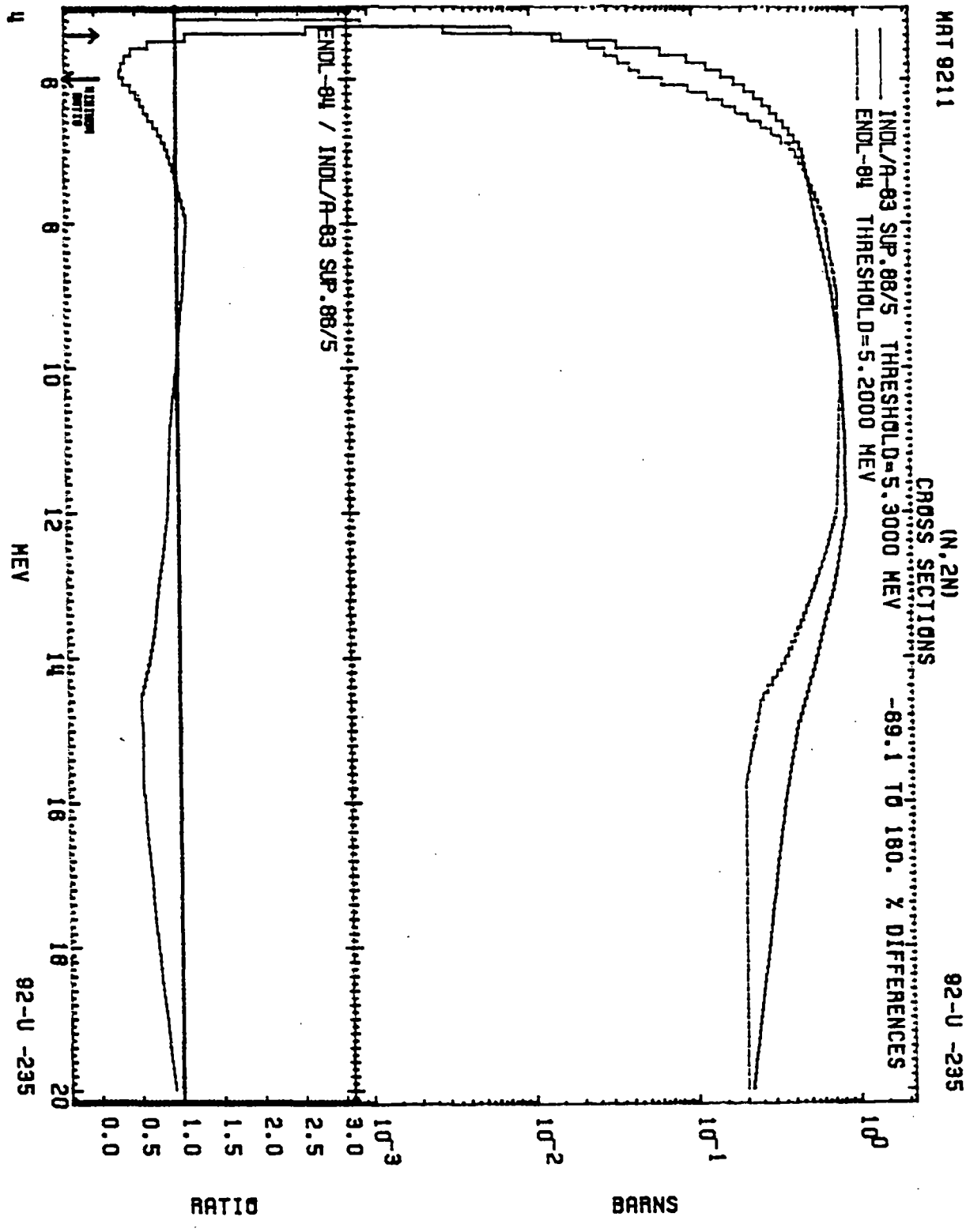


Figure 15: ^{235}U cross section comparison between INDL/A and ENDL-84 evaluations

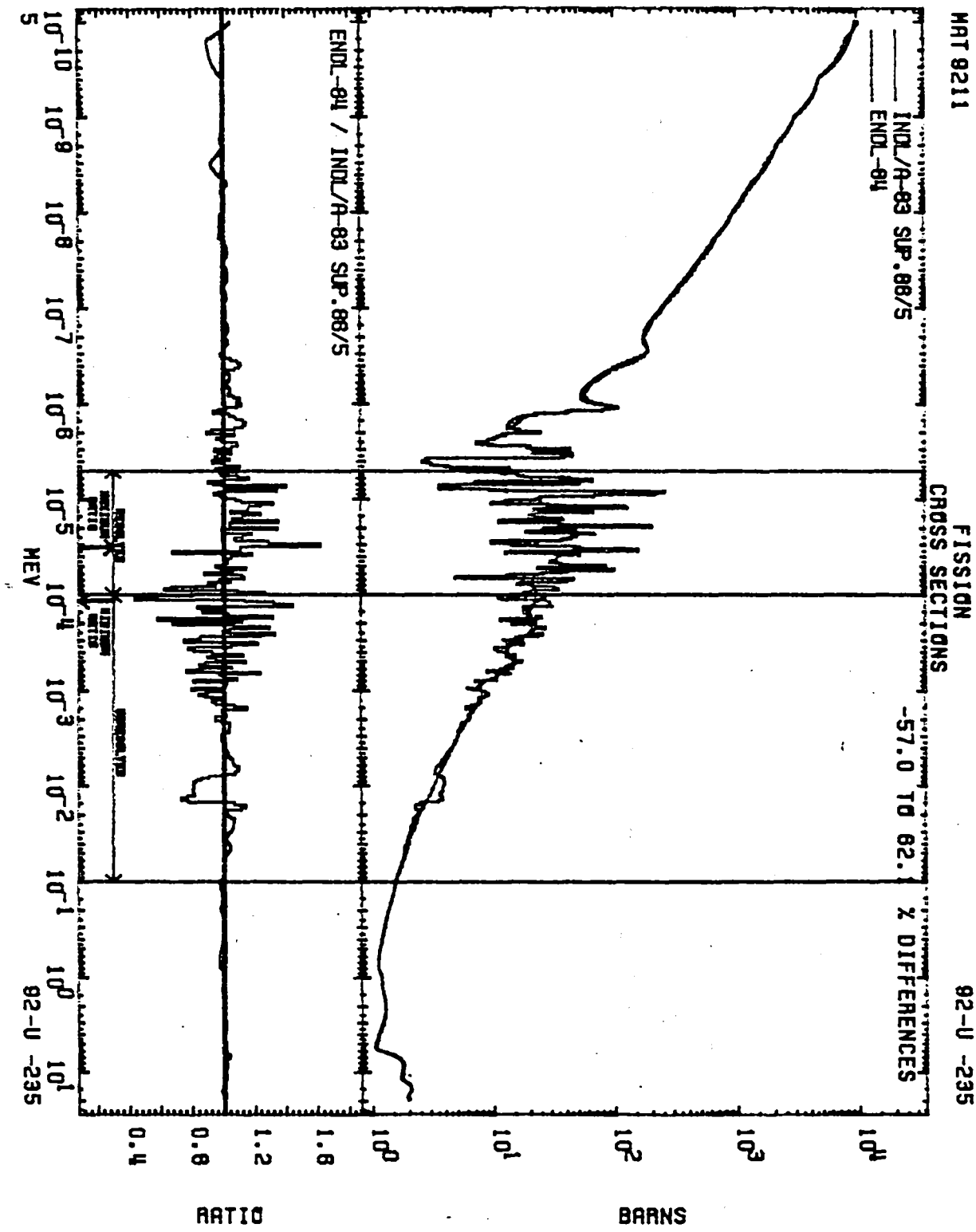


Figure 16: ^{235}U cross section comparison between INDL/A and ENDL-84 evaluations

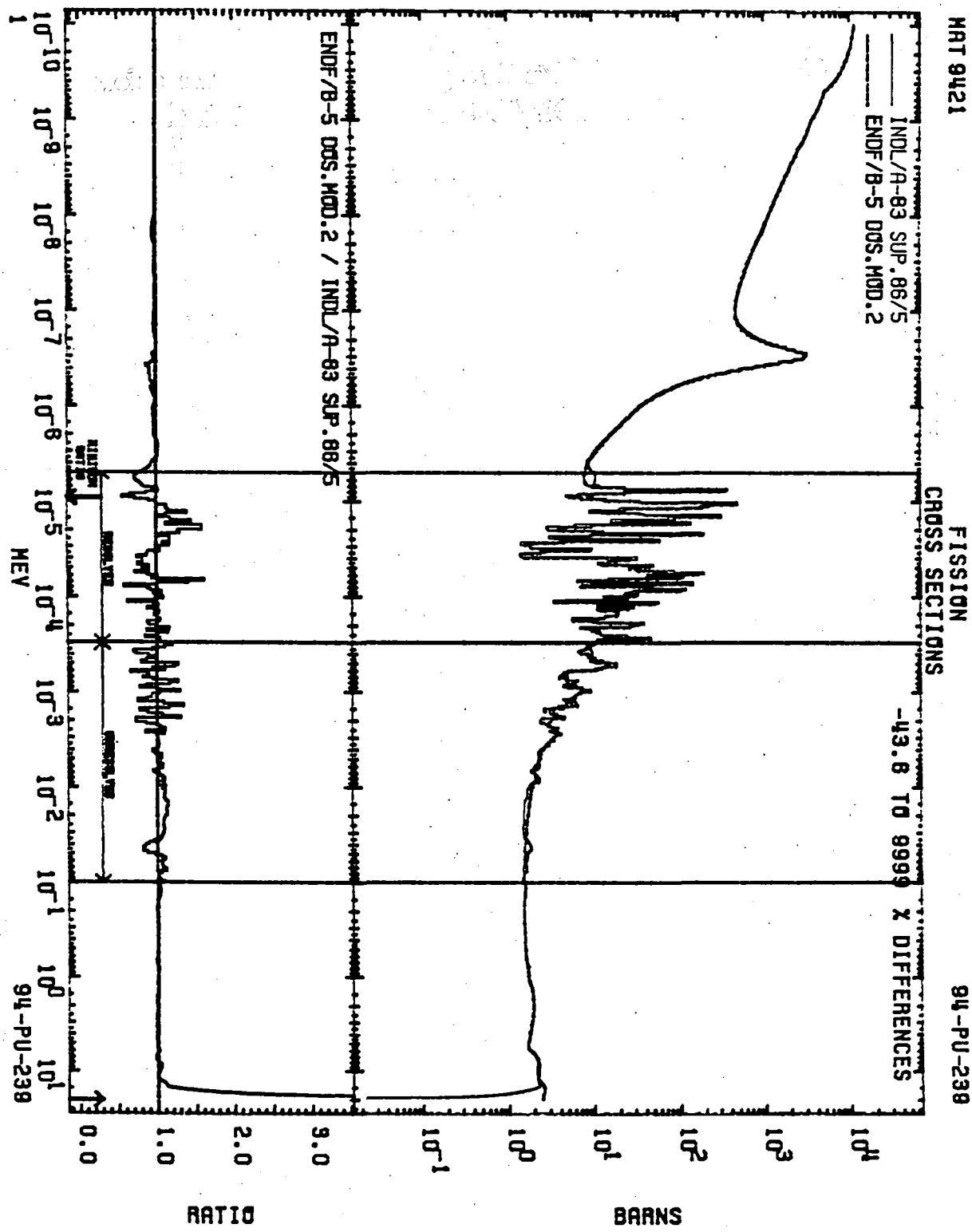


Figure 18: ^{239}Pu cross section comparison between INDL/A and ENDF/B-5 evaluations

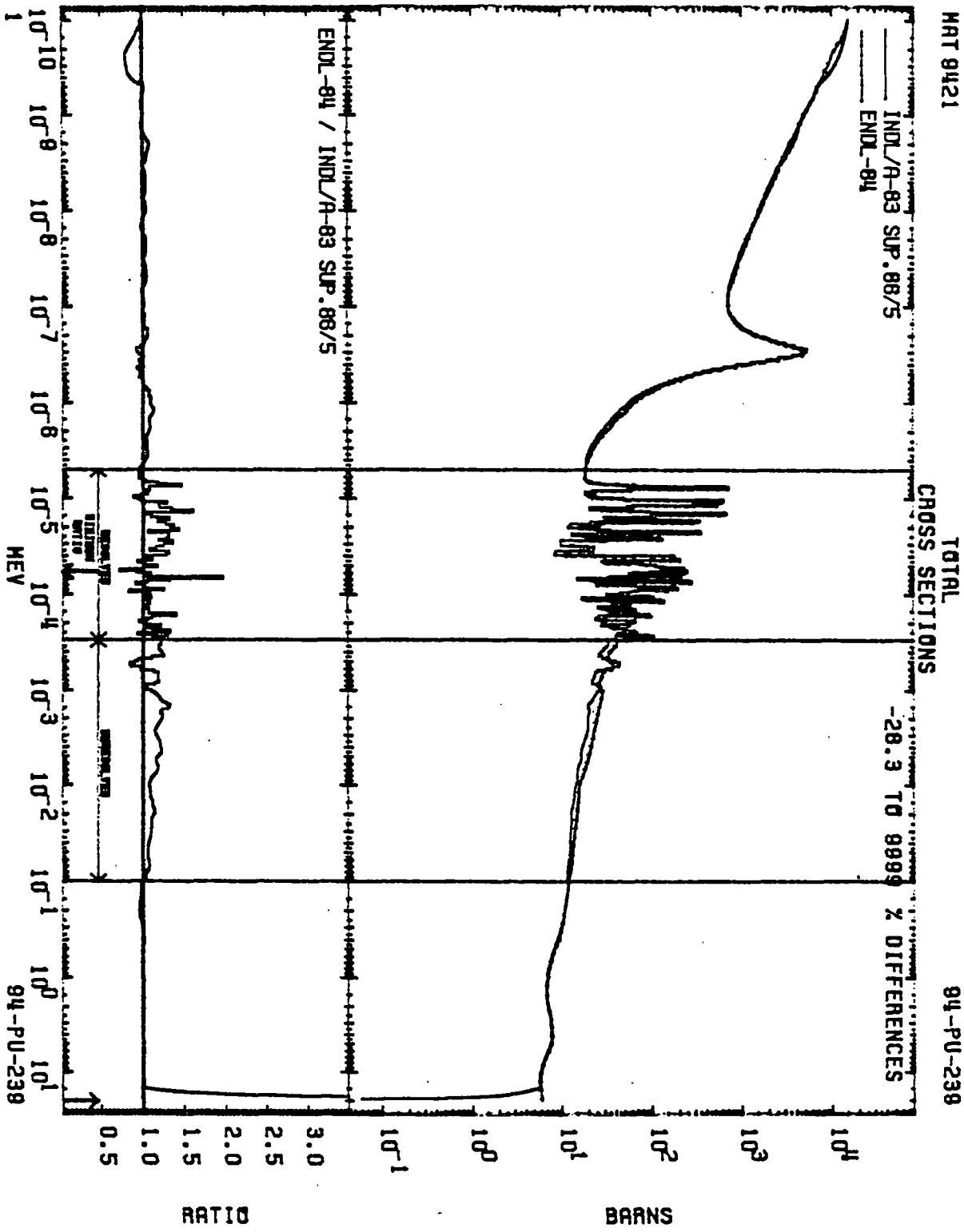


Figure 19: ^{239}Pu cross section comparison between INDL/A and ENDL-84 evaluations

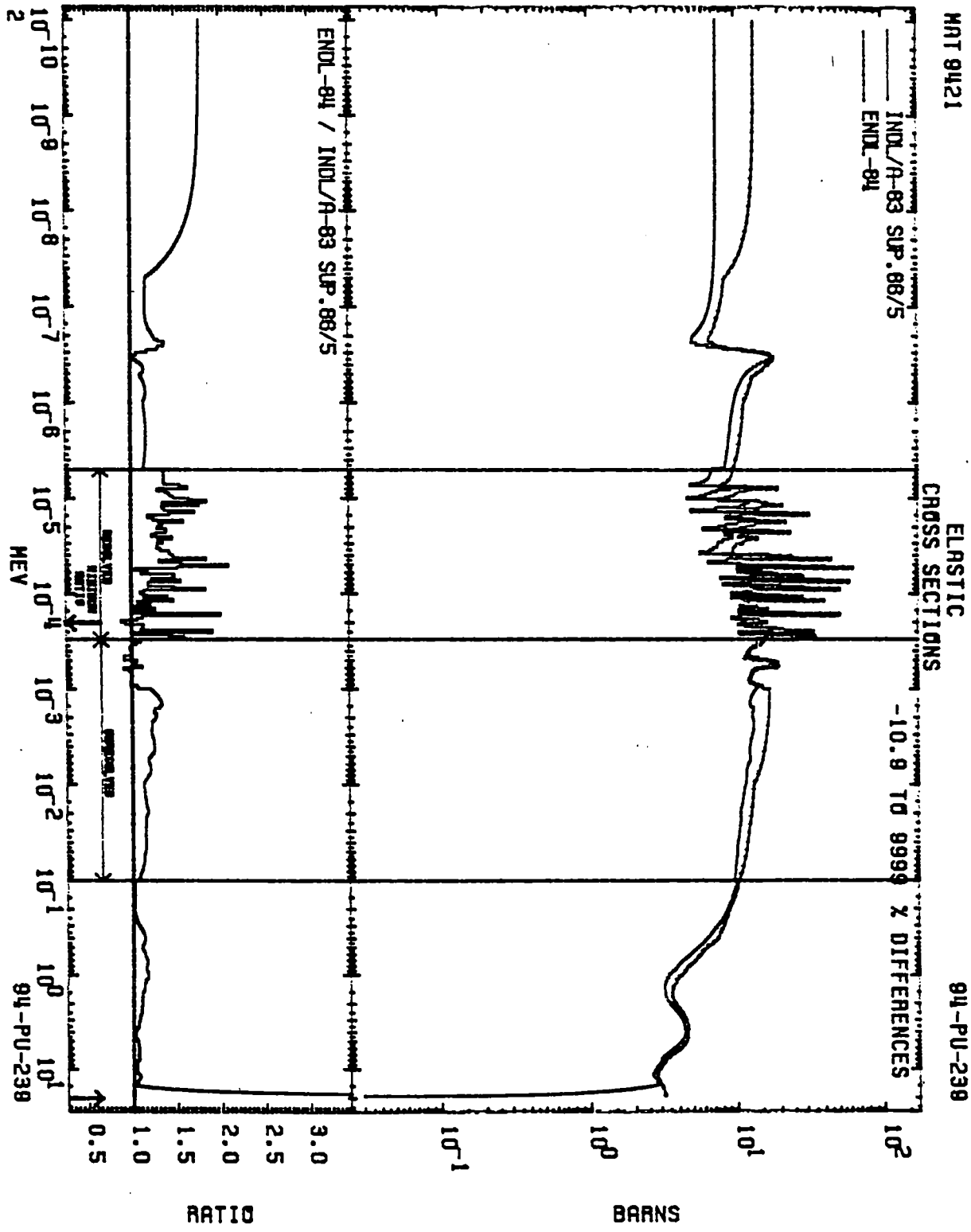


Figure 20: ^{239}Pu cross section comparison between INDL/A and ENDL-84 evaluations

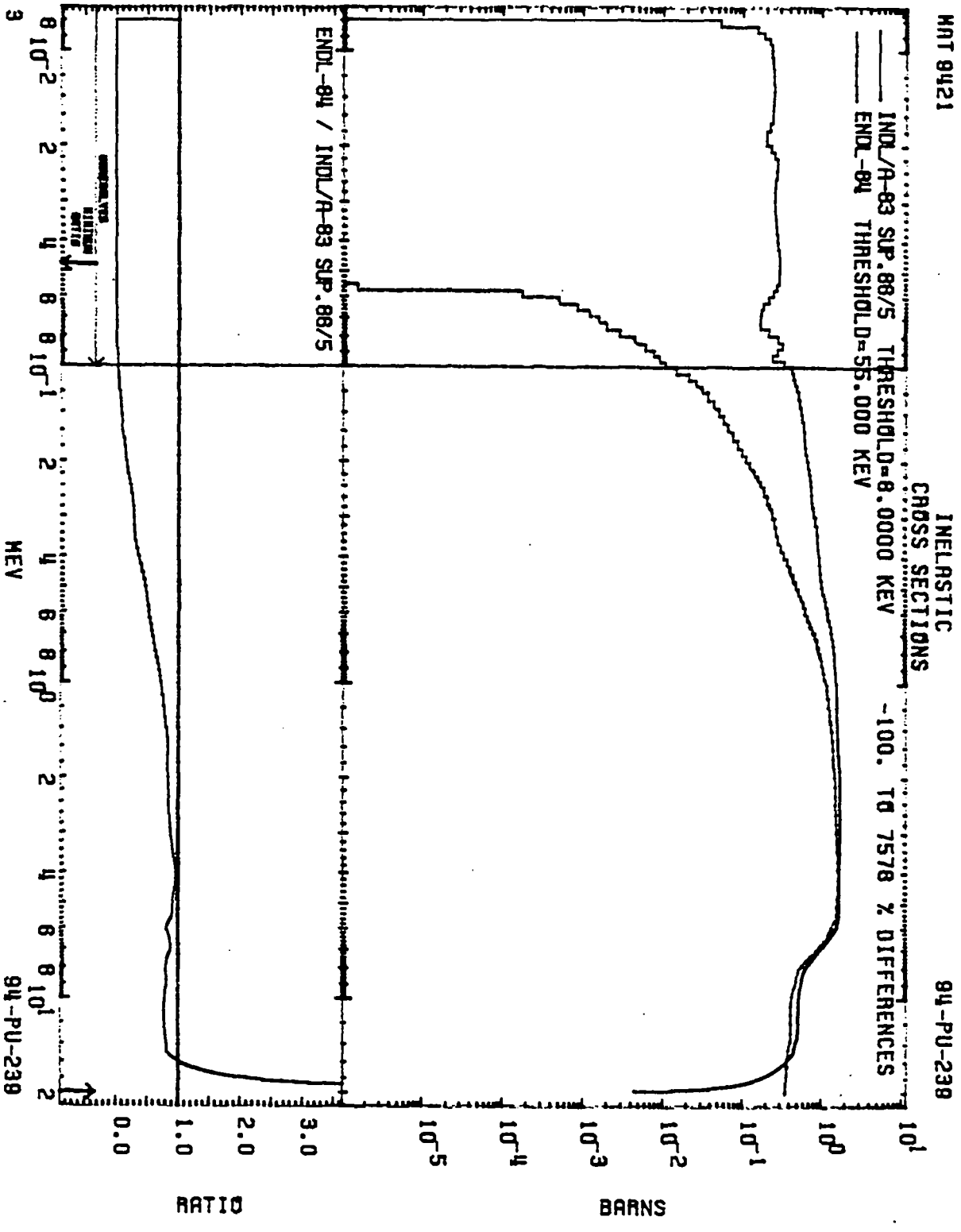


Figure 21: ^{239}Pu cross section comparison between INDL/A and ENDL-84 evaluations

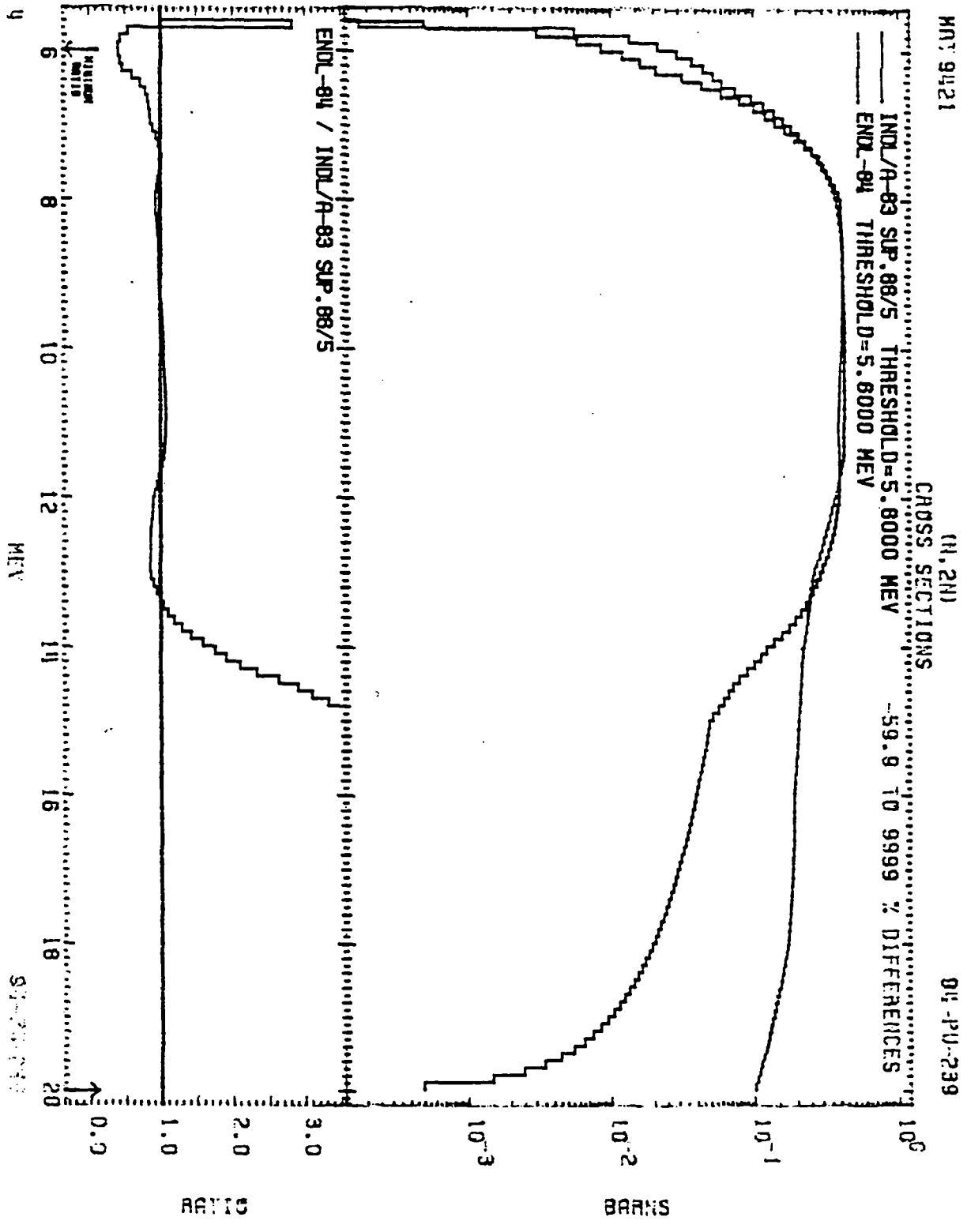


Figure 22: ^{239}Pu cross section comparison between INDL/A and ENDL-84 evaluations

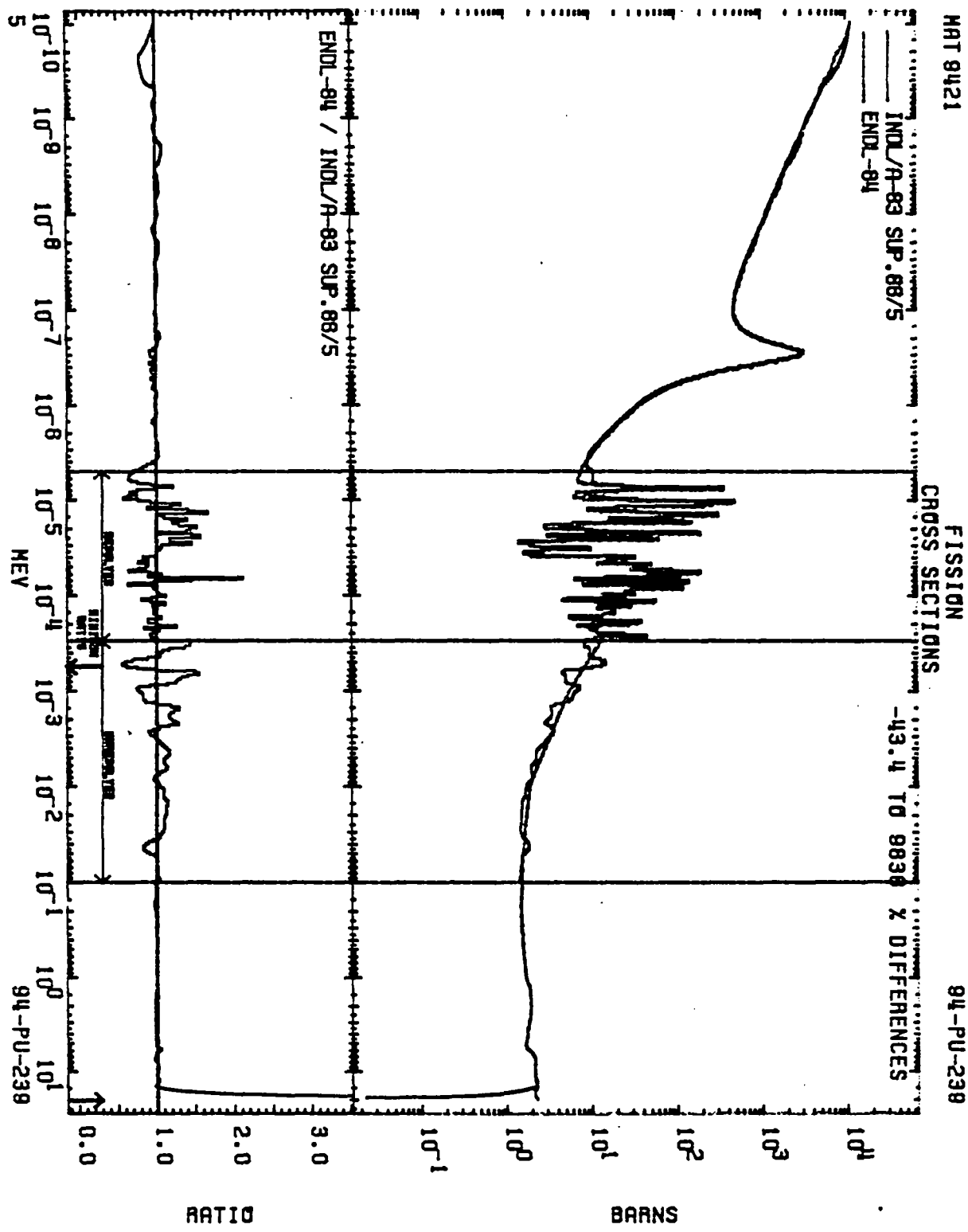


Figure 23: ^{239}Pu cross section comparison between INDL/A and ENDL-84 evaluations

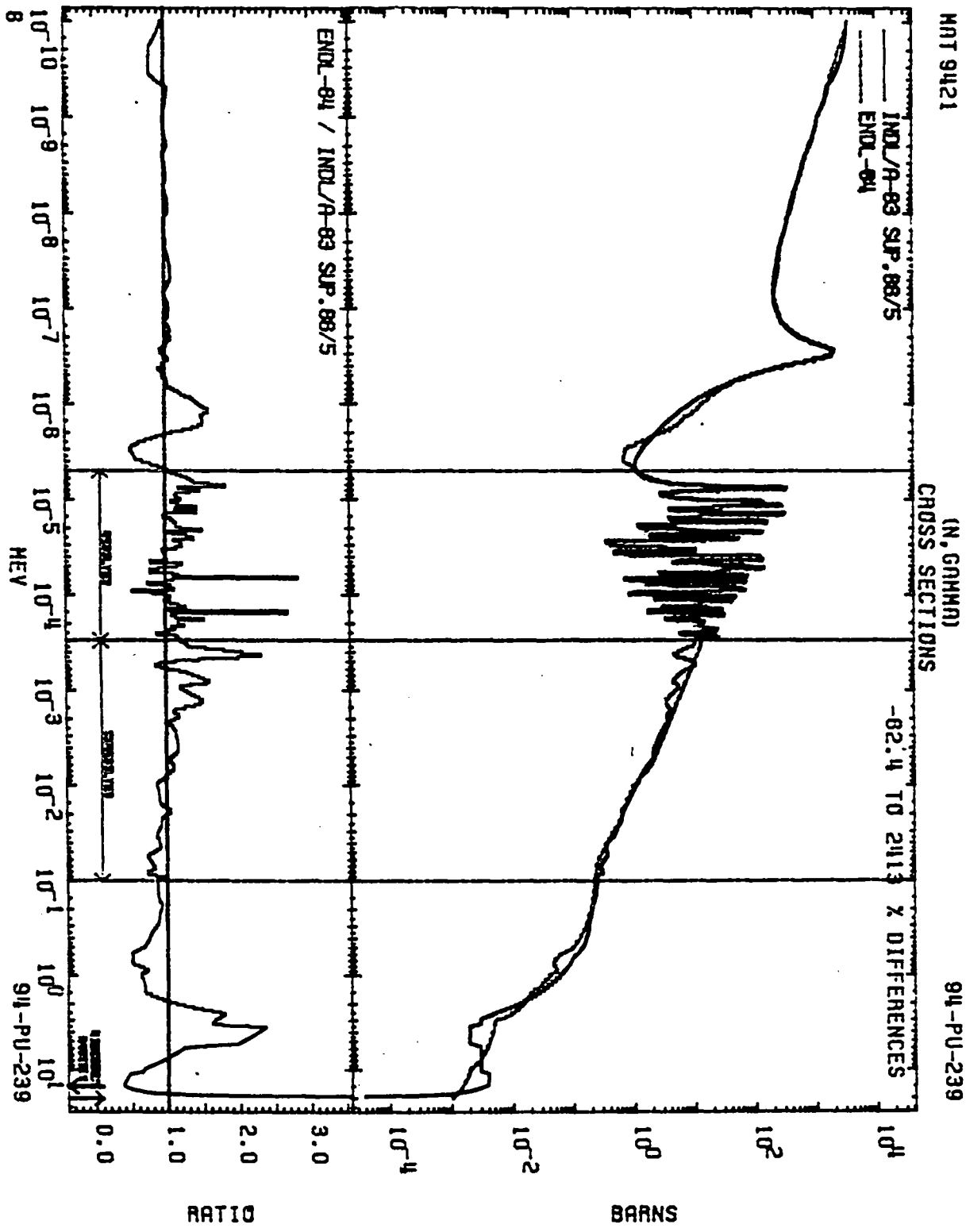


Figure 24: ^{239}Pu cross section comparison between INDL/A and ENDL-84 evaluations

C – Comparison of ^{240}Pu fine group cross sections form INDL/A with ENDL-84 data

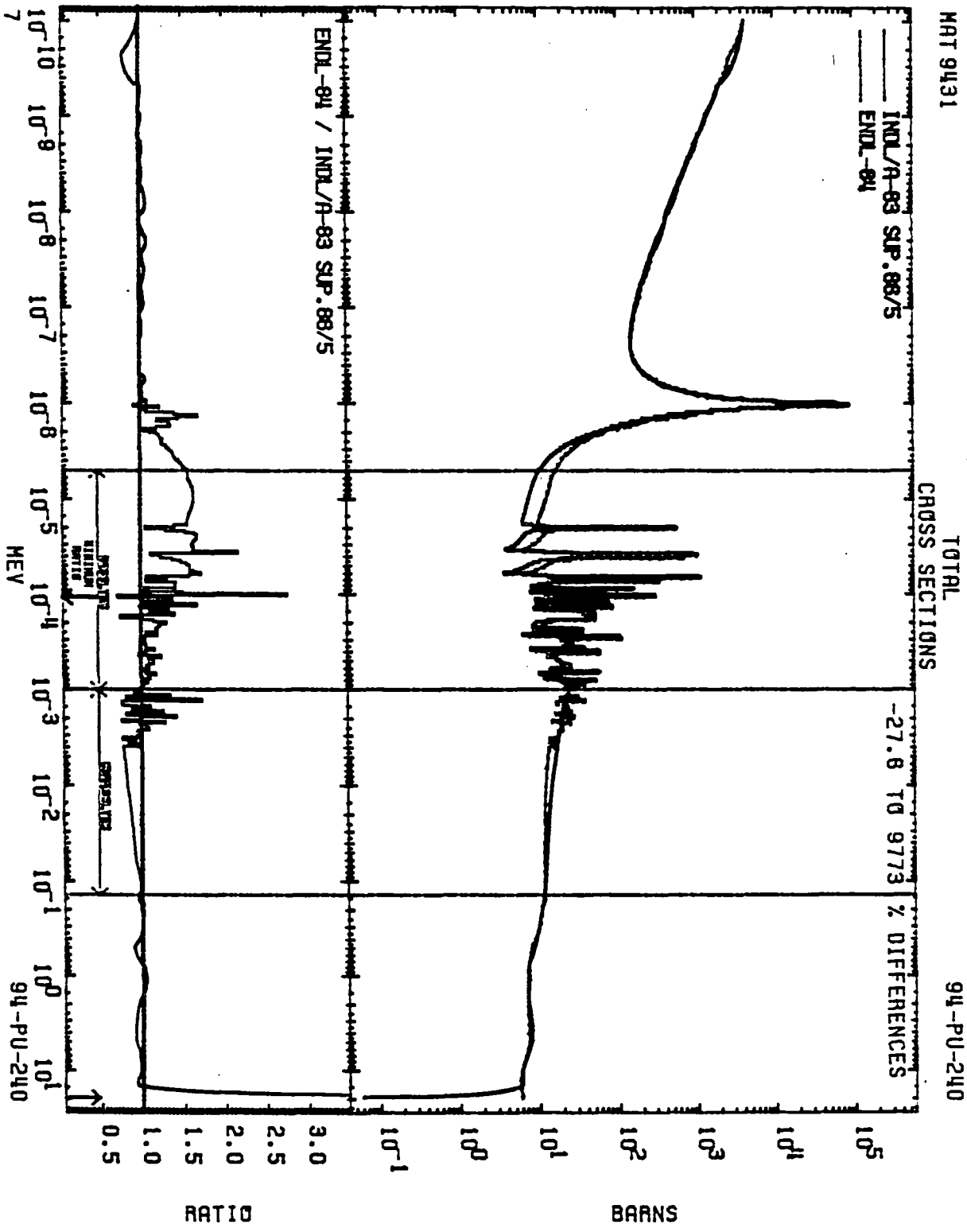


Figure 25: ^{240}Pu cross section comparison between INDL/A and ENDL-84 evaluations

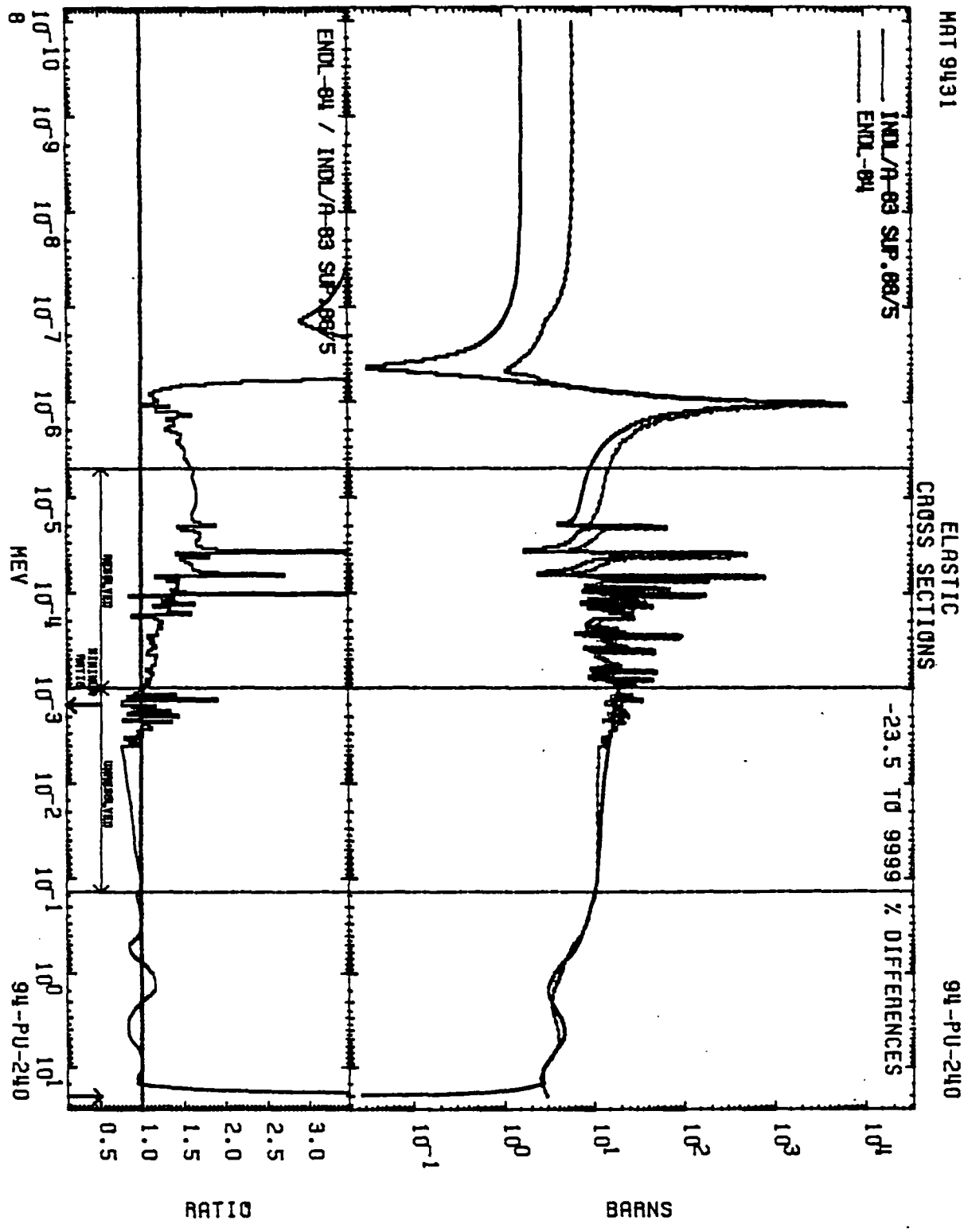


Figure 26: ^{240}Pu cross section comparison between INDL/A and ENDL-84 evaluations

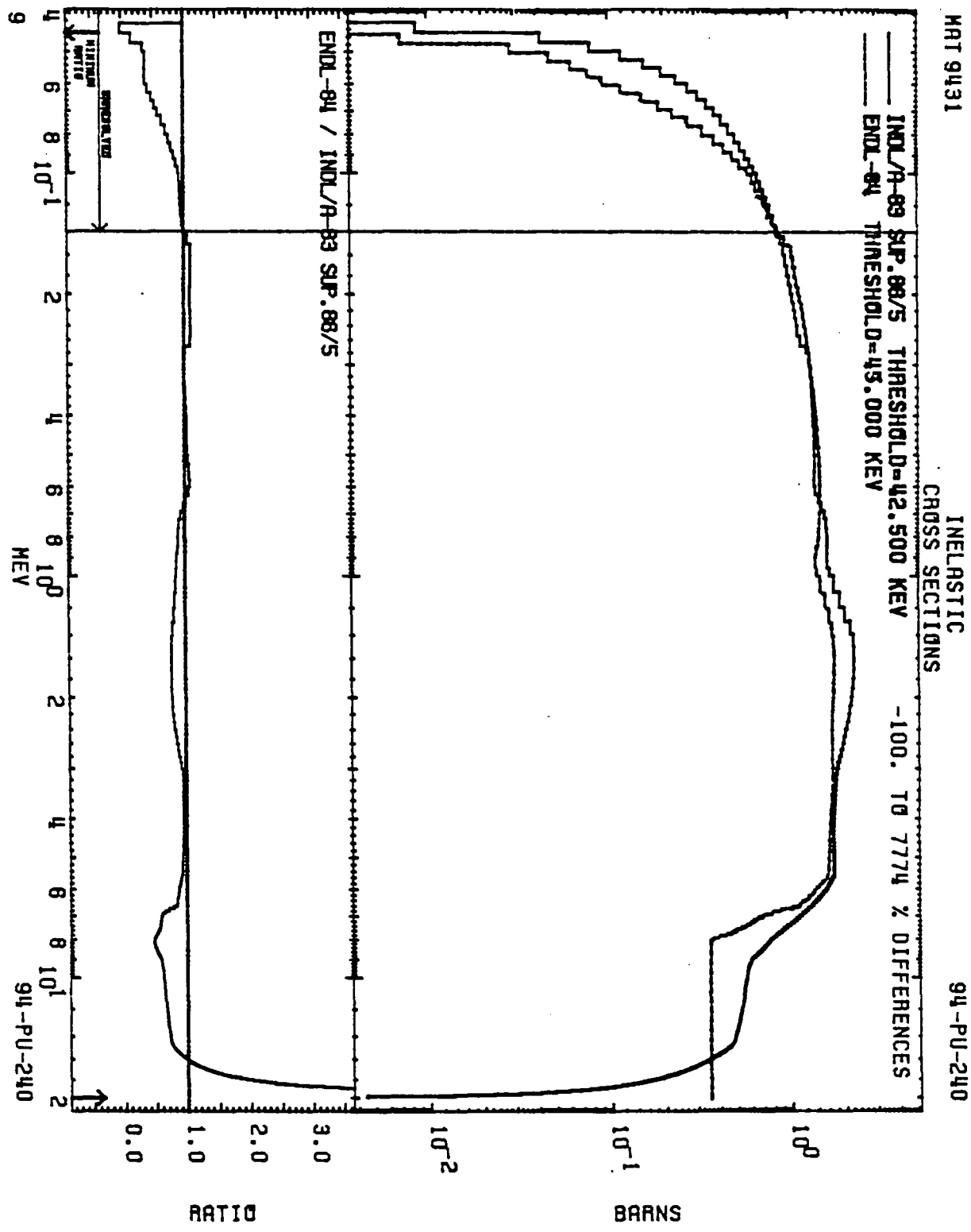


Figure 27: ^{240}Pu cross section comparison between INDL/A and ENDL-84 evaluations

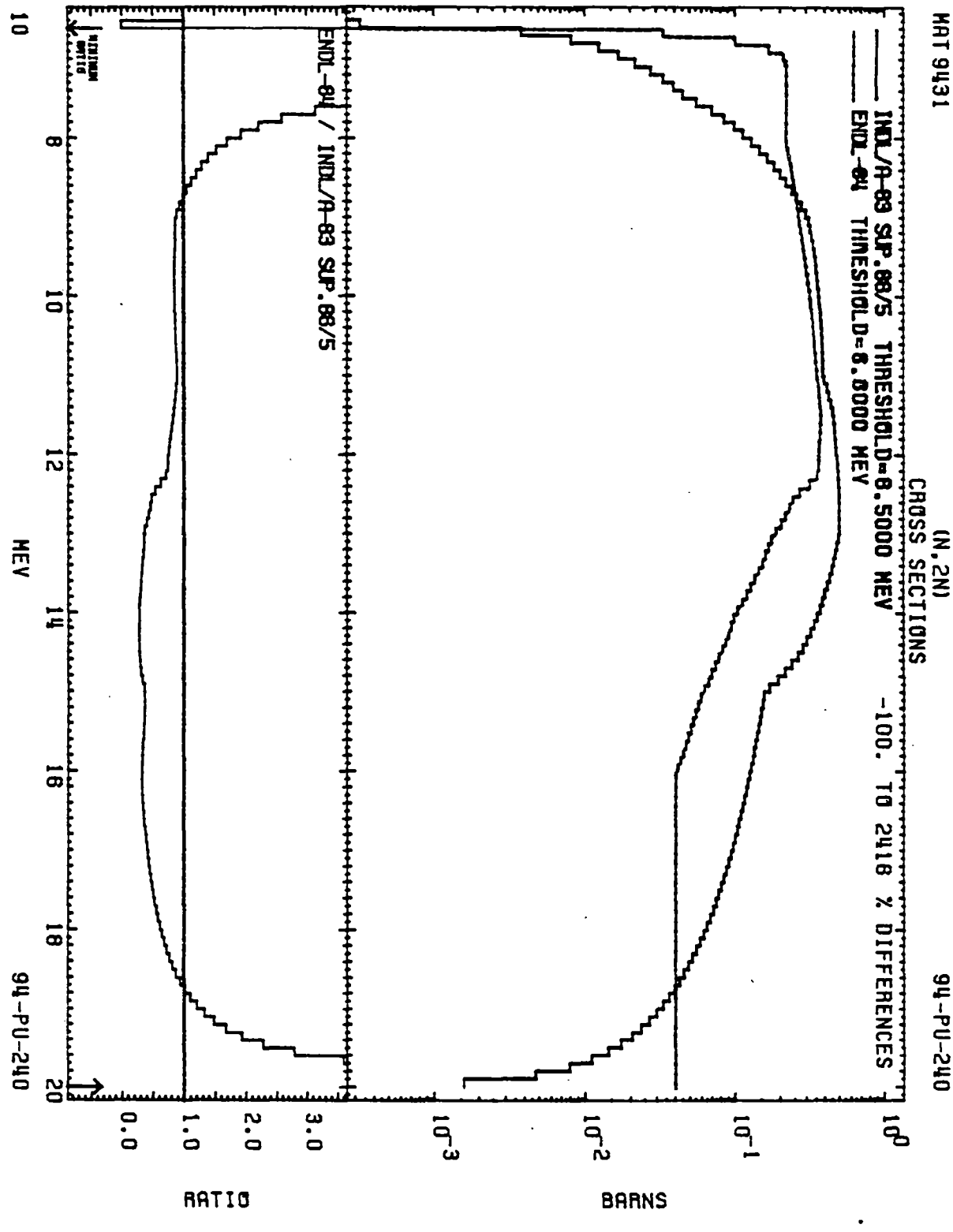
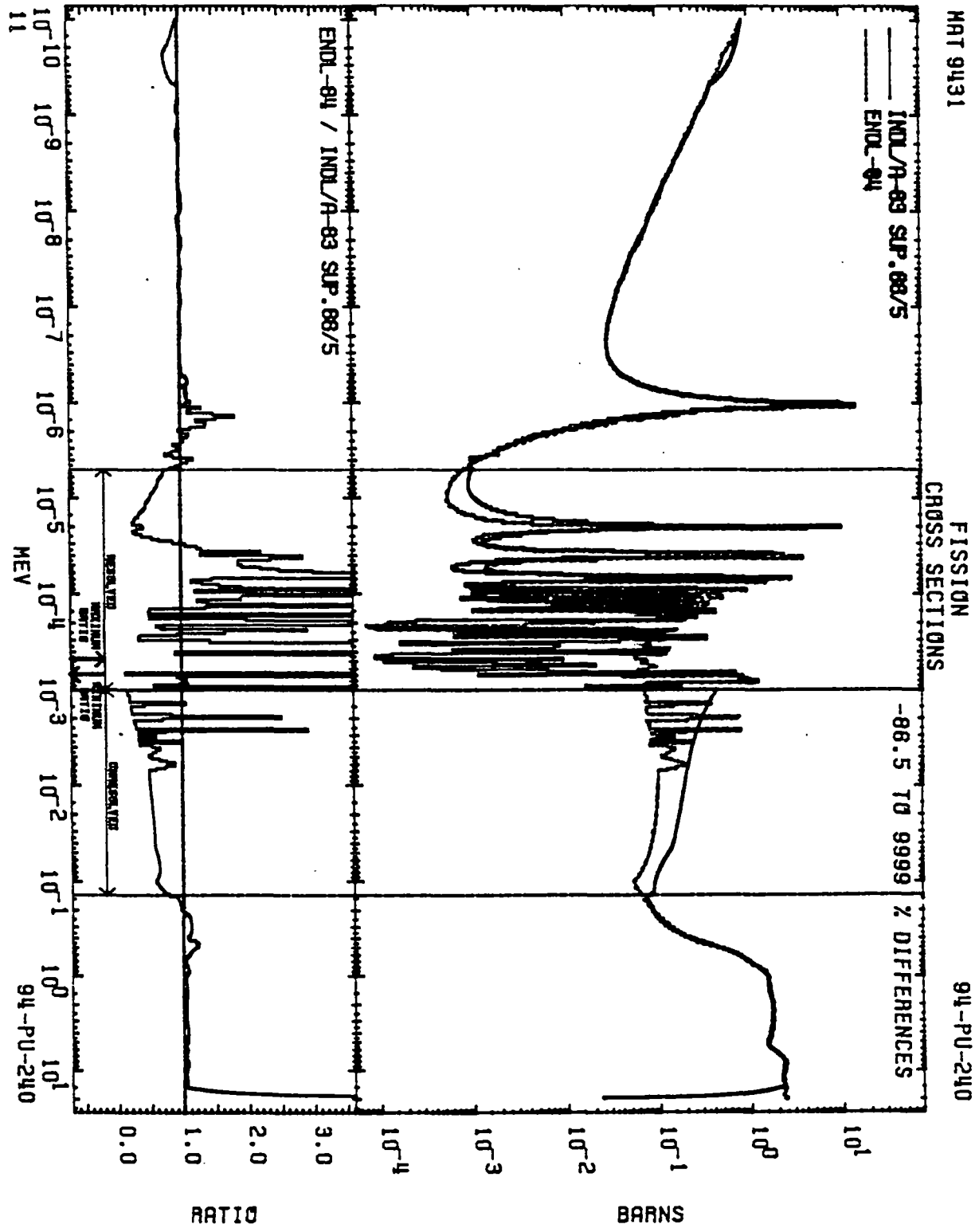


Figure 28: ^{240}Pu cross section comparison between INDL/A and ENDL-84 evaluations



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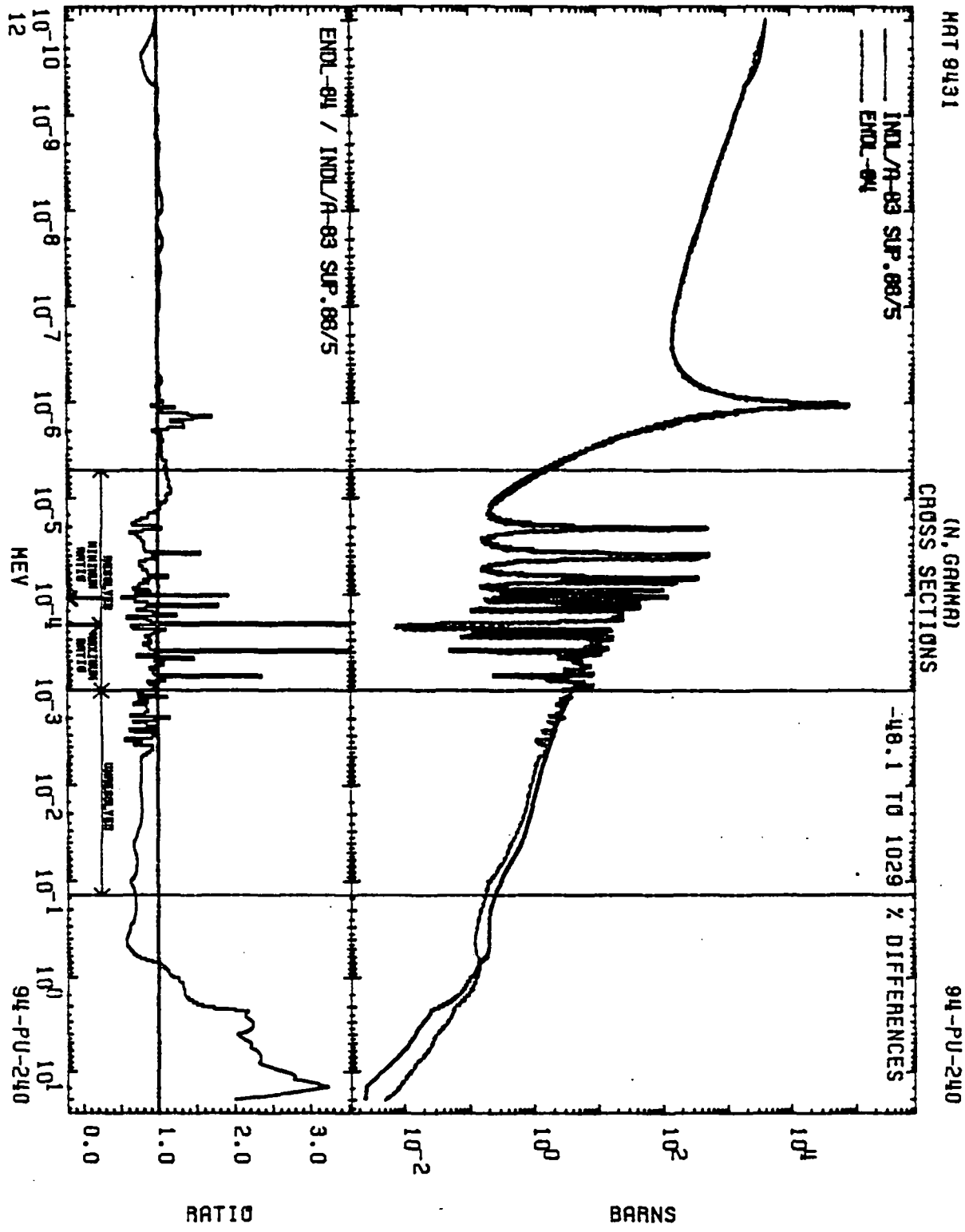


Figure 30: ^{240}Pu cross section comparison between INDL/A and ENDL-84 evaluations

D – Comparison of ^{241}Pu fine group cross sections form INDL/A with ENDL-84 data

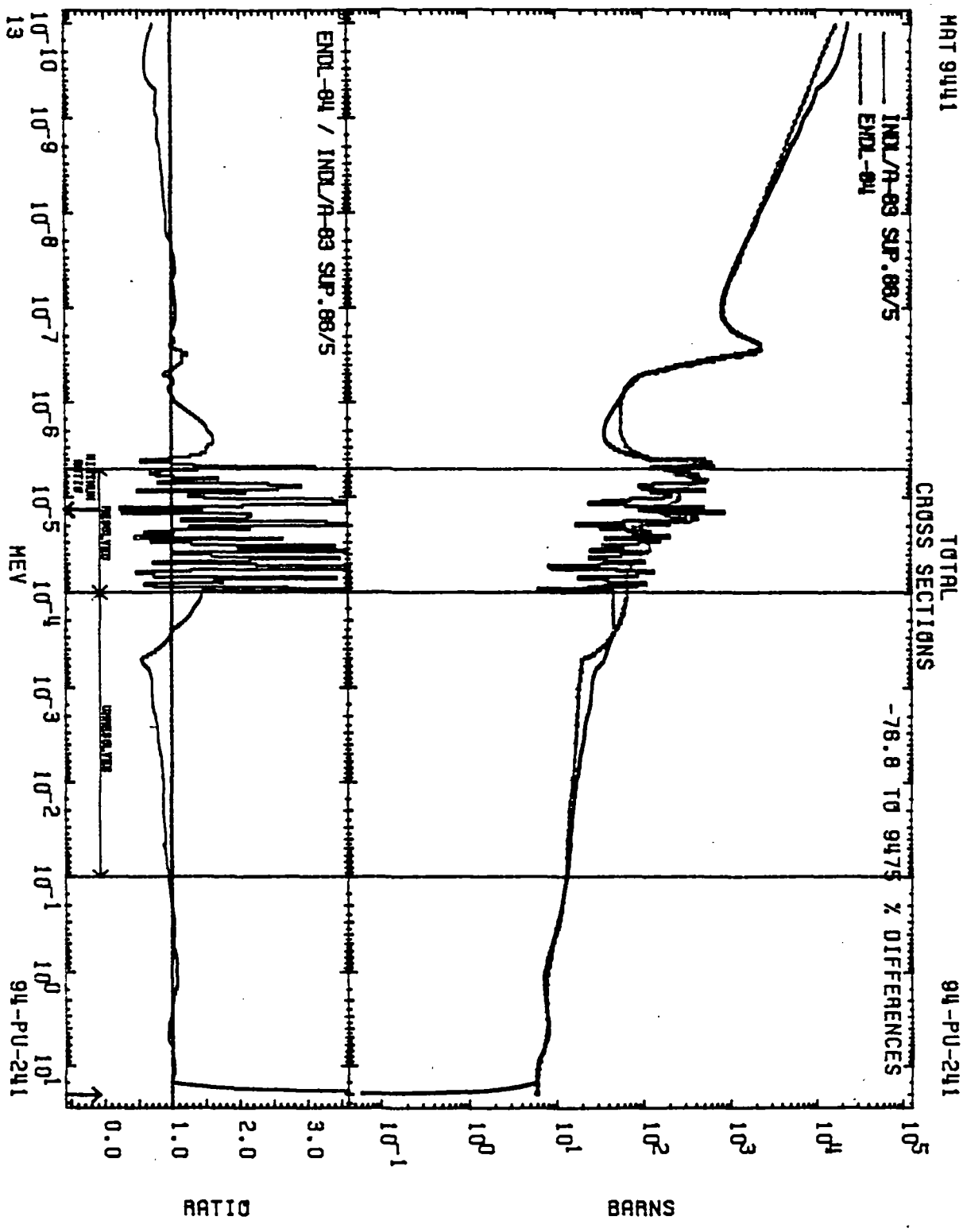


Figure 31: ^{241}Pu cross section comparison between INDL/A and ENDL-84 evaluations

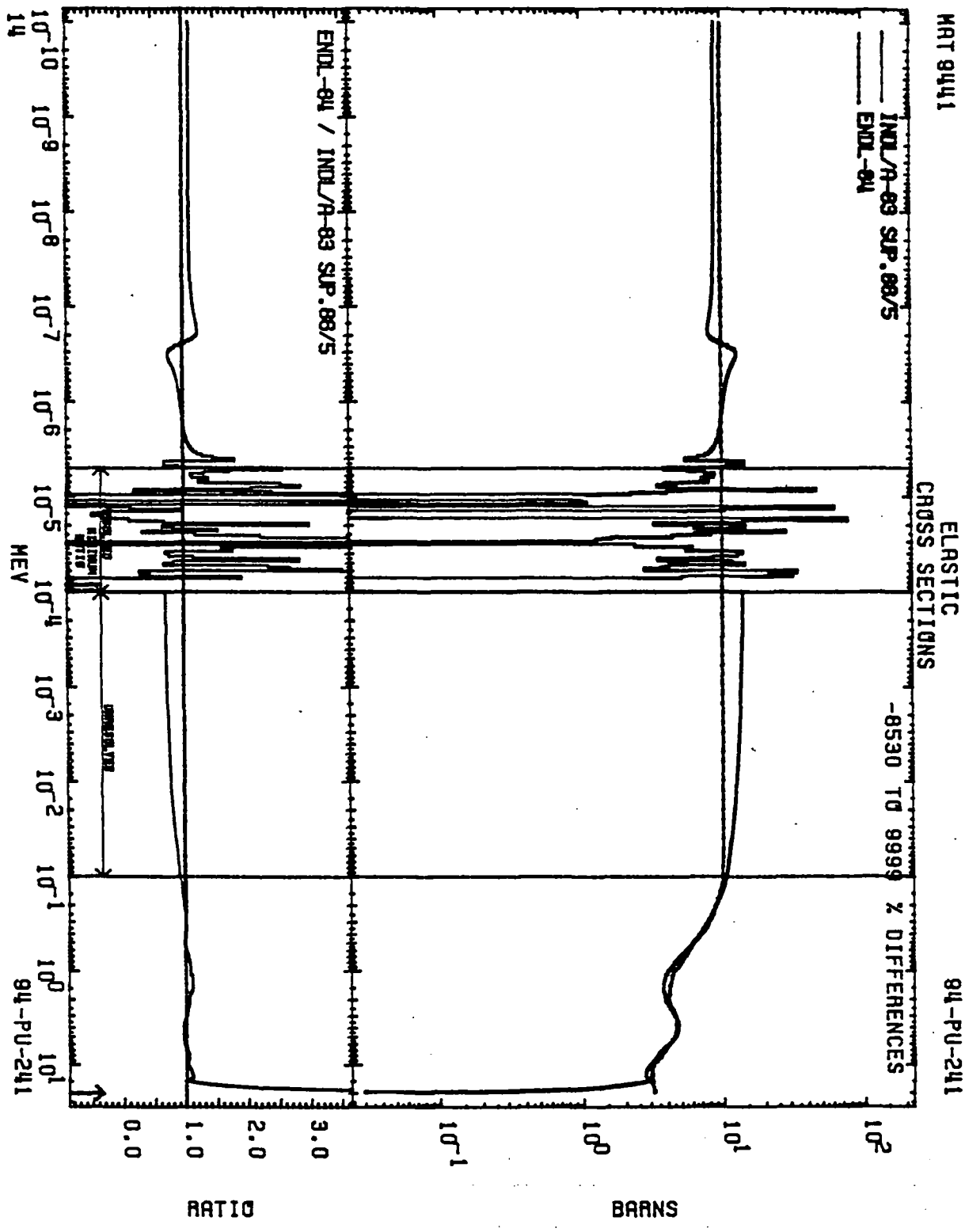
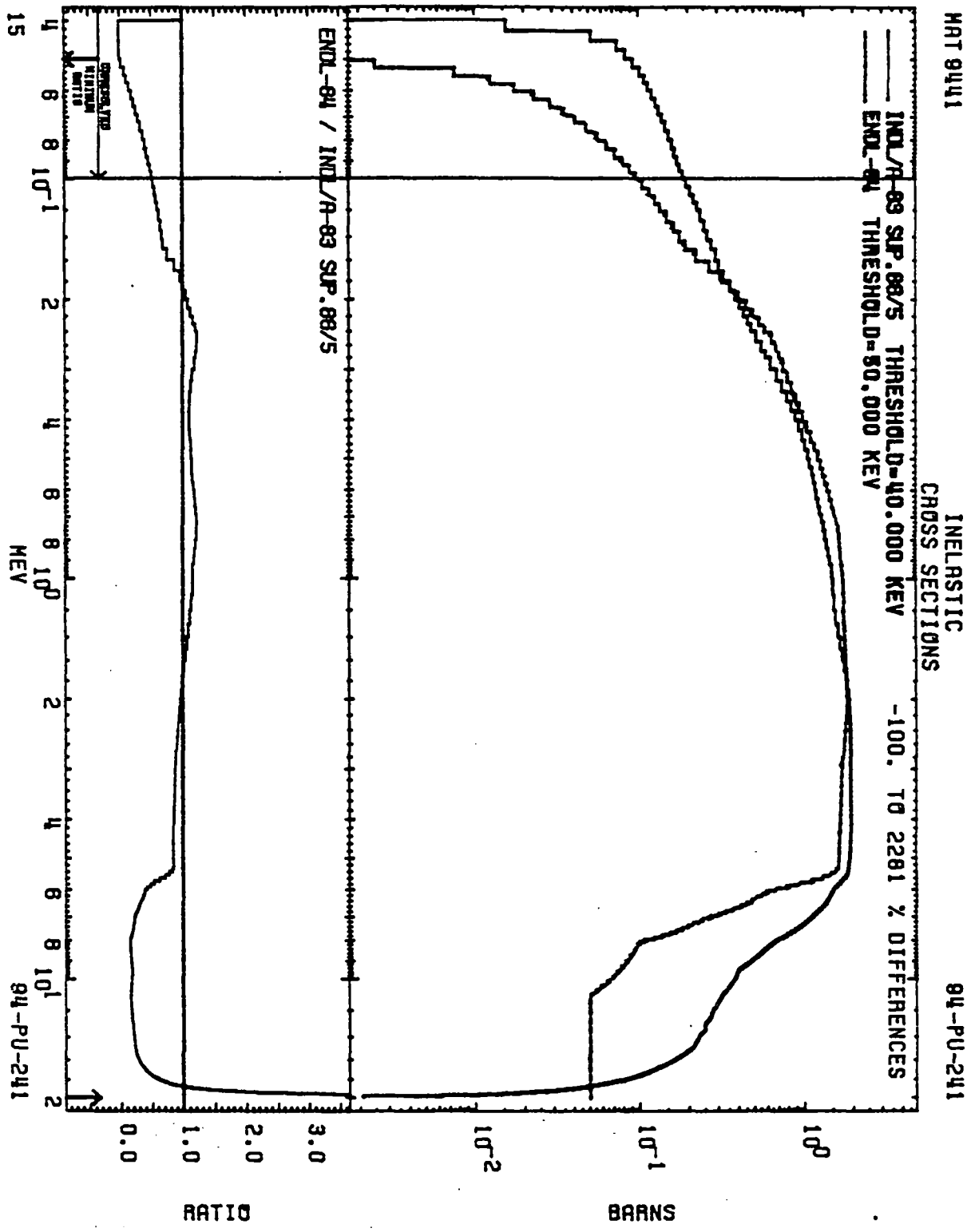


Figure 32: ^{241}Pu cross section comparison between INDL/A and ENDL-84 evaluations



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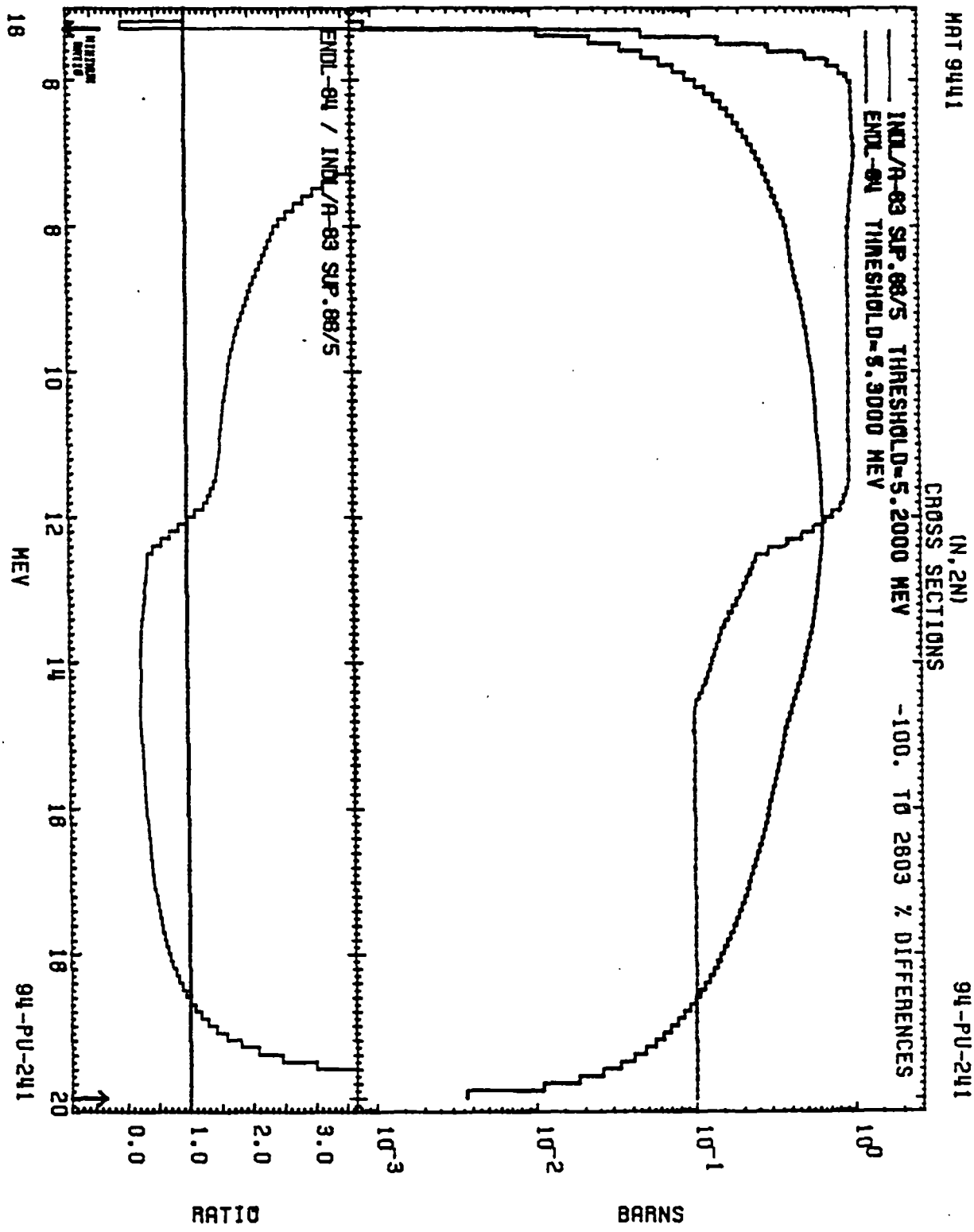

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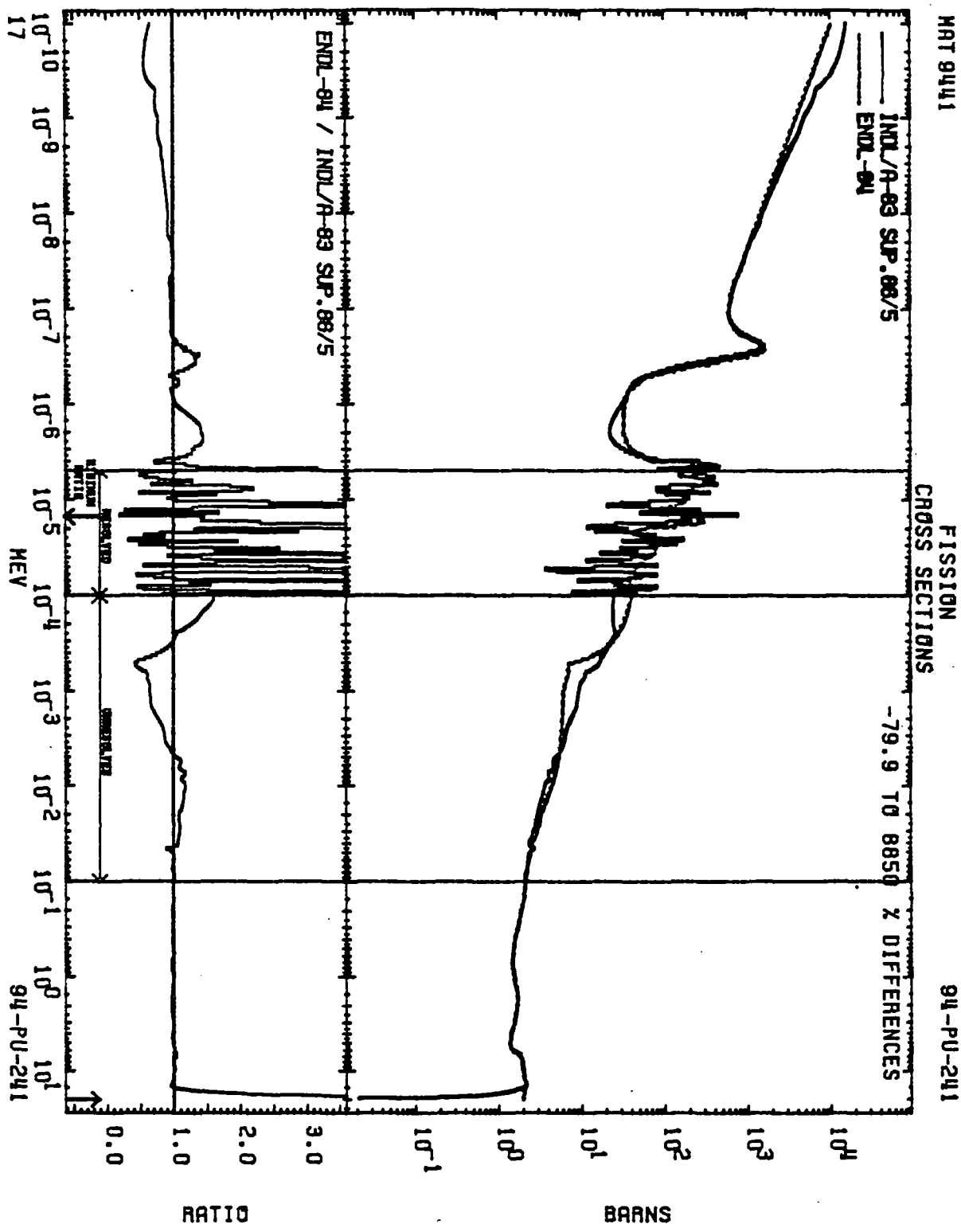

Figure 35: ^{241}Pu cross section comparison between INDL/A and ENDL-84 evaluations

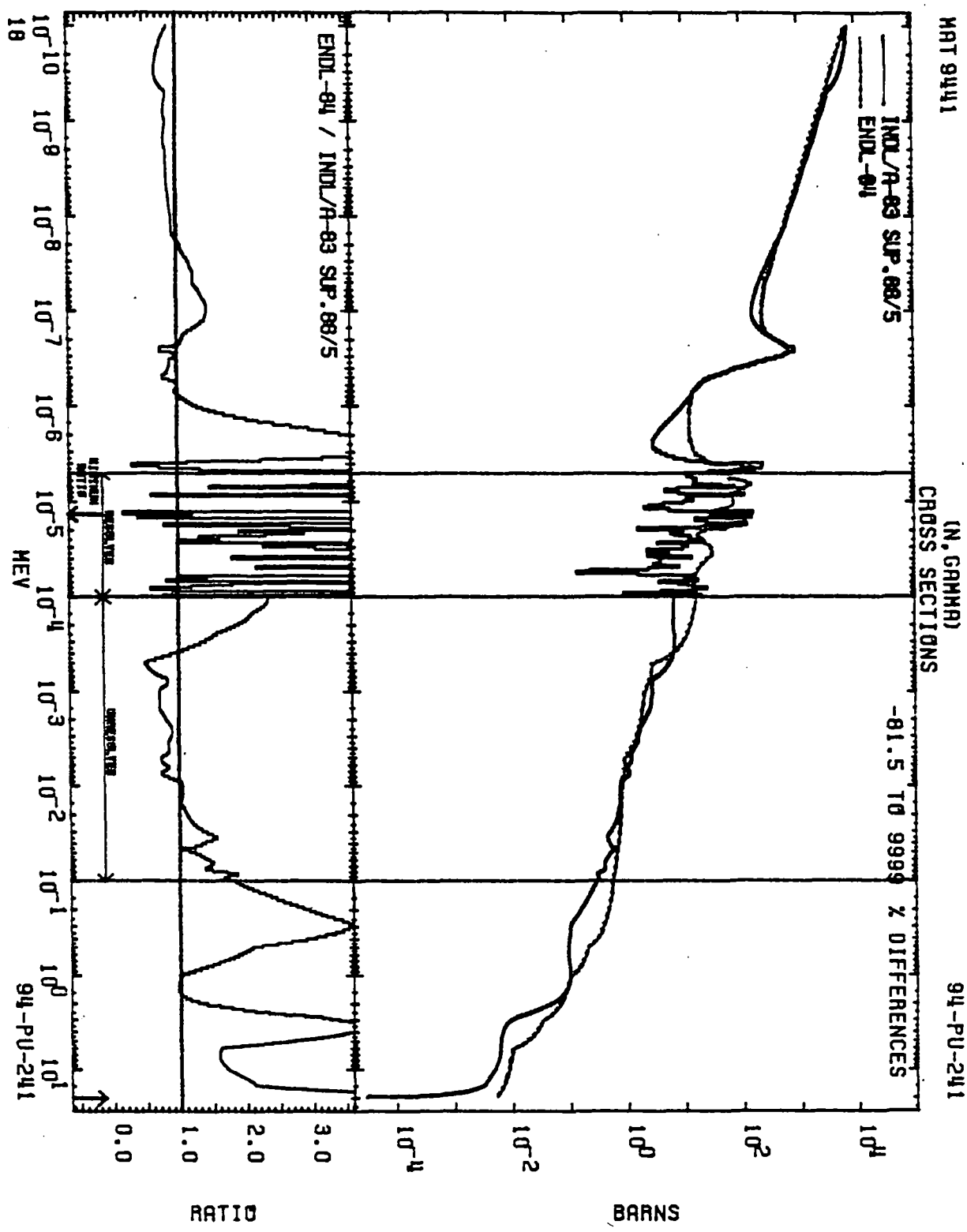

Figure 36: ^{241}Pu cross section comparison between INDL/A and ENDL-84 evaluations

E – Comparison of ^{242}Pu fine group cross sections form INDL/A with ENDF/B-5 and ENDL-84 data

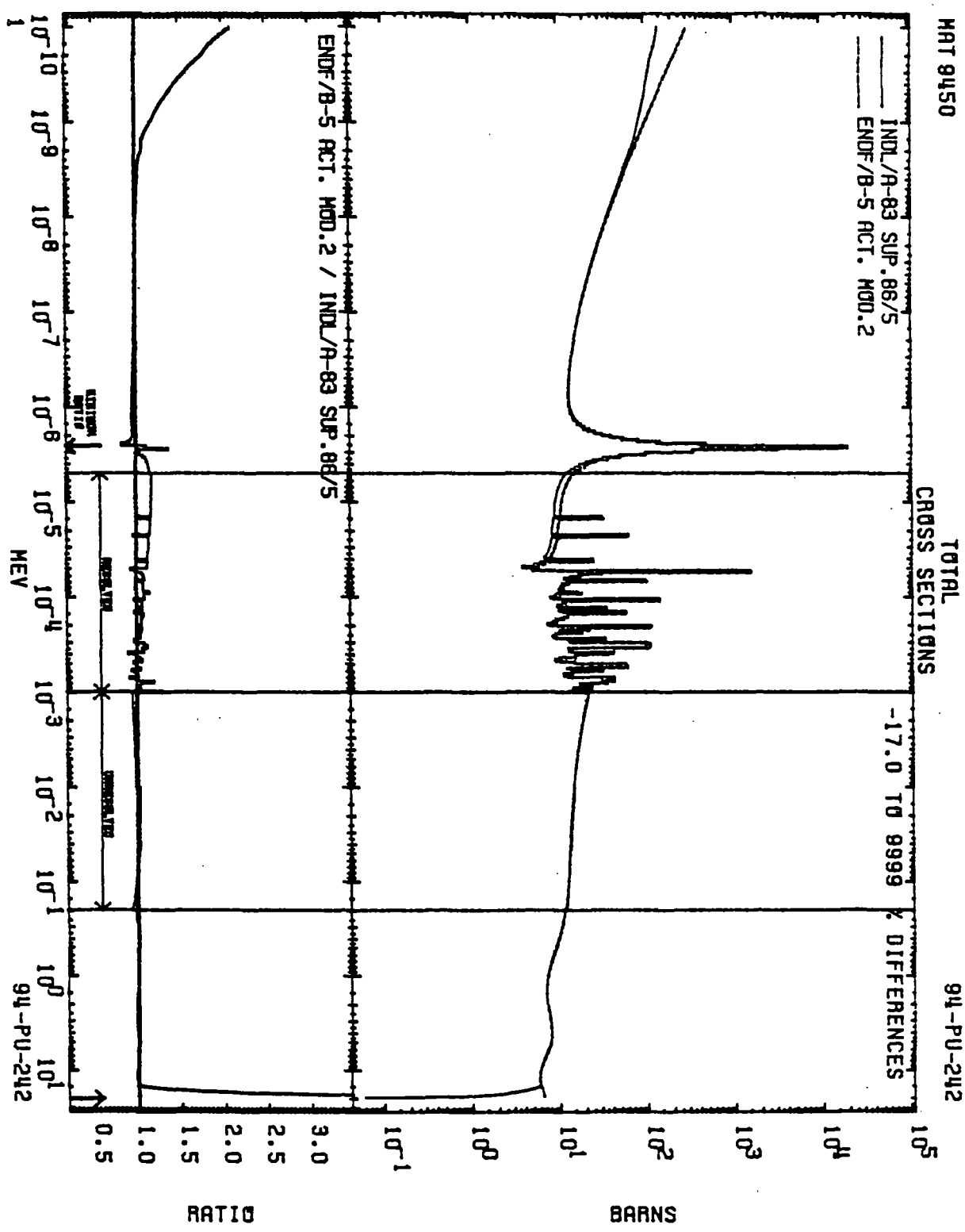

Figure 37: ^{242}Pu cross section comparison between INDL/A and ENDF/B-5 evaluations

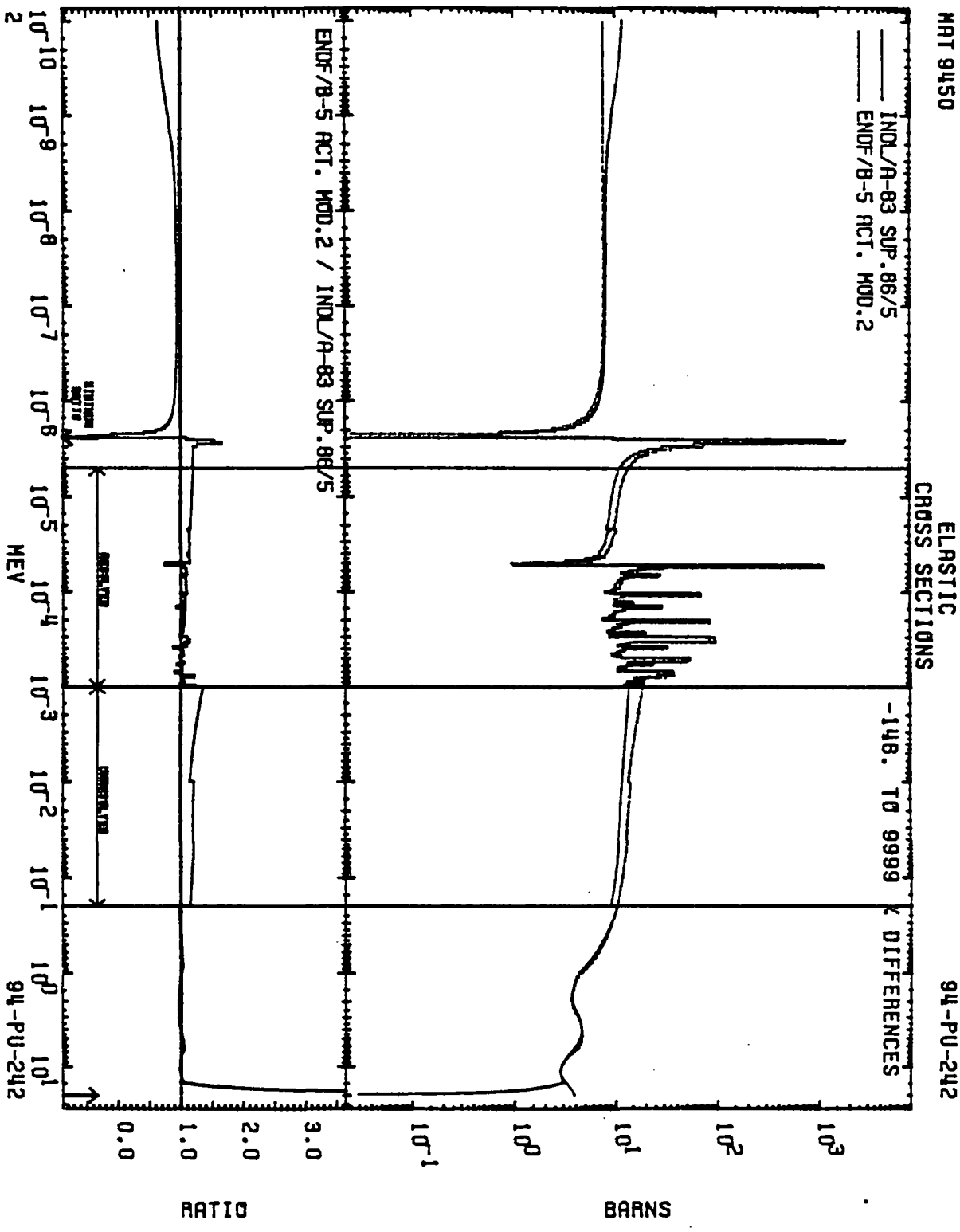

Figure 38: ^{242}Pu cross section comparison between INDL/A and ENDF/B-5 evaluations

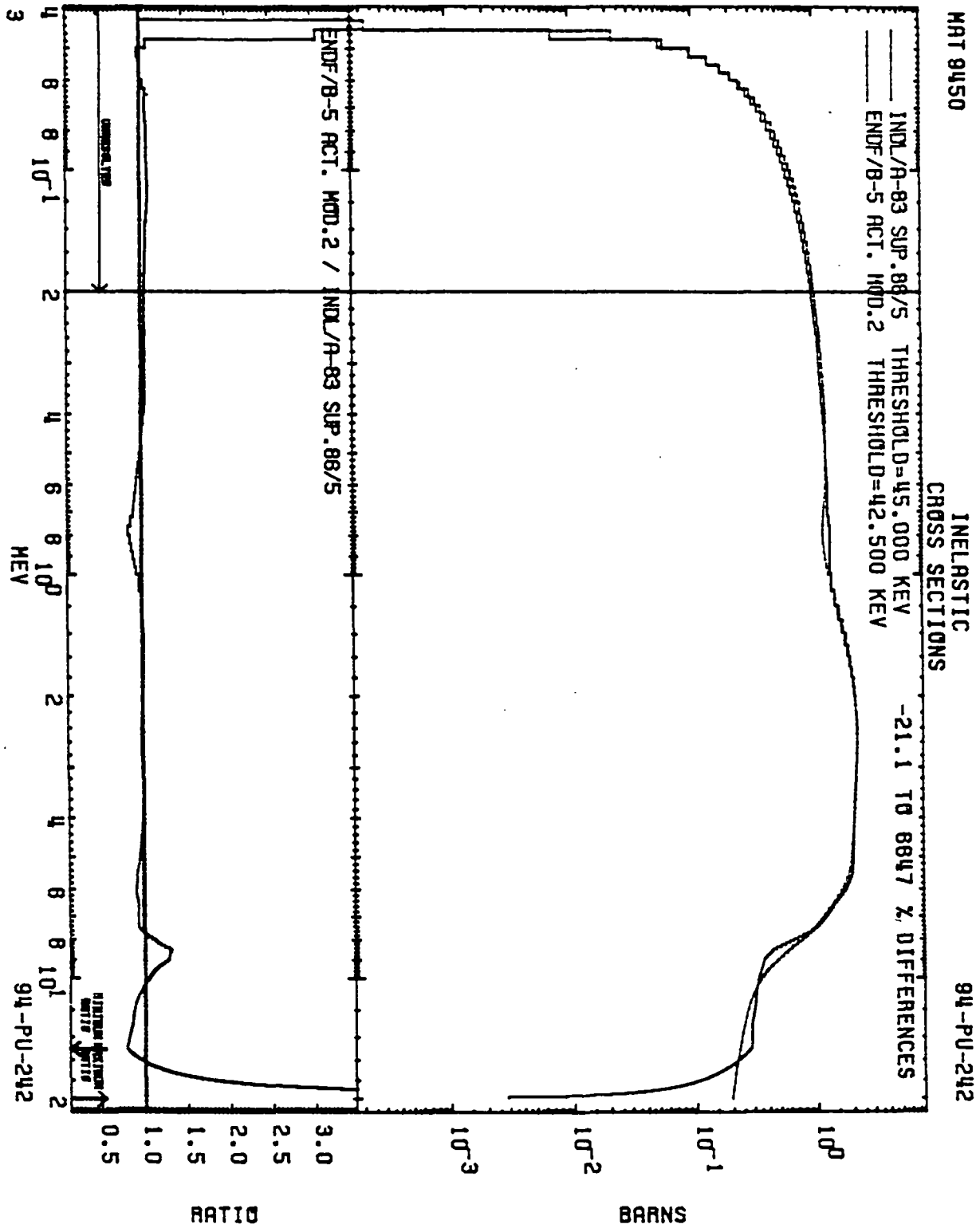

Figure 39: ^{242}Pu cross section comparison between INDL/A and ENDF/B-5 evaluations

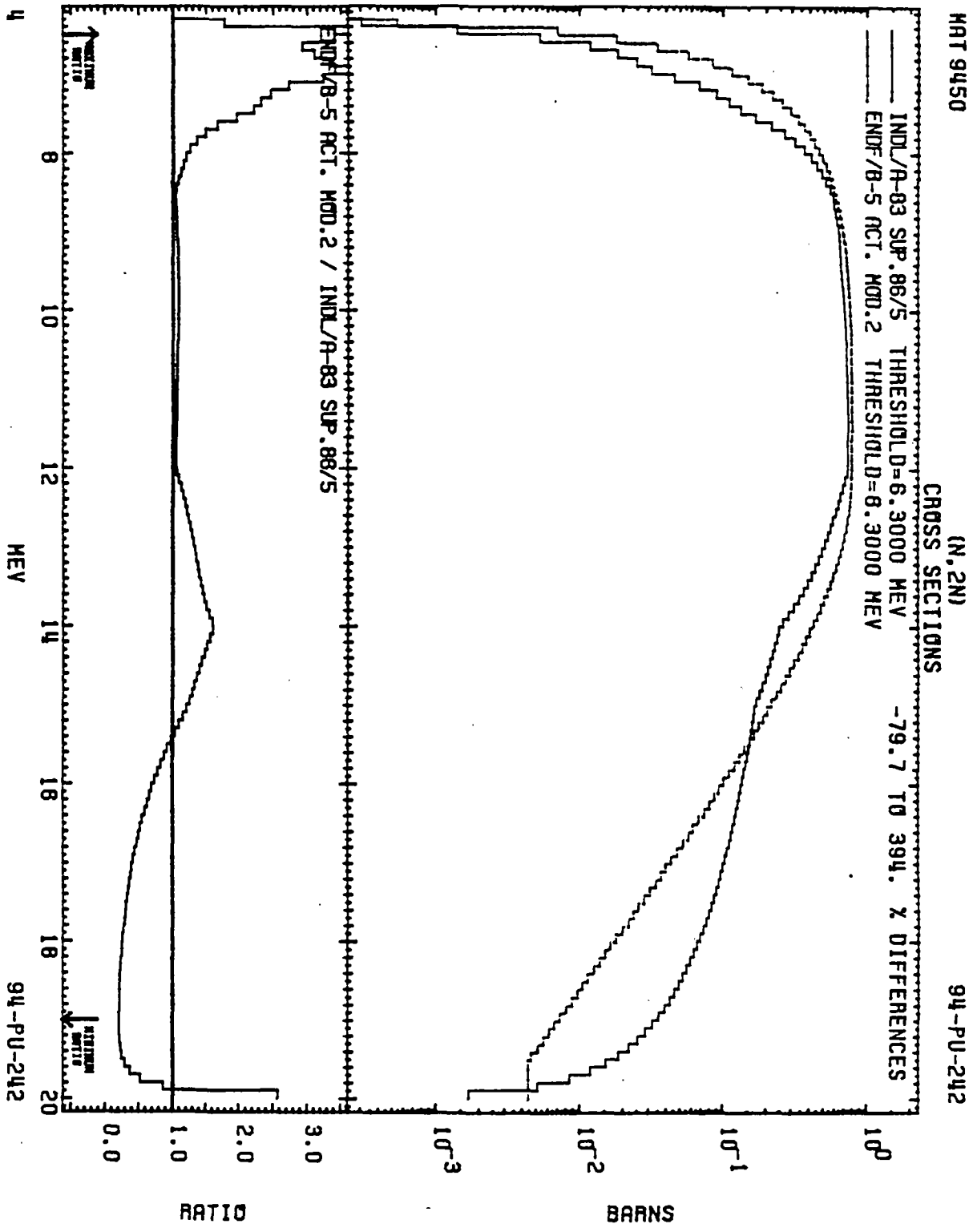

Figure 40: ^{242}Pu cross section comparison between INDL/A and ENDF/B-5 evaluations

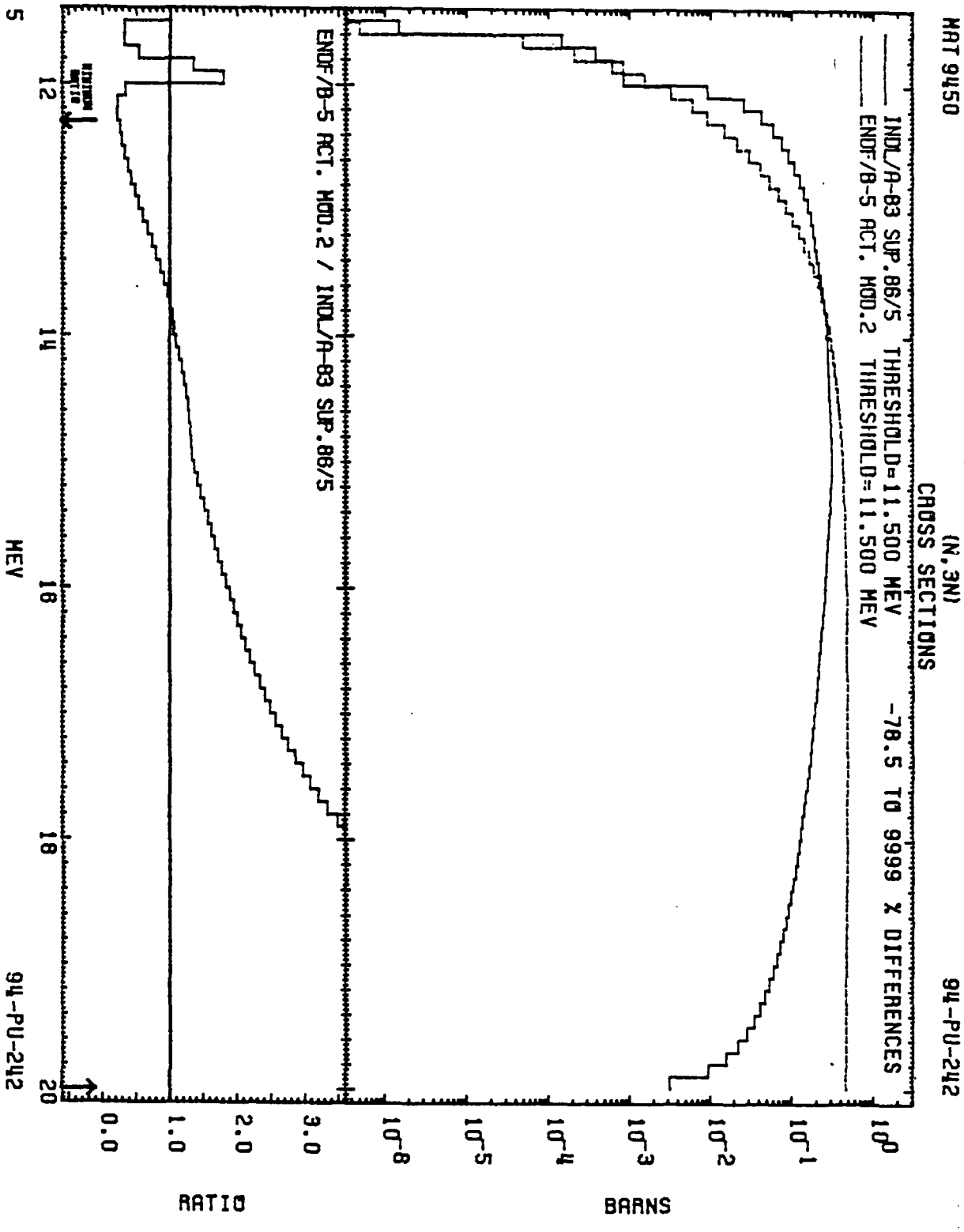

Figure 41: ^{242}Pu cross section comparison between INDL/A and ENDF/B-5 evaluations

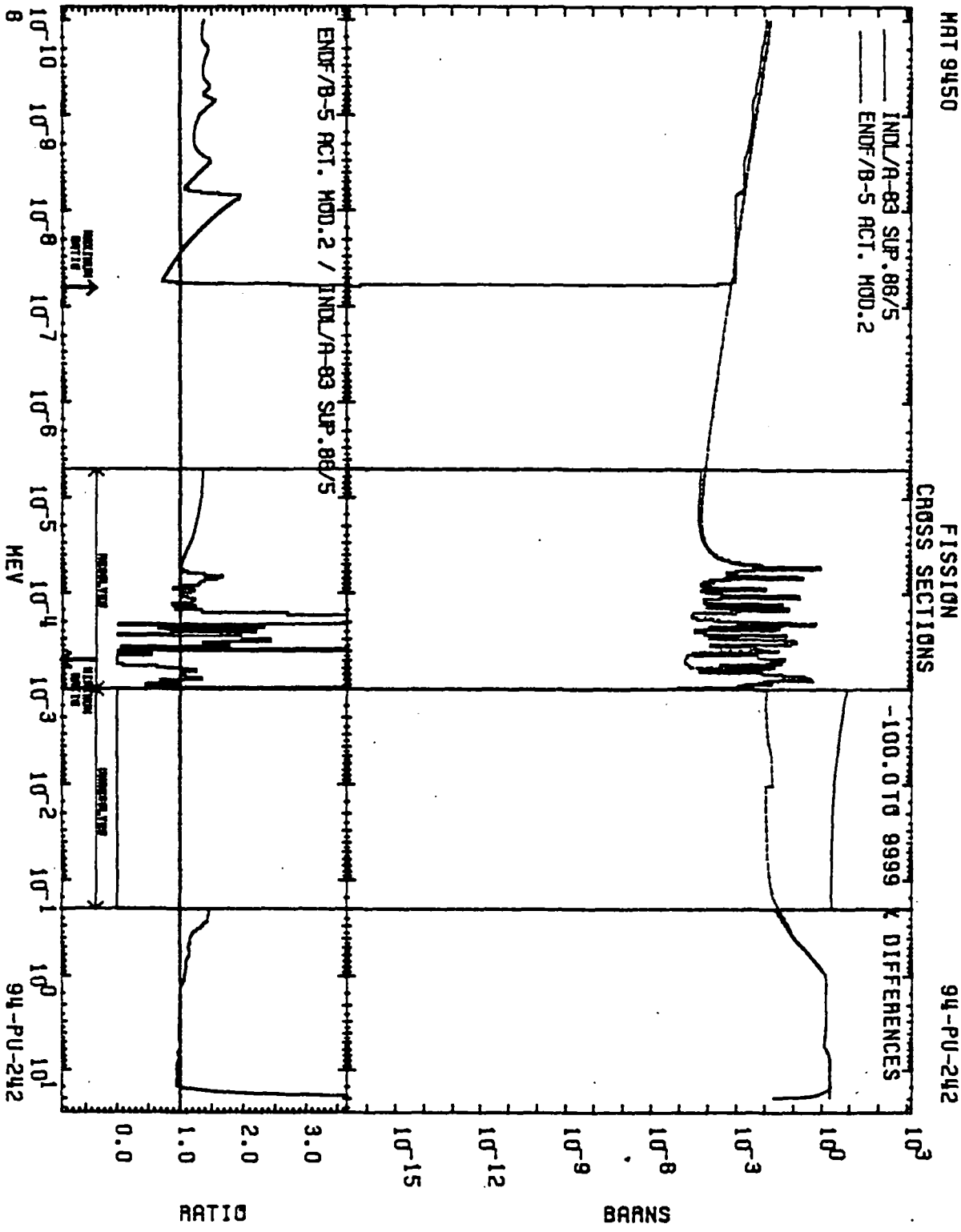

Figure 42: ^{242}Pu cross section comparison between INDL/A and ENDF/B-5 evaluations

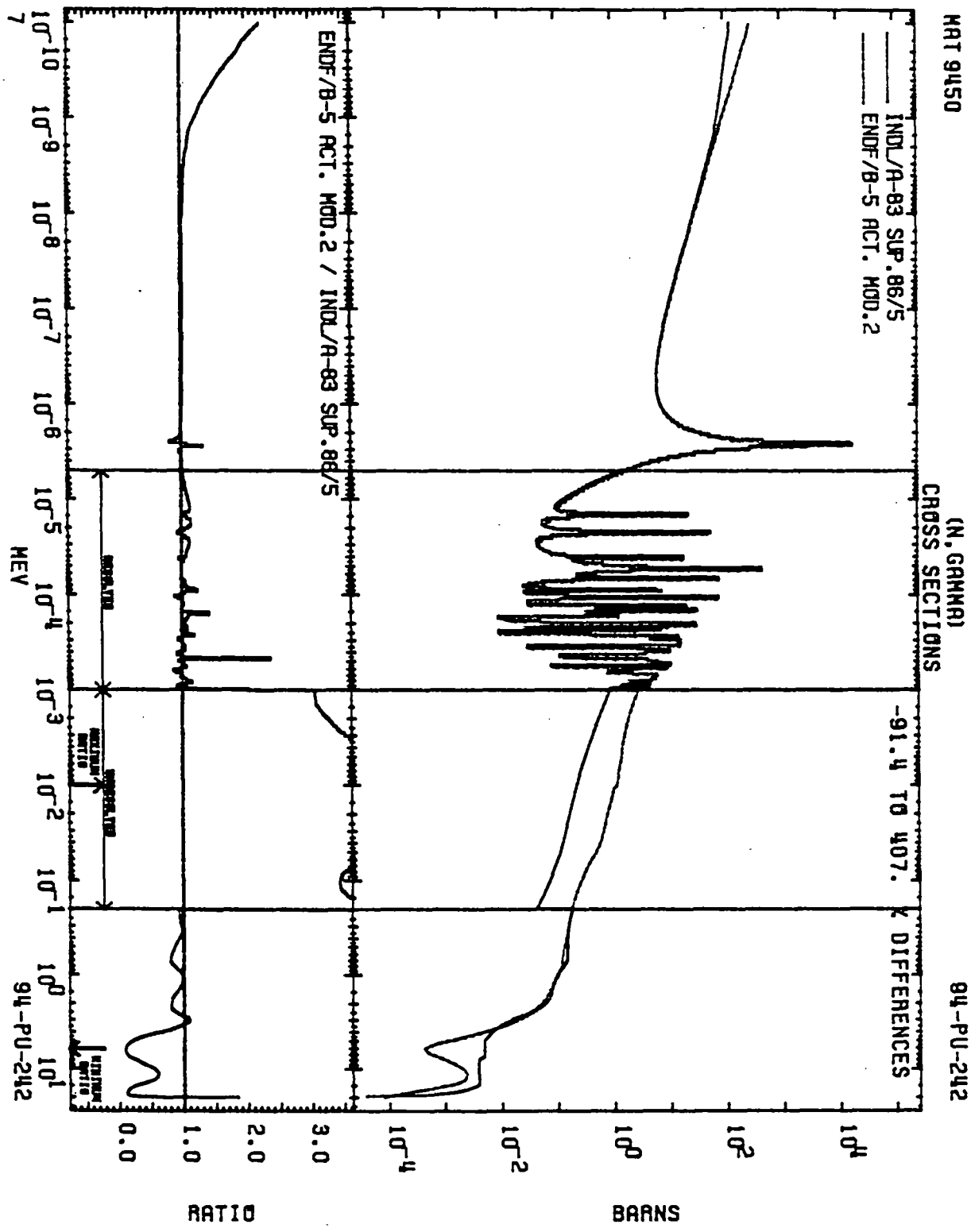

Figure 43: ^{242}Pu cross section comparison between INDL/A and ENDF/B-5 evaluations

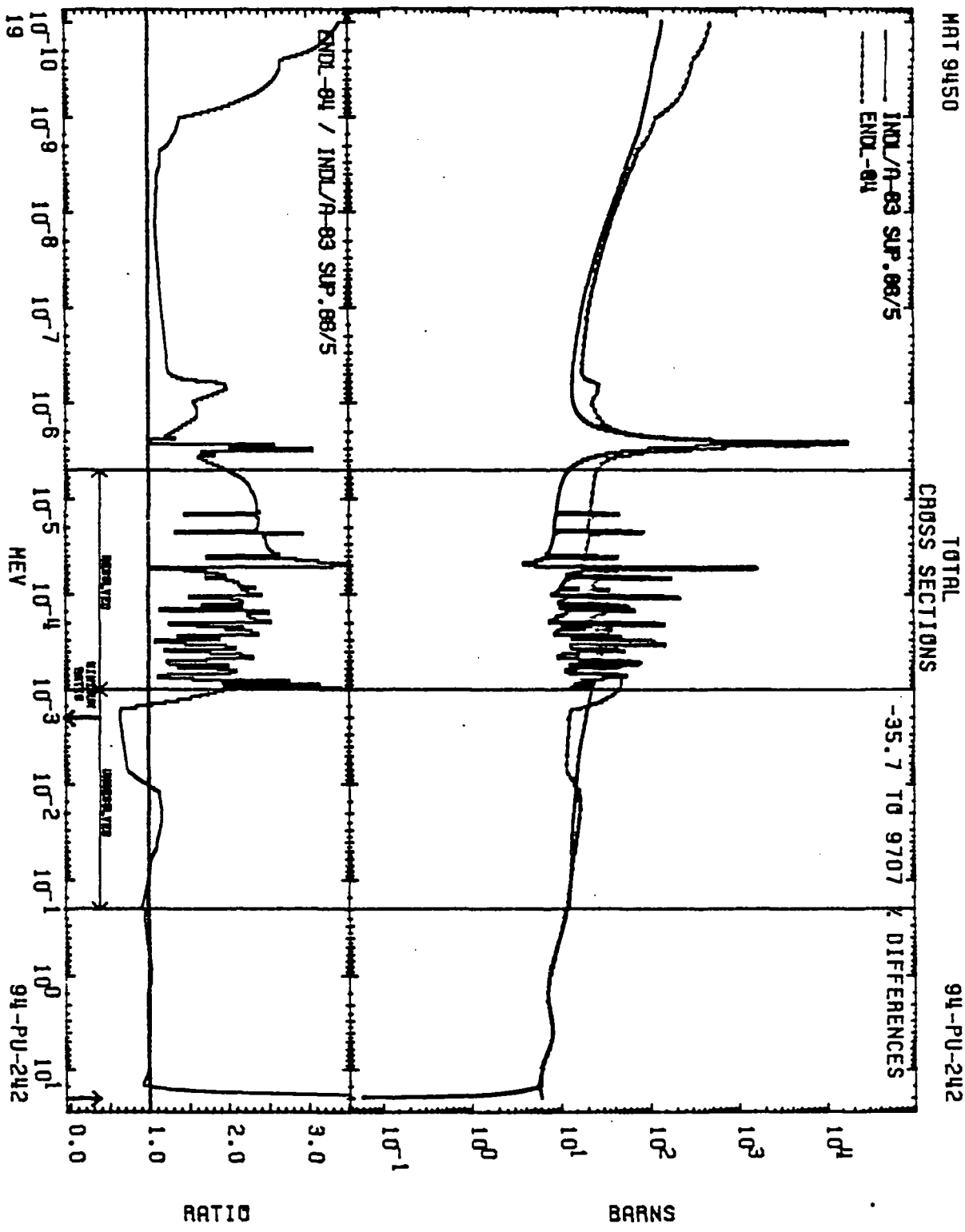

Figure 44: ^{242}Pu cross section comparison between INDL/A and ENDL-84 evaluations

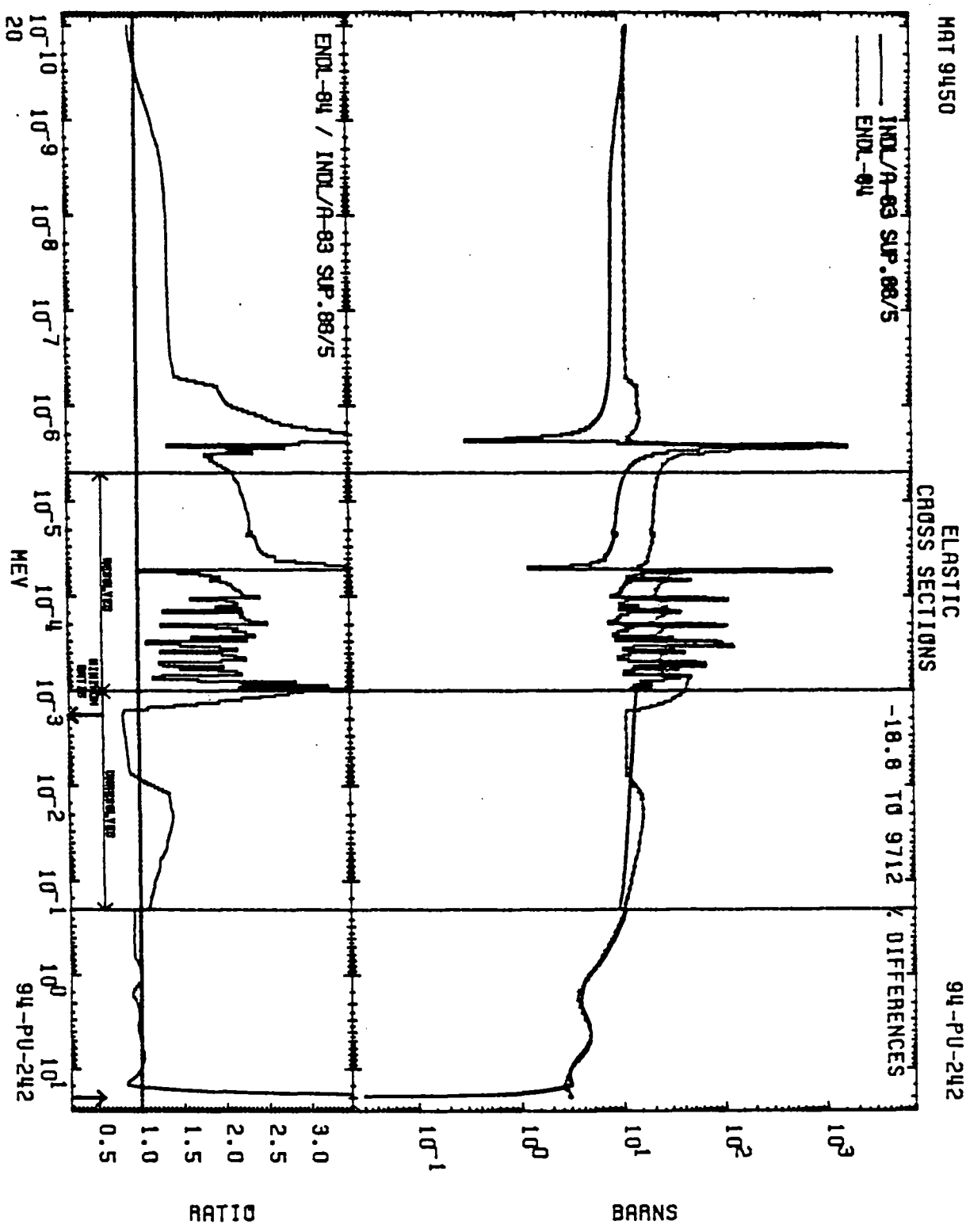

Figure 45: ^{242}Pu cross section comparison between INDL/A and ENDL-84 evaluations

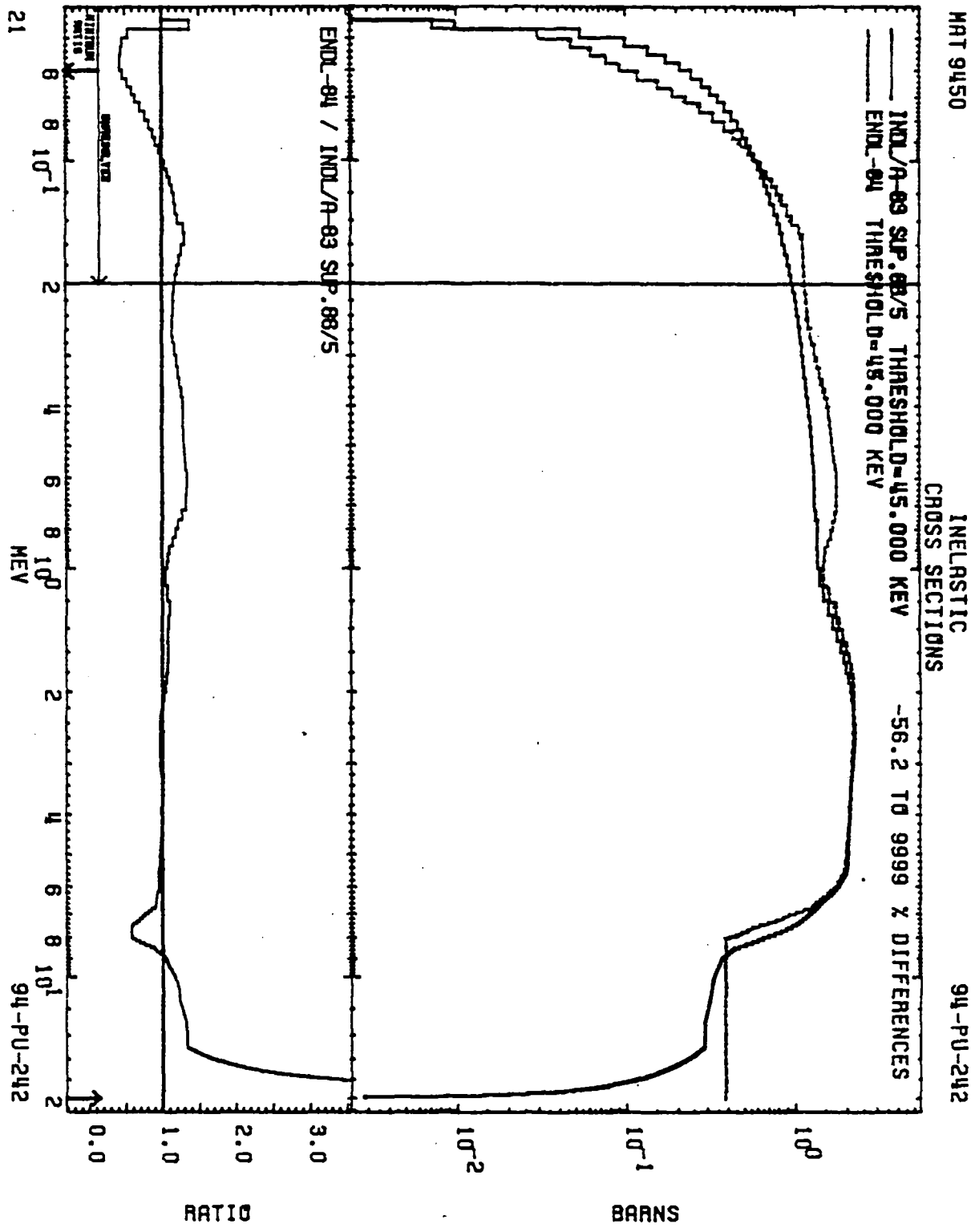

Figure 46: ^{242}Pu cross section comparison between INDL/A and ENDL-84 evaluations

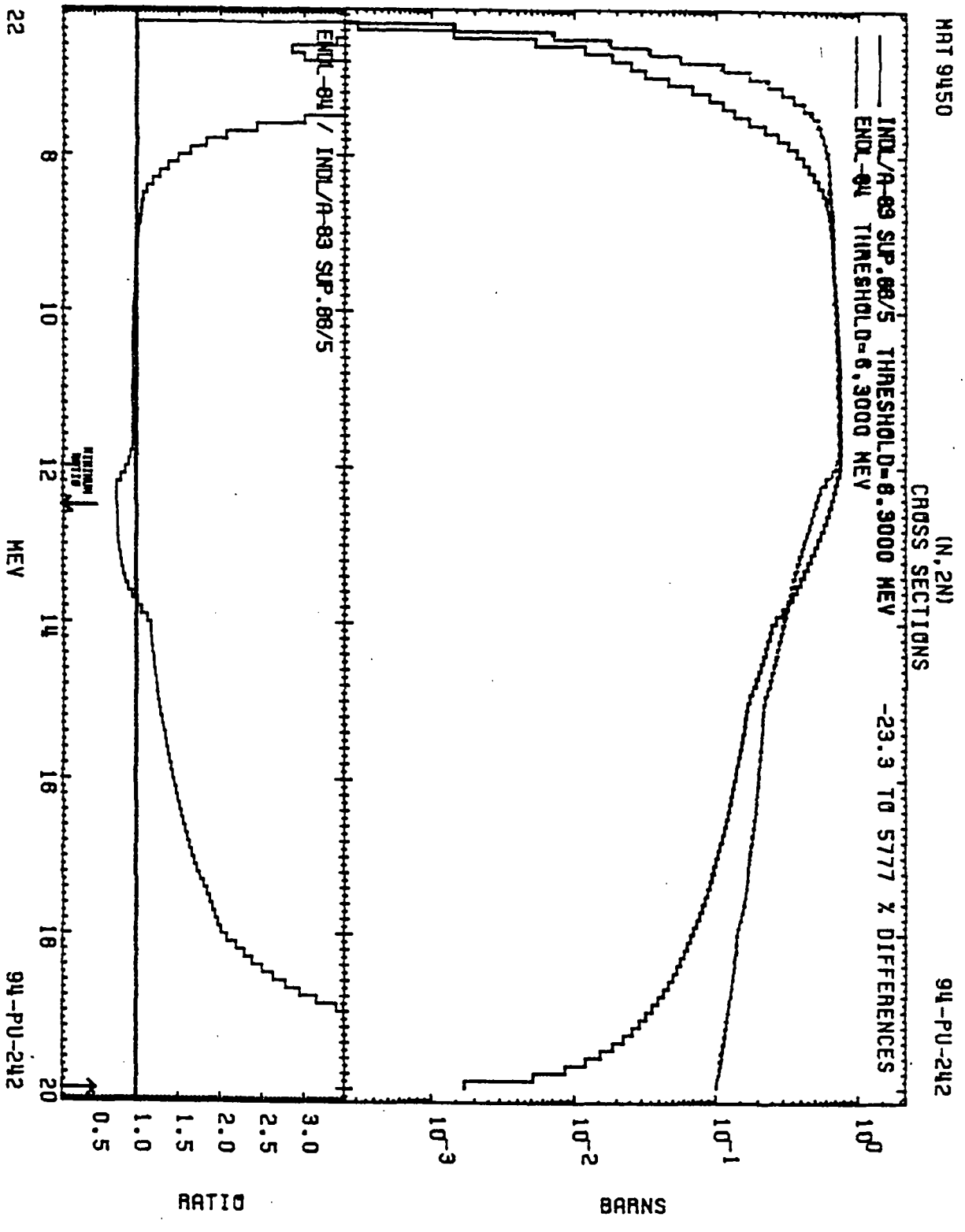

Figure 47: ^{242}Pu cross section comparison between INDL/A and ENDL-84 evaluations

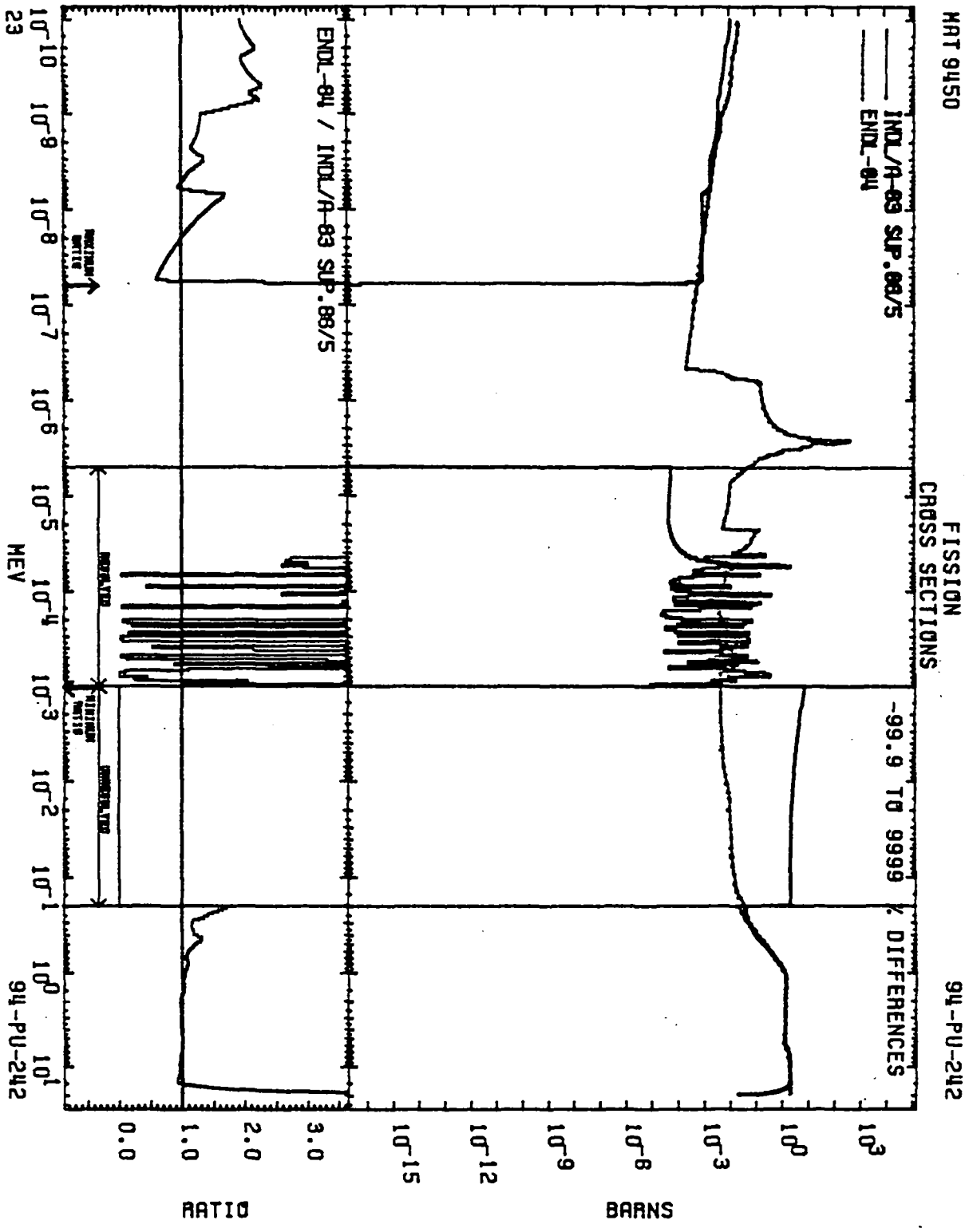

Figure 48: ^{242}Pu cross section comparison between INDL/A and ENDL-84 evaluations

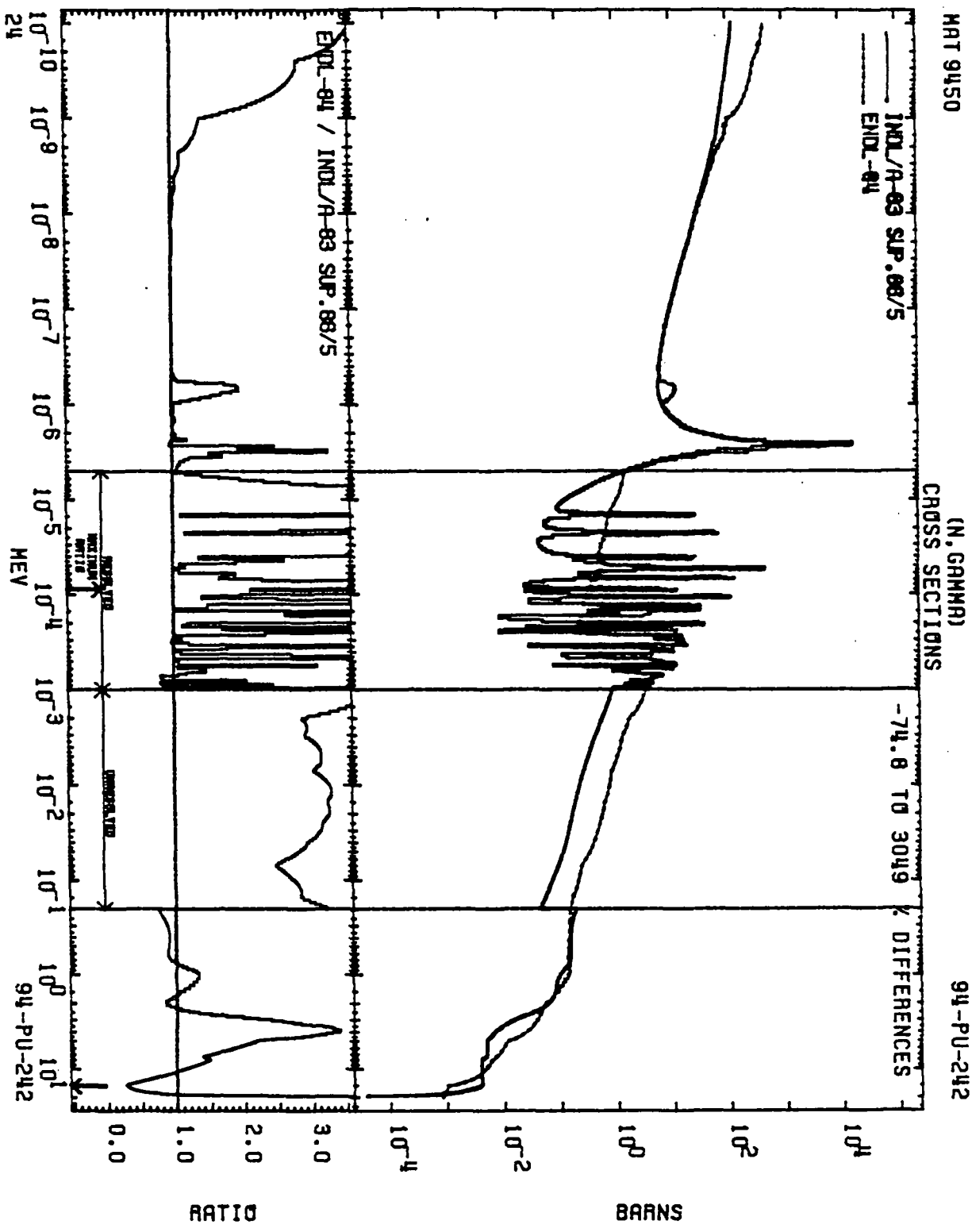

Figure 49: ^{242}Pu cross section comparison between INDL/A and ENDL-84 evaluations

F – Comparison of the averaged cross sections from the ENDL-84 library using different evaluations of the ^{252}Cf spontaneous fission spectrum

Spectra averages (millibarns) from ENDL-84 library

SPECTRA	Cf-252 fiss (NBS)	Cf-252 fiss (IAEA)
NUMBER OF GROUPS	620	640
SPECTRA ENERGY RANGE IS FROM	1.0000-10	1.0000-10
TO (MEV)	18.0	20.0
SPECTRA AVERAGED ENERGY (MEV)	2.1194	2.1226
STANDARD DEVIATION (MEV)	1.7141	1.6998

ISOTOPE	MAT	GROUPS	THRESHOLD (MEV)	REACTION	SPECTRA AVERAGES (MILLIBARNS)		Diff. (%)
90-TH-231	7863	640		TOTAL	7259.	7250.	-0.12
90-TH-231	7863	640		ELASTIC	4449.	4434.	-0.34
90-TH-231	7863	250	0.05	INELASTIC-TOTAL	2431.	2439.	0.33
90-TH-231	7863	149	5.1	(N,2N)	78.17	75.51	-3.40
90-TH-231	7863	640		FISSION	203.5	204.6	0.54
90-TH-231	7863	640		(N,GAMMA)	95.84	96.16	0.33
90-TH-231	7863	640		251	638.2	638.3	0.02
90-TH-232	7864	640		TOTAL	7417.	7417.	0.00
90-TH-232	7864	640		ELASTIC	4996.	4990.	-0.12
90-TH-232	7864	250	0.05	INELASTIC-TOTAL	2219.	2226.	0.32
90-TH-232	7864	135	6.5	(N,2N)	24.20	23.56	-2.64
90-TH-232	7864	190	1.0	FISSION	81.17	81.59	0.52
90-TH-232	7864	640		(N,GAMMA)	95.94	95.89	-0.05
90-TH-232	7864	640		251	517.0	519.4	0.46
90-TH-233	7865	640		TOTAL	7184.	7175.	-0.13
90-TH-233	7865	640		ELASTIC	4449.	4434.	-0.34
90-TH-233	7865	250	0.05	INELASTIC-TOTAL	2416.	2424.	0.33
90-TH-233	7865	152	4.8	(N,2N)	94.77	91.64	-3.30
90-TH-233	7865	640		FISSION	127.3	127.7	0.31
90-TH-233	7865	640		(N,GAMMA)	95.68	95.97	0.30
90-TH-233	7865	640		251	640.0	640.3	0.05
92-U -233	7866	640		TOTAL	7214.	7218.	0.06
92-U -233	7866	640		ELASTIC	4197.	4195.	-0.05
92-U -233	7866	235	0.1	INELASTIC-TOTAL	1053.	1060.	0.66
92-U -233	7866	140	6.0	(N,2N)	7.621	7.428	-2.53
92-U -233	7866	640		FISSION	1896.	1896.	0.00
92-U -233	7866	640		(N,GAMMA)	60.44	59.82	-1.03
92-U -233	7866	640		251	546.1	547.8	0.31
92-U -234	7867	640		TOTAL	7588.	7589.	0.01
92-U -234	7867	640		ELASTIC	5044.	5031.	-0.26
92-U -234	7867	253	0.04	INELASTIC-TOTAL	1168.	1175.	0.60
92-U -234	7867	132	6.8	(N,2N)	9.320	9.073	-2.65
92-U -234	7867	235	0.1	FISSION	1230.	1236.	0.49
92-U -234	7867	640		(N,GAMMA)	136.6	136.6	0.00
92-U -234	7867	640		251	523.0	524.6	0.31
92-U -235	7868	640		TOTAL	7534.	7538.	0.05
92-U -235	7868	640		ELASTIC	4613.	4610.	-0.07
92-U -235	7868	267	0.02	INELASTIC-TOTAL	1582.	1587.	0.32
92-U -235	7868	148	5.2	(N,2N)	15.54	15.06	-3.09
92-U -235	7868	640		FISSION	1232.	1234.	0.16
92-U -235	7868	640		(N,GAMMA)	91.46	91.55	0.10
92-U -235	7868	640		251	523.0	524.6	0.31
92-U -236	7869	640		TOTAL	7611.	7610.	-0.01
92-U -236	7869	640		ELASTIC	5055.	5042.	-0.26
92-U -236	7869	250	0.05	INELASTIC-TOTAL	1782.	1790.	0.45
92-U -236	7869	135	6.5	(N,2N)	19.01	18.45	-2.95
92-U -236	7869	640		FISSION	586.7	591.0	0.73
92-U -236	7869	640		(N,GAMMA)	168.2	168.1	-0.06

Spectra averages (millibarns) from ENDL-84 library

SPECTRA	Cf-252 fiss (NBS)	Cf-252 fi (IAEA)
NUMBER OF GROUPS	620	640
SPECTRA ENERGY RANGE IS FROM	1.0000-10	1.0000-10
TO (MEV)	18.0	20.0
SPECTRA AVERAGED ENERGY (MEV)	2.1194	2.1226
STANDARD DEVIATION (MEV)	1.7141	1.6998

ISOTOPE	MAT	GROUPS	THRESHOLD (MEV)	REACTION	SPECTRA AVERAGES (MILLIBARNS)		Diff. (%)
92-U -236	7869	640		251	522.9	524.5	0.31
92-U -237	7870	640		TOTAL	7524.	7514.	-0.13
92-U -237	7870	640		ELASTIC	4658.	4646.	-0.26
92-U -237	7870	267	0.02	INELASTIC-TOTAL	2075.	2081.	0.29
92-U -237	7870	149	5.1	(N,2N)	32.19	31.15	-3.23
92-U -237	7870	640		FISSION	655.9	652.7	-0.49
92-U -237	7870	640		(N,GAMMA)	103.0	103.3	0.29
92-U -237	7870	640		251	541.0	542.6	0.30
92-U -238	7871	640		TOTAL	7781.	7779.	-0.03
92-U -238	7871	640		ELASTIC	4834.	4823.	-0.23
92-U -238	7871	252	0.04	INELASTIC-TOTAL	2527.	2533.	0.24
92-U -238	7871	140	6.0	(N,2N)	23.65	22.98	-2.83
92-U -238	7871	207	0.4	FISSION	321.7	324.1	0.75
92-U -238	7871	640		(N,GAMMA)	74.85	74.97	0.16
92-U -238	7871	640		251	554.5	555.9	0.25
92-U -239	7872	640		TOTAL	7756.	7746.	-0.13
92-U -239	7872	640		ELASTIC	4665.	4653.	-0.26
92-U -239	7872	250	0.05	INELASTIC-TOTAL	2422.	2427.	0.21
92-U -239	7872	152	4.8	(N,2N)	84.85	82.00	-3.36
92-U -239	7872	640		FISSION	527.9	527.1	-0.15
92-U -239	7872	640		(N,GAMMA)	55.66	56.27	1.10
92-U -239	7872	640		251	540.7	542.4	0.31
92-U -240	7873	640		TOTAL	7690.	7693.	0.04
92-U -240	7873	640		ELASTIC	5367.	5356.	-0.20
92-U -240	7873	280	0.01	INELASTIC-TOTAL	1953.	1966.	0.67
92-U -240	7873	141	5.9	(N,2N)	45.76	44.23	-3.34
92-U -240	7873	190	1.0	FISSION	242.3	244.1	0.74
92-U -240	7873	640		(N,GAMMA)	80.95	81.23	0.35
92-U -240	7873	640		251	522.9	524.5	0:31
93-NP-235	8307	640		TOTAL	7653.	7652.	-0.01
93-NP-235	8307	640		ELASTIC	4802.	4789.	-0.27
93-NP-235	8307	257	0.03	INELASTIC-TOTAL	1518.	1522.	0.26
93-NP-235	8307	130	7.0	(N,2N)	6.339	6.205	-2.11
93-NP-235	8307	640		FISSION	1303.	1311.	0.61
93-NP-235	8307	640		(N,GAMMA)	22.38	22.51	0.58
93-NP-235	8307	640		251	538.2	539.8	0.30
93-NP-236	8308	640		TOTAL	7930.	7929.	-0.01
93-NP-236	8308	640		ELASTIC	4803.	4790.	-0.27
93-NP-236	8308	257	0.03	INELASTIC-TOTAL	1016.	1024.	0.79
93-NP-236	8308	143	5.7	(N,2N)	16.96	16.40	-3.30
93-NP-236	8308	640		FISSION	2062.	2066.	0.19
93-NP-236	8308	640		(N,GAMMA)	32.85	32.87	0.06
93-NP-236	8308	640		251	538.2	539.8	0.30
93-NP-237	7874	640		TOTAL	7798.	7797.	-0.01
93-NP-237	7874	640		ELASTIC	4803.	4790.	-0.27
93-NP-237	7874	257	0.03	INELASTIC-TOTAL	1518.	1522.	0.26
93-NP-237	7874	135	6.5	(N,2N)	6.364	6.229	-2.12
93-NP-237	7874	640		FISSION	1303.	1311.	0.61

Spectra averages (millibarns) from ENDL-84 library

SPECTRA	Cf-252 fiss (NBS)	Cf-252 fi (IAEA)
NUMBER OF GROUPS	620	640
SPECTRA ENERGY RANGE IS FROM	1.0000-10	1.0000-10
TO (MEV)	18.0	20.0
SPECTRA AVERAGED ENERGY (MEV)	2.1194	2.1226
STANDARD DEVIATION (MEV)	1.7141	1.6998

ISOTOPE	MAT	GROUPS	THRESHOLD (MEV)	REACTION	SPECTRA AVERAGES (MILLIBARNS)		Diff. (%)
93-NP-237	7874	640		(N,GAMMA)	166.5	167.0	0.30
93-NP-237	7874	640		251	538.2	539.8	0.30
93-NP-238	8309	640		TOTAL	7798.	7797.	-0.01
93-NP-238	8309	640		ELASTIC	4803.	4790.	-0.27
93-NP-238	8309	257	0.03	INELASTIC-TOTAL	1468.	1479.	0.75
93-NP-238	8309	145	5.5	(N,2N)	20.51	19.88	-3.07
93-NP-238	8309	640		FISSION	1474.	1475.	0.07
93-NP-238	8309	640		(N,GAMMA)	32.85	32.87	0.06
93-NP-238	8309	640		251	538.2	539.8	0.30
94-PU-238	7875	640		TOTAL	7761.	7754.	-0.09
94-PU-238	7875	640		ELASTIC	4642.	4631.	-0.24
94-PU-238	7875	250	0.05	INELASTIC-TOTAL	959.7	959.0	-0.07
94-PU-238	7875	130	7.0	(N,2N)	2.285	2.244	-1.79
94-PU-238	7875	640		FISSION	2047.	2052.	0.24
94-PU-238	7875	640		(N,GAMMA)	110.3	110.3	0.00
94-PU-238	7875	640		251	543.0	544.7	0.31
94-PU-239	7876	640		TOTAL	7732.	7731.	-0.01
94-PU-239	7876	640		ELASTIC	4710.	4700.	-0.21
94-PU-239	7876	248	0.05	INELASTIC-TOTAL	1193.	1200.	0.59
94-PU-239	7876	144	5.6	(N,2N)	7.205	6.994	-2.93
94-PU-239	7876	640		FISSION	1781.	1782.	0.06
94-PU-239	7876	640		(N,GAMMA)	41.53	41.46	-0.17
94-PU-239	7876	640		251	522.9	524.5	0.31
94-PU-240	7877	640		TOTAL	7442.	7438.	-0.05
94-PU-240	7877	640		ELASTIC	4402.	4389.	-0.30
94-PU-240	7877	252	0.04	INELASTIC-TOTAL	1533.	1534.	0.07
94-PU-240	7877	134	6.6	(N,2N)	5.228	5.075	-2.93
94-PU-240	7877	640		FISSION	1413.	1421.	0.57
94-PU-240	7877	640		(N,GAMMA)	88.92	89.17	0.28
94-PU-240	7877	640		251	522.9	524.5	0.31
94-PU-241	7878	640		TOTAL	8102.	8097.	-0.06
94-PU-241	7878	640		ELASTIC	4826.	4816.	-0.21
94-PU-241	7878	250	0.05	INELASTIC-TOTAL	1524.	1528.	0.26
94-PU-241	7878	147	5.3	(N,2N)	45.32	43.74	-3.49
94-PU-241	7878	640		FISSION	1593.	1595.	0.13
94-PU-241	7878	640		(N,GAMMA)	113.5	112.9	-0.53
94-PU-241	7878	640		251	522.9	524.5	0.31
94-PU-242	7880	640		TOTAL	7912.	7910.	-0.03
94-PU-242	7880	640		ELASTIC	4880.	4868.	-0.25
94-PU-242	7880	252	0.04	INELASTIC-TOTAL	1814.	1816.	0.11
94-PU-242	7880	137	6.3	(N,2N)	10.52	10.22	-2.85
94-PU-242	7880	640		FISSION	1126.	1133.	0.62
94-PU-242	7880	640		(N,GAMMA)	81.48	81.50	0.02
94-PU-242	7880	640		251	594.9	596.6	0.29
94-PU-243	7881	640		TOTAL	7804.	7793.	-0.14
94-PU-243	7881	640		ELASTIC	4585.	4566.	-0.41
94-PU-243	7881	246	0.06	INELASTIC-TOTAL	2032.	2037.	0.25
94-PU-243	7881	150	5.0	(N,2N)	67.64	65.33	-3.42

Spectra averages (millibarns) from ENDL-84 library

SPECTRA	Cf-252 fiss (NBS)	Cf-252 fi (IAEA)
NUMBER OF GROUPS	620	640
SPECTRA ENERGY RANGE IS FROM	1.0000-10	1.0000-10
TO (MEV)	18.0	20.0
SPECTRA AVERAGED ENERGY (MEV)	2.1194	2.1226
STANDARD DEVIATION (MEV)	1.7141	1.6998

ISOTOPE	MAT	GROUPS	THRESHOLD (MEV)	REACTION	SPECTRA AVERAGES (MILLIBARNS)		Diff. (%)
94-PU-243	7881	640		FISSION	1076.	1083.	0.65
94-PU-243	7881	640		(N,GAMMA)	41.48	41.36	-0.29
94-PU-243	7881	640		251	594.6	596.3	0.29
95-AM-241	7882	640		TOTAL	7615.	7612.	-0.04
95-AM-241	7882	640		ELASTIC	4732.	4720.	-0.25
95-AM-241	7882	250	0.05	INELASTIC-TOTAL	1192.	1191.	-0.08
95-AM-241	7882	141	5.9	(N,2N)	15.87	15.28	-3.72
95-AM-241	7882	640		FISSION	1500.	1512.	0.80
95-AM-241	7882	640		(N,GAMMA)	157.4	156.6	-0.51
95-AM-241	7882	640		251	629.1	629.6	0.08
95-AM-242	7883	640		TOTAL	7547.	7529.	-0.24
95-AM-242	7883	640		ELASTIC	4531.	4510.	-0.46
95-AM-242	7883	235	0.1	INELASTIC-TOTAL	1068.	1072.	0.37
95-AM-242	7883	144	5.6	(N,2N)	27.56	26.56	-3.63
95-AM-242	7883	640		FISSION	1825.	1825.	0.00
95-AM-242	7883	640		(N,GAMMA)	95.43	95.65	0.23
95-AM-242	7883	640		251	596.7	598.7	0.34
95-AM-243	7884	640		TOTAL	7667.	7660.	-0.09
95-AM-243	7884	640		ELASTIC	4589.	4570.	-0.41
95-AM-243	7884	240	0.08	INELASTIC-TOTAL	1888.	1892.	0.21
95-AM-243	7884	136	6.4	(N,2N)	25.57	24.76	-3.17
95-AM-243	7884	640		FISSION	1123.	1132.	0.80
95-AM-243	7884	640		(N,GAMMA)	41.52	41.41	-0.26
95-AM-243	7884	640		251	595.3	597.0	0.29
96-CM-242	7885	640		TOTAL	7707.	7712.	0.06
96-CM-242	7885	640		ELASTIC	4537.	4532.	-0.11
96-CM-242	7885	252	0.04	INELASTIC-TOTAL	1627.	1629.	0.12
96-CM-242	7885	130	7.0	(N,2N)	14.64	14.23	-2.80
96-CM-242	7885	640		FISSION	1433.	1442.	0.63
96-CM-242	7885	640		(N,GAMMA)	95.43	95.65	0.23
96-CM-242	7885	640		251	596.7	598.7	0.34
96-CM-243	7886	640		TOTAL	8258.	8235.	-0.28
96-CM-243	7886	640		ELASTIC	4585.	4566.	-0.41
96-CM-243	7886	246	0.06	INELASTIC-TOTAL	1608.	1602.	-0.37
96-CM-243	7886	143	5.7	(N,2N)	26.73	25.83	-3.37
96-CM-243	7886	640		FISSION	1996.	2000.	0.20
96-CM-243	7886	640		(N,GAMMA)	41.00	40.94	-0.15
96-CM-243	7886	640		251	594.6	596.3	0.29
96-CM-244	7887	640		TOTAL	7654.	7658.	0.05
96-CM-244	7887	640		ELASTIC	4537.	4532.	-0.11
96-CM-244	7887	252	0.04	INELASTIC-TOTAL	1661.	1663.	0.12
96-CM-244	7887	132	6.8	(N,2N)	15.06	14.62	-2.92
96-CM-244	7887	640		FISSION	1399.	1406.	0.50
96-CM-244	7887	640		(N,GAMMA)	41.88	41.85	-0.07
96-CM-244	7887	640		251	595.6	597.3	0.29
96-CM-245	7888	640		TOTAL	7964.	7942.	-0.28
96-CM-245	7888	640		ELASTIC	4653.	4635.	-0.39
96-CM-245	7888	246	0.06	INELASTIC-TOTAL	1519.	1515.	-0.26

Spectra averages (millibarns) from ENDL-84 library

SPECTRA	Cf-252 fiss (NBS)	Cf-252 fi (IAEA)
NUMBER OF GROUPS	620	640
SPECTRA ENERGY RANGE IS FROM	1.0000-10	1.0000-10
TO (MEV)	18.0	20.0
SPECTRA AVERAGED ENERGY (MEV)	2.1194	2.1226
STANDARD DEVIATION (MEV)	1.7141	1.6998

ISOTOPE	MAT	GROUPS	THRESHOLD (MEV)	REACTION	SPECTRA AVERAGES (MILLIBARNS)		Diff. (%)
96-CM-245	7888	144	5.6	(N,2N)	27.46	26.53	-3.39
96-CM-245	7888	640		FISSION	1723.	1725.	0.12
96-CM-245	7888	640		(N,GAMMA)	41.00	40.94	-0.15
96-CM-245	7888	640		251	568.8	571.5	0.47
96-CM-246	7889	640		TOTAL	7622.	7621.	-0.01
96-CM-246	7889	640		ELASTIC	4510.	4499.	-0.24
96-CM-246	7889	253	0.04	INELASTIC-TOTAL	1701.	1701.	0.00
96-CM-246	7889	136	6.4	(N,2N)	23.55	22.82	-3.10
96-CM-246	7889	640		FISSION	1346.	1356.	0.74
96-CM-246	7889	640		(N,GAMMA)	41.51	41.40	-0.26
96-CM-246	7889	640		251	595.0	596.6	0.27
96-CM-247	7890	640		TOTAL	7912.	7904.	-0.10
96-CM-247	7890	640		ELASTIC	4541.	4530.	-0.24
96-CM-247	7890	250	0.05	INELASTIC-TOTAL	1224.	1227.	0.25
96-CM-247	7890	149	5.1	(N,2N)	41.05	39.63	-3.46
96-CM-247	7890	640		FISSION	2066.	2067.	0.05
96-CM-247	7890	640		(N,GAMMA)	40.85	40.74	-0.27
96-CM-247	7890	640		251	595.6	597.3	0.29
96-CM-248	7891	640		TOTAL	7633.	7634.	0.01
96-CM-248	7891	640		ELASTIC	4522.	4508.	-0.31
96-CM-248	7891	252	0.04	INELASTIC-TOTAL	1668.	1673.	0.30
96-CM-248	7891	138	6.2	(N,2N)	25.59	24.77	-3.20
96-CM-248	7891	640		FISSION	1342.	1352.	0.75
96-CM-248	7891	640		(N,GAMMA)	74.85	74.98	0.17
96-CM-248	7891	640		251	594.4	596.0	0.27
97-BK-249	7892	640		TOTAL	7629.	7627.	-0.03
97-BK-249	7892	640		ELASTIC	4521.	4508.	-0.29
97-BK-249	7892	250	0.05	INELASTIC-TOTAL	2115.	2119.	0.19
97-BK-249	7892	138	6.2	(N,2N)	29.36	28.46	-3.07
97-BK-249	7892	640		FISSION	888.6	896.3	0.87
97-BK-249	7892	640		(N,GAMMA)	74.90	75.04	0.19
97-BK-249	7892	640		251	594.8	596.4	0.27
98-CF-249	7893	640		TOTAL	7858.	7850.	-0.10
98-CF-249	7893	640		ELASTIC	4522.	4509.	-0.29
98-CF-249	7893	252	0.04	INELASTIC-TOTAL	1509.	1514.	0.33
98-CF-249	7893	144	5.6	(N,2N)	29.54	28.55	-3.35
98-CF-249	7893	640		FISSION	1721.	1724.	0.17
98-CF-249	7893	640		(N,GAMMA)	74.89	75.03	0.19
98-CF-249	7893	640		251	594.8	596.4	0.27
98-CF-250	7894	640		TOTAL	8023.	8009.	-0.17
98-CF-250	7894	640		ELASTIC	4522.	4509.	-0.29
98-CF-250	7894	252	0.04	INELASTIC-TOTAL	1432.	1432.	0.00
98-CF-250	7894	134	6.6	(N,2N)	15.55	15.09	-2.96
98-CF-250	7894	640		FISSION	1978.	1977.	-0.05
98-CF-250	7894	640		(N,GAMMA)	74.93	75.08	0.20
98-CF-250	7894	640		251	595.1	596.8	0.29
98-CF-251	7895	640		TOTAL	7862.	7855.	-0.09
98-CF-251	7895	640		ELASTIC	4522.	4509.	-0.29

Spectra averages (millibarns) from ENDL-84 library

SPECTRA	Cf-252 fiss (NBS)	Cf-252 fi (IAEA)
NUMBER OF GROUPS	620	640
SPECTRA ENERGY RANGE IS FROM	1.0000-10	1.0000-10
TO (MEV)	18.0	20.0
SPECTRA AVERAGED ENERGY (MEV)	2.1194	2.1226
STANDARD DEVIATION (MEV)	1.7141	1.6998

ISOTOPE	MAT	GROUPS	THRESHOLD (MEV)	REACTION	SPECTRA AVERAGES (MILLIBARNS)		Diff. (%)
98-CF-251	7895	263	0.02	INELASTIC-TOTAL	1495.	1500.	0.33
98-CF-251	7895	149	5.1	(N,2N)	48.38	46.72	-3.43
98-CF-251	7895	640		FISSION	1721.	1724.	0.17
98-CF-251	7895	640		(N,GAMMA)	74.92	75.07	0.20
98-CF-251	7895	640		251	594.8	596.4	0.27
98-CF-252	7896	640		TOTAL	7782.	7787.	0.06
98-CF-252	7896	640		ELASTIC	4640.	4635.	-0.11
98-CF-252	7896	250	0.05	INELASTIC-TOTAL	1169.	1167.	-0.17
98-CF-252	7896	138	6.2	(N,2N)	11.86	11.48	-3.20
98-CF-252	7896	640		FISSION	1922.	1934.	0.62
98-CF-252	7896	640		(N,GAMMA)	39.76	39.68	-0.20
98-CF-252	7896	640		251	575.9	578.4	0.43