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Mark 14-30A

TECHNICAL DIVISION
SAVANNAH RIVER LABORATORY

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May 9, 1985

TO: W. E. JULE, 773-A

FROM: J. R. CHANDLER, 773-41A
JRC

Evaluation of ^{235}U, ^{238}U, ^{6}Li, and ^{27}Al Cross Sections

Introduction and Summary

Good nuclear data are essential for accurate prediction of reactor parameters. Several cross section libraries are currently available for use with GLASS physics calculations. In recent Mark 15 and Mark 22 studies, cross section data were developed to provide more accurate buckling calculations for Mark 15 and Mark 22 charges.[1,2,3] This report documents evaluation of these new data for universal application. All new data described in this report are in the JOSHUA.STD dataset. The names for these new cross section data in JOSHUA.STD are U235E4/5, U238MOD, LI6E4, and AL27E4.

Use of the new data is recommended for all GLASS calculations involving SR lattices. GLASS results using the various sets of cross section data in JOSHUA.STD were evaluated against experimental data for Mark 22, Mark 15, Mark 16-30A, Mark 14-30A, Mark VR, and Mark VE lattices. Use of the new data consistently gives better results for material buckling than use of any of the older data. Use of the new data provides material buckling results which agree with experimental buckling results. In most of the cases studied use of the new data gives results equal to older data results for the quantities ρ^{28}, C*, δ^{25}, and δ^{28}.

*This copy contains microfiche for all GLASS calculations supporting DPST-85-288.

Discussion

Data Development

Most SRL cross section data is from the HAMMER code data library or from ENDF/B-III (Evaluated Nuclear Data Files B-Version III). HAMMER was the 1960's predecessor of the GLASS code and had cross sections for nuclides used in SR calculations. ENDF data is a set of data that is evaluated and maintained by the Cross Section Evaluation Working Group (CSEWG), a group of experts representing national laboratories and other interested organizations. Three versions of ENDF data are of use to SRL: ENDF/B-III of 1972; ENDF/B-IV of 1975; and ENDF/B-V of 1979. (Currently SRL processing codes are not compatible with ENDF/B-V formats.) Recent studies of Mark 15 and Mark 22 reactivity calculations led to generation of new SRL cross section data from ENDF data for ^{235}U, ^{238}U, ^{6}Li, and and ^{27}Al.[1,2] A summary of these developments follows.

Three cross section libraries for ^{235}U currently exist in the JOSHUA.STD.MULTIGRP dataset versions STANDARD and STD37. The libraries have the names U235, U235OLD, and U235E4/5. The origin of the U235OLD library is not well known. It is from the HAMMER code library in the mid-1960's but its experimental basis is unknown. The U235 library was developed in 1972 from ENDF/B-III data.[4] The only significant change made to the ENDF/B-III ^{235}U data at that time was 5% reduction in capture and fission cross sections over the energy range 0.101 - 25 keV. The U235E4/5 library was developed recently in an attempt to improve Mark 22 reactivity calculations.[2] The U235E4/5 data consists of pure ENDF/B-IV cross sections and modified ENDF/B-IV ν data. ENDF data for ^{235}U were modified between version IV and version V but the major impact of these modifications on SR lattice calculations is an increase in η

(where $\eta = \dfrac{\nu\sigma f}{\sigma_f+\sigma_c}$ = number of neutrons emitted per neutron absorbed).

The ν modification was derived so thermal η of U235E4/5 is equal to thermal η of ENDF/B-V. MULTIGRP data for ^{235}U are illustrated in Figures 1-4.

Three cross section libraries for ^{238}U exist in the JOSHUA.STD.MULTIGRP dataset versions STANDARD and STD37. The libraries have the names U238, U238OLD, and U238MOD. The U238OLD library is from the HAMMER code library, just as U235OLD. The U238 library was developed in 1972 with the U235 library.[4] Several cross sections from ENDF/B-III were reduced when the U238 library was generated. A detailed accounting of the changes is given in Reference 4. The most recent ^{238}U library is U238MOD. This library is identical to the U238 library except the 6.65 eV resonance capture width is adjusted from 23.0 meV to 21.0 meV. This adjustment yields agreement between measured and calculated Mark 15 buckling (see Table V).[1] MULTIGRP data for ^{238}U are illustrated in Figures 5-7.

Two cross section libraries for ^{27}Al exist in the JOSHUA.STD.MULTIGRP dataset versions STANDARD and STD37. The libraries have the names AL and AL27E4. The AL library is from the HAMMER code library. The AL27E4 library is pure ENDF/B-IV data. The major difference between the AL27E4 and AL cross sections is that AL27E4 has a smaller epithermal neutron absorption cross section. MULTIGRP data for ^{27}Al are illustrated in Figures 8 and 9.

Four cross section libraries exist for ^{6}Li having the names LI6, LI6X, LI6E4, and LI6XE4. However, the LI6 and LI6X libraries are identical, as are the LI6E4 and LI6XE4 libraries. This allows inner and outer Mark 22 target contents to be distinguished in GLASS depletion edits. The LI6 and LI6X libraries are from the HAMMER code library whereas LI6E4 and LI6XE4 are ENDF/B-IV data. There is no major difference between the data, but the advantage of using ENDF data is knowing the experimental basis for the data. MULTIGRP data for ^{6}Li are illustrated in Figures 10 and 11.

Lattice Measurements

Physics parameters were measured for several lattices in the Process Development Pile (PDP) and Subcritical Experiment (SE) facility in the 1960's and 1970's. Results from Mark VE, Mark 14-30A, Mark 16-30A, Mark 15, and Mark 22 measurements are listed in Tables I-V. Results for Mark VR are listed in Appendix A. Details of the lattice contents and geometries along with results are given in References 1 and 4-11.

Evaluation of data was based on comparison of GLASS calculations of lattice integral parameters with results of SE and PDP experiments. Integral parameters used in the comparison include the lattice material buckling and the quantities ρ^{28}, C*, δ^{25} and δ^{28}. The quantities ρ^{28}, C*, δ^{25}, and δ^{28} represent neutron activation ratios in ^{235}U and ^{238}U, and are defined below:

$$\rho^{28} = \frac{\text{Epithermal } ^{238}\text{U captures}}{\text{Thermal } ^{238}\text{U captures}}$$

$$C^* = \frac{\text{Total } ^{238}\text{U captures}}{\text{Total } ^{235}\text{U fissions}}$$

$$\delta^{25} = \frac{\text{Epithermal } ^{235}\text{U fissions}}{\text{Thermal } ^{235}\text{U fissions}}$$

$$\delta^{28} = \frac{\text{Total } ^{238}\text{U fissions}}{\text{Total } ^{235}\text{U fissions}}$$

Measurements of C* were made indirectly as a ratio of C* in the lattice of interest to C* in a reference position having a well-thermalized spectrum. This ratio was then multiplied by an approximate expression for the $^{238}U/^{235}U$ activity ratio in the thermal reference position.

$$C^*_{\text{reported}} = \frac{N_{238}\ (g\sigma_a)_{238}}{N_{235}\ (g\sigma_f)_{235}} \times \frac{C^*_{\text{lattice}}}{C^*_{\text{thermal reference}}}$$

Here $(g\sigma_a)_{238}$ and $(g\sigma_f)_{235}$ are effective (average) thermal cross sections based on the product of Wescott g factors and 2200 m/sec cross sections from SRL libraries. Thus experimental values reported for C* depend on ^{235}U and ^{238}U cross section data. In order to account for changes in $(g\sigma_a)_{238}$ and $(g\sigma_f)_{235}$ due to evaluation of new differential data for ^{238}U and ^{235}U, experimental values of C* were corrected as shown below:

$$C^*_{\text{corrected}} = C^*_{\text{reported}} \frac{(\bar{\sigma}_a\ ^{238}/\bar{\sigma}_f\ ^{235})\ \text{new evaluation}}{(\bar{\sigma}_a\ ^{238}/\bar{\sigma}_f\ ^{235})\ \text{old evaluation}}$$

where $\bar{\sigma}_a{}^{238}$ is the average thermal absorption cross section for ^{238}U and $\bar{\sigma}_f{}^{235}$ is the average thermal fission cross section for ^{235}U. Values for $\bar{\sigma}_a{}^{238}$ and $\bar{\sigma}_f{}^{235}$ were determined from GLASS calculations of thermal neutron spectra in dilute solutions of natural uranium and D_2O. Corrected C* values for each cross section set are listed in Tables II and A-2 as "measured" data.

Lattice Calculations

GLASS calculations for Mark VE, Mark 14-30A, Mark 16-30A, Mark 15, and Mark 22 lattices were made using fuel and target contents listed in Table VI. The contents for Mark VE, Mark 30A, and Mark 15 assemblies are the same as listed in References 1, 7 and 11. Contents of Mark 14, Mark 16, and Mark 22 assemblies were modified to account for NTG errors.[7,10] Details of the Mark 14 and Mark 16 modifications are given in Reference 12. It was assumed that the Mark 16B low-k NTG recalibration results apply to Mark 14 and Mark 16 assemblies since they also have 3 fuel tubes and probably had similar NTG standards problems. However, the Mark 14 and Mark 16 fuel modifications only affect material buckling results and change them by only 10μB. Details of the Mark 22 modifications, which have significant effect, are given in Reference 2.

GLASS buckling calculations for Mark 22, Mark 15, Mark 16-30A, Mark 14-30A, and Mark VE included representation of the stacked slug columns as axially homogeneous arrays of concentric cylinders. This homogenization included the following:

- o Aluminum end caps and uranium metal homogenized over the total length of the slug.
- o Nickel coating, present on Mark 15 and Mark VE, homogenized with the aluminum cladding.

For purposes of comparison with foil activation experiments, GLASS activation calculations ignored the presence of end caps. Instead, slugs were treated as homogeneous columns of uranium metal at density 18.9 gm/cc.

Results

Overall results are better when the new cross sections are used. Comparisons using results from current calculations are presented in Tables I-V. Use of the new cross section data provides very good agreement between calculated and measured buckling. Calculated buckling improvements are 20-30 μB except for the Mark 22 lattice where the improvement is about 90 μB. Results for ρ^{28}, C*, δ^{25}, and δ^{28} are generally about the same when the new data are used in place of older standard data.

Comparisons between calculations using RAHAB (the predecessor to GLASS) and measured data are described in Reference 4 for older SRL cross section data and Mark VE, Mark 14-30A, and Mark 16-30A lattices. Current GLASS calculations using what is believed to be the same data do not agree with results presented in Reference 4 (see Tables VII and VIII). Notes[13] of D. A. Sharp made in 1973 list several corrections (Tables IX and X) to the buckling results presented in Reference 4 for Mark 14-30A and Mark 16-30A. These corrections to Reference 4 yield results which closely agree with current Mark 14-30A and Mark 16-30A results. Similar corrections applied to Reference 4 Mark VE results would yield good agreement with current calculations. Additionally, D. A. Sharp evaluated integral transport (RAHAB) and Monte Carlo calculations for experimentally measured lattices (Table XI).[13] No calculational differences were found for uniform lattices. In mixed lattices, the integral transport option gave a systematically low ratio of Mark 14/Mark 30A and Mark 16/Mark 30A fissions. Monte Carlo gives a higher ratio, in good agreement with experiment, resulting in a higher calculated buckling and better agreement with measured buckling. Integral transport (the GLASS option used in this study) buckling results should be corrected when comparing with measured buckling. The correction factors are listed in Tables II and III. It should be noted that when cell cross sections from GLASS are to be used in GRIMHX, as JASON does, the GLASS option (Monte Carlo or integral transport) will not have much effect.

Use of the new data is recommended for all GLASS calculations involving SR lattices because the origins of the data are well documented, and because the new data give good agreement with integral measurements.

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13. D. A. Sharp, personal notes.

TABLE I

MARK VE MATERIAL BUCKLING

Cross Section Set	FZ-0 B_m^2	BZ B_m^2
Original SRL Standard[1]	730 μB	614 μB
1972 SRL Evaluation[2]	792	688
Current SRL Standard[3]	848	752
Proposed SRL Standard[4]	866	772
Experiment	883	800 ±10

1 U235OLD, U238OLD, LI6, AL

2 U235, U238, LI6, AL

3 U235OLD, U238, LI6, AL

4 U235E4/5, U238MOD, LI6E4, AL27E4

TABLE II

LATTICE BUCKLING AND MARK 30A ACTIVATION PARAMETERS IN MARK 14-30A LATTICES

		CALCULATIONS WITH VARIOUS DATA VERSIONS			
Parameter	Experiment	Original[1] SRL Standard	1972 SRL[2] Evaluation	Current[3] SRL Standard	Proposed[4] SRL Standard
FZ-0 B_m^2	585 μB	431 μB	459 μB	493 μB	519 μB
Corrected[5] B_m^2		495 μB	523 μB	557 μB	583 μB
ρ^{28}	0.626	0.613	0.595	0.598	0.596
C* measured		5.12	5.04	5.03	5.00
C* calculated		5.12	4.96	5.00	4.91
δ^{25}	0.0806	0.077	0.078	0.076	0.081
δ^{28}	0.152	0.143	0.137	0.142	0.139

1 U235OLD, U238OLD, LI6, AL

2 U235, U238, LI6, AL

3 U235OLD, U238, LI6, AL

4 U235E4/5, U238MOD, LI6E4, AL27E4

5 Correction factor +64 μB was applied to correct for RAHAB-Monte Carlo calculation differences.

TABLE III

LATTICE BUCKLING AND MARK 30A ACTIVATION PARAMETERS IN MARK 16-30A LATTICES

		CALCULATIONS WITH VARIOUS DATA VERSIONS			
Parameter	Experiment	Original[1] SRL Standard	1972 SRL[2] Evaluation	Current[3] SRL Standard	Proposed[4] SRL Standard
FZ-0 B_m^2	557 μB	422 μB	447 μB	485 μB	518 μB
Corrected[5] B_m^2		457 μB	482 μB	520 μB	553 μB
ρ^{28}	0.717	0.698	0.680	0.679	0.681
C*	Not measured				
δ^{25}	0.079	0.086	0.088	0.086	0.091
δ^{28}	0.160	0.146	0.140	0.145	0.142

1 U235OLD, U238OLD, LI6, AL

2 U235, U238, LI6, AL

3 U235OLD, U238, LI6, AL

4 U235E4/5, U238MOD, LI6E4, AL27E4

5 Correction factor +35 μB was applied to correct for RAHAB-Monte Carlo calculation differences.

TABLE IV

MARK 15 MATERIAL BUCKLING

Cross Section Set	B_m^2, μB
Original SRL Standard[1]	540
1972 SRL Evaluation[2]	630
Current SRL Standard[3]	705
Proposed SRL Standard[4]	726
Mk15 Demonstration Charge Design[5]	729
LTR Measurement	728

1 U235OLD, U238OLD, LI6, AL

2 U235, U238, LI6, AL

3 U235OLD, U238, LI6, AL

4 U235E4/5, U238MOD, LI6E4, AL27E4

5 U235OLD, U238MOD, LI6, AL

TABLE V

MARK 22 MATERIAL BUCKLING

^{6}Li Targets, gm/ft Inner	Outer	Experiment	Original[1] SRL Standard	Current[2] SRL Standard	Proposed[3,4] SRL Standard
3.339	1.223	262 μB	174 μB	147 μB	239 μB
3.339	1.129	374	283	256	347
3.339	0.923	608	527	499	588
2.946	1.223	348	259	232	324
2.946	1.129	473	377	341	431
2.946	0.923	702	611	584	673
2.641	1.223	439	336	309	400
2.641	1.129	550	443	416	506
2.641	0.923	796	687	660	749

1 U235OLD, U238OLD, LI6, AL

2 U235, U238, LI6, AL

3 U235E4/5, U238MOD, LI6E4, AL27E4

4 These results differ slightly from results listed in Reference 2 due to an error in control rod coolant temperature and purity in Reference 2 calculations.

TABLE VI

ASSEMBLY CONTENTS, GM/FT

ASSEMBLY	^{234}U	^{235}U	^{236}U	^{238}U	^{6}LI
Mark VE	0.959	136.7	1.013	14330	-
Mark 14[1] (PDP)	2.566	135.0	35.11	20.20	-
Mark 14[1] (SE)	2.943	136.2	35.86	19.00	-
Mark 16[1] (PDP)	6.006	240.7	178.2	71.48	0.263
Mark 16[1] (SE)	7.975	234.6	175.0	71.92	0.320
Mark 30A(PDP)	-	36.60	-	25567	-
Mark 30A(SE)	-	38.04	-	26748	-
Mark 15	-	190.9	4.011	17157	-
Mark 22[1]	3.567	243.3	30.81	26.19	Varies[2]

1 Contents listed in References 7-10 were modified to account for NTG errors

2 See Table V

TABLE VII

COMPARISON OF ACTIVATION PARAMETER CALCULATIONS FOR MARK 30A ASSEMBLIES

MARK 14-30A FZO LATTICE

Parameter	1972 Calculation-Ref. 4: 1972[1] Standard	1972 Calculation-Ref. 4: 1972[2] Evaluation	Current Calculation: 1972[1] Standard	Current Calculation: 1972[2] Evaluation
ρ^{28}	0.606	0.599	0.613	0.595
C*	5.03	4.90	5.12	4.96
δ^{25}	0.0752	0.0781	0.077	0.078
δ^{28}	0.136	0.141	0.143	0.137

MARK 16-30A FZO LATTICE

Parameter	1972 Calculation-Ref. 4: 1972[1] Standard	1972 Calculation-Ref. 4: 1972[2] Evaluation	Current Calculation: 1972[1] Standard	Current Calculation: 1972[2] Evaluation
ρ^{28}	0.702	0.696	0.698	0.680
δ^{25}	0.0870	0.0909	0.086	0.088
δ^{28}	0.138	0.151	0.146	0.140

1 U235OLD, U238OLD, LI6, AL

2 U235, U238, LI6, AL

TABLE VIII

COMPARISON OF B_m^2 CALCULATIONS

LATTICE	CROSS SECTION DATA	CALCULATION DATE	B_m^2
MK14-30A FZ0	1972 Standard[1]	1972 - Ref. 4	465
		1973 - Ref. 13	439
		1985	431
	1972 Evaluation[2]	1972 - Ref. 4	492
		1973 - Ref. 13	466
		1985	459
MK16-30A FZ0	1972 Standard[1]	1972 - Ref. 4	457
		1973 - Ref. 13	430
		1985	422
	1972 Evaluation[2]	1972 - Ref. 4	481
		1973 - Ref. 13	454
		1985	447
Mark VE FZ0	1972 Standard[1]	1972 - Ref. 4	774
		1985	724
	1972 Evaluation[2]	1972 - Ref. 4	824
		1985	786
Mark VE BZ	1972 Standard[1]	1972 - Ref. 4	661
		1985	614
	1972 Evaluation[2]	1972 - Ref. 4	734
		1985	688

[1] U235OLD, U238OLD, LI6, AL

[2] U235, U238, LI6, AL

TABLE IX

CORRECTIONS TO 1972 MARK 14-30A BUCKLING CALCULATION

RAHAB Calculation (2/23/72) Used U235, U238, and TESTALIB D-D_2O Experimental Mark 30A	492 μB
Use of STANDARD D-D_2O instead of TESTALIB	- 6 μB
Changed atomic weights to correct error	- 6 μB
ϵ[1]	- 4 μB
Use of production Mark 30A	- 10 μB
RAHAB (2/20/73)	466 μB

[1]Definition of ϵ is unknown.

TABLE X

CORRECTIONS TO 1972 MARK 16-30A BUCKLING CALCULATION

RAHAB Calculation (2/23/72) Used U235, U238, and TESTALIB D-D_2O Experimental Mark 30A	481 μB
Use of STANDARD D-D_2O instead of TESTALIB	- 12 μB
Changed atomic weights	- 6 μB
Use of production Mark 30A	- 9 μB
RAHAB (2/20/73)	454 μB

TABLE XI

COMPARISON OF INTEGRAL TRANSPORT AND MONTE CARLO MIXED LATTICE CALCULATIONS[1]

LATTICE	CALCULATED PARAMETER	RAHAB2	MONTE2
Mark 14-30A	B_m^2	465	529 ± 7
Mark 14-30A	$\frac{\text{Target Thermal Flux}}{\text{Fuel Thermal Flux}}$	0.874	$0.843 \pm .004$
Mark 16-30A	B_m^2	447	483 ± 7
Mark 16-30A	$\frac{\text{Target Thermal Flux}}{\text{Fuel Thermal Flux}}$	1.210	$1.242 \pm .006$

1. This table taken from Reference 13.
2. Measured Mark 14-30A $B_m^2 = 585 \mu B$.
3. Measured Mark 16-30A $B_m^2 \simeq 557 \mu B$.

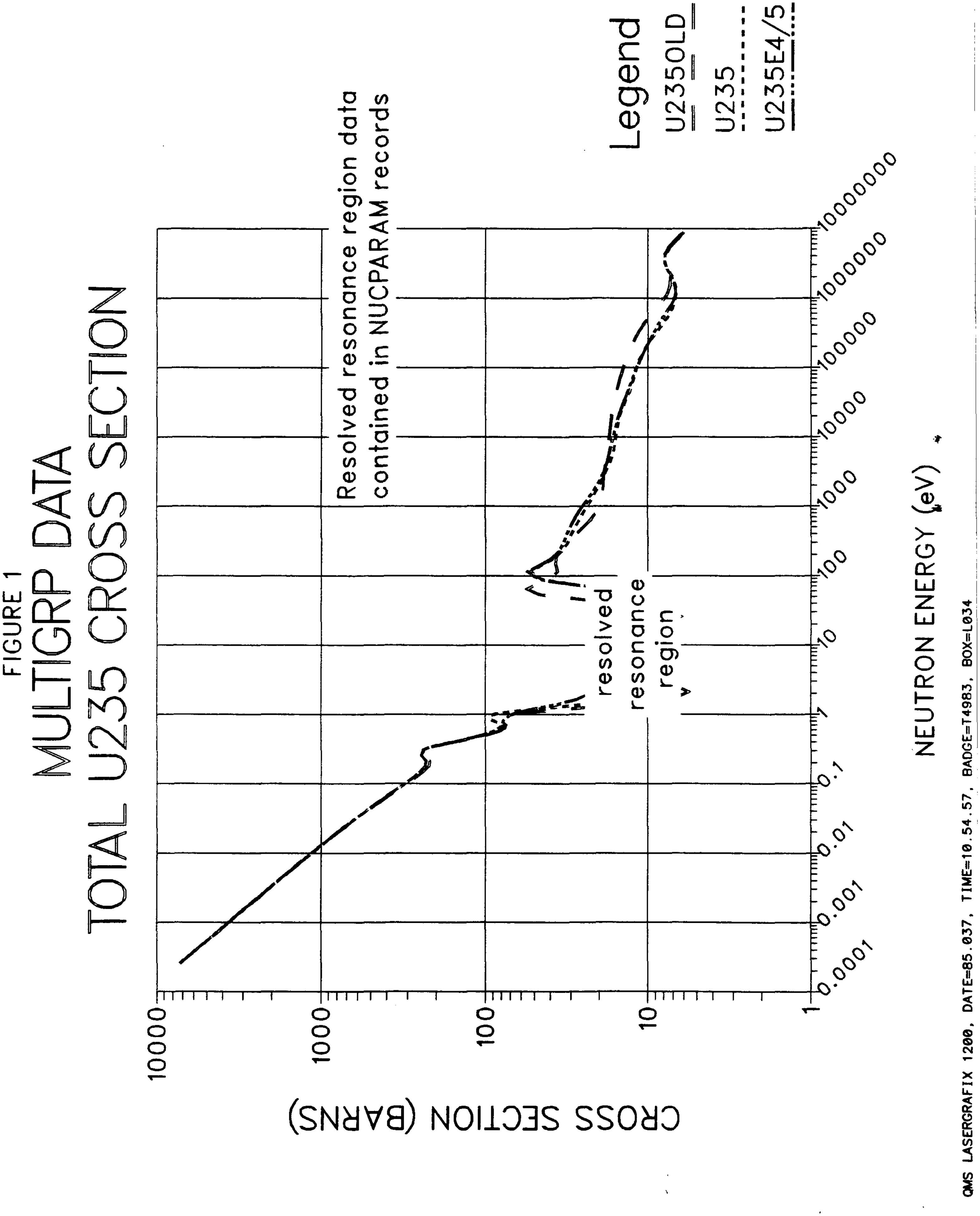
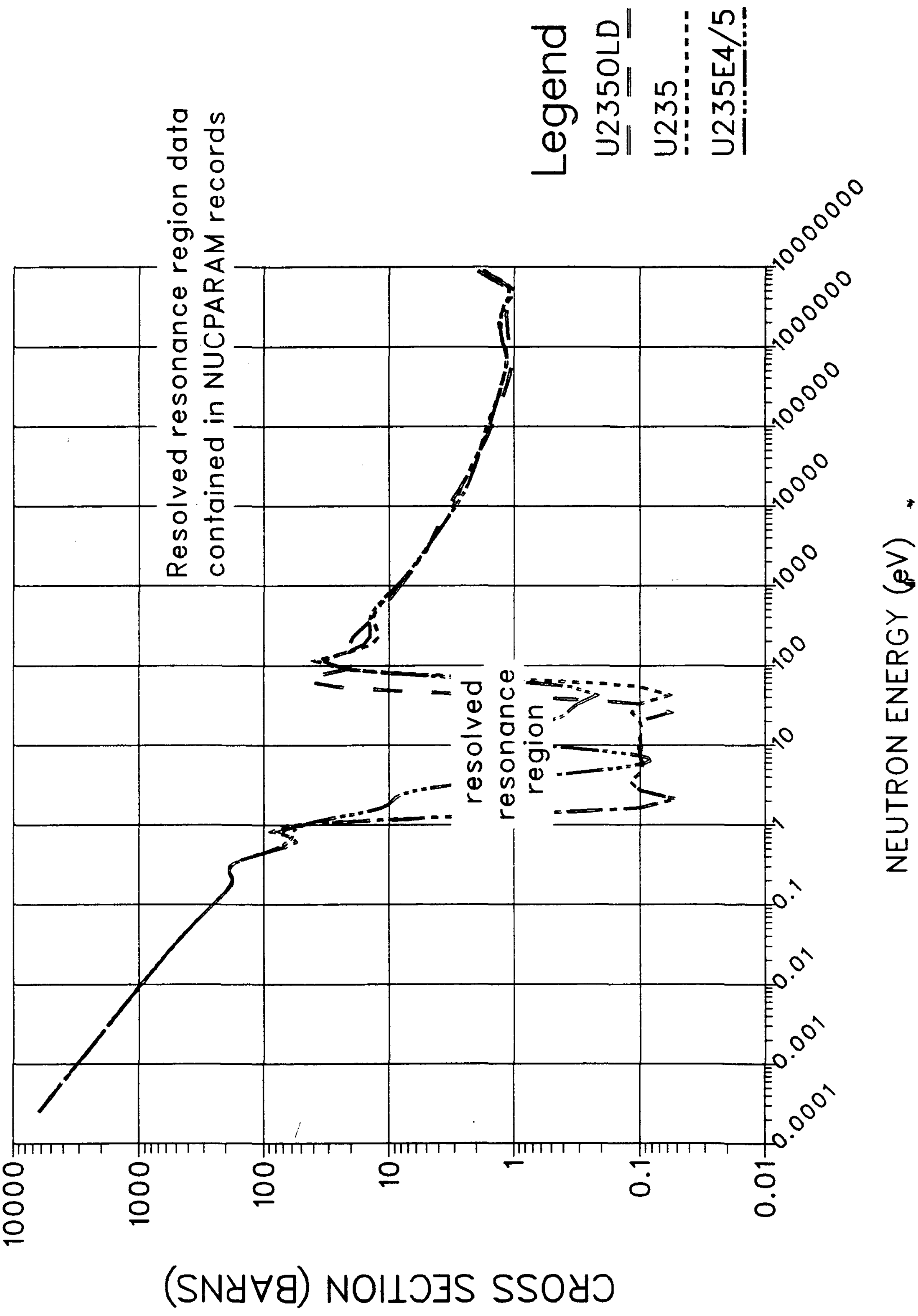


FIGURE 1
MULTIGRP DATA
TOTAL U235 CROSS SECTION
Resolved resonance region data
contained in NUCPARAM records
resolved
resonance
region
Legend
U235OLD
U235
U235E4/5
CROSS SECTION (BARNS)
NEUTRON ENERGY (eV)
10000
1000
100
10
1
0.0001
0.001
0.01
0.1
1
10
100
1000
10000
100000
1000000
10000000
QMS LASERGRAFIX 1200, DATE=85.037, TIME=10.54.57, BADGE=T4983, BOX=L034

FIGURE 2
MULTIGRP DATA
U235 FISSION CROSS SECTION



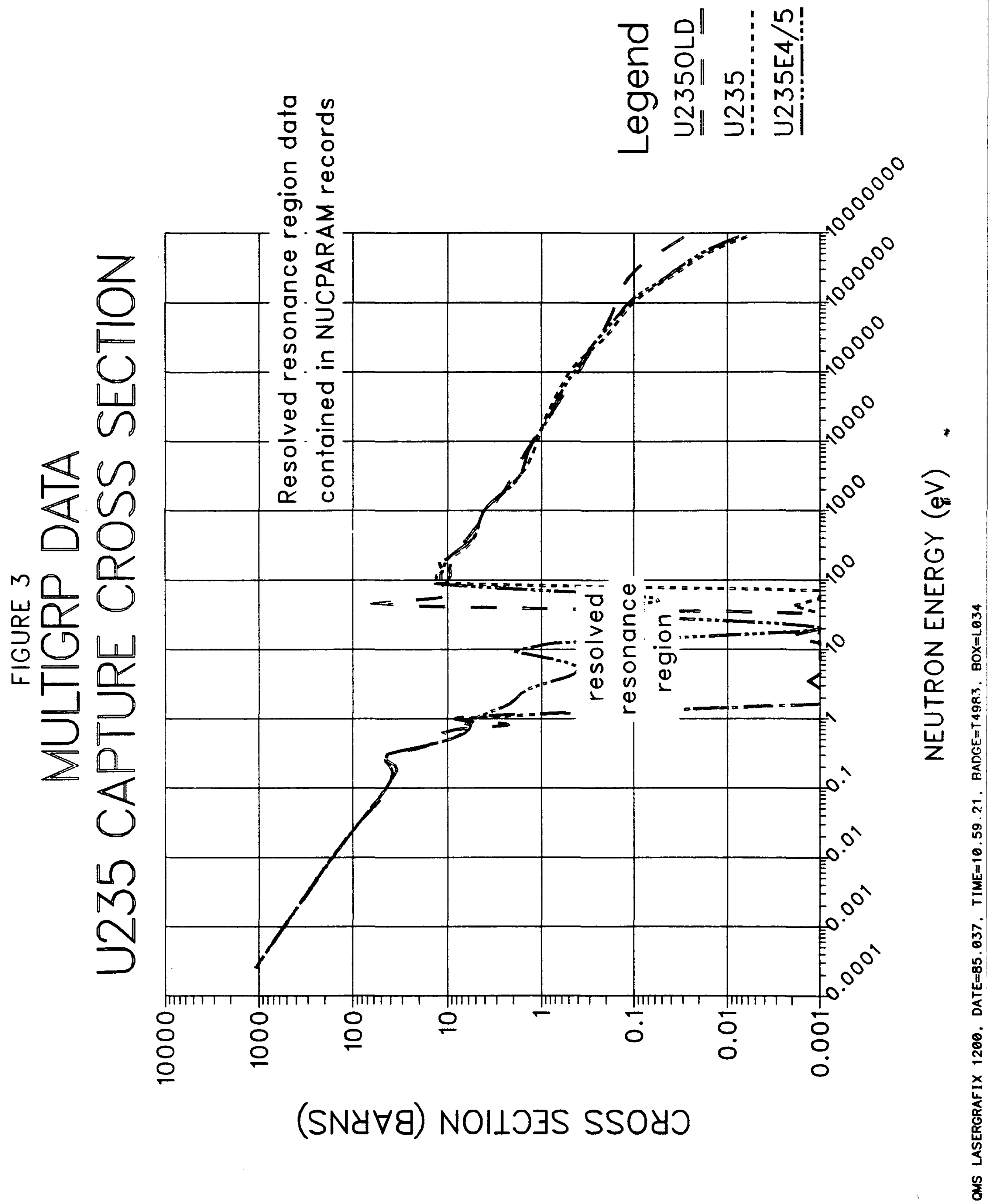


FIGURE 3
MULTIGRP DATA
U235 CAPTURE CROSS SECTION
Resolved resonance region data
contained in NUCPARAM records
resolved
resonance
region
Legend
U235OLD
U235
U235E4/5
CROSS SECTION (BARNS)
NEUTRON ENERGY (eV)
10000
1000
100
10
1
0.1
0.01
0.001
0.0001
0.001
0.01
0.1
1
10
100
1000
10000
100000
1000000
10000000
QMS LASERGRAFIX 1200, DATE=85.037, TIME=10.59.21, BADGE=T4983, BOX=L034

FIGURE 4
MULTIGRP DATA
U235 NU

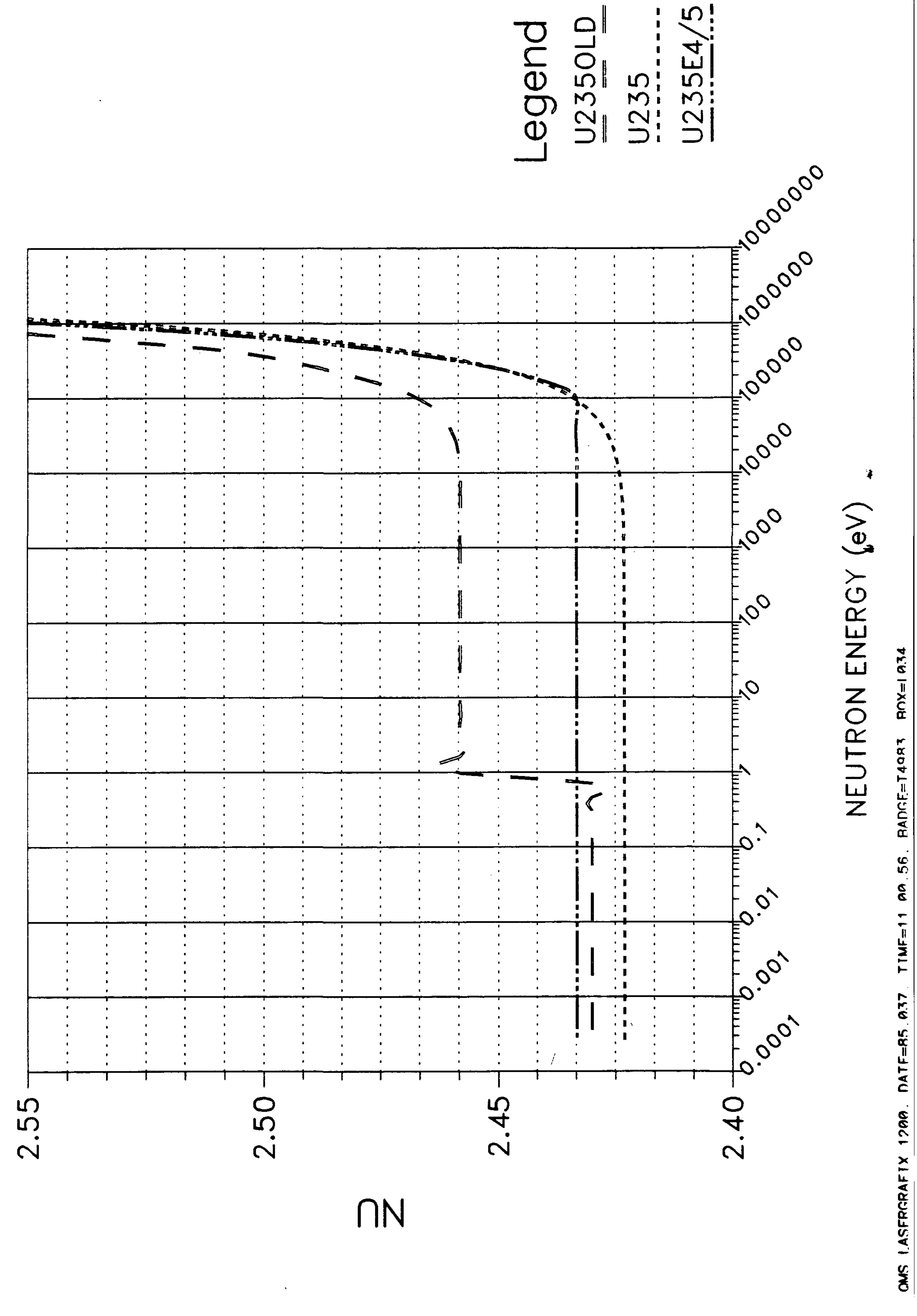
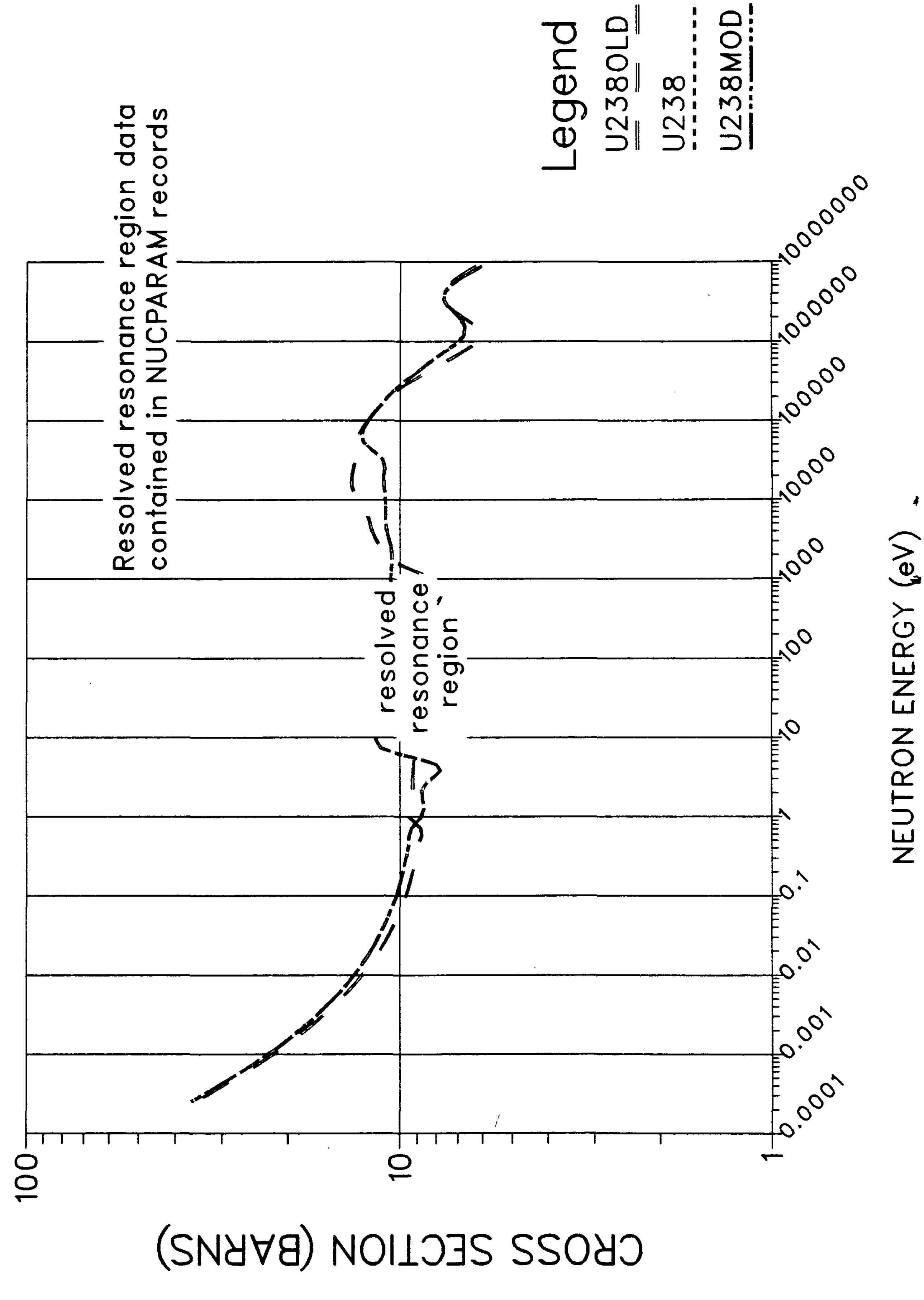


FIGURE 5
MULTIGRP DATA
TOTAL U238 CROSS SECTION



QMS LASERGRAFIX 1200. DATE=85.037. TIME=16.50.15. BADGF=T4983. BOX=L034

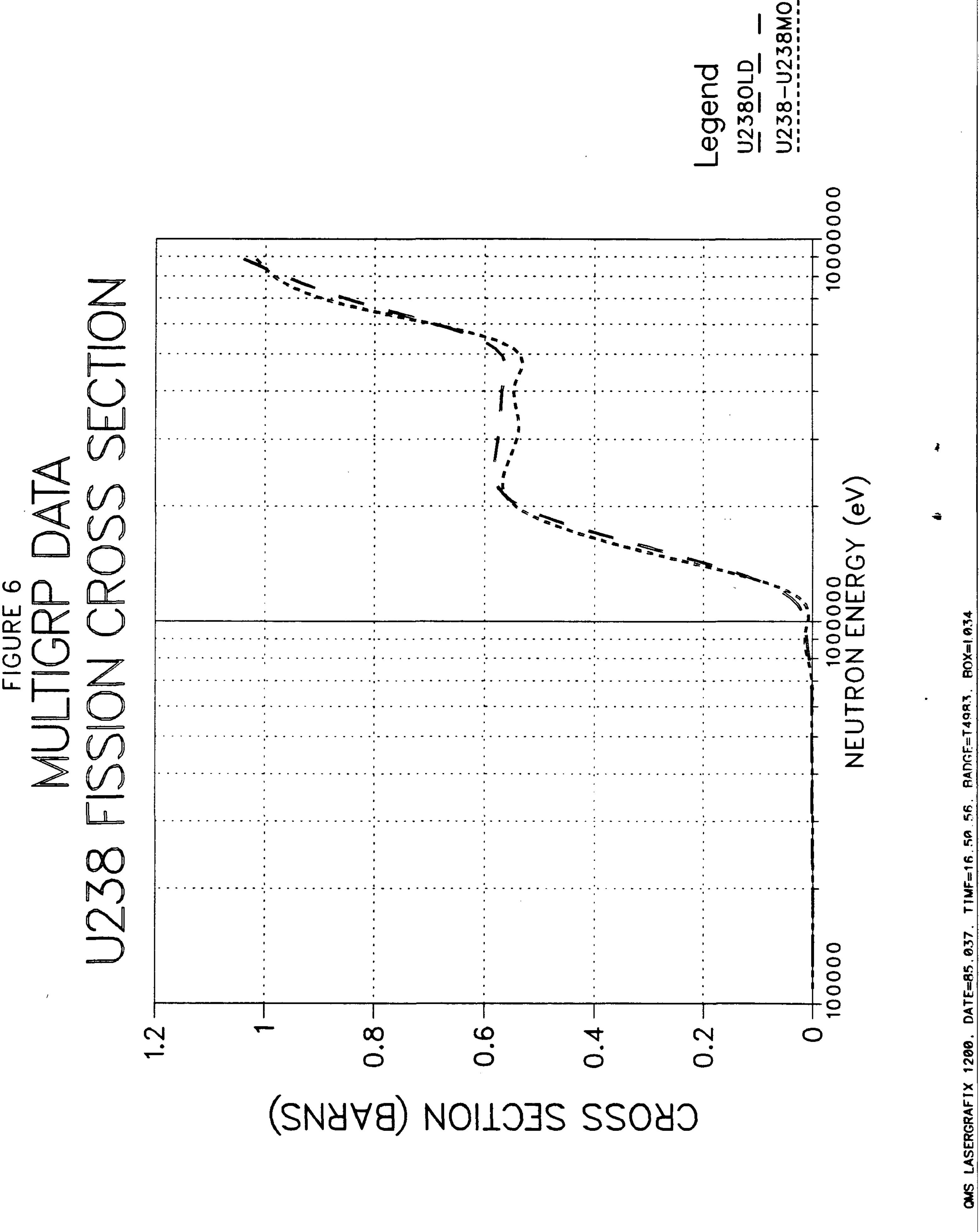


FIGURE 6
MULTIGRP DATA
U238 FISSION CROSS SECTION
Legend
U238OLD
U238-U238MOD
CROSS SECTION (BARNS)
1.2
1
0.8
0.6
0.4
0.2
0
100000
1000000
10000000
NEUTRON ENERGY (eV)
QMS LASERGRAFIX 1200. DATE=85.037. TIME=16.50.56. BADGE=T4983. BOX=1034

FIGURE 7

MULTIGRP DATA

U238 CAPTURE CROSS SECTION

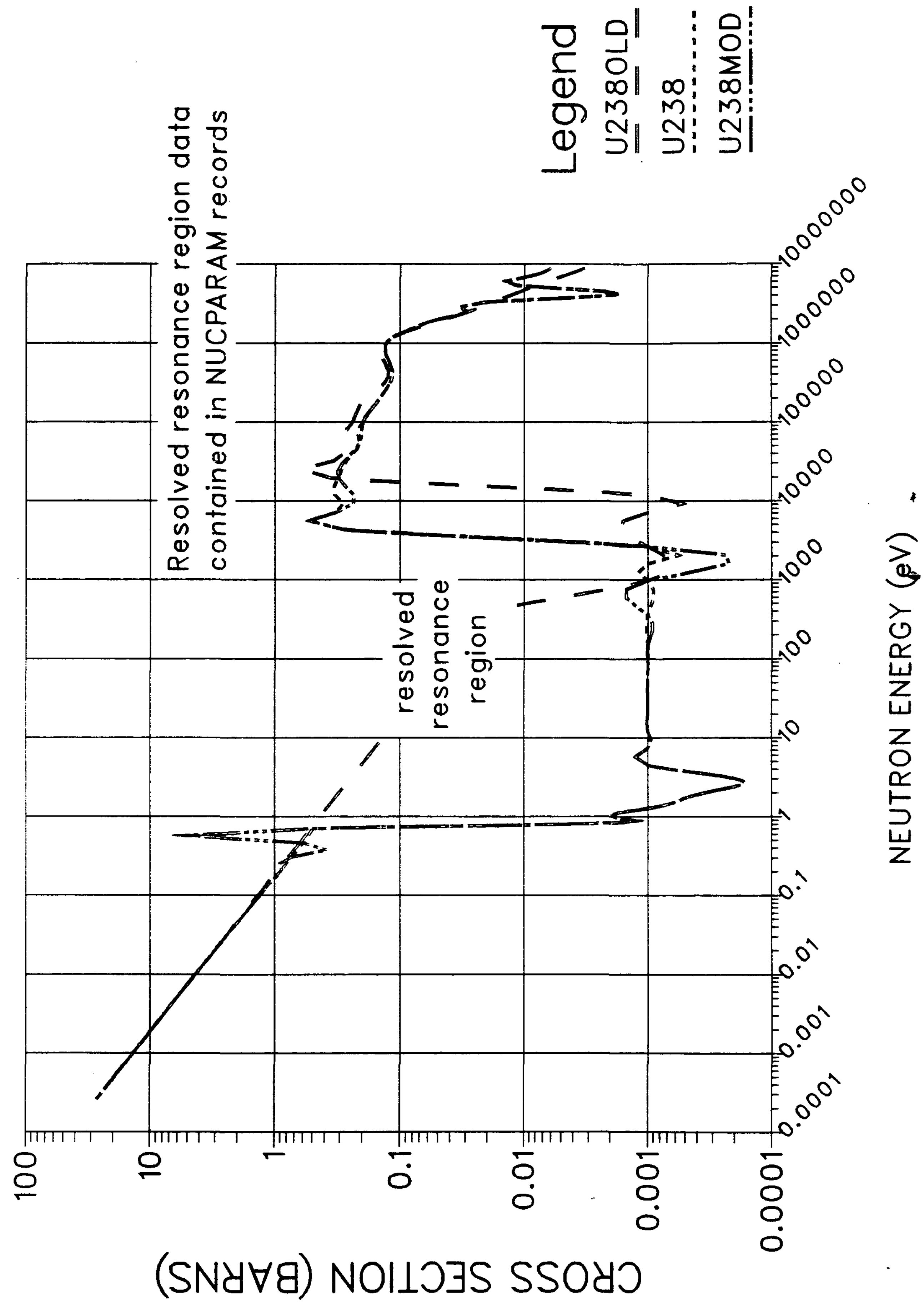
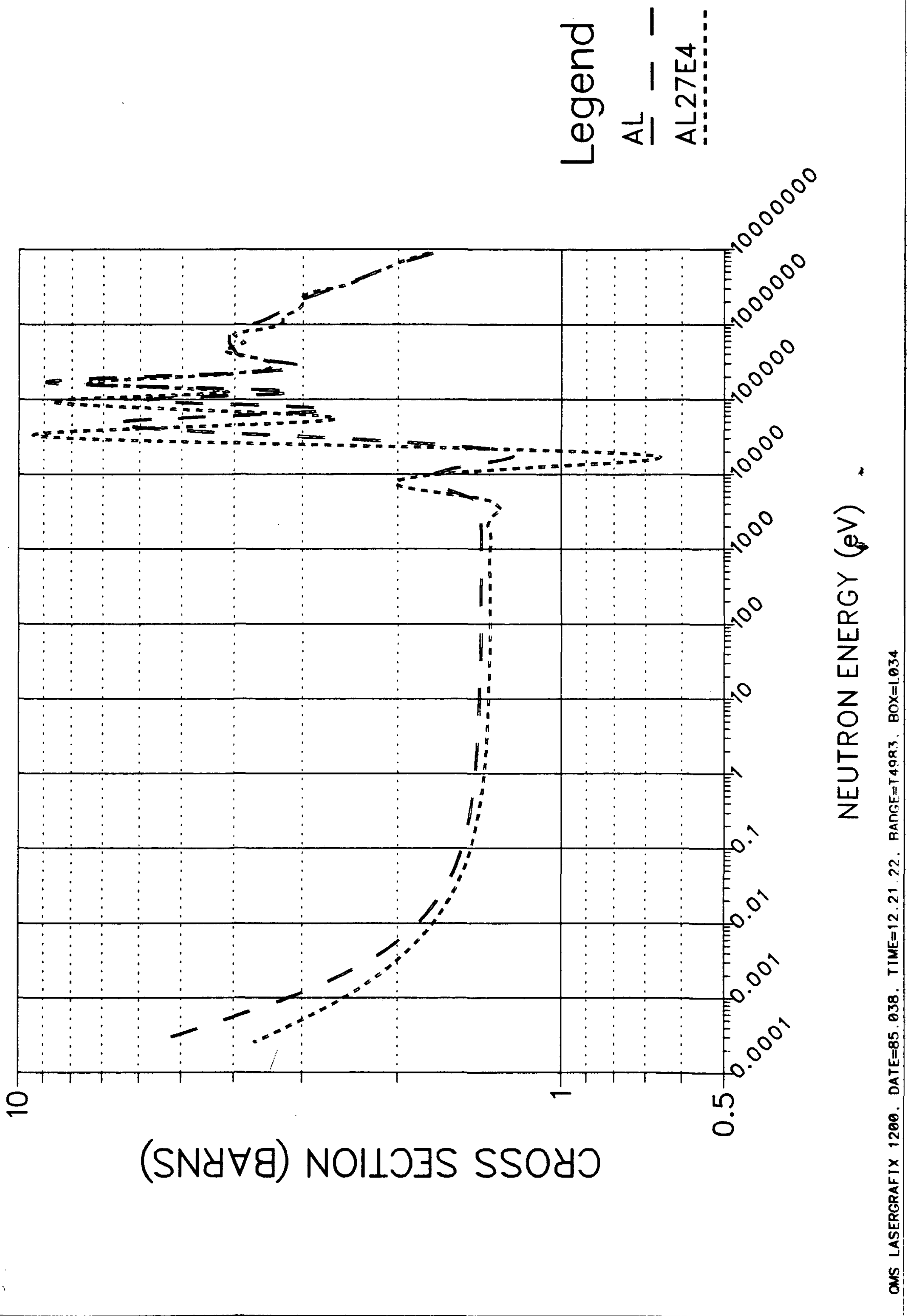


FIGURE 8
MULTIGRP DATA
TOTAL AL CROSS SECTION



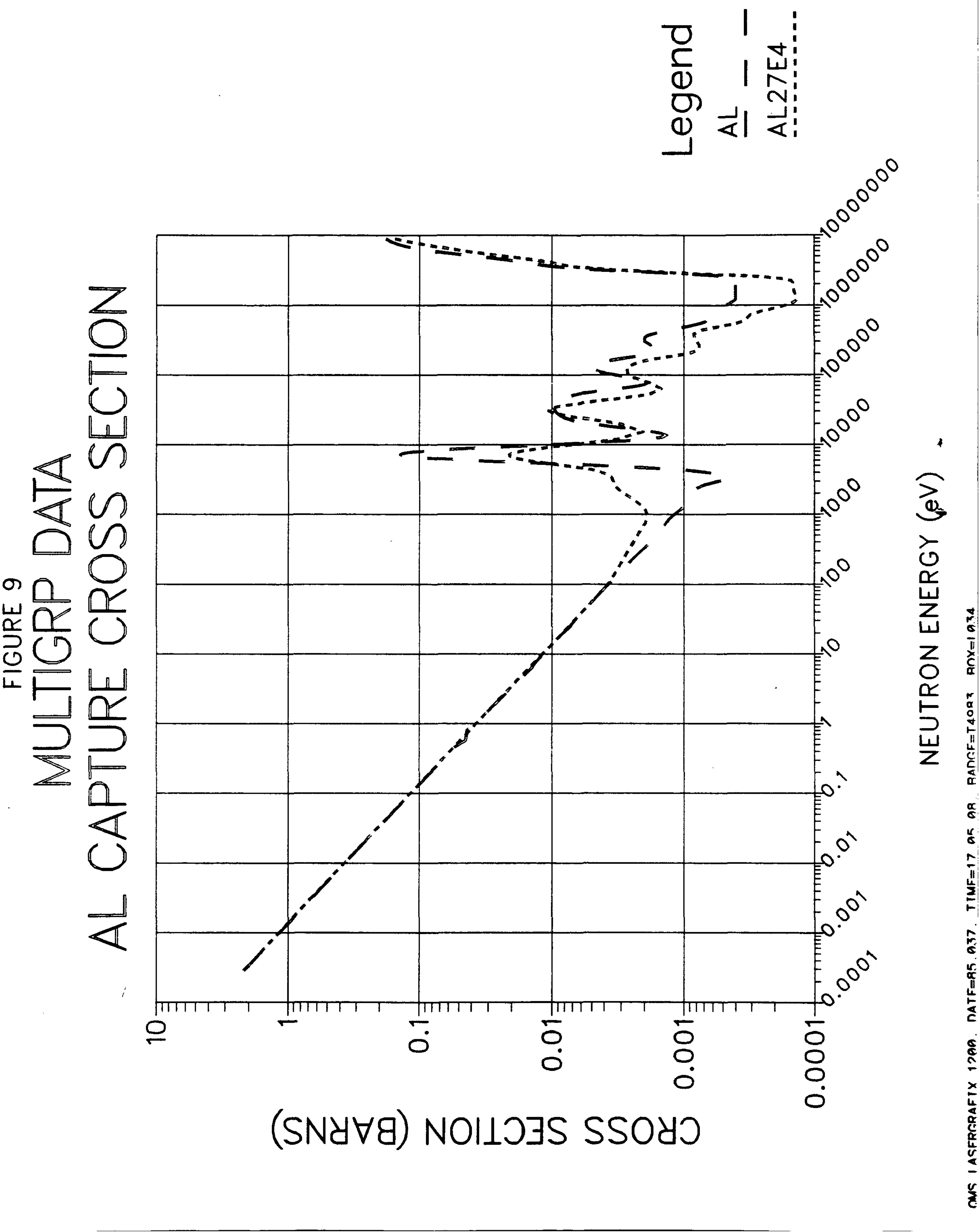


FIGURE 9
MULTIGRP DATA
AL CAPTURE CROSS SECTION
Legend
AL
AL27E4
CROSS SECTION (BARNS)
10
1
0.1
0.01
0.001
0.0001
NEUTRON ENERGY (eV)
0.0001
0.001
0.01
0.1
1
10
100
1000
10000
100000
1000000
10000000

FIGURE 10
MULTIGRP DATA
TOTAL LI6 CROSS SECTION

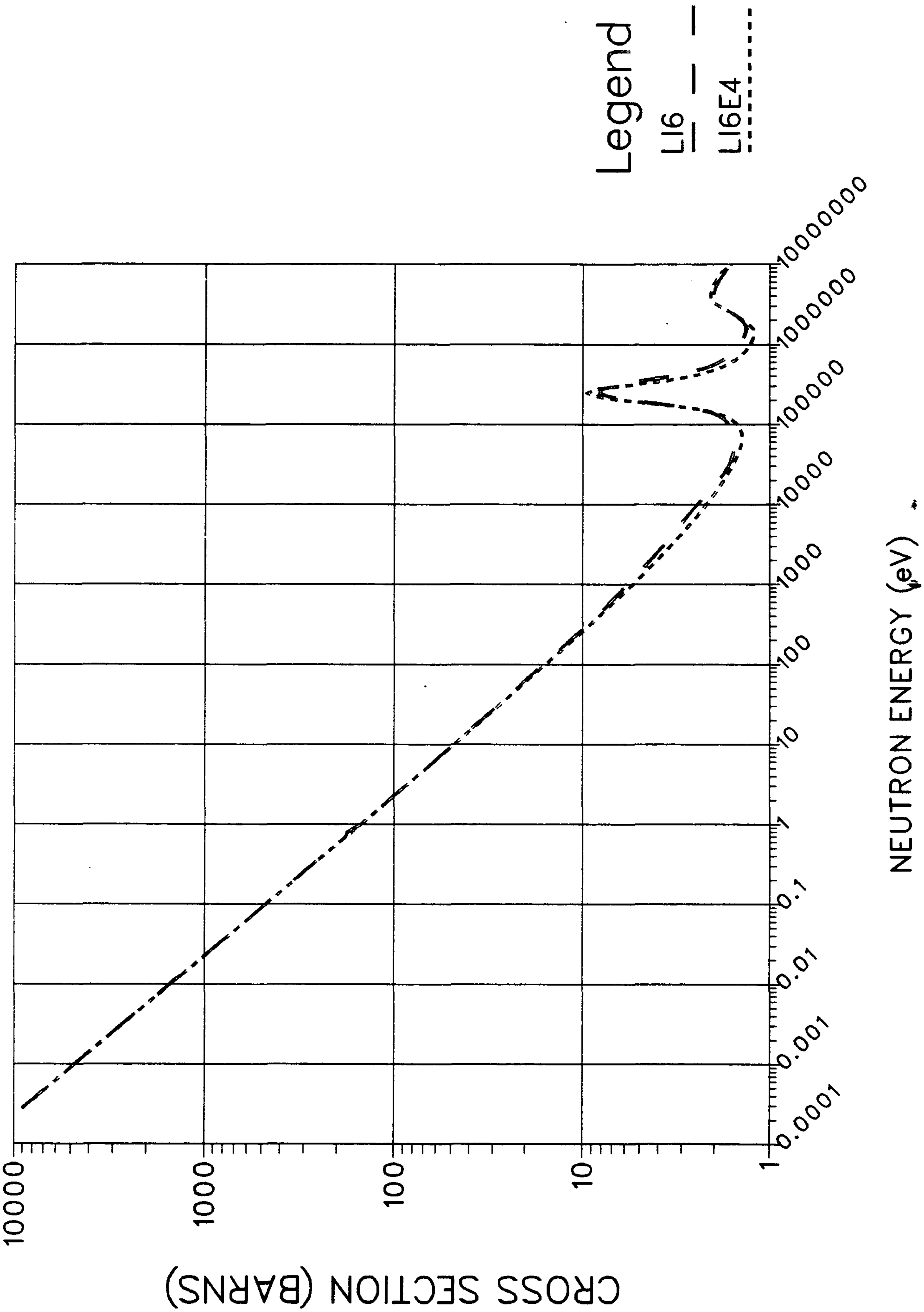
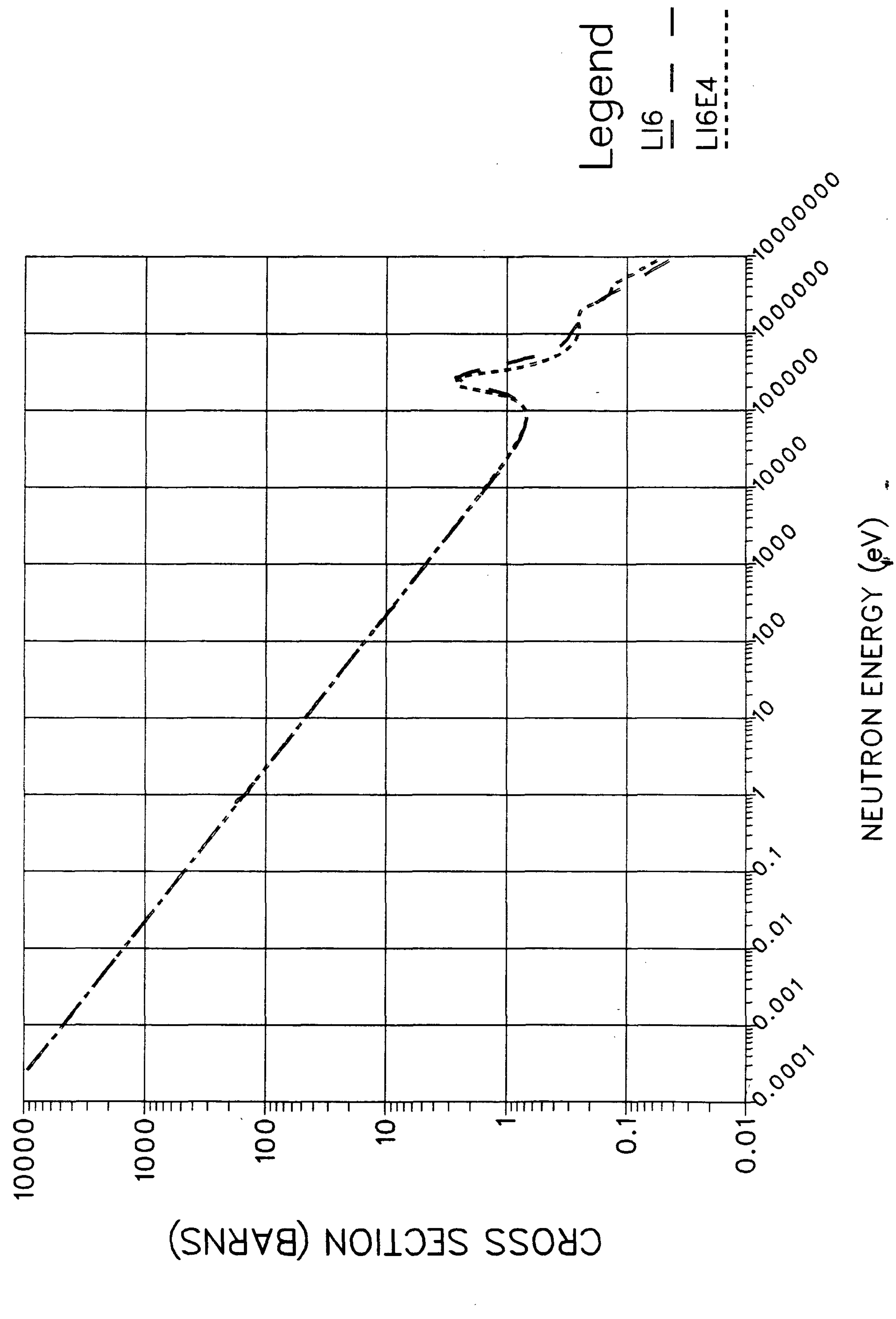


FIGURE 11
MULTIGRP DATA
LI6 CAPTURE CROSS SECTION



QMS LASERGRAFIX 1200, DATE=85.037, TIME=17.06.35, BADGE=T4983, BOX=L034

APPENDIX A

MARK VR CALCULATIONS

Physics parameters were measured for the Mark VR lattice in the Subcritical Experiment (SE) facility in 1971. Documentation for these measurements is not as complete as documentation for the other lattices considered in this study. Buckling results from recent calculations differ significantly from measured Mark VR results and also differ from calculations reported in Reference 4. Since input for the Reference 4 calculations is not documented, comparison with current input is not possible. For these reasons Mark VR results were not included in the main text of this report. However, for completeness Mark VR results are presented in this Appendix.

Mark VR assembly contents used for current calculations are listed in Table A-1. The source for Table A-1 is Reference 11, which is unpublished. The Mark VR assemblies used in the experiments had their end caps machined to bare uranium metal, so no end cap homogenization was performed for either buckling or activation calculations. However, the nickel coating on the slug surface was homogenized with the aluminum cladding. Details of Mark VR assembly geometry are contained in Reference 5 and 11.

Results of Mark VR calculations are listed in Table A-2. Comparisons between current calculations using cross section data from 1972 and calculations made in 1972 and provided in Table A-3. Mark VR results uphold the trend showing the new data provide better results, even when the calculational discrepancy is taken into account. In fact, normalization of current results to match results from 1972 calculations yields 691 μB buckling for use of the new data. This is in good agreement with the measured value 684 μB.

TABLE A-1

MARK VR CONTENTS[1]

REGION NAME	ISOTOPE	CONCENTRATION (10^{24} ATOMS/cm^3)
Coolant[2]		
Inner Housing	Al	6.051 (10^{-2})
	^{10}B	3.044 (10^{-7})
Coolant[2]		
Cladding	Al	5.969 (10^{-2})
	^{10}B	3.253 (10^{-7})
	Ni	1.507 (10^{-3})
Inner Fuel	^{234}U	3.405 (10^{-6})
	^{235}U	4.1659(10^{-4})
	^{236}U	1.544 (10^{-5})
	^{238}U	4.740 (10^{-2})
Cladding	Al	5.989 (10^{-2})
	^{10}B	3.264 (10^{-7})
	Ni	1.136 (10^{-3})
Coolant[2]		
Cladding	Al	5.970 (10^{-2})
	^{10}B	3.254 (10^{-7})
	Ni	1.488 (10^{-3})
Outer Fuel	Same as "Inner Fuel"	Same as "Inner Fuel"
Cladding	Al	5.971 (10^{-2})
	^{10}B	3.254 (10^{-7})
	Ni	1.456 (10^{-3})
Moderator	D	6.625 (10^{-2})
	H	2.594 (10^{-4})
	O	3.325 (10^{-2})

1. Taken from Reference 11.

2. Coolant isotopes and concentrations are the same as the moderator.

TABLE A-2

MARK VR PARAMETERS

Parameter	Experiment	Original[1] SRL Standard	1972 SRL[2] Evaluation	Current[3] SRL Standard	Proposed[4] SRL Standard
BZ B_m^2	684μB	393μB	470μB	534μB	553μB
ρ^{28}	1.31	1.32	1.30	1.29	1.31
C* measured		1.14	1.12	1.12	1.11
C* calculated		1.16	1.12	1.12	1.11
δ^{25}	0.119	0.128	0.133	0.128	0.139
δ^{28}	0.085	0.084	0.081	0.084	0.082

1 U235OLD, U238OLD, LI6, AL

2 U235, U238, LI6, AL

3 U235OLD, U238, LI6, AL

4 U235E4/5, U238MOD, LI6E4, AL27E4

TABLE A-3

COMPARISONS OF PRESENT AND PAST
MARK VR CALCULATIONS

PARAMETER	1972 CALCULATION-REF. 4		CURRENT CALCULATION	
	1972[1] STANDARD	1972[2] EVALUATION	1972[1] STANDARD	1972[2] EVALUATION
B_m^2	532μB	608μB	393μB	470μB
$\rho 28$	1.27	1.28	1.32	1.30
C*	1.14	1.12	1.16	1.12
δ^{25}	0.124	0.132	0.128	0.133
δ^{28}	0.0840	0.0840	0.084	0.081

1 U235OLD, U238OLD, LI6, AL

2 U235, U238, LI6, AL