ORNL-TM-4257
(ENDF-191)

ADLER-III: A PROGRAM TO CALCULATE CROSS SECTIONS FROM ADLER-ADLER RESONANCE PARAMETERS

R. Q. Wright

OAK RIDGE NATIONAL LABORATORY

OPERATED BY UNION CARBIDE CORPORATION • FOR THE U.S. ATOMIC ENERGY COMMISSION

ORNL-TM-4257
(ENDF-191)

Contract No. W-7405-eng-26

NEUTRON PHYSICS DIVISION

ADLER-III: A PROGRAM TO CALCULATE CROSS SECTIONS FROM ADLER-ADLER RESONANCE PARAMETERS*

R. Q. Wright**

JUNE 1973

*Packaged in RSIC as PSR-58

**Mathematics Division

NOTE:

This work partially supported by
DEFENSE NUCLEAR AGENCY
Under Subtasks PB052-03
and PE074

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

TABLE OF CONTENTS

LIST OF FIGURES

ACKNOWLEDGMENTS

The author is grateful to G. deSaussure and L. M. Petrie for many valuable discussions during the course of this work and to G. B. Raiford for typing the report.

ABSTRACT

ADLER-III is a revision and extension of the ADLER program described in "ENDF/B Processing Codes for the Resonance Region," BNL-50296 (ENDF 148) by M. R. Bhat. The program calculates total, capture, and fission cross sections according to the Adler-Adler formalism using the corresponding parameters from file 2 of ENDF/B data. Doppler broadening is provided through the use of the ψ and φ functions. The program can read the resonance parameters from an ENDF/B tape (BCD mode) or the resonance parameters may be input on cards if desired.

COMPUTER CODE ABSTRACT

1. Name of Code: ADLER-III
2. Computer for which code is designed: IBM 360/75 or 360/91
3. Nature of problem solved: ADLER-III calculates total, capture and fission cross-sections from the corresponding Adler-Adler parameters in the ENDF/B File 2 and also Doppler broadens cross-sections.
4. Method of solution: Adler-Adler formalism. Doppler broadening is provided through the use of the ψ and φ functions.
5. Restrictions on the complexity of the problem: This program can handle resonance data up to a maximum of 10 isotopes with a total number of 500 resonances. It further assumes that the resolved resonance parameters are given for one energy range which is the same for all the isotopes of an element. The mesh points at which the cross-sections are calculated can be varied.
6. Typical running time: Calculations of cross-sections of one isotope with 70 resonances and 40 mesh points between resonances takes: (360/91)

   (a) without Doppler broadening, 7 secs

   (b) with Doppler broadening, 21 secs
7. Unusual features of the program: None
8. Related auxiliary program: None
9. Status: ADLER-III is in production use on the IBM 360/91 at the ORNL Computing Center.
10. Reference: R. Q. Wright, "ADLER-III: A Program to Calculate Cross Sections from Adler-Adler Resonance Parameters," ORNL-TM-4257 (ENDF-191) (1973).

11. Machine requirements: ADLER-III requires 120K bytes on the IBM-360/91.

12. Programming language: FORTRAN IV.

13. Operating system: IBM OS/360 with the FORTRAN H compiler.

14. Programming information: ADLER-III consists of 11 subroutines on $\sim$ 1000 source cards. Very little (if any) effort should be required to convert the program to another operating system (e.g., CDC-6600) since the program is essentially machine independent.

15. User information: The code may be obtained from the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory.

I. DESCRIPTION OF CALCULATIONAL PROCEDURE

ADLER-III is a revision and extension of the ADLER program described in ref. 1. The program calculates total, capture and fission cross sections according to the Adler-Adler formalism using the corresponding parameters from file 2 of ENDF/B data.[2] This work is part of an effort toward providing Adler-Adler processing capability in the ORNL AMPX[3] system.

The Adler-Adler formulae have been discussed in a number of reports.[4-8] The formulae are:

$$\sigma^{(x)}(E) = \frac{2C}{E}(1 - \cos\omega)$$

$$+ \frac{C}{\sqrt{E}} \sum_k \frac{\nu_k (G_k^{(x)} \cos\omega + H_k^{(x)} \sin\omega) + (\mu_k - E)(H_k^{(x)} \cos\omega - G_k^{(x)} \sin\omega)}{(\mu_k - E)^2 + \nu_k^2}$$

$$+ \frac{C}{\sqrt{E}} \left(A_1^{(x)} + \frac{A_2^{(x)}}{E} + \frac{A_3^{(x)}}{E^2} + \frac{A_4^{(x)}}{E^3} + B_1^{(x)} E + B_2^{(x)} E^2\right) \qquad (1)$$

where

$\sigma^{(x)}(E)$ = cross section for reaction type "x" at energy E.

$C = \pi\lambda\!\!\!^{-2} E;\quad \lambda\!\!\!^{-} = \frac{1}{k}$

where k is the wave number of the incident neutron in the center-of-mass system.

$k = 2.196771\times10^{-3}\left(\frac{A}{A+1}\right)\sqrt{E(\text{eV})}$

A = the atomic weight ratio.

$\frac{2C}{E}(1-\cos\omega)$ = potential scattering cross section; this term is non-zero only for scattering and total cross sections.

$\omega = 2k\hat{a}$ where $\hat{a}$ corresponds to the effective scattering radius in units of 10^{-12} cm;

$= 2WN\sqrt{E(eV)}$ and $WN = 2.196771(\frac{A}{A+1.0}) \times 10^{-3} \times \hat{a}$,

$A_1^{(x)}, A_2^{(x)}, A_3^{(x)}, A_4^{(x)}$ and $B_1^{(x)}, B_2^{(x)}$ are background parameters which give contributions due to "tails" of resonances lying outside of the energy range under consideration. They are associated with the type of reaction "x".

$G_k^{(x)}, H_k^{(x)}$ and ν_k, μ_k are Adler-Adler parameters, characteristic of the k-th resonance and the type of reaction indicated by "x".

Procedure

(1) After all the data for one element are read in, the resonance energies of all the isotopes are arranged in increasing order.

(2) For each pair of resonances which lie within the limits of the energy range in the input data, a variable number of energy points which lie between these two resonances are generated, and the total, capture and fission cross sections are calculated and printed out. This is done for all the resonances in an element.

In general, the number of points per resonance is

$$NP = 4*MM.$$

Presently 40 points per resonance are used but the mesh may be varied by setting MM (card ADLR278) to a different value.

(3) Only s-wave resonances are handled by the program.

(4) The Doppler broadened cross section is written as:[9]

$$E^{1/2}\sigma_{\Delta}(E) = \frac{1}{\Delta\sqrt{\pi}}\int_0^{\infty} E'^{1/2}\sigma(E')e^{-\left(\frac{E-E'}{\Delta}\right)^2} dE' \qquad (2)$$

where

$$\Delta = \sqrt{\frac{4KTE}{A}}$$

$\sigma_{\Delta}(E)$ = the Doppler broadened cross section at energy E.

K = Boltzmann's constant (8.6167×10^{-5} eV/°K).

T = effective temperature in °K.

A = atomic weight ratio.

The Doppler broadened cross section is obtained by performing the convolution (Eq. 2) on the cross section formula (Eq. 1):[7]

$$\sigma_{\Delta}^{(x)}(E) = \frac{2C}{E}(1 - \cos\omega)$$

$$+ \frac{C}{\sqrt{E}}\sum_k \frac{\psi(X,\beta)}{\nu_k}\left\{G_k^{(x)}\cos\omega + H_k^{(x)}\sin\omega\right\}$$

$$+ \frac{C}{\sqrt{E}}\sum_k \frac{\varphi(X,\beta)}{\nu_k}\left\{H_k^{(x)}\cos\omega - G_k^{(x)}\sin\omega\right\} \qquad (3)$$

where $\beta = \frac{\Delta}{\nu}$ and $X = \frac{\mu - E}{\nu}$.

Our definitions of the ψ and φ functions follow that of Dresner:[10] first we define the variable $\theta = \frac{2}{\beta}$ or $\theta = \frac{\Gamma}{\Delta}$

then

$$\psi(x,\theta) \equiv \frac{\theta}{2\sqrt{\pi}} \int_{-\infty}^{\infty} \frac{\exp\left[-\frac{1}{4}\theta^2(x-y)^2\right]}{1+y^2}\,dy \tag{4}$$

$$\varphi(x,\theta) \equiv \frac{\theta}{2\sqrt{\pi}} \int_{-\infty}^{\infty} \frac{\exp\left[-\frac{1}{4}\theta^2(x-y)^2\right]}{1+y^2}\,y\,dy \tag{5}$$

Note that the function φ differs from the usual χ function in that χ has an additional factor of 2 not present in φ.

A discussion of the properties of the ψ and χ functions may be found in ref. 10. Numerical techniques for the evaluation of the ψ and χ functions are discussed in ref. 11.

In ADLER-III the ψ and φ functions are evaluated using subroutine VOGAM. The arguments to this subroutine are X, β, ψ, and φ. We have replaced subroutine PFCN (in ADLER) since VOGAM is considerably faster (on the IBM-360/91) and gives equivalent results. VOGAM uses three different numerical techniques: Taylor's series, asymptotic series, and a rational approximation. The accuracy of the results from subroutine VOGAM has been verified by comparison with subroutine PFCN and with results given in ref. 11.

The potential scattering term (non-zero only for scattering and total) and the background terms are not Doppler broadened, as they are slowly varying and not affected by Doppler broadening. The code assumes $\omega = 0$ for fission and capture so equation (3) reduces to:

$$\sigma_{\Delta}^{(x)}(E) = \frac{C}{\sqrt{E}} \sum_k \frac{1}{\nu_k} \left\{ G_k^{(x)} \psi(X,\beta) + H_k^{(x)} \varphi(X,\beta) \right\} \tag{6}$$

which is identical to equation (7) of ref. 7 (except for the factor C which is included in G and H). Thus, ADLER-III uses equation (3) for the total cross section and equation (6) for the fission and capture cross sections.

II. INPUT DESCRIPTION

This section describes the input data and designates the columns to be used for each item. Standard FORTRAN conventions used to describe the data formats and symbol names, as they appear in the code, are also listed.

Card No. 1 (7A4,A2,E10.0,8X,I2)

Item	Cols.	Name	Description
1	1-30	TITLE	Any suitable title for the case.
2	31-40	TEFF	Effective temperature of the sample in degrees Kelvin. Also if TEFF > 0.0 cross sections will be Doppler-broadened TEFF ≤ 0.0 cross sections are not Doppler-broadened.
3	49-50	NDFB	Unit number of ENDF/B library tape. If NDFB = 0 or 5 read resonance parameters on unit 5 (card reader).

Card No. 2 (2I5) {NDFB ≠ 0 or 5}

Item	Cols.	Name	Description
1	1-5	MATNO	ENDF/B material number.
2	6-10	IDTAP	ENDF/B tape identification number.

This concludes the input data if resonance parameters are read from an ENDF/B tape.

Card(s) No. 3 {NDFB = 0 or 5}

Follow the ENDF/B File 2 format for Adler-Adler parameters. This is described in detail in ref. 2.

III. SAMPLE PROBLEM

The input cards for the sample problem, U-233, are given in Table 1. These Adler-Adler resonance parameters are documented in refs. 7 and 8, except that the capture parameters have been revised above 21 eV (ref. 12). It should be noted that the revised capture parameters do not produce any negative cross sections. The fission cross section calculated by ADLER-III using the parameters of Table 1 and a temperature of 293°K is given in the Appendix (Fig. 1) for neutron energies between 0.79 and 60.0 eV. In a similar fashion, the radiative capture cross section for this energy region is shown in Fig. 2.

Table 1. Sample Problem Input Data

```
ADLER-III TEST OF U-233  5-73.      293.0
    92233.   231.0375             0             0             1             00233 2151
    92233.        1.0             0             1             1             00233 2151
     0.790       60.0             1             4             0             00233 2151
       2.5    1.00925             0             0             1             00233 2151
 231.0375         0.0             6             0            18             30233 2151
       0.0        0.0           0.0           0.0           0.0           0.00233 2151
       0.0        0.0           0.0           0.0           0.0           0.00233 2151
       0.0        0.0           0.0           0.0           0.0           0.00233 2151
       0.0        0.0             0             0             1             00233 2151
       2.5        0.0             0             0           840            700233 2151
 0.0         0.0         0.0         0.0        -2.8100E 00 3.6000E-01
-3.5773E-03-8.4062E-04-2.8100E 00 3.6000E-01 2.1802E-04 7.3099E-07
 0.0         0.0         0.0         0.0         1.5000E-01 5.0000E-02
-3.5026E-07-3.0458E-07 1.5000E-01 5.0000E-02 2.4366E-07-1.3706E-07
 0.0         0.0         0.0         0.0         1.4300E 00 2.9000E-01
 7.7454E-05-2.8737E-05 1.4300E 00 2.9000E-01 2.7412E-06-9.8531E-06
 0.0         0.0         0.0         0.0         1.7800E 00 1.2000E-01
 1.7401E-04-3.5026E-07 1.7800E 00 1.2000E-01 3.9732E-05 9.1373E-06
 0.0         0.0         0.0         0.0         2.2900E 00 5.0000E-02
 6.5088E-05 2.8143E-05 2.2900E 00 5.0000E-02 5.6804E-05 3.1524E-06
 0.0         0.0         0.0         0.0         3.3700E 00 3.5000E-01
 4.2824E-05-2.0727E-05 3.3700E 00 3.5000E-01 0.0        0.0
 0.0         0.0         0.0         0.0         3.6300E 00 8.0000E-02
 3.3534E-05 1.0356E-05 3.6300E 00 8.0000E-02 1.4711E-05 2.2234E-06
 0.0         0.0         0.0         0.0         4.6100E 00 4.0000E-01
 9.5226E-05-5.3423E-05 4.6100E 00 4.0000E-01 5.9393E-06-2.3300E-06
 0.0         0.0         0.0         0.0         5.8000E 00 1.5000E-01
 3.0671E-05-2.5980E-05 5.8000E 00 1.5000E-01 5.4976E-06-1.0965E-06
 0.0         0.0         0.0         0.0         6.8000E 00 9.0000E-02
 2.6897E-04 4.7712E-05 6.8000E 00 9.0000E-02 8.9104E-05 4.0813E-06
 0.0         0.0         0.0         0.0         7.4700E 00 1.0000E-01
 1.0325E-05-1.0203E-06 7.4700E 00 1.0000E-01 2.8478E-06 1.1422E-06
 0.0         0.0         0.0         0.0         8.6900E 00 2.8000E-01
 3.4524E-05 7.3403E-06 8.6900E 00 2.8000E-01 4.4925E-06-1.8427E-06
 0.0         0.0         0.0         0.0         9.1400E 00 1.5000E-01
 2.6620E-05-3.1189E-05 9.1400E 00 1.5000E-01 5.2387E-06-1.4315E-06
 0.0         0.0         0.0         0.0         1.0350E 01 1.6000E-01
 4.2015E-04 6.3961E-07 1.0350E 01 1.6000E-01 8.2891E-05 3.5636E-06
 0.0         0.0         0.0         0.0         1.1300E 01 2.0000E-01
 5.0880E-05 1.3203E-05 1.1300E 01 2.0000E-01 4.7819E-06-3.9443E-06
 0.0         0.0         0.0         0.0         1.2760E 01 1.7000E-01
 3.6380E-04 7.5276E-05 1.2760E 01 1.7000E-01 5.0408E-05 4.5839E-06
 0.0         0.0         0.0         0.0         1.3450E 01 2.0000E-01
 6.3900E-05 2.2128E-05 1.3450E 01 2.0000E-01 1.5122E-05-5.9240E-06
 0.0         0.0         0.0         0.0         1.3650E 01 1.4000E-01
 6.8515E-05-4.0692E-05 1.3650E 01 1.4000E-01 3.8377E-06 1.3554E-06
 0.0         0.0         0.0         0.0         1.5280E 01 1.2000E-01
 1.8240E-04-5.4017E-05 1.5280E 01 1.2000E-01 6.9840E-05-5.9545E-06
 0.0         0.0         0.0         0.0         1.6130E 01 2.2000E-01
```

Table 1 (cont'd)

1.5419E-04	-7.2946E-06	1.6130E 01	2.2000E-01	7.4622E-06	-4.4621E-06
0.0	0.0	0.0	0.0	1.6520E 01	1.3000E-01
1.4397E-04	4.0539E-05	1.6520E 01	1.3000E-01	3.0260E-05	6.7312E-06
0.0	0.0	0.0	0.0	1.7930E 01	1.2000E-01
6.5728E-05	6.4266E-06	1.7930E 01	1.2000E-01	1.4863E-05	2.2691E-06
0.0	0.0	0.0	0.0	1.8420E 01	2.1000E-01
8.9363E-05	2.7869E-06	1.8420E 01	2.1000E-01	7.7820E-06	-1.6295E-06
0.0	0.0	0.0	0.0	1.8860E 01	1.5000E-01
3.2931E-04	-7.0799E-05	1.8860E 01	1.5000E-01	5.4063E-05	6.8530E-06
0.0	0.0	0.0	0.0	2.0530E 01	2.0000E-01
1.6982E-04	9.9445E-06	2.0530E 01	2.0000E-01	2.8859E-05	1.2488E-06
0.0	0.0	0.0	0.0	2.1850E 01	1.3000E-01
1.9831E-04	1.3109E-04	2.1850E 01	1.3000E-01	5.1395E-05	0.0
0.0	0.0	0.0	0.0	2.2230E 01	2.4000E-01
8.1051E-04	-1.1816E-04	2.2230E 01	2.4000E-01	7.5210E-05	0.0
0.0	0.0	0.0	0.0	2.2940E 01	3.8000E-01
1.4708E-04	5.5433E-06	2.2940E 01	3.8000E-01	-3.1245E-06	0.0
0.0	0.0	0.0	0.0	2.3540E 01	3.2000E-01
9.9734E-05	-1.4448E-04	2.3540E 01	3.2000E-01	8.5739E-06	0.0
0.0	0.0	0.0	0.0	2.5130E 01	1.9000E-01
1.2300E-04	-5.7885E-05	2.5130E 01	1.9000E-01	2.1985E-05	0.0
0.0	0.0	0.0	0.0	2.6190E 01	1.8000E-01
-1.1117E-06	2.5067E-05	2.6190E 01	1.8000E-01	0.0	0.0
0.0	0.0	0.0	0.0	2.6570E 01	2.4000E-01
1.0776E-04	2.9422E-05	2.6570E 01	2.4000E-01	1.7126E-05	0.0
0.0	0.0	0.0	0.0	2.7180E 01	7.7000E-01
2.1808E-05	-6.4829E-05	2.7180E 01	7.7000E-01	0.0	0.0
0.0	0.0	0.0	0.0	2.8310E 01	1.1000E-01
2.0620E-05	2.2006E-05	2.8310E 01	1.1000E-01	1.0414E-05	0.0
0.0	0.0	0.0	0.0	2.9050E 01	2.7000E-01
2.6891E-04	7.3418E-05	2.9050E 01	2.7000E-01	1.9547E-05	0.0
0.0	0.0	0.0	0.0	2.9600E 01	8.0000E-02
1.9600E-05	1.6447E-05	2.9600E 01	8.0000E-02	4.5687E-06	0.0
0.0	0.0	0.0	0.0	3.0810E 01	1.4000E-01
3.8224E-05	9.7160E-05	3.0810E 01	1.4000E-01	1.1550E-05	0.0
0.0	0.0	0.0	0.0	3.1110E 01	5.7000E-01
2.7774E-04	-1.5477E-04	3.1110E 01	5.7000E-01	1.9904E-05	0.0
0.0	0.0	0.0	0.0	3.2000E 01	1.2000E-01
1.3031E-04	3.3153E-05	3.2000E 01	1.2000E-01	2.9346E-05	0.0
0.0	0.0	0.0	0.0	3.2950E 01	3.0000E-01
8.0637E-05	-2.6133E-05	3.2950E 01	3.0000E-01	4.5819E-06	0.0
0.0	0.0	0.0	0.0	3.3950E 01	6.1000E-01
2.5702E-04	1.1379E-04	3.3950E 01	6.1000E-01	0.0	0.0
0.0	0.0	0.0	0.0	3.4460E 01	4.0000E-01
2.8717E-04	-6.4449E-05	3.4460E 01	4.0000E-01	1.6611E-05	0.0
0.0	0.0	0.0	0.0	3.5190E 01	1.2000E-01
5.4078E-05	-5.8936E-06	3.5190E 01	1.2000E-01	-1.4110E-06	0.0
0.0	0.0	0.0	0.0	3.5400E 01	1.5000E-01
1.3249E-06	-5.4931E-05	3.5400E 01	1.5000E-01	2.7553E-06	0.0
0.0	0.0	0.0	0.0	3.6540E 01	1.1000E-01
9.8105E-05	2.8448E-05	3.6540E 01	1.1000E-01	3.2456E-05	0.0
0.0	0.0	0.0	0.0	3.7440E 01	2.7000E-01
1.3918E-04	1.4940E-05	3.7440E 01	2.7000E-01	6.1719E-06	0.0
0.0	0.0	0.0	0.0	3.9140E 01	6.0000E-01

Table 1 (cont'd)

1.6234E-04	-1.5275E-04	3.9140E 01	6.0000E-01	5.7352E-06	0.0
0.0	0.0	0.0	0.0	4.0460E 01	4.2000E-01
7.5855E-05	4.2412E-05	4.0460E 01	4.2000E-01	1.9545E-06	0.0
0.0	0.0	0.0	0.0	4.1030E 01	9.0000E-02
3.9077E-05	1.2625E-05	4.1030E 01	9.0000E-02	1.5898E-05	0.0
0.0	0.0	0.0	0.0	4.2630E 01	1.2000E-01
9.5150E-05	3.3382E-05	4.2630E 01	1.2000E-01	2.4474E-05	0.0
0.0	0.0	0.0	0.0	4.3460E 01	1.8000E-01
6.2667E-05	8.1931E-06	4.3460E 01	1.8000E-01	8.3776E-06	0.0
0.0	0.0	0.0	0.0	4.4570E 01	4.6000E-01
9.4495E-05	1.4940E-05	4.4570E 01	4.6000E-01	0.0	0.0
0.0	0.0	0.0	0.0	4.5190E 01	5.3000E-01
4.1270E-06	-4.5808E-05	4.5190E 01	5.3000E-01	0.0	0.0
0.0	0.0	0.0	0.0	4.6020E 01	8.0000E-02
4.0067E-05	-1.9188E-05	4.6020E 01	8.0000E-02	2.1155E-05	0.0
0.0	0.0	0.0	0.0	4.7220E 01	2.4000E-01
1.1617E-04	1.0523E-05	4.7220E 01	2.4000E-01	6.4137E-06	0.0
0.0	0.0	0.0	0.0	4.8660E 01	1.2000E-01
2.5521E-04	-2.2843E-07	4.8660E 01	1.2000E-01	1.1179E-04	0.0
0.0	0.0	0.0	0.0	4.9060E 01	1.4000E-01
2.7732E-05	-2.6635E-05	4.9060E 01	1.4000E-01	-2.3300E-06	0.0
0.0	0.0	0.0	0.0	5.0250E 01	4.6000E-01
9.7495E-05	-3.8407E-05	5.0250E 01	4.6000E-01	-1.1117E-06	0.0
0.0	0.0	0.0	0.0	5.1160E 01	2.8000E-01
2.6468E-05	1.4056E-05	5.1160E 01	2.8000E-01	-3.1545E-07	0.0
0.0	0.0	0.0	0.0	5.1870E 01	7.3000E-01
5.3286E-05	-1.0957E-04	5.1870E 01	7.3000E-01	0.0	0.0
0.0	0.0	0.0	0.0	5.2890E 01	2.5000E-01
6.5271E-05	-1.1839E-04	5.2890E 01	2.5000E-01	1.4011E-05	0.0
0.0	0.0	0.0	0.0	5.3960E 01	1.1000E-01
7.3099E-05	-3.4966E-05	5.3960E 01	1.1000E-01	1.4848E-05	0.0
0.0	0.0	0.0	0.0	5.4800E 01	1.0000E-01
8.9698E-05	2.5524E-05	5.4800E 01	1.0000E-01	3.1159E-05	0.0
0.0	0.0	0.0	0.0	5.6080E 01	2.7000E-01
9.7343E-05	2.0110E-04	5.6080E 01	2.7000E-01	8.9698E-06	0.0
0.0	0.0	0.0	0.0	5.6350E 01	3.0000E-01
3.4231E-04	-8.5921E-05	5.6350E 01	3.0000E-01	3.5482E-05	0.0
0.0	0.0	0.0	0.0	5.7480E 01	4.9000E-01
4.3972E-04	2.6498E-05	5.7480E 01	4.9000E-01	1.4936E-05	0.0
0.0	0.0	0.0	0.0	5.8510E 01	2.3000E-01
1.3898E-04	3.5529E-05	5.8510E 01	2.3000E-01	1.2875E-05	0.0
0.0	0.0	0.0	0.0	6.1300E 01	4.0000E-01
2.0224E-04	7.1408E-05	6.1300E 01	4.0000E-01	0.0	0.0
0.0	0.0	0.0	0.0	6.2500E 01	2.0000E-01
1.6426E-04	-2.2782E-05	6.2500E 01	2.0000E-01	0.0	0.0
0.0	0.0	0.0	0.0	6.4300E 01	4.0000E-01
3.4725E-04	-3.3580E-05	6.4300E 01	4.0000E-01	0.0	0.0

REFERENCES

1. M. R. Bhat, "ENDF/B Processing Codes for the Resonance Region," BNL-50296 (ENDF-148) (June 1971).

2. M. K. Drake, Editor, "Data Formats and Procedures for the ENDF Neutron Cross Section Library," BNL-50274 (ENDF-102) (October 1970).

3. N. M. Greene, J. L. Lucius, W. E. Ford, III, J. E. White, R. Q. Wright, and L. M. Petrie, "AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B," ORNL-TM-3706 (AMPX-1) (to be published).

4. D. B. Adler and F. T. Adler, "Analysis of Neutron Resonances in Fissile Elements: Program CODILLI, CURVEPLOT and SIGMA," Report COO-1546-3 (Sept. 1966), Dept. of Physics, University of Illinois, Urbanna, Illinois.

5. F. T. Adler and D. B. Adler, <u>Proc. of Conf. on Neutron Cross Section and Technology</u>, Book 2, p. 873, U.S.A.E.C. Report No. CONF-660303 (1966).

6. F. T. Adler and D. B. Adler, "Multilevel Expansion and its Application to Neutron Resonance Analysis," COO-2022-18, Department of Physics, University of Illinois, Urbanna, Illinois (July 1972).

7. G. deSaussure, "Analytical Description of the Neutron Cross Sections of U-233 Below 60 eV," ORNL-TM-2745 (November 1969).

8. G. deSaussure, et al., "Proceedings - Second International Conference on Nuclear Data for Reactors," IAEA-CN-26/94, Helsinki, Vol. II, pp. 757-775 (June 1970).

9. A. W. Solbrig, Jr., <u>Am. J. Phys. 29</u>, 257-261 (1961).

10. L. Dresner, Resonance Absorption in Nuclear Reactors, Pergamon Press (1960).

11. R. B. Canright, Jr. and Thor T. Semler, "Comparison of Numerical Techniques for the Evaluation of the Doppler Broadening Functions $\psi(X,\theta)$ and $\chi(X,\theta)$," NASA TM X-2559 (June 1972).

12. G. deSaussure, ORNL, personal communication to R. Q. Wright, April 23, 1973.

APPENDIX

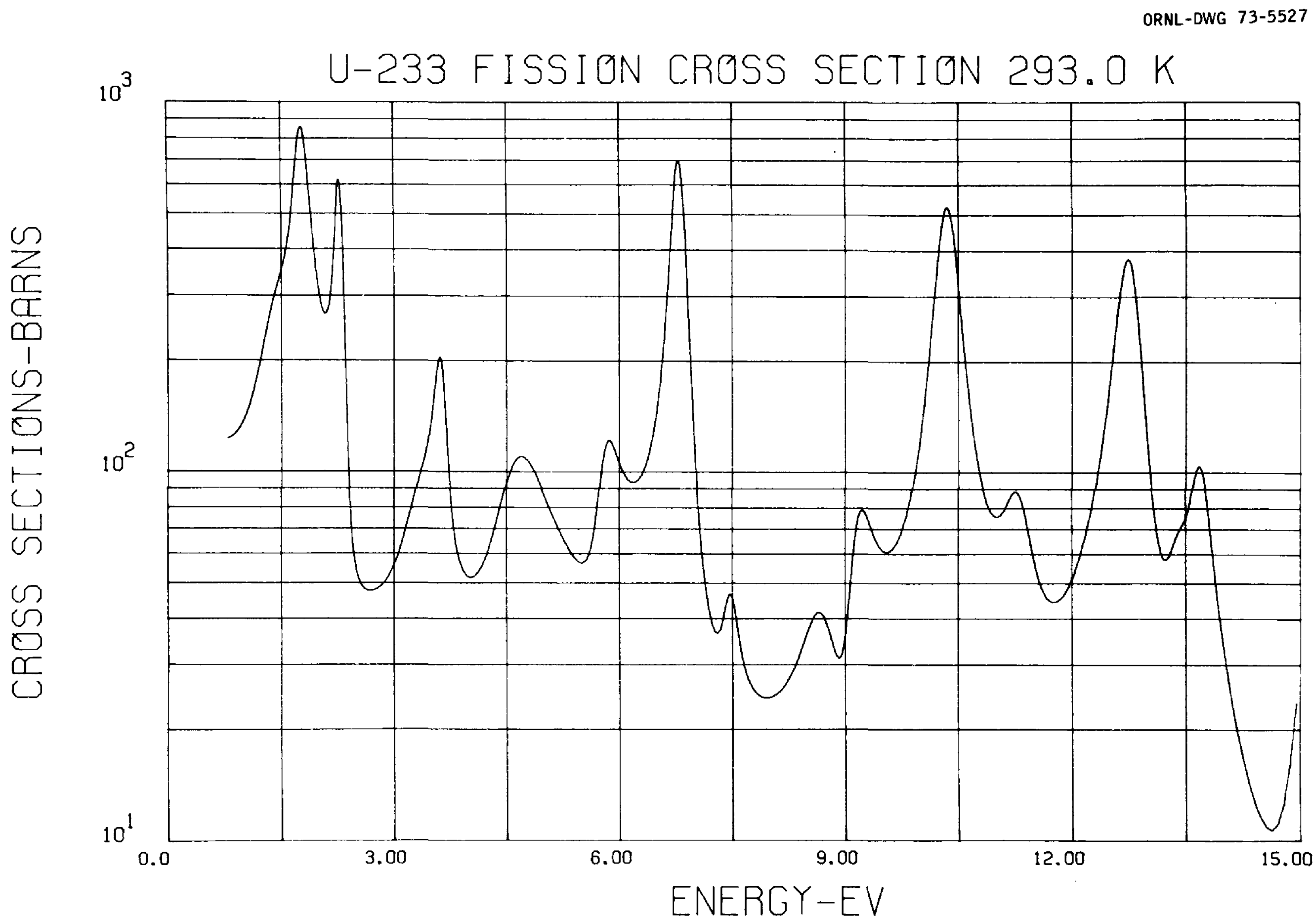

Fig. 1a. ^{233}U Fission Cross Section for E = 0.79 to 15.0 eV.

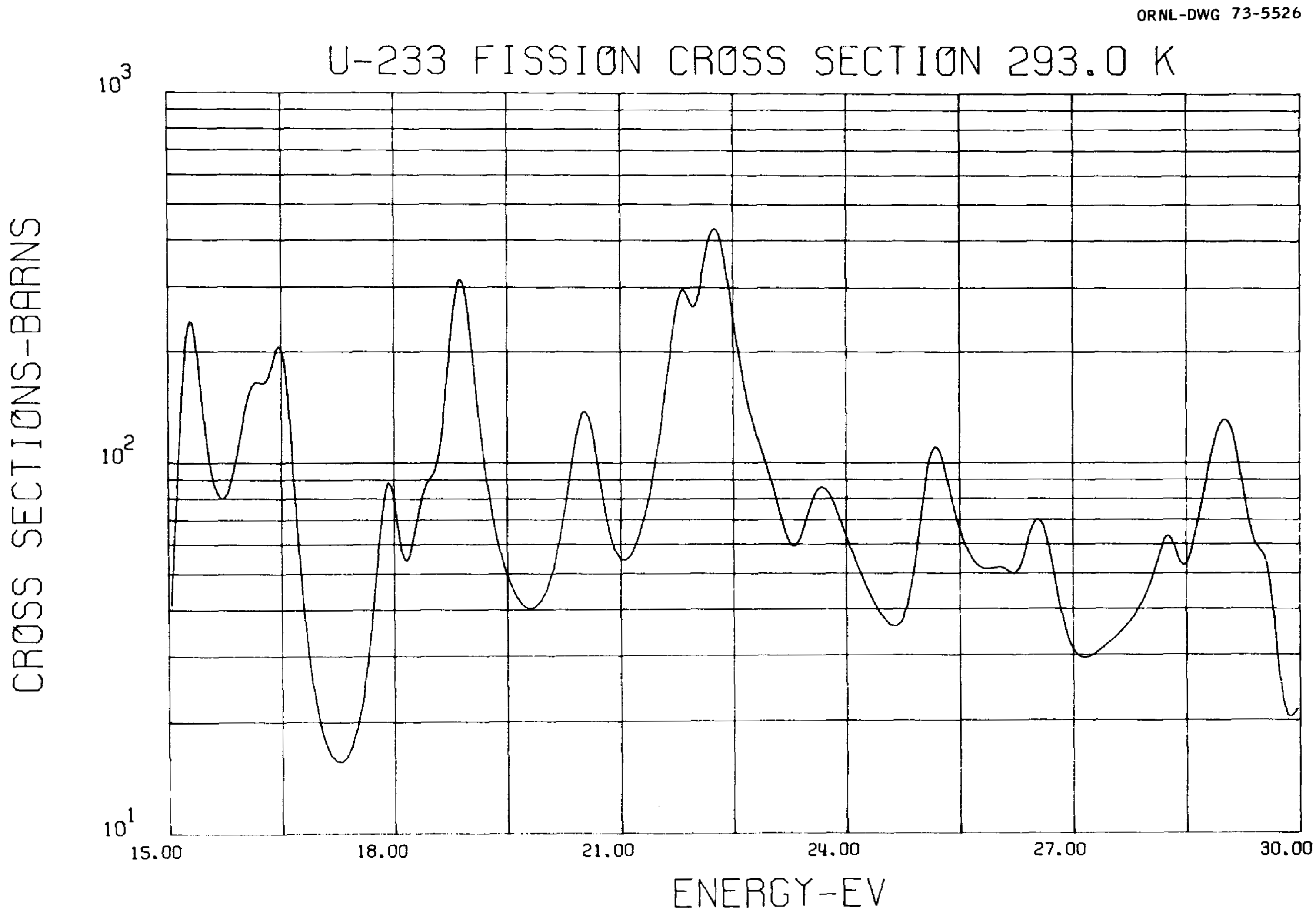

Fig. 1b. ^{233}U Fission Cross Section for E = 15.0 to 30.0 eV.

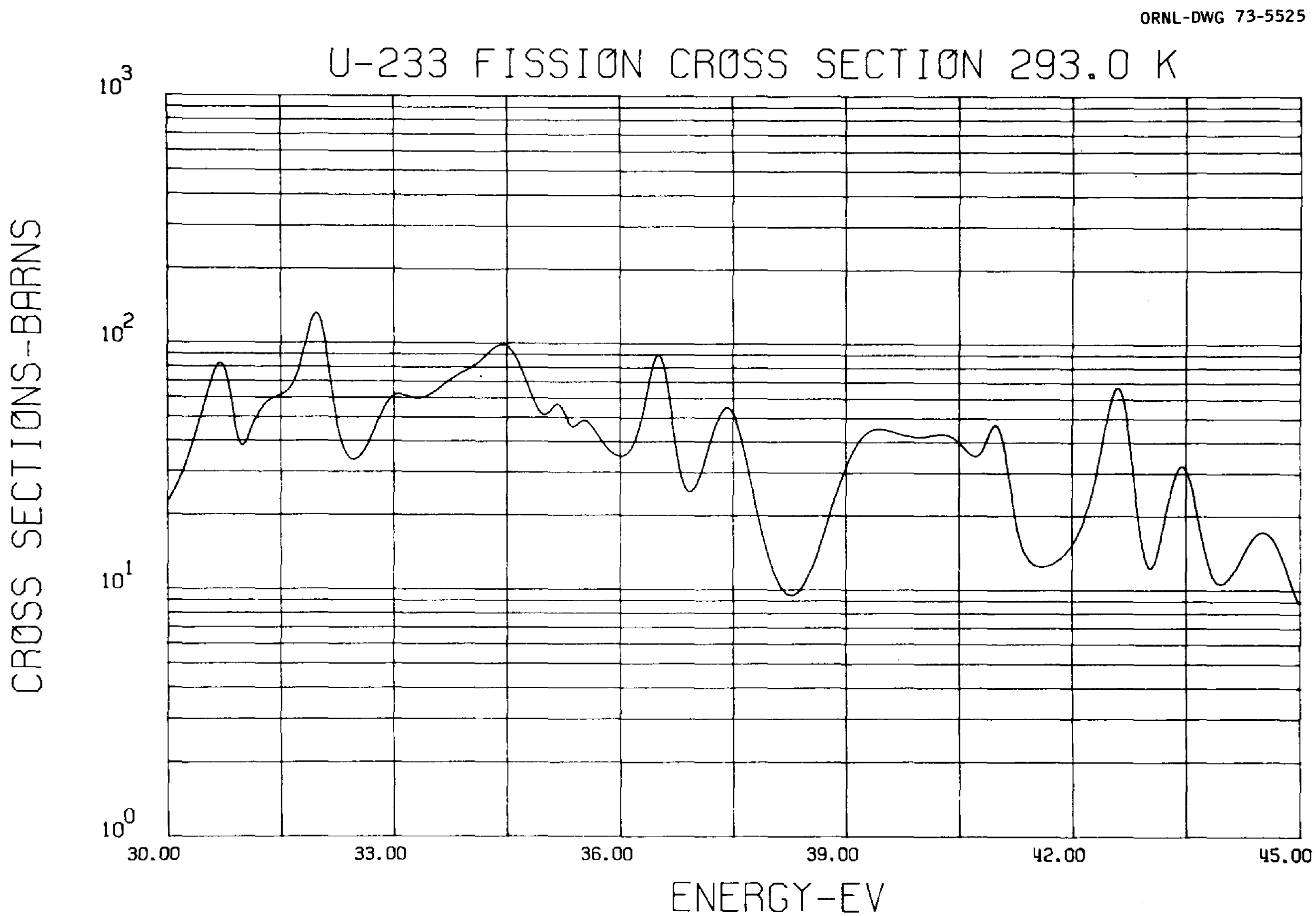

Fig. 1c. ^{233}U Fission Cross Section for E = 30.0 to 45.0 eV.

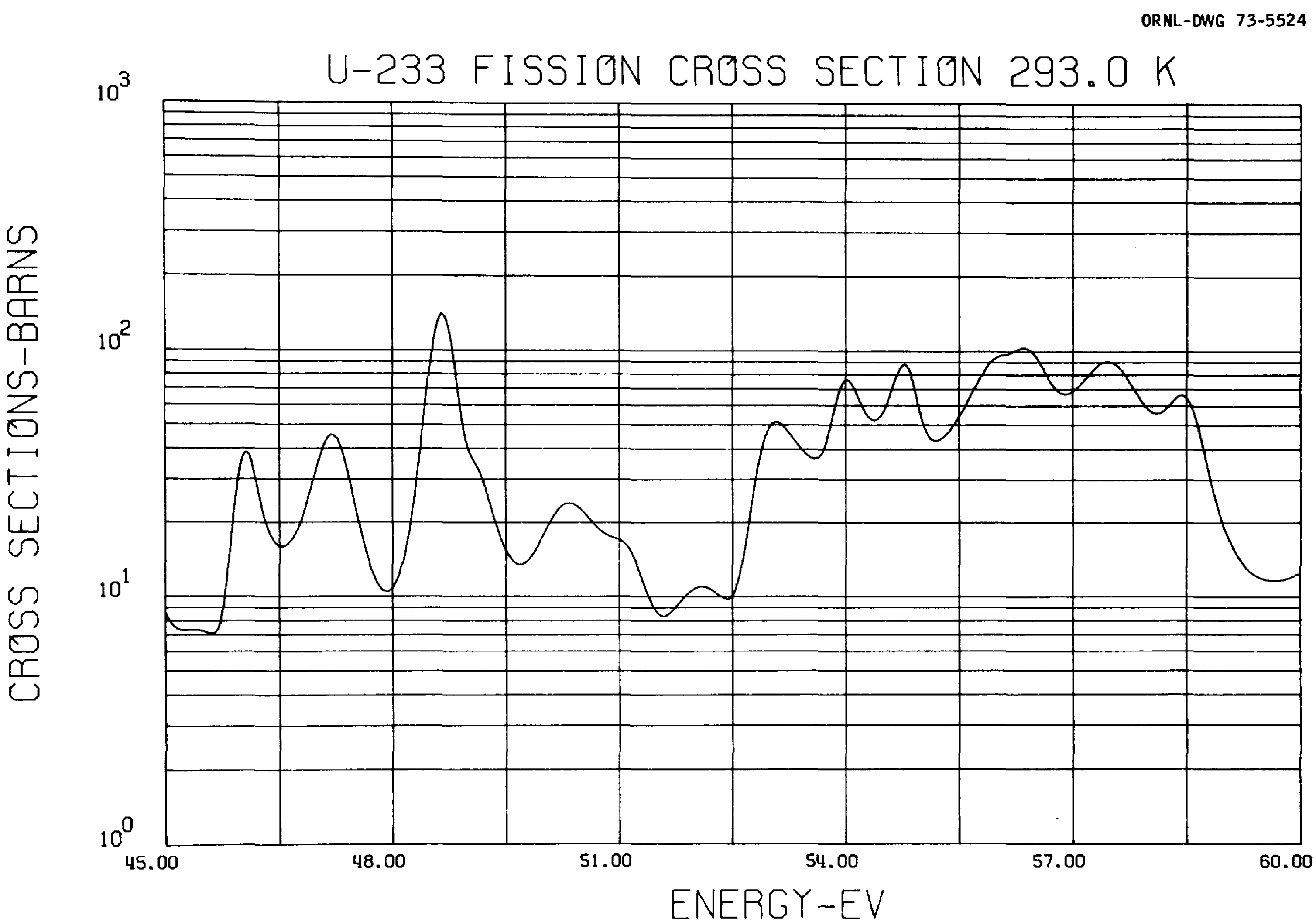

Fig. 1d. ^{233}U Fission Cross Section for E = 45.0 to 60.0 eV.

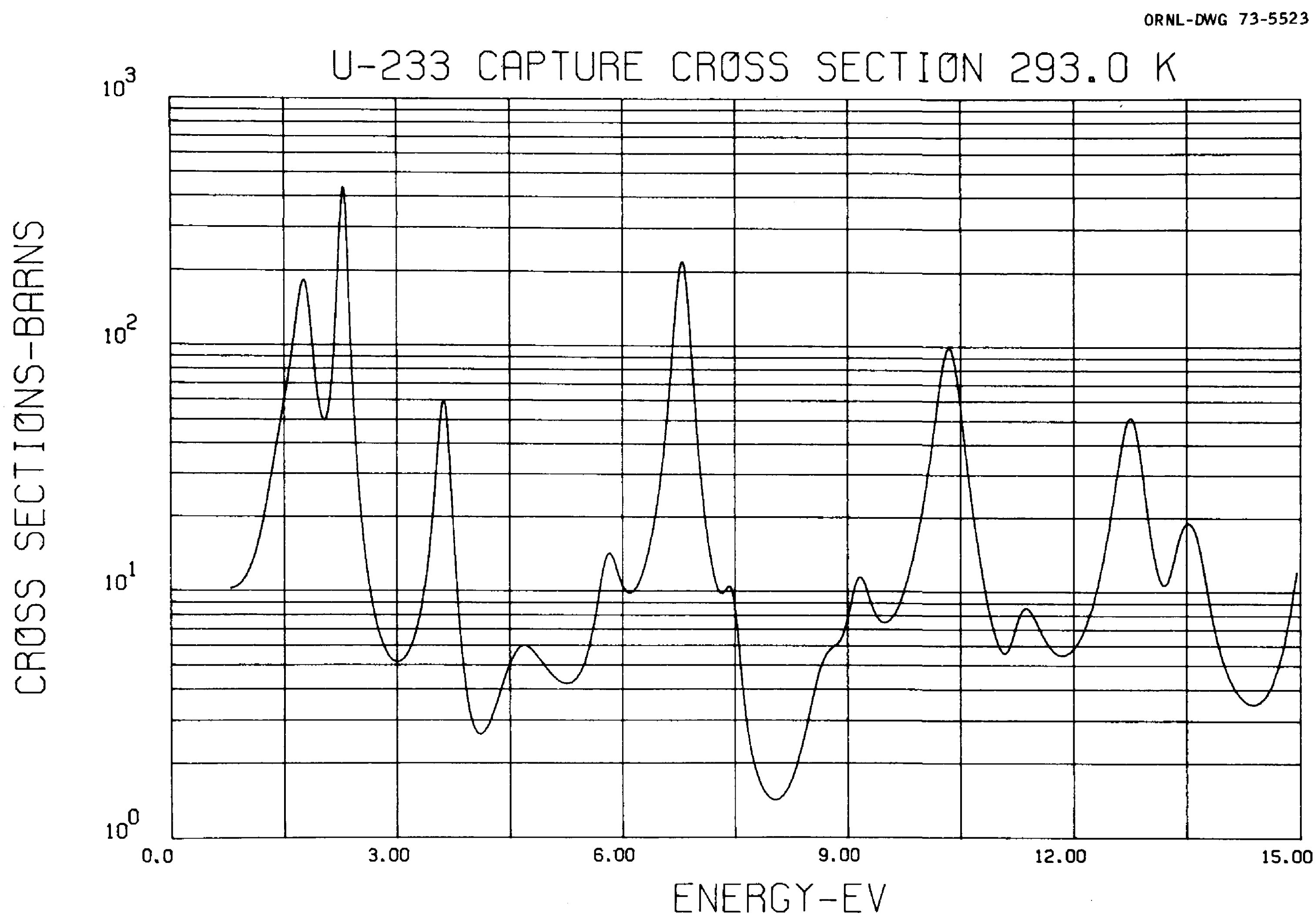

Fig. 2a. ^{233}U Capture Cross Section for E = 0.79 to 15.0 eV.

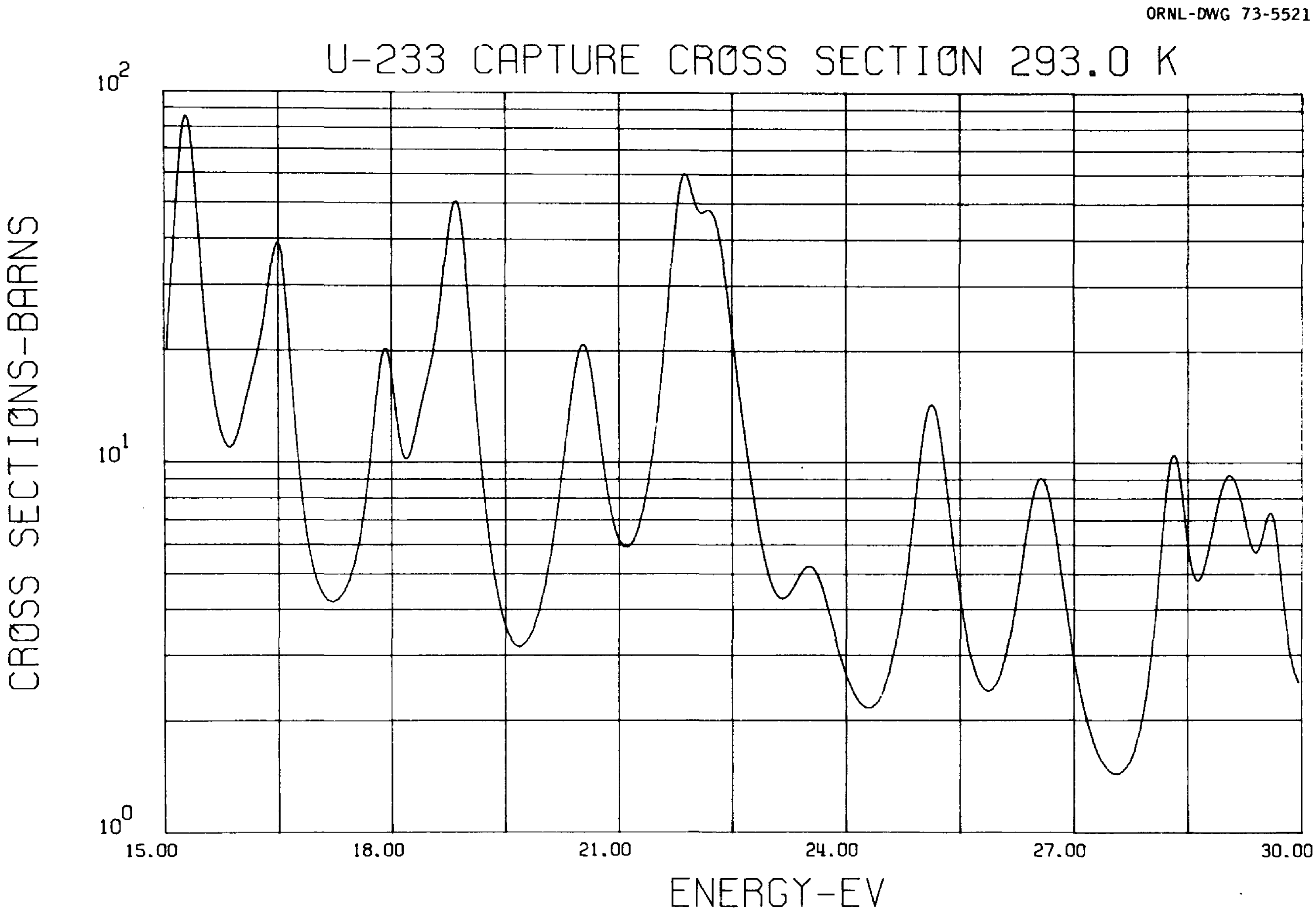

Fig. 2b. ^{233}U Capture Cross Section for E = 15.0 to 30.0 eV.

ORNL-DWG 73-5522

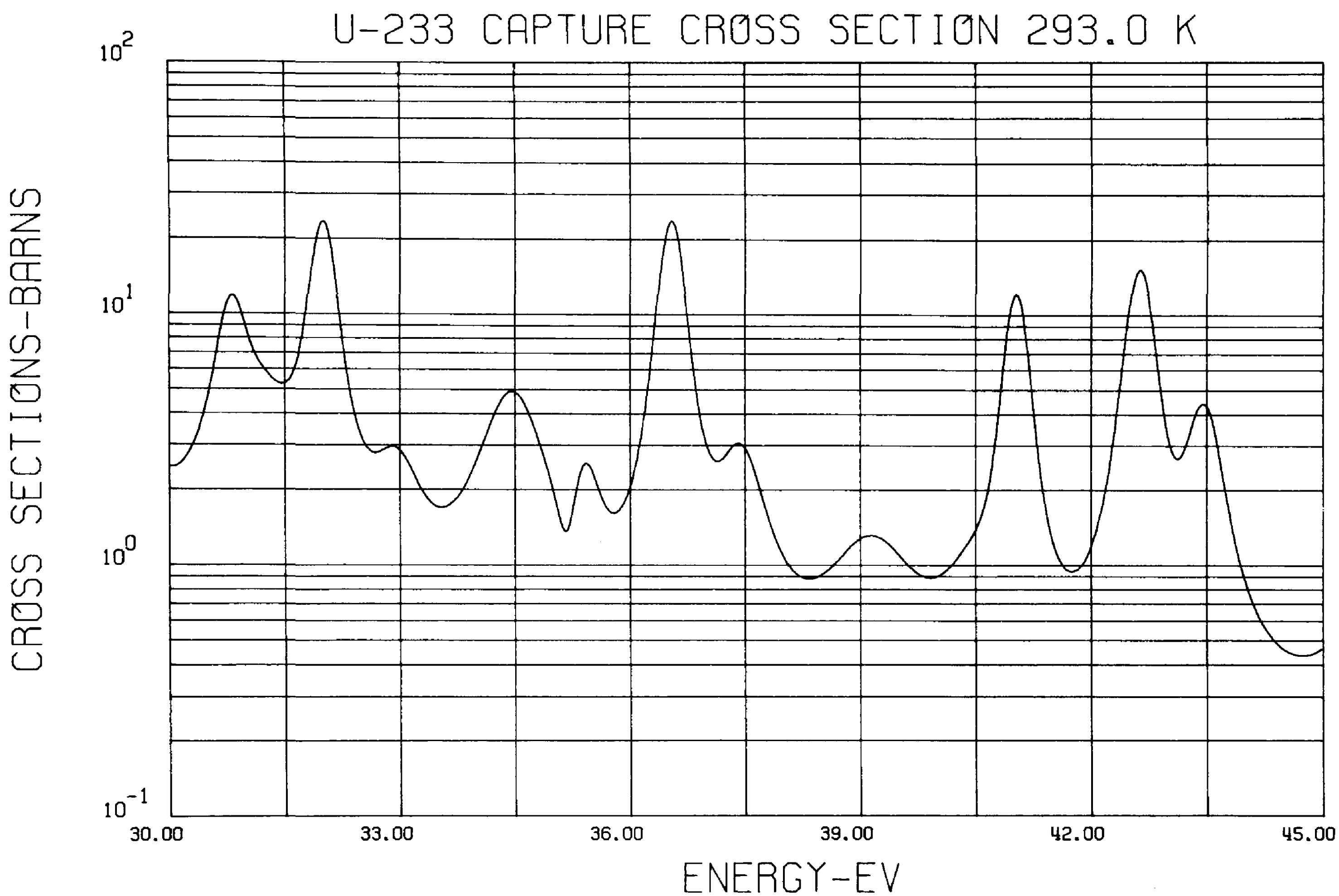

Fig. 2c. ^{233}U Capture Cross Section for E = 30.0 to 45.0 eV.

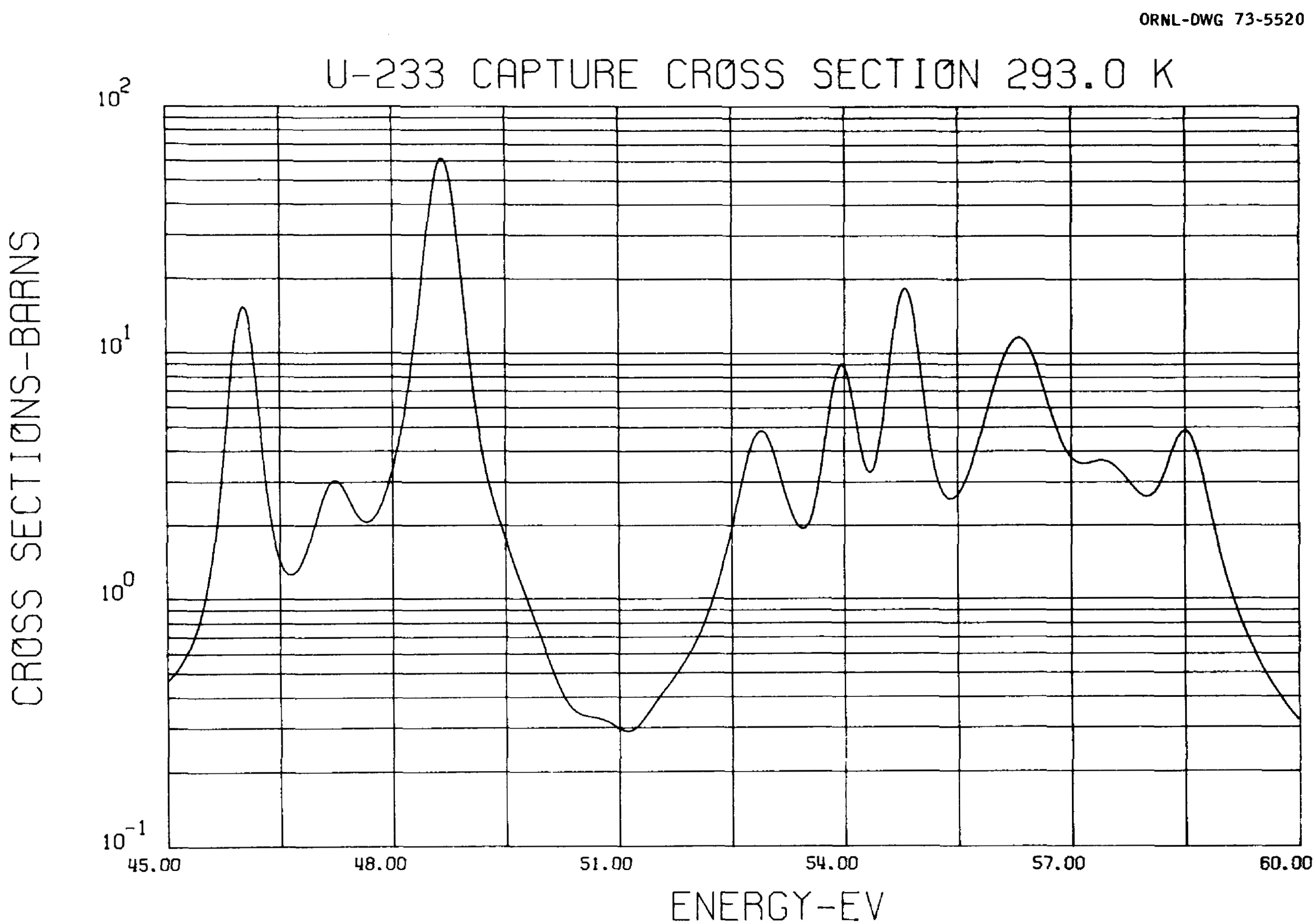

Fig. 2d. ^{233}U Capture Cross Section for E = 45.0 to 60.0 eV.

INTERNAL DISTRIBUTION

1-3. L. S. Abbott
4. R. G. Alsmiller, Jr.
5. D. E. Bartine
6. L. L. Bennett
7. A. A. Brooks
8. H. P. Carter
9-11. C. E. Clifford
12. F. L. Culler
13. G. deSaussure
14. W. E. Ford, III
15. N. M. Greene
16. W. E. Kinney
17. J. L. Lucius
18. F. C. Maienschein
19. F. R. Mynatt
20. R. W. Peelle
21. F. G. Perey
22. A. M. Perry
23. L. M. Petrie
24. A. H. Snell
25. D. Steiner
26. J. G. Sullivan
27. M. L. Tobias
28. D. B. Trauger
29. D. R. Vondy
30. J. E. White
31. G. E. Whitesides
32-36. R. Q. Wright
37. A. Zucker
38. H. Feshbach (Consultant)
39. P. F. Fox (Consultant)
40. C. R. Mehl (Consultant)
41. H. T. Motz (Consultant)
42-191. Radiation Shielding Information Center
192-193. Central Research Library
194. Document Reference Section
195-199. Laboratory Records
200. Laboratory Records - ORNL RC
201-210. Math. Division Library
211. ORNL Patent Office

EXTERNAL DISTRIBUTION

212-213. Technical Information Center (TIC)

214. Research and Technical Support Division, ORO

215. AEC Patent Office, OR

216-356. Given DNA Radiation Transport Distribution (Updated 4-3-73)

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
NUCLEAR DIVISION

POST OFFICE BOX X
OAK RIDGE, TENNESSEE 37830

SPECIAL 4TH CLASS RATE BOOKS

If a change of address is needed, indicate the change, include the zip code, and identify the document; tear off this cover and return it to the above address.