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Recent Critical Experiments
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SUMMARY

In connection with the program of the measurement of eta for U^{233} , several critical experiments have been performed by R. Gwin and D. W. Magnuson of ORNL with light water solutions of uranyl nitrate (highly enriched in either U^{233} or U^{235}) in an essentially bare sphere 27 inches in diameter. This report presents the results of two multigroup-diffusion-theory calculations for the above experiments performed by C. B. Mills and associates at Los Alamos. Assumed cross sections, material concentrations, detailed neutron balances and a comparison with elementary theory are included. The agreement between the calculated and experimental multiplication constants is excellent for the multigroup calculation but only fair for the elementary calculation. The latter method overestimates the fast leakage so that the computed multiplication constant is less than that found experimentally.

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MULTIGROUP DIFFUSION THEORY CALCULATIONS FOR RECENT CRITICAL EXPERIMENTS

C. W. Nestor, Jr.

Introduction

The results of several elementary calculations on specific critical experiments were presented and discussed in a previous report.¹ The results presented here were obtained at Los Alamos Scientific Laboratory* with the techniques described in a recent report,² and a set of cross sections (with certain revisions) which has yielded good agreement between predicted and experimental multiplication constants for a wide range of reactor types.³

Discussion of Method and Data

The multigroup diffusion theory machine code FIRE is described in detail in a recent report,² so only an outline of its features will be given here. There are three major items of interest, namely, the treatment of slowing down, the treatment of the extrapolated boundary, and the inclusion of the variation with incident neutron energy of the number of neutrons per fission.

The slowing down of neutrons by hydrogen is treated by the Coveyou-Macauley (or Selengut-Goertzel) method.^{2,4} As this procedure overestimates the age, the high-energy transport cross sections of hydrogen have been adjusted to produce an age to indium resonance of 30.2 cm^2 .

The extrapolation distance is taken to be $0.7104 \lambda_{tr}^{(g)}$, where $\lambda_{tr}^{(g)}$ is the transport mean free path in energy group g . The large variation in the hydrogen cross section leads to a considerable variation in the extrapolation distance and consequently a large variation in the effective buckling over the energy range considered.

* By C. B. Mills and associates.

An increase in the number of neutrons per fission with energy of the incident neutron of about 1 per 8 Mev has been observed for U^{235} ; (5) while explicit account of this variation is not included in the FIRE code, the fission cross sections in the upper energy groups have been modified appropriately. For this reason, the "fission" cross sections in the listing (Appendix 1) are larger than the corresponding absorption cross sections in some groups. The variation of η with energy (based on the variation of α with energy) is given in Fig. 1.

Results

Values of ν_c , the number of neutrons per fission required for criticality, are listed in Table 1 together with calculated values of η , experimental values of ν and η , and calculated and experimental values of the ratio of $\eta(U^{233})/\eta(U^{235})$. Cumulative leakage and fissioning spectra are plotted in Figs. 2 and 3. A detailed neutron balance for each reactor is given in Table 2. If the assumption is made that all nonthermal absorptions occur just above thermal energy the fast nonleakage probability, $K(B^2)$, is just

$$K(B^2) = 1 - L_f \quad (1)$$

where L_f is the fast leakage per source neutron. A calculation of the fast nonleakage probability (tabulated below under "Elementary") from the empirical formula of Trubey, Moran and Weinberg⁶ leads to a smaller value for $K(B^2)$ than is calculated by the multigroup method. Values of thermal utilization, products of fast effect and resonance escape probability, thermal and fast nonleakage probabilities, and effective multiplication constants deduced from the multigroup results are given in Table 3.

An estimation of the effect of the value of the neutron age on the computed value of η can be made with the equation

$$\frac{\eta_1}{\eta_2} = \frac{1 + B^2 \tau_1}{1 + B^2 \tau_2} \quad (2)$$

For instance, if $\tau_1 = 27.0 \text{ cm}^2$, (the theoretical age in H_2O), $\tau_2 = 30.2 \text{ cm}^2$, and $\eta_2 = 2.27$, then $\eta_1 = 2.23$; for U^{235} , with $\eta_2 = 2.07$, then $\eta_1 = 2.03$.

Acknowledgments:

It is a pleasure to acknowledge the helpful cooperation and major assistance provided by C. B. Mills of Los Alamos Scientific Laboratory in performing the multigroup calculations.

Table 1. Comparison of Calculated and Experimental Values of ν and η

Fissile material	ν Calculated	ν BNL-325 (WCS)*	$1 + \alpha$ Thermal avg.	η Thermal avg. cal- culated	η 2200 m/sec BNL-325 (WCS)*	η_3/η_5 Calcu- lated	η_3/η_5 BNL-325 (WCS)*
U^{233}	2.490	2.51 ± 0.03	1.098	2.268	2.28 ± 0.02	-	
U^{235}	2.464	2.47 ± 0.03	1.194	2.064	2.07 ± 0.02	1.099	1.10 ± 0.02

* "World Consistent Set"

Table 2. Multigroup Neutron Balance in 27-inch Diameter Reactor
(Based on 100 Neutrons Absorbed by Fuel)

Losses:		U^{233}	U^{235}
Absorptions			
fuel	fast	9.4	6.9
	thermal	90.6	93.1
nonfuel	fast	5.2	4.8
	thermal	80.1	63.9
Leakages			
	fast	36.7	33.2
	thermal	4.6	3.7
Total losses		226.6	205.6
Productions:			
fast fission		20.5	13.0
thermal fission		206.1	192.6
Total productions		226.6	205.6
η:			
Fast		2.17	1.88
Thermal		2.27	2.07
Average		2.27	2.06

Table 3. Comparison of Multigroup and Elementary Calculations

Fissile Material:	Elementary		Multigroup	
	23	25	23	25
f	0.5323	0.5943	0.5323	0.5943
ϵp	1.016 ⁽⁷⁾	1.006 ⁽⁷⁾	1.014	0.9922
η	2.27	2.07	2.268	2.064
$K(B^2)$	0.8150	0.8150	0.8381	0.8384
P_{NL}^{Th}	0.9735	0.9772	0.9735	0.9772
k_e	0.974	0.986	0.999	0.997

Appendix 1

1. Reactor Characteristics and Nuclear Data

Sphere of 34.6 cm radius (computed from volume)

$B^2 = 7.1705 \times 10^{-3} \text{ cm}^{-2}$ (2.5 cm extrapolation distance) for elementary calculations

Element	Concentration, atoms/barn centimeter	
	<u>U²³³</u>	<u>U²³⁵</u>
U fissile	4.35×10^{-5}	4.81×10^{-5}
H	6.62×10^{-2}	6.63×10^{-2}
O	3.31×10^{-2}	3.315×10^{-2}
N	1.17×10^{-4}	1.75×10^{-4}

2. FIRE Group Structure

Group	Upper Energy Limit, ev	Group	Upper Energy limit, ev
1	$10^7 (u = 0)$	10	61.4
2	3×10^6	10	22.6
3	1.4	12	8.32
4	0.9	13	3.06
5	0.4	14	1.13
6	0.1	15	0.414
7	1.7×10^4	16	0.152
8	3.35×10^3	17	0.056
9	454	18 (Thermal)	0.0252

3. Microscopic Cross Sections

U²³³

XI TOT	ABSPT	TRANSPT	FISSION
0.0000 00	1.8000 00	4.2500 00	2.1139 00
0.0000 00	1.9100 00	4.5000 00	1.9763 00
0.0000 00	1.9999 00	4.8000 00	1.9731 00
0.0000 00	2.0900 00	5.6999 00	1.9943 00
0.0000 00	2.4700 00	8.3999 00	2.2668 00
0.0000 00	3.7800 00	1.2000 01	3.2428 00
0.0000 00	6.7700 00	1.6819 01	6.0000 00
0.0000 00	1.0200 01	2.0250 01	9.0000 00
0.0000 00	2.7400 01	3.7450 01	2.4000 01
0.0000 00	5.6950 01	6.7000 01	5.3199 01
0.0000 00	1.1895 02	1.2900 02	1.0200 02
0.0000 00	1.0995 02	1.2000 02	9.0000 01
0.0000 00	3.4394 02	3.5400 02	2.9000 02
0.0000 00	1.6095 02	1.7100 02	1.3200 02
0.0000 00	2.0480 02	2.1480 02	1.8319 02
0.0000 00	3.1880 02	3.2879 02	2.9439 02
0.0000 00	5.0220 02	5.1219 02	4.5879 02
0.0000 00	5.1600 02	5.2600 02	4.7000 02

INEL I+1	INEL I+2	INELI+3	INELI+4	INELI+5	INELI+6
2.0000-01	2.6999-01	4.4999-01	3.0999-01	3.9999-02	
1.8000-01	5.0000-01	3.4999-01	5.0000-02	0.0000 00	
4.4999-01	3.0000-01	5.9999-02	0.0000 00	0.0000 00	
2.9000-01	5.0000-02	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
0.0000 00	0.0000 00	0.0000 00	0.0000 00	0.0000 00	

U-235

XI TOT	ABSPT	TRANSPT	FISSION
0.0000 00	1.2600 00	4.2500 00	1.4518 00
0.0000 00	1.3000 00	4.5000 00	1.3044 00
0.0000 00	1.3300 00	4.6500 00	1.2600 00
0.0000 00	1.3500 00	5.1999 00	1.2195 00
0.0000 00	1.6600 00	7.8999 00	1.4359 00
0.0000 00	3.1500 00	1.2399 01	2.5000 00
0.0000 00	5.5000 00	1.5099 01	4.1999 00
0.0000 00	1.1099 01	2.1100 01	7.9020 00
0.0000 00	2.5500 01	3.5500 01	1.7497 01
0.0000 00	9.3569 01	1.0357 02	5.9714 01
0.0000 00	1.1150 02	1.2150 02	5.8151 01
0.0000 00	4.7040 01	5.7039 01	2.4163 01
0.0000 00	3.2550 01	4.2549 01	2.4653 01
0.0000 00	7.3100 01	8.3099 01	6.1591 01
0.0000 00	2.0440 02	2.1439 02	1.6579 02
0.0000 00	3.1359 02	3.2360 02	2.6591 02
0.0000 00	5.6130 02	5.7129 02	4.7081 02
0.0000 00	6.0400 02	6.1400 02	5.0587 02

INEL I+1	INEL I+2	INELI+3	INELI+4	INELI+5	INELI+6
2.6999-01	3.7000-01	6.4999-01	4.4000-01	5.9999-02	
2.4000-01	6.6999-01	4.4999-01	6.9999-02	0.0000 00	
5.5000-01	4.0000-01	6.9999-02	0.0000 00	0.0000 00	
3.4999-01	7.9999-02	0.0000 00	0.0000 00	0.0000 00	
7.9999-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
5.0000-02	0.0000 00	0.0000 00	0.0000 00	0.0000 00	
0.0000 00	0.0000 00	0.0000 00	0.0000 00	0.0000 00	

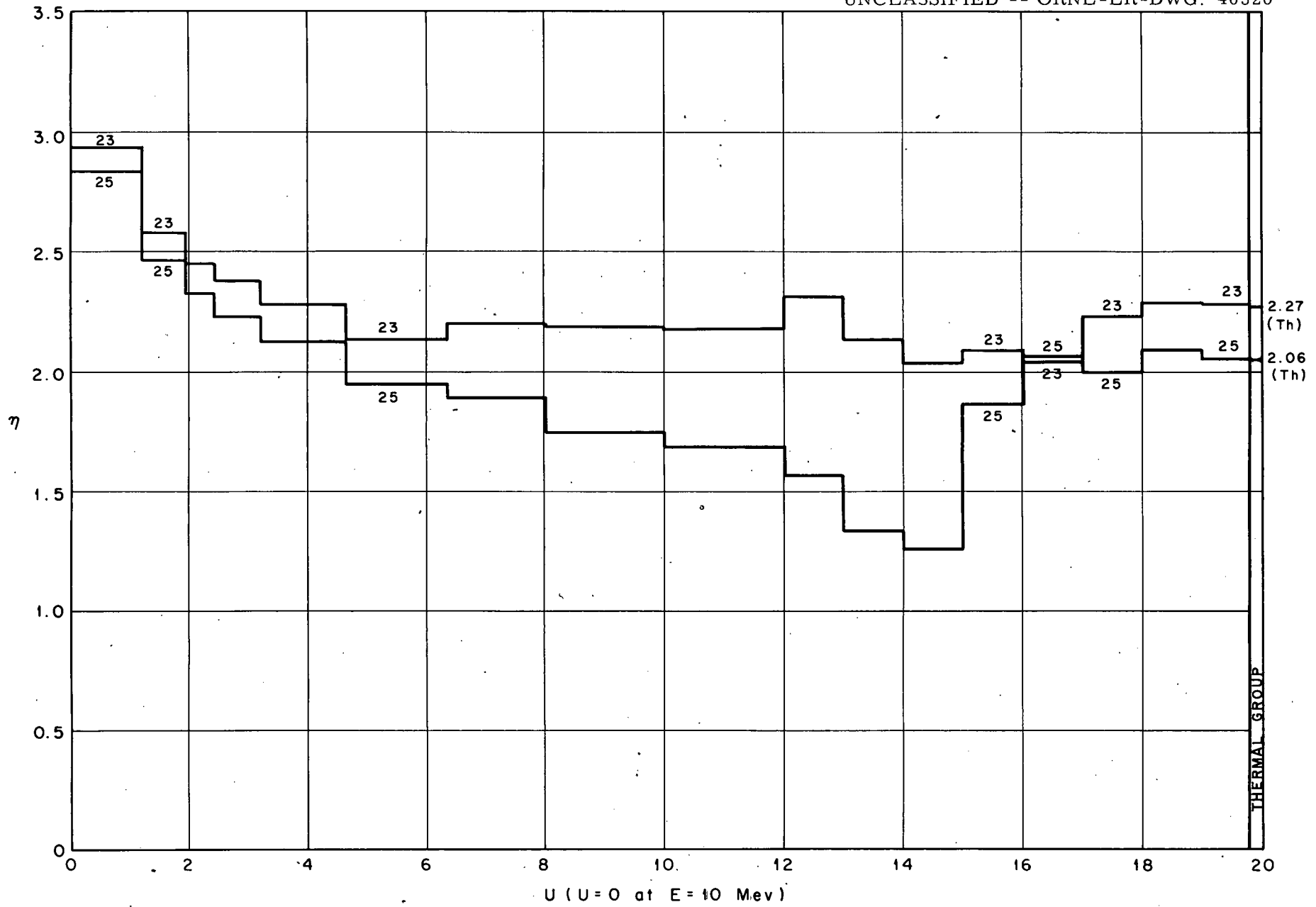


Fig. 1. Variation of η_{23} and η_{25} with Lethargy.

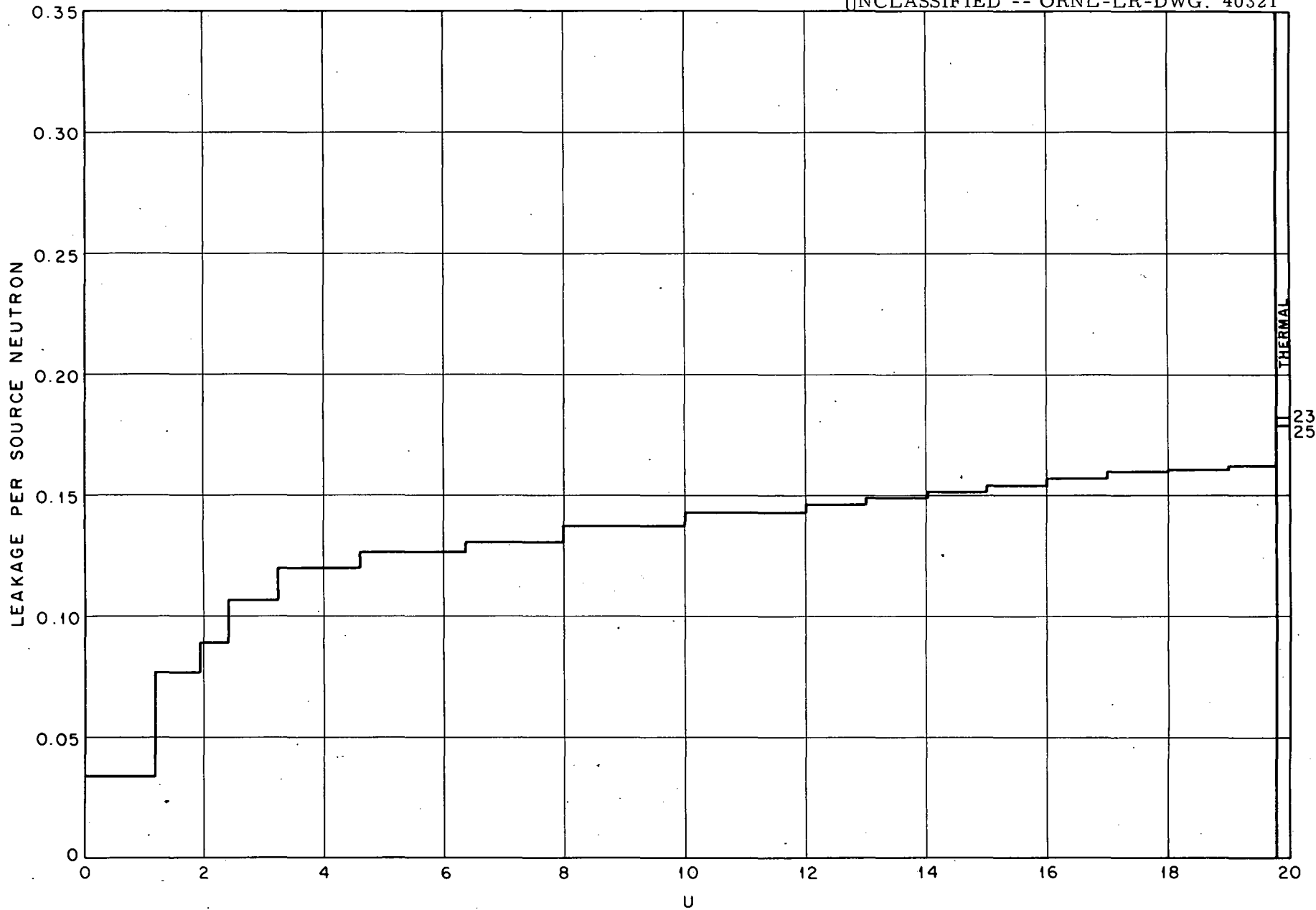


Fig. 2. Cumulative Leakage Spectrum.

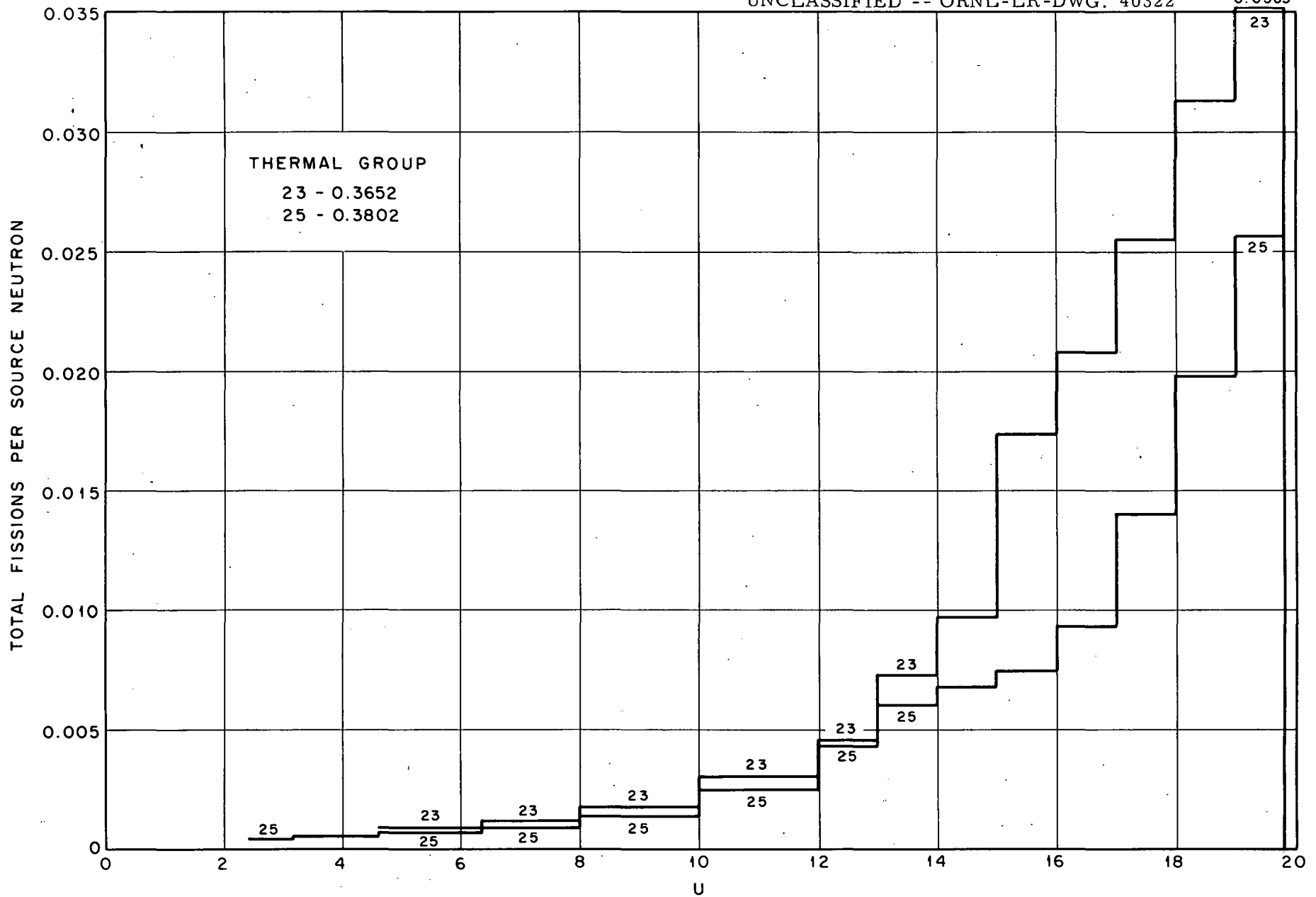


Fig. 3. Cumulative Fast Fissioning Spectrum

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