OLD BARNYARD: A CROSS SECTION CODE FOR SCHOOL USE

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December 1967

AIR FORCE INSTITUTE OF TECHNOLOGY

AIR UNIVERSITY
UNITED STATES AIR FORCE

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AFIT Technical Report 67-17
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This report replaces AFIT TR 66-6 (same title)

ABSTRACT

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OLD BARNYARD is a digital computer code which calculates few group fast and thermal neutron cross sections and constants. The first of two sequential chains calculates the neutron flux needed to determine the group cross sections. The code uses the moments method to find the fast flux and solves the Wilkins equation numerically for the thermal flux. The code is written in Fortran for a small computer, the IBM 1620, and is intended for use by students and professors in support of classroom assignments.

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List of Symbols

Symbol	Meaning
A	Mass of the scattering nuclide, amu
a_n or b_n	Coefficient of the n^{th} term of a series
E	Energy, ev
$F\{g(x)\}$	Fourier transform of $g(x)$
f	Thermal utilization
k	Boltzmann's Constant, ev per degree Kelvin
L^2	Thermal diffusion length squared, cm^2
M^2	Migration area, cm^2
m	Mass of scattering nuclide, amu
N	Number density of the mixture, atoms per barn cm (atoms x $10^{-24}/cm^3$)
$N(x)$	Number of neutrons at normalized velocity x
n	Neutron density, neutrons/cm^3
p	Resonance escape probability
P_n or P_ℓ	Legendre Polynomials
R	Reaction rate, events/cm^3-sec
$\vec{r}$	Neutron path vector between collisions, cm
$\bar{r}^2$	Neutron mean square path, cm^2
S	Neutron Source
T	Temperature, degrees Kelvin
U	Lethargy
V	Neutron velocity, cm/sec
X	Normalized neutron velocity

List of Symbols

Symbol	Meaning
Δ	The absorption parameter, amu
$\delta(x)$	Dirac Delta function
δ_m^n	Kronecker Delta
ξ	Average logarithmic energy decrement per collision
λ	Mean free path, cm
φ	Neutron flux, neutrons/cm^2-sec
Θ_{mn}^p	The p^{th} derivative of the m^{th} moment of the fourier transformed flux in group n
$\Psi(E)$	The deviation of the neutron flux from the Maxwellian distribution
$M(E)$	The Maxwellian flux distribution
μ	Cosine of the scalar scattering angle
τ	Neutron age
σ	Microscopic neutron cross section, barns
Σ	Macroscopic neutron cross section, cm^{-1}

Subscripts for the cross section are

a	Absorption
r	Removal
s	Scattering
Sj→n	Scattering transfer matrix element
t	Total
MGT	Multigroup transport
Tr	Common transport

PREFACE

Many professors of nuclear engineering, myself included teach reactor physics or neutron physics as a problem oriented course. Such problems generally require, as input, group cross sections and cross section related constants such as the diffusion coefficient or neutron age. However, these problems, unlike their counterparts in the real world, usually involve only one or a few energy groups. This few group distinction has always made the acquisition of group cross sections a particularily frustrating experience. One borrows on experimental value from here, a calculated value from there, and combines them with some fast cross sections collapsed by hand from a many group set (which inevitably doesn't contain all the elements of interest.) Although sophisticated methods are always available, data for one problem of a 10 problem assignment rarely warrants the effort to obtain them; nor does the accuracy required in a school problem warrant their sophistication.

The code described in this report is an attempt to provide reasonably good cross sections from a single, easily used, and unsophisticated (therefore inexpensive) source. The code is written in a version of the Fortran language, the Kingston Version of Fortran II (Kindstran.) Kingstran has not received wide publicity and, I find, is in limited use at other schools. However the language may be translated to Fortran IV with only very minor modifications. The AFIT version has been compiled for the IBM 1620, a digital computer, installed ub 146 (55%) of the 268 schools listed in the 1964 Oak Ridge Institute of Nuclear Studies report, EDUCATIONAL PROGRAMS AND FACILITIES IN

NUCLEAR SCIENCE AND ENGINEERING.

The AFIT 1620 has a 40K memory, card input-output, floating divide, indirect address and the usual additional instructions (TNS, TNF, MF.) Our object decks should execute without change on any similar machine. It is our present plan to honor requests for copies of our source or object decks directly. Please send such requests to me at the address below.

Robert H. Hansen carried out the moments reduction study as his M.S. thesis. Ernest Park Sims added the thermal portion of the code in conjunction with his M.S. thesis research, and put the code in its preliminary form. Daniel P. Cannon made the final revisions and thoroughly checked the code out. In addition he provided the bulk of this revised document. This effort was his M.S. thesis. In addition to project direction, I provided the theory chapter.

This report is a revision of an interim report which was published as AFIT TR 66-6 (May 1966.) This revision contains extensive code changes as well as increased documentation.

Finally I feel compelled to make some comment about the name OLD BARNYARD. It was originally coined by Capt. Sims as a joke; however, I find it is a name which the student remembers and more importantly associates with cross sections and age. I have yet to have a student ask "What was the name of that code which generates cross sections?"

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OLD BARNYARD

A CROSS SECTION CODE FOR SCHOOL USE

I. Introduction

Many nuclear engineering instructors regard the assignment and completion of realistic problems as vital to student learning. However, instructors who assign their students such problems are continually faced with the problem of providing group neutron cross sections and cross section related constants such as the diffusion coefficient and age. The alternative to providing such data is to require each student to find or calculate his own values. This alternative is unsatisfactory for both the student and the instructor. For the student, except for the first few times, finding or calculating group cross section is repetitious, time consuming, and of little educational value. For the instructor this practice is guaranteed to produce a considerable variation in final answers to evaluate.

An instructor usually can assume a Maxwell-Boltzmann distribution to determine thermal group constants or he can use experimental values, as available. Epithermal and fast group values are more difficult to obtain however, especially since only a few broad groups are desired for learning purposes. Computer codes are currently available, such as GAM (Ref 13), to provide fast cross sections, but they usually require large, fast and expensive computers not always available. Even when such codes are locally available, they usually employ sophisticated methods which are sketchily documented. Such a situation is acceptable when the codes are used by professionals but quite unacceptable when the codes are used by students.

As a result the instructor ends up borrowing some epithermal and fast group values from some few group published sets such as those in ANL 5800 (Ref 1.) However, these may not include all the elements of interest, and the group energy boundaries are not flexible.

Statement of the Problem

A need exists for a few-group neutron cross section code for school use in support of reactor criticality and neutron transport problems which meets the following criteria: calculates cross sections and related constants to a fair degree of accuracy (say 10%), uses methods with which a student can be expected to be familiar, is throughly documented, and can be used on small (as well as large) digital computers.

Purpose and Method

The purpose of this study is to produce such a code. The computer code described in this report calculates both fast and thermal cross sections and related constants such as age and diffusion length. It is written in Fortran and has been executed on an IBM 1620 computer where it requires about 15 minutes for completion. The cross sections are found by calculating the energy dependent flux in an infinite homogeneous mixture of the isotopes or compounds specified. The input cross section data is obtained from eleven group libraries of fast cross sections plus 2200 meter per second thermal values. The energy dependent flux is calculated by the neutron transport, moments method, for above thermal energies and by a Wilkins equation calculation for thermal energies.

The eleven group input was selected as a compromise between accuracy and amount of calculation after a systematic reduction in the number of fine groups (Ref 8.) The above thermal calculations were further simplified by computing resonance absorption with a resonance integral taken from the formulas given by Murray (Ref 16:63) and Isbin (Ref 11:459.) The Wilkins spectra is normalized to the moments spectra at 1.125 ev.

The code itself is described in Chapter II, The Code. This is followed by a chapter on operating instructions and information on preparing input. Thirteen sample problems are given in Chapter IV. The final chapter attempts to trace the theory of both epithermal and thermal calculations in textbook fashion.

II. The Code

The digital computer code Old Barnyard is written in the Kingston version of Fortran II, sometimes called "Kingstran" (Ref 6). The code consists of two chains which must be run sequentially. The first chain computes the energy dependent flux, both fast and thermal. This first chain also computes neutron age, resonance parameters, and the thermal constants. The output of this chain consists of the group flux, expressed as flux per unit lethargy, for 11 epithermal groups plus 12 thermal values at half lethargy intervals. The flux is normalized to a maximum value of 100. The neutron age is also computed both to the Indium resonance and to 1.125 ev (which is taken here as the arbitrary thermal region upper boundary). Should a resonance calculation be performed the scattering per absorber atom, and effective resonance integral are also calculated.

The thermal flux values are computed only if there is moderator in the system. If a thermal calculation is performed, in addition to the output above, the code computes the thermal diffusion length squared, the migration area, the most probable and average thermal neutron velocities, and the absorption parameter of the system. Chain one also includes, as an option, a plot of the flux as a function of lethargy over the entire range of interest.

The second chain collapses the cross sections to the desired group structure using the fluxes calculated in chain one. The operator may choose between microscopic and macroscopic cross sections for chain two output. A listing of the source decks is contained in

Appendices B and D. The source decks shown there have been executed on an IBM 1620 with 40 K memory, card input-output, floating divide, indirect addressing and the usual additional instructions (TNS, TNF, MF).

As a convenience to the reader, the more important quanities calculated by the code are listed in figure 1 along with the appropriate equation numbers from Chapter V of this report. Figure 1 also lists page number references in three popular texts for the same quantities. The texts are by Glasstone and Edlund (Ref 7), Isbin (Ref 11), and Murray (Ref 16).

Quanity to be Calculated	Equation Number in Chapter V	Textbook Reference		
		Murray	Isbin	G & E
1. Fast flux and age	66, 67, 68	60		181
2. Resonance Integral	71 thru 74		459	253
3. Resonance escape probability	75	63	454	253
4. Thermal utilization	78, 79	87	507	264
5. Absorption parameter	82			
6. Thermal flux	101, 103, 105			
7. Most probable thermal neutron velocity	109		47	38
8. Thermal diffusion length squared	118	66	236	116
9. Migration area	122		255	216
10. Collapsed broad group cross sections	125			

Figure 1

Summary of Important Quanities Calculated

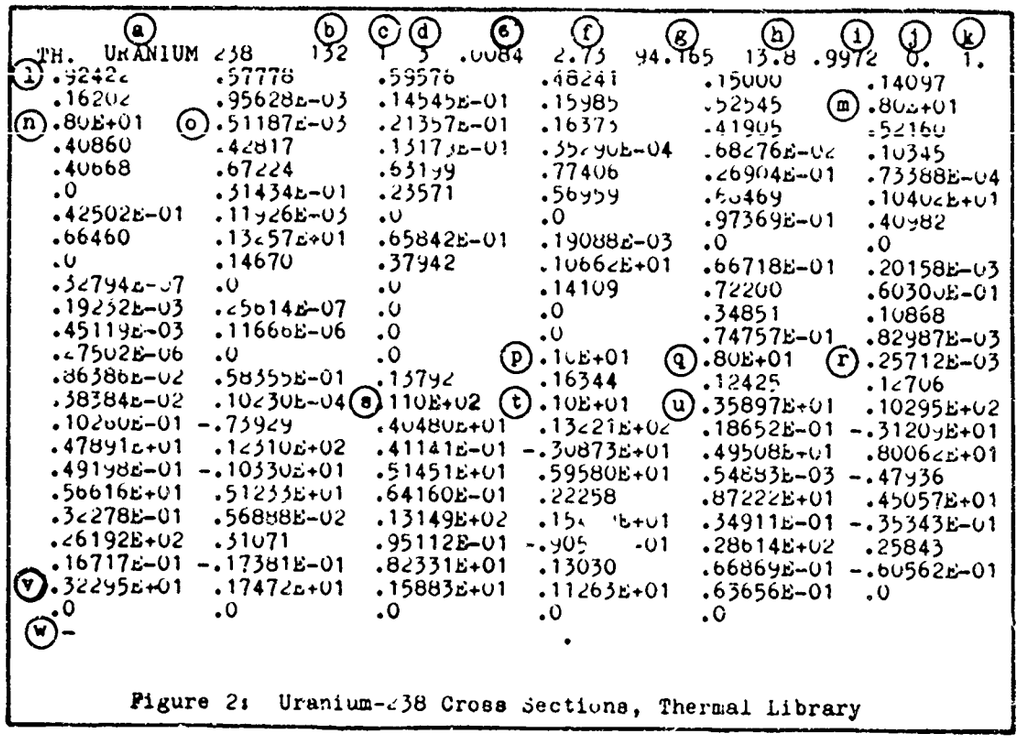
The input to chain I consists of a cross section library ("fine" groups) of 11 fast groups and the 2200 m/sec thermal cross section. Related thermal constants are included such as the average logarithmic energy decrement ξ, the absorption parameter at T = 293 deg Kelvin, and the quanity $(1-\bar{\mu})$ where $\bar{\mu} \equiv 2/3A$. There are two cross section libraries provided with the code. The thermal values in these libraries were obtained from ANL 5800 (Ref 1) and BNL 325 and the supplements to it (Ref 9). The eleven fast group cross sections were generated by the General Atomics 68 group moments code, GAM-1 (Ref 13). Therefore, the eleven group cross sections, used as input, are already flux averaged cross sections. As such, they are a function of the material composition used to generate them. Thus the two cross section libraries provided represent flux weighting over both a thermal and a fast reactor core composition. The Air Force Nuclear Engineering Test Facility (NETF) core, *an MTR type reactor*, was selected as the thermal standard; and the Godiva core, *a bare unmoderated system*, was selected as the fast reactor standard for input to GAM-1. The "thermal" library of 11 fast groups was generated by GAM working with the NETF core plus a trace of the nuclide to be considered. The calculation was repeated using each nuclide in the library as the trace nuclide. The "fast" library was generated in the same way except that the Godiva core was used in place of the NETF core. The "thermal" library is intended for use with thermal reactor problems; the fast library is used with fast reactor problems. The libraries include inelastic (n,n*) and (n-2n) cross sections and these effects are included in the moments methods calculations. The libraries also include fission cross sections for use in chain two. The user may elect to generate his own library

using some other material composition. The only restriction is that the 11 fine group boundaries are the same and the data order is that shown in Figure 2.

The libraries provided with the code currently include the following nuclides:

1. Hydrogen
2. Deuterium
3. Beryllium
4. Boron
5. Carbon
6. Oxygen
7. Nitrogen
8. Aluminum
9. Magnesium
10. Chromium
11. Manganese
12. Iron
13. Nickel
14. Copper
15. Zirconium
16. Molybdenum
17. Thorium-232
18. Uranium-235
19. Uranium-238
20. Plutonium-239
21. Fission Products (of Uranium-235)
22. Xenon-135
23. Samarium-149

The fission products entries includes all product nuclides of thermal Uranium-235 fission except Xenon-135 and Samarium-149. These two nuclides are included directly in the libraries.



	a		b	c	d	e	f	g	h	i	j	k
TH. URANIUM 238			132	1	3	.0084	2.73	94.165	13.8	.9972	0.	1.
l	.92426		.57778		.59576		.48241		.15000		.14097	
	.16202		.95628E-03		.14545E-01		.15985		.52545	m	.80E+01	
n	.80E+01	o	.51187E-03		.21357E-01		.16375		.41905		.52160	
	.40860		.42817		.13173E-01		.35290E-04		.68276E-02		.10345	
	.40668		.67224		.63199		.77406		.26904E-01		.73388E-04	
	.0		.31434E-01		.23571		.56959		.50469		.10402E+01	
	.42502E-01		.11926E-03		.0		.0		.97369E-01		.40982	
	.66460		.13257E+01		.65842E-01		.19088E-03		.0		.0	
	.0		.14670		.37942		.10662E+01		.66718E-01		.20158E-03	
	.32794E-07		.0		.0		.14109		.72200		.60300E-01	
	.19252E-03		.25614E-07		.0		.0		.34851		.10868	
	.45119E-03		.11660E-06		.0		.0		.74757E-01		.82987E-03	
	.27502E-06		.0		.0	p	.10E+01	q	.80E+01	r	.25712E-03	
	.86386E-02		.58355E-01		.13792		.16344		.12425		.12706	
	.38384E-02		.10230E-04	s	.110E+02	t	.10E+01	u	.35897E+01		.10295E+02	
	.10200E-01		-.75929		.40480E+01		.13221E+02		.18652E-01		-.31209E+01	
	.47891E+01		.12510E+02		.41141E-01		-.30873E+01		.49508E+01		.80062E+01	
	.49198E-01		-.10330E+01		.51451E+01		.59580E+01		.54883E-03		-.47936	
	.56616E+01		.51233E+01		.64160E-01		.22258		.87222E+01		.45057E+01	
	.32278E-01		.56888E-02		.13149E+02		.15[illegible]+01		.34911E-01		-.35343E-01	
	.26192E+02		.31071		.95112E-01		-.905[illegible]-01		.28614E+02		.25843	
	.16717E-01		-.17381E-01		.82331E+01		.13030		.66869E-01		-.60562E-01	
v	.32295E+01		.17472E+01		.15883E+01		.11263E+01		.63656E-01		.0	
	.0		.0		.0		.0		.0			
w	-											

Figure 2: Uranium-238 Cross Sections, Thermal Library

Figure 2 shows the U-238 cross sections from the "thermal" library. The entries are explained below as coded in the Figure.

a) Library and nuclide identification (Thermal library, U-238). Columns 1 thru 20

b) Number of fast cross sections below which begins on the second line. Does not include the $\nu\sigma_f$ entries for fissionable nuclides.

c) Epithermal-fast absorption code. o = No fast absorption cross sections provided (assumed negligible); 1 = Fast absorption cross sections provided

d) Type of inelastic scatter: 1 = inelastic; 2 = (n,2n); 3 = both; 4 = neither

e) Thermal average logarithmic energy decrement, ξ

f) Thermal microscopic absorption cross section, σ_a

g) Thermal absorption parameter, Δ

h) Thermal microscopic scattering cross section, σ_s in barns

i) Thermal $(1-\bar{\mu})$ where $\bar{\mu} = 2/3A$

j) Thermal value of the product of the number of neutrons per fission and the microscopic fission cross section, $\nu\sigma_f$

k) Epithermal-fast fission code: 0 = no; 1 = yes

l) Microscopic, fast, absorption cross sections: σ_a^1, σ_a^2,$\cdots\sigma_a^{11}$, barns

m) Number of groups from which neutrons inelastically scatter

n) Number of groups to which neutrons inelastically scatter

o) Microscopic, fast, inelastic cross sections in the following order:

$$\sigma^{in}_{1-1} \; ; \; \sigma^{in}_{1-2} \; ; \; \ldots \; \sigma^{in}_{1-9} \; ;$$

$$\sigma^{in}_{2-2} \; ; \; \sigma^{in}_{2-3} \; ; \; \ldots \; \sigma^{in}_{2-10}$$

$$\vdots$$

$$\sigma^{in}_{7-7} \; ; \; \sigma^{in}_{7-8} \; ; \; \ldots \; \sigma^{in}_{7-th}$$

$$\sigma^{in}_{8-8} \; ; \; \sigma^{in}_{8-9} \; ; \; \ldots \; \sigma^{in}_{8-th}$$

p) Number of groups in which (n,2n) occurs

q) Number of groups to which (n,2n) scatter occurs

r) Microscopic, fast, (n-2n) cross sections in the same order as inelastic

s) Number of groups from which neutrons elastically scatter

t) Number of groups to which neutrons elastically scatter

u) Microscopic Po and P1 elastic scatter cross sections in the following order (See the note below):

$$\sigma^{Po}_{1-1} \; ; \; \sigma^{P1}_{1-1} \; ; \; \sigma^{Po}_{1-2} \; ; \; \sigma^{P1}_{1-2}$$

$$\sigma^{Po}_{2-2} \; ; \; \sigma^{P1}_{2-2} \; ; \; \sigma^{Po}_{2-3} \; ; \; \sigma^{P1}_{2-3}$$

.

.

$$\sigma^{Po}_{11-11} \; ; \; \sigma^{P1}_{11-11} \; ; \; \sigma^{Po}_{11-th} \; ; \; \sigma^{P1}_{11-th}$$

NOTE: The microscopic Po elastic scatter cross section is the all angle elastic scatter cross section σ_s. The microscopic P1 elastic scatter cross section is the product of $\bar{\mu}$ times σ_s where $\bar{\mu}$ is the average cosine of the scattering angle. GAM-I outputs the value $3\sigma P1$ and this is the value used in the cross section library. Therefore, in the code all $\sigma P1$ are divided by 3.

v) Microscopic, fast, $\nu\sigma_f$ in the order $\nu\sigma_f^1$, $\nu\sigma_f^2$, ... $\nu\sigma_f^{11}$

w) Grid background card for the spectrum printout.

NOTE: The grid card is considered part of the library for <u>each</u> nuclide.

Since flux weighting cross sections involves division by the group flux, the problem of division by near zero quantities in the lower energy groups of fast reactor systems might occur. To prevent any problem the code scans the fast fluxes for a minimum value of 10^{-12}. A message is typed giving the group number of the first fine group in

which the neutron flux is less than the minimum value. If the total flux in a broad group (the sum of all fine group fluxes in a given broad group) is less than the minimum, there is no output for that broad group.

Should the user desire cross sections of "thermal" type materials, such as carbon or hydrogen, in the presence of a fast flux spectrum, he can obtain these simply by substituting the cross section library of the "thermal" nuclide when executing chain two.

The special set of Kingstran subroutines* provided with the code must be used to permit the chain operation.

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*At AFIT, if new object decks are produced (from the source decks) using the Kingstran compiler, seven (7) Kingstran trailer cards must be added to the back of the chain one and chain two object decks produced. This is necessary to use the special set of subroutines.

III. Operating Instructions: IBM 1620

Input data for chain one and chain two are prepared according to instructions contained in the next two sections respectively. To execute the code follow the operating instructions below:

1. Set all console switches to PROGRAM except the Parity switch is set to STOP.
2. Place the special Old Barnyard short subroutines in the read hopper. Press RESET and LOAD.
3. When the last card of the subroutines is reached, the console READER NO FEED light will illuminate steadily. Press READER START and read in last card.
4. Remove the subroutines from the out hopper. Place the Old Barnyard chain one object deck in the reader hopper. Place the data cards for chain one on top of the chain one object deck. Press LOAD and PUNCH START.
5. Chain one program will run automatically. When the last data card enters the reader, the console READER NO FEED light will illuminate steadily. Press READER START to read in last card.
6. The console typewriter will indicate the beginning of each section of the code. After all calculations to be executed are complete, the typewriter will type "SET SWITCH 1 ON FOR SPECTRUM PRINTOUT. PRESS START." If a plot of the neutron spectrum is desired, set switch 1 ON and press START. If no spectrum plot is desired, set switch 1 OFF and press START.

7. At the end of chain one the typewriter will type "END OF CHAIN 1. TO COLLAPSE CROSS SECTIONS LOAD CHAIN 2." If no cross sections are desired, the program is finished. Remove chain one and data cards and, if no cross sections are desired the chain one output cards, from the out hoppers.

8. If cross sections are desired, place Old Barnyard Chain 2 object deck in the read hopper. Place data cards for chain two on top of it. Press RESET and LOAD. Chain two will run automatically. After the first data card is read in a typewritten message summarizes the group structure requested. When the last card enters the reader the console READER NO FEED light will illuminate steadily. Press READER START to read in the last card. Remove chain two and data cards from the out hopper.

9. When chain two is complete the typewriter will type "END OF PROGRAM". Remove output cards from the output hopper. It is not necessary to press NON-PROCESS RUN OUT.

10. List output answer cards on the 407 lister. Put switch number 4 on the right side of the 407 UP. This causes automatic advance of a new sheet of paper for each section of output answers. If the paper advances at unwanted times (as it sometimes does) put switch 4 DOWN. The best plan is to put switch 4 UP for chain one output, DOWN for chain two outputs.

11. To run a second problem go back to step (1) and repeat. NOTE: if anything should go wrong during a run (such as an erroneous data card, or an I/O error, or a SKIP CHECK), the particular chain in use may be restarted by the following procedure:

1) Press STOP.

2) Remove remaining data cards, if any, from the read hopper.

3) Clear the reader of any internal data cards by pressing NON-PROCESS RUN OUT.

4) Replace corrected data cards in the read hopper.

5) Press RESET and INSERT.

6) Type on the console typewriter the numbers 4900936.

7) Press RESET and START or punch the R-S key on the typewriter.

8) Press READER START to read in data cards.

9) Program will run.

Preparation of Input Data, Chain One

The following data cards are required to operate chain one. Input data may be punched in any format. If the E format is used for data, no blank spaces are permitted between the last digit of the mantissa and the E of the exponent, but blanks are permitted between the E and the exponent. If the exponent is positive the plus sign is optional. Individual numbers on a card must be separated by one or more blank spaces; commas are not allowed. Examples of correct and incorrect methods are shown below.

Correct	Incorrect
1 1.5	1,1.5 or 1, 1.5
1.67E+01	1.67 E+01
1.67E 01	

Card No. 1:

This card contains up to 65 columns of alphameric data of the User's choosing. Identification, problem title, date, etc., are

possible information to be placed on this card. The data on this card will be typed out on the console typewriter and will be punched at the head of the output data.

EXAMPLE: AFIT Subcritical Core. 5 April 67 Mackrill

Card No. 2:

This card has six (6) numbers as follows;

a) A number between two (2) and twenty-one (21) indicating the source of neutrons the user wants to use. The meaning of the numbers is shown below:

Number	Source
2	U^{235} fission (Cranberg Spectrum)
3	U^{233} fission
4	Pu^{239}
5	Pu^{241}
6	cf^{252}
7	Pu-Be (Whitmore-Baker)
8	Pu-Be (Cochrane-Henry)
9	Ra-Be (Hill)
10	The user's own source (see Card No. 4)
11-21	A unit source in group 1-11 respectively

EXAMPLE: 15 = A unit source in group 5.

b) A number giving the number of nuclides to be used in the problem

EXAMPLE: 2 for Water (H and O)

2 for BeO (Be and O)

2 for C_2H_6 (C and H)

3 for an Al_2O_3-H_2O mixture (Al, O, and H)

1 for pure carbon.

c) A number giving the total nuclei density for the problem in atoms times 10^{-24} per cm^3. (atoms per barn-cm)

EXAMPLE: For Carbon 0.08025

For Water (H_2O) 0.1005

d) A number giving the temperature of the system in degrees Kelvin.

e) A number indicating whether or not a resonance escape calculation should be performed; as below:

0 = NO

1 = YES

f) A number indicating whether or not the mixture is considered homogeneous or heterogeneous; as below:

0 = homogeneous

1 = heterogeneous

Card No. 3:

This "card" is a small deck of cards containing the neutron source data for the program. It is included as part of the input library and normally requires no preparation on the part of the user. The source deck must always be included even when the user desires to read in his own source spectrum. A listing of the sources is shown in Appendix E.

Card No. 4:

This card is required only if the first number on card no. 2 is 10; i.e., only if the user wants to read in and use his own source. This card contains eleven numbers representing the fraction of the source in each of the eleven epithermal-fast groups.

Card No. 5:

This card is included only if the fifth number on card no. 2 is 1; i.e., if a resonance escape calculation is to be performed. This card contains three (3) numbers as follows:

a) A number indicating whether the resonance nuclide is U^{238}, Th^{232} or W (Tungsten) as shown below:

1 = U or W

2 = Th

b) A number giving the nuclei density of the resonance nuclide(s) in atoms per barn-cm.

c) A number indicating whether the effective resonance integral is being supplied by the user, or whether it is to be calculated by the program:

0 = to be calculated

non-zero = value of RI_{eff} to be used

Card No. 6:

This card is included only if the sixth number on card no. 2 is 1; i.e., only if the mixture is considered heterogeneous. This card contains two (2) numbers giving the "F" and "E" factors, in that order, for calculating the effect of heterogeneous cell structure on thermal

utilization. (See the section on thermal utilization for the equations used to calculate the "F" and "E" factors).

Cards number 7 and 8 are required for each nuclide in the sequence 7, 8, 7, 8, ...

Card No. 7:

This card contains six numbers as follows:

a) Twenty (20) columns of alphameric identification to identify the n^{th} nuclide of the problem. This is optional, for the user's use only, and may be left blank (the next number must always start after column 20 however).

b) A number giving the nuclei density of the n^{th} nuclide in the problem in atoms per barn-cm.

c) A number indicating whether the nuclei is a moderator, fuel, or other as below:

1 = moderator

2 = fuel

3 = other

d) A number indicating whether microscopic or macroscopic cross sections are to be output for this n^{th} nuclide. A number is always required here even if cross section averaging is not to be performed (chain two of the program). The number should be:

1 = microscopic

2 = macroscopic

e) A number indicating if this is a resonant nuclide (U^{238}, Th^{232} or W). The number should be:

0 = NO

1 = YES

Card No. 8:

This "card" is a deck of cards containing the eleven (11) group plus thermal cross sections of the n^{th} nuclide. This deck is selected from one of the two program libraries provided, and requires no preparation on the part of the user. The thermal library is for use with thermal reactor problems, and the fast library is for use with fast reactor problems. The Uranium-238 cross section deck from the thermal library is shown in Figure 2. The grid background card (the last card of each cross section deck) must be included for each nuclide.

Figure 3 shows the arrangement of the entire input deck for chain one. The following are some notes on chain one data:

a) The total nuclei density on the first card should be the sum of the individual nuclide densities on the separate card no. 7's.

b) The nuclei density of the resonance nuclide on card no. 3 should agree with that on card no 7 for the resonance absorber.

Preparation of Input Data, Chain Two

Chain two of the program requires the preparation of only one additional card. This card has up to twelve (12) numbers. The first is the number of epithermal-fast broad groups for which cross sections are to be calculated. The remaining numbers are the numbers of the lowest fine group in each of the selected broad groups. The last number on this card is eleven (11) indicating that the last epithermal-fast broad group extends down to .414 ev. Thus if the first number is five there will be five more numbers on the card. If the first number is eleven, there will be eleven more numbers on the card.

EXAMPLES:

```
 1  11
 2   6  11
11   1   2   3   4   5   6   7   8   9  10  11
```

This additional card is the first input data card for chain two. It is followed by the collection of cards 7 and 8 from chain one as shown in Figure 4.

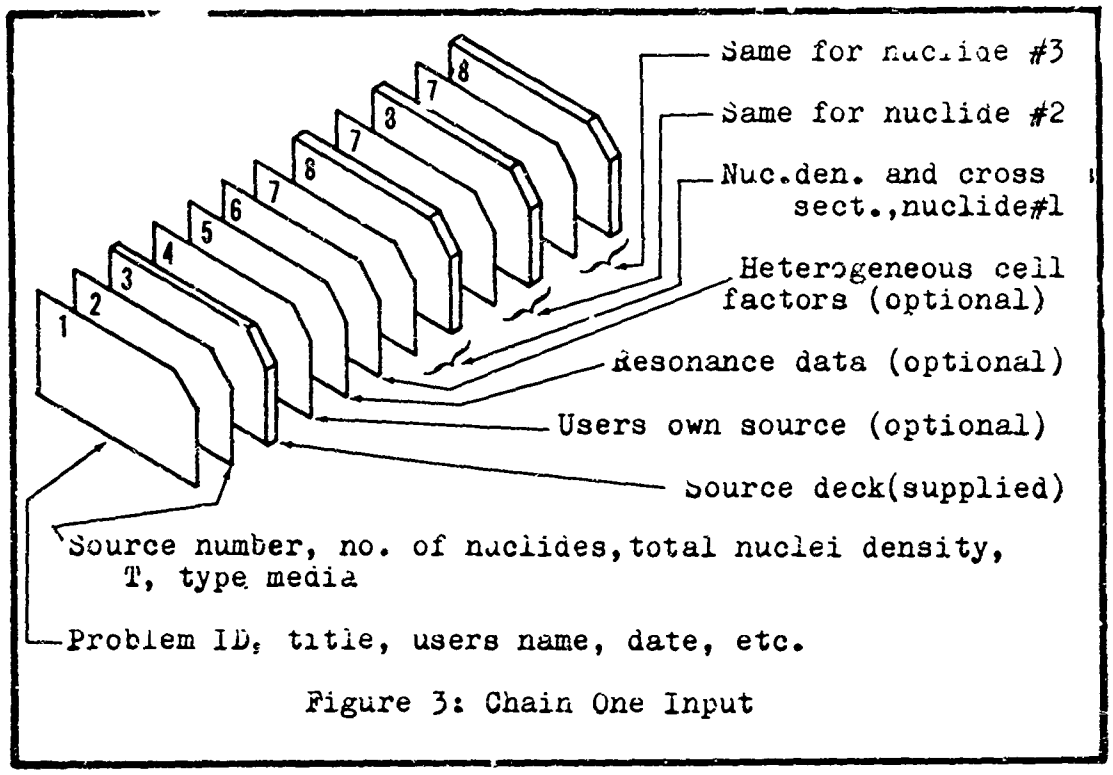


Figure 3: Chain One Input

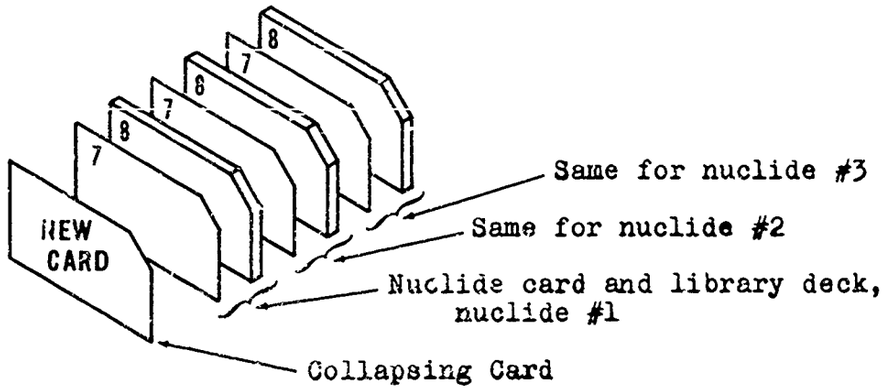


Figure 4: Chain Two Input

IV. Sample Problems

In the preceeding chapter, instructions were given showing how to prepare input for the code and how to run the code using the IBM 1620 digital computer. This chapter is devoted to exhibiting "OLD BARNYARD" output for selected problems. Further explanation of some of the problems is given below. The following is a list of the sample problems included in this chapter:

(1) Pure Carbon

a) 1 fast group, T = 293 deg. Kelvin

b) 2 fast groups, T = 293 deg Kelvin

c) 1 fast group, T = 1273 deg. Kelvin

(2) Water

a) 1 fast group

b) 2 fast groups

(3) Pure Uranium-235

a) 1 fast group

b) 3 fast groups

(4) Air Force Nuclear Engineering Test Facility (NETF) core

a) 1 fast group

b) 2 fast groups

(5) Air Force Institute of Technology (AFIT) Subcritical core

a) 1 fast group.

b) 2 fast groups

(6) Romaska Core

a) 1 fast group

b) 3 fast groups.

The first two problems listed above are self explanatory. Problem three, pure Uranium-235 is shown in figures 10 and 11 for one and three fast groups respectively. This is, of course, a fast system with no moderator present. The reader will note that in this case thermal calculations are not performed, as was explained earlier in Chapter II. Therefore, only ten epithermal group fluxes are output plus the age to the Indium resonance in chain one. Also, the flux plot is terminated at lethargy 16 in this case.

The Air Force Nuclear Engineering Test Facility (NETF) is a ten megawatt, approximately 90% enriched, unpressurized, thermal research reactor. The core has a rectangular parallelepiped arrangement of MTR flat-plate type fuel elements, cooled and moderated by light water.

The Air Force Institute of Technology (AFIT) subcritical reactor is a natural uranium reactor with cylindrical, aluminum encased, fuel elements arranged in a cylindrical pattern. The core is moderated with light water.

Finally, the Romaska reactor is a Russian direct energy converter reactor with a UC_2, C core. The Romaska core has a Carbon-Uranium atom ratio of approximately 6 to 1. Thus, the Romaska core is a fast reactor system. Output for this core is shown in figures 16 and 17 for one and three fast groups respectively. Since there is some moderator in the system, thermal calculations are performed even though the thermal flux is very low.

JOB OLD BARNYARD, CHAIN 1

CARBON 1 FAST GROUP 21 FEB, 1967

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 293.0 KELVIN

ATOM FRACTIONS ARE

CARBON 1.00000000

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	5.3003E-02
2	1.00	3.6788E+06	1.0824E-01	2.8939E-01
3	1.50	2.2313E+06	2.1044E-01	7.6130E-01
4	2.00	1.3534E+06	2.3139E-01	1.3565E+00
5	2.50	8.2085E+05	1.8048E-01	1.3975E+00
6	3.00	4.9787E+05	1.1483E-01	1.4450E+00
7	5.00	6.7379E+04	1.2439E-01	1.3667E+00
8	8.00	3.3546E+03	7.2094E-03	1.1381E+00
9	12.00	6.1442E+01	.0000E+00	1.0195E+00
10	16.00	1.1254E+00	.0000E+00	1.0454E+00
11	17.00	4.1399E-01	.0000E+00	1.1516E+00
12	17.50	2.5110E-01	.0000E+00	2.1834E+00
13	18.00	1.5230E-01	.0000E+00	1.7762E+01
14	18.50	9.2374E-02	.0000E+00	6.5395E+01
15	19.00	5.6028E-02	.0000E+00	1.0000E+02
16	19.50	3.3983E-02	.0000E+00	8.7568E+01
17	20.00	2.0612E-02	.0000E+00	5.4531E+01
18	20.50	1.2502E-02	.0000E+00	2.7603E+01
19	21.00	7.5826E-03	.0000E+00	1.2321E+01
20	21.50	4.5991E-03	.0000E+00	5.0958E+00
21	22.00	2.7895E-03	.0000E+00	2.0123E+00
22	22.50	1.6919E-03	.0000E+00	7.7271E-01
23	23.00	1.0262E-03	.0000E+00	2.9172E-01

AGE TO INDIUM RESONANCE (1.46EV) IS	3.2282E+02	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	3.2706E+02	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	3.0762E+03	CM2
TOTAL MIGRATION AREA IS	3.4032E+03	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	2.5493E+03	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	2.5200E+03	M/SEC
ABSORPTION PARAMETER IS	2.4099E-02	

Figure 5a: Carbon, 1 Fast Group, Chain 1

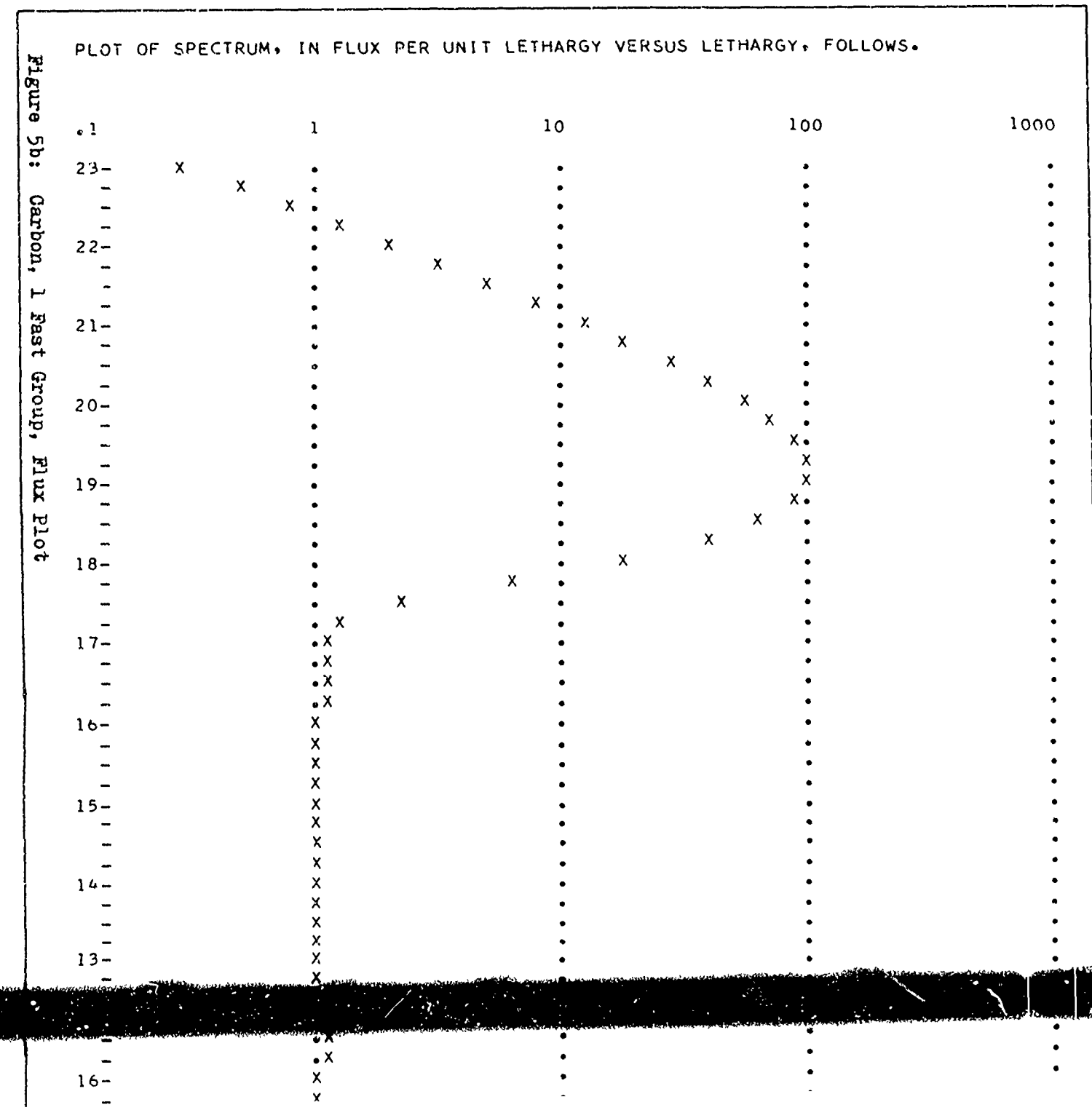


Figure 5b: Carbon, 1 Fast Group, Flux Plot

B.

C.

```
     JOB  OLD BARNYARD CHAIN 22

THE TOTAL NUMBER OF BROAD GROUPS IS  2
OUTPUT WILL BE FOR  2  BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 11 12


NUCLIDE IS CARBON

     ITS NUMBER DENSITY IS 8.0230E-02 PER BARN-CM


MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER                                          ELASTIC      ELASTIC       TOTAL
FROM   TO       INELASTIC          N-2N           (P0)         (P1)       TRANSFER

  1     1        .00068          .00000        4.21159       .28014       4.21227
  1     2        .00000          .00000         .04126      -.01204        .04126
  2     2        .00000          .00000        4.80000       .26688       4.80000


GROUP      SIGTR          SIGR          SIGA        NUSIGF       SOURCE
   1      3.98343        .04126        .00000       .00000      1.00000
   2      4.53661        .00349        .00349       .00000


MAXWELL-BOLTZMAN FACTOR = 1.128,     AVERAGE X =  1.1468881F+00
```

Figure 5c: Carbon, 1 Fast Group, Chain 2

JOB OLD BARNYARD, CHAIN 1

CARBON 2 FAST GROUPS 21 FEB, 1967

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 293.0 KELVIN

ATOM FRACTIONS ARE

CARBON 1.00000000

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	5.3003E-02
2	1.00	3.6788E+06	1.0824E-01	2.8939E-01
3	1.50	2.2313E+06	2.1044E-01	7.6130 01
4	2.00	1.3534E+06	2.3139E-01	1.3565E+00
5	2.50	8.2085E+05	1.8048E-01	1.3975E+00
6	3.00	4.9787E+05	1.1483E-01	1.4450E+00
7	5.00	6.7379E+04	1.2439E-01	1.3667E+00
8	8.00	3.3546E+03	7.2094E-03	1.1381E+00
9	12.00	6.1442E+01	.0000E+00	1.0195E+00
10	16.00	1.1254E+00	.0000E+00	1.0454E+00
11	17.00	4.1399E-01	.0000E+00	1.1516E+00
12	17.50	2.5110E-01	.0000E+00	2.1834E+00
13	18.00	1.5230E-01	.0000E+00	1.7762E+01
14	18.50	9.2374E-02	.0000E+00	6.5395E+01
15	19.00	5.6028E-02	.0000E+00	1.0000E+02
16	19.50	3.3983E-02	.0000E+00	8.7568E+01
17	20.00	2.0612E-02	.0000E+00	5.4531E+01
18	20.50	1.2502E-02	.0000E+00	2.7603E+01
19	21.00	7.5826E-03	.0000E+00	1.2321E+01
20	21.50	4.5991E-03	.0000E+00	5.0958E+00
21	22.00	2.7895E-03	.0000E+00	2.0123E+00
22	22.50	1.6919E-03	.0000E+00	7.7271E-01
23	23.00	1.0262E-03	.0000E+00	2.9172E-01

AGE TO INDIUM RESONANCE (1.46EV) IS	3.2282E+02	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	3.2706E+02	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	3.0762E+03	CM2
TOTAL MIGRATION AREA IS	3.4032E+03	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	2.5493E+03	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	2.5200E+03	M/SEC
ABSORPTION PARAMETER IS	2.4099E-02	

Figure 6a: Carbon, 2 Fast Groups, Chain 1

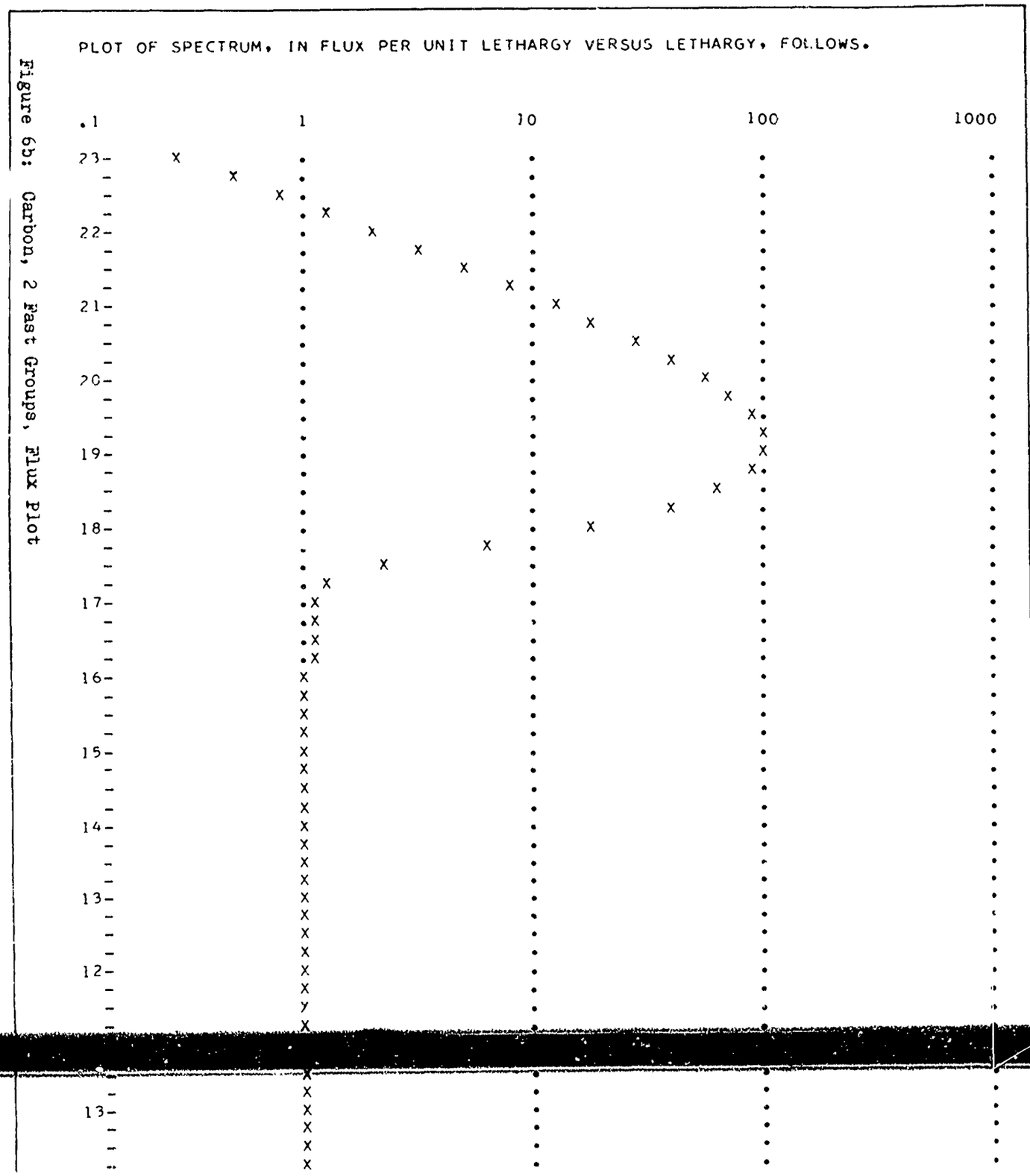


Figure 6b: Carbon, 2 Fast Groups, Flux Plot

A.

B.

JOB OLD BARNYARD CHAIN 2Z

THE TOTAL NUMBER OF BROAD GROUPS IS 3
OUTPUT WILL BE FOR 3 BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 6 11 12

NUCLIDE IS CARBON

ITS NUMBER DENSITY IS 8.0230E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00403	.00000	2.14892	.32287	2.15295
1	2	.00061	.00000	.24404	-.06270	.24465
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	4.52366	.28357	4.52366
2	3	.00000	.00000	.04834	-.01410	.04834
3	3	.00000	.00000	4.80000	.26688	4.80000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	2.13743	.24465	.00000	.00000	.86840
2	4.30254	.04834	.00000	.00000	.13160
3	4.53661	.00349	.00349	.00000	

MAXWELL-BOLTZMAN FACTOR = 1.128, AVERAGE X = 1.1468881E+00

Figure 6c: Carbon, 2 Fast Groups, Chain 2

JOB OLD BARNYARD, CHAIN 1

CARBON AT 1000 DEG CENTIGRADE 1 FAST GROUP

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 1274.0 KELVIN

ATOM FRACTIONS ARE

CARBON 1.00000000

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	6.6410E-02
2	1.00	3.6788E+06	1.0824E-01	3.6259E-01
3	1.50	2.2313E+06	2.1044E-01	9.5386E-01
4	2.00	1.3534E+06	2.3129E-01	1.6996E+00
5	2.50	8.2085E+05	1.9048E-01	1.7509E+00
6	3.00	4.9787E+05	1.1483E-01	1.8105E+00
7	5.00	6.7379E+04	1.2439E-01	1.7124E+00
8	8.00	3.3546E+03	7.2094E-03	1.4260E+00
9	12.00	6.1442E+01	.0000E+00	1.2773E+00
10	16.00	1.1254E+00	.0000E+00	1.3098E+00
11	17.00	4.1399E-01	.0000E+00	6.1758E+01
12	17.50	2.5110E-01	.0000E+00	9.9540E+01
13	18.00	1.5230E-01	.0000E+00	8.9865E+01
14	18.50	9.2374E-02	.0000E+00	5.6999E+01
15	19.00	5.6028E-02	.0000E+00	2.9177E+01
16	19.50	3.3983E-02	.0000E+00	1.3114E+01
17	20.00	2.0612E-02	.0000E+00	5.4468E+00
18	20.50	1.2502E-02	.0000E+00	2.1567E+00
19	21.00	7.5826E-03	.0000E+00	8.2955E-01
20	21.50	4.5991E-03	.0000E+00	3.1352E-01
21	22.00	2.7895E-03	.0000E+00	1.1724E-01
22	22.50	1.6919E-03	.0000E+00	4.3556E-02
23	23.00	1.0262E-03	.0000E+00	1.6119E-02

AGE TO INDIUM RESONANCE (1.46EV) IS	3.2476E+02	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	3.2902E+02	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	6.3847E+03	CM2
TOTAL MIGRATION AREA IS	6.7137E+03	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	5.3968E+03	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	5.1951E+03	M/SEC
ABSORPTION PARAMETER IS	1.1561E-02	

Figure 7a: Carbon, 1 Fast Group, 1000 Deg. Kelvin, Chain 1

Figure 7b: Carbon, 1 Fast Group, 1000 Deg. Kelvi. Flux Plot

A.

B.

JOB OLD BARNYARD CHAIN 2Z

THE TOTAL NUMBER OF BROAD GROUPS IS 2
OUTPUT WILL BE FOR 2 BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 11 12

NUCLIDE IS CARBON

ITS NUMBER DENSITY IS 7.9990E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00068	.00000	4.21159	.28014	4.21227
1	2	.00000	.00000	.04126	-.01204	.04126
2	2	.00000	.00000	4.80000	.26688	4.80000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	3.98543	.04126	.00000	.00000	1.00000
2	4.53665	.00353	.00353	.00000	

MAXWELL-BOLTZMAN FACTOR = 1.128, AVERAGE X = 1.1343391E+00

Figure 7c: Carbon, 1 Fast Group, 1000 Deg. Kelvin, Chain 2

JOB OLD BARNYARD, CHAIN 1

WATER 1 FAST GROUP 21 FEB. 1967

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 293.0 KELVIN

ATOM FRACTIONS ARE

HYDROGEN	.66666666
OXYGEN	.33333333

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+05	2.3023E-02	5.6299E-01
2	1.00	3.6788E+06	1.0824E-01	2.1187E+00
3	1.50	2.2313E+06	2.1044E-01	4.1606E+00
4	2.00	1.3534E+06	2.3139E-01	4.1480E+00
5	2.50	8.2085E+05	1.8048E-01	3.5712E+00
6	3.00	4.9787E+05	1.1483E-01	3.0086E+00
7	5.00	6.7379E+04	1.2439E-01	1.6096E+00
8	8.00	3.3546E+03	7.2094E-03	8.5745E-01
9	12.00	6.1442E+01	.0000E+00	7.6440E-01
10	16.00	1.1254E+00	.0000E+00	7.5715E-01
11	17.00	4.1399E-01	.0000E+00	8.3546E-01
12	17.50	2.5110E-01	.0000E+00	1.8262E+00
13	18.00	1.5230E-01	.0000E+00	1.7363E+01
14	18.50	9.2374E-02	.0000E+00	6.5127E+01
15	19.00	5.6028E-02	.0000E+00	1.0000E+02
16	19.50	3.3983E-02	.0000E+00	8.7712E+01
17	20.00	2.0612E-02	.0000E+00	5.4671E+01
18	20.50	1.2502E-02	.0000E+00	2.7690E+01
19	21.00	7.5826E-03	.0000E+00	1.2365E+01
20	21.50	4.5991E-03	.0000E+00	5.1154E+00
21	22.00	2.7895E-03	.0000E+00	2.0205E+00
22	22.50	1.6919E-03	.0000E+00	7.7599E-01
23	23.00	1.0262E-03	.0000E+00	2.9300E-01

AGE TO INDIUM RESONANCE (1.46EV) IS	2.5224E+01	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	2.5338E+01	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	6.2752E+00	CM2
TOTAL MIGRATION AREA IS	3.1614E+01	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	2.5493E+03	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	2.5111E+03	M/SEC
ABSORPTION PARAMETER IS	1.7454E-02	

Figure 8a: Water, 1 Fast Group, Chain 1

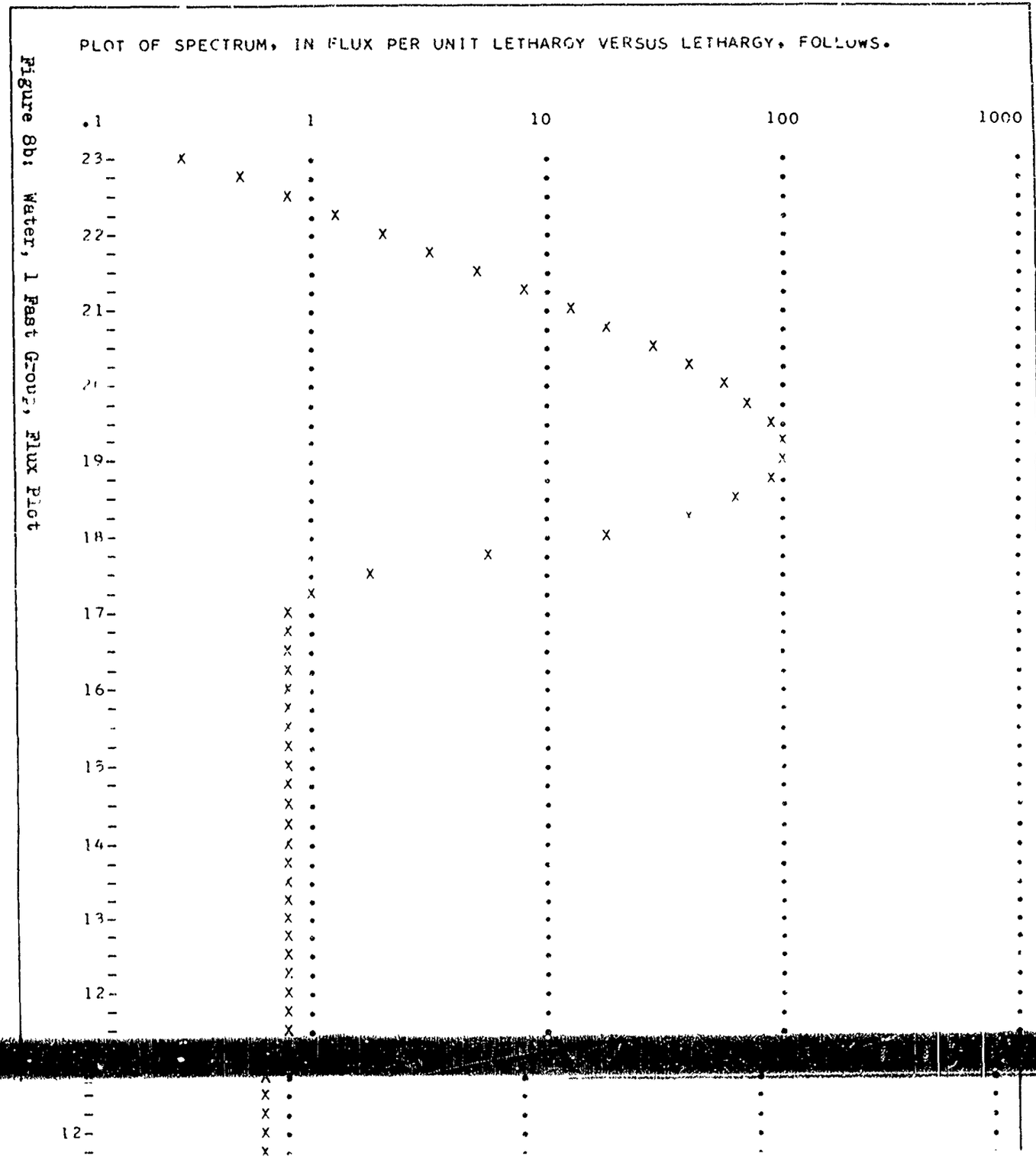


Figure 8b: Water, 1 Fast Group, Flux Plot

A.

B_c

```
     JOB  OLD BARNYARD CHAIN 27

THE TOTAL NUMBER OF BROAD GROUPS IS  2
OUTPUT WILL BE FOR  2  BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 11 12


NUCLIDE IS HYDROGEN

     ITS NUMBER DENSITY IS 6.6800E-02 PER BARN-CM


MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER                                          ELASTIC      ELASTIC       TOTAL
FROM   TO       INELASTIC        N-2N             (P0)         (P1)       TRANSFER

  1     1        .00000         .00000        10.63433      7.21949      10.63433
  1     2        .00000         .00000          .70545       .31223        .70545
  2     2        .00000         .00000        38.00000     25.13320      38.00000


GROUP      SIGTR          SIGR          SIGA        NUSIGF       SOURCE
   1      3.81308        .71048        .00502       .00000      1.00000
   2     13.15730        .29050        .29050       .00000


NUCLIDE IS OXYGEN

     ITS NUMBER DENSITY IS 3.3400E-02 PER BARN-CM


MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER                                          ELASTIC      ELASTIC       TOTAL
FROM   TO       INELASTIC        N-2N             (P0)         (P1)       TRANSFER

  1     1        .00118         .00000         3.34723       .25328       3.34840
  1     2        .00000         .00000          .01400      -.00420        .01400
  2     2        .00000         .00000         4.20000       .17514       4.20000


GROUP      SIGTR          SIGR          SIGA        NUSIGF       SOURCE
   1      3.11941        .02009        .00609       .00000      1.00000
   2      4.02661        .00175        .00175       .00000


MAXWELL-BOLTZMAN FACTOR = 1.128,    AVERAGE X =  1.1428491E+00
```

Figure 8c: Water, 1 Fast Group, Chain 2

JOB OLD BARNYARD, CHAIN 1

WATER 2 FAST GROUPS 21 FEB, 1967

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 293.0 KELVIN

ATOM FRACTIONS ARE

HYDROGEN	.66666666
OXYGEN	.33333333

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	5.6299E-01
2	1.00	3.6788E+06	1.0824E-01	2.1187E+00
3	1.50	2.2313E+06	2.1044E-01	4.1606E+00
4	2.00	1.3534E+06	2.3139E-01	4.1480E+00
5	2.50	8.2085E+05	1.8048E-01	3.5712E+00
6	3.00	4.9787E+05	1.1483E-01	3.0086E+00
7	5.00	6.7379E+04	1.2439E-01	1.6096E+00
8	8.00	3.3546E+03	7.2094E-03	8.5745E-01
9	12.00	6.1442E+01	.0000E+00	7.6440E-01
10	16.00	1.1254E+00	.0000E+00	7.5715E-01
11	17.00	4.1399E-01	.0000E+00	8.3546E-01
12	17.50	2.5110E-01	.0000E+00	1.8262E+00
13	18.00	1.5230E-01	.0000E+00	1.7363E+01
14	18.50	9.2374E-02	.0000E+00	6.5127E+01
15	19.00	5.6028E-02	.0000E+00	1.0000E+02
16	19.50	3.3983E-02	.0000E+00	8.7712E+01
17	20.00	2.0612E-02	.0000E+00	5.4671E+01
18	20.50	1.2502E-02	.0000E+00	2.7690E+01
19	21.00	7.5826E-03	.0000E+00	1.2365E+01
20	21.50	4.5991E-03	.0000E+00	5.1154E+00
21	22.00	2.7895E-03	.0000E+00	2.0205E+00
22	22.50	1.6919E-03	.0000E+00	7.7599E-01
23	23.00	1.0262E-03	.0000E+00	2.9300E-01

AGE TO INDIUM RESONANCE (1.46EV) IS	2.5224E+01	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	2.5338E+01	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	6.2752E+00	CM2
TOTAL MIGRATION AREA IS	3.1614E+01	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	2.5493E+03	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	2.5111E+03	M/SEC
ABSORPTION PARAMETER IS	1.7454E-02	

Figure 3a: Water, 2 Fast Groups, Chain 1

PLOT OF SPECTRUM, IN FLUX PER UNIT LETHARGY VERSUS LETHARGY, FOLLOWS.

Fig

Figure 9b: Water, 2 Fast Groups, Flux Plot

B.

JOB OLD BARNYARD CHAIN 2Z

THE TOTAL NUMBER OF BROAD GROUPS IS 3
OUTPUT WILL BE FOR 3 BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 6 11 12

NUCLIDE IS HYDROGEN

ITS NUMBER DENSITY IS 6.6800E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00000	.00000	1.82518	1.46149	1.82518
1	2	.00000	.00000	1.45849	.72167	1.45849
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	15.76453	10.73447	15.76453
2	3	.00000	.00000	1.19781	.53015	1.19781
3	3	.00000	.00000	38.00000	25.13320	38.00000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	1.10052	1.45849	.00000	.00000	.86840
2	5.70625	1.20634	.00853	.00000	.13160
3	13.15730	.29050	.29050	.00000	

NUCLIDE IS OXYGEN

ITS NUMBER DENSITY IS 3.3400E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00244	.00000	2.44289	.39193	2.44532
1	2	.00043	.00000	.11221	.00256	.11264
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	3.90007	.15472	3.90007
2	3	.00000	.00000	.02377	-.00713	.02377
3	3	.00000	.00000	4.20000	.17514	4.20000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	2.17829	.12746	.01481	.00000	.86840
2	3.77624	.02377	.00000	.00000	.13160
3	4.02661	.00175	.00175	.00000	

MAXWELL-BOLTZMAN FACTOR = 1.128, AVERAGE X = 1.1428491E+00

Figure 9c: Water, 2 Fast Groups, Chain 2

JOB OLD BARNYARD, CHAIN 1

PURE URANIUM-235 1 FAST GROUP 21 FEB, 1967

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 293.0 KELVIN

ATOM FRACTIONS ARE

URANIUM 235 1.00000000

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	6.3595E+00
2	1.00	3.6788E+06	1.0824E-01	3.3570E+01
3	1.50	2.2313E+06	2.1044E-01	6.6729E+01
4	2.00	1.3534E+06	2.3139E-01	8.9339E+01
5	2.50	8.2085E+05	1.8048E-01	1.0000E+02
6	3.00	4.9787E+05	1.1483E-01	9.9793E+01
7	5.00	6.7379E+04	1.2439E-01	6.6472E+01
8	8.00	3.3546E+03	7.2094E-03	4.7531E+00
9	12.00	6.1442E+01	.0000E+00	3.8288E-03
10	16.00	1.1254E+00	.0000E+00	8.0036E-08

AGE TO INDIUM RESONANCE (1.46EV) IS 1.9578E+01 CM2

Figure 10a: Uranium-235, 1 Fast Group, Chain 1

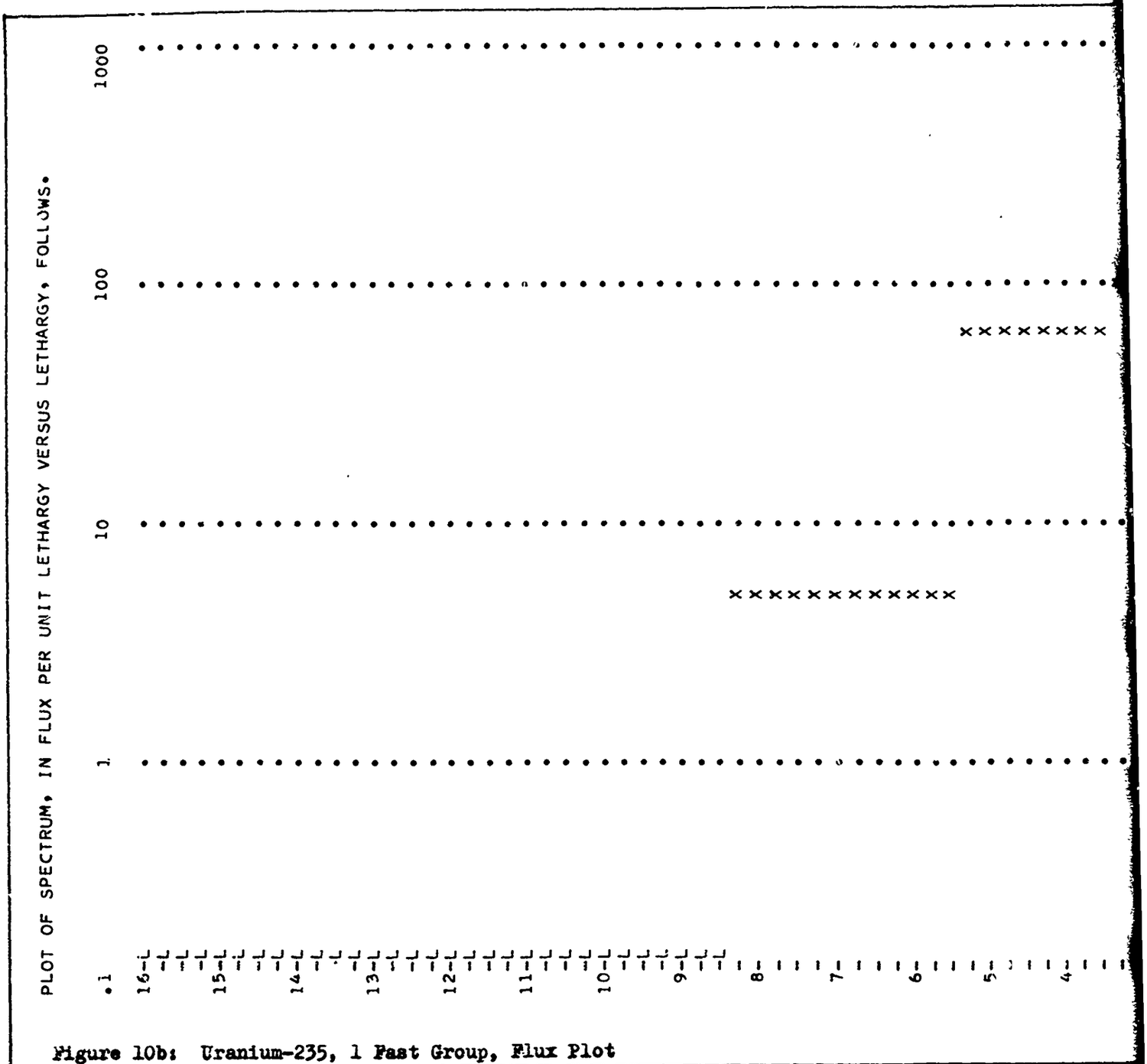
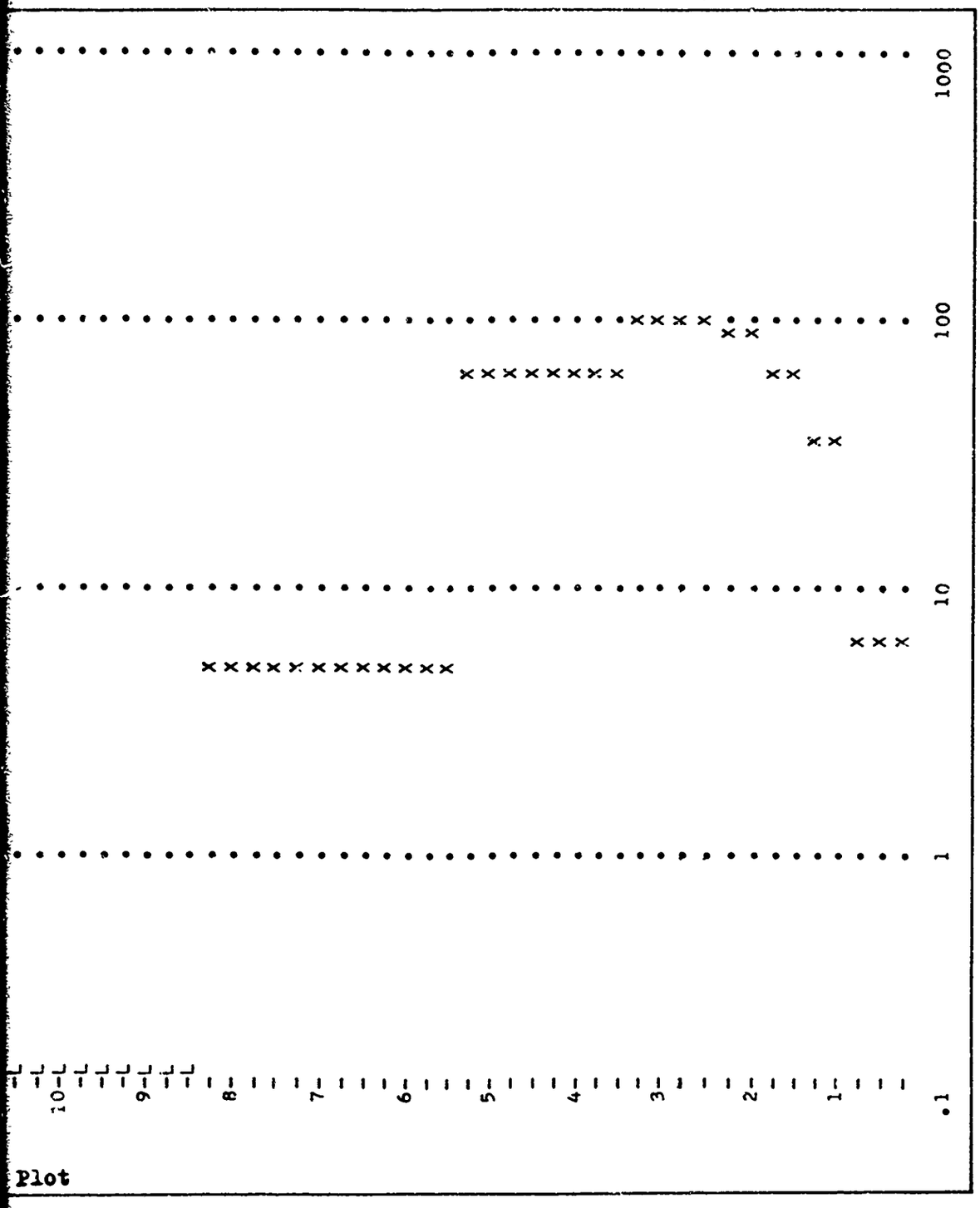


Figure 10b: Uranium-235, 1 Fast Group, Flux Plot

A.



B.

JOB OLD BARNYARD CHAIN 2Z

THE TOTAL NUMBER OF BROAD GROUPS IS 1
OUTPUT WILL BE FOR 1 BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 11 12

NUCLIDE IS URANIUM 235

ITS NUMBER DENSITY IS 4.7900E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	1.23713	.00000	5.64214	1.68886	6.87927
1	2	.00000	.00000	.00000	.00000	.00000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	6.63000	1.43959	1.43959	3.61426	1.00000

Figure 10c: Uranium-235, 1 Fast Group, Chain 2

JOB OLD BARNYARD, CHAIN 1

PURE URANIUM-235 3 FAST GROUPS 21 FEB.1967

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 293.0 KELVIN

ATOM FRACTIONS ARE

URANIUM 235 1.00000000

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	6.3595E+00
2	1.00	3.6788E+06	1.0824E-01	3.3570E+01
3	1.50	2.2313E+06	2.1044E-01	6.6729E+01
4	2.00	1.3534E+06	2.3139E-01	8.9339E+01
5	2.50	8.2085E+05	1.8048E-01	1.0000E+02
6	3.00	4.9787E+05	1.1483E-01	9.9793E+01
7	5.00	6.7379E+04	1.2439E-01	6.6472E+01
8	8.00	3.3546E+03	7.2094E-03	4.7531E+00
9	12.00	6.1442E+01	.0000E+00	3.8288E-03
10	16.00	1.1254E+00	.0000E+00	8.0036E-08

AGE TO INDIUM RESONANCE (1.46EV) IS 1.9578E+01 CM2

Figure 11a: Uranium-235, 3 Fast Groups, Chain 1

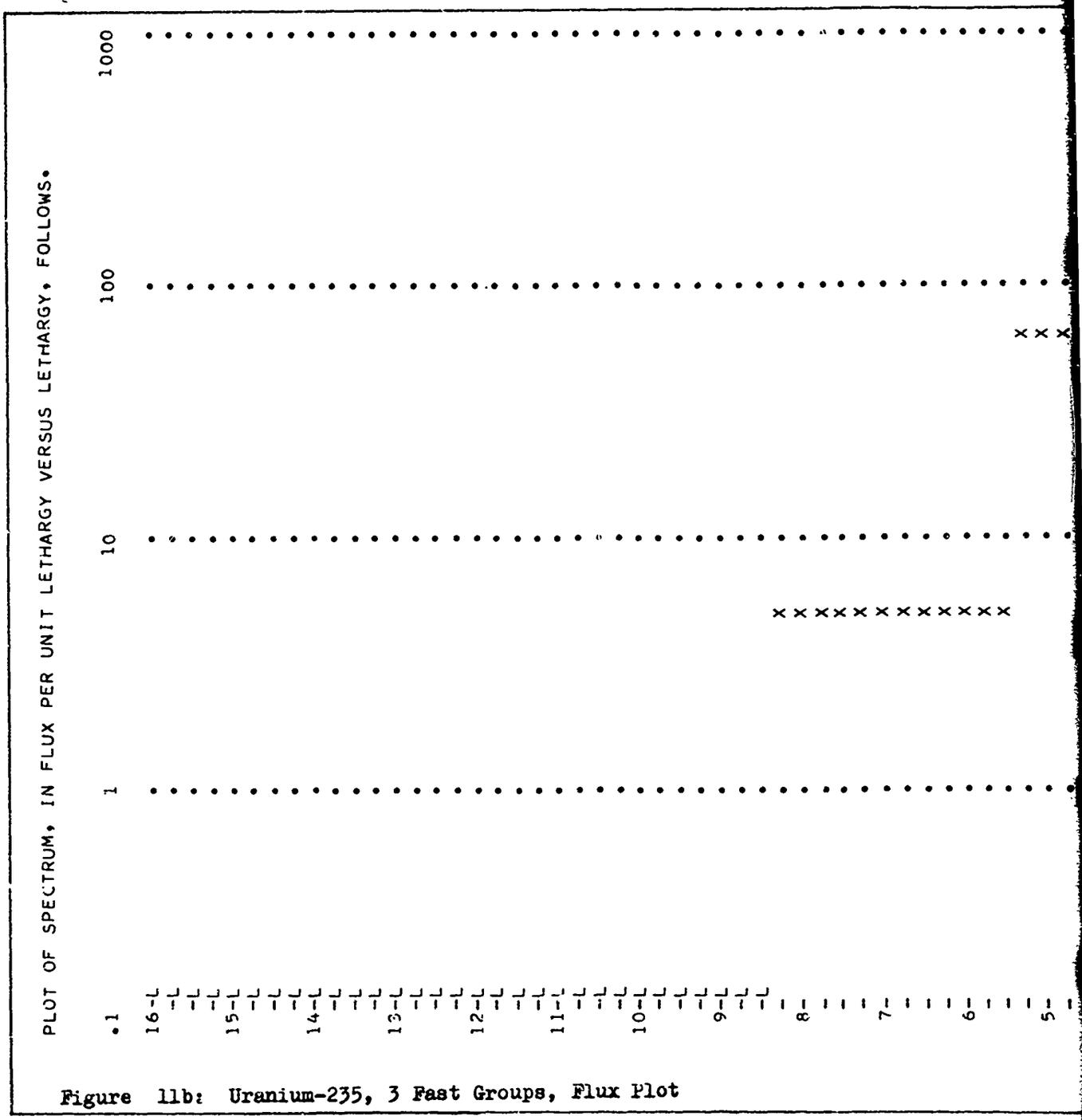
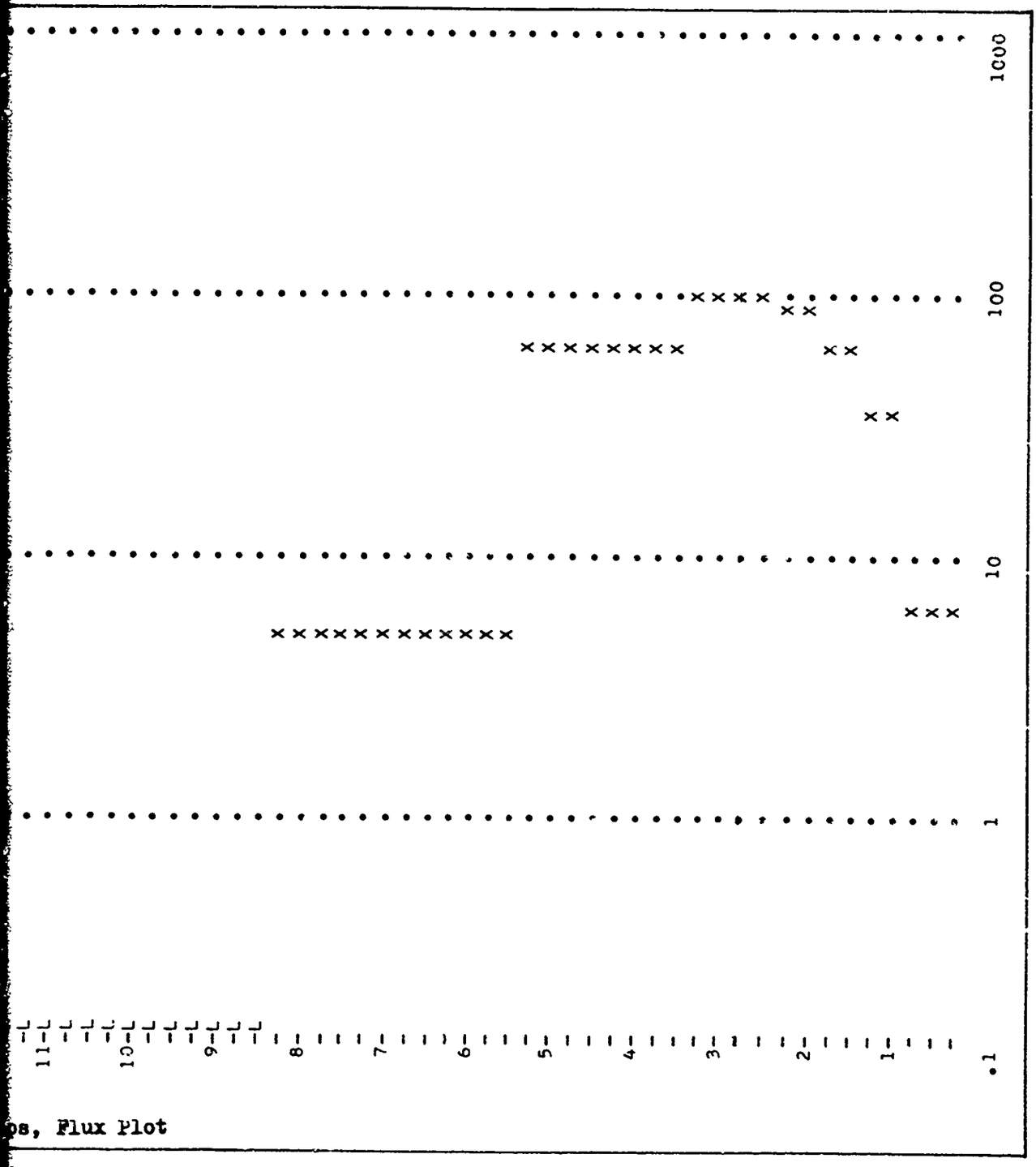


Figure 11b: Uranium-235, 3 Fast Groups, Flux Plot

A.



os, Flux Plot

B.

JOB OLD BARNYARD CHAIN 2Z

THE TOTAL NUMBER OF BROAD GROUPS IS 3
OUTPUT WILL BE FOR 3 BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 3 6 11 12

NUCLIDE IS URANIUM 235

ITS NUMBER DENSITY IS 4.7900E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.05410	.00000	4.76586	4.03780	4.81996
1	2	1.17413	.00000	.02081	-.61377	1.19494
1	3	.70041	.00000	.00000	.00000	.70041
1	4	.00000	.00000	.00000	.00000	.00000
2	2	.44240	.00000	4.07620	1.57196	4.51861
2	3	.97542	.00000	.01892	-.06592	.99434
2	4	.00000	.00000	.00000	.00000	.00000
3	3	.80920	.00000	7.47117	1.23983	8.28037
3	4	.00000	.00000	.00000	.00000	.00000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	4.57927	3.18334	1.28798	3.91628	.34170
2	5.26345	2.25088	1.25654	3.26017	.52670
3	8.71479	1.67426	1.67426	3.85258	.13160

Figure 11c: Uranium-235, 3 Fast Groups, Chain 2

JOB OLD BARNYARD, CHAIN 1

NETF CORE (AN MTR TYPE REACTOR) 1 FAST GROUP 21 FEB, 1967

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 293.0 KELVIN

ATOM FRACTIONS ARE

HYDROGEN	.48160826
OXYGEN	.24079822
ALUMINUM	.27604967
URANIUM 235	.00139125
URANIUM 238	.00015263

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	1.3110E+01
2	1.00	3.6738E+06	1.0824E-01	5.0607E+01
3	1.50	2.2313E+06	2.1044E-01	1.0000E+02
4	2.00	1.3534E+06	2.3139E-01	9.9896E+01
5	2.50	8.2085E+05	1.8048E-01	8.6102E+01
6	3.00	4.9787E+05	1.1483E-01	7.2588E+01
7	5.00	6.7379E+04	1.2439E-01	3.8833E+01
8	8.00	3.3546E+03	7.2094E-03	2.0604E+01
9	12.00	6.1442E+01	.0000E+00	1.8118E+01
10	16.00	1.1254E+00	.0000E+00	1.7140E+01
11	17.00	4.1399E-01	.0000E+00	1.7765E+01
12	17.50	2.5110E-01	.0000E+00	1.9334E+01
13	18.00	1.5230E-01	.0000E+00	2.5553E+01
14	18.50	9.2374E-02	.0000E+00	3.5858E+01
15	19.00	5.6028E-02	.0000E+00	3.6817E+01
16	19.50	3.3983E-02	.0000E+00	2.6404E+01
17	20.00	2.0612E-02	.0000E+00	1.4587E+01
18	20.50	1.2502E-02	.0000E+00	6.8143E+00
19	21.00	7.5826E-03	.0000E+00	2.8722E+00
20	21.50	4.5991E-03	.0000E+00	1.1384E+00
21	22.00	2.7895E-03	.0000E+00	4.3532E-01
22	22.50	1.6919E-03	.0000E+00	1.6308E-01
23	23.00	1.0262E-03	.0000E+00	6.0405E-02

AGE TO INDIUM RESONANCE (1.46EV) IS	6.3752E+01	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	6.3939E+01	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	3.1354E+00	CM2
TOTAL MIGRATION AREA IS	6.7075E+01	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	2.8887E+03	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	3.6731E+03	M/SEC
ABSORPTION PARAMETER IS	1.0226E+00	
SCATTERING PER RESONANCE ATOM IS	6.9315E+04	BARNS
EFFECTIVE RESONANCE INTEGRAL IS	2.5056E+02	BARNS

Figure 12a: NETF, 1 Fast Group, Chain 1

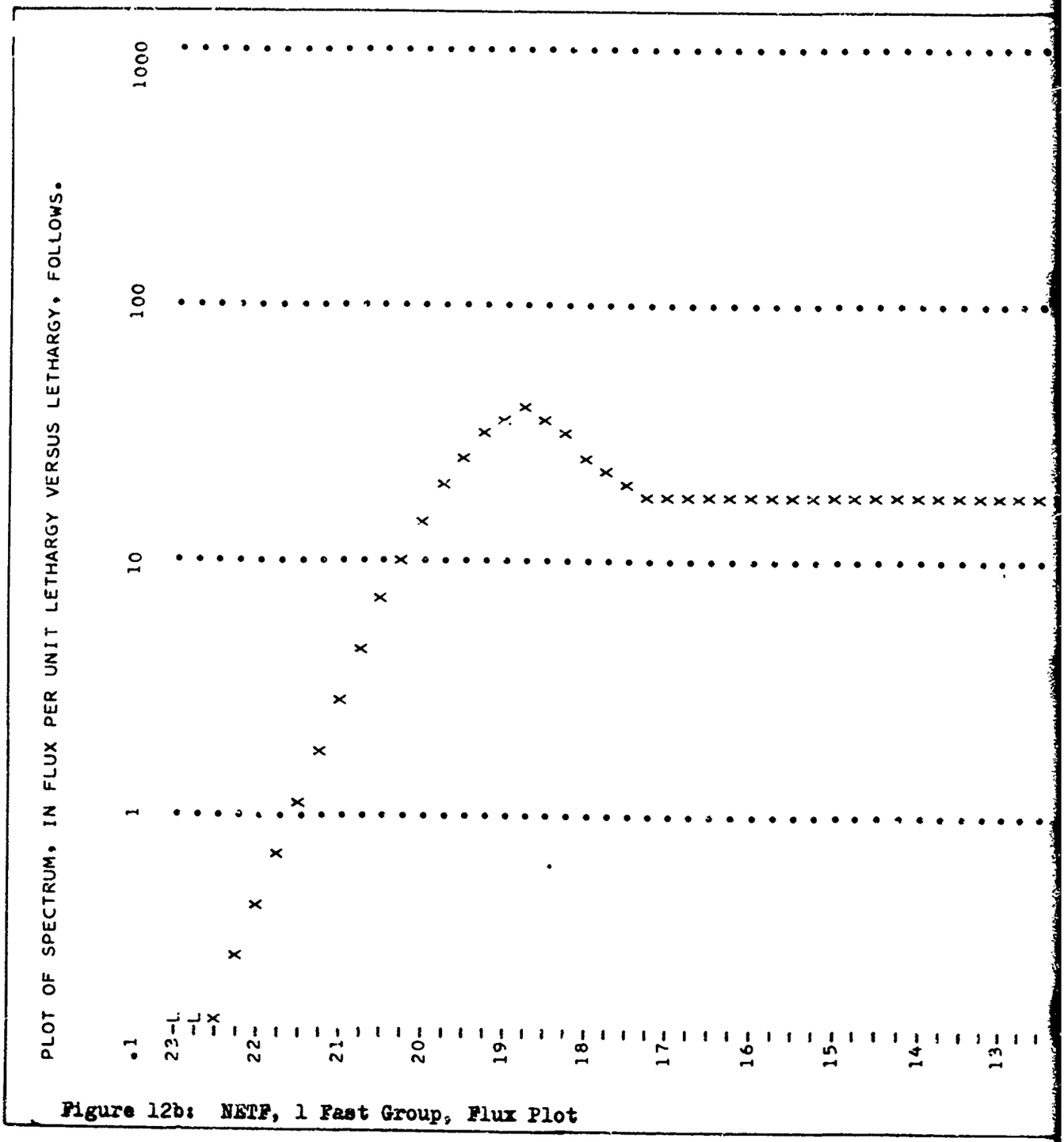
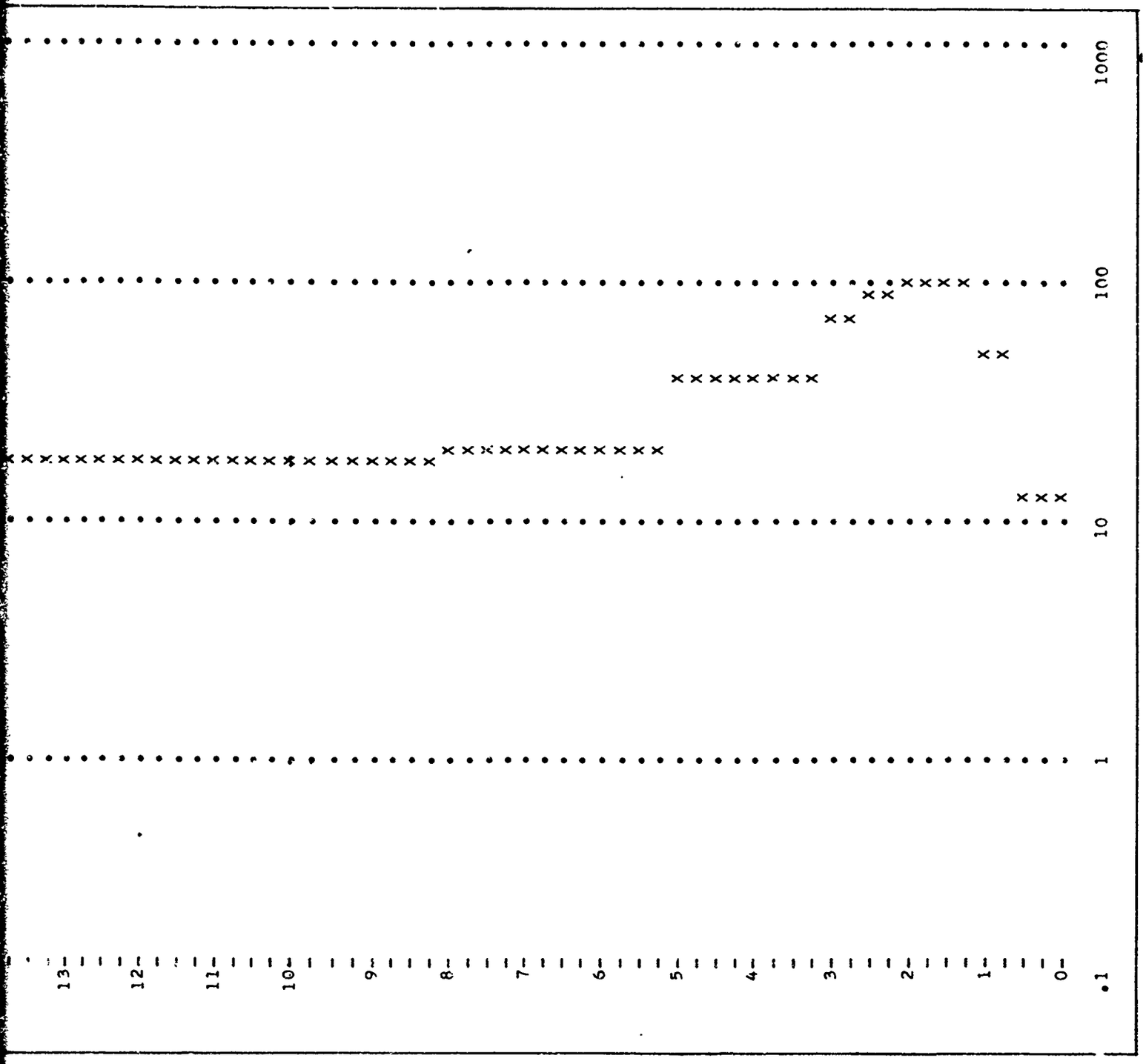


Figure 12b: NETF, 1 Fast Group, Flux Plot

A.



B.

JOB OLD BARNYARD CHAIN 2Z

THE TOTAL NUMBER OF BROAD GROUPS IS 2
OUTPUT WILL BE FOR 2 BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 11 12

NUCLIDE IS HYDROGEN

ITS NUMBER DENSITY IS 4.0713E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00000	.00000	10.55896	7.16066	10.55896
1	2	.00000	.00000	.66517	.29354	.66517
2	2	.00000	.00000	38.00000	25.13320	38.00000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	3.77468	.66992	.00475	.00000	1.00000
2	13.06540	.19860	.19860	.00000	

NUCLIDE IS OXYGEN

ITS NUMBER DENSITY IS 2.0336E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00116	.00000	3.34652	.25410	3.34767
1	2	.00000	.00000	.01311	-.00393	.01311
2	2	.00000	.00000	4.20000	.17514	4.20000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	3.11666	.01916	.00605	.00000	1.00000
2	4.02606	.00120	.00120	.00000	

Figure 12c: NETF, 1 Fast Group, Chain 2

NUCLIDE IS ALUMINUM

ITS NUMBER DENSITY IS 2.3336E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.05433	.00000	.00010	.00093	.05443
1	2	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	1.40000	.03444	1.40000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	.06197	.00847	.00847	.00000	1.00000
2	1.50972	.14416	.14416	.00000	

NUCLIDE IS URANIUM 235

ITS NUMBER DENSITY IS 1.1761E-04 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.80758	.00000	7.81200	1.19759	8.61957
1	2	.00000	.00000	.00363	-.00108	.00363
2	2	.00000	.00000	14.80000	.04144	14.80000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	24.27599	16.85293	16.84930	27.34165	1.00000
2	420.44480	405.68824	405.68824	829.68092	

Figure 12d: NETF, 1 Fast Group, Chain 2, Continued

NUCLIDE IS URANIUM 238

ITS NUMBER DENSITY IS 1.2903E-05 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.95131	.00806	12.89255	1.33711	13.85997
1	2	.00000	.00000	.00208	-.00063	.00208
2	2	.00000	.00000	13.80000	.03864	13.80000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	21.26361	8.74012	8.74610	.40161	1.00000
2	15.39440	1.63304	1.63304	.00000	

MAXWELL-BOLTZMAN FACTOR = 1.128, AVERAGE X = 1.6717270E+00

Figure 12e: NETF, 1 Fast Group, Chain 2, Continued

JOB OLD BARNYARD. CHAIN 1

NETF CORE (AN MTR TYPE REACTOR) 2 FAST GROUPS 28 FEB, 1967

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 293.0 KELVIN

ATOM FRACTIONS ARE

HYDROGEN	.48160826
OXYGEN	.24079822
ALUMINUM	.27604967
URANIUM 235	.00139125
URANIUM 238	.00015263

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	1.3110E+01
2	1.00	3.6788E+06	1.0824E-01	5.0607E+01
3	1.50	2.2313E+06	2.1044E-01	1.0000E+02
4	2.00	1.3534E+06	2.3139E-01	9.9896E+01
5	2.50	8.2085E+05	1.8048E-01	8.6102E+01
6	3.00	4.9787E+05	1.1483E-01	7.2588E+01
7	5.00	6.7379E+04	1.2439E-01	3.8833E+01
8	3.00	3.3546E+03	7.2094E-03	2.0604E+01
9	12.00	6.1442E+01	.0000E+00	1.8118E+01
10	16.00	1.1254E+00	.0000E+00	1.7140E+01
11	17.00	4.1399E-01	.0000E+00	1.7765E+01
12	17.50	2.5110E-01	.0000E+00	1.9334E+01
13	18.00	1.5230E-01	.0000E+00	2.5553E+01
14	18.50	9.2374E-02	.0000E+00	3.5858E+01
15	19.00	5.6028E-02	.0000E+00	3.6817E+01
16	19.50	3.3983E-02	.0000E+00	2.6404E+01
17	20.00	2.0612E-02	.0000E+00	1.4587E+01
18	20.50	1.2502E-02	.0000E+00	6.8143E+00
19	21.00	7.5826E-03	.0000E+00	2.8722E+00
20	21.50	4.5991E-03	.0000E+00	1.1384E+00
21	22.00	2.7895E-03	.0000E+00	4.3532E-01
22	22.50	1.6919E-03	.0000E+00	1.6308E-01
23	23.00	1.0262E-03	.0000E+00	6.0405E-02

AGE TO INDIUM RESONANCE (1.46EV) IS	6.3752E+01	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	6.3939E+01	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	3.1354E+00	CM2
TOTAL MIGRATION AREA IS	6.7075E+01	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	2.8887E+03	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	3.6731E+03	M/SEC
ABSORPTION PARAMETER IS	1.0226E+00	
SCATTERING PER RESONANCE ATOM IS	6.9315E+04	BARNS
EFFECTIVE RESONANCE INTEGRAL IS	2.5056E+02	BARNS

Figure 13a: NETF, 2 Fast Groups, Chain 1

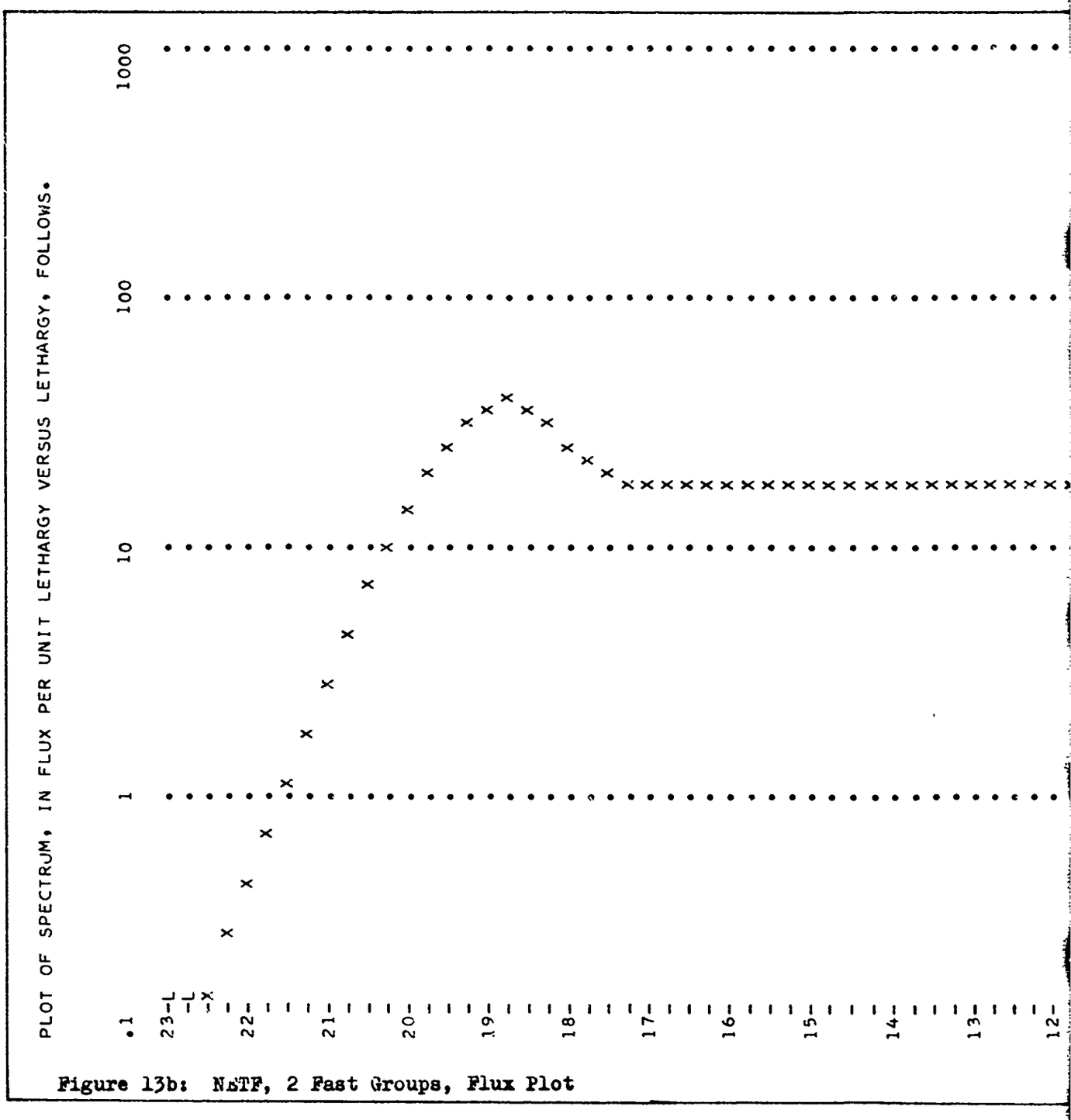


Figure 13b: NETF, 2 Fast Groups, Flux Plot

A.

B.

JOB OLD BARNYARD CHAIN 2Z

THE TOTAL NUMBER OF BROAD GROUPS IS 3
OUTPUT WILL BE FOR 3 BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 6 11 12

NUCLIDE IS HYDROGEN

ITS NUMBER DENSITY IS 4.0713E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00000	.00000	1.82604	1.46253	1.82604
1	2	.00000	.00000	1.46266	.72397	1.46266
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	15.74018	10.70556	15.74018
2	3	.00000	.00000	1.13920	.50273	1.13920
3	3	.00000	.00000	38.00000	25.13320	38.00000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	1.10220	1.46266	.00000	.00000	.86840
2	5.67925	1.14734	.00814	.00000	.13160
3	13.06540	.19860	.19860	.00000	

NUCLIDE IS OXYGEN

ITS NUMBER DENSITY IS 2.0356E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00236	.00000	2.44589	.39171	2.44825
1	2	.00042	.00000	.11264	.00257	.11306
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	3.90808	.15421	3.90808
2	3	.00000	.00000	.02245	-.00673	.02245
3	3	.00000	.00000	4.20000	.17514	4.20000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	2.18157	.12760	.01454	.00000	.86840
2	3.78305	.02245	.00000	.00000	.13160
3	4.02606	.00120	.00120	.00000	

Figure 13c: NETF, 2 Fast Groups, Chain 2

NUCLIDE IS ALUMINUM

ITS NUMBER DENSITY IS 2.3336E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.13057	.00000	.00024	.00224	.13081
1	2	.00000	.00000	.00000	.00000	.00000
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	.00000	.00000	.00000
2	3	.00000	.00000	.00000	.00000	.00000
3	3	.00000	.00000	1.40000	.03444	1.40000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	.13489	.00432	.00432	.00000	.86840
2	.01143	.01143	.01143	.00000	.13160
3	1.50972	.14416	.14416	.00000	

NUCLIDE IS URANIUM 235

ITS NUMBER DENSITY IS 1.1761E-04 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.77234	.00000	4.31206	2.30494	5.08440
1	2	.85655	.00000	.00884	-.03074	.86540
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.22226	.00000	10.29996	.43032	10.52222
2	3	.00000	.00000	.00621	-.00184	.00621
3	3	.00000	.00000	14.80000	.04144	14.80000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	4.95087	2.14068	1.27528	3.55094	.86840
2	38.04826	27.95452	27.94830	44.29636	.13160
3	420.44680	405.68824	405.68824	829.68092	

Figure 13d. NETF, 2 Fast Groups, Chain 2 Continued

NUCLIDE IS URANIUM 238

ITS NUMBER DENSITY IS 1.2903E-05 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	1.03487	.01530	4.95898	2.44602	6.02445
1	2	1.06085	.00406	.01103	.01275	1.08000
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.13573	.00000	18.53863	.53775	18.67436
2	3	.00000	.00000	.00356	-.00107	.00356
3	3	.00000	.00000	13.80000	.03864	13.80000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	5.03425	1.46857	.40793	.96514	.86840
2	32.82964	14.69195	14.68839	.00000	.13160
3	15.39440	1.63304	1.63304	.00000	

MAXWELL-BOLTZMAN FACTOR = 1.128, AVERAGE X = 1.6717270E+00

Figure 13e: NETF, 2 Fast Groups, Chain 2 Continued

JOB OLD BARNYARD, CHAIN 1

AFIT SUBCRITICAL REACTOR CORE 1 FAST GROUP

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 293.0 KELVIN

ATOM FRACTIONS ARE

HYDROGEN	.46238076
OXYGEN	.23119038
ALUMINUM	.08513929
URANIUM 235	.00161649
URANIUM 238	.22934297

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	9.4603E+00
2	1.00	3.6788E+06	1.0824E-01	4.2270E+01
3	1.50	2.2313E+06	2.1044E-01	8.4042E+01
4	2.00	1.3534E+06	2.3139E-01	9.3733E+01
5	2.50	8.2085E+05	1.8048E-01	1.0000E+02
6	3.00	4.9787E+05	1.1483E-01	9.9286E+01
7	5.00	6.7379E+04	1.2439E-01	6.0211E+01
8	8.00	3.3546E+03	7.2094E-03	3.1922E+01
9	12.00	6.1442E+01	.0000E+00	2.6493E+01
10	16.00	1.1254E+00	.0000E+00	1.7623E+01
11	17.00	4.1399E-01	.0000E+00	1.8736E+01
12	17.50	2.5110E-01	.0000E+00	2.1139E+01
13	18.00	1.5230E-01	.0000E+00	3.3059E+01
14	18.50	9.2374E-02	.0000E+00	5.8141E+01
15	19.00	5.6028E-02	.0000E+00	6.8442E+01
16	19.50	3.3983E-02	.0000E+00	5.3053E+01
17	20.00	2.0612E-02	.0000E+00	3.0782E+01
18	20.50	1.2502E-02	.0000E+00	1.4872E+01
19	21.00	7.5826E-03	.0000E+00	6.4232E+00
20	21.50	4.5991E-03	.0000E+00	2.5928E+00
21	22.00	2.7895E-03	.0000E+00	1.0053E+00
22	22.50	1.6919E-03	.0000E+00	3.8067E-01
23	23.00	1.0262E-03	.0000E+00	1.4218E-01

AGE TO INDIUM RESONANCE (1.46EV) IS	2.9947E+01	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	3.0004E+01	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	1.8123E+00	CM2
TOTAL MIGRATION AREA IS	3.1817E+01	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	2.5493E+03	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	3.2063E+03	M/SEC
ABSORPTION PARAMETER IS	5.8382E-01	
SCATTERING PER RESONANCE ATOM IS	7.6640E+01	BARNS
EFFECTIVE RESONANCE INTEGRAL IS	2.0765E+01	BARNS

Figure 14a: AFIT Subcrit., 1 Fast Group, Chain 1

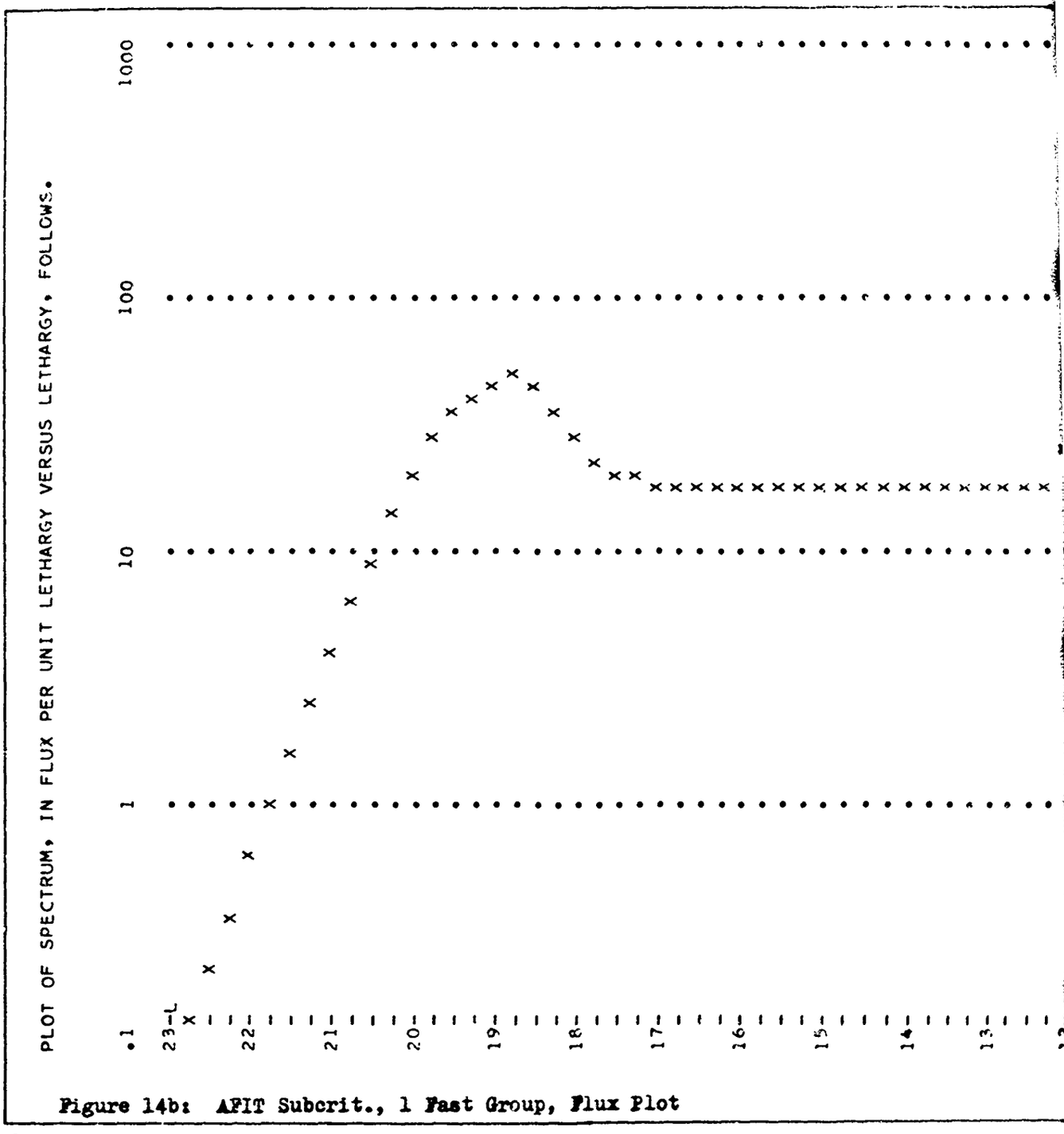


Figure 14b: AFIT Subcrit., 1 Fast Group, Flux Plot

A.

B.

JOB OLD BARNYARD CHAIN 2Z

THE TOTAL NUMBER OF BROAD GROUPS IS 2
OUTPUT WILL BE FOR 2 BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 11 12

NUCLIDE IS HYDROGEN

ITS NUMBER DENSITY IS 3.5040E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00000	.00000	11.44837	7.72347	11.44837
1	2	.00000	.00000	.55438	.24351	.55438
2	2	.00000	.00000	38.00000	25.13320	38.00000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	4.03172	.55834	.00396	.00000	1.00000
2	13.09431	.22751	.22751	.00000	

NUCLIDE IS OXYGEN

ITS NUMBER DENSITY IS 1.7520E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00068	.00000	3.52317	.24283	3.52385
1	2	.00000	.00000	.01084	-.00325	.01084
2	2	.00000	.00000	4.20000	.17514	4.20000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	3.29896	.01469	.00385	.00000	1.00000
2	4.02623	.00137	.00137	.00000	

Figure 14c: AFIT Subcrit., 1 Fast Broup, Chain 2

NUCLIDE IS ALUMINUM

ITS NUMBER DENSITY IS 6.4520E-03 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.04298	.00000	.00006	.00055	.04303
1	2	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	1.40000	.03444	1.40000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	.05048	.00800	.00800	.00000	1.00000
2	1.53071	.16515	.16515	.00000	

NUCLIDE IS URANIUM 235

ITS NUMBER DENSITY IS 1.2250E-04 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.70799	.00000	8.06126	1.04389	8.76924
1	2	.00000	.00000	.00300	-.00089	.00300
2	2	.00000	.00000	14.80000	.04144	14.80000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	23.28539	15.55915	15.55615	25.34965	1.00000
2	479.51266	464.75410	464.75410	950.47764	

Figure 14d: AFIT Subcrit., 1 Fast Group, Chain 2, Continued

NUCLIDE IS URANIUM 238

ITS NUMBER DENSITY IS 1.7380E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.77921	.00474	13.38083	1.19954	14.16952
1	2	.00000	.00000	.00172	-.00052	.00172
2	2	.00000	.00000	13.80000	.03864	13.80000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	13.77623	.80574	.80875	.28073	1.00000
2	15.63216	1.87080	1.87080	.00000	

MAXWELL-BOLTZMAN FACTOR = 1.128, AVERAGE X = 1.4592663E+00

Figure 14e: AFIT Subcrit., 1 Fast Group, Chain 2, Continued

JOB OLD BARNYARD, CHAIN 1

AFIT SUBCRITICAL REACTOR CORE 2 FAST GROUPS

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 293.0 KELVIN

ATOM FRACTIONS ARE

HYDROGEN	.46238076
OXYGEN	.23119038
ALUMINUM	.08513929
URANIUM 235	.00161649
URANIUM 238	.22934297

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	9.4603E+00
2	1.00	3.6788E+06	1.0824E-01	4.2270E+01
3	1.50	2.2313E+06	2.1044E-01	8.4042E+01
4	2.00	1.3534E+06	2.3139E-01	9.3733E+01
5	2.50	8.2085E+05	1.8048E-01	1.0000E+02
6	3.00	4.9787E+05	1.1483E-01	9.9286E+01
7	5.00	6.7379E+04	1.2439E-01	6.0211E+01
8	8.00	3.3546E+03	7.2094E-03	3.1922E+01
9	12.00	6.1442E+01	.0000E+00	2.6493E+01
10	16.00	1.1254E+00	.0000E+00	1.7623E+01
11	17.00	4.1399E-01	.0000E+00	1.8736E+01
12	17.50	2.5110E-01	.0000E+00	2.1139E+01
13	18.00	1.5230E-01	.0000E+00	3.3059E+01
14	18.50	9.2374E-02	.0000E+00	5.8141E+01
15	19.00	5.6028E-02	.0000E+00	6.8442E+01
16	19.50	3.3983E-02	.0000E+00	5.3053E+01
17	20.00	2.0612E-02	.0000E+00	3.0782E+01
18	20.50	1.2502E-02	.0000E+00	1.4872E+01
19	21.00	7.5826E-03	.0000E+00	6.4232E+00
20	21.50	4.5995E-03	.0000E+00	2.5928E+00
21	22.00	2.7895E-03	.0000E+00	1.0053E+00
22	22.50	1.6919E-03	.0000E+00	3.8067E-01
23	23.00	1.0262E-03	.0000E+00	1.4216E-01

AGE TO INDIUM RESONANCE (1.46EV) IS	2.9947E+01	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	3.0004E+01	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	1.8123E+00	CM2
TOTAL MIGRATION AREA IS	3.1817E+01	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	2.5493E+03	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	3.2063E+03	M/SEC
ABSORPTION PARAMETER IS	5.8382E-01	
SCATTERING PER RESONANCE ATOM IS	7.6640E+01	BARNS
EFFECTIVE RESONANCE INTEGRAL IS	2.0765E+01	BARNS

Figure 15a: AFIT Subcrit., 2 Fast Groups, Chain 1

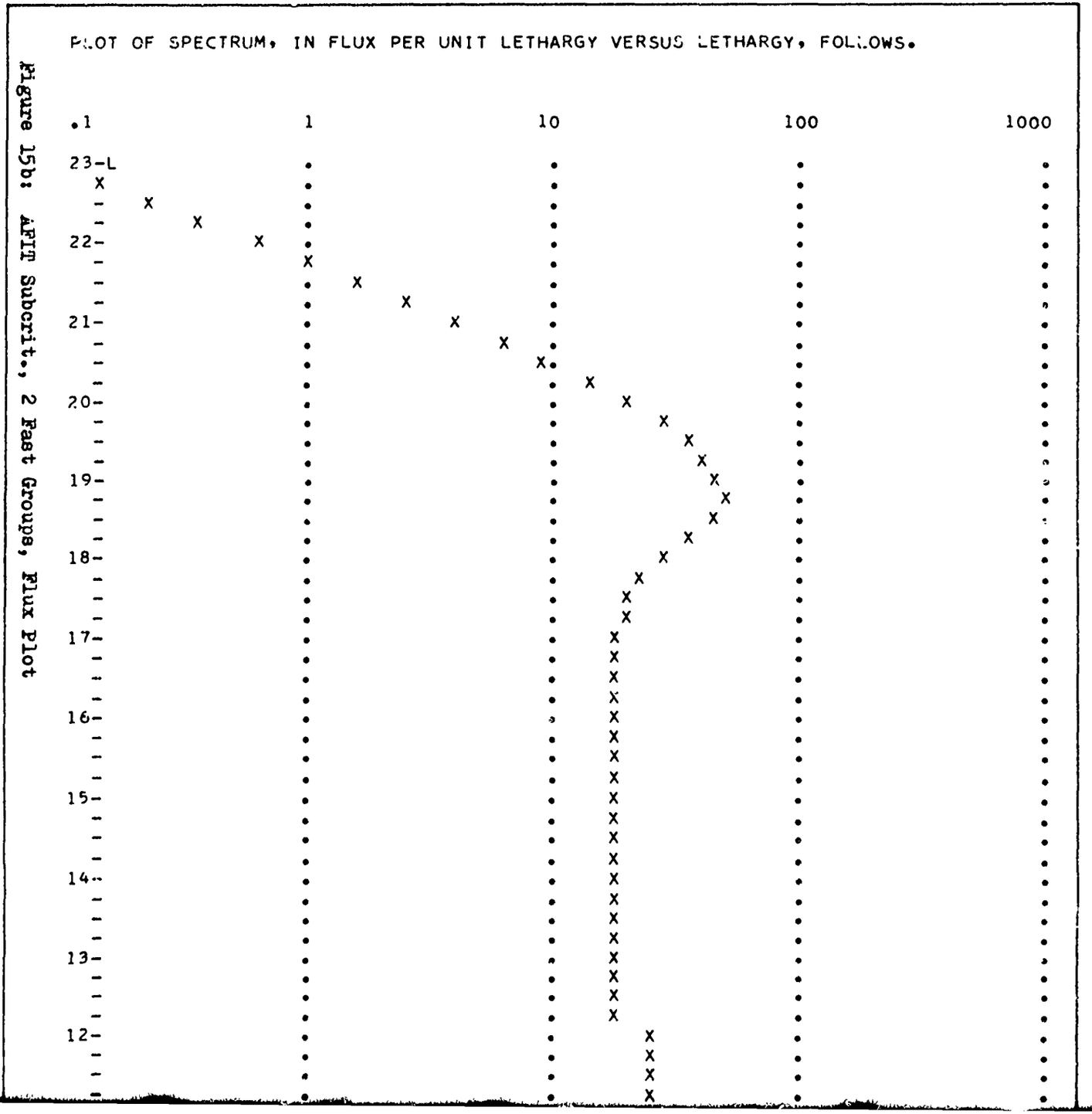


Figure 15b: AFIT Subcrit., 2 Fast Groups, Flux Plot

B.

JOB OLD BARNYARD CHAIN 2Z

THE TOTAL NUMBER OF BROAD GROUPS IS 3
OUTPUT WILL BE FOR 3 BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 6 11 12

NUCLIDE IS HYDROGEN

ITS NUMBER DENSITY IS 3.5040E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00000	.00000	1.79736	1.45938	1.79736
1	2	.00000	.00000	1.73146	.88732	1.73146
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	15.60306	10.54419	15.60306
2	3	.00000	.00000	.84521	.37127	.84521
3	3	.00000	.00000	38.00000	25.13320	38.00000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	1.18213	1.73147	.00000	.00000	.86840
2	5.53885	.85125	.00604	.00000	.13160
3	13.09431	.22751	.22751	.00000	

NUCLIDE IS OXYGEN

ITS NUMBER DENSITY IS 1.7520E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00168	.00000	2.56149	.41282	2.56317
1	2	.00030	.00000	.15174	.00346	.15204
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	3.94807	.15183	3.94807
2	3	.00000	.00000	.01653	-.00495	.01653
3	3	.00000	.00000	4.20000	.17514	4.20000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	2.31611	.16321	.01118	.00000	.86840
2	3.81772	.01653	.00000	.00000	.13160
3	4.02623	.00137	.00137	.00000	

Figure 15c: AFIT Subcrit., 2 Fast Groups, Chain 2

NUCLIDE IS ALUMINUM

ITS NUMBER DENSITY IS 6.4520E-03 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.12489	.00000	.00017	.00159	.12507
1	2	.00000	.00000	.00000	.00000	.00000
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	.00000	.00000	.00000
2	3	.00000	.00000	.00000	.00000	.00000
3	3	.00000	.00000	1.40000	.03444	1.40000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	.12679	.00331	.00331	.00000	.86840
2	.01046	.01046	.01046	.00000	.13160
3	1.53071	.16515	.16515	.00000	

NUCLIDE IS URANIUM 235

ITS NUMBER DENSITY IS 1.2250E-04 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.69454	.00000	4.28541	2.16289	4.97995
1	2	.86675	.00000	.01191	-.04141	.89866
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.24984	.00000	10.03586	.47857	10.28571
2	3	.00000	.00000	.00457	-.00136	.00457
3	3	.00000	.00000	14.80000	.04144	14.80000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	5.02510	2.16663	1.26798	3.47598	.86840
2	32.86496	23.05647	23.05189	36.82483	.13160
3	479.51266	464.75410	464.75410	950.47764	

Figure 15d: AFIT Subcrit., 2 Fast Groups, Chain 2, Continued

NUCLIDE IS URANIUM 238

ITS NUMBER DENSITY IS 1.7380E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.92311	.01087	5.04366	2.33414	5.98852
1	2	1.05061	.00289	.01486	.01718	1.07124
1	3	.00000	.00000	.00000	.00000	.00000
2	2	.15256	.00000	17.74682	.59531	17.89938
2	3	.00000	.00000	.00262	-.00079	.00262
3	3	.00000	.00000	13.80000	.03864	13.80000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	5.06187	1.42467	.36719	.81584	.86840
2	18.34789	1.04302	1.04040	.00000	.13160
3	15.63216	1.87080	1.87080	.00000	

MAXWELL-BOLTZMAN FACTOR = 1.128, AVERAGE X = 1.4592663E+00

Figure 15e: AFIT Subcrit., 2 Fast Groups, Chain 2, Continued

JOB OLD BARNYARD, CHAIN 1

ROMASKA CORE (A DIRECT ENERGY CONVERTER REACTOR) 1 FAST GROUP

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 1273.0 KELVIN

ATOM FRACTIONS ARE

CARBON	.86548694
URANIUM 235	.12213834
URANIUM 238	.01361218

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	4.4849E+00
2	1.00	3.6788E+06	1.0824E-01	2.3895E+01
3	1.50	2.2313E+06	2.1044E-01	5.8781E+01
4	2.00	1.3534E+06	2.3139E-01	8.9394E+01
5	2.50	8.2035E+05	1.8048E-01	9.1312E+01
6	3.00	4.9787E+05	1.1483E-01	9.0867E+01
7	5.00	6.7379E+04	1.2439E-01	1.0000E+02
8	8.00	3.3546E+03	7.2094E-03	3.4544E+01
9	12.00	6.1442E+01	.0000E+00	6.9175E-02
10	16.00	1.1254E+00	.0000E+00	9.7373E-07
11	17.00	4.1399E-01	.0000E+00	1.0213E-11
12	17.50	2.5110E-01	.0000E+00	4.7077E-14
13	18.00	1.5230E-01	.0000E+00	3.0930E-16
14	18.50	9.2374E-02	.0000E+00	2.9839E-18
15	19.00	5.6028E-02	.0000E+00	4.2305E-20
16	19.50	3.3983E-02	.0000E+00	8.6734E-22
17	20.00	2.0612E-02	.0000E+00	2.5074E-23
18	20.50	1.2502E-02	.0000E+00	9.9264E-25
19	21.00	7.5826E-03	.0000E+00	5.2209E-26
20	21.50	4.5991E-03	.0000E+00	3.5424E-27
21	22.00	2.7895E-03	.0000E+00	3.0158E-28
22	22.50	1.6919E-03	.0000E+00	3.1401E-29
23	23.00	1.0262E-03	.0000E+00	3.9066E-30

AGE TO INDIUM RESONANCE (1.46EV) IS	9.6172E+01	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	9.5923E+01	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	1.7292E-01	CM2
TOTAL MIGRATION AREA IS	9.6096E+01	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	.0000E+00	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	2.1557E+04	M/SEC
ABSORPTION PARAMETER IS	2.7659E+02	
SCATTERING PER RESONANCE ATOM IS	4.3363E+02	BARNS
EFFECTIVE RESONANCE INTEGRAL IS	4.6972E+01	BARNS

Figure 16a: Romaska, 1 Fast Group, Chain 1

A.

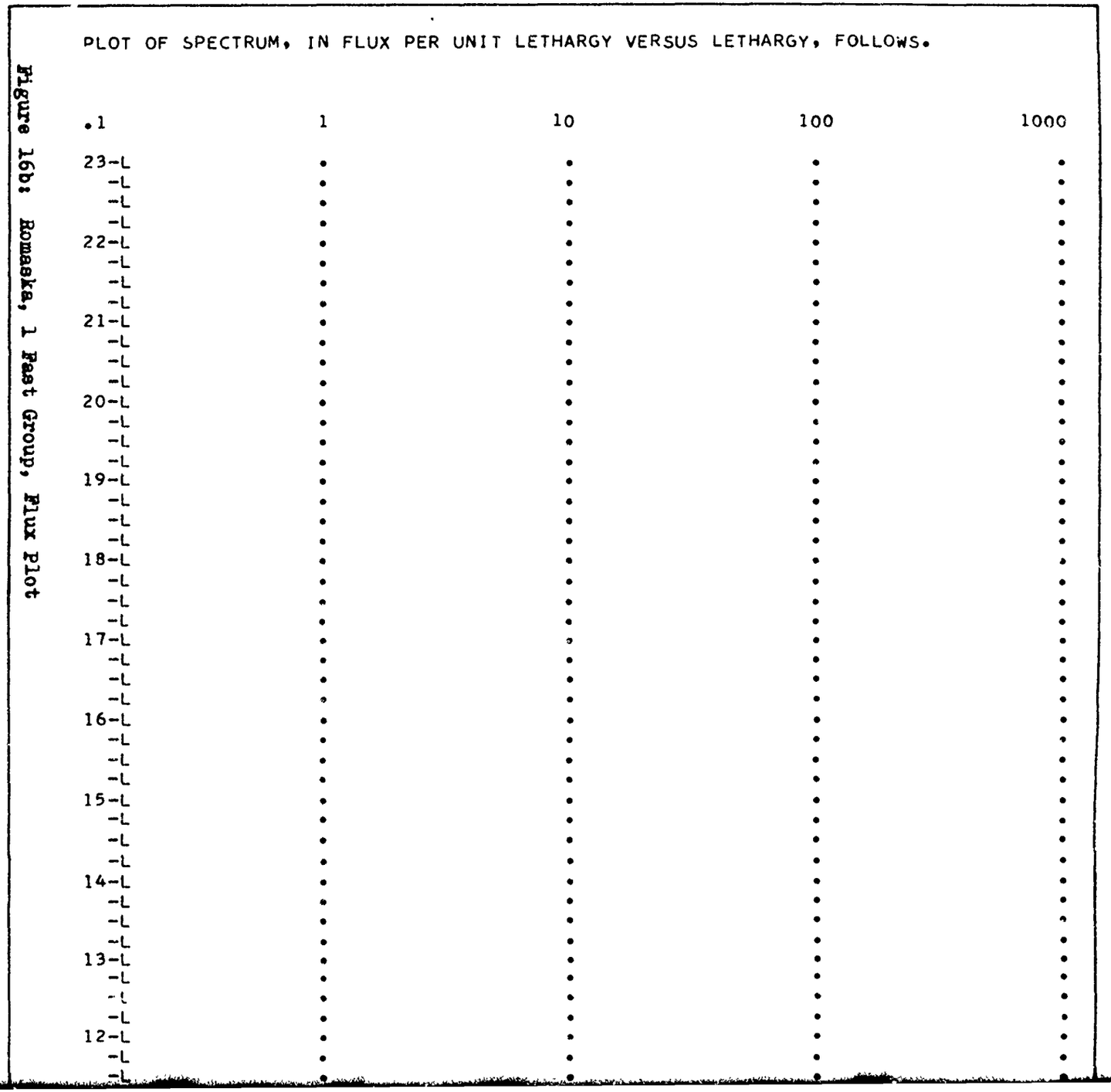


Figure 16b: Romeska, 1 Fast Group, Flux Plot

B.

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     JOB  OLD BARNYARD CHAIN 2Z

THE TOTAL NUMBER OF BROAD GROUPS IS  1
OUTPUT WILL BE FOR  1  BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 11 12


NUCLIDE IS CARBON

    ITS NUMBER DENSITY IS 6.9940E-02 PER BARN-CM


MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER                                          ELASTIC       ELASTIC       TOTAL
FROM   TO       INELASTIC          N-2N           (P0)          (P1)       TRANSFER

  1     1        .00211          .00000        3.51316        .28160        3.51527
  1     2        .00000          .00000         .00000         .00000         .00000

GROUP      SIGTR          SIGR          SIGA       NUSIGF       SOURCE
   1      3.23367        .00000        .00000        .00000      1.00000


NUCLIDE IS URANIUM 235

    ITS NUMBER DENSITY IS 9.8700E-03 PER BARN-CM


MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER                                          ELASTIC       ELASTIC       TOTAL
FROM   TO       INELASTIC          N-2N           (P0)          (P1)       TRANSFER

  1     1        .94420          .00000        6.76122       1.42560       7.70542
  1     2        .00000          .00000         .00000         .00000         .00000

GROUP      SIGTR          SIGR          SIGA       NUSIGF       SOURCE
   1      7.99061       1.71079       1.71079       4.01466      1.00000
```

Figure 16c: Romaska, 1 Fast Group, Chain 2

NUCLIDE IS URANIUM 238

ITS NUMBER DENSITY IS 1.1000E-03 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.94824	.00285	8.09556	1.67050	9.04950
1	2	.00000	.00000	.00000	.00000	.00000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	7.57219	.19318	.19603	.26603	1.00000

Figure 16d: Romaska, 1 Fast Group, Chain 2, Continued

JOB OLD BARNYARD, CHAIN 1

ROMASKA CORE (A DIREC ENERGY CONVERTER REACTOR) 3 FAST GROUP

THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).

SYSTEM TEMPERATURE IS 1273.0 KELVIN

ATOM FRACTIONS ARE

CARBON	.86548694
URANIUM 235	.12213834
URANIUM 238	.01361218

GROUP	LETHARGY	ENERGY, EV	SOURCE	RELATIVE GROUP FLUX, /UNIT U
1	.50	6.0653E+06	2.3023E-02	4.4849E+00
2	1.00	3.6788E+06	1.0824E-01	2.3895E+01
3	1.50	2.2313E+06	2.1044E-01	5.8781E+01
4	2.00	1.3534E+06	2.3139E-01	8.9394E+01
5	2.50	8.2085E+05	1.8048E-01	9.1312E+01
6	3.00	4.9787E+05	1.1483E-01	9.0867E+01
7	5.00	6.7379E+04	1.2439E-01	1.0000E+02
8	8.00	3.3546E+03	7.2094E-03	3.4544E+01
9	12.00	6.1442E+01	.0000E+00	6.9175E-02
10	15.00	1.1254E+00	.0000E+00	9.7373E-07
11	17.00	4.1399E-01	.0000E+00	1.0213E-11
12	17.50	2.5110E-01	.0000E+00	4.7077E-14
13	18.00	1.5230E-01	.0000E+00	3.0930E-16
14	18.50	9.2374E-02	.0000E+00	2.9839E-18
15	19.00	5.6028E-02	.0000E+00	4.2305E-20
16	19.50	3.3983E-02	.0000E+00	8.6734E-22
17	20.00	2.0612E-02	.0000E+00	2.5074E-23
18	20.50	1.2502E-02	.0000E+00	9.9264E-25
19	21.00	7.5826E-03	.0000E+00	5.2209E-26
20	21.50	4.5991E-03	.0000E+00	3.5424E-27
21	22.00	2.7895E-03	.0000E+00	3.0158E-28
22	22.50	1.6919E-03	.0000E+00	3.1401E-29
23	23.00	1.0262E-03	.0000E+00	3.9066E-30

AGE TO INDIUM RESONANCE (1.46EV) IS	9.6172E+01	CM2
AGE TO ARBITRARY THERMAL (1.12EV) IS	9.5923E+01	CM2
THERMAL DIFFUSION LENGTH SQUARED IS	1.7292E-01	CM2
TOTAL MIGRATION AREA IS	9.6096E+01	CM2
MOST PROBABLE THERMAL NEUTRON VELOCITY IS	.0000E+00	M/SEC
AVERAGE THERMAL NEUTRON VELOCITY IS	2.1557E+04	M/SEC
ABSORPTION PARAMETER IS	2.7659E+02	
SCATTERING PER RESONANCE ATOM IS	4.3363E+02	BARNS
EFFECTIVE RESONANCE INTEGRAL IS	4.6972E+01	BARNS

Figure 17a: Romaska, 3 Fast Group, Chain 1

PLOT OF SPECTRUM, IN FLUX PER UNIT LETHARGY VERSUS LETHARGY, FOLLOWS.

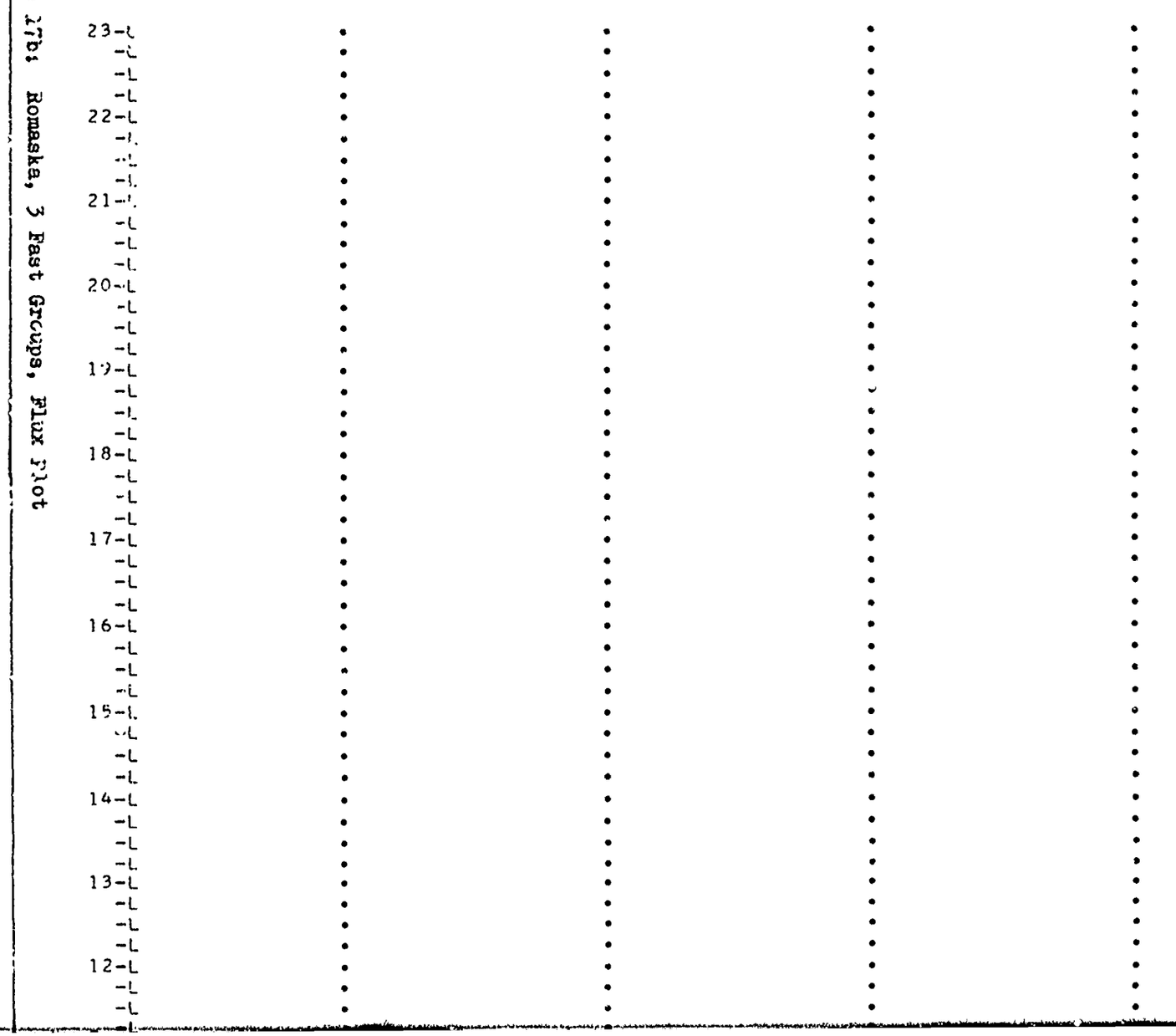

Figure 17b: Romaska, 3 Fast Groups, Flux Plot

A.

B.

JOB OLD BARNYARD CHAIN 2Z

THE TOTAL NUMBER OF BROAD GROUPS IS 3
OUTPUT WILL BE FOR 3 BROAD GROUPS
THE BOUNDARY FINE GROUPS ARE 3 6 11 12

NUCLIDE IS CARBON

ITS NUMBER DENSITY IS 6.9940E-02 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.00808	.00000	1.30261	.28939	1.31069
1	2	.01226	.00000	.41154	-.1`988	.42379
1	3	.00306	.00000	.00000	.00000	.00306
1	4	.00000	.00000	.00000	.00000	.00000
2	2	.00000	.00000	2.24744	.38214	2.24744
2	3	.00000	.00000	.31718	-.08037	.31718
2	4	.00000	.00000	.00000	.00000	.00000
3	3	.00000	.00000	4.19494	.29727	4.19494
3	4	.00000	.00000	.00000	.00000	.00000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	1.62804	.42686	.00000	.00000	.34170
2	2.26286	.31718	.00000	.00000	.52670
3	3.89767	.00000	.00000	.00000	.13160

NUCLIDE IS URANIUM 235

ITS NUMBER DENSITY IS 9.8700E-03 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.05023	.00000	4.75120	4.02975	4.80143
1	2	1.16352	.00000	.02243	-.66161	1.18596
1	3	.71014	.00000	.00000	.00000	.71014
1	4	.00000	.00000	.00000	.00000	.00000
2	2	.45118	.00000	4.07086	1.57852	4.52204
2	3	.97199	.00000	.01834	-.06391	.99034
2	4	.00000	.00000	.00000	.00000	.00000
3	3	.58971	.00000	8.24010	1.10727	8.82980
3	4	.00000	.00000	.00000	.00000	.00000

Figure 17c: Romaska, 3 Fast Groups, Chain 2

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	4.61893	3.18565	1.28955	3.89875	.34170
2	5.25574	2.24831	1.25797	3.26821	.52670
3	9.69603	1.97351	1.97351	4.36480	.13160

NUCLIDE IS URANIUM 238

ITS NUMBER DENSITY IS 1.1000E-03 PER BARN-CM

MICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.

SCATTER FROM	TO	INELASTIC	N-2N	ELASTIC (P0)	ELASTIC (P1)	TOTAL TRANSFER
1	1	.06161	.00339	4.52201	3.86047	4.59039
1	2	1.54761	.02155	.02559	-.69836	1.61630
1	3	.96850	.00665	.00000	.00000	.98180
1	4	.00000	.00000	.00000	.00000	.00000
2	2	.61045	.00000	5.29216	1.94066	5.90260
2	3	1.09910	.00000	.02288	.02651	1.12198
2	4	.00000	.00000	.00000	.00000	.00000
3	3	.37443	.00000	9.84668	1.32406	10.22111
3	4	.00000	.00000	.00000	.00000	.00000

GROUP	SIGTR	SIGR	SIGA	NUSIGF	SOURCE
1	4.60208	3.17380	.60728	1.71666	.34170
2	5.31470	1.37928	.25730	.39585	.52670
3	9.30674	.10969	.10969	.00000	.13160

Figure 17d: Romaska, 3 Fast Groups, Chain 2, Continued

V. The Calculation of Cross Sections and Related Constants

In this section the equations and mathematical expressions used for the important calculations in the code will be developed. After a brief discussion of flux weighting, the theoretical methods for predicting the energy dependent flux will be described. This theory consists of two major subdivisions, that for epithermal, or above thermal, energies, and that for thermal energies. Finally, the methods used to calculate group cross sections, both epithermal and thermal, will be presented.

Flux Weighting

The microscopic neutron cross section, represented by the symbol σ, can be defined as the probability, measured in cross sectional area (barns), that a neutron will interact with a nucleus in a specified manner. The macroscopic cross section Σ, is defined as

$$\Sigma = N\sigma \tag{1}$$

where N is the number of nuclei per cm^3. Thus, the macroscopic cross section is a measure of the probability that a neutron will interact with a nucleus per centimeter of path. If a material of macroscopic cross section Σ is also populated by a neutron density n ($_0n^1/cm^3$) all with constant velocity v (cm/sec), then the reaction rate R (events/cm^3 - sec) is

$$R = \Sigma nv \tag{2}$$

The product nv in equation (2) is defined as the neutron flux, neutrons per cm^2 per sec.

However, the neutron cross section is not a constant; rather, it

is a function of energy. Therefore, the reaction rate in the presence of polyenergetic neutrons becomes

$$R = \int_0^\infty \Sigma(E)\varphi(E)dE \tag{3}$$

where $\varphi(E)$ is the energy dependent neutron flux, $n(E)v(E)$, and where E is the energy. The total flux is, by definition

$$\varphi = \int_0^\infty \varphi(E)dE \tag{4}$$

In terms of this total flux, φ, the reaction rate can be written

$$R = \bar{\Sigma}\varphi \tag{5}$$

where $\bar{\Sigma}$ is some average macroscopic cross section over the energy range of interest. It follows, by equating the right hand sides of equations (3) and (5), that this average cross section must be given by

$$\bar{\Sigma} = \frac{\int_0^\infty \Sigma(E)\varphi(E)dE}{\int_0^\infty \varphi(E)dE} \tag{6}$$

or, in other words, the average cross section is a flux weighted average of the energy dependent cross section. Of course a similar expression to Eq (6) can be written in terms of the microscopic cross section, i.e.,

$$\bar{\sigma} = \frac{\int_0^\infty \sigma(E)\varphi(E)dE}{\int_0^\infty \varphi(E)dE} \tag{7}$$

Equation 6 can be generalized to any energy range (limits on the integral), say E_1 to E_2 in order to produce an average cross section applicable to that energy range. Such averages are called energy group cross sections, and may be expressed as

$$\Sigma_n = \frac{\int_{E_n^-}^{E_n^+} \Sigma(E)\varphi(E)dE}{\int_{E_n^-}^{E_n^+} \varphi(E)dE} \qquad (8)$$

where Σ_n is the group cross section of the nth energy group

E_n^- is the lower energy boundary of the nth group

E_n^+ is the upper energy boundary of the nth group.

Thus, in order to determine group cross sections the variation of neutron flux with energy must be known. The calculation necessary to determine the energy dependent flux far exceeds the calculation necessary to determine group cross sections from equation (8) above. In this code all of Chain I is devoted to the calculation of energy dependent flux. The shorter Chain II utilizes this flux to calculate group cross sections.

The flux calculations performed in Chain I are in solution to the energy dependent, Boltzmann transport equation under very limiting assumptions and highly idealized boundary conditions. These assumptions and limitations are imposed for the sole purpose of obtaining a usable solution. To attempt a solution for energy dependent flux without such limitations would involve (for our purposes) a prohibitive amount of calculation. In addition the Boltzmann equation is to be applied, in this code, to two very different situations, the above thermal energy region where target nuclei kinetic energies or velocities are negligible with respect to the neutron velocities, and the thermal region where the velocities of the target nuclei are not negligible with respect to the neutron velocities. These two situations

require different assumptions and boundary conditions and hence different solutions to the Boltzmann equation. *As it will be seen the above thermal, energy dependent flux is determined by the method of moments while the thermal flux is determined by the method of Wigner and Wilkins.*

In either case we start with the steady state Boltzman.. equation written for an infinite, non-multiplying homogeneous medium* which scatters and absorbs neutrons.

$$\mu \frac{\partial}{\partial x} \varphi(x,u,\mu) + \Sigma_t(u)\varphi(x,u,\mu) = S(x,u,\mu)$$

$$+ \int d\Omega' \int_0^u du' \, \Sigma_s(u')\varphi(x,u',\mu')f(u' \rightarrow u;\mu_o) \qquad (9)$$

where x is spatial position of the neutrons,

u is the neutron's lethargy or logarithm of the energy, (Ref 7:146),

μ or μ' is the cosine of the scalar angle, cosine θ, between the neutron's direction and the x axis,

μ_o is the cosine of the scalar angle, cosine θ, through which a neutron is scattered,

$\varphi(x,u,\mu)$ is neutron flux in n^1/cm^2 - sec-steradian-unit lethargy

$\Sigma_t(u)$ is the lethargy dependent macroscopic total cross section, cm^{-1}

$\Sigma_s(u)$ is the lethargy dependent macroscopic scattering cross section, cm^{-1},

$d\Omega'$ is the differential solid angle = $\sin\theta \, d\theta \, d\phi = -d(\cos\theta)d\phi = -d\mu d\theta$

$f(u' \rightarrow u;\mu_o)$ is the scattering frequency function or probability

---

*The cross sections to be generated by this code will frequently be applied to media which are either multiplying, hetrogeneous or both and always to media which are finite. Without suitable corrections this can obviously lead to error.

that a neutron with initial lethargy u′ and direction μ' will, when scattered, emerge within a unit lethargy interval about u and within a unit solid angle about μ,

$S(x,u,\mu)$ is the source of neutrons per unit volume, per unit lethargy, per unit solid angle.

The scattering frequency function and the macroscopic scattering cross section under the integral in equation (9) are frequently taken as a product

$$\Sigma_s(u', \mu_o) \equiv \Sigma_s(u')\ f(u' \rightarrow u, \mu_o) \tag{10}$$

where $\Sigma_s(u',\mu_o)$ *is defined as the macroscopic* <u>*differential*</u> *scattering cross section in* cm^{-1}*/steradian.* The Boltzmann equation then becomes,

$$\mu \frac{\partial \varphi}{\partial x}(x,u,\mu) + \Sigma_t(u)\varphi(x,u,\mu) = S(x,u,\mu)$$

$$+ \int d\Omega' \int_o^u \Sigma_s(u', \mu_o)\varphi(x,u',\mu')du' \tag{11}$$

The two directional coordinates, μ, μ' and the cosine scalar angle between them, μ_o, are shown in Figure 18. The cosine angle μ_o is related to μ and μ' by the law of cosines for a spherical triangle.

<u>Fast Flux Spectrum and Age</u>

In the energy region where target nuclei motion may be neglected (the epithermal, above-thermal or fast region) the Boltzmann equation can be most easily solved by assuming that the source is isotropic and consists of a plane of infinite area located at the coordinate position x = 0, i.e.,

$$S(x,u,\mu) = \frac{S(u)}{4\pi}\delta(x) \tag{12}$$

where $\delta(x)$ is the Dirac delta function at x = 0.

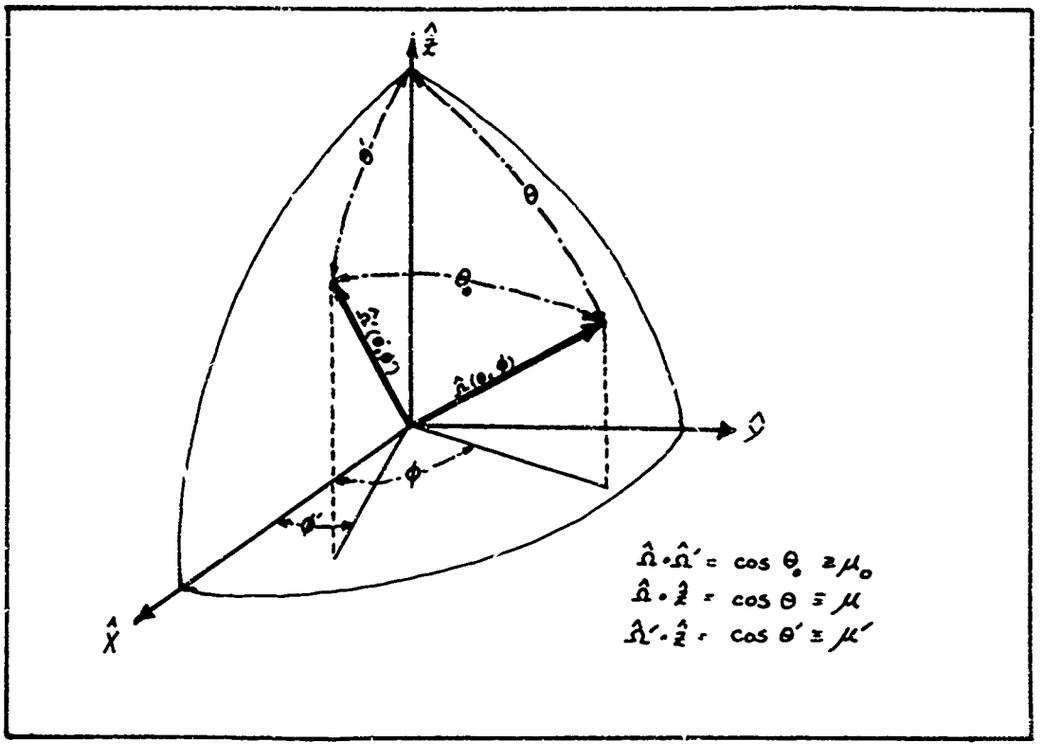

Figure 18: Neutron Scattering Diagram

With this assumption the method of moments may be applied to find the fast flux as a function of energy. However, as it will be seen, the minimum calculation necessary for the flux simultaneously produces the neutron age as a function of energy. Since this parameter is of some importance in elementary calculations it is also tabulated by the code.

The appearance of the cosine of the scalar scattering angle u, in the Boltzmann equation, equation (9), immediately suggests the use of Legendre polynomials. The Legendre polynomials are

$$P_0(u) = 1$$

$$P_1(u) = u$$

$$P_2(u) = \tfrac{1}{2}(3u^2 - 1)$$

and in general,

$$P_n(u) = \frac{1}{2^n} \sum_{j=0}^{m} \frac{(-1)^j (2n-j)!}{j!(n-2j)!(n-j)!} u^{n-2j} \tag{13}$$

where $m = \frac{n}{2}$ if n is even or zero or $m = \frac{1}{2}(n-1)$ if n is odd.

These polynomials obey the orthonality condition:

$$\int_{-1}^{+1} P_n(u) P_m(u) du = \frac{2}{2n+1} \delta_m^n \tag{14}$$

where $\delta_m^n \equiv$ Kronecker Delta $= \begin{cases} 0, & m \neq n \\ 1, & m = n \end{cases}$

Other properties which will be useful are (Ref 12:115)

$$\int_0^{2\pi} P_n(u_o) d\phi = \int_0^{2\pi} P_n(u) P_n(u') d\phi \tag{15}$$

where ϕ is the plane angle shown in Figure 18, and

$$uP_n(u) = \frac{n+1}{2n+1} P_{n+1}(u) + \frac{n}{2n+1} P_{n-1}(u) \tag{16}$$

For further information on Legendre polynomials, the reader is referred to Churchill (Ref 3:200-203).

The flux in equation (11) can be expanded in terms of these Legendre polynomials as

$$\varphi(x,u,u) = \sum_{n=0}^{\infty} \frac{2n+1}{4\pi} \varphi_n(x,u) P_n(u) \tag{17}$$

Similarily, the scattering term may be expressed as

$$\Sigma_s(u,u_o) = \sum_{m=0}^{\infty} \frac{2m+1}{4\pi} \Sigma_s^m(u') P_m(u_o) \tag{18}$$

Note that both the indexes m and n are required since the integral term in equation (11) involves both the flux and scattering terms.

By using the orthoganality condition, equation (14), it is also easy to show that $\varphi_0(x,u)$ is the all angle flux and $\varphi_1(x,u)$ is the net neutron current of diffusion theory. Similarly applying the orthogonality condition to the scattering term results in

$$\Sigma_s^0(u') = \Sigma_s(u') \tag{19}$$

$$\Sigma_s^1(u') = \overline{u}\,\Sigma_s(u') \tag{20}$$

where $\Sigma_s(u')$ is all-angle total scattering cross section at lethargy u', and $\overline{u}$ is the average cosine of the angle of scatter in the laboratory system.

The substitution of the source given by equation (12) and of the expansions (17) and (18) into equation (11) yields

$$\sum_{n=0}^{\infty} \left\{ \frac{2n+1}{4\pi} \frac{\partial \varphi_n(x,u)}{\partial x} \mu P_n(\mu) + \frac{2n+1}{4\pi} \Sigma_t(u)\, \varphi_n(x,u)\, P_n(\mu) = \frac{S(u)}{4\pi}\, \delta(x) \right.$$

$$\left. + \int d\Omega' \int_0^u du' \sum_{m=0}^{\infty} \left[\frac{2m+1}{4\pi} \Sigma_s^m(u') P_m(\mu_0) \right] \left[\frac{2n+1}{4\pi} \varphi_n(x,u') P_n(\mu') \right] \right\} \tag{21}$$

If the recurrence relation (16) is used in the first term of (21) and if the integral relationship (15) as well as the orthogonality property (14) are applied to the integral term of (21), we obtain

$$\sum_{n=0}^{\infty} \left\{ \frac{\partial \varphi_n(x,u)}{\partial x} \left[(n+1)\, P_{n+1}(\mu) + n\, P_{n-1}(\mu) \right] + \right.$$

$$(2n+1)\, \Sigma_t(u) \varphi_n(x,u)\, P_n(u) = S(u)\, \delta(x) +$$

$$\left. (2n+1)\, P_n(u) \int_0^u du'\, \Sigma_s^n(u')\, \varphi_n(x,u') \right\} \tag{22}$$

Equation (22), a single equation with an infinite number of terms, may

be transformed into an infinite number of coupled equations, each with a finite number of terms by employing the orthogonality condition given by equation (14). This transformation is carried out by operating on equation (22), term by term, with

$$\int_{-1}^{+1} P_\ell(u)\langle \text{eq.}(22)\rangle \, du, \qquad \ell = 0,1,2,\ldots. \tag{23}$$

The results are:

$\ell = 0$:

$$\frac{\partial \varphi_1(x,u)}{\partial x} + \Sigma_t \varphi_0(x,u) = \int_0^u \Sigma_s^0(u') \varphi_0(x,u')du' + S_0(u)\,\delta(x)$$

$\ell = 1$:

$$\frac{2}{3}\frac{\partial \varphi_2(x,u)}{\partial x} + \frac{1}{3}\frac{\partial \varphi_0(x,u)}{\partial x} + \Sigma_t \varphi_1(x,u) = \int_0^u \Sigma_s^1(u') \varphi_1(x,u')du'$$

$\ell = 2$:

$$\frac{3}{5}\frac{\partial \varphi_3(x,u)}{\partial x} + \frac{2}{5}\frac{\partial \varphi_1(x,u)}{\partial x} + \Sigma_t \varphi_2(x,u) = \int_0^u \Sigma_s^2(u') \varphi_2(x,u')du'$$

$$\vdots \qquad\qquad \vdots \qquad\qquad \vdots \qquad\qquad \vdots \tag{24}$$

If equations (24) are truncated by assuming that $\varphi_{\ell+1}(x,u)$, $\varphi_{\ell+2}(x,u)$, $\ldots$, $\varphi_\infty(x,u)$ are zero, then a solution of the remaining finite set is possible and the result is known as the P_ℓ approximation. The index ℓ may have any value, e.g. P_5. The flux in such an approximation would be found by summing

$$\varphi(x,u,u) = \sum_{\ell=0}^{5} \frac{2n+1}{4\pi} \varphi_\ell(x,u)\, P_\ell(u) \tag{25}$$

However in this case we are not so much interested in the flux,

$\varphi(x,u,u)$, as we are in the volume, angle integral of the flux, i.e.,

$$\varphi^{(o)}(u) = \int_{-\infty}^{+\infty} dx \int d\Omega\, \varphi(x,u,u) \tag{26}$$

which is the basic energy dependence of the flux, which we seek. Equation (26) is merely the Zero moment of the spatial distribution of the total (all-angle) flux. The higher moments would be

$$\varphi^{(1)}(u) = \int_{-\infty}^{+\infty} dx \int d\Omega\, x\, \varphi(x,u,u) \Big/ \int_{-\infty}^{+\infty} dx \int d\Omega\, \varphi(x,u,u)$$

$$\varphi^{(2)}(u) = \int_{-\infty}^{+\infty} dx \int d\Omega\, x^2\, \varphi(x,u,u) \Big/ \int_{-\infty}^{+\infty} dx \int d\Omega\, \varphi(x,u,u) \tag{27}$$

$$\vdots \qquad\qquad \vdots \qquad\qquad \vdots$$

For the plane source which we have assumed, it is obvious that the odd moments $\varphi^{(1)}$, $\varphi^{(3)}$, ... must be zero in value.

Although we cannot calculate $\varphi(x,u,u)$ exactly (a P_∞ calculation), we can calculate the moments exactly using some properties of the Fourier integral transform on the spatial variable, i.e., (Ref 18)

$$\mathcal{F}\{\varphi_\ell(x,u)\} = \int_{-\infty}^{+\infty} \varphi_\ell(x,u)\, e^{-ipx}\, dx \equiv \vartheta_\ell(p,u) \tag{28}$$

It can be seen that if equation (28) is written for $\ell = 0$ and the special case $p = 0$, we have exactly the zero moment of equation (26), i.e.,

$$\varphi^{\circ}(u) = \left[\, \vartheta_o(p,u) \,\right]_{p=0} \tag{29}$$

(Recall that $\varphi_o(x,u) = \int \varphi(x,u,\Omega)\, d\Omega$.)

Thus we Fourier transform equations (24) using the properties (Ref 18)

$$\mathcal{F}\left\{\frac{\partial \varphi_\ell(x,u)}{\partial x}\right\} = ip\,\theta(p,u) \tag{30}$$

and

$$\mathcal{F}\{\delta(x)\} = 1$$

to obtain

$$ip\,\theta_1(p,u) + \Sigma_t(u)\,\theta_0(p,u) = S(u) + \int_0^u \Sigma_t(u')\theta_0(p,u')du' \tag{31}$$

$$\frac{1}{3}ip\,\theta_0(p,u) + \frac{2}{3}ip\,\theta_2(p,u) + \Sigma_t(u)\,\theta_1(p,u) = \int_0^u \Sigma_s^1(u')\,\theta_1(p,u')\,du' \tag{32}$$

$$\frac{2}{5}ip\,\theta_1(p,u) + \frac{3}{5}ip\,\theta_3(p,u) + \Sigma_t(u)\,\theta_2(p,u) = \int_0^u \Sigma_s^2(u')\,\theta_2(p,u')du' \tag{33}$$

. . .

We note that the transformed P_n equations are valid for any arbitrary p. We are, however, interested in the specific point p = 0. This suggests the use of the Maclaurin expansion. The Maclaurin expansions for $\theta_0(p,u)$ and $\theta_1(p,u)$ are chosen as

$$\theta_0(p,u) = \theta_0^0(u) + (-ip)\,\theta_0^1(u) + \frac{(-ip)^2}{2!}\theta_0^2(u) + \ldots. \tag{34}$$

$$\theta_1(p,u) = \theta_1^0(u) + (-ip)\,\theta_1^1(u) + \frac{(-ip)^2}{2!}\theta_1^2(u) + \ldots. \tag{35}$$

where the argument of the Maclaurin expansion is (-ip) rather than p

for reasons which will become apparent in the discussion on age below. The primes denoting differentiation in the expa n have been replaced by supercripts. That is

$$\theta_o''(u) \equiv \theta_o^2(u) \tag{36}$$

Under this notation we note that the zero moment is symbolized

$$\Theta^\circ = \theta_o^\circ(u) \tag{37}$$

Davison and Sykes (Ref 5:343) show that the n-th moment of the neutron flux can only involve spherical harmonics of order n or less. Further, they show that due to the odd-even nature of the functions involved

$$\int_{-\infty}^{\infty} \chi^n \phi_\ell(x,u)dx \neq 0 \tag{38}$$

only when both n and ℓ are both even or both odd. Therefore, in the equations for the second moment of the flux (to be derived) only θ_o°, θ_o^2, ... and θ_1^1, θ_1^3, ... will be non zero. Thus the Maclaurin expansions become

$$\theta_o(p,u) = \theta_o^\circ(u) + \frac{(-ip)^2}{2}\theta_o^2(u) + \ldots \tag{39}$$

$$\theta_1(p,u) = (-ip)\,\theta_1^1(u) + \frac{(-ip)^3}{3}\theta_1^3(u) + \ldots \tag{40}$$

By substituting equations (39) and (40) into the transformed P_n equations, equations (31), (32) and (33) and equating like powers of -ip, we finally obtain the moments equations which are:

$$(-ip)^\circ \left\{ \Sigma_t(u)\,\theta_o^\circ(u) = S(u) + \int_o^u \Sigma_s^\circ(u')\,\theta_o^\circ(u')\,du' \right\} \tag{41}$$

$$(-ip)^1 \left\{ \Sigma_t(u)\,\theta_1^1(u) = \frac{\theta_o^\circ(u)}{3} + \int_o^u \Sigma_s^1(u')\,\theta_1^1(u')\,du' \right\} \tag{42}$$

$$(-ip)^2 \left\{ \Sigma_t(u)\, \theta_0^2(u) = 2\, \theta_1^1(u) + \int_0^u \Sigma_s^0(u')\, \theta_0^2(u')\, du' \right\} \tag{43}$$

$$(-ip)^3 \left\{ \Sigma_t(u)\, \theta_1^3(u) = \theta_0^2(u) + 2\, \theta_2^2(u) + \int_0^u \Sigma_s^1(u')\, \theta_1^3(u')\, du' \right\} \tag{44}$$

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It must be emphasized that these moments equations are rigorous. The spherical harmonic expansion was quite general and not terminated at some finite ℓ.

Equation (41) alone is sufficient to calculate the zero moment, $\theta_0^0(u)$ (which, again, is the volume angle integral of the energy dependent flux); equations (41 and (42) alone are sufficient to calculate $\theta_1^1(u)$; and equations (41), (42) and (43) alone are sufficient to calculate $\theta_0^2(u)$. This permits a direct determination of neutron age. Neutron age in a slab geometry is defined as one half of the second spatial moment of the flux,

$$\tau = \tfrac{1}{2}\Theta^{(2)} \tag{45}$$

But the second spatial moment is given by equation (27) which can also be expressed as

$$\Theta^{(2)}(u) = \frac{\int_{-\infty}^{+\infty} dx \int d\Omega\ x^2\, \varphi(x,u,\mu)}{\int_{-\infty}^{+\infty} dx \int d\Omega\, \varphi(x,u,\mu)} = \frac{\int_{-\infty}^{+\infty} x^2\, \varphi_0(x,u)\, dx}{\int_{-\infty}^{+\infty} \varphi_0(x,u)\, dx} =$$

$$\frac{\mathcal{F}\{ x^2\, \varphi_0(x,u) \}_{p=0}}{\mathcal{F}\{ \varphi_0(x,u) \}_{p=0}} \tag{46}$$

Since

$$\mathcal{F}\{x^q\, \vartheta_\ell(x,u)\} = i^q \frac{\partial^q \theta_\ell(p,u)}{\partial p^q} \tag{47}$$

(another Fourier transform property, Ref 18), it follows that

$$\tau = \tfrac{1}{2}\Theta^{(2)} = \tfrac{1}{2}\frac{i^2\left[\dfrac{\partial^2 \theta_0(p,u)}{\partial p^2}\right]_{p=0}}{[\theta_0(p,u)]_{p=0}} = \tfrac{1}{2}\frac{i^2[(-i)^2\,\theta_0^2(u)]}{[\theta_0^0(u)]} = \tfrac{1}{2}\frac{\theta_0^2(u)}{\theta_0^0(u)}$$

So that a simultaneous solution of (41), (42) and (43) yields both the desired energy dependent flux $[\theta_0^0(u)]$ as well as neutron age $[\tfrac{1}{2}\theta_0^2(u)/\theta_0^0(u)]$. It is important to note that these quantities were determined without truncating equations (31), (32)....

Recall that the purpose here is to develop a few-group machine program to generate group cross sections and other related reactor constants. Therefore, the moments equations must be put into multi-group notation, and integrated over a specific lethargy range u_n^- to u_n^+. Consider the zero moment equation, Eq (41) In group form it is

$$\int_{u_n^-}^{u_n^+} \Sigma_t(u)\,\vartheta_0^0(u)\,du = \int_{u_n^-}^{u_n^+} S(u)\,du + \int_{u_n^-}^{u_n^+} du \int_0^u \Sigma_s^0(u')\,\theta_0^0(u')\,du' \tag{49}$$

From the definition of a flux weighted group cross section, Eq (8), it is easy to see that the first term becomes

$$\int_{u_n}^{u_n^+} \Sigma_t(u)\,\vartheta_0^0(u)\,du = \Sigma_{t_n}\,\vartheta_{0_n}^0 \tag{50}$$

where Σ_{t_n} is the macroscopic total cross section of the n-th group and

θ°_{on} is the total group flux. (Remember that in this case the transformed flux has the property of being equal to the spatially integrated all angle flux in real space.) Similarly the second term is

$$\int_{u_n^-}^{u_n^+} S(u)\, du = \chi_n \tag{51}$$

where χ_n is the fraction of the source emitted in the n-th group providing S(u) is a normalized source. The last term

$$\int_{u_n^-}^{u_n^+} du \int_{o}^{u} \Sigma_s^{\circ}(u')\, \theta_o^{\circ}(u')\, du' \tag{52}$$

can be considered as a group of n double integrals where the integral from o to u on du' has been broken into n intervals, $\Delta u_1'$, $\Delta u_2'$, $\Delta u_3'$,... It is important that the integration be carried out on the variable u first since the limits of integration on u, i.e., u_n^- to u_n^+, are actually functions of u' and the maximum Δu of the scattering nucleus. Thus the last term becomes

$$\int_{u_n^-}^{u_n^+} du \int_{\Delta u_1'} \Sigma_s^{\circ}(u')\, \theta_o^{\circ}(u')\, du' + \int_{u_n^-}^{u_n^+} du \int_{\Delta u_2'} \Sigma_s^{\circ}(u')\, \theta_o^{\circ}(u')\, du'$$

$$\ldots + \int_{u_n^-} du \int_{\Delta u_{n-1}'} \Sigma_s^{\circ}(u')\, du' +$$

$$\int_{u_n^-}^{u_n^+} du \int_{\Delta u_n'} \Sigma_s^{\circ}(u')\, \theta_o^{\circ}(u')\, du' \tag{53}$$

Now, each double integral, say the j-th, is a measure of the neutron scatter transport from the j-th to the n-th group. Thus this last term, Eq (53), can be written as

$$\Sigma^{o}_{s1-n}\,\theta^{o}_{01} + \Sigma^{o}_{s2-n}\,\theta^{o}_{02} + \cdots \Sigma^{o}_{sj-n}\,\theta^{o}_{0j} +$$

$$\cdots + \Sigma^{o}_{sn}\,\theta^{o}_{0n} = \sum_{j=1}^{n} \Sigma^{o}_{sj-n}\,\theta^{o}_{0j} \qquad (54)$$

Substituting Eq (50), (51) and (54) into Eq (49) and rearranging terms yields

$$(\Sigma_{tn} - \Sigma_{sn-n})\,\theta^{o}_{0n} = \chi_n + \sum_{j=1}^{n-1} \Sigma^{o}_{sj-n}\,\theta^{o}_{0j}$$

or

$$\Sigma_{rn}\,\theta^{o}_{0n} = \chi_n + \sum_{j=1}^{n-1} \Sigma^{o}_{sj-n}\,\theta^{o}_{0j} \qquad (55)$$

where Σ_{rn} is the macroscopic total removal cross section from the n-th group, that is scatter removal plus absorption removal. Note that this equation is essentially a neutron balance. Treating the first moment equation similarly yields

$$(\Sigma_{tn} - \Sigma^{1}_{sn-n})\,\theta^{1}_{1n} = \frac{\theta^{o}_{0n}}{3} \sum_{j=1}^{n-1} \Sigma_{sj-n}\,\theta^{1}_{1j} \qquad (56)$$

Equation (56) is the heart of the difference between the multigroup moments method and the age diffusion methods. Note that the difference term multiplying θ^{1}_{1n} on the left is not the common transport cross section, in that it only subtracts away the within group scattering term. Rather than assume that collision densities, $(\Sigma_t \phi_0)$, are slowly varying, this method explicitly accounts for the linearly anisotropic term $\bar{\mu}\,\Sigma_s$. Using the term

$$\Sigma_{trn} = (\Sigma_{tn} - \Sigma^{1}_{sn-n}) \qquad (57)$$

Eq (56) becomes

$$\Sigma_{trn}\,\theta^{2}_{0n} = \frac{\theta^{o}_{0n}}{3} + \sum_{j=1}^{n-1} \Sigma^{1}_{sj-n}\,\theta^{1}_{1j} \qquad (58)$$

Finally the second moment equation can be written

$$\Sigma_{rn}\,\theta^2_{on} = 2\,\theta^1_{1n} + \sum_{j=1}^{n-1} \Sigma^o_{sj-n}\,\theta^2_{oj} \tag{59}$$

The group age, τ_n, may be defined in a similar manner,

$$\tau_n = \frac{\int_{\Delta n} \tau(u)\,\theta^o_o(u)\,du}{\int_{\Delta n} \theta^o_o(u)\,du} \tag{60}$$

However, group age must be more carefully interpreted than group cross sections. The only average age which has any meaning is the average age of neutrons which leave the group. Therefore, τ_n, is always to be associated with the lower energy boundary of the group. By equation (48) group age may be written as

$$\tau_n = \frac{1}{2}\frac{\theta^2_{on}}{\theta^o_{on}} \tag{61}$$

This may be rewritten by dividing equation (59) by $2\Sigma_{rn}\theta^o_{on}$ to yield

$$\tau_n = \frac{1}{\Sigma_{rn}\,\theta^o_{on}}\left[\theta^1_{1n} + \frac{1}{2}\sum_{j=1}^{n-1} \Sigma^o_{sj-n}\,\theta^2_{oj}\right] \tag{62}$$

But, θ^2_o can be expressed as

$$\theta^2_{oj} = 2\tau_j\,\theta^o_{oj} \tag{63}$$

Therefore, the group age expression becomes

$$\tau_n = \frac{1}{\Sigma_{rn}\,\theta^o_{on}}\left[\theta^1_{1n} + \sum_{j=1}^{n-1} \theta^o_{oj}\,\tau_j\right] \tag{64}$$

Finally by defining α_n as

$$\alpha_n = N\,\sigma_{rn}\,\theta^o_{on} \tag{65}$$

$$\alpha_n = \chi_n + \sum_{j=1}^{n-1} \frac{\sigma_{s\,j-n}}{\sigma_{r\,j}} \alpha_j \tag{66}$$

$$\theta^1_{1n} = \frac{\alpha_n}{3\sigma_{tr\,n}\,\sigma_{r\,n}\,N^2} + \frac{1}{\sigma_{tr\,n}} \sum_{j=1}^{n-1} \sigma^1_{s\,j-n}\,\theta^1_{1\,j} \tag{67}$$

$$\tau_n = \frac{\theta^1_{1n}}{\alpha_n} + \frac{1}{\alpha_n} \sum_{j=1}^{n-1} \frac{\sigma^o_{s\,j-n}}{\sigma_{r\,n}}\,\alpha_j\,\tau_j \tag{68}$$

where N = total nuclei density, nuclei/cm^3 x 10^{-24}, and the σ's are the respective microscopic cross sections in barns with the subscript notation the same as in previous equations. The group fluxes, the ϕ^o_{on} and the group ages can be determined by solving the above equations. Cross section data is read in from a cross section library, which will be discussed later, and calculation is begun with group one where the scatter in term is zero. Calculation proceeds consecutively through all remaining groups down to lethargy 17 (0.414 ev). All group fluxes are then normalized to $\phi(16) = 1$ in order to be normalized to the thermal flux at the same value. It should be noted however that the input cross sections are generated from a modified version of GAM (Ref 13) which does not include the effects of resonance absorption. Therefore, resonance absorption must be accounted for in the code. This is done by calculating the resonance escape probability.

Resonance

The resonance escape probability is the probability that a neutron will escape resonance capture during slowing down and is given by Murray as (Ref 16:62)

$$p = \exp\left[-\int_E^{E_o} (\Sigma_a/\xi\,\Sigma_t\,\frac{dE}{E}\right] \tag{69}$$

where p is the resonance escape probability

E_o is the reference energy, usually 10 Mev

ξ is the average logarithmic energy decrement per collision

Σ_a is the macroscopic absorption cross section, cm^{-1}

Σ_t is the macroscopic total cross section, cm^{-1}

Murray also points out that the exponent in Eq (69) may be written as

$$\frac{N^A}{\xi\,\Sigma_s}\int_E^{E_o} \frac{\sigma_a^A}{1+\frac{N^A\sigma_a^A}{\Sigma_s}}\,\frac{dE}{E} \tag{70}$$

where N^A is the nuclide density of the resonance absorber x 10^{-24} per cm^3

σ_a^A is the microscopic absorption cross section of the resonant absorber, barns

and Σ_s is the macroscopic scattering cross section, cm^{-1}

Note that the term in the integral (70) depends on the total scattering cross section per absorber atom Σ_s/N^A, which is a function of the moderator-fuel ratio. The integral term in (70) above, defined as the resonance integral, is usually approximated by empirical expressions. Isbin (Ref 11:459) lists some of these expressions:

(for U^{238})

$$(RI)_{eff} = 2.69\left(\frac{\Sigma_s}{N^{238}}\right)^{0.471} \qquad 0 \le \frac{\Sigma_s}{N^{238}} \le 4000 \tag{71}$$

$$\ln(RI)_{eff} = 5.64 - \frac{163}{(\Sigma_s/N^{238})^{0.65}} \qquad \frac{\Sigma_s}{N^{238}} > 4000 \tag{72}$$

and $(RI)_{eff} = 260$ for $\Sigma_s/N^{238} = \infty$,

and for Th^{232}

$$(RI)_{eff} = 8.33\left(\frac{\Sigma_s}{N^{232}}\right) \qquad 0 \leq \frac{\Sigma_s}{N^{232}} \leq 1500 \qquad (73)$$

$$(RI)_{eff} = 70 \qquad \frac{\Sigma_s}{N^{232}} > 1500 \qquad (74)$$

where $(RI)_{eff}$ is the effective resonance integral in barns.

Σ_s is the macroscopic scattering cross section of the mixture over the resonance region, cm^{-1}.

The resonance escape probability given in equation (69) then becomes

$$p = \exp\left[-(RI)_{eff}\ N^A/\xi\Sigma_s\right] \qquad (75)$$

where the $\xi\Sigma_s$ is for the mixture.

The effect of resonance absorption is then introduced into equations (55), (56) and (59), the moments equations, by increasing the σ_{r9} and σ_{r10} (resonance takes place in groups 9 and 10), thru the equations (Ref 1:179)

$$\sigma_{rn} = \sigma_{rn} + \sigma_{an}^{res} \qquad (76)$$

$$\text{where } \sigma_{an}^{res} = A_n(RI)_{eff}/\Delta u \qquad (77)$$

where σ_{rn} = microscopic removal cross section of group n, barns

A_n = an empirical factor to apportion the resonance absorption between groups 9 and 10.

Δu = the lethargy interval over which resonance is effective. Murray gives a value of 5.5.

This has the effect of reducing the downscatter to groups 10 and 11 by appropriate factors without violating neutron conservation within the

groups. The code allows the user the option of providing his own value of $(R\bar{I})_{eff}$ should he so desire.

The equations for determining the fast flux spectrum and age are now complete. The equations specifying the thermal flux spectrum are developed in the next section.

Thermal Flux Spectrum

In elementary reactor physics the thermal flux spectrum is usually assumed to be in thermal equilibrium with the moderator producing a Maxwellian distribution. This is not true however since absorption removes neutrons and thus disturbs the equilibrium. On the other hand the moments equation used for epithermal energies can no longer be used since they take into account only downscatter, whereas the possibility of upscatter must be considered when the energy of the moderater nuclei is comparable to the neutron energy.

Wilkins (Ref 20) derived a second order differential equation which describes the thermal neutron spectrum in an infinitely heavy Maxwellian gas and allows absorption. The thermal spectrum generated by this code is calculated by solving the Wilkins equation. This solution is expressed in terms of a cross section determined parameter, the absorption parameter, which in turn (in the case of a mixture of nuclides) is expressed in terms of the thermal utilization, another cross section parameter. These two parameters are developed next, followed by a development of the Wilkin's equation.

Thermal Utilization

The thermal utilization f, is defined as the ratio of thermal neutrons absorbed in fuel to all thermal neutrons absorbed or,

$$\frac{1}{f} = \frac{[vf\ \Sigma_a]_{fuel} + [vf\ \Sigma_a]_{mod}}{[vf\ \Sigma_a]_{fuel}} = 1 + \frac{[vf\ \Sigma_a]_{mod}}{[vf\ \Sigma_a]_{fuel}} \qquad (78)$$

where vf is the volume fraction. However, the thermal neutron flux in the fuel elements of a heterogenous reactor will be depressed due to the strong absorption in the fuel. This effect complicates the calculation of the thermal utilization. The proper expression in the case of heterogeneous reactors is (Ref 16:87-88)

$$\frac{1}{f} = 1 + \frac{[vf\ \Sigma_a]_{mod}}{[vf\ \Sigma_a]_{fuel}} F + (E - 1) \qquad (79)$$

where $F = K_0 w \coth (K_0 w)$ (in the case of slab geometry)

$E = K_1(W-w) \coth [K_1(W-w)]$ (For cylindrical geometry see Ref 16:87-88)

$K_0^2 = \Sigma_0/D_0$

$K_1^2 = \Sigma_1/D_1$

W = half the distance between fuel element midpoints

w = the distance from the midpoint of the fuel element to the boundary of the fuel element

and the subscripts 0 and 1 refer to fuel and moderator respectively. In effect F is actually the ratio of the flux at the fuel surface to the average flux in the fuel while (E - 1) accounts for the fact that the flux in the moderator rises from a minimum at the fuel element to a maximum midway between elements.

In the code the F and E factors must be read in for heterogeneous cell calculations but, they are assumed equal to unity for homogeneous systems. In the case of heterogeneous reactors in which thin closely spaced fuel plates are used the system can be assumed homogeneous from a calculational point of view.

The Absorption Parameter is defined as

$$\Delta = \frac{2A\sigma_a(E)\sqrt{E}/kT}{\sigma_s(E)} \tag{80}$$

where A = nuclide mass, amu

k = Boltzmann's constant in MeV/(°K)^4

T = temperature, deg. Kelvin

Since Δ is used only for thermal calculations, $\sigma_s(E)$ is assumed constant and $\sigma_a(E)$ is assumed to vary as 1/v. Then

$$\Delta = \frac{2A\sigma_{ao}}{\sigma_s} \tag{81}$$

where σ_{ao} = microscopic absorption cross section at energy kT in barns. Weinberg and Wigner (Ref 19:337) point out that for a mixture of nuclides the absorption parameter should be divided by the factor (1 - f). Therefore, to find the absorption parameter for a mixture of nuclides the following equation is used:

$$\Delta = \frac{1}{(1-f)} \frac{\sum_i \Sigma_T^i \Delta^i}{\sum_i \Sigma_T^i} \tag{82}$$

where Σ_T^i = total macroscopic cross section of the i-th nuclide, and the summing variable i, refers to all moderator nuclides.

The Wilkins Equation

The Boltzmann transport equation for steady state conditions in an infinite homogeneous medium was given in equation (8). It is repeated here for convenience.

$$u \frac{\partial}{\partial x} \varphi(x,u,u) + \Sigma_t(u)\, \varphi(x,u,u)\, S(x,u,u)$$

$$\int d\Omega' \int_0^u \Sigma_s(u')\, \varphi(x,u',u')\, f(u' - u; u_0)\, du' \qquad (9)$$

However in the thermal case we shall take the source, $S(x,u,u)$ as zero, and introduce the neutrons as a boundary condition by specifying some arbitrary number of neutrons slowing down past the upper energy limit of the thermal group. In the absence of a source and because the media is infinite the variation of flux with both spatial position and direction disappears. That is $\phi(x,u,u) = \phi(u)$ and $\frac{\partial \phi(x,u,u)}{\partial x} = 0$. Thus the Boltzmann equation becomes

$$[\Sigma_a(u) + \Sigma_s(u)]\, \varphi(u) = \int_{-\infty}^{+\infty} \Sigma_s(u' - u)\, \varphi(u')\, du \qquad (83)$$

or in terms of energy

$$[\Sigma_a(E) + \Sigma_s(E)]\, \varphi(E) = \int_0^{\infty} \Sigma_s(E' - E)\, \varphi(E')\, dE' \qquad (84)$$

where the limits on the integral are correctly o to ∞ because of the the possibility of upscatter as well as downscatter.

where

$$\Sigma_s(E) = \int_0^{\infty} \Sigma_s(E - E')\, dE' = \int_{-\infty}^{\infty} \Sigma_s(u')\, f(u' \rightarrow u)\, du' \qquad (85)$$

and $\Sigma_s(E - E')\, dE' / \Sigma_s(E)$ is the probability that a neutron at energy E, will be scattered into the energy range E' to E' + dE'.

Both Hurwitz (Ref 10) and Wilkins (Ref 20) solve this equation for a heavy gaseous moderator. Hurwitz's solution is more easily followed and his method will be used to develop the Wilkins equation.

According to the principle of detailed balance for neutrons, in

the case of zero absorption

$$\Sigma(E' - E)\, M(E') = \Sigma(E - E')\, M(E) \tag{86}$$

where M(E) = the Maxwellian flux at energy E. As Hurwitz states, "This [is] a property of the kernel which is independent of the flux". Using this, equation (84) can be rewritten in terms of the new variable

$$\Psi(E) = \Phi(E)/M(E) \tag{87}$$

where $\Psi(E)$ is the deviation from a Maxwellian flux. Thus equation (84) becomes

$$\Sigma_a(E)\, \Psi(E) = \int_0^\infty \Sigma(E - E')\, [\Psi(E') - \Psi(E)]dE' \tag{88}$$

Hurtitz then uses a Taylor series expansion of $\Psi(E')$ about the point $E' = E$ to give

$$\left[\frac{\Sigma_a(E)}{\Sigma_s(E)}\right] \Psi(E) = \overline{\Delta E}\, \frac{d\Psi}{dE} + \frac{1}{2}\, \overline{\Delta E^2}\, \frac{d^2\Psi}{dE^2} \tag{89}$$

where $\overline{\Delta E} = 2(2kT - E)/A$ (Ref. 10:283)

$\overline{\Delta E^2} = 4EkT/A$

and A = mass number of the moderator nuclei.

Substituting for $\overline{\Delta E}$ and $\overline{\Delta E^2}$ in equation (89) yields

$$EkT\, \frac{d^2\Psi}{dE^2} + (2kT - E)\, \frac{d\Psi}{dE} - \frac{\Delta}{4}\Psi = 0 \tag{90}$$

where Δ is the absorption parameter defined by equation (81).

It is convenient at this point to introduce a new variable x which is a dimensionless velocity defined by

$$x = \sqrt{E/kT} \tag{91}$$

In terms of this new variable

$$x^2 N(x) = 2\, E\Phi(E) = 2\Phi(u) \tag{92}$$

and $$\Psi(E) = Cx^{-2} \exp(x^2)\, N(x) \tag{93}$$

where $N(x)$ = number of neutrons at velocity x, and C is a constant.

Now noting that

$$\Psi(E) = f(x) \tag{94}$$

$$\Psi'(E) = f'(x)\,\frac{dx}{dE} \tag{95}$$

$$\Psi''(E) = f''(x)\left[\frac{dx}{dE}\right]^2 f'(x)\left[\frac{d^2x}{dE^2}\right]\left[\frac{dx}{dE}\right] \tag{96}$$

where $$\frac{dx}{dE} = \frac{1}{2xkT} \text{ and } \frac{d^2x}{dE^2} = -\frac{1}{4(kT)^2 x^3} \tag{97}$$

and where the primes denote differentiation with respect to the respective independent variables.

We find that

$$\Psi'(E) = \frac{\exp(x^2)}{2\,kT}\left\{ x^{-3} N'(x) + (2x^{-2} - 2x^{-4})\, N(x)\right\} \tag{98}$$

$$\Psi''(E) = \frac{\exp(x^2)}{4(kT)^2}\left\{ x^{-4} N''(x) + (4x^{-3} - 5x^{-5})\, N'(x) + (8x^{-6} - 8x^{-4} + 4x^{-2})\, N(x)\right\} \tag{99}$$

Substituting Eq (95) and (96) into Eq (87) results in the well known Wilkins equation

$$xN''(x) + (2x^2 - 1)N'(x) + (4x - \Delta)N(x) = 0 \tag{100}$$

This equation is solved by setting

$$N(x) = x^2 \exp(-x^2)M(x) \tag{101}$$

which converts Eq (100) into

$$xM''(x) + (3 - 2x^2)M'(x) - \Delta M(x) = 0 \tag{102}$$

This equation is solved by the power series method. The method is straightforward and for an expansion about the point $x = 0$ results in

$$M(x) = \sum_{n=0}^{\infty} a_n x^n \qquad (103)$$

where $a_1 = a_0 \frac{\Delta}{3}$

$$a_2 = a_0 \frac{\Delta^2}{24}$$

$$a_n = \frac{1}{n(n+2)} [2(n-2)\, a_{n-2} + \Delta\, a_{n-1}]$$

and a_0 is arbitrarily set equal to 1. Equation (102) can also be expanded about the point $x = a$. In this case by letting $x - a = V$, Eq (102) is converted to

$$(V+a)\, M''(x) + (3 - 2V^2 - 4Va - 2a^2)M'(x) - \Delta M(x) = 0 \qquad (104)$$

and the solution is

$$M(x) = \sum_{n=0}^{\infty} b_n (x-a)^n \qquad (105)$$

where

$$b_n = \frac{1}{a(1-n)n} [b_{n-1}(n-1)(n-1-2a^2) -$$

$$b_{n-2} \{4a(n-2) + \Delta\} - b_{n-3}\, 2(n-3)]$$

In the code the values $M(x)$, $M'(x)$, $N(x)$ and $x^2N(x)$, (which is $\propto \phi(u)$) are calculated at every 0.25 lethargy intervals beginning at $u = 23$ and continuing to $u = 16$. The calculation employs Eq (103) for the first point then uses Eq (105) for all subsequent points, the point $x = a$ used in Eq (105) being the point x in the previous calculation. Terms are added until relative contributions of additional terms to $M(x)$ and $M'(x)$ are less than 10^{-6} relative value. All thermal fluxes are normalized to $\phi(16) = 1$ to match the epithermal fluxes from the moments calculation. To arrange for the proper incrementing of the variable x, such that Δx corresponds to 0.25 leth-

argy units, the following is used:

$$x_{(u=23)} = \exp\left[(-11.5)(10^{12}/8.616T)^{\frac{1}{2}}\right] \tag{106}$$

$$x_2 = x_1 + \Delta x \tag{107}$$

$$\Delta x = x_1\,[\exp(.125) - 1] \tag{108}$$

T is temperature in $^{\circ}$Kelvin

Thermal Constants

In the code the thermal range is arbitrarily defined as 0 to 1.25 ev which is ∞ to 16 on the lethargy scale. All thermal constants are calculated for this interval.

Most Probable Neutron Velocity

The most probable neutron thermal neutron velocity is found by locating the peak of the N(x) vs x curve by the code. However, since values of N(x) are found for 0.25 lethargy units, the value of v_p is not exact. The true probable velocity is

$$v_p = \left[x(16{,}500\ T)^{\frac{1}{2}}\right]_{x=x_p} \tag{109}$$

(T in $^{\circ}K$)

Average Neutron Velocity

The average neutron velocity is found by Eq (103) except that the equation is evaluated at $x = \bar{x}$, which is the average x, rather than at $x = x_p$. The average normalized neutron velocity $\bar{x}$ is

$$\bar{x} = \frac{\int_0^{x(u=16)} x\,N(x)\,dx}{\int_0^{x(u=16)} N(x)\,dx} \tag{110}$$

The integral over N(x) can be found by integrating Eq (100)

$$\int_0^x N(t)\,dt = \frac{xN'(x) + 2(x^2 - 1)\,N(x)}{\Delta} \tag{111}$$

where

$$N'(x) = 2\left[x^3 \exp(-x^2)\, M(x)\right]\left\{\frac{1}{x^2} - 1\right\} + x^2 \exp(-x^2)\, M'(x) \qquad (112)$$

Thermal Diffusion Length Squared

The thermal diffusion length squared is defined as one sixth of the mean square distance a neutron travels from the time it becomes thermal until capture. The total distance $\vec{r}$, a neutron travels in making n collisions is

$$\vec{r} = \vec{r}_1 \hat{\theta}_1 + \vec{r}_2 \hat{\theta}_2 + \vec{r}_3 \hat{\theta}_3 + \ldots + \vec{r}_n \hat{\theta}_n \qquad (113)$$

where $r_i \hat{\theta}_i$ is the magnitude and direction vector of the neutron between collisions. The total mean square path is the scalar, or dot, product of equation (113) with itself. Note that the cross product terms will involve the dot products of the various neutron direction vectors $\hat{\theta}_i$, and $\hat{\theta}_i \cdot \hat{\theta}_j = \cos\theta_{ij}$ where θ_{ij} is the angle between $\hat{\theta}_i$ $\hat{\theta}_j$. But for isotropic scatter the angle θ_{ij} between two successive paths i and j is completely random. Thus, there are as many positive cosines as negative ones, and on the average, cancel each other out. So

$$\overline{r^2} = \sum_{k=1}^{K} r_k^2 \qquad (114)$$

where k is the collision number and K is the total number of collisions. It can be shown that any r_k^2 is twice the mean free path squared for that collision (16:278), therefore

$$\overline{r^2} = \sum_{k=1}^{K} 2\lambda_k^2 = 2K\,\overline{\lambda^2} = 2K/\overline{\Sigma_t^2} \qquad (115)$$

where $\bar{\lambda}$ is the average neutron mean free path over the thermal range, cm

$\bar{\Sigma}_t$ is the total average thermal macroscopic total cross section cm^{-1}.

Sims (Ref 17:14-16) shows that n, the number of collisions a neutron makes is equal to the ratio of the scattering rate to the absorption rate. Hence, equation (115) becomes

$$\bar{r}^2 = \frac{2 \int_0^{1.125 \text{ ev}} \Sigma_s \phi(E)\, dE}{\bar{\Sigma}_t^2 \int_0^{1.125 \text{ ev}} \Sigma_a(E)\, \phi(E)\, dE} \tag{116}$$

Now, the thermal diffusion length squared, L^2, is one-sixth this value. The quantity $\bar{\Sigma}_t$ can be found by assuming that Σ_s is constant and Σ_a has a 1/v dependence in the thermal range. Since $x = \sqrt{E/kT}$, L^2 can be written as

$$L^2 = \frac{\Sigma_s \int_0^{x_0} x\, N(x)\, dx}{3\, \bar{\Sigma}^2\, \Sigma_{ao} \int_0^{x_0} N(x)\, dx} \tag{117}$$

or

$$L^2 = \frac{\Sigma_s\, \bar{x}}{3\, \bar{\Sigma}^2\, \Sigma_{ao}} \tag{118}$$

where x_0 = x corresponding to 1.125 ev,

$\Sigma_{ao} = [x\, \Sigma_a(x)]_{x=x_p}$, the most probable value. However, $\bar{\Sigma}$ is itself an average and can be rewritten as

$$\bar{\Sigma} = \Sigma_s + \bar{\Sigma}_a \tag{119}$$

where

$$\Sigma_s = \frac{\int_0^{x_o} \Sigma_s(x)\ XN(x)\ dx}{\int_0^{x_o} x\ N(x)\ dx} \tag{120}$$

Hence the equation for L^2 can finally be written as (Ref 17:32)

$$L^2 = \frac{\sigma_s\ \bar{x}}{3\ \sigma_{ao}\ [(\sigma_s + \sigma_{ao}(x)\ N]^2} \tag{121}$$

where N is again the nuclei density x 10^{-24} cm^{-3}

σ is the microscopic cross section with subscripts the same as before in barns

$\bar{x}$ is the average normalized neutron velocity and the energy range for the calculations is understood to range from 0 to 1.125 ev in the code.

Migration Area

The total migration area is

$$M^2 = L^2 + \tau_{Th} \tag{122}$$

where τ_{Th} is the age to thermal.

In the code the migration area is found by adding the moments methods age to lethargy 16 (1.125 ev) to the value of L^2 computed from lethargy 16 to ∞.

Thermal Cross Sections

The 2200 m/sec cross sections for the appropriate nuclides are input data to the code from cross section libraries. However, these cross sections must be adjusted to compensate for temperatures above

293 deg. Kelvin. Assuming that σ_a has a 1/v dependence and σ_s is constant, the thermal cross sections are adjusted accordingly and then used in the thermal calculations. Finally the thermal cross sections are adjusted to account for deviation of the thermal spectrum from Maxwellian. The proper absorption cross section is

$$\sigma_a = \frac{\sigma_{ao}}{\bar{x}} \tag{123}$$

and the transport cross section is

$$\sigma_{tr} = \sigma_a + \sigma_s (1 - \bar{u}) \tag{124}$$

where the term $(1 - \bar{u})$ is also input data from the cross section libraries.

Actually, an iterative procedure should be used. In Eq (123), $\sigma_{ao} = [x\, \sigma_a(x)]_{x=x_p}$. However, the thermal cross sections read into the program are 2200 meter/sec cross section values. They are then adjusted for temperature, assuming that the flux is Maxwellian. But, when the flux is not Maxwellian the most probable thermal neutron velocity x_p is not equal to x_p for a Maxwell-Boltzmann distribution. Therefore, the value of x_p used in calculating σ_{ao} may not be correct. The correct procedure would be:

(1) adjust the absorption cross section using equation (123) at the end of the thermal calculations.

(2) re-input the value of $\bar{\sigma}_a$ calculated by equation (123).

(3) recalculate the thermal spectrum using the new value of $\bar{\sigma}_a$.

(4) return to step (1) above until $\bar{\sigma}_a$ remains constant.

To keep computer run time to a minimum the iterative procedure

shown above is not used.

Cross Section Collapsing

The theory presented in the previous paragraphs is all incorporated in Chain I of OLD BARNYARD to produce flux as a function of energy, both epithermal as well as thermal, in an infinite arbitrary mixture of nuclides. This energy dependent flux is expressed as a histogram of 11 epithermal values and 9 thermal values. These flux values are employed in Chain II to calculate flux weighted average cross sections over any number of epithermal groups from 1 to 11 and a thermal group. The exact number of epithermal groups, called broad groups to distinguish them from the input fine groups, are at the specification of the user. The boundaries of the broad groups are restricted to values which are boundaries of the fine groups.

Broad group cross sections are calculated from the formula

$$\sigma_n^{BG} = \frac{\sum_i \varphi_i \sigma_n^i}{\sum_i \varphi_i} \qquad (125)$$

where σ_n^{BG} is the microscopic cross section of n-th kind for a broad group

φ_i is the total flux of the i-th fine group

σ_n^i is the microscopic cross section of the n-th kind for the i-th fine group.

The interrelationships which exist among some of the cross sections output by the code is not always obvious. The following explanation, therefore, may be useful.

Total Transfer

The group to group total transfer cross section is that value which when multiplied by the losing group flux gives the rate of transfer from j to k and is the sum of inelastic, elastic (P0) and twice the n - 2n cross sections from j to k. Note that elastic (P1), does not contribute to total transfer.

Removal

The removal cross section is that value which when multiplied by its broad group flux gives the rate of removal of neutrons from that group. Thus the removal cross section is the sum of absorption, inelastic, elastic (P0) and the n - 2n removal cross sections.

Transport

The transport cross section is defined as

$$\Sigma_{tr} = \Sigma_t - \bar{u}\,\Sigma_s = \Sigma_t - \Sigma_s^1 \qquad (126)$$

Thus the transport cross section is the total cross section minus the elastic (P1) cross section. The transport cross section is calculated as the sum of inelastic, the n - 2n, absorption, and elastic (P0) cross section. The transport cross section thus includes both within group and out-of-group scatter terms.

Summary

"OLD BARNYARD" was written to assist professors and students engaged in a reactor physics course. It does this by providing the few-group cross sections and related constants usually required to solve problems assigned in such a course.

The code calculates the energy dependent flux necessary to flux weigh the group cross sections in the first of two sequential chains. The epithermal flux is calculated by the neutron transport, moments method, and the thermal flux is calculated by solving the Wilkins equation by the power series method. Resonance effects are included by the use of empirical equations. The second chain collapses the eleven fine epithermal groups to any number of broad groups from one to eleven.

Finally one should always keep in mind the limitations of these calculations. The accuracy of any output is dependent on the input cross sections employed. It must be remembered that the input, eleven group, cross sections are the result of collapsing the 68 group cross sections from GAM-1 using two specific standard problems. This largely limits the accuracy of the code. Also, the calculation of the most probable thermal neutron velocity is only accurate within 0.25 lethargy units. At the present time resonance calculations are effectively limited to one nuclide. This is because chain two includes resonance absorption in calculating the broad group cross sections. If two resonance nuclides are included in chain two, the resonance absorption effect is, in effect, counted twice by assigning all the resonance absorption occurring in the mixture to each of the resonance nuclides. Finally, the equations developed in this chapter and used in the code inherently apply limitations. Recall that the moments equations were based on an infinite homogeneous medium and a plane source of infinite dimensions at the origin. Similarly the Wilkins equation is applicable to an infinitely heavy Maxwellian gas and a source free medium.

Bibliography

1. Argonne National Laboratory. Reactor Physics Constants. USAEC Document ANL-5800. Lemont, Illinois, July 1963.

2. Bridgman, C. J. An Analytical Solution of the Neutron Moments Equations. Unpublished dissertation. Raleigh, N. C.: North Carolina State College, 1962.

3. Churchill, R. V. Fourier Series and Boundary Value Problems. (Second Edition). New York: McGraw-Hill Book Company, Inc., 1963.

4. Cohen, E. R. "The Neutron Velocity Spectrum in a Heavy Moderator." Nuclear Science and Engineering, 2:227-245 (May 1957).

5. Davison, B., and J. B. Sykes. Neutron Transport Theory. London: Oxford University Press, 1957.

6. Field, J. A. A.[1], D. A. Jardine[2], E. S. Lee[1], J. A. N. Lee[3], and D. G. Robinson[2]. Kingston Fortran II Specifications. Document distributed by the authors, July 15, 1964.

    1. Department of Electrical Engineering, University of Toronto, Toronto, Ontario.

    2. Dupont of Canada Ltd., Research Centre, Kingston, Ontario.

    3. Computing Centre, Queens University, Kingston, Ontario.

7. Glasstone, S. and M. C. Edlund. The Elements of Nuclear Reactor Theory. Princeton, New Jersey: D. Van Nostrand Company, Inc., 1952.

8. Hansen, R. H. Theoretical Calculation of Neutron Age. Defense Documentation Centre Document AD 603603, Thesis, Air Force Institute of Technology, Dayton, Ohio 1964.

9. Hughes, D. J., and R. B. Schwartz. Neutron Cross Sections. USAEC Document BNL-325 (Second Edition). Upton, New York: Brookhaven National Laboratory, 1958.

10. Hurwitz, H., Jr., M. S. Nelkin, and G. J. Habetler. "Neutron Thermalization I. Heavy Gaseous Moderator." Nuclear Science and Engineering, 1:280-312 (August 1956).

11. Isbin, H. S. Introductory Reactor Theory. New York: Reinhold Publishing Corporation, 1963.

12. Jahnke, E., and F. Emde. Tables of Functions. (Fourth Edition). New York: Dover Publications, 1945.

13. Joanou, G. D., and J. S. Dudek. GAM-1: A Consistent P1 Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Constants. USAEC Document GA-1850. San Diego, California: John Jay Hopkins Laboratory for Pure and Applied Sciences, 1961.

14. Marshak, R. E. "Theory of the Slowing Down of Neutrons by Elastic Collision with Atomic Nuclei." Reviews of Modern Physics, 19: 185-238 (July 1947).

15. Meghreglian, R. V., and D. K. Holmes. Reactor Analysis. New York: McGraw-Hill Book Company, Inc., 1957.

16. Murray, R. L. Nuclear Reactor Physics. Englewood Cliffs, N. J.: Prentice-Hall, Inc., 1957.

17. Sims, E. P. The Thermal Neutron Spectrum in a Finite-Mass Medium. Defense Documentation Center Document AD 621119, Thesis, Air Force Institute of Technology, Dayton, Ohio, 1965.

18. Sneddon, I. N. Fourier Transforms. New York: McGraw-Hill Book Company, Inc., 1951.

19. Weinberg, A. M., and E. P. Wigner. The Physical Theory of Neutron Chain Reactors. Chicago, Illinois: University of Chicago Press, 1958.

20. Wilkins, J. E., Jr., Effect of the Temperature of the Moderator on the Velocity Distribution of Neutrons for a Heavy Moderator. USAEC Document CP-2401, 1944.

21 The Staff of the Bateman Manuscript Project. Tables of Integral Transforms, Vol I. (Compiled from the notes of Harry Bateman), New York: McGraw-Hill Book Company, Inc., 1954.

Appendix A

Glossary of Symbols Used in Chain One

The following is a list of all variables used in chain one of the code. The abbreviation "af" is defined as atom fraction. See Appendix F for an example of the scattering matrices. In the scattering matrices the elements of the matrix, a_{jk}, are defined as follows: a diagonal element if j=k, an upper triangular element if j<k, a lower triangular element if j>k.

Name	Location	Meaning or Use
AN	0804	a_n in the recursion formula of eq (100)
	0805	b_n in the recursion formula of eq (102)
AN1	0503+01	a_{n-1} in the recursion formulas of eq (100) and eq (106)
AN2	0503+05	a_{n-2} in the recursion formulas above
AN3	0805	a_{n-3} in the recursion formulas above
AREA	0810+01	The area under the N(x) vs X curve (see eq 108)
AREAX	0809-03	Summing variable to find the area under the xN(x) vs X curve
ATOMS	1061-01	The last five alphameric characters of the n^{th} nuclides identification
	1025+03	A summing variable for use in the first moment equation, eq (64)

$$\sum_{j=1}^{k-1} \frac{\Sigma_s^o + \Sigma_{n,n^*} + 2\Sigma_{n2n\ j-k}}{\Sigma_r}$$

Name	Location	Meaning or Use
C	0810	N'(x) see eq (109)
DELTA	1061+02	Absorption parameter of the n^{th} nuclide
	0503	Absorption parameter of the mix
DENBR	1061+01	Total nuclide density of the mix (atoms per barn-cm)
DENS	1061-01	A code number used in chain 2
	1051+04	The quantity $[1/3(\Sigma_{MGTn} \cdot \Sigma_{Rn})]$ for use in eq (65)
DSUM	0806+04	Summing variable for terms of the derivative of M(x) from the Wilkins equation
DTERM	0806	Derivative of the n^{th} term in the power series expansion of the Wilkins equation
E	1002+01	The E factor for heterogeneous cell calculations: E = 1 for homogeneous mixture
ELSQRD	0851+02	Thermal diffusion length squared
EMSQRD	0852+01	The migration area
F	1029+01	A normalizing factor such that F[FLX(10)] = 1
	0808+01	The value of N(x) for the point x = p
	0816+01	Flux per unit lethargy $x^2N(x)$
	0810+04	A normalizing factor for thermal fluxes such that ϕ(u=16) = 1
	0109+02	Storage for the maximum group flux. Used to normalize the flux spectrum to a maximum value of 100
FF	1002+01	The F factor for heterogeneous calculations: F = 1 for homogeneous mixture
FIS	1061+02	A code number: Is this a fission nuclide: 0 = NO, 1 = YES
FLX	1027	The group flux times the removal cross section. It is α in the moments equations
	1023	The group flux per unit lethargy

Name	Location	Meaning or Use
I	0500+08	Internal index used to read data for all nuclides in the mix
	0503+09	Index used to identify the energy group in the thermal calculations
II	0814+01	A gate for locating the peak of the N(x) vs x curve
	0108	An initial value for the index J in SS 109+01 if a thermal calculation was not performed
IIN	1006+03	The maximum number of groups, below any scattering group, to which scatter occurs
IN	1006+01	The number of groups from which scatter occurs
INDEX	1043+01	Code indicating what source has just been read in
J	1065+01	Internal index used in various DO loops
JJJ	0995	A code number for the type of resonance absorber: 1 = U or W, 2 = Th
K	1065+03	A gate to determine if the desired source has been read in: 0 = still seeking source, 1 = found source. Read rest of source deck into a dead end
	3014	Converts log (FLUX) to an integer for flux spectrum printout
KK	1007-02	The group number of the lowest group to which scatter occurs, from a given group
	0112	A sorting variable in outputting the fast fluxes
	3001+01	A gate to separate thermal and epithermal fluxes and a counter to sort the epithermal fluxes
KKK	0500+01	A gate to initialize TSIG9, TSIG10, and PSIG9
KKZ	1067+02	A gate to index KZ by increments of 2 after the first elastic (P0) and elastic (P1) are stored
KZ	1067+01	A counter used in sorting the XSAM's

Name	Location	Meaning or Use
LE	2002+01	Index used to convert the k index of $\sigma_{j\to k}$ of elastic (P0) and (P1) for proper storage in the lower half of the scattering matrices
LGP	1001	Subscripted variable to read the problem I. D. card
	1047	Used to print out the group with the unit source
	1050	Identifies source as users own
LQ	1061+02	A code number: indicates type of elastic scatter: 1 = (n,n*), 2 = (n,2n), 3 = both, 4 = neither
LS	1061+02	The number of entries in the cross section library starting on the second card (except $\nu\sigma_f$ if present)
	2002	Index used to convert the j index of σ_{j-k} of elastic (P0) and (P1), see LE above
M	1067+01	A code number: indicates type of media: 0 = homogeneous, 1 = heterogeneous
MOD	1067+01	A code number: Is nuclide: 1 = moderator, 2 = fuel, 3 = other
NA	1061+02	A code number: Is epithermal absorption appreciable: 0 = NO, 1 = YES
NEW	1061+01	Number between 2 and 21 indicating source to be used
NRNUC	1060+01	The number of nuclides in the problem
P	1061-01	Nuclide density of the n^{th} nuclide (atoms per barn-cm)
	1061+01	Atom Fraction of the n^{th} nuclide
	1051+07	A summing variable for the $\sigma^1_{sj-k}\theta^1_{lj}$ in eq (65)
	0503+06	The point x=a for use in eq (102) for a series expansion of the Wilkins equation about the point x=a
	0808	The value of the variable x for the n^{th} thermal group

Name	Location	Meaning or Use
	0813	Lower lethargy boundary for the various thermal group fluxes
PREVF	0503+02	A storage variable for N(x) from the previous calculation
	0816	N(x) at the peak of the N(x) vs x curve
PREFX	0809-02	Temporary storage of xN(x) from the previous calculation. Used to find the area under the xN(x) vs x curve
PROB	0728	The resonance escape probability
PSI	1061+02	The average logarithmic energy decrement ξ
PSIGF	500+02	Sum of the products of the atom fraction times the microscopic thermal absorption cross section of fuel nuclides only
PSIGM	0500+03	Same as PSIGF except for moderator nuclides only
PSIGO	0500+04	Same as PSIGF except for other nuclides only
PSIG9	0705+04	Sum of the products of ξ times average of the scattering cross section of groups 9 and 10 for the mixture (Average scattering cross section in the resonance region)
PU	1045	A dead end to read through the remaining undesired sources after the desired one has been found
	0995	Nuclide density of the resonance nuclide(s) in atoms per barn-cm. Immediately converted to atom fraction
RES	1060+01	A code number: Is a resonance calculation to be performed: 0 = NO, 1 = YES
RHO	1051+01	A holding variable for the lower lethargy (upper E) boundary of the n^{th} epithermal group
RIEFF	0995	A code number: Is the resonance integral to be calculated: 0 = calculated by code, non-zero = value of resonance integral to be used.
SA	1027+01	First moment of the fourier transformed flux

Name	Location	Meaning or Use
SAM	1061-01	The first five alphameric characters of the nuclide's identification
	1025+02	The summation term in eq (64). Also, with τ_j, it is used in eq (66)
SAMM	1061-01	The second five alphameric characters of the nuclide's I. D.
	0730	Recriprocal of 3 times the total nuclide density
	1023+01	The age to the Indium resonance (1.46ev)
SCAT	0601+01	The scattering per resonance atom
SIGA	1061+02	The σ_a^{th} for the n^{th} nuclide
SIGATH	0851+01	The σ_a^{th} for the mixture
SIGDEL	0500+06	The sum of $(\sigma_T \Delta)$ a.f. of the mix where a.f. is atom fraction
SIGS	1061+02	The σ_s^{th} of the n^{th} nuclide
SIGSTH	1066+04	The σ_s^{th} of the mixture
SIGT	1066+03	The σ_T^{th} of the n^{th} nuclide
SIGTT	0991+01	The sum of the product of the atom fraction times the σ_T^{th} of all nuclides except fuels
SIC9	0705	The total elastic scatter from group 9 for the n^{th} nuclide. See Appendix F
SIG10	0705+02	The same as SIG9 except the scatter is from group 10
SOURCE	1043+01	The fraction of the source in each of the 11 epithermal groups
SR	1005	The removal cross section
SSN	1009+01	Scattering matrix for (n,n*) and elastic (P0) cross sections. (a 12 x 13 matrix): Upper triangle (J, K+1) for (n,n*), diagonal and lower triangle (LS,LE) for elastic (P0)
STR	1004+02	The multigroup transport cross section, $\Sigma_T - \bar{\mu}\Sigma^1_{sj-j}$

Name	Location	Meaning or Use
SUM	0806+03	Summing variable for terms in the power series expansion of the Wilkins equation. See eq (100) and (102).
S2N	1009+01	Same as SSN except it is for (n,2n) and elastic (P1) values
T	1027+02	The group age τ_n. To use in eq (66)
TEMPTR	1060+01	The system temperature T, in degrees Kelvin
TERM	806+02	The individual terms, $a_n X^n$, in the solution to the Wilkins equation, eq (100) and (102)
TERMU	1061+02	$(1-\tilde{\mu})$ thermal of the n^{th} nuclide
TOTWT	1027+04	A summing variable for the summation term in eq (66)
TSIG9	705+01	The total scattering cross section from group 9 for the mix
TSIG10	705+03	Same as TSIG9 except it applies to group 10
U	1065+02	Subscripted variable for the upper lethargy (lower E) boundaries of the 11 epithermal groups
UTIL	502	The thermal utilization
VF	1061+02	The $\nu\sigma_f^{th}$ of the n^{th} nuclide
VPROB	0815	The most probable thermal neutron velocity
VBAR	854+01	The average thermal neutron velocity
X(K)	1027+03	The lethargy width of the n^{th} group
	603+01	Storage for the symbols used in the flux spectrum printout
XBAR	810+02	The average normalized thermal neutron velocity $\bar{x}$
XSAM	0997	Temporary storage for the cross section library
	1030	Storage for the fast and thermal fluxes prior to their output
	0698	Temporary storage for the grid for spectrum printout

Name	Location	Meaning or Use
	0120	This prevents computer underflow if the flux is too low. Note log (0) = $-\infty$
XTON	0806	x^{n-1} in equation (100) or (102)
	0806+01	x^{n} in equation (100) or (102)
Y	0503+08	The value of x corresponding to lethargy U = 23
	0809+02	The point x for the next Wilkins calculation: $x_2 = x_1 + x_1$ [exp (.125-1)]
	0810	The quantity (x-a) in equation no. (102)

Appendix B

Chain One Source Deck

```
C     THIS CARD ORIGINS THE PROGRAM AT ADDRESS 10970 IN MEMORY.  IN ORDER
C     TO DO THIS YOU MUST USE A SPECIAL SET OF SHORT SUBROUTINES.
$     JOB  OLD BARNYARD, CHAIN 1
$     ORIGIN 10970
C     STORING IN COMMON THOSE DATA USED IN CHAIN 2 OR IN SUBPROGRAMS
      COMMON LGP(13),KE,FLX(11),U(11),SOURCE(11),DENBR,NRNUC,TEMPTR,
     1XBAR,PSIGM,RIEFF,RES
      DIMENSION SA(11),STR(11),SR(11),XSAM(168),SSN(12,13),S2N(12,13),X(
     111),T(12)

C     READING IN THE FIRST TWO DATA CARDS
 1001 READ 1060,LGP
      PUNCH1060,LGP
 1060 FORMAT(13A5)
      READ 1067,NEW,NRNUC,DENBR,TEMPTR,RES,M
      PUNCH1065
 1065 FORMAT(//)
      TYPE1060,LGP

C     READING IN THE DECK OF SOURCES
      READ 1070,U
      K=0
      DO 1046 J=1,10
      IF(K)1045,1043,1045
 1043 READ1060,LGP
      READ1070,INDEX,SOURCE
 1070 FORMAT(12N)
      IF(INDEX-NEW)1046,1044,1046
 1044 K=1
      GO TO 1046
 1045 READ 1060,PU
      READ,PU
 1046 CONTINUE
      IF(NEW-10)1050,1049,1047

C     SETTING THE UNIT SOURCE IN ITS GROUP (IF APPLICABLE)
 1047 SOURCE(NEW-10)=1.
      LGP(10)=NEW-10
      PUNCH 1071,LGP
 1071 FORMAT(9A5,I3,3A5)
      GO TO 1048

C     READING IN THE USERS OWN SOURCE (IF APPLICABLE)
 1049 READ1070,SOURCE
 1050 PUNCH 1060,LGP
 1048 TYPE 600
 600  FORMAT(/16HBEGIN EPITHERMAL)
```

```
      PUNCH 1058,TEMPTR
 1058 FORMAT(/22HSYSTEM TEMPERATURE IS ,F6.1,7H KELVIN)
      PUNCH 1059
 1059 FORMAT(/18HATOM FRACTIONS ARE/)

C     CLEARING THE MATRICES OF ANY PREVIOUS PROBLEM
      DO 1000 J=1,11
      STR(J)=0.
 1000 SR(J)=0.
      DO 1002 J=1,12
      DO 1002 K=1,13
      SSN(J,K)=0.
 1002 S2N(J,K)=0.
      FF=1.
      E=1.

C     READING IN THE RESONANCE DATA CARD (IF APPLICABLE)
      IF(RES)995,994,995
  995 READ,JJJ,PU,RIEFF
      PU=PU/DENBR

C     (IF APPLICABLE)
C     READ THE F AND E FACTORS FOR HETEROGENOUS CELL CALCULATIONS
  994 IF(M)500,500,501
  501 READ,FF,E

C     SETTING UP THE DO LOOP FOR EACH OF THE NUCLIDES
  500 PROB=1.
      KKK=0
      PSIGF=0.
      PSIGM=0.
      PSIGO=0.
      SIGTT=.0
      SIGDEL=.0
      SIGSTH=0.
      DO 1020 I=1,NRNUC

C     READING IN THE NUCLIDE DATA CARDS
      READ 1061, SAM, SAMM, ATOMS, P, MOD, DENS
 1061 FORMAT(5X,3A5,10N)
      P=P/DENBR
      READ 1061,SAM,SAMM,ATOMS,LS,NA,LQ,PSI,SIGA,DELTA,SIGS,TERMU,VF,FIS
      PUNCH1066,SAM,SAMM,ATOMS,P
 1066 FORMAT(5X,3A5,F15.8)

C     CORRECTING THE THERMAL CROSS SECTIONS FOR TEMPERATURE AND
C     CALCULATING THE THERMAL SIGMA TOTAL FOR EACH NUCLIDE
      SIGA=SIGA*2.2E+05/(1.28E+04*SQRT(TEMPTR))
      DELTA=DELTA*2.2E+05/(1.28E+04*SQRT(TEMPTR))
      SIGT=SIGA+SIGS

C     CALCULATING TOTAL THERMAL SCATTERING SIGMA
      SIGSTH=SIGSTH+P*SIGS

C     CALCULATING THERMAL UTILIZATION AND ABSORPTION PARAMETER
      GO TO (993,992,996),MOD
  992 PSIGF=PSIGF+P*SIGA
```

```
      GO TO 997
  996 PSIGO=PSIGO+P*SIGA
      GO TO 991
  993 PSIGM=PSIGM+P*SIGA
  991 SIGDEL=SIGDEL+SIGT*P*DELTA
      SIGTT=SIGTT+P*SIGT

C     READING IN THE WHOLE CROSS SECTION LIBRARY FOR THE NUCLIDE
  997 READ 1067,(XSAM(J),J=1,LS)
 1067 FORMAT(6N)
      KZ=0
      KKZ=0

C     CHECKING FOR ABSORPTION
      IF(NA)1006,1006,1004
 1004 DO1005 J=1,11
      KZ=KZ+1
      STR(J)=STR(J)+XSAM(KZ)*P
 1005 SR(J)=SR(J)+XSAM(KZ)*P
 1006 KZ=KZ+1
      IN=XSAM(KZ)
      KZ=KZ+1
      IIN=XSAM(KZ)
      DO 1017 J=1,IN
      KK=J+IIN
      IF(KK-12)1008,1008,1007
 1007 KK=12
 1008 DO 1017 K=J,KK
      GO TO (1009,1010,1009,1011),LQ
 1009 KZ=KZ+1

C     INELASTIC CROSS SECTION, GROUP TO GROUP
      SSN(J,K+1)=SSN(J,K+1)+XSAM(KZ)*P
      GO TO 1012
 1010 KZ=KZ+1

C     N-2N CROSS SECTIONS, GROUP TO GROUP
      S2N(J,K+1)=S2N(J,K+1)+XSAM(KZ)*P
      GO TO 1012
 1011 IF(KKZ)2001,2000,2001
 2000 KZ=KZ+1
      KKZ=1
      GOTO2002
 2001 KZ=KZ+2

C     PUTTING ELASTIC (P0) AND ELASTIC (P1) IN THE BOTTOM OF SSN AND S2N
 2002 LS=13-J
      LE=13-K
      SSN(LS,LE)=SSN(LS,LE)+XSAM(KZ)*P
      S2N(LS,LE)=S2N(LS,LE)+XSAM(KZ+1)*P/3.
 1012 STR(J)=STR(J)+XSAM(KZ)*P
      IF(K-J)1016,1013,1016
 1013 GO TO (1017,1014,1017,1015),LQ
 1014 SR(J)=SR(J)-XSAM(KZ)*P
      GO TO 1017
 1015 STR(J)=STR(J)-XSAM(KZ+1)*P/3.
      GO TO 1017
```

```
 1016 SR(J)=SR(J)+XSAM(KZ)*P
 1017 CONTINUE
      GO TO(1018,1018,1019,700),LQ
 1018 LQ=4
      GO TO 1006
 1019 LQ=3
      GO TO 1006

C     READING THE FISSION CROSS SECTIONS USED IN CHAIN 2 INTO DUMMY STORAG
  700 IF(FIS)698,698,699
  699 READ 1067,(XSAM(J),J=41,51)

C     READING IN THE GRID BACKGROUND FOR THE SPECTRUM PRINTOUT
  698 READ 2222,(XSAM(J),J=41,120)
 2222 FORMAT(80A1)

C     CHECKING FOR RESONANCE CALCULATION
      IF(KKK)705,701,705
  701 IF(RES)704,1020,704
  704 KKK=1

C     CALCULATING PSI*SIGMA-S FOR THE MIXTURE
      TSIG9=0.
      TSIG10=0.
      PSIG9=0.
  705 SIG9=SSN(4,4)+SSN(4,3)+SSN(4,2)+SSN(4,1)-TSIG9
      TSIG9=SSN(4,4)+SSN(4,3)+SSN(4,2)+SSN(4,1)
      SIG10=SSN(3,3)+SSN(3,2)+SSN(3,1)-TSIG10
      TSIG10=SSN(3,3)+SSN(3,2)+SSN(3,1)
      PSIG9=PSIG9+PSI*P*(SIG9+SIG10)*.5
 1020 CONTINUE

C     ALL CROSS SECTIONS HAVE BEEN CALCULATED AT THIS POINT

      IF(KKK)720,730,720
  720 TYPE 601
  601 FORMAT(/15HBEGIN RESONANCE)

C     CALCULATING SCATTERING PER RESONANCE ATOM (IF APPLICABLE)
      SCAT=.5*(TSIG9+TSIG10)/PU

C     SELECTING THE EMPIRICAL EQUATION FOR RESONANCE INTEGRAL
      IF (RIEFF)728,721,728
  721 GO TO(722,725),JJJ
  722 IF(SCAT-4000.)723,723,724
  723 RIEFF=2.69*SCAT**.471
      GO TO 728
  724 RIEFF=EXPF(5.64-163./(SCAT**.65))
      GO TO 728
  725 IF(SCAT-4500.)726,726,727
  726 RIEFF=8.33*SCAT**.253
      GO TO 728
  727 RIEFF=70.

C     CALCULATING RESONANCE ESCAPE PROBABILITY
  728 PROB=EXPF(-PU*RIEFF/PSIG9)
```

```
C       ADJUSTING GROUP 9 AND 10 REMOVAL CROSS SECTION
C       FOR RESONANCE ESCAPE
        SR(9)=SR(9)+RIEFF*.1*PU/LOGF(61.442/1.1254)
        SR(10)=SR(10)+R EFF*.9*PU/LOGF(61.442/1.1254)
        STR(9)=SR(9).SS.    ,4)-S2N(4,4)
        STR(10)=STR(10)+SSN(3,3)-S2N(3,3)

C       HANSENS FINITE DIFFERENCE EQUATIONS FOR EPITHERMAL FLUX AND AGE
  730   SAMM=1./(3.*DENBR*DENBR)
        PUNCH 1051
 1051   FORMAT(//57X,14HRELATIVE GROUP/5HGROUP,7X,8HLETHARGY,7X,10HENERGY,
       1 2V,8X,6HSOURCE,6X,14HFLUX,   /UNIT U/)
        RHO=.0
        DO 1029 K=1,11
        LE=13-K
        DENS=SAMM/(STR(K)*SR(K))
        ATOMS=0.
        TOTWT=0.
        P=0.
        T(K)=0.
        FLX(K)=0.
        SA(K)=0.
        IF(K-1)1027,1027,1025
 1025   DO 1026 J=1,K-1
        LS=13-J
        SAM=FL)(J)*(SSN(LS,LE)+SSN(J,K+1)+2.*S2N(J,K+1))/SR(J)
        ATOMS=ATOMS+SAM
        TOTWT=TOTWT+SAM*T(J)
 1026   P=P+S2N(LS,LE)*SA(J)
 1027   FLX(K)=ATOMS+SOURCE(K)
        SA(K)=FLX(K)*DENS+P/STR(K)
        T(K)=(SA(K)+TOTWT)/FLX(K)
        X(K)=U(K)-RHO
        RHO=U(K)
 1029   CONTINUE
        F=SR(10)*4./FLX(10)

C       CALCULATING TOTAL FLUX PER GROUP
        DO 1023 K=1,11
 1023   FLX(K)=FLX(K)/SR(K)
        DO 1030 K=1,10
 1030   XSAM(40-K)=F*FLX(K)/X(K)

C       NOTICE THAT TOTAL FLUX PER GROUP HAS BEEN LEFT IN STORAGE IN FLX(11)

C       CALCULATING AGE TO INDIUM RESONANCE
        SAMM=(T(10)-T(9))*0.935+T(9)

C       CHECKING FOR MODERATOR IN THE SYSTEM

        IF(PSIGM)1024,108,1024
C       STARTING THE THERMAL CALCULATION
 1024   TYPE 602
  602   FORMAT(/13HBEGIN THERMAL)

C       THE ABSORPTION PARAMETER AND (1/1-F)
        UTIL=0.
```

```
      IF(PSIGF)502,503,502
  502 UTIL=1./(((PSIGM+PSIGO)/PSIGF)*FF+E)
  503 DELTA=SIGDEL/SIGTT*(1./(1.-UTIL))
      AREAX=.0
      PREFX=.0
      PREVF=0.
      II=0
      AN2=1.
      P=0.

C     ARRANGING FOR INCREMENTS OF 0.25 LETHARGY UNITS
      ETODU=EXP(.125)-1.

C     Y HERE EQUALS X.  SINCE X WAS SUBSCRIPTED ABOVE
      Y=EXP(-11.5)*SQRT(1.0E+12/(8.616*TEMPTR))
      I=0

C     DOING THE SERIES SOLUTION TO WILKINS EQUATION
      AN1=AN2*DELTA*.333333
      DSUM=AN1
  802 SUM=AN2+AN1*Y
      XTON=Y
      AN3=0.
      N=4
  803 N=N+1
      XN=N-3
      IF (P)805,804,805

C     THE RECURSION RELATION FOR X=0
  804 AN=(2.*(XN-2.)*AN2+DELTA*AN1)/(XN*(XN+2.))
      GO TO 806

C     THE GENERAL RECURSION RELATION
  805 AN=(AN1*(XN-1.)*(XN+1.-2.*P*P)-AN2*(4.*P*(XN-2.)+DELTA)-AN3*2.*(X
     1N-3.))/(P*XN*(1.-XN))
      AN3=AN2
  806 DTERM=XN*AN*XTON
      XTON=XTON*Y
      TERM=AN*XTON
      SUM=SUM+TERM
      DSUM=DTERM+DSUM
      AN2=AN1
      AN1=AN

C     CHECKING FOR CONVERGENCE OF THE SERIES
      IF(ABS(TERM/SUM)-1.0E-06)807,807,803
  807 IF(ABS(DTERM/DSUM)-1.0E-06)808,808,803
  808 P=Y+P

C     FINDING N(X) FOR THE POINT X=P
      F=EXP(-P*P)*SUM*P**2

C     LOCATING THE PEAK OF THE N(X) CURVE
      IF(II)816,814,816
  814 IF(F-PREVF)815,815,816

C     CALCULATING THE MOST PROBABLE VELOCITY
```

```
 815  VPROB=SQRT(1.6477E+04*TEMPTR)*P
      II=1
 816  PREVF=F

C     CALCULATING FLUX PER UNIT U WHICH IS X*X*N(X)
      F=F*P*P
      I=I+1

C     STORING THE VALUES OF THERMAL FLUX
      XSAM(I)=F

C     CALCULATING THE AREA UNDER THE XN(X) CURVE BY TRAPEZOIDAL RULE
      F=F/P
      AREAX=AREAX+(F+PREFX)*Y*.5
      PREFX=F
      IF(I-29)809,810,810
 809  AN2=SUM
      AN1=DSUM
      Y=P+P*ETODU
      Y=Y-P
      GO TO 802

C     CALCULATING THE AREA UNDER THE FLUX CURVE ANALYTICALLY
 810  C=2.*F*(1./(P*P)-1.)+P*P*EXP(-P*P)*DSUM
      AREA=(P*C+2.*(P*P-1.)*F/P)/DELTA

C     FINDING THE AVERAGE NORMALIZED VELOCITY
      XBAR=AREAX/AREA

C     NORMALIZING THE FLUXES AND MATCHING THE THERMAL AND EPITHERMAL
C     FLUXES IF A THERMAL CALCULATION WAS PERFORMED
      F=1./XSAM(29)
      DO 812 J=1,29
 812  XSAM(J)=F*XSAM(J)
      GO TO 109
 108  II=30
      TYPE 1028
 1028 FORMAT(36HTHERE IS NO MODERATOR IN THE SYSTEM./36HNO THERMAL CALCU
     1LATION WILL BE DONE.)
 109  F=0.

C     SETTING THE MAXIMUM FLUX TO 100
      DO 105 J=II,39
      IF(F-XSAM(J))106,106,105
 106  F=XSAM(J)
 105  CONTINUE
      DO 111 J=II,39
 111  XSAM(J)=XSAM(J)*100./F

C     OUTPUTTING THE FLUXES
      P=16.5
      DO 131 K=1,23
      IF(K-10)112,112,113
 112  KK=40-K
 1022 PUNCH 1052,K,U(K),1.0E+07*EXP(-U(K)),SOURCE(K),XSAM(KK)
      GO TO 131
 113  IF(PSIGM)115,131,115
```

```
 115  P=P+.5
      J=47-K-K
 813  PUNCH 1052,K,P,1.0E+07*EXP(-P),0.,XSAM(J)
 131  CONTINUE
 1052 FORMAT(I3,F15.2,3X,1P3E16.4)

C     OUTPUTTING THE THERMAL CONSTANTS
      PUNCH 850,SAMM
      IF(PSIGM)130,860,130
 130  PUNCH 851,T(10)
 850  FORMAT(//5X,35HAGE TO INDIUM RESONANCE (1.46EV) IS,1PE20.4,4H CM2)
 851  FORMAT(/5X,36HAGE TO ARBITRARY THERMAL (1.12EV) IS, 1PE19.4,4H CM2
     1)

C     CALCULATING THE DIFFUSION LENGTH SQUARED AND MIGRATION AREA
      SIGATH=PSIGM+PSIGF+PSIGO
      ELSQRD=.33333*SIGSTH*XBAR/(SIGATH*(DENBR*(SIGSTH+SIGATH/XBAR))**2)
      PUNCH 852,ELSQRD
 852  FORMAT(/5X,35HTHERMAL DIFFUSION LENGTH SQUARED IS,1PE20.4,4H CM2)
      EMSQRD=T(10)+ELSQRD
      PUNCH 853,EMSQRD
 853  FORMAT(/5X,23HTOTAL MIGRATION AREA IS,1PE32.4,4H CM2)
      PUNCH 854,VPROB
 854  FORMAT(/5X,41HMOST PROBABLE THERMAL NEUTRON VELOCITY IS,1PE14.4,6H
     1 M/SEC)
      VBAR =SQRT(1.6477E+04*TEMPTR)*XBAR
      PUNCH 855,VBAR
 855  FORMAT(/5X,35HAVERAGE THERMAL NEUTRON VELOCITY IS,1PE20.4,6H M/SEC
     1)
      PUNCH 856,DELTA
 856  FORMAT(/5X,23HABSORPTION PARAMETER IS,1PE32.4)

C     OUTPUTTING THE RESULTS OF THE RESONANCE CALCULATION
      IF(RES)859,860,859
 859  PUNCH 857,SCAT
 857  FORMAT(/5X, 32HSCATTERING PER RESONANCE ATOM IS,1PE23.4,6H BARNS)
      PUNCH 858,RIEFF
 858  FORMAT(/5X,31HEFFECTIVE RESONANCE INTEGRAL IS,1PE24.4,6H BARNS)
 860  PUNCH 861
 861  FORMAT(78X,1H-)

C     WARNING THE USER TO SET SWITCH FOR SPECTRUM OUTPUT
      TYPE 862
 862  FORMAT(/52HPUT SWITCH 1 ON FOR SPECTRUM PRINTOUT.  PRESS START.)
      PAUSE
      IF(SENSE SWITCH 1)3000,2999

C     THE SPECTRUM PUNCHOUT SUBPROGRAM
 2999 TYPE 863
 863  FORMAT(/28HNO SPECTRUM WILL BE PUNCHED.)
      GO TO 1091
 3000 TYPE 603
 603  FORMAT(/14HBEGIN SPECTRUM)

C     STORING THE SYMBOLS FOR THE GRAPH
      X(1)=1HX
      X(2)=1HL
```

```
      X(3)=1HH
      PUNCH 300
  300 FORMAT(69HPLOT OF SPECTRUM, IN FLUX PER UNIT LETHARGY VERSUS LETHA
     1RGY, FOLLOWS,//)
      PUNCH 301

C     ADJUSTING MINIMUM FLUX TO PREVENT COMPUTER UNDERFLOW AND
C     REDUCING FLUXES TO LOG SCALE 80 UNITS WIDE
      DO 3001 J=11,39
      IF(XSAM(J)-4.982E-03)120,3001,3001
  120 XSAM(J)=4.982E-03
 3001 XSAM(J)=((LOG(XSAM(J))/2.30259)+1.)*20.+.5
      KK=9
      I=0

C     ONE PASS THROUGH THE LOOP FOR EACH LINE OF OUTPUT
      DO 3005 J=11,93
      IF(KK)3011,3014,3011

C     CHECKING FOR EPITHERMAL OR THERMAL FLUXES
 3011 IF(J-29)3009,3009,3012
 3012 PU=J
      P =23.2-PU/4.

C     ADJUSTING FOR DIFFERENT WIDTH GROUPS
      IF(P-U(KK))3013,3013,3014
 3013 KK=KK-1
 3014 K=XSAM(39-KK)
      GO TO 3008
 3009 K=XSAM(J)
 3008 I=I+1

C     NUMBERS FOR THE LETHARGY SCALE
      LE=23-J/4

C     CHECKING FOR VALUES TOO LARGE FOR SCALE
      IF(K-79)3015,3016,3006
 3006 N=3
      K=79
      GO TO 3004

C     CHECKING FOR VALUES TOO SMALL  FOR SCALE
 3015 IF(K-3)3019,3017,3016
 3017 GO TO (3018,3016,3016,3016,3018),I
 3018 N=2
      K=4
      GO TO 3004
 3019 IF(K-2)3018,3017,3017
 3016 N=1
 3004 KZ=K+39
      KKZ=K+41
      GO TO(3002,3003,3003,3003,3002),I

C     OUTPUTTING THE SPECTRUM GRAPH
 3002 PUNCH 302,LE,(XSAM(LS),LS=43,KZ),X(N),(XSAM(LS),LS=KKZ,120)
  302 FORMAT(I2,78A1)
      I=1
```

```
      GO TO 3005
 3003 PUNCH 2222,(XSAM(LS),LS=41,KZ),X(N),(XSAM(LS),LS=KKZ,120)
 3005 CONTINUE
      PUNCH 301
 301  FORMAT(/2H.1,17X,1H1,18X,2H10,18X,3H100,15X,4H1000/)
 1091 TYPE 1092
 1092 FORMAT(/14HEND OF CHAIN 1//39HTO COLLAPSE CROSS SECTIONS LOAD CHAI
     1N 2/)

C     THIS PUNCH MAKES THE 407 LISTER ADVANCE A SHEET OF PAPER
      PUNCH 861

C     THIS PUNCH AVOIDS RUNNING OUT THE LAST PUNCHED OUTPUT CARD
      PUNCH 1065
      STOP
      END
$     EOJ
```

Appendix C

Glossary of Symbols Used in Chain Two

The following is a list of all variable names used in chain two of the program. Their location in the code is referenced by statement numbers, and their meaning or use in the code at that location is given. Some variable terms are carried over to chain two from chain one through the use of the COMMON statement at the beginning of the chains. See Appendix F for a sample of the scattering matrices.

Name	Location	Meaning or Use
A	0054-01	A summing variable for all the group to group elastic (P0) flux weighted cross sections ($\Sigma_i \sigma^{Po}_{j-k}$ for all j in the "scatter from" broad group, and all k in the "scatter to" broad group)
	1109+02	A summing variable for all the flux weighted fine group absorption cross sections in any given broad group
ATOMS	0701	The last five alphameric characters of the nuclides name
DENS	0701	A number indicating whether microscopic or macroscopic cross sections are desired: 1 = micro, 2 = macro
	0061	If macroscopic cross sections are desired, this variable is set equal to the nuclide density (atoms per barn-cm)
DELTA	0702	Thermal absorption parameter of the n^{th} nuclide. Not used in chain two calculations
FIS	0702	A code number: Are epithermal $\nu\sigma_f$ included for this nuclide: 0 = NO, 1 = YES
FLX	0003	The fraction of the total broad group flux that exists in each fine group within that broad group

Name	Location	Meaning or Use
I	0310	An integer number used to decrease the requested number of broad groups if the flux in a broad group is too low (done to prevent division by near zero quanities)
	0307	Index of the DO loop of SS 301-02 to SS 305. Set equal to 12 to immediately the DO loop if NBG is changed
	1108	Index to determine if thermal broad group will be output
	0205+01	Index of outermost of four DO loops. It indexes through the epithermal broad groups. For scatter from group n to group m it is the value of n
IIN	1006+01	The number of groups from which scatter occurs
IRES	1115	A code number: Is this a resonance nuclide: 0 = NO, 1 = YES
J	0601+01	A subscript to determine the boundaries of the broad groups
	0002-03	An index used in normalizing the fraction of the broad group flux that exists in each fine group
	1112	Index for clearing matrices
	0049	Index to put the cross section library into XSAM
	1004	Index to sort the σ_a into their proper SA
	1006+04	Index of the outer DO loop used to load the cross sections into the scattering matrices. It is the j index of σ_{j-k}
	1120	Index of the third of four nested DO loops. For scatter from broad group n to broad group m, it accounts for all fine groups in n
	4040	Index of the outer DO loop for broad group output
K	0604+03	Temporary storage for the requested number of broad groups

Name	Location	Meaning or Use
	0002-01	Index of inner DO loop to account for all fine groups in each broad group
	1008	Index of the inner DO loop used to load the cross sections into the scattering matrices. It is the k index of σ_{j-k}
	1122	Index of the fourth of four nested DO loops. For scatter from broad group n to broad group m, it accounts for all fine groups in m
	1109	Index of the inner DO loop for broad group output. It indexes through the fine groups within any given broad group
KK	1070+04	Both a gate and an index to identify the first fine group with a flux less than 10^{-12} (if any)
	1007-02	The group number of the lowest fine group, to which scatter occurs, from a given broad group
	1120+01	An index used with the variable k in SS 1122. It assists in proper scanning of the scattering matrices (Example: prevents scanning on or below the diagonal for (n,n*) or (n,2n) cross sections)
KKZ	1067+02	A gate to cause KZ to increment properly after reading in the first elastic (P0) and (P1) values into the scattering matrices: $\Delta KZ = 2$ for elastic scatter, $\Delta KZ = 1$ for all others
KZ	1067+01	A counter to sort the cross sections into the proper matrices
L	0205+02	Index of the second of four nested DO loops. For scatter from group n to group m, it is the value of m
LE	2002+01	Index used to convert the k index of σ_{j-k} for elastic (P0) and (P1) cross sections. These cross sections are read into the lower half of SSN and S2N
LGP	0604+01	A subscripted variable for the numbers of the fine groups that form the boundaries of the broad groups
LQ	0702	A code number: indicates type of inelastic scatter: 1 = (n,n*), 2 = (n,2n), 3 = both, 4 = neither

Name	Location	Meaning or Use
LS	0702	The total number of entries in the nuclide cross section library starting with the second library card (does not include epithermal $\nu\sigma_f$ of fission nuclides)
	2002	Index used to convert the j index of σ_{j-k} for elastic (P0) and (P1) cross sections. These cross sections are read into the lower half of the SSN and S2N matrices
M	1112-02	An address variable to cause the program to branch to the proper statement number after clearing the matrices. It is a statement number that can be varied. Used with GO TO M statements
MOD	0701	Is nuclide moderator, fuel, or other? Not used in chain 2 however it must be read in so that correct number is read into DENS
MXL	0604+04	The number of broad groups requested plus one. This allows for output of thermal group values
N	0004+01	Index of a DO loop (SS 0004+01 to SS 0055) used to read through all nuclides of the mixture
NA	0702	A code number: Is epithermal absorption appreciable: 0 = NO, 1 = YES
NBG	0604+01	The total number of broad groups for chain 2 output
NRNUC	0004+01	The total number of nuclides in the mixture. COMMON from chain 1
P	0701	Nuclide density in atoms per barn-cm
PSI	0702	The average logarithmic energy decrement ξ. Not used in chain 2
R	0054	A summing variable for the group to group elastic (P1) flux weighted cross sections. See use of A in SS 0054-01
RES	0704	A code indicating whether or not a resonance calculation was performed in chain 1. If yes (RES = 1) the resonance absorption will be included in calculating the absorption cross sections of the resonance nuclide. COMMON from chain 1

Name	Location	Meaning or Use
RIEFF	0700	The effective resonance integral of the resonance nuclide(s). COMMON from chain 1
SA	1005	Microscopic epithermal absorption cross section
SAM	0003-01	The sum of all fine group fluxes in a given broad group
	0701	The first five alphameric characters of the nuclide name
	1122+01	A summing variable for group to group (n,n*) flux weighted cross sections. See the use of A in SS 0054-01
	0315	Dummy storage for the void background cards
SAMM	0701	The second five alphameric characters of the nuclide name
	1122+02	A summing variable for group to group (n,2n) flux weighted cross sections. See the use of A in SS 0054-01
	1109+01	A summing variable for the broad group flux weighted $\nu\sigma_f$
SIGA	0702	Microscopic thermal absorption cross section
SIGS	0702	Microscopic thermal scattering cross section
SIGTR	0314+02	Thermal transport cross section: $[(1-\bar{\mu})\sigma_a+\sigma_a]$* DENS
SOURCE	0004	The fraction of the total source in each broad group
SR	0901	The broad group flux weighted removal cross section. The absorption contribution is added in SS 0045-01
SSN	1009+01	Scattering matrix (12 x 13) for (n,n*) and elastic (P0) cross sections. Upper triangle (J, k+1) is for (n,n*). Diagonal and lower triangle (LS, LE) is for elastic (P0)
STR	0054+02	The broad group flux weighted common transport cross section. The absorption contribution is added in SS 0045

Name	Location	Meaning or Use
S2N	1010+02	Scattering matrix (12 x 13) for (n,2n) and elastic (P1) cross sections. Upper triangle (J, k+1) is for (n,2n). Diagonal and lower triangle is for elastic (P1)
TERMU	0702	Thermal $(1-\overline{\mu})$ of the n^{th} nuclide
TT	0054+01	A summing variable for the flux weighted group to group total transfer cross sections. See the use of A in SS 0054-01
VF	0702	Thermal $\nu\sigma_f$ of the n^{th} nuclide
XBAR	1130	The average normalized thermal neutron velocity $\overline{x}$. COMMON from chain 1
XSAM	0049	Temporary storage for the entries of the nuclide cross section library

Appendix D

Chain Two Source Deck

```
C       THIS IS PART 2 OF OLD BARNYARD. CROSS SECTION COLLAPSING
$       JOB  OLD BARNYARD CHAIN 2
$       ORGIN 10970

C       NOTICE COMMON STATEMENT IS SAME AS IN CHAIN 1
        COMMON LGP(13),KE,FLX(11),U(11),SOURCE(11),DENBR,NRNUC,TEMPTR,XBAR
       1,PSIGM,RIEFF,RES
        DIMENSION SA(11),STR(11),SR(11),XSAM(200),SSN(12,13),S2N(12,13),
       1VSF(11)
        TYPE 604
 604    FORMAT(/16HBEGIN COLLAPSING)

C       READING IN NUMBER OF BROAD GROUPS AND THEIR BOUNDARY FINE GROUPS
        READ 1070,NBG,(LGP(J),J=2,NBG+1)
 1070   FORMAT(12N)

C       CHECK FOR ZERO FLUX AND REVISE THE GROUP STRUCTURE IF NECESSARY
        K=NBG
        MXL=NBG+1
        LGP(NBG+2)=12
        KK=0
        DO 305 J=1,LGP(NBG+1)
        IF(KK)301,301,305
 301    IF(FLX(J)-1.0E-12)303,305,305
 303    KK=J
        DO 305 I=2,KK
        IF(J-(LGP(I)+1))310,310,305
 310    NBG=I-1
 307    I=12
 305    CONTINUE
        IF(KK)308,308,309
 309    TYPE 1100,KK,NBG
 1100   FORMAT(/37HFLUX APPROXIMATELY ZERO IN FINE GROUP,I3,2X,27HTO PREVE
       1NT DIVISION BY ZERO,/39HWHEN FLUX WEIGHTING CROSS SECTIONS ONLY,I
       23,2X,28HEPITHERMAL BROAD GROUPS WILL,/,71HBE OUTPUT. THE FOLLOWING
       3FINE GROUP BOUNDARIES WILL BE USED ACCORDING TO,/,14HSENSE SWITCH
       41)
        TYPE 1101,(LGP(J),J=2,K+1),12
        TYPE 1102,(LGP(J),J=2,NBG+1),12
 1101   FORMAT(2HON,3X,12I3)
 1102   FORMAT(3HOFF,2X,12I3)
        TYPE 1103
 1103   FORMAT(30HWHERE 12 IS THE THERMAL GROUP.,/,31HSET SENSE SWITCH 1,P
       1PESS START.)
        PAUSE
        IF(SENSE SWITCH 1)1105,1106
 1106   LGP(NBG+2)=12
        K=NBG
        PSIGM=0.
```

```
      MXL=NBG+1
      GO TO 308
 1105 LGP(K+2)=12
      MXL=K+1
 308  LGP(1)=0
      IF(PSIGM)1107,1108,1107
 1108 I=0
      TYPE 1909
 1909 FORMAT(44HTHERE WILL BE NO THERMAL BROAD GROUP OUTPUT.)
      GO TO 1110
 1107 I=1
 1110 PUNCH 1111,MXL-1+I,NBG+I,(LGP(J),J=2,K+2)
      TYPE 1111,MXL-1+I,NBG+I,(LGP(J),J=2,K+2)
 1111 FORMAT(35HTHE TOTAL NUMBER OF BROAD GROUPS IS,I3,/,18HOUTPUT WILL
     1BE FOR,I3,2X,12HBROAD GROUPS,/,28HTHE BOUNDARY FINE GROUPS ARE,12
     2I3,/,30HWHERE 12 IS THE THERMAL GROUP.)

C     NORMALIZING THE FINE GROUP FLUXES IN EACH BROAD GROUP AND
C     ADDING UP THE SOURCE IN EACH BROAD GROUP
      DO 2 J=1,NBG
      SA(J)=0.
      DO 2 K=LGP(J)+1,LGP(J+1)
 2    SA(J)=SA(J)+FLX(K)
      DO 4 J=1,NBG
      SAM=0.
      DO 3 K=LGP(J)+1,LGP(J+1)
      SAM=SAM+SOURCE(K)
 3    FLX(K)=FLX(K)/SA(J)
 4    SOURCE(J)=SAM

C     ONE TRIP THROUGH THE DO LOOP FOR EACH NUCLIDE
      DO 55 N=1,NRNUC

C     CLEARING THE MATRICES AND SETTING SOME TERMS TO ZERO FOR LATER USE
      ASSIGN 704 TO M
      IRES=0
 1112 DO 1002 J=1,12
      DO 1002 K=1,13
      SSN(J,K)=0.
 1002 S2N(J,K)=0.
 1113 DO 1000 J=1,11
      STR(J)=0.
      SR(J)=0.
      SA(J)=0.
 1000 VSF(J)=0.
 1114 A=0.
      SAMM=0.
      SAM=0.
      R=0.
      TT=0.
      GO TO M

C     READING IN THE NUCLIDE DATA CARDS
 704  IF(RES)1115,701,1115
 701  READ 1061,SAM,SAMM,ATOMS,P,MOD,DENS
      GO TO 702
 1115 READ 1061,SAM,SAMM,ATOMS,P,MOD,DENS,IRES
```

```
 1061 FORMAT(5X,3A5,10N)
  702 READ 1061,SAM,SAMM,ATOMS,LS,NA,LQ,PSI,SIGA,DELTA,SIGS,TERMU,VF,FIS
      PUNCH 200,SAM,SAMM,ATOMS,P
  200 FORMAT(//11HNUCLIDE IS ,3A5,//25H     ITS NUMBER DENSITY IS,1PE11.4
     1,12H PER BARN-CM,//)

C     CHECKING FOR MICRO OR MACROSCOPIC CROSS SECTIONS
      IF(DENS-1.)61,62,61
   61 DENS=P
      PUNCH 202
  202 FORMAT(46HMACROSCOPIC CROSS SECTIONS, IN PER CM, FOLLOW.)
      GO TO 49
   62 PUNCH 201
  201 FORMAT(45HMICROSCOPIC CROSS SECTIONS, IN BARNS, FOLLOW.)

C     READING IN THE WHOLE CROSS SECTION LIBRARY FOR THE NUCLIDE
   49 READ 1067,(XSAM(J),J=1,LS)
 1067 FORMAT(6N)
      KZ=0
      KKZ=0

C     CHECKING FOR ABSORPTION
      IF(NA)1006,1006,1004
 1004 DO 1005 J=1,11
      KZ=KZ+1
 1005 SA(J)=SA(J)+XSAM(KZ)*FLX(J)*DENS
      IF(IRES)700,1006,700
  700 SA(9)=SA(9)+((RIEFF/LOGF(61.442/1.1254))*.1*FLX(9)*DENS)
     .SA(10)=SA(10)+((RIEFF/LOGF(61.442/1.1254))*.9*FLX(10)*DENS)
 1006 KZ=KZ+1
      IN=XSAM(KZ)
      KZ=KZ+1
      IIN=XSAM(KZ)
      DO 1017 J=1,IN
      KK=J+IIN
      IF(KK-12)1008,1008,1007
 1007 KK=12
 1008 DO 1017 K=J,KK
      GO TO (1009,1010,1009,1011),LQ
 1009 KZ=KZ+1

C     INELASTIC CROSS SECTION,GROUP TO GROUP
      SSN(J,K+1)=SSN(J,K+1)+XSAM(KZ)
      GO TO 1017
 1010 KZ=KZ+1

C     N-2N CROSS SECTIONS, GROUP TO GROUP
      S2N(J,K+1)=S2N(J,K+1)+XSAM(KZ)
      GO TO 1017
 1011 IF(KKZ)2001,2000,2001
 2000 KZ=KZ+1
      KKZ=1
      GO TO 2002
 2001 KZ=KZ+2

C     PUTTING ELASTIC (P0) AND ELASTIC (P1) IN THE BOTTOM OF SSN AND S2N
 2002 LS=13-J
```

```
      LE=13-K
      SSN(LS,LE)=SSN(LS,LE)+XSAM(KZ)
      S2N(LS,LE)=S2N(LS,LE)+XSAM(KZ+1)/3.
 1017 CONTINUE
      GO TO (1018,1018,1019,1020),LQ
 1018 LQ=4
      GO TO 1006
 1019 LQ=2
      GO TO 1006
 1020 PUNCH 205
 205  FORMAT(//7HSCATTER,34X,7HELASTIC,6X,7HELASTIC,7X,5HTOTAL/8HFROM  T
     1O,6X,9HINELASTIC,7X,4HN-2N,9X,4H(PO),9X,4H(P1),6X,8HTRANSFER/)

C     OUTPUTTING GROUP TO GROUP VALUES
      DO 53 I=1,NBG
      DO 53 L=I,MXL
      ASSIGN 1120 TO M
      GO TO 1114
 1120 DO 54 J=LGP(I)+1,LGP(I+1)
      KK=LGP(L)+1
      IF(KK-J)1121,1122,1122
 1121 KK=J
 1122 DO 54 K=KK,LGP(L+1)
      SAM=SAM+SSN(J,K+1)*FLX(J)*DENS
      SAMM=SAMM+S2N(J,K+1)*FLX(J)*DENS
      LS=13-J
      LE=13-K
      A=A+SSN(LS,LE)*FLX(J)*DENS
 54   R=R+S2N(LS,LE)*FLX(J)*DENS
      TT=TT+SAM+2.*SAMM+A
      STR(I)=STR(I)+SAM+SAMM+A-R
      IF(I-L)900,901,900
 901  SR(I)=SR(I)-SAMM
      GO TO 53
 900  SR(I)=SR(I)+SAM+SAMM+A
 53   PUNCH 103,I,L,SAM,SAMM,A,R,TT

C     THERMAL GROUP TO GROUP VALUES
      SIGS=SIGS*DENS
      IF(PSIGM)56,903,56
 56   PUNCH 103,MXL,MXL,0.,0.,SIGS,(1.-TERMU)*SIGS,SIGS
 103  FORMAT(I3,I5,5F13.5)

C     CALCULATING AND OUTPUTTING BROAD GROUP VALUES
 903  PUNCH 220
 220  FORMAT(//5HGROUP,5X,5HSIGTR,9X,4HSIGR,9X,4HSIGA,5X,6HNUSIGF,6X,
     16HSOURCE)
 421  IF(FIS)4040,4040,4041
 4041 READ 1067,(VSF(I),I=1,11)
 4040 DO 46 J=1,NBG
      ASSIGN 1109 TO M
      GO TO 1114
 1109 DO 45 K=LGP(J)+1,LGP(J+1)
      SAMM=SAMM+VSF(K)*FLX(K)
      A=A+SA(K)
      SR(J)=SR(J)+SA(K)
 45   STR(J)=STR(J)+SA(K)
```

```
 46   PUNCH 102,J,STR(J),SR(J),A,DENS*SAMM,SOURCE(J)

C     OUTPUTTING THERMAL BROAD GROUP VALUES
      IF(PSIGM)314,315,314
 314  VF=(VF/XBAR)*DENS
      SIGA=(SIGA/XBAR)*DENS
      SIGTR=(SIGS*TERMU+SIGA)
      PUNCH 102,MXL,SIGTR,SIGA,SIGA,VF
 102  FORMAT(I4,5F12.5)

C     READING AND IGNORING THE GRID BACKGROUND CARD IN EACH LIBRARY DECK
 315  READ 1061,SAM
 55   CONTINUE
      IF(PSIGM)1130,1093,1130
 1130 PUNCH 111,XBAR
 111  FORMAT(//48HMAXWELL-BOLTZMAN FACTOR = ),128,     AVERAGE X = ,N)
 1093 TYPE 1063
 1063 FORMAT(/15HEND OF PROGRAM./)
      PUNCH 1065
 1065 FORMAT(//1X)
      STOP
      END
$     EOJ
```

Appendix E

List of Neutron Source Deck

```
.5 1. 1.5 2. 2.5 3. 5. 8. 12. 16. 17.
THE SOURCE USED IS U235 FISSION (CRANBERG SPECTRUM).
2 .023023 .10824 .21044 .23139 .18048 .11483 .12439 .0072094 .0 .0 .0
THE SOURCE USED IS U233 FISSION.
3 .021495 .10472 .20811 .23202 .18252 .11674 .1267  .0073845 .0 .0 .0
THE SOURCE USED IS PU239 FISSION.
4 .024567 .11047 .21062 .22996 .17917 .11412 .1239 .0071986 .0 .0 .0
THE SOURCE USED IS PU241 FISSION.
5 .026852 .11615 .215 .22949 .11077 .1190 .0068574 .0 .0 .0
THE SOURCE USED IS CF252 FISSION.
6 .046142 .14786 .2292 .21936 .15472 .09531 .099137 .0055827 .0 .0 .0
THE SOURCE USED IS PO-BE (WHITMORE-BAKER).
7 .2342 .357 .261 .1025 .036 .0085 .0 .0 .0 .0 .0
THE SOURCE USED IS PO-BE (COCHRANE-HENRY).
8 .257 .369 .2514 .079 .028 .010 .0 .0 .0 .0 .0
THE SOURCE USED IS RA-BE (HILL).
9 .239 .365 .189 .109 .065 .033 .0 .0 .0 .0 .0
THE SOURCE USED IS YOUR OWN AS SHOWN BELOW.
10. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0.
THE SOURCE USED IS A UNIT SOURCE IN GROUP NO.
10 .0 .0 .0 .0 .0 .0 .0 .0 .0 .0 .0
```

Appendix F

Inelastic and Elastic (Po) Scattering Matrix

J ↓ K →	1	2	3	4	5	6	7	8	9	10	11	12	13
1		in 1-1	in 1-2	in 1-3	in 1-4	in 1-5	in 1-6	in 1-7	in 1-8	in 1-9	in 1-10	in 1-11	in 1-12
2	Po 11-12	Po 11-11	in 2-2	in 2-3	in 2-4	in 2-5	in 2-6	in 2-7	in 2-8	in 2-9	in 2-10	in 2-11	in 2-12
3	Po 10-12	Po 10-11	Po 10-10	in 3-3	in 3-4	in 3-5	in 3-6	in 3-7	in 3-8	in 3-9	in 3-10	in 3-11	in 3-12
4	Po 9-12	Po 9-11	Po 9-10	Po 9-9	in 4-4	in 4-5	in 4-6	in 4-7	in 4-8	in 4-9	in 4-10	in 4-11	in 4-12
5	Po 8-12	Po 8-11	Po 8-10	Po 8-9	Po 8-8	in 5-5	in 5-6	in 5-7	in 5-8	in 5-9	in 5-10	in 5-11	in 5-12
6	Po 7-12	Po 7-11	Po 7-10	Po 7-9	Po 7-8	Po 7-7	in 6-6	in 6-7	in 6-8	in 6-9	in 6-10	in 6-11	in 6-12
7	Po 6-12	Po 6-11	Po 6-10	Po 6-9	Po 6-8	Po 6-7	Po 6-6	in 7-7	in 7-8	in 7-9	in 7-10	in 7-11	in 7-12
8	Po 5-12	Po 5-11	Po 5-10	Po 5-9	Po 5-8	Po 5-7	Po 5-6	Po 5-5	in 8-8	in 8-9	in 8-10	in 8-11	in 8-12
9	Po 4-12	Po 4-11	Po 4-10	Po 4-9	Po 4-8	Po 4-7	Po 4-6	Po 4-5	Po 4-4	in 9-9	in 9-10	in 9-11	in 9-12
10	Po 3-12	Po 3-11	Po 3-10	Po 3-9	Po 3-8	Po 3-7	Po 3-6	Po 3-5	Po 3-4	Po 3-2	in 10-10	in 10-11	in 10-12
11	Po 2-12	Po 2-11	Po 2-10	Po 2-9	Po 2-8	Po 2-7	Po 2-6	Po 2-5	Po 2-4	Po 2-3	Po 2-2	in 11-11	in 11-12
12	Po 1-12	Po 1-11	Po 1-10	Po 1-9	Po 1-8	Po 1-7	Po 1-6	Po 1-5	Po 1-4	Po 1-3	Po 1-2	Po 1-1	

where $\overset{Po}{m-n}$ is the location of the inelastic (Po) scattering cross section from group m to group n in the matrix.

Figure 19: Inelastic and Elastic (Po) Scattering Matrix