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NEW BARNYARD: A MULTIGROUP NEUTRON CROSS SECTION CODE

THESIS

GNE/PHYS 69-8

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NEW BARNYARD: A MULTIGROUP NEUTRON CROSS SECTION CODE

THESIS

Presented to the Faculty of the School of Engineering of

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by

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First Lieutenant USAF

Graduate Nuclear Engineering

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PREFACE

Initially, my thesis project was to set up Gulf General Atomic's multigroup neutron cross section code, GGC-4, on an IBM 7094 digital computer. I was to understand the theory used in the code, be able to operate the code well enough to document its operation for others (produce a local user's manual) and compile a source book of neutron group cross sections for weapons problems.

I obtained the GGC-4 code and the GGC-4 cross section library (data for 45 nuclides) from the Air Force Weapons Laboratory where a CDC 6600 digital computer was used to copy the code and the cross section library onto magnetic tapes. I was unable to program the GGC-4 code on the IBM 7094 due to what appeared to be insufficient computer memory, but with the possibility of it being due to a"compatibility" problem with the CDC 6600 output tape being used on the IBM 7094. I was eventually successful in obtaining data for the 45 nuclides from the GGC-4 data tapes.

My thesis advisor, Dr. C. J. Bridgman, and I decided to write our own code for the IBM 7094. This code calculates the zero moment of the neutron flux which is used to flux weight the GGC-4 cross section data. This code was the bulk of my thesis work. In the course of this work, I was able to obtain two new ENDF/B neutron cross section data tapes from Oak Ridge National Laboratory. I was able to use these ENDF/B data tapes with minor modifications to my original

moments code.

A few words should be said about the title of my code, New Barnyard. The Resident School of Engineering at AFIT has a cross section code called Old Barnyard that calculates group cross sections for the energy range 10 MeV to thermal. Old Barnyard's cross section library was updated by using the cross section results from my code as the "new" input library to Old Barnyard. Thus, the idea occurred to me of calling my code New Barnyard.

I am indebted to several people in connection with this thesis: to my advisor, Dr. C. J. Bridgman, for guidance and counseling; to Lieutenant Robert Barry of the Air Force Weapons Laboratory, Kirtland, AFB, for help with the GGC-4 library tapes; to my classmates, Captains Jim Fisk and Robert Winchester, and Lieutenants Gary Knutson and Fred Damm, for many interesting discussions on transport theory, and to my typist, Mrs. Bobbie Thompson. Finally, I express special thanks to my wife, Cely, for her encouragement and good humor throughout my thesis study.

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LIST OF SYMBOLS

The following mathematical symbols and nomenclature are used in this study.

SYMBOL	MEANING
D	Diffusion Coefficient, cm
ev	Electron Volts
E	Energy, ev
$F\,[g(x)]$	Fourier Transform of $g(x)$
N	Number density of the mixture, Atoms per barn-cm (atoms x $10^{-24}/\text{cm}^3$)
$P_n P_1$	Legendre Polynomials
R	Reaction rate, events/cm^3-sec
S	Neutron source
x	Spatial dimension, cm
$\delta(x)$	Dirac delta function
θ_o^o	Zero moment of the neutron flux
θ_{on}^P	Pth derivative of the zero moment of the neutron flux in group n
μ	Cosine of the scalar scattering angle
σ	Microscopic cross section, barn
$d\Omega$	Differential solid angle

LIST OF SYMBOLS (Contd')

SYMBOL	MEANING
$\Sigma_s (E' \to E, \mu_o)$	Macroscopic differential scattering transfer cross section, cm^{-1}/steradian
$\Sigma_{sn}(E' \to E)$	Macroscopic scattering transfer cross section coefficient for scattering from energy E' to energy E associated with the n-th Legendre polynomial term in the expansion of the differential scattering transfer cross section, cm^{-1}
$\Sigma_{sn}(i \to j)$	Macroscopic scattering transfer cross section coefficient for scattering from group i to group j for the n-th Legendre polynomial term
PN $(i \to j)$	PN scattering transfer cross section which is equal to $(2n + 1)*$ Σ_{sn} $(i \to j)$ with N=n.
$Ⓗ^{(o)}$	Zero moment
Σ^K_{tr}	Macroscopic transport cross section for the K-th broad group
Σ^K_a	Macroscopic absorption cross section for the K-th broad group
Σ^K_t	Macroscopic total cross section for the K-th broad group
Σ^K_f	Macroscopic fission cross section for the K-th broad group
ν	Average number of fission neutrons per fission event
NNUK	Number of nuclides in the problem

LIST OF SYMBOLS (Contd')

SYMBOL	MEANING
NBBG	Number of broad groups in the problem
ϕ	Neutron flux, neutrons/cm^2-sec
$\bar{\mu}$	Average cosine of the scattering angle
χ_n	Fraction of the neutron source emitted in the n-th group

LIST OF TABLES

ABSTRACT

New Barnyard, a moments code, was written to calculate multigroup neutron macroscopic cross sections. Total, absorption, fission, and scattering transfer (group to group) cross sections can be calculated. The transport cross section, diffusion coefficient, and the average cosine of the scattering angle can also be calculated for each group. The energy range of these cross sections extends from 14.918 MeV to .4139 ev. Two versions of the moments code were written so that two different data sources could be used. With one version P0, P1, P2, and P3 elastic scattering transfer cross sections can be calculated for 22 broad groups, and with the other version P0 through P8 elastic scattering transfer cross sections can be calculated for 20 broad groups.

New Barnyard was written in the Fortran IV language for use on an IBM 7094 digital computer. The code calculates the zero moment of the neutron flux which is then used to flux weight basic neutron cross section data over the energy limits of each group. Unlike some other moments codes, no first and second moments of neutron flux are calculated, and no B_L, P_L, age, or resonance calculations are performed. These calculations were excluded to achieve simplicity and speed of calculation. New Barnyard uses the GGC-4 cross section library as well as the new ENDF/B data from ORNL (Oak Ridge National Laboratory).

Excellent agreement was found when the results of New Barnyard were compared with results from other cross

ABSTRACT (Contd')

section codes. Results from the two versions of New Barnyard showed relative differences generally much less than 10% for comparisons between absorption and between total cross sections. Larger relative differences were found in the comparisons of the PN scattering transfer cross sections due mainly to the fact that the number of terms used in the Legendre polynomial expansion of the differential scattering cross sections for each set of data on the two data tape sources were different.

NEW BARNYARD: A MULTIGROUP NEUTRON CROSS SECTION CODE

I. Introduction

A need exists for multigroup neutron cross sections for weapons problems. Although many sets of group cross sections exist, most are based on reactor spectra and consequently are of limited use from about 2 to 15 MeV, the principle region of interest for weapon physics.

Purpose and Method

The purpose of this study is to produce a code that will calculate group neutron cross sections in this higher neutron energy range. The computer code described in this study calculates group neutron cross sections and related constants for the energy range .4139 ev to 14.918 MeV. It is written in Fortran IV and has been executed on an IBM 7094 computer. The cross sections are determined by first calculating the energy dependent flux in an infinite homogeneous mixture of the isotopes or compounds specified. This flux is then used to flux weight cross section data. The input cross section data is obtained from 99 group libraries of fast cross sections. The energy dependent flux is calculated by the method of moments.

Sequence of Development

The basic theory for the numerical equations used in New Barnyard is given in Chapter II. The cross section data tapes, the numerical equations, and the special output features of New Barnyard are presented in Chapter III. The operating instructions for using New Barnyard on an IBM 7094 are given in Chapter IV. In Chapter V, results from other cross section codes are given and then the conclusions reached on the validity and accuracy of the code are stated.

II. Theory

In this chapter a brief review of the theory of flux weighting is given followed by a theoretical discussion of the calculation of the zero moment of the neutron flux which is used to flux weight cross sections in New Barnyard.

Flux Weighting

The reaction rate R (events/cm^3 sec) in the presence of polyenergetic neutrons is

$$R = \int_0^\infty \phi(E)\Sigma(E)\,dE \tag{1}$$

where E is the energy,

$\phi(E)$ is the energy dependent flux in ${}_0n^1/cm^2$-sec,

$\Sigma(E)$ is the energy dependent macroscopic cross section in cm^{-1}.

The total flux is, by definition

$$\phi = \int_0^\infty \phi(E)\,dE \tag{2}$$

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

$$R = \overline{\Sigma}\phi \tag{3}$$

where $\overline{\Sigma}$ is some average macroscopic cross section over the energy range of interest. It follows, by equating the reaction rates, i.e., the right hand sides of equations (1) and (3) and substituting from (2), that this average cross section must be given by

$$\overline{\Sigma} = \frac{\int_0^\infty \Sigma(E)\ \phi\ (E)\ dE}{\int_0^\infty \phi\ (E)\ dE} \tag{4}$$

or, in other words, the average cross section is a flux weighted average of the energy dependent cross section.

Equation (4) can be generalized to any range (limits on the integral), say from 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

$$\Sigma^n = \frac{\int_{E_n^-}^{E_n^+} \Sigma(E)\ \phi\ (E)\ dE}{\int_{E_n^-}^{E_n^+} \phi\ (E)\ dE} \tag{5}$$

where Σ^n is the group cross section for the n-th energy group,

E_n^- is the lower energy boundary of the n-th group,

E_n^+ is the upper energy boundary of the n-th group.

From inspection of equation (5) it is seen that the variation of neutron flux with energy must be known in order to determine group cross sections.

<u>Zero Moment of the Neutron Flux</u>

The flux calculations performed in this code are in solution to the energy dependent Boltzmann transport equation for an above-thermal energy region (14.918 MeV to

.4139 ev). The energy dependent flux for this energy range is calculated by the method of moments. A brief review of the zero moment of the neutron flux will be discussed here. A more detailed analysis of the method of moments can be found in AFIT Technical Report 67-17 (Ref 1).

The steady state Boltzmann equation written for an infinite, non-multiplying homogeneous medium which scatters and absorbs neutrons is

$$\mu\frac{\partial\phi}{\partial x}(X,E,\mu) + \Sigma_t(E)\phi(X,E,\mu) = S(X,E,\mu) + \int d\Omega' \int_0^\infty dE\ \Sigma_s(E'\rightarrow E,\mu_0)\phi(X,E',\mu') \tag{6}$$

Where X is the spatial positions of the neutrons

E is the neutron's energy,

μ_0 is the cosine of the scalar angle, cosine θ_0, through which a neutron is scattered,

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

$\phi(X,E,\mu)$ is the neutron flux in $_0n'/cm^2$-sec-steradian-unit energy

Σ_t (E) is the energy dependent macroscopic total cross section in cm^{-1},

$\Sigma_s(E'\rightarrow E,\mu_0)$ is the energy dependent macroscopic differential scattering transfer cross section for a neutron with an initial energy E' and an initial direction μ' that scatters into a unit energy interval about E and within a unit solid angle about μ in cm^{-1}/steradian.

In the above-thermal energy region **the target nuclei** motion may be neglected with respect to the neutron energies. **The** moments method assumes that an isotropic source consisting of a plane of infinite area is located at the coordinate position x = 0, i.e.,

$$S\,(X,E,\mu) = \frac{S(E)}{4\pi}\,\delta(X) \tag{7}$$

Where $\delta(X)$ is the Dirac data function at X = 0. The flux in equation (6) is expanded in terms of Legendre polynomial as

$$\phi(X,E,\mu) = \sum_{m=o}^{\infty} \frac{2m+1}{4\pi}\,\phi_m(X,E)P_m(\mu) \tag{8}$$

Similarily, the differential scattering transfer cross section is expressed as

$$\Sigma_s(E'\rightarrow E,\mu_o) = \sum_{n=o}^{\infty} \frac{2n+1}{4\pi}\,\Sigma_{sn}(E'\rightarrow E)P_n(\mu_o) \tag{9}$$

The coefficients, $\Sigma_{sn}(E'\rightarrow E)$, in the polynomial expansion in equation (9) are referred to as PN scattering transfer cross sections where PN is given by

$$PN(E' \rightarrow E) = (2n+1) \Sigma_{sn}(E' \rightarrow E) \tag{10}$$

where N = n.

The substitution of the source given by equation (7) and of the expansions (8) and (9) into equation (6) yields

$$\sum_{m=0}^{\infty} \left\{ \frac{2m+1}{4\pi} \frac{\partial\phi(X,E)}{\partial X} \mu P_m(\mu) + \frac{2m+1}{4\pi} \Sigma_t(E) \phi_m(X,E) P_m(\mu) = \frac{S(E)}{4\pi} \delta(X) + \right.$$

$$\left. \int d\Omega' \int_0^{\infty} dE' \sum_{n=0}^{\infty} \left[\frac{2n+1}{4\pi} \Sigma_{sn} (E' \rightarrow E) P_n(\mu_0) \right] \left[\frac{2m+1}{4\pi} \phi_m(X,E') P_m(\mu') \right] \right\} \tag{11}$$

Equation (11) is simplified by using some of the special properties of Legendre polynomials (Ref 2: 115) to obtain

$$\sum_{m=0}^{\infty} \left\{ \frac{\partial\phi_m(X,E)}{\partial X} \left[(m+1) P_{m+1}(\mu) + m P_{m-1}(\mu) \right] + \right.$$

$$(2m+1) \Sigma_t(E) \phi_m(X,E) P_m(\mu) = S(E)\delta(X) +$$

$$(2m+1) P_m(\mu) \int_0^{\infty} dE' \; \Sigma_{sm}(E' \rightarrow E) \phi_m(X,E') \tag{12}$$

Equation (12), a single equation with an infinite number of terms, is transformed into an infinite number of coupled equations, each with a finite number of terms, by operating on equation (12), term by term, with

$$\int_{-1}^{+1} P_{\ell}(\mu) \langle \text{eq. (12)} \rangle d\mu \quad \ell = 0,1,2,\ldots$$

The results for $\ell = 0$ are:

$$\frac{\partial \phi_1(X,E)}{\partial X} + \Sigma_t \phi_o(X,E) = \int_o^\infty \Sigma_{so}(E' \longrightarrow E)\phi_o(X,E')dE' + S_o(E)\delta(X) \tag{13}$$

Equation (13) is the basis for the calculation of the zero moment of the neutron flux.

The basic energy dependence of the flux is expressed as the volume-angle integral of the flux, i.e.,

$$Ⓗ^{(o)}(E) = \int_{-\infty}^{+\infty} dX \int d\Omega \ \phi(X,E,\mu) \tag{14}$$

or integrating over $d\Omega$ to give

$$Ⓗ^{(o)}(E) = \int_{-\infty}^{+\infty} dX \ \phi_o(X,E) \tag{15}$$

where $\phi_o(X,E)$ is the all angle flux. Equation (15) is the zero moment of the spatial distribution of the total (all angle) flux. This "zero moment flux" can be calculated exactly by using some properties of the Fourier integral transform (Ref 3) on the spatial variable in equation (13), i.e.,

$$F\left\{ \phi_\ell(X,E) \right\} = \int_{-\infty}^{+\infty} \phi_\ell(X,E)e^{-ipx} \, dx \equiv \theta_\ell(P,E) \tag{16}$$

It can be seen that if equation (16) is written for $\ell=0$ and the special case $P = 0$, then it is equivalent to the zero moment of equation (15), i.e.,

$$\Theta^{o}(E) = \Big[\theta_o(P,E)\Big]_{P=0} \qquad (17)$$

Therefore, equation (13) is Fourier transformed using the properties (Ref 3)

$$F\left\{\frac{\partial\phi_o(X,E)}{\partial X}\right\} = iP\ \theta_o(P,E) \qquad (18)$$

and

$$F\left\{\delta(X)\right\} = 1$$

To obtain

$$iP\ \theta_1(P,E) = \Sigma_t(E)\theta_o(P,E) = S(E) + \int_0^{\infty}\Sigma_{so}(E'\rightarrow E)\theta_o(P,E')dE' \qquad (19)$$

Equation (19) is valid for any arbitrary P but from equation (17) the zero moment flux is obtained for $P=0$; therefore, Maclaurin expansions for $\theta_o(P,E)$ and $\theta_1(P,E)$ are chosen as

$$\theta_o(P,E) = \theta_o^o(E) + (-iP)\theta_o^1(E) + \frac{(-iP)^2}{2!}\theta_o^2(E) + \ldots \qquad (20)$$

$$\theta_1(P,E) = \theta_1^o(E) + (-iP)\theta_1^1(E) + \frac{(-iP)^2}{2!}\theta_1^2(E) + \ldots \qquad (21)$$

Where the argument of the Maclaurin expansion is (-iP) rather than P. The primes denoting differentiation in the expansion have been replaced by superscripts. That is,

$$\theta_o''(E) = \theta_o^2(E) \qquad (22)$$

Under this notation the zero moment flux is symbolized

$$\Theta^{o}(E) = \theta_{o}^{o}\ (E) \tag{23}$$

Which is the energy dependent flux of interest.

Davison and Sykes, (Ref 4:343) 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

$$\int_{-\infty}^{\infty} X^{n}\ \phi_{\ell}\ (X,E)\,dX \neq 0 \tag{24}$$

only when both n and ℓ are both even or both odd. Thus the Maclaurin expansions become

$$\theta_{o}(P,E) = \theta_{o}^{o}(E) + \frac{(-iP)^{2}}{2!}\ \theta_{o}^{2}(E) + \ldots \tag{25}$$

$$\theta_{1}(P,E) = (-iP)\theta_{1}^{1}(E) + \frac{(-iP)^{3}}{3!}\ \theta_{1}^{3}(E) + \ldots \tag{26}$$

By substituting equations (25) and (26) into equation (19) and equating like powers of $(-iP)^{o}$ the zero moment equation is obtained

$$\Sigma_{t}(E)\theta_{o}^{o}(E) = S(E) + \int_{o}^{\infty} \Sigma_{so}(E' \rightarrow E)\theta_{o}^{o}(E')\,dE' \tag{27}$$

where $\theta_{o}^{o}(E)$ is the zero moment of the neutron flux and is the energy dependent flux necessary to calculate the group cross sections.

Solution of the Zero Moment Equation

Equation (27) is solved numerically by expressing it in multigroup notation. In group form it is

$$\int_{E_n^-}^{E_n^+} \Sigma_t(E)\theta_o^o(E)\,dE = \int_{E_n^-}^{E_n^+} S(E)\,dE + \int_{E_n^-}^{E_n^+} dE \int_0^{\infty} \Sigma_{so}(E' \to E)\theta^o(E')\,dE' \tag{28}$$

From the definition of a flux weighted group cross section, equation (5), it is seen that the first term becomes

$$\int_{E_n^-}^{E_n^+} \Sigma_t(E)\theta_o^o(E)\,dE = \Sigma_t^n\,\theta_{on}^o \tag{29}$$

Where Σ_t^n is the macroscopic total cross section of the n-th energy group, and θ_{on} is the total group flux. Similarly the second term is

$$\int_{E_n^-}^{E_n^+} S(E)\,dE = \chi_n \tag{30}$$

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

$$\int_{E_n^-}^{E_n^+} dE \int_o^{\infty} \Sigma_{so}(E' \to E)\theta_o^o(E')\,dE' \tag{31}$$

is considered as a group of n double integrals where the integral from 0 to ∞ on dE' has been broken into n finite intervals; $\Delta E'_1, \Delta E'_2, \Delta E'_3$, ... over the energy range .4139 ev to 14.918 MeV. It is important that the integration on the variable E be performed first since the limits of integration on E, i.e., E_n^- to E_n^+, are actually functions of E' and the maximum ΔE of the scattering nucleus. The last term becomes

$$\int_{E_n^-}^{E_n^+} dE \int_{\Delta E'_1} \Sigma_{so}(E' \rightarrow E)\, \theta_o^o(E')\, dE' +$$

$$\int_{E_n^-}^{E_n^+} dE \int_{\Delta E'_2} \Sigma_{so}(E' \rightarrow E)\, \theta_o^o(E')\, dE' \ldots \int_{E_n^-}^{E_n^+} dE \int_{\Delta E'_n} \Sigma_{so}(E' \rightarrow E)\, \theta_o^o(E')\, dE' \tag{32}$$

Each double integral, say the j-th, in equation (32) is a measure of the neutron scatter transport from the j-th to the n-th group. Recalling the definition of a flux weighted cross section in a slightly different form

$$\Sigma_{so}(j \rightarrow n) = \frac{\displaystyle\int_{E_n^-}^{E_n^+} dE \int_{E_j^-}^{E_j^+} \Sigma_{so}(E' \rightarrow E)\, \theta_o^o(E')\, dE'}{\displaystyle\int_{E_j^-}^{E_j^+} \theta_o^o(E')\, dE'} \tag{33}$$

It is seen that equation (32) can be expressed as

$$\sum_{j=1}^{n} \Sigma_{so}(j\rightarrow n)\theta^{o}_{oj} \tag{34}$$

Substituting equations (29), (30), and (34) into question (28) and rearranging terms yields the numerical zero moment equation,

$$\left[\Sigma_{t}^{n} - \Sigma_{so}(n\rightarrow n)\right] \theta^{o}_{on} = \chi_{n} + \sum_{j=1}^{n-1} \Sigma_{so}(j\rightarrow n)\theta^{o}_{oj} \tag{35}$$

The total group flux, θ^{o}_{on}, is determined by solving equation (35).

New Barnyard reads in a 99 group cross section set (or sets) from a cross section library tape, which is discussed in Chapter III, calculates a 99 group macroscopic cross section set, **then flux calculation is begun with** group one (highest energy group) where the scatter in term is zero. Calculation proceeds consecutively through all remaining groups down to .4139 ev. It is assumed that no "scatter up" in the neutron's energy occurs in this above-thermal energy region. Once the total flux has been calculated for each of the 99 groups, it is used to flux weight the 99 group macroscopic cross section set over each of the desired broad group limits.

III. The Code

In this chapter a few comments about New Barnyard are made followed by the contents and structure of the two data tape sources. Next, the numerical equations used to calculate the group cross sections and related constants are given and finally the special features of the output are mentioned.

New Barnyard is written in Fortran IV for an IBM 7094 digital computer. Two source decks are available in order to use the GGC-4 cross section data tape (Ref 5) and the two ENDF/B cross section data tapes (Ref 6). The GGC-4 source deck (i.e., the deck that uses the GGC-4 data tape) and the ENDF/B source deck are listed in appendices A and B. Extra comment cards have been positioned throughout the source decks to make them easier to read. Appendix C lists the variables and the meaning or use of each variable used in the two source decks. The code reads in cross section data and then determines the zero moment of the neutron flux which the code uses to calculate broad group cross sections and other related constants over the energy range .4139 ev to 14.918 MeV.

The Cross Section Data Tapes

As mentioned earlier, either the GGC-4 data tape or the ENDF/B data tapes can be used in New Barnyard. Both data tape sources have been compiled recently (1966-1967) and the results for absorption or total cross sections obtained from the two source decks for the same problem are within approximately 5 to 10% of each other. The ENDF/B data is updated from time to time by R.S.I.C. (The Radiation Shielding Information Center) at Oak Ridge National Laboratory. At this writing they are also preparing data for nuclides other than those that are contained on the ENDF/B data tapes described here. New data tapes can be obtained through R.S.I.C.*

Both data tape sources contain total, absorption, fission** and scattering transfer microscopic cross sections for 99 energy groups. The 99 "fine" group structure is shown in Appendix D. The P0 transfer cross sections on the GGC-4 tape are listed separately as elastic, inelastic, n-2n, and total transfer. The ENDF/B tapes list only a P0 transfer cross section which is the sum of the elastic, inelastic and 2(n-2n) transfer cross sections. The n-2n

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* ENDF/B data tapes are obtained from R.S.I.C, Oak Ridge National Laboratory, Post Office Box X, Oak Ridge, Tennessee, 37830

** The ENDF/B data tapes list $\nu\sigma_f$ for 99 groups. The GGC-4 data tape list ν and σ_f separately.

transfer cross section is multiplied by 2 to conserve neutrons. It is assumed that both neutrons are emitted in the same energy group. The GGC-4 tape has P0 through P3 elastic scattering transfer cross sections. The ENDF/B tapes have P0 through P8 elastic scattering transfer cross sections; however, the P0, as mentioned above, also includes inelastic and n-2n transfer cross sections. Both data sources include only P0 transfer cross sections for the n-2n and inelastic reactions. The GGC-4 tape also contains other information such as the 99 energy group boundaries, fission source spectrums, one dimensional cross section arrays (for (n,α), (n,γ) and other similar reactions) and resonance data. The 99 group fission source spectrums for several nuclides are listed in Appendix F.

The GGC-4 tape has data for the following nuclides*:

1. Hydrogen	7. Boron(natural)
2. Deuterium	8. Boron-10
3. Helium	9. Carbon
4. Lithium-6	10. Nitrogen
5. Lithium-7	11. Oxygen
6. Beryllium	12. Sodium

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* It was mentioned that data for 45 nuclides was on the GGC-4 data tape; this is true but data for some nuclides is repeated. This repeated data was obtained by Gulf General Atomic from other sources.

13. Magnesium
14. Aluminum
15. Silicon
16. Sulfur
17. Calcium
18. Titanium
19. Chromium
20. Manganese
21. Iron
22. Cobalt
23. Nickel
24. Copper
25. Molybdenum
26. Cadmium
27. Tungsten(natural)
28. Tungsten-180
29. Tungsten-182
30. Tungsten-183
31. Tungsten-184
32. Tungsten-186
33. Lead
34. Uranium-233
35. Uranium-235
36. Uranium-238
37. Plutonium-241

The ENDF/B tapes have data for the following nuclides:

1. Hydrogen
2. Deuterium
3. Lithium-6
4. Lithium-7
5. Beryllium
6. Boron-10
7. Carbon
8. Nitrogen
9. Oxygen
10. Sodium
11. Magnesium
12. Aluminum
13. Titanium
14. Chromium
15. Manganese
16. Iron
17. Nickel
18. Tungsten-182
19. Tungsten-183
20. Tungsten-184
21. Tungsten-186
22. Vanadium
23. Uranium-235
24. Uranium-238
25. Plutonium-238
26. Plutonium-239
27. Plutonium-240
28. Plutonium-241
29. Plutonium-242

Tables XIII and XIV show the structures of the data tapes and Table XV lists comments about these tapes. Appendix E contains these three tables.

Each of the 99 group cross section sets contained in the two data tape sources was calculated by flux weighting energy dependent cross sections with a 1/E flux dependence over the limits of each fine group. The specific calculations of these 99 group cross section sets can be found in the description of the GGC-4 code (Ref 5).

Since repeated reference is made to PN (P0, P1, etc.) scattering transfer cross sections, a review of what is meant by the PN elastic scattering transfer cross sections is included here. The differential scattering transfer cross sections for elastic scattering of neutrons, $\sigma_S(E' \to E, \mu_o)$ are experimentally determined and are usually tabulated as a plot of $\sigma_S(E' \to E, \mu_o)$ versus μ_o where μ_o is the cosine of the scattering angle. BNL 400 (Ref 7) shows typical differential scattering transfer cross section curves for nuclides with Z numbers from 1 to 22. In order to cut down on the amount of data that would have to be retained for each of these curves, a Legendre polynomial expansion of the differential scattering transfer cross section is performed as follows:

$$\sigma_S(E' \to E, \mu_o) = \sum_{k=o}^{K} \frac{2n+1}{4\pi} \sigma_{sk}(E' \to E) P_k(\mu_o) \qquad (36)$$

$P_k(\mu_o)$ is the k-th Legendre polynomial and $\sigma_{sk}(E' \to E)$ is the k-th scattering transfer cross section coefficient. These coefficients, $\sigma_{sk}(E' \to E)$, can be calculated by equating equation (36) to the experimental values of $\sigma_S(E' \to E, \mu_o)$. The PN fine group scattering transfer cross sections are then calculated by weighting coefficients $\sigma_{sk}(E' \to E)$ with $1/E'$ over the limits of each fine group; that is,

$$PN(i \rightarrow j) = \frac{(2k+1)\int_{E_{i-1}}^{E_i}\int_{E_{j-1}}^{E_j} \sigma_{sk}(E' \rightarrow E)\,\frac{1}{E'}\,dE'\,dE}{\int_{E_{i-1}}^{E_i}\frac{dE'}{E'}} \tag{37}$$

$k = 0, 1, 2, \ldots K$

$i = 2, 3, \ldots 99$

$j = 2, 3, \ldots 100$

where N = k. This polynomial expansion of $\sigma_s(E' \rightarrow E, \mu_o)$ can lead to negative cross section values when the number of terms (K+1) used in the expansion is small. Thus, there is an advantage of having the ENDF/B data since it has P0 through P8 elastic scattering transfer cross section coefficients.

<u>The Numerical Equations Used in the Code</u>

After the code reads the 99 group set (or sets) of microscopic cross sections specified for a particular problem, it calculates a 99 group set of macroscopic cross sections using the relation

$$\Sigma^n = \sum_{i=1}^{NNUK} N_i\,\sigma_i^n \tag{36}$$

where σ_i is the microscopic cross section in barns for the i-th nuclide and the n-th fine group, N_i is the number density in nuclei/cm-barn for the i-th nuclide, NNUK is the number of nuclides for the problem. The code then uses the fine group total macroscopic cross sections and the fine group P0 total transfer macroscopic cross sections and calculates the total flux for each of the 99 groups from the zero moment equation (eq. (35)) which is repeated here for convenience:

$$\theta_{on}^{o} = \frac{1}{(\Sigma_t^n - \Sigma_{so}(n \rightarrow n))} \left[\chi_n + \sum_{j=1}^{n-1} \Sigma_{so}(j \rightarrow n)\theta_j^o \right] \qquad (35)$$

$n=1, 2, 3, \ldots 99$

Next, the code flux weights the 99 group macroscopic cross section set over the limits of each of the broad groups. The specific cross sections calculated along with the numerical equations used to calculate them are listed below: (In the following equations, the subscripts n and j denote fine groups (any one of the 99 groups) and K and L denote broad groups.)

1. Absorption cross section

$$\Sigma_a^K = \frac{\sum_{n \in K} \theta_{on}^{o} \Sigma_a^n}{\sum_{n \in K} \theta_{on}^{o}} \qquad K = 1, 2, 3, \ldots \text{NBBG}$$

2. Total cross section

$$\Sigma_t^K = \frac{\sum\limits_{n\in K} \theta_{on}^o \Sigma_t^n}{\sum\limits_{n\in K} \theta_{on}^o} \qquad K=1,\ 2,\ 3,\ \ldots NBBG$$

3. Fission cross section

$$\Sigma_f^K = \frac{\sum\limits_{n\in K} \theta_{on}^o \Sigma_f^n}{\sum\limits_{n\in K} \theta_{on}^o} \qquad K=1,\ 2,\ 3,\ \ldots NBBG$$

4. Nu* (Fission cross section)

$$(\nu \Sigma_f)^K = \frac{\sum\limits_{n\in K} \theta_{on}^o \nu^n \Sigma_f^n}{\sum\limits_{n\in K} \theta_{on}^o} \qquad K=1,\ 2,\ 3,\ldots NBBG$$

5. Scattering transfer cross section

$$PN(K\rightarrow L) = \frac{\sum\limits_{n\in K} \theta_{on}^o \sum\limits_{j\in L} PN(n\rightarrow j)}{\sum\limits_{n\in K} \theta_{on}^o} \qquad K=1,\ 2,\ 3,\ldots NBBG$$

Where NBBG is the number of broad groups,

θ_{on}^o is the total flux for the n-th group,

$\sum\limits_{n\in K}$ is the summation symbol denoting that the sum is over all the fine groups in the broad group K,

χ_n is the fraction of the source neutrons emitted in the n-th group (source normalized to 1),

$\Sigma_{SO}(j \rightarrow n)$ is the P0 total scattering transfer macroscopic cross section for a neutron which scatters from group j to group n,

Σ_a^n is the macroscopic absorption cross section for the n-th group,

Σ_t^n is the macroscopic total cross section for the n-th group,

Σ_f^n is the macroscopic fission cross section for the n-th group,

ν_n is the average number of fission neutrons produced per fission event in the n-th group,

$PN(n \rightarrow j)$ is equal to $(2N+1)\Sigma_{SN}(n \rightarrow j)$ where N=0, 1, 2, 3 for the GGC-4 data tape and N=0, 1, 2, ...8 for the ENDF/B data tapes,

$\Sigma_{SN}(n \rightarrow j)$ is the scattering transfer macroscopic cross section coefficient for a neutron which scatters from group n to group j. The subscript N denotes the N-th scattering transfer cross section coefficient (term) in the Legendre polynomial expansion of the differential scattering transfer cross section. For N=0, this can be the elastic, inelastic, n-2n, or the total scattering transfer cross section when using the GGC-4 data tape but only the total scattering transfer cross section when using the ENDF/B data tapes.

The code also calculates the macroscopic transport cross section, the diffusion coefficient, and the average cosine of the scattering angle for each broad group. These three constants are defined as follows:

transport cross section, $\Sigma_{tr} \equiv \Sigma_t - \Sigma_{s1}$

diffusion coefficient, $D \equiv \dfrac{1}{3 \times \Sigma_{tr}}$

average cosine of the scattering angle, $\bar{\mu} \equiv \dfrac{\Sigma_{s1}}{\Sigma_{so}}$

The multigroup calculations performed in New Barnyard for these constants to obtain broad group constants are:

1. Transport cross section

$$\Sigma^K_{tr} = \Sigma^K_t - P1(K \rightarrow K)/3 \quad K=1, 2, 3, \ldots NBBG$$

(Recall that the PN coefficients, defined in both data tapes, include the factor 2n+1.)

2. Diffusion coefficient

$$D^K = \frac{1}{3 \times \Sigma_{tr}} \qquad K=1, 2, 3, \ldots NBBG$$

3. Average cosine of the scattering angle

$$\bar{\mu}^K = \frac{(P1(K \rightarrow K)/3)}{P0(K \rightarrow K)} \quad K=1, 2, 3, \ldots NBBG$$

Output Features of the Two Source Decks

Both source decks calculate total, absorption, and fission cross sections for each broad group as well as the transport cross section, diffusion coefficient, and the average cosine of the scattering angle for each broad group.

The ENDF/B source deck can also provide P0 through P8 macroscopic elastic scattering transfer broad group cross sections; however, the P0 transfer cross sections include inelastic and n-2n transfer cross sections as mentioned earlier. The order of the PN transfer cross sections desired is specified in the input data. The maximum number of broad groups is 20 due to the fact that insufficient core storage occurrs when more than 20 broad groups are used.

The GGC-4 source deck also provides a list of the nuclides on the data tape, the 99 fine group structure, the broad group structure, the P0 elastic, inelastic, n-2n, and total transfer broad group cross sections, and it lists P0 through P3 elastic scattering transfer broad group cross sections. This deck can calculate group cross sections for 22 broad groups.

IV. Operating Instructions

This chapter contains the information necessary to use New Barnyard (either source deck) on an IBM 7094 computer. The input data cards required for each source deck are given. Next, the control cards required by the IBM 7094 are shown, and finally the composite deck (control cards, source deck, data cards) is described.

The user should consider the following factors when selecting which source deck to use:

1. The GGC-4 source deck (i.e., the deck that uses the GGC-4 data tape) runs approximately twice as fast on the computer as the ENDF/B source deck.
2. Results for absorption or total cross sections from the two source decks for the same problem are approximately within 5 to 10% of each other.
3. P0 through P3 elastic scattering transfer cross sections can be obtained from the GGC-4 source deck P0 through P8 elastic scattering transfer cross sections can be obtained from the ENDF/B source deck.
4. The P0 output of the GGC-4 source deck lists separately the elastic, inelastic, 2(n-2n), and total transfer cross sections. The P0 output of the ENDF/B source deck lists only the total transfer cross sections.

As a convenience to the user of New Barnyard, sample problems including input data cards and computer output from both source decks are presented in Appendix G.

<u>Preparation of Input Data, Source Deck Using GGC-4 Data Tape</u>

<u>CARD NUMBER 1</u>:	Format (18A4)
Columns 1-72	Problem description - Anything can be punched on this card. This information will appear at the top of each page of results.
Symbol*	BXCX(I), I=1, 18
<u>CARD NUMBER 2</u>:	Format(3I3)
Columns 1-3	The number of broad energy groups
Symbol	NBBG, NBBG $\leq$ 22
Columns 4-6	The number of nuclides in the problem
Symbol	NNUK
Columns 7-9	This value is either 1 or 0. If it is 1, the code will calculate the flux. If it is 0, the code will have a flux input.
Symbol	KKK

---

* The symbol (variable) used in the code for this specified data.

CARD NUMBER 3:	Format (22I3)
Columns 1-3 Columns 4-6 ⋮ etc.	Lower broad group boundaries-The boundaries are the numbers of the lowest fine group in each of the selected broad groups. For example, if one of the lower energy bounaries was 10.00 MeV, the value 4 would be specified. The number of values punched on this card is equal to the number of broad groups, NBBG.
Symbol	LBGB (I), I = 1, NBBG

Each of the actual energy boundaries selected for a particular problem must correspond to one of the 99 energy groups; therefore, the broad group boundaries should be selected from the 99 fine group structure (Appendix D).

Sometimes it is desirable to know the broad group scattering transfer cross section for scatter into a "thermal dump" group. This can be done in New Barnyard by specifying a broad group for the energy range .4139 ev to 0. The lower broad group boundary value that would be input for this group would be 100. This procedure is valid because both data sources give scattering transfer cross section values for scatter into this "100th" group; however, one should be aware of the fact that no 100th group total or absorption cross section values are included on the data tapes.

CARD NUMBER 4:	Format (1F12.7, 1E13.6)
Columns 1-12	Identification number of the nuclide on the data tape - The I.D. number for the nuclides are shown in Table I.
Symbol	AID (IXL), IXL = 1, NNUK
Columns 13-25	The number density (nuclei/cm-barn) of the nuclide (nuclei/cm-barn) = $\frac{\text{nuclei}}{\text{cm}^3} * 10^{-24} \frac{\text{cm}^2}{\text{barn}}$
Symbol	DENT(IXL), IXL = 1, NNUK

Card 4 is repeated for each nuclide in the problem. All card 4's precede card 5. It is necessary to start with the card with the lowest I.D. number, followed by the cards with increasing I.D. numbers, since the nuclides on the data tape occur in the order of increasing I.D. numbers.

CARD NUMBER 5:	Format (6E12.6)
Columns 1-12 Columns 13-24 ⋮ etc.	Either a flux spectrum or a source spectrum is specified depending on the value on card number 1 for KKK (columns 7-9). 99 values (6 per card) must be specified.
Symbol	FLUX (I) or SSSS(I), I=1,99

If a flux spectrum is included as input data, none of the values can be zero. Appendix F contains 99 group source spectrums for fissionable nuclides. The source spectrum used must be normalized to 1 as is the case for the spectrums in Appendix F.

Preparation of Input Data, Source Deck for ENDF/B Data Tapes

The first three input data cards are the same as the data cards for the GGC-4 source deck with the exception that the maximum allowable number of broad groups is 20 (i.e., NBBG≤20).

Card 4 is repeated for each nuclide. All Card 4's precede Card 5. It is necessary to start for the card with the lowest material number followed by the cards with increasing material numbers, since the nuclides on each of the two data tapes occur in the order of increasing material number. If both data tapes are being used for a particular problem it is only necessary to have the cards arranged such that the material number (nuclides) that are on the same tape be placed in increasing order.

CARD NUMBER 4:	Format (1x,1A4,2I2,1I6,1E13.4)
Columns 2-5	Material number of the nuclide on the data tape-The material numbers for the nuclides are shown in Tables **II and III.**

Symbol	MATNO
Columns 6-7	Order of the PN scattering transfer cross sections desired-The values allowed are 0 through 8. This value must be the same for each nuclide used in the problem.
Symbol	LORDER
Columns 8-9	Logical unit number of the data tape-If using data tape B (Table II) the input value will be 1. If using data tape C (Table III) the input value will be 2. (No decimals are used with these values since I format)
Symbol	N
Columns 10-15	The number of data records before reaching the nuclide of interest.
Symbol	NOR
Columns 16-28	The number density (nuclei/cm-barn) of the nuclide $\left(\frac{\text{nuclei}}{\text{cm-barn}} = \frac{\text{nuclei}}{\text{cm}^3} * \frac{10^{-24}\text{cm}^2}{\text{barn}}\right)$
Symbol	DENT

The number of data records before reaching the nuclide of interest is obtained with the aid of the "No. of data Records" columns in Tables II and III. For example, if C-12

TABLE I

Data Tape A, GGC-4 Data

	Master	Duplicate
Tape No.	2602	OS122
Block Size*	256	256
Track	7	7
Density	556BPI	556BPI

Nuclide Identification Number	Nuclide Description	P-N
1.0000000	Hydrogen	3
1.2000000	Deuterium	3
2.0000000	Helium	3
3.0062000	Lithium-6	3
3.0072000	Lithium-7	3
4.0000000	Beryllium	3
5.0000000	Boron	3
5.0099999	Boron-10	3
6.0200000	Carbon	3
7.0000000	Nitrogen	3
8.0200000	Oxygen	3
11.0000000	Sodium	3
12.0000000	Magnesium	3
13.0000000	Aluminum	3
14.0000000	Silicon	3
16.0000000	Sulfur	3
20.0000000	Calcium	3
22.0000000	Titanium	3
23.9999990	Chromium	3
25.0000000	Manganese	3
26.0000000	Iron	3
27.0000000	Cobalt	3
27.9999990	Nickel	3
29.0000000	Copper	3
41.9999990	Molybdenium	3
48.0000000	Cadmium	3
73.9999990	Tungsten	3
74.1799990	Tungsten - 180	3
74.1820000	Tungsten - 182	3

* Binary tape

Table I (Contd')

Nuclide Identification Number	Nuclide Description	P-N
74.1821000	Tungsten-182 (Resonance data included)	3
74.1829990	Tungsten-183	3
74.1831000	Tungsten-183 (Resonance data included	3
74.1840000	Tungsten-184	3
74.1841000	Tungsten-184 (Resonance data included)	3
74.1859990	Tungsten-186	3
74.1861000	Tungsten-186 (Resonance data included)	3
82.0000000	Lead	3
92.2334990	Uranium-233	3
92.2350000	Uranium-235 (NASA Data)	3
92.2351990	Uranium-235	3
92.2379990	Uranium-238 (NASA Data)	3
92.2381000	Uranium-238 (Resonance data included)	3
92.2381990	Uranium-238 (ORNL data)	3
92.2382990	Uranium-238 (ORNL Resonance data included)	3
94.2411990	Plutonium-241	3

TABLE II

Data Tape B, ENDF/B Data

	Master	Duplicate
Tape No.	1152	2601
Block size*	80	80
Track	7	7
Density	800BPI	800BPI

Material Number	Material	P-N	No. of Data Records	Total Records
1003	H-2	8	2892	2892
1005	Li-6	8	1537	4429
1006	Li-7	8	1531	5960
1007	Be	8	1479	7439
1009	B-10	8	1447	8886
1010	C-12	8	1392	10278
1012	N-14	8	1072	11350
1013	O-16	8	991	12341
1014	Mg	8	1048	13389
1015	Al-27	8	1079	14468
1016	Ti	8	1081	15549
1017	V	8	1136	16685
1018	Cr	8	963	17648
1019	Mn	8	1110	18758
1020	Fe	8	1079	19837
1021	Ni	8	975	20812
1044	U-235	8	1197	22009
1047	U-238	8	1172	23181

----

*BCD Tape

TABLE III

Data Tape C, ENDF/B Data

	Master	Duplicate
Tape No.	1148	05578
Block size*	80	80
Track	7	7
Density	800BPI	800BPI

Material Number	Material	P-N	No. of Data Records	Total Records
1001	H-1	8	8286	8286
1050	Pu-238	8	1133	9419
1051	Pu-239	8	1160	10579
1053	Pu-240	8	1157	11736
1054	Pu-241	8	1215	12951
1055	Pu-242	8	1151	14102
1059	Na-23	8	1073	15175
1060	W-182	8	1139	16314
1061	W-183	8	1148	17462
1062	W-184	8	1151	18613
1063	W-186	8	1173	19786

----
*BCD Tape

and N-14 were the nuclides needed for a particular problem, the value 8886 would be punched on the data card for C-12, and the value 0 would be punched on the data card for N-14. When the computer reads the value 8886 it "skips" 8886 data records; that is, the data tape is advanced until the nuclide of interest is reached and then the computer begins to read the data from the tape. When the computer reads 0, no data records are skipped. If two nuclides of interest are separated by other nuclides, the number of data records that have to be skipped is the sum of the records for the nuclides separating the two. For example, if H-2, and Be were the nuclides of interest, 3068 (1537+1531) would be specified on the data card for H-2 and 0 would be specified on the data card for Be.

CARD NUMBER 5:	Format (6E12.6)
Columns 1-12	Either a flux or source spec. is specified. (Same as the GGC-4 source deck)
Symbol	FLUX(I) or SSSS(I), I=1,99

Control Cards for the IBM 7094

A brief explanation of the control cards used in the two source decks is given here to insure proper usage of these source decks on an IBM 7094 digital computer located at the Digital Computation Division (ASNCD), Wright-Patterson AFB, Ohio. As a convenience to the user, the only control card that will have to be prepared is the $JOB card, the first card of the composite deck; all other control cards have been prepared and positioned in the source decks. The user should be aware of the fact that control cards are subject to change and that each computer facility has its own characteristic control cards. The user should always have on hand the current literature for the computer being used.

Control Cards, GGC-4 Source Deck

The control cards used with the GGC-4 source deck are as follows:

```
Columns 1       8       16                      31

        $JOB            Priority, Time, Lines   Job I.D.
        $SETUP  1       OS122,NORING
        $IBJOB          MAP,FIOCS
        $IBMAP  FILES
                ENTRY   .UN01.
        .UN01.  PZE     UNIT01
        .UNIT01 FILE    ,,READY,INPUT,BIN,BLK=256
                ENTRY   .UN02.
        .UN02.  PZE     UNIT02
        UNIT02  FILE    ,,READY,INPUT,BIN,BLK=256
                ENTRY   .UN03.
        .UN03.  PZE     UNIT03
        UNIT03  FILE    ,,READY,INOUT,BIN,BLK=256
                ENTRY   .UN04.
        .UN04.  PZE     UNIT04
        UNIT04  FILE    ,,READY,INOUT,BIN,BLK=256
                ENTRY   .UN07.
        .UN07.  PZE     UNIT07
        UNIT07  FILE    ,,READY,INOUT,BIN,BLK=256
                ENTRY   .UN08.
        .UN08   PZE     UNIT08
        UNIT08  FILE    ,,READY,INOUT,BIN,BLK-256
                END
```

```
Columns 1       8

        $IBFTC  MAIN
        $IBFTC  REWE
        $IBFTC  ONEE
        $IBFTC  CSAVE
        $DATA
        $EOF
```

The $JOB card is an orange card supplied by the computer facility. The priority number should always be 0 unless the user is authorized some other number. The time number is the estimated 7094 time* needed in minutes. The line number is the estimated number of lines** that will be output. This number should include all printing and card punching that will be done by the 7094. If either the time estimate or the line estimate is exceeded, the computer run will be terminated. The job identification should include the user's account number, office symbol, and name. After the user's name, there should appear a slash followed by the name Bridgman. The data tapes are filed under this name and it has to be present on the $JOB card to use the GGC-4 data tape. An example of the $JOB card is shown:

```
$JOB   0,3,2000      68-564-00  AFIT-SE  JONES/BRIDGMAN
```

---

* An estimate of 3 minutes is usually adequate when using the GGC-4 source deck.

** An estimate of 2000 lines is adequate for the GGC-4 source deck.

Control Cards, ENDF/B Source Deck

The control cards for the ENDF/B source deck are almost the same as the control cards for the GGC-4 source deck. The exceptions are the following:

(1) No file is defined for logical unit 7 (only 3 scratch tapes are used)

(2) 2 $SETUP cards are used for the two input data tapes.

(3) Usually an adequate time estimate for the $JOB card is 5 minutes.

Composite Deck Setup

The composite deck setup for either source deck is shown below. The user has to supply the $JOB card and the input data cards.

```
   $JOB
   $SETUP (2 cards for the ENDF/B source deck)
   $IBJOB
   $IBMAP FILES
          ENTRY  .UN01.
             :
            END
   $IBFTC  MAIN
(Source Deck - includes all subroutines)
   $DATA
(Input data cards)
   $EOF
```

The $SETUP Card is used to indicate the use of a data tape. The number 1 on this card refers to the logical number of the tape used in the program (GGC-4 Source deck). The number 0S122 is the number of the duplicate copy of the GGC-4 data tape. The word NORING that follows the tape number is used to indicate that the tape is not to be written on. The $IBJOB card as used here is to have a print out (map) of the storage location of each variable used in the program. The word FIOCS is used to cut down on the core storage requirement of the program. The $IBMAP card and all the cards that follow up to and including the END card are used to define files for the input data tape and the 5 scratch tapes used by the program. The $IBFTC cards are required for the main program and the subroutines used. The $IBFTC MAIN card precedes the GGC-4 source deck and each of the other $IBFTC cards precede a subroutine in the source deck. The names REWE, ONEE, and CSAVE that appear on these cards are arbitrary; any name (6 letters or less) other than the actual name of the subroutine can be used. The $DATA card is required when input data cards are used. The $EOF card is the end-of-file card and it is the last card of the composite deck.

V. Some Sample Results and Conclusions

In this chapter results of the GGC-4 version of New Barnyard are compared with results from other cross section codes; also, group cross sections calculated from the ENDF/B version of New Barnyard are compared with group cross sections calculated from the GGC-4 version. Throughout this chapter repeated reference will be made to the zero moment of the neutron flux, zero moment flux, energy dependent flux, and flux. These terms all mean the same thing. The reader is reminded that the GGC-4 version of New Barnyard refers to the source deck that uses the GGC-4 data tape and, the ENDF/B version of New Barnyard refers to the source deck that uses the two ENDF/B data tapes.

The Zero Moment of the Neutron Flux for Aluminum

It was pointed out in Chapter II that in order to calculate group cross sections, the energy dependent flux had to be known. This flux is obtained in New Barnyard by solving the zero moment equation (35). The validity of this flux calculation was checked by comparing New Barnyard's calculation of the zero moment flux for aluminum with Gulf General Atomic's GAM-1 (Ref 9) calculation of the zero moment flux for the same material. GAM-1 calculates group neutron cross sections and other related constants such as age for the energy range 10 MeV to .4139 ev. GAM-1 uses a 68 group cross section library data tape. In order to have a better basis for comparison, the GGC-4 version of New Barnyard was modified slightly to use GAM-1's library data tape. Table IV shows the results of these two flux calculations. The ratio of the GAM-1 flux to the New Barnyard flux for each energy group is 4.000, as shown in Table IV . This indicates that the shape of the two flux spectrums are identical but with the GAM-1 flux having a magnitude 4 times larger than the New Barnyard flux. This factor of 4 is unimportant, however, since in the flux weighting process for the calculation of group cross sections this factor will always cancel and either flux will produce the same group cross sections.

<u>Seven Group Cross sections for Carbon</u>

The validity of the flux weighting process to calculate group cross sections was checked by comparing a 7 group macroscopic cross section set for carbon calculated by the GGC-4 version of New Barnyard with a 7 group macroscopic cross section set for carbon calculated by the GGC-4 code. The GGC-4 code was run on a CDC 6600 computer at the A.F. Weapons Lab. The same flux spectrum was input to each of the codes. The group cross sections obtained from each of the codes were exactly identical. Table V shows the broad group structure for this cross section set. Table VI shows the results from both codes for the total cross sections and Table VII lists the PO (total transfer) scattering cross sections from both codes.

TABLE IV

Flux Spectrum for Aluminum Calculated

in New Barnyard and GGC-4.

Group	New Barnyard Flux (on^1/cm^2-group)	GAM-1 Flux (on^1/cm^2 group)	Ratio of GAM-1 Flux to New Barnyard Flux
1	9.2397×10^2	3.6959×10^{-1}	4.000
5	2.4437×10^0	9.7748×10^0	4.000
10	8.9699×10^0	3.5880×10^1	4.000
15	1.8522×10^1	7.4089×10^1	4.000
20	1.3666×10^1	5.4664×10^1	4.000
25	$6,7206 \times 10^1$	2.6882×10^2	4.000
30	2.3125×10^1	9.2500×10^1	4.000
35	2.4929×10^1	9.9715×10^1	4.000
40	2.4753×10^1	9.9013×10^1	4.000
45	2.4753×10^1	9.9013×10^1	4.000
50	2.3750×10^1	9.4999×10^1	4.000
55	2.1040×10^1	8.4162×10^1	4.000
60	1.7247×10^1	6.8988×10^1	4.000
65	1.1993×10^1	4.7972×10^1	4.000

TABLE V

Broad Group Structure of the 7 Group Cross Section Set for Carbon.

Group	Energy (ev)		Interval
1	1.4918×10^{7}	to	1.3498×10^{7}
2	1.3498×10^{7}	to	1.2214×10^{7}
3	1.2214×10^{7}	to	1.1052×10^{7}
4	1.1052×10^{7}	to	1.0000×10^{7}
5	1.0000×10^{7}	to	6.0653×10^{6}
6	6.0653×10^{6}	to	3.0119×10^{6}
7	3.0119×10^{6}	to	1.0026×10^{6}

TABLE VI

Total Cross Sections for Carbon Calculated in New Barnyard and GGC-4

Group	New Barnyard Total Cross Section (cm^{-1})	GGC-4 Total Cross Section (cm^{-1})
1	1.0730×10^{-1}	1.0730×10^{-1}
2	1.1078×10^{-1}	1.1078×10^{-1}
3	1.1223×10^{-1}	1.1223×10^{-1}
4	9.8612×10^{-2}	9.8612×10^{-2}
5	1.0006×10^{-1}	1.0006×10^{-1}
6	1.4518×10^{-1}	1.4518×10^{-1}
7	1.6053×10^{-1}	1.6053×10^{-1}

TABLE VII

P0 Scattering Transfer Cross Sections for Carbon Calculated in New Barnyard and GGC-4

Group	New Barnyard P0 Total Transfer Cross Section (cm^{-1})	GGC-4 P0 Total Transfer Cross Section (cm^{-1})
FROM 1 TO 1	3.6984×10^{-2}	3.6984×10^{-2}
FROM 1 TO 2	1.3377×10^{-2}	1.3377×10^{-2}
FROM 1 TO 3	6.6004×10^{-3}	6.6004×10^{-3}
FROM 1 TO 4	4.0538×10^{-3}	4.0538×10^{-3}
FROM 1 TO 5	1.6083×10^{-2}	1.6083×10^{-2}
FROM 1 TO 6	3.5487×10^{-3}	3.5487×10^{-3}
FROM 1 TO 7	1.0219×10^{-2}	1.0219×10^{-2}
FROM 2 TO 2	3.8940×10^{-2}	3.8940×10^{-2}
FROM 2 TO 3	1.4444×10^{-2}	1.4444×10^{-2}
FROM 2 TO 4	8.1390×10^{-3}	8.1390×10^{-3}
FROM 2 TO 5	1.8409×10^{-2}	1.8409×10^{-2}
FROM 2 TO 6	5.6734×10^{-3}	5.6734×10^{-3}
FROM 2 TO 7	7.8967×10^{-3}	7.8967×10^{-3}
FROM 3 TO 3	3.8735×10^{-2}	3.8735×10^{-2}
FROM 3 TO 4	1.5851×10^{-2}	1.5851×10^{-2}
FROM 3 TO 5	2.3362×10^{-2}	2.3362×10^{-2}
FROM 3 TO 6	1.1659×10^{-2}	1.1659×10^{-2}
FROM 3 TO 7	4.9461×10^{-3}	4.9461×10^{-3}
FROM 4 TO 4	3.3846×10^{-2}	3.3846×10^{-2}
FROM 4 TO 5	2.6936×10^{-2}	2.6936×10^{-2}
FROM 4 TO 6	2.6107×10^{-2}	2.6107×10^{-2}
FROM 4 TO 7	2.5326×10^{-3}	2.5326×10^{-3}
FROM 5 TO 5	4.2742×10^{-2}	4.2742×10^{-2}
FROM 5 TO 6	3.5086×10^{-2}	3.5086×10^{-2}
FROM 5 TO 7	1.9360×10^{-2}	1.9360×10^{-2}
FROM 6 TO 6	9.3408×10^{-2}	9.3408×10^{-2}
FROM 6 TO 7	5.0676×10^{-2}	5.0676×10^{-2}
FROM 7 TO 7	1.3782×10^{-1}	1.3782×10^{-1}

Six Group Cross Section Set for Nitrogen

A comparison of cross sections calculated in the ENDF/B version of New Barnyard and the GGC-4 version of New Barnyard was made for several problems. The 6 group cross section set for nitrogen that is presented here for comparison is typical of the cross section comparisons obtained for other nuclides and mixtures. The cross sections calculated by the GGC-4 version of New Barnyard were used as the standard for the relative difference calculations between these two cross section sets. Table VIII shows the 6 group structure of the cross sections. The absorption cross sections obtained from the two versions of New Barnyard and the relative differences between these cross sections are shown in Table IX ; similarly, Table X shows the total cross sections and their relative differences. Table XI and XII show the comparisons for the P0, P1, P2, and P3 scattering transfer cross sections. The P0 cross section is the total transfer cross section which is the sum of the elastic inelastic, and 2 (n-2n) transfer cross section. The P1 through P3 cross sections are elastic scattering cross sections only.

By inspection of Tables IX and X it is seen that the largest relative differences are 13% and 14% which occurred for the 3rd and 4th group absorption cross sections respectively. Even these relative differences aren't too great when one considers the following factors:

(1) The cross section data tapes are from two independent sources, namely GGA (Gulf General Atom) and ORNL (Oak Ridge National Lab.).

(2) Each group flux is dependent on the total and P0 fine group cross sections. If there exists a relative difference for each group cross section, then there exists a relative difference for each group flux. Since broad group cross sections are obtained by flux weighting fine group cross sections, it is logical to argue that the relative differences build up rapidly since when multiplying or adding variables that have relative differences (errors) associated with them these relative differences are added together.

The comparisons of the P0 scattering transfer cross section are quite good. Table XI shows that most relative differences are less than 5%; however, the relative differences for the P0 cross sections for transfer from 1 to 3, 1 to 4, 1 to 5, and 1 to 6, were 41, 99, 100, and 100%

respectively, but these four cross section values are very small in magnitude and most likely ORNL did not list cross section values as small as GGA's. The largest relative difference in the comparisons between the P1 cross sections was 21%. It is seen from Table XII that the relative differences in the comparisons between P2 cross sections are good except for cross section values for transfer from 1 to 1 and 1 to 2. Note that in the P3 cross section comparison, even larger relative differences than for P2 are common and that for 3 of the cross section comparisons (denoted by an asterisk) the algebraic signs differ between the cross sections. All of these large relative differences and alternating algebraic signs can be explained by the fact that ORNL used an 8th order Legendre polynomial expansion of the fine group differential scattering transfer cross sections and GGA used a 6th order *Legendre polynomial expansion.

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* GGA only listed P0 through P3 cross sections on the GGC-4 data tape.

TABLE VIII

Broad Group Structure of the 6 Group Cross Section Set for Nitrogen

Group	Energy Interval (ev)
1	1.4918×10^{7} to 3.0119×10^{6}
2	3.0119×10^{6} to 6.0810×10^{5}
3	6.0810×10^{5} to 1.2277×10^{5}
4	1.2277×10^{5} to 2.6126×10^{3}
5	2.6126×10^{3} to 4.7851×10^{1}
6	4.7851×10^{3} to 3.9279

TABLE IX

Comparison Between Absorption Cross Selections Which Were Calculated in the GGC-4 Version and the ENDF/B Version of New Barnyard for Nitrogen

Group	GGC-4 Absorption Cross Section (cm^{-1})	ENDF/B Absorption Cross Section (cm^{-1})	Relative Difference (%)
1	1.8192×10^{-5}	1.7910×10^{-5}	2
2	3.9154×10^{-6}	3.6110×10^{-6}	8
3	5.3592×10^{-7}	4.6538×10^{-7}	13
4	1.2364×10^{-7}	1.4085×10^{-7}	14
5	9.2645×10^{-7}	9.8802×10^{-7}	7
6	4.3170×10^{-6}	4.5740×10^{-6}	6

TABLE X

Comparisons Between Total Cross Sections Which Were Calculated in the GGC-4 Version and the ENDF/B Version of New Barnyard for Nitrogen

Group	GGC-4 Total Cross Section (cm^{-1})	ENDF/B Total Cross Section (cm (cm^{-1})	Relative Difference (%)
1	8.7827×10^{-5}	8.4124×10^{-5}	4
2	9.9716×10^{-5}	9.6426×10^{-5}	3
3	1.7344×10^{-4}	1.5963×10^{-4}	8
4	3.4983×10^{-4}	3.4664×10^{-4}	1
5	4.9423×10^{-4}	4.9424×10^{-4}	0
6	5.3625×10^{-4}	5.3478×10^{-4}	0

TABLE XI

Comparison Between P0 and Between P1 Scattering Transfer Cross Sections Which Were Calculated in the GGC-4 Version and the ENDF/B Version of New Barnyard for Nitrogen

Group	GGC-4 P0 Transfer Cross Section (CM^{-1})	ENDF/B P0 Transfer Cross Section (CM^{-1})	P0 Relative Difference %	GGC-4 P1 Transfer Cross Section (CM^{-1})	ENDF/B P1 Transfer Cross Section (CM^{-1})	P1 Relative Difference %
From 1 to 1	4.7845×10^{-5}	4.8938×10^{-5}	2	4.0280×10^{-5}	5.8782×10^{-5}	2
From 1 to 2	2.1630×10^{-5}	1.7194×10^{-5}	20	-2.1332×10^{-5}	-2.2410×10^{-5}	5
From 1 to 3	1.4876×10^{-7}	8.7676×10^{-8}	41	0.0	0.0	0
From 1 to 4	2.1350×10^{-8}	1.6494×10^{-10}	99	0.0	0.0	0
From 1 to 5	1.0350×10^{-10}	0.0	100	0.0	0.0	0
From 1 to 6	4.1896×10^{-13}	0.0	100	0.0	0.0	0
From 2 to 2	8.4995×10^{-5}	8.2104×10^{-5}	3	4.3963×10^{-5}	4.3874×10^{-5}	0
From 2 to 3	1.0805×10^{-5}	1.0760×10^{-5}	0	-9.4342×10^{-6}	-9.6751×10^{-6}	3
From 2 to 4	0.0	0.0	0	0.0	0.0	0
From 2 to 5	0.0	0.0	0	0.0	0.0	0
From 2 to 6	0.0	0.0	0	0.0	0.0	0
From 3 to 3	1.5751×10^{-4}	1.4480×10^{-4}	8	4.8841×10^{-5}	4.1714×10^{-5}	15
From 3 to 4	1.5394×10^{-5}	1.4376×10^{-5}	6	-1.3633×10^{-5}	-1.2710×10^{-5}	7
From 3 to 5	0.0	0.0	0	0.0	0.0	0
From 3 to 6	0.0	0.0	0	0.0	0.0	0
From 4 to 4	3.3730×10^{-4}	3.3410×10^{-4}	1	6.2527×10^{-5}	6.2461×10^{-5}	0
From 4 to 5	1.2409×10^{-5}	1.2403×10^{-5}	0	-1.1164×10^{-5}	-1.1156×10^{-5}	0
From 4 to 6	0.0	0.0	0	0.0	0.0	0
From 5 to 5	4.7704×10^{-4}	4.7689×10^{-4}	0	8.5092×10^{-5}	8.5863×10^{-5}	0
From 5 to 6	1.6266×10^{-5}	1.6367×10^{-5}	0	-1.4655×10^{-5}	-1.4750×10^{-5}	0
From 6 to 6	5.0523×10^{-4}	5.0350×10^{-4}	0	1.0005×10^{-4}	1.0049×10^{-5}	0

TABLE **XII**

Comparisons Between P2 and Between P3 Scattering Transfer Cross Sections Which Were Calculated in the GGC-4 Version and the ENDF/B Version of New Barnyard for Nitrogen

Group	GGC-4 P2 Transfer Cross Section (CM^{-1})	ENDF/B P2 Transfer Cross Section (CM^{-1})	P2 Relative Difference (%)	GGC-4 P3 Transfer Cross Section (CM^{-1})	ENDF/B P3 Transfer Cross Section (CM^{-1})	P3 Relative Difference (%)
From 1 to 1	4.1080×10^{-5}	7.7636×10^{-5}	89	3.3199×10^{-5}	5.6592×10^{-5}	70
From 1 to 2	3.3534×10^{-6}	2.0285×10^{-6}	40	-1.8488×10^{-7}	-1.2627×10^{-5}	32
From 1 to 3	0.0	0.0	0	0.0	0.0	0
From 1 to 4	0.0	0.0	0	0.0	0.0	0
From 1 to 5	0.0	0.0	0	0.0	0.0	0
From 1 to 6	0.0	0.0	0	0.0	0.0	0
From 2 to 2	1.7421×10^{-5}	2.2396×10^{-5}	3	3.8061×10^{-6}	5.3094×10^{-6}	40
From 2 to 3	1.5530×10^{-6}	-1.8053×10^{-6}	2	8.6145×10^{-8}	-5.0983×10^{-9}*	41
From 2 to 4	0.0	0.0	0	0.0	0.0	0
From 2 to 5	0.0	0.0	0	0.0	0.0	0
From 2 to 6	0.0	0.0	0	0.0	0.0	0
From 3 to 3	4.9520×10^{-6}	3.9846×10^{-6}	19	3.3335×10^{7}	-5.2558×10^{-8}*	84
From 3 to 4	-1.6490×10^{-6}	-1.4656×10^{-6}	11	-9.5588×10^{-8}	-1.5059×10^{-7}	56
From 3 to 5	0.0	0.0	0	0.0	0.0	0
From 3 to 6	0.0	0.0	0	0.0	0.0	0
From 4 to 4	3.9850×10^{-6}	3.3925×10^{-6}	15	-5.5014×10^{-8}	-5.1122×10^{-8}	7
From 4 to 5	-1.1462×10^{-6}	-1.2008×10^{-6}	4	-7.8139×10^{-8}	-5.3329×10^{-8}	26
From 4 to 6	0.0	0.0	0	0.0	0.0	0
From 5 to 5	5.1923×10^{-6}	4.1422×10^{-6}	20	-7.7961×10^{-8}	-2.3663×10^{-8}	70
From 5 to 6	-1.4863×10^{-6}	-1.5628×10^{-6}	5	-9.0747×10^{-8}	-6.4528×10^{-8}	29
From 6 to 6	6.3895×10^{-6}	5.2714×10^{-6}	17	-1.2026×10^{-8}	9.2380×10^{-8}*	250

* The absolute value was taken so that a comparison could be made.

Conclusions

The comparison of the flux calculated by New Barnyard with the flux calculated by GAM-1 and the comparison of the group cross sections calculated by New Barnyard with the group cross sections calculated by GGC-4 give confidence in the "correctness" or validity of the calculations performed by New Barnyard; that is, New Barnyard does what it is supposed to do, and does it right. Some thought, however, should now be given to the **usefulness** of the zero moment of the neutron flux for flux weighting cross sections.

Recall that the zero moment equation was based on an infinite medium and a plane source of infinite dimensions at the origin. Assumptions such as these are valid for reactor calculations where one at least crudely has a plane source (fission neutrons in the core) and an infinite medium (shielding around the core). When considering one dimensional thin shield transport calculations for neutrons with energies above thermal, where the mean free path of a neutron is larger than the thickness of the shield, the zero moment equation leads to an inaccurate flux spectrum. Thus, for a thin shield calculation a better approximation for the flux is one equal to the source spectrum for the problem since relatively few neutrons are slowed down when passing through the shield. It was for cases such as this that

New Barnyard was programmed with the option of inputting a flux spectrum.

One should also recall that no resonance calculations are performed in New Barnyard; however, resonances in general are negligible above a few kev and it is above this energy range that New Barnyard is primarily intended.

Finally, recall the zero moment flux was used to flux weight P0 through P8 scattering transfer cross sections. Cross section codes such as GAM-1 and GGC-4 usually flux weight the PN cross sections with a corresponding ϕ_N flux which comes from the expansion of the three dimensional flux, ϕ (X, E, μ), in a Legendre polynomial series followed by either a B_L (Ref 10) or a P_L (Ref 11) solution to the Boltzmann equation. In general, as N becomes large ϕ_N becomes small; thus, flux weighting the PN cross sections with ϕ_N fluxes produces PN group cross sections smaller than the corresponding PN group cross sections flux weighted by the zero moment flux.

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APPENDIX A

GGC-4 Source Deck of New Barnyard

On the following pages is listed the source deck that uses the GGC-4 cross section library data tape. It is written in the Fortran IV language for use on the IBM 7094 digital computer. It consists of a main program and 3 separate subroutines.

Input data from data cards and from the GGC-4 cross section data tape are "read in" in the main program. Five scratch tapes are used to store P0, P1, P2, P3, inelastic, n-2n, and total P0 cross section arrays until they are needed in the main program. The flux calculation and the broad group calculations of the transport cross section, diffusion coefficient, and the average cosine of the scattering angle are performed in the main program.

The subroutine named REW is used to rewind the scratch tapes and to set a double subscripted scattering transfer cross section symbol to zero. The subroutine named ONE is used to calculate 99 group macroscopic transfer cross section sets. The subroutine named CSAV is called in the main program to flux weight the cross sections.

Extra comment cards have been added to the listing that follows so that the program will be easier to read. A glossary of computer program symbols is given in Appendix C.

```
C      NEW BARNYARD
C      MULTIGROUP NEUTRON MACROSCOPIC CROSS SECTION CODE
C1     THIS SOURCE DECK USES THE GGC-4 CROSS SECTION DATA TAPE
       DIMENSION BXCX(18),NTID(3),AT(90),DAD(21),DD(3,45),LEN(4),
      1SIGQ(100),TRA(310),GTH(110),ENG(103),UL(206),LLBGB(22),
      2SIGA(100),SIGT(100),SS(37),TOT(22),ABBS(22),SISO(22,22),
      3PO(22,22),XINELS(22,22),XN2N(22,22),ESS(40),FTOT(22),AID(31),
      4DENT(31),SSSS(99),P1(22,22),P2(22,22),P3(22,22)
       COMMON TTT(5050),TT(100,100),LDF,LD,LT,NBBG,LBGB(99),FLUX(100)
       EQUIVALENCE (ESS(7),SS(1))
       MG=1
       MSS=2
       MT=3
       MTT=4
       MMT=7
       MMMT=8
C2     MG, MSS, MT, MTT, MMT, MMMT ARE THE TAPE NUMBERS USED IN
C      THIS SOURCE DECK
.      FPL=1.0
       FNBT=31568.00
C3     READ FIRST DATA CARD
       READ (5,2)(BXCX(I),I=1,18)
C4     READ SECOND DATA CARD
       READ(5,6)NBBG,NNUK,KKK
     2 FORMAT(18A4)
    6  FORMAT(3I3)
       REWIND MG
C5     THE FOLLOWING CARDS THROUGH CARD NO. 10 ARE FOR READING
C      THE GGC-4 DATA TAPE AND FOR PRINTING OUT  IMPORTANT INFORMATION
C      FROM THE DATA TAPE
       READ (MG) (NTID(I),I=1,3)
       NBT=NTID(1)
       NEP=NTID(2)
       NGT=NTID(3)
       NEV=NEP-1
       NES=NEP+1
       LNX=NEV+6
       NBTC=FNBT+0.1
       IF(FNBT .LT. 0.) NBTC=FNBT-0.1
       IF(NBT-NBTC)4,5,4
     4 WRITE(6,981)NBT,NBTC
   981 FORMAT(1H1//5X,22HFAST DATA TAPE NUMBER I6,35H WAS LOADED INSTEAD
      1 OF TAPE NUMBER I6,1X,22H WHICH WAS SPECIFIED. //
      25X,20HPROBLEM TERMINATED.
       CALL EXIT
     5 READ (MG)(AT(I),I=1,90)
       WRITE(6,1002)(BXCX(I),I=1,18)
  1002 FORMAT(1H1,18A4)
       WRITE(6,905)NBT,(AT(I),I=1,90)
   905 FORMAT(26HOFAST DATA TAPE NUMBER  =  I6/
      X23HOTAPE DESCRIPTION.....  //(1X,18A4))
       READ (MG)NNOT
       ASSIGN 809 TO NPR1
       IF(FPL)808,808,750
  808  ASSIGN 129 TO NPR1
       GO TO 810
  750  CONTINUE
```

```
804   LRINK=0
      WRITE(6,894)NBT
  894 FORMAT(/35HOCONTENTS OF FAST DATA TAPE NUMBER   I6//
     X5X,11HNUCLIDE NO.  5X,20HNUCLIDE DESCRIPTION  /)
  810 DO 129 IXL=1,NNCT
      READ(MG)(DAD(I),I=1,21)
      DD(1,IXL)=DAD(19)
      DD(2,IXL)=DAD(20)
      DD(3,IXL)=DAD(21)
C     DAD(19)=NUCLIDE ID NO.$DAD(20)=NO. OF RESOLVED RESONANCES
C     DAD(21)=NO. OF UNRESOLVED RESONANCES.
      IF(IXL.GT.16.AND.IXL.LE.30) GO TO 129
      GO TO NPR1,(809,129)
  809 LRINK=LRINK+1
      IF(LRINK-46)805,807,805
  807 WRITE(6,895)NBT
  805 CONTINUE
      WRITE(6,897)DAD(19),(DAD(I),I=1,18)
  895 FORMAT(35H1CONTENTS OF FAST DATA TAPE NUMBER   I6//
     X5X,11HNUCLIDE NO. 5X,20HNUCLIDE DESCRIPTION  /)
  897 FORMAT(1X,F13.7,5X,18A4,/)
  129 CONTINUE
C     READ FAST DATA TAPE--RESONANCE TABLES
      READ(MG)(LEN(I),I=1,4)
      LBS=LEN(4)
      READ(MG)(TTT(I),I=1,LBS)
C     READ FAST DATA TAPE---ENERGIES,LETHARGIES,DELTA U
      JMM=2*NES+NEP
      READ(MG)(TRA(I),I=1,JMM)
      READ(MG)(GTH(I),I=1,NGT)
      READ(MG)(NTID(I),I=1,1)
      DO 84 I=1,NEP
      ENG(I)=TRA(I)
      IX=I+NES
      UL(I)=TRA(IX)
 84   IZ=IX+NES
      ENG(NES)=TRA(NES)
      UL(NES)=TRA(2*NES)
      DO 8  I=1,99
    8 LBGB(I)= 0
C6    READ THIRD DATA CARD
      READ(5,7)(LLBGB(I),I=1,NBBG)
      DO 9 I=1,NBBG
    9 LBGB(I)=LLBGB(I)
    7 FORMAT(24I3)
      WRITE(6,1002)(BXCX(I),I=1,18)
      WRITE(6,747)
      WRITE(6,748)(I,ENG(I),ENG(I+1),UL(I),UL(I+1),LBGB(I),I=1,50)
 747  FORMAT(22HOFINE GROUP STRUCTURE  /80HOGROUP                   ENERGY INTE
     1RVAL(E.V.)                      LETHARGY INTERVAL     ,39H          LO
     2WER BD. GRP BOUND.                                   )
  748 FORMAT(I5,6X,1PE13.6,4H TO 1PE13.6,6X,1PE13.5,4H TO 1PE12.5,
     X1I22)
      WRITE(6,1002)(BXCX(I),I=1,18)
      WRITE(6,747)
      WRITE(6,748)(I,ENG(I),ENG(I+1),UL(I),UL(I+1),LBGB(I),I=51,99)
      WRITE(6,1002)(BXCX(I),I=1,18)
```

```
      WRITE(6,1009)
 1009 FORMAT(22H0BROAD GROUP STRUCTURE /46H0BROAD GROUP              ENER
     XGY INTERVAL(E.V.)
      ENG(101)=0.0
      DO 1008 I=1,NBBG
      IF(I .NE. 1) GO TO 1005
      JNY=LBGB(I)+1
      WRITE(6,1007)I,ENG(1),ENG(JNY)
      GO TO 1008
 1005 K=I-1
      JNYN=LBGB(K)+1
      JNYY=LBGB(I)+1
      WRITE(6,1007)I,ENG(JNYN),ENG(JNYY)
 1008 CONTINUE
 1007 FORMAT(I5,10X,1PE13.6,4H TO 1PE13.6)
C     SKIP FISSION SPECTRA SOURCE DATA ON TAPE
      NSP=NTID(1)
      DO 10 I=1,NSP
   10 READ(MG)DUMMY
      DO 27 IXL=1,NNUK
C7    READ FOURTH DATA CARD
      READ(5,11)AID(IXL),DENT(IXL)
   11 FORMAT(1F12.7,1E13.6)
      DO 31 JZ=1,NNOT
      J2=JZ
C8    THE NEXT CARD CHECKS TO SEE IF THE ID NUMBERS  THAT ARE INPUT
C     MATCH AN ID NUMBER ON THE DATA TAPE
      IF(ABS(AID(IXL)-DD(1,J2))-0.00001)27,27,31
   31 CONTINUE
      WRITE(6,982)AID(IXL),NBT,(DD(1,I),I=1,NNOT)
  982 FORMAT(1H1//5X, 15H NUCLIDE NUMBER F9.4 ,31H NOT ON FAST DATA TA
     XPE NUMBER   I6//5X,27HNUCLIDES ON DATA TAPE ARE--//(30X,F9.4))
   27 CONTINUE
      READ(MG) (NTID(I),I=1,1)
C     SKIP RESONANCE DATA
      NMORE=2*NTID(1)
      DO 160 I=1,NMORE
  160 READ(MG)DUMMY
C     ZERO OUT CROSS SECTION ARRAYS
      DO 15  I=1,100
      SIGQ(I)=0.0
      SIGA(I)=0.0
      ENG(I)=0.0
   15 SIGT(I)=0.0
      DO 111 JSD=1,NNUK
  777 READ(MG)(ESS(I),I=1,37)
C9    THE FOLLOWING CARDS THROUGH CARD NO. 66 ARE USED TO FIND THE
C     DATA FOR THE PROBLEM  ON THE DATA   TAPE
      IF(ABS(AID(JSD)-SS(13))-0.00001)20,20,21
   21 NRK=SS(30)+0.1
C10   SKIP NRK RECORDS ON THE DATA TAPE
      DO 666 I=1,NRK
  666 READ(MG)DUMMY
      GO TO 777
C11   THE NEXT 4 CARDS ARE FOR WRITING DATA ON THE SCRATCH TAPES
   20 WRITE(MT)(ESS(I),I=1,37)
      WRITE(MTT)(ESS(I),I=1,37)
```

```
      WRITE(MMT)(ESS(I),I=1,37)
      WRITE(MMMT)(ESS(I),I=1,37)
      SSS=SS(13)
      IF(SS(26))39,39,85
 85   WRITE(6,40) SSS
 40   FORMAT(12H1NUCLIDE NO. F10.7,1X,13HHAS 1-D ARRAY
      NX=SS(26)+0.1
C     SS(26)=NO. OF ONE-DIMENSIONAL ARRAYS
      LNX=6+NEV
      NT=NX*LNX
C     READ 1-D CROSS SECTIONS
      READ(MG)(TTT(I),I=1,NT)
C12   THE NEXT TWO CARDS CALCULATES THE FINE GROUP MACROSCOPIC CROSS
C     SECTIONS
      DO 33 I=1,NEV
   33 SIGA(I)=SIGA(I)+TTT(I+6)*DENT(JSD)
      IS1=LNX+6
      IF(SS(25))8997,8997,8996
C13   SS(25) INDICATES WHETHER A FISSION CROSS SECTION IS INCLUDED FOR
C     THIS NUCLIDE
 8996 DO 8999 I=1,NEV
C14   THE NEXT 3 CARDS CALCULATE THE FINE GROUP MACRO. CROSS SECTION
      IS=IS1+I
 8999 SIGQ(I)=SIGQ(I)+TTT(IS)*DENT(JSD)
      IS1=IS+6
C15   THE NEXT 3 CARDS ASSIGN THE FINE GROUP NU VALUES TO ENG(I)
      DO 9010  I=1,NEV
      IS=IS1+I
 9010 ENG(I)=ENG(I)+TTT(IS)
 8997 CONTINUE
   39 IF(SS(16))91,91,41
   91 WRITE(6,50)SSS
 50   FORMAT(12H0NUCLIDE NO. F10.7,1X,37HDOES NOT HAVE P-0,P-1,P-2,P-3 A
     XRRAY
      GO TO 98
   41 WRITE(6,42)SSS
 42   FORMAT(12H0NUCLIDE NO. F10.7,1X,29HHAS P-0,P-1,P-2,AND P-3 ARRAY )
      LT=SS(16)+0.1
C     READ AND STORE P-0 ARRAY
      READ(MG)(TTT(I),I=1,LT)
      DO 4000 I=1,LT
C17   THE NEXT CARD CALCULAFES MACRO. CROSS SECTIONS
 4000 TTT(I)=TTT(I)*DENT(JSD)
      WRITE(MT)SS(13),SS(17),SS(18),LT,(TTT(I),I=1,LT)
C16   READ AND STORE P1 ARRAY
      READ(MG)(TTT(I),I=1,LT)
      DO 4444  I=1,LT
 4444 TTT(I)=TTT(I)*DENT(JSD)
      WRITE(MMMT)SS(13),SS(17),SS(18),LT,(TTT(I),I=1,LT)
C17   READ AND STORE P2 ARRAY
      READ(MG)(TTT(I),I=1,LT)
      DO 4445  I=1,LT
 4445 TTT(I)=TTT(I)*DENT(JSD)
      WRITE(MMMT)SS(13),SS(17),SS(18),LT,(TTT(I),I=1,LT)
C19   READ AND STORE P3 ARRAY
      READ(MG)(TTT(I),I=1,LT)
      DO 4446  I=1,LT
```

```
4446 TTT(I)=TTT(I)*DENT(JSD)
     WRITE(MMMT)SS(13),SS(17),SS(18),LT,(TTT(I),I=1,LT)
C    READ AND STORE INELASTIC ARRAY
  98 IF(SS(19))82,82,73
  82 WRITE(6,53)SSS
 53  FORMAT(12HONUCLIDE NO. F10.7,1X,29HDOES NOT HAVE INELASTIC ARRAY )
     GO TO 89
  73 WRITE(6,43)SSS
 43  FORMAT(12HONUCLIDE NO. F10.7,1X,19HHAS INELASTIC ARRAY          )
     LT=SS(19)+0.1
     READ(MG)(TTT(I),I=1,LT)
     DO 4001 I=1,LT
4001 TTT(I)=TTT(I)*DENT(JSD)
     WRITE(MTT)SS(13),SS(20),SS(21),LT,(TTT(I),I=1,LT)
C    READ AND STORE N-2N ARRAY
  89 IF(SS(22))92,92,83
 92  WRITE(6,54)SSS
 54  FORMAT(12HONUCLIDE NO. F10.7,1X,24HDOES NOT HAVE N-2N ARRAY     )
     GO TO 74
  83 WRITE(6,44)SSS
 44  FORMAT(12HONUCLIDE NO. F10.7,1X,14HHAS N-2N ARRAY               )
     LT=SS(22)+0.1
     READ(MG)(TTT(I),I=1,LT)
     DO 4002 I=1,LT
4002 TTT(I)=TTT(I)*DENT(JSD)
     WRITE(MMT)SS(13),SS(23),SS(24),LT,(TTT(I),I=1,LT)
C    READ SIGMA TOTAL
  74 READ (MG)(TTT(I),I=1,NEV)
C    CALCULATE TOTAL MACROSCOPIC CROSS SECTION
     DO 45 I=1,NEV
  45 SIGT(I)=SIGT(I)+TTT(I)*DENT(JSD)
C    READ TOTAL ISOTROPIC SCATTER ARRAY(P-0+INEL+2*N-2N)
     IF(SS(27))105,105,94
 105 WRITE(6,60)SSS
 60  FORMAT(12HONUCLIDE NO. F10.7,1X,29HDOES NOT HAVE TOTAL ISO.ARRAY)
     CALL EXIT
  94 WRITE(6,46)SSS
 46  FORMAT(12HONUCLIDE NO. F10.7,1X,28HHAS TOTAL ISO. SCATTER ARRAY
     LT=SS(27)+0.1
     READ(MG)(TTT(I),I=1,LT )
     DO 4004 I=1,LT
4004 TTT(I)=TTT(I)*DENT(JSD)
     WRITE(MSS)SS(13),SS(28),SS(29),LT,(TTT(I),I=1,LT)
C    READ SIGMA SCATTER TOTAL FOR P-0 ARRAY
     IF(SS(16))111,111,106
 106 READ(MG)(TTT(I),I=1,NEV)
C    READ SIGMA SCATTER TOTAL FOR P-1 ARRAY
     READ(MG)(TTT(I),I=1,NEV)
 111 CONTINUE
     CALL REW(MSS)
C21  SUBROUTINE REW REWINDS A TAPE AND SETS TT(K,KK) EQUAL TO 0.
     DO 120 JJ=1,NNUK
     READ(MSS)SS(13),SS(28),SS(29),LT,(TTT(I),I=1,LT)
     LDF=SS(28)    +0.1
C22  LDF = THE NUMBER OF GROUPS SCATTERED FROM
     LD=SS(29)     +0.1
C23  LD= NUMBER OF GROUPS SCATTERED TO
```

```
      CALL ONE(1)
C24   SUBROUTINE ONE SETS UP THE FINE GROUP MACRO. SCATTERING
C     CROSS SECTIONS FOR A TWO ARRAY VARIABLE, TT(K,KK)
  120 CONTINUE
C     CALCULATE FLUX OR READ IN FLUX
      IF(KKK-1)125,126,126
  125 READ(5,127)(FLUX(I),I=1,99)
  127 FORMAT(6E12.6)
      GO TO 315
  126 READ(5,127)SSSS
C25   THE FOLLOWING CARDS THROUGH CARD NO. 29 ARE FOR CALCULATING
C     THE FLUX
      FLUX(1)=(1./(SIGT(1)-TT(1,1)))*SSSS(1)
      DO 29 LL=2,99
      SUM=0.0
      KKKK=LL-1
      DO 30 J=1,KKKK
   30 SUM=SUM+FLUX(J)*TT(J,LL)
   29 FLUX(LL)=(1./(SIGT(LL)-TT(LL,LL)))*(SSSS(LL)+SUM)
  315 MM=1
      TT(100,100)=0.0
      FLUX(100)=FLUX(99)
C26   THE NEXT 8 CARDS ARE FOR PRINTING OUT THE FLUX
      WRITE(6,1002)(BXCX(I),I=1,18)
      WRITE(6,211)
  211 FORMAT(1H0/7H  GROUP,12H       FLUX
      WRITE(6,215)(I,FLUX(I),I=1,50)
  215 FORMAT(1I6,1PE20.6)
      WRITE(6,1002)(BXCX(I),I=1,18)
      WRITE(6,211)
      WRITE(6,215)(I,FLUX(I),I=51,99)
C     CALCULATE BROAD GROUP MACRO. ABS. AND TOTAL CROSS SECTION
C27   THE FOLLOWING CARDS THROUGH 9019 CALCULATE THE BROAD GROUP ABS.
C     TOTAL,FISSION,AND NU*FISSION CROSS SECTIONS
      DO 201 I=1,NBBG
      SUM=0.0
      BUM=0.0
      SUMM=0.0
      SUMMM=0.0
      TUM=0.0
      III=LBGB(I)
      DO 200 II=MM,III
      BUM=BUM+ENG(II)*FLUX(II)*SIGQ(II)
      SUMMM=SUMMM+SIGQ(II)*FLUX(II)
      SUM=SUM+SIGT(II)*FLUX(II)
      SUMM=SUMM+FLUX(II)
  200 TUM=TUM+SIGA(II)*FLUX(II)
      FTOT(I)=SUMMM/SUMM
      TOT(I)=SUM/SUMM
      ABBS(I)=TUM/SUMM
      TRA(I)=BUM/SUMM
  201 MM=LBGB(I)+1
      WRITE(6,1002)(BXCX(I),I=1,18)
      WRITE(6,312)
  312 FORMAT(48H0BROAD GROUP AVERAGED MACROSCOPIC CROSS SECTIONS/1H0)
      WRITE(6,217)
  217 FORMAT(6H0GROUP,20H    SIGMA ABSORPTION,20H       SIGMA TOTAL  ,20
```

```
      XH      SIGMA FISSION
      WRITE(6,218)(I,ABBS(I),TOT(I),FTOT(I),I=1,NBBG)
 218  FORMAT(1I4,1PE20.6,1PE18.6,1PE18.6)
      WRITE(6,1002)(BXCX(I),I=1,18)
      WRITE(6,9009)
 9009 FORMAT(6HOGROUP,28H          NU*SIGMA FISSION
 9019 FORMAT(1I4,1PE28.6)
      WRITE(6,9019)(I,TRA(I),I=1,NBBG)
C     CALCULATE TOTAL BROAD GROUP ISO.TRANSFER SCATTER CROSS SECTION
      CALL CSAV(SISO)
C28   THE SUBROUTINE CSAV FLUX WEIGHTS  FINE GROUP SCATTERING TRANSFER
C     CROSS SECTIONS
C     CALCULATE P-0 MACRO. CROSS SECTIONS
      CALL REW(MT)
      DO 2010 JM=1,NNUK
      READ(MT)(ESS(I),I=1,37)
      IF(SS(16))2010,2010,3000
 3000 READ(MT)SS(13),SS(17),SS(18),LT,(TTT(I),I=1,LT)
      LDF=SS(17)    +0.1
      LD=SS(18)     +0.1
      CALL ONE(1)
 2010 CONTINUE
      CALL CSAV(P0)
C29   THE NEXT 10 CARDS ARE FOR CALCULATING THE BROAD GROUP INELLASTIC
C     SCATTERING TRANSFER CROSS SECTIONS
      CALL REW(MTT)
      DO 3021 JM=1,NNUK
      READ(MTT)(ESS(I),I=1,37)
      IF(SS(19))3021,3021,3022
 3022 READ(MTT)SS(13),SS(20),SS(21),LT,(TTT(I),I=1,LT)
      LDF=SS(20)    +0.1
      LD=SS(21)     +0.1
      CALL ONE(1)
 3021 CONTINUE
      CALL CSAV(XINELS)
C30   THE NEXT 10 CARDS ARE FOR CALCULATING THE BROAD GROUP  N-2N
C     SCATTERING TRANSFER CROSS SECTIONS
      CALL REW(MMT)
      DO 3027 JM=1,NNUK
      READ(MMT)(ESS(I),I=1,37)
      IF(SS(22))3027,3027,3028
 3028 READ(MMT)SS(13),SS(23),SS(24),LT,(TTT(I),I=1,LT)
      LDF=SS(23)    +0.1
      LD=SS(24)     +0.1
      CALL ONE(2)
 3027 CONTINUE
      CALL CSAV(XN2N)
C31   THE NEXT 10 CARDS ARE FOR CALCULATING THE P1 SCATTERING TRANSFER
C     CROSS SECTIONS
      CALL REW(MMMT)
      DO 7330  II=1,NNUK
      READ(MMMT)(ESS(I),I=1,37)
      IF(SS(16))7330,7330,7331
 7331 READ(MMMT)SS(13),SS(17),SS(18),LT,(TTT(I),I=1,LT)
      READ(MMMT)DUMMY
      READ(MMMT) DUMMY
      LDF=SS(17)+0.1
```

```
      LD=SS(18)+0.1
      CALL ONE(1)
 7330 CONTINUE
      CALL CSAV(P1)
C32   THE BROAD GROUP  P2 SCATTERING TRANSFER CROSS SECTION CALC FOLLOW
      CALL REW(MMMT)
      DO 7332  II=1,NNUK
      READ(MMMT)(ESS(I),I=1,37)
      IF(SS(16))7332,7332,7333
 7333 READ(MMMT)DUMMY
      READ(MMMT)SS(13),SS(17),SS(18),LT,(TTT(I),I=1,LT)
      READ(MMMT)DUMMY
      LDF=SS(17)+0.1
      LD=SS(18)+0.1
      CALL ONE(1)
 7332 CONTINUE
      CALL CSAV(P2)
C33   THE P3 BROAD GROUP SCATTERING TRANSFER CROSS SECT CALC FOLLOW
      CALL REW(MMMT)
      DO 7334  II=1,NNUK
      READ(MMMT)(ESS(I),I=1,37)
      IF(SS(16))7334,7334,7335
 7335 READ(MMMT)DUMMY
      READ(MMMT)DUMMY
      READ(MMMT)SS(13),SS(17),SS(18),LT,(TTT(I),I=1,LT)
      LDF=SS(17)+0.1
      LD=SS(18)+0.1
      CALL ONE(1)
 7334 CONTINUE
      CALL CSAV(P3)
      WRITE(6,1002)(BXCX(I),I=1,18)
C34   THE NEXT 33 CARDS ARE FOR PRINTING OUT BROAD GROUP CROSS SECTION
      WRITE(6,312)
      WRITE(6,551)
 551  FORMAT(100H0GROUP                       P-0           INELASTIC          N-2N
     *        TOTAL SCATTER
      LPC=0
      DO 4100 LL=1,NBBG
      DO 3200 I=LL,NBBG
      LPC=LPC+1
      IF(LPC-25)3200,555,555
 555  WRITE(6,1002)(BXCX(N),N=1,18)
      WRITE(6,312)
      WRITE(6,551)
      LPC=0
 3200 WRITE(6,38)LL,I,P0(LL,I),XINELS(LL,I),XN2N(LL,I),SISO(LL,I)
 4100 CONTINUE
 38   FORMAT(5H0FROM,1I3,3H TO,1I3,1PE15.6,1PE15.6,1PE15.6,1PE15.6)
      WRITE(6,1002)(BXCX(I),I=1,18)
      WRITE(6,312)
      WRITE(6,5511)
 5511 FORMAT(54H0                        P-1              P-2              P-3  )
      LPC=0
      DO 1400  LL=1,NBBG
      DO 2300  I=LL,NBBG
      LPC=LPC+1
      IF(LPC-25)2300,5555,5555
```

```
5555 WRITE(6,1002)(BXCX(N),N=1.18)
     WRITE(6,312)
     WRITE(6,5511)
     LPC=0
2300 WRITE(6,3888)LL,I,P1(LL,I),P2(LL,I),P3(LL,I)
1400 CONTINUE
3888 FORMAT(5H0FROM,1I3,3H TO,1I3,1PE15.6,1PE15.6,1PE15.6)
     WRITE(6,1002)(BXCX(I),I=1,18)
     WRITE(6,1424)
1424 FORMAT(65H0GROUP   AVERAGE COS(THETA)   DIFF. COEFF.   SIGMA TRANS
    *PORT                                                         )
C35  THE NEXT 5 CARDS ARE FOR CALCULATING  THE TRANSPORT CROSS SECTION
C    THE DIFFUSION COEFF. AND THE AVG. COSINE OF THE SCATT. ANGLE
1425 FORMAT(1I5,1PE17.4,1PE18.4,1PE17.4)
     DO 1431  LL=1,NBBG
     SIGTR=TOT(LL)-P1(LL,LL)/3.
     D=1./(3.*SIGTR)
     XMUBAR=(P1(LL,LL)/3.)/SISO(LL,LL)
1431 WRITE(6,1425)LL,XMUBAR,D,SIGTR
     REWIND MG
     STOP
     END
$IBFTC REWE
     SUBROUTINE REW(N)
C36  SUBROUTINE REW REWINDS A DATA TAPE AND SETS TT(K,KK)=0.0
     COMMON TTT(5050),TT(100,100),LDF,LD,LT,NBBG,LBGB(99),FLUX(100)
     REWIND N
     DO 3043 K=1,100
     DO 3043 KK=1,100
3043 TT(K,KK)=0.0
     RETURN
     END
$IBFTC ONEE
     SUBROUTINE ONE(JJJ)
C37  THIS SUBROUTINE SETS UP THE MACRO. SCATTERING TRANSFER CROSS
C    SECTIONS IN TERMS OF TT(K,KK)
     COMMON TTT(5050),TT(100,100),LDF,LD,LT,NBBG,LBGB(99),FLUX(100)
     N=1
     KK=0
     NSN=LD
     BB=JJJ
     DO 122 K=1,LDF
     DO 123 L=N,LD
     KK=KK+1
123  TT(K,KK)=BB*TTT(L)+TT(K,KK)
     N=LD+1
     KK=K
     NNN=K+NSN
     IF(NNN-100)101,101,124
124  NDIF=NNN-100
     LD=NSN+LD-NDIF
     GO TO 122
101  LD=NSN+LD
122  CONTINUE
     RETURN
     END
$IBFTC CSAVE
```

```
      SUBROUTINE CSAV(SIG)
C38   THIS SUB. FLUX WEIGHTS THE FINE GROUP SCATTERING TRANSFR CROSS SEC
      COMMON TTT(5050),TT(100,100),LDF,LD,LT,NBBG,LBGB(99),FLUX(100)
      DIMENSION SIG(22,22)
      N=1
      NNNN=1
      DO 41 LL=1,NBBG
      DO 32 I=LL,NBBG
      X=0.0
      FF=0.0
      KK=LBGB(LL)
      DO 25 J=N,KK
      F=0.0
      MMM=LBGB(I)
      DO 10 K=NNNN,MMM
      IF(J.GT.K) GO TO 10
      F=FLUX(J)*TT(J,K)+F
 10   CONTINUE
      FF=F+FF
   25 X=X+FLUX(J)
      NNNN=LBGB(I)+1
      SIG(LL,I)=FF/X
   32 CONTINUE
      N=LBGB(LL)+1
      NNNN=LBGB(LL)+1
   41 CONTINUE
      RETURN
      END
```

APPENDIX B

ENDF/B Source Deck of New Barnyard

On the following pages is listed the source deck that uses the two ENDF/B data tapes. It is written in the Fortran IV language for use on the IBM 7094 digital computer. It consists of a main program and 4 separate subroutines.

Input data from data cards is "read in" in the main program. The cross section data required for the problem is obtained in the subroutine named FIB which reads the data from the data tape. Three scratch tapes are used to store the PN scattering transfer cross sections until they are needed in the main program. The flux calculation, and the broad group calculations of the transport cross section, diffusion coefficient, and the average cosine of the scattering angle are performed in the main program.

The subroutines named REW, ONE, and CSAV perform the same calculations as the subroutines with the same names in the GGC-4 source deck. These subroutines were discussed briefly in Appendix A.

Extra comment cards have been added to the listing that follows so that it will be easier to read. A glossary of computer program symbols is given in Appendix C.

```
C       NEW BARNYARD
C       MULTIGROUP NEUTRON MACROSCOPIC CROSS SECTION CODE
C       THIS SOURCE DECK USES THE ENDF/B CROSS SECTION DATA TAPES
        DIMENSION A(21),P0(20,20),P1(20,20),P2(20,20),P3(20,20),
      1TOT(22),ABBS(22),SSSS(99),BXCX(18)
        COMMON DENT(29),NNUK,TT(103,100),B(10300),NBBG,FLUX(100),
      *LBGB(22),ID,N
        DATA C1/4H..../
C1      SETUP SCRATCH TAPE NUMBERS AND READ IN DATA FROM DATA CARDS
        NTCH1=3
        NTCH2=4
        NGP=100
        NGP3=NGP+3
        N1=0
        ITSN=8
        MODE=2
        READ(5,1006)(BXCX(I),I=1,18)
C0      CARDS FROM CARD NUMBERS 1006 TO 34 ARE FOR READING IN DATA  AND
C01     FOR STORING THIS DATA ON A SCRATCH TAPE
 1006   FORMAT(18A4)
 1002   FORMAT(1H1,18A4)
        READ(5,9871) NBBG,NNUK,KKK
        READ(5,7)(LBGB(I),I=1,NBBG)
    7   FORMAT(24I3)
 9871   FORMAT(3I3)
 2      FORMAT(1X,A4,2I2,1I6,1E13.4)
        DO 34  JKX=1,NNUK
        READ(5,2) MATNO,LORDER, N, NOR, DENT(JKX)
 61     CONTINUE
        N1=N1+100
        X=MATNO
        LOR1=9
        IF(NOR.EQ.0) GO TO 1001
C2      SKIP NOR RECORDS ON THE ENDF/B DATA TAPE
        DO 62  I=1,NOR
 62     READ(N,24)DUMMY
 24     FORMAT(1A4)
C3      READ AND CHECK DATA FROM ENDF/B DATA TAPE
 1001   READ(N,20)(A(I),I=1,21)
        WRITE(6,20)(A(I),I=1,21)
        IF(A(1).NE.C1.AND.A(11).NE.X) GO TO 67
        GO TO 14
 67     WRITE(6,63)
        GO TO 33
 63     FORMAT(37H0 ERROR IN SKIPPING DATA ---CHECK NOR)
 14     L=0
 15     CONTINUE
        CALL FIB(NGP,NGP3)
C       SUBROUTINE FIB IS FOR READING THE ENDF/B DATA TAPE
        ID=N1+L
        IF(L.GT.LORDER) GO TO 166
C4      WRITE CROSS SECTION DATA FOR THIS PROBLEM ON ITSN SCRATCH TAPE
        WRITE(ITSN)(B(K),K=1,10300)
C5      WRITE CROSS SECTION DATA  INFO. THAT WILL BE USED FOR THIS PROBLEM
        WRITE(6,23) MATNO,L,ID
 166    IF(L.EQ.8) GO TO 34
        READ(N,20)(A(I),I=1,21)
```

```
      WRITE(6,20)(A(I),I=1,21)
      IF(A(1) .EQ. C1) GO TO 16
      WRITE(6,64)
 64   FORMAT(27HOA(1) IS NOT EQUAL TO ....
      GO TO 33
 16   CONTINUE
      L=L+1
      IF(L.EQ.LOR1) GO TO 300
      IF(MODE.NE.0) GO TO 15
 300  CONTINUE
 34   CONTINUE
 20   FORMAT(9A4,A1,10A4,A3)
 21   FORMAT(5X,A4,4X,1HP,I1,3X,I5,2X,8H RECORDS)
 22   FORMAT(1H1,26X,23H **** MATERIAL NUMBER  ,A4,6X,I3,8H  GROUPS,
     *   5X,1HP,I1,6H **** /)
 23   FORMAT(1H ,28H THE ID NUMBER FOR MATERIAL ,A4,4H   P,I1,4H  IS,I6)
      CALL REW(ITSN)
C     READ P-0 DATA FROM THE BINARY SCRATCH TAPE
      DO 73  I=1,NNUK
      READ(ITSN)(B(K),K=1,10300)
      DO 75  K=1,NGP
      DO 75 J=1,NGP3
      KK=NGP3*(K-1)+J
C6    CARD 75 SETS UP A 2 ARRAY VARIABLE FOR THE SCATTERING TRANSFER
C60   CROSS SECTIONS AND IT CALCULATES THE MACROSCOPIC CROSS SECTION
 75   TT(J,K)=B(KK)*DENT(I)+TT(J,K)
      JY=LORDER
      DO 74  JI=1,JY
C7    SKIP JYN RECORDS ON THE BINARY SCRATCH TAPE
 74   READ(ITSN)DUMMY
 73   CONTINUE
C     CALCULATE FLUX OR READ IN FLUX
      IF(KKK-1)125,126,126
C8    READ IN FLUX VALUES
  125 READ(5,127)(FLUX(I),I=1,99)
  127 FORMAT(6E12.6)
      GO TO 315
C9    READ IN SOURCE VALUES
 126  READ(5,127)SSSS
C10   THE FOLLOWING CARDS THROUGH CARD NO. 29 ARE FOR CALCULATING THE
C100  FLUX
      FLUX(1)=(1./(TT(3,1)-TT(4,1)))*(SSSS(1)+SUM)
      DO 29 LL=2,99
      SUM=0.0
      KKKK=LL-1
      DO 30 J=1,KKKK
      NYN=LL-J+4
 30   SUM=SUM+FLUX(J)*TT(NYN,LL)
 29   FLUX(LL)=(1./(TT(3,LL)-TT(4,LL)))*(SSSS(LL)+SUM)
  315 MM=1
      FLUX(100)=FLUX(99)
C11   PRINT OUT  THE FLUX SPECTRUM
      WRITE(6,1002)(BXCX(I),I=1,18)
      WRITE(6,211)
 211  FORMAT(1H0/7H  GROUP,12H        FLUX
      WRITE(6,215)(I,FLUX(I),I=1,50)
 215  FORMAT(I6,1PE20.6)
```

```
      WRITE(6,1002)(BXCX(I),I=1,18)
      WRITE(6,211)
      WRITE(6,215)(I,FLUX(I),I=51,99)
C     CALCULATE BROAD GROUP MACRO. ABS. AND TOTAL CROSS SECTION
C12   THE FOLLOWING CARDS THROUGH CARD NO. 218 ARE FOR CALCULATING THE
C120  TOTAL, ABSORPTION, AND NU*FISSION CROSS SECTIONS
      DO 201 I=1,NBBG
      SUM=0.0
      BUM=0.0
      SUMM=0.0
      TUM=0.0
      III=LBGB(I)
      DO 200 II=MM,III
      SUM=SUM+TT(3,II)*FLUX(II)
      BUM=BUM+TT(2,II)*FLUX(II)
      SUMM=SUMM+FLUX(II)
  200 TUM=TUM+TT(1,II)*FLUX(II)
      TOT(I)=SUM/SUMM
      ABBS(I)=TUM/SUMM
      SSSS(I)=BUM/SUMM
  201 MM=LBGB(I)+1
      WRITE(6,1002)(BXCX(I),I=1,18)
      WRITE(6,312)
  312 FORMAT(48H0BROAD GROUP AVERAGED MACROSCOPIC CROSS SECTIONS/1H0)
      WRITE(6,217)
  217 FORMAT(6H0GROUP,20H    SIGMA ABSORPTION,20H        SIGMA TOTAL   ,20
     *H  NU*SIGMA FISSION                 )
      WRITE(6,218)(I,ABBS(I),TOT(I),SSSS(I),I=1,NBBG)
  218 FORMAT(1I4,1PE20.6,1PE18.6,1PE20.6)
      CALL CSAV(P0)
C13   SUBROUTINE CSAV IS FOR FLUX WEIGHTING THE SCATTERING TRANSFER
C130  CROSS SECTIONS
      DO 94  LL=1,NBBG
      DO 95  I=LL,NBBG
      P1(LL,I)=0.0
      P2(LL,I)=0.0
   95 P3(LL,I)=0.0
   94 CONTINUE
C     THE NEXT CARD IS A CHECK TO SEE IF P1 CROSS SECTIONS ARE WANTED
      IF(LORDER.GE.1) GO TO 78
      GO TO 86
   78 CALL REW(ITSN)
C14   SUBROUTINE REW IS FOR REWINDING THE BINARY SCRATCH TAPE AND FOR
C140  SETTING  T(KK,K) EQUAL TO 0.
C     READ P-1 DATA FROM THE BINARY SCRATCH TAPE
C     OBTAIN    P-1 SCATTERING TRANSFER MACROSCOPIC CROSS SECTIONS
C     FOR THE 99 GROUPS
      JY=LORDER-1
      CALL ONE(ITSN,1,JY)
C15   SUBROUTINE ONE IS FOR READING THE DATA FROM THE SCRATCH TAPE AND
C150  AND FOR ASSIGNING THE CROSS SECTION DATA TO A VARIABLE
C1500 THAT HAS TWO DIMENSIONS AND  FOR CALCULATING  MACRO. CROSS SECTS
      CALL CSAV(P1)
      IF(LORDER.GE.2) GO TO 83
      GO TO 86
   83 CALL REW(ITSN)
C     READ P-2 DATA FROM THE BINARY SCRATCH TAPE
```

```
C       OBTAIN     P-2 SCATTERING TRANSFER MACROSCOPIC CROSS SECTIONS
C       FOR THE 99 GROUPS
        JY=LORDER-2
        CALL ONE(ITSN,2,JY)
        CALL CSAV(P2)
        IF(LORDER.GE.3) GO TO 84
        GO TO 86
 84     CALL REW(ITSN)
C       READ P-3 DATA FROM THE BINARY SCRATCH TAPE
C       OBTAIN     P-3 SCATTERING TRANSFER MACROSCOPIC CROSS SECTIONS
C       FOR THE 99 GROUPS
        JY=LORDER-3
        CALL ONE(ITSN,3,JY)
        CALL CSAV(P3)
 86     DO 7899  LL=1,NBBG
        DO 7870  I=LL,NBBG
C16     WRITE OUT THE BROAD GROUP SCATTERING TRANSFER CROSS SECTIONS
C160    ON A SCRATCH TAPE FOR TEMPORARY STORAGE
 7870   WRITE(NTCH1)P0(LL,I),P1(LL,I),P2(LL,I),P3(LL,I)
 7899   CONTINUE
        IF(LORDER.LT.4) GO TO 500
 85     DO 51 LL=1,NBBG
        DO 52  I=LL,NBBG
        P0(LL,I)=0.0
        P1(LL,I)=0.0
        P2(LL,I)=0.0
 52     P3(LL,I)=0.0
 51     CONTINUE
        CALL REW(ITSN)
C       READ P-4 DATA FROM THE BINARY SCRATCH TAPE
C       OBTAIN     P-4 SCATTERING TRANSFER MACROSCOPIC CROSS SECTIONS
C       FOR THE 99 GROUPS
        JY=LORDER-4
        CALL ONE(ITSN,4,JY)
        CALL CSAV(P0)
        IF(LORDER.GE.5) GO TO 87
        GO TO 50
 87     CALL REW(ITSN)
C       READ P-5 DATA FROM THE BINARY SCRATCH TAPE
C       OBTAIN     P-5 SCATTERING TRANSFER MACROSCOPIC CROSS SECTIONS
C       FOR THE 99 GROUPS
        JY=LORDER-5
        CALL ONE(ITSN,5,JY)
        CALL CSAV(P1)
        IF(LORDER.GE.6) GO TO 88
        GO TO 50
 88     CALL REW(ITSN)
C       READ P-6 DATA FROM THE BINARY SCRATCH TAPE
C       OBTAIN     P-6 SCATTERING TRANSFER MACROSCOPIC CROSS SECTIONS
C       FOR THE 99 GROUPS
        JY=LORDER-6
        CALL ONE(ITSN,6,JY)
        CALL CSAV(P2)
        IF(LORDER.GE.7) GO TO 89
        GO TO 50
 89     CALL REW(ITSN)
C       READ P-7 DATA FROM THE BINARY SCRATCH TAPE
```

```
C       OBTAIN    P-7 SCATTERING TRANSFER MACROSCOPIC CROSS SECTIONS
C       FOR THE 99 GROUPS
        JY=LORDER-7
        CALL ONE(ITSN,7,JY)
        CALL CSAV(P3)
 50     DO 789 LL=1,NBBG
        DO 787  I=LL,NBBG
C17     WRITE OUT THE BROAD GROUP SCATTERING TRANSFER CROSS SECTIONS
C       FOR P4 THROUGH P7
 787    WRITE(NTCH2)P0(LL,I),P1(LL,I),P2(LL,I),P3(LL,I)
 789    CONTINUE
 500    REWIND NTCH1
        IF(LORDER.LT.4) GO TO 501
        REWIND NTCH2
C18     CARDS FROM CARD NO. 501 TO 551 ARE FOR PRINTING OUT CROSS SECTION
C       VALUES
 501    WRITE(6,1002)(BXCX(I),I=1,18)
        WRITE(6,312)
        WRITE(6,551)
        LPC=0
        DO 4100 LL=1,NBBG
        DO 3200 I=LL,NBBG
        READ(NTCH1)P0(LL,I),P1(LL,I),P2(LL,I),P3(LL,I)
        LPC=LPC+1
        IF(LPC-25)3200,555,555
 555    WRITE(6,1002)(BXCX(NZ),NZ=1,18)
        WRITE(6,312)
        WRITE(6,551)
        LPC=0
 3200   WRITE(6,38)LL,I,P0(LL,I),P1(LL,I),P2(LL,I),P3(LL,I)
 4100   CONTINUE
 38     FORMAT(5H0FROM,1I3,3H TO,1I3,1PE15.6,1PE15.6,1PE15.6,1PE15.6)
 3888   FORMAT(5H0FROM,1I3,3H TO,1I3,1P4E15.6)
 5511   FORMAT(90H0GROUP                        P-4             P-5             P-6
       *           P-7                                                            )
 551    FORMAT(100H0GROUP                        P-0             P-1             P-2
       *           P-3                                                            )
        WRITE(6,1424)
 1424   FORMAT(1H1,65H GROUP   AVERAGE COS(THETA)   DIFF. COEFF.   SIGMA T
       *RANSPORT
 1425   FORMAT(1I5,1PE17.4,1PE18.4,1PE17.4)
C19     BEGIN CALCULATION OF THE TRANSPORT CROSS SECTION, DIFFUSION
C       COEFFICIENT AND THEAVG. COSINE OF THE   SCATTERING ANGLE
        DO 1431  LL=1,NBBG
        SIGTR=TOT(LL)-P1(LL,LL)/3.
        D=1./(3.*SIGTR)
        XMUBAR=(P1(LL,LL)/3.)/P0(LL,LL)
 1431   WRITE(6,1425)LL,XMUBAR,D,SIGTR
        IF(LORDER.LT.4) GO TO 33
C20     READ THE SCRATCH TAPE AND PRINT OUT THE BROAD GROUP SCATTERING
C       TRANSFER CROSS SECTIONS
        WRITE(6,1002)(BXCX(I),I=1,18)
        WRITE(6,312)
        WRITE(6,5511)
        LPC=0
        DO 1400  LL=1,NBBG
        DO 2300  I=LL,NBBG
```

```
      READ(NTCH2)P0(LL,I),P1(LL,I),P2(LL,I),P3(LL,I)
      LPC=LPC+1
      IF(LPC-25)2300,5555,5555
 5555 WRITE(6,1002)(BXCX(NZ),NZ=1,18)
      WRITE(6,312)
      WRITE(6,5511)
      LPC=0
 2300 WRITE(6,3888)LL,I,P0(LL,I),P1(LL,I),P2(LL,I),P3(LL,I)
 1400 CONTINUE
      IF(LORDER.EQ.8) GO TO 90
      GO TO 33
 90   CALL REW(ITSN)
C     READ P-8 DATA FROM THE BINARY SCRATCH TAPE
C     OBTAIN    P-8 SCATTERING TRANSFER MACROSCOPIC CROSS SECTIONS
C     FOR THE 99 GROUPS
      JY=LORDER-8
      CALL ONE(ITSN,8,JY)
      CALL CSAV(P0)
      WRITE(6,1002)(BXCX(I),I=1,18)
      WRITE(6,312)
      WRITE(6,5595)
 5595 FORMAT(25H0GROUP                  P-8
      LPC=0
      DO 5678  LL=1,NBBG
      DO 5679  I=LL,NBBG
      LPC=LPC+1
      IF(LPC-25)5679,4568,4568
 4568 WRITE(6,1002)(BXCX(NZ),NZ=1,18)
      WRITE(6,312)
      WRITE(6,5595)
      LPC=0
 5679 WRITE(6,4569)LL,I,P0(LL,I)
 4569 FORMAT(5H0FROM,1I3,3H TO,1I3,1PE15.6)
 5678 CONTINUE
 33   STOP
      END
$IBFTC FIBB
      SUBROUTINE FIB(NGP,NGP3)
C22   THIS SUBROUTINE IS FOR READING THE ENDF/B  DATA TAPE AND FOR
C     ASSIGNING THE CROSS SECTION VALUES TO THE VARIABLE B(K)
      DIMENSION M(6),TEMP(6),WORD(6,1)
      COMMON DENT(29),NNUK,TT(103,100),B(10300),NBBG,FLUX(100),
     *LBGB(22),ID,N
 5    FORMAT(6(I2,1X,E9.0))
      JT=NGP*NGP3
      J=0
 10   READ(N,5)(M(K),TEMP(K),K=1,6)
      DO 7  K=1,6
 7    WORD(K,1)=TEMP(K)
      DO 15  K=1,5
      IF(WORD(K,1) .EQ. 0.0 .AND. WORD(K+1,1) .EQ. 0.0) GO TO 20
      NA=K+1
      GO TO 15
 20   NA=K
      GO TO 21
 15   CONTINUE
 21   DO 35  K=1,NA
```

```
      IF(M(K).GT.1) GO TO 30
      J=J+1
      B(J)=TEMP(K)
      IF(J.GE.JT) GO TO 40
      GO TO 35
 30   IF(M(K).EQ.0) GO TO 35
      M(K)=M(K)-1
      J=J+1
      B(J)=TEMP(K)
      IF(J.GE.JT) GO TO 40
      GO TO 30
 35   CONTINUE
      IF(J.LT.JT) GO TO 10
 40   CONTINUE
      RETURN
      END
$IBFTC REWE
      SUBROUTINE REW(NN)
C23   THIS SUBROUTINE IS FOR REWINDING TAPES AND FOR SETTING
C     TT(KK,K) EQUAL TO 0.
      COMMON DENT(29),NNUK,TT(103,100),B(10300),NBBG,FLUX(100),
     *LBGB(22),ID,N
      REWIND NN
      DO 3043  K=1,100
      DO 3043  KK=1,103
 3043 TT(KK,K)=0.0
      RETURN
      END
$IBFTC ONEE
      SUBROUTINE ONE(ITSN,JN,JY)
C24   THIS SUBROUTINE IS FOR READING THE CROSS SECTIONDATA FROM A
C     BINARY SCRATCH TAPE AND ASSIGNING THE DATA TO A TWO ARRAY
C     VARIABLE,   TT(J,K)
      COMMON DENT(29),NNUK,TT(103,100),B(10300),NBBG,FLUX(100),
     *LBGB(22),ID,N
      DO 80  JM=1,NNUK
      DO 79  I=1,JN
 79   READ(ITSN)DUMMY
      READ(ITSN)(B(K),K=1,10300)
      DO 81  K=1,100
      DO 81  J=1,103
      KK=103*(K-1)+J
 81   TT(J,K)=B(KK)*DENT(JM)+TT(J,K)
      IF(JY.EQ.0) GO TO 80
      DO 82  J=1,JY
 82   READ(ITSN)DUMMY
 80   CONTINUE
      RETURN
      END
$IBFTC CSAVE
      SUBROUTINE CSAV(SIG)
C25   THIS SUBROUTINE FLUX WEIGHTS THE FINE GROUP CROSS SECTIONS TO
C     TO OBTAIN BROAD GROUP CROSS SECTIONS
      DIMENSION SIG(20,20)
      COMMON DENT(29),NNUK,TT(103,100),B(10300),NBBG,FLUX(100),
     *LBGB(22),ID,N
      NB=1
```

```
      NNNN=1
      DO 41 LL=1,NBBG
      DO 32 I=LL,NBBG
      X=0.0
      FF=0.0
      KK=LBGB(LL)
      DO 25  J=NB,KK
      F=0.0
      MMM=LBGB(I)
      DO 10 K=NNNN,MMM
      IF(J.GT.K) GO TO 10
      NNY=K-J+4
      F=FLUX(J)*TT(NNY,K)+F
10    CONTINUE
      FF=F+FF
   25 X=X+FLUX(J)
      NNNN=LBGB(I)+1
      SIG(LL,I)=FF/X
   32 CONTINUE
      NB=LBGB(LL)+1
      NNNN=LBGB(LL)+1
   41 CONTINUE
      RETURN
      END
```

APPENDIX C

Glossary of Computer Program Symbols

The following Fortran IV nomenclature is used in both source decks of New Barnyard. The program themselves are listed in Appendices A and B.

Symbol	Meaning or Use
A(I)	: Alphameric variable* for temporary storage of information concerning cross section data for a nuclide
ABBS(I)	: Absorption cross section for the I-th broad group
AID(IXL)	: Nuclide I.D. number for the IXL-th nuclide
AT(I)	: Alphameric variable used for data tape description
B(K)	: Single array variable used for the temporary storage of cross section data
BUM	: Summing Variable
BXCX(I)	: Alphameric variable used for problem description
C1	: Alphameric variable used to check alphameric data from the data tapes
CSAV	: Subroutine that flux weights fine group cross sections

---

* Alphameric variable refers to a variable which is "read in" under an A format.

Appendix C (Contd')

Symbol	Meaning or use
D(I)	: Diffusion coefficient for the I-th broad group
DAD(I)	: Alphameric variable used for nuclide description
DD(1, IXL)	: Nuclide I.D. number of the IXL-th nuclide
DD(2, IXL)	: Resonance parameter of the IXL-th nuclide
DD(3,IXL)	: Resonance parameter of the IXL-th nuclide
DENT(IXL)	: Number density of IXL-th nuclide
DUMMY	: The first variable of a data record
ENG(I)	: Convenience variable for temporary storage of the i-th fine group boundary and also the i-th fine group average number of neutrons emitted per fission event
ESS(I)	: Alphameric variable used for nuclide description
F, FF	: Summing variables
FIB	: Subroutine used to read the ENDF/B data tapes
FLUX(I)	: Neutron flux for the I-th fine group
FNBT	: Data tape number on the GGC-4 data tape
FTOT(I)	: Fission cross section for the I-th broad group

Appendix C (Contd')

Symbol	Meaning or Use
FPL	: Convenience variable for problem constant
GTH(I)	: Energy boundary of the I-th thermal fine group
ID	: Convenience variable used as an identification number for a particular set of data
III	: Fine group number
IS, IX, I	: Summing variables
ISI	: Convenience variable for a problem constant
ITSN	: Logical unit number of scratch tape
J2	: Convenience variable for a problem constant
JNVN, JNY, JNYY	: Broad group numbers
JY, JT	: Convenience variable for problem constant
KK	: Convenience variable for designating the KK-th dimension of a dimensioned variable and also used for a fine group number
KKK	: Indicator for performing a flux calculation or inputting a flux spectrum
KKKK	: Convenience variable for temporary storage of a fine group number
LBS	: Convenience variable for problem constant

Appendix C (Contd')

Symbol	Meaning or Use
LD	: Number of fine groups scattered to
LDF	: Number of fine groups scattered from
LEN(I)	: I-th resonance parameter
LLBBG(I),LBGB(I)	: Lower broad group boundaries of the I-th group
LNX	: Length of I-D cross section array
LORDER	: Order of the PN scattering transfer cross section
LORI	: Convenience variable for problem constant
LPC, LRINK	: Summing variables
LT	: Length of cross section array
M(K)	: Convenience variable used for temporary storage of cross section data
MATNO	: Material number of a nuclide
MODE	: Convenience variable for problem constant
MG	: Logical unit number of GGC-4 data tape
MM	: Convenience variable for temporary storage of a fine group number
MMMT, MMT, MSS, MT, MTT	: Logical unit numbers of scratch tapes

Appendix C (Contd')

Symbol	Meaning or Use
N	: Logical unit number of ENDF/B data tape
NA	: Summing variable
NB	: Convenience variable used for temporary storage of a fine group number
NBBG	: Number of broad groups for a particular problem
NBT	: Data tape number on the GGC-4 data tape
NBTC	: Data tape number on the GGC-4 data tape
NEP	: Number of fine group energy boundaries
NES	: Convenience variable for problem constant
NEV	: Number of fine groups
NGP, NGP3	: Convenience variable for problem constant
NGT	: Number of thermal energy boundaries
NMORE	: Number of records that contain resonance data
NNOT	: Number of nuclides for which data is listed on the GGC-4 data tape
NNNN	: Convenience variable used for temporary storage of a fine group number

Appendix C (Contd')

Symbol	Meaning or Use
NNUK	: Number of nuclides for a particular problem
NOR	: Number of data records
NPR1	: Convenience variable for problem constant
NRK	: Number of data records for a nuclide
NSP	: Number of fission sources on the GGC-4 data tape
NT	: Convenience variable for problem constant
NTCH1, NTCH2	: Logical unit number of scratch tapes
NTID(I)	: Convenience variable used for temporary storage of miscellaneous data from the GGC-4 data tape
NX	: Number of I-D cross section arrays for a nuclide
ONE	: Subroutine for calculating macroscopic scattering transfer cross sections
P0(LL,I) P1(LL,I), P2(LL,I), P3(LL,I)	: Convenience variables used for temporary storage of PN scattering transfer cross section for scattering transfer cross section for scattering from broad group LL to broad group I
REW	: Subroutine used for rewinding scratch tapes and setting a two dimensional scattering transfer cross section variable to zero

Appendix C (Contd')

Symbol	Meaning or Use
SIGA (I)	: Absorption cross section for the I-th fine group
SIGT (I)	: Total cross section for the I-th fine group
SIGQ (I)	: Fission cross section for the I-th fine group
SIGTR (I)	: Transport cross section for the I-th broad group
SISO(LL,I)	: Total P0 scattering transfer cross section for scattering from broad group LL to broad group I
SS(I)	: Convenience variable used for temporary storage of specific data for a nuclide
SSS	: Nuclide I.D. number
SSSS(I)	: Convenience variable used for temporary storage of the source spectrum and also the fission cross section times Nu.
SUM, SUMM, SUMMM	: Summing variables
TEMP (K)	: Convenience variable used for the temporary storage of cross section data
TTT(I)	: Total cross section for the I-th broad group
TRA(I)	: Convenience variable for temporary storage of energies and lethargies

Appendix C (Contd')

Symbol	Meaning or Use
TT(K,KK)	: Scattering transfer cross section for scatter from fine group K to fine group KK (GGC-4 source deck)
TT(K,KK)	: Scattering transfer cross section for scatter from fine group J to fine group KK where K is equal to KK+4-J (ENDF/B deck)
TT(3,I)	: Total cross section for the I-th fine group (ENDF/B deck)
TT(2,I)	: Fission cross section times Nu for the I-th fine group (ENDF/B deck)
TT(1,I)	: Absorption cross section for the I-th fine group (ENDF/B deck)
TTT(I)	: Convenience variable used for temporary storage of cross section data and resonance data
XINELAS(LL,I)	: Inelastic scattering transfer for cross section for scatter from broad group LL to broad group I
XMUBAR(I)	: Average cosine of the scattering angle for broad group I
XN2N(LL,I)	: N-2N scattering transfer cross section for scatter from broad group LL to broad group I

APPENDIX D

The 99 Fine Group Structure

On the following pages are listed the 99 fine group boundaries and the corresponding 99 group lethargies used in both versions of New Barnyard. The 99 group structure is calculated as follows:

Let E_5, the fifth energy point, be equal to 10 MeV and let it be the reference energy. Therefore,

$$E_5 = 10^7 \text{ ev} \qquad \text{(D1)}$$

$$U_5 = 0$$

The first energy point is taken to be $E_1 = E_5 \; e^{-(-.4)}$. The next 49 points are determined with a uniform lethargy mesh U = 0.1 i.e.,

$$U_1 = -.4$$

$$U_i = U_{i-1} + 0.1 \qquad i = 2, 3, \ldots 50$$

$$\text{(D2)}$$

and the next 50 points are given by

$$U_i = U_{i-1} + 0.25 \qquad i = 51, 52, \ldots 100$$

$$\text{(D3)}$$

Energies associated with these lethargies are given by

$$E_i = E_5 \; e^{-U_i}.$$

GROUP	ENERGY INTERVAL(E.V.)			LETHARGY INTERVAL		
1	1.491825E 07	TO	1.349859E 07	-4.00000E-01	TO	-3.00000E-01
2	1.349859E 07	TO	1.221403E 07	-3.00000E-01	TO	-2.00000E-01
3	1.221403E 07	TO	1.105171E 07	-2.00000E-01	TO	-1.00000E-01
4	1.105171E 07	TO	1.000000E 07	-1.00000E-01	TO	0.
5	1.000000E 07	TO	9.048374E 06	0.	TO	1.00000E-01
6	9.048374E 06	TO	8.187308E 06	1.00000E-01	TO	2.00000E-01
7	8.187308E 06	TO	7.408182E 06	2.00000E-01	TO	3.00000E-01
8	7.408182E 06	TO	6.703200E 06	3.00000E-01	TO	4.00000E-01
9	6.703200E 06	TO	6.065307E 06	4.00000E-01	TO	5.00000E-01
10	6.065307E 06	TO	5.488116E 06	5.00000E-01	TO	6.00000E-01
11	5.488116E 06	TO	4.965853E 06	6.00000E-01	TO	7.00000E-01
12	4.965853E 06	TO	4.493290E 06	7.00000E-01	TO	8.00000E-01
13	4.493290E 06	TO	4.065697E 06	8.00000E-01	TO	9.00000E-01
14	4.065697E 06	TO	3.678794E 06	9.00000E-01	TO	1.00000E 00
15	3.678794E 06	TO	3.328711E 06	1.00000E 00	TO	1.10000E 00
16	3.328711E 06	TO	3.011942E 06	1.10000E 00	TO	1.20000E 00
17	3.011942E 06	TO	2.725318E 06	1.20000E 00	TO	1.30000E 00
18	2.725318E 06	TO	2.465970E 06	1.30000E 00	TO	1.40000E 00
19	2.465970E 06	TO	2.231302E 06	1.40000E 00	TO	1.50000E 00
20	2.231302E 06	TO	2.018965E 06	1.50000E 00	TO	1.60000E 00
21	2.018965E 06	TO	1.826835E 06	1.60000E 00	TO	1.70000E 00
22	1.826835E 06	TO	1.652989E 06	1.70000E 00	TO	1.80000E 00
23	1.652989E 06	TO	1.495686E 06	1.80000E 00	TO	1.90000E 00
24	1.495686E 06	TO	1.353353E 06	1.90000E 00	TO	2.00000E 00
25	1.353353E 06	TO	1.224564E 06	2.00000E 00	TO	2.10000E 00
26	1.224564E 06	TO	1.108032E 06	2.10000E 00	TO	2.20000E 00
27	1.108032E 06	TO	1.002589E 06	2.20000E 00	TO	2.30000E 00
28	1.002589E 06	TO	9.071796E 05	2.30000E 00	TO	2.40000E 00
29	9.071796E 05	TO	8.208501E 05	2.40000E 00	TO	2.50000E 00
30	8.208501E 05	TO	7.427359E 05	2.50000E 00	TO	2.60000E 00
31	7.427359E 05	TO	6.720552E 05	2.60000E 00	TO	2.70000E 00
32	6.720552E 05	TO	6.081007E 05	2.70000E 00	TO	2.80000E 00
33	6.081007E 05	TO	5.502323E 05	2.80000E 00	TO	2.90000E 00
34	5.502323E 05	TO	4.978708E 05	2.90000E 00	TO	3.00000E 00
35	4.978708E 05	TO	4.504921E 05	3.00000E 00	TO	3.10000E 00
36	4.504921E 05	TO	4.076221E 05	3.10000E 00	TO	3.20000E 00
37	4.076221E 05	TO	3.688317E 05	3.20000E 00	TO	3.30000E 00
38	3.688317E 05	TO	3.337327E 05	3.30000E 00	TO	3.40000E 00
39	3.337327E 05	TO	3.019739E 05	3.40000E 00	TO	3.50000E 00
40	3.019739E 05	TO	2.732373E 05	3.50000E 00	TO	3.60000E 00
41	2.732373E 05	TO	2.472353E 05	3.60000E 00	TO	3.70000E 00
42	2.472353E 05	TO	2.237078E 05	3.70000E 00	TO	3.80000E 00
43	2.237078E 05	TO	2.024192E 05	3.80000E 00	TO	3.90000E 00
44	2.024192E 05	TO	1.831564E 05	3.90000E 00	TO	4.00000E 00
45	1.831564E 05	TO	1.657268E 05	4.00000E 00	TO	4.10000E 00
46	1.657268E 05	TO	1.499558E 05	4.10000E 00	TO	4.20000E 00
47	1.499558E 05	TO	1.356856E 05	4.20000E 00	TO	4.30000E 00
48	1.356856E 05	TO	1.227734E 05	4.30000E 00	TO	4.40000E 00
49	1.227734E 05	TO	1.110900E 05	4.40000E 00	TO	4.50000E 00
50	1.110900E 05	TO	8.651698E 04	4.50000E 00	TO	4.75000E 00

GROUP	ENERGY INTERVAL(E.V.)	LETHARGY INTERVAL
51	8.651698E 04 TO 6.737949E 04	4.75000E 00 TO 5.00000E 00
52	6.737949E 04 TO 5.247520E 04	5.00000E 00 TO 5.25000E 00
53	5.247520E 04 TO 4.086773E 04	5.25000E 00 TO 5.50000E 00
54	4.086773E 04 TO 3.182782E 04	5.50000E 00 TO 5.75000E 00
55	3.182782E 04 TO 2.478753E 04	5.75000E 00 TO 6.00000E 00
56	2.478753E 04 TO 1.930455E 04	6.00000E 00 TO 6.25000E 00
57	1.930455E 04 TO 1.503440E 04	6.25000E 00 TO 6.50000E 00
58	1.503440E 04 TO 1.170880E 04	6.50000E 00 TO 6.75000E 00
59	1.170880E 04 TO 9.118823E 03	6.75000E 00 TO 7.00000E 00
60	9.118823E 03 TO 7.101746E 03	7.00000E 00 TO 7.25000E 00
61	7.101746E 03 TO 5.530846E 03	7.25000E 00 TO 7.50000E 00
62	5.530846E 03 TO 4.307427E 03	7.50000E 00 TO 7.75000E 00
63	4.307427E 03 TO 3.354627E 03	7.75000E 00 TO 8.00000E 00
64	3.354627E 03 TO 2.612587E 03	8.00000E 00 TO 8.25000E 00
65	2.612587E 03 TO 2.034684E 03	8.25000E 00 TO 8.50000E 00
66	2.034684E 03 TO 1.584614E 03	8.50000E 00 TO 8.75000E 00
67	1.584614E 03 TO 1.234098E 03	8.75000E 00 TO 9.00000E 00
68	1.234098E 03 TO 9.611169E 02	9.00000E 00 TO 9.25000E 00
69	9.611169E 02 TO 7.485186E 02	9.25000E 00 TO 9.50000E 00
70	7.485186E 02 TO 5.829468E 02	9.50000E 00 TO 9.75000E 00
71	5.829468E 02 TO 4.539995E 02	9.75000E 00 TO 1.00000E 01
72	4.539995E 02 TO 3.535751E 02	1.00000E 01 TO 1.02500E 01
73	3.535751E 02 TO 2.753646E 02	1.02500E 01 TO 1.05000E 01
74	2.753646E 02 TO 2.144542E 02	1.05000E 01 TO 1.07500E 01
75	2.144542E 02 TO 1.670171E 02	1.07500E 01 TO 1.10000E 01
76	1.670171E 02 TO 1.300730E 02	1.10000E 01 TO 1.12500E 01
77	1.300730E 02 TO 1.013010E 02	1.12500E 01 TO 1.15000E 01
78	1.013010E 02 TO 7.889328E 01	1.15000E 01 TO 1.17500E 01
79	7.889328E 01 TO 6.144214E 01	1.17500E 01 TO 1.20000E 01
80	6.144214E 01 TO 4.785119E 01	1.20000E 01 TO 1.22500E 01
81	4.785119E 01 TO 3.726654E 01	1.22500E 01 TO 1.25000E 01
82	3.726654E 01 TO 2.902321E 01	1.25000E 01 TO 1.27500E 01
83	2.902321E 01 TO 2.260330E 01	1.27500E 01 TO 1.30000E 01
84	2.260330E 01 TO 1.760347E 01	1.30000E 01 TO 1.32500E 01
85	1.760347E 01 TO 1.370960E 01	1.32500E 01 TO 1.35000E 01
86	1.370960E 01 TO 1.067704E 01	1.35000E 01 TO 1.37500E 01
87	1.067704E 01 TO 8.315290E 00	1.37500E 01 TO 1.40000E 01
88	8.315290E 00 TO 6.475955E 00	1.40000E 01 TO 1.42500E 01
89	6.475955E 00 TO 5.043478E 00	1.42500E 01 TO 1.45000E 01
90	5.043478E 00 TO 3.927865E 00	1.45000E 01 TO 1.47500E 01
91	3.927865E 00 TO 3.059024E 00	1.47500E 01 TO 1.50000E 01
92	3.059024E 00 TO 2.382370E 00	1.50000E 01 TO 1.52500E 01
93	2.382370E 00 TO 1.855392E 00	1.52500E 01 TO 1.55000E 01
94	1.855392E 00 TO 1.444981E 00	1.55000E 01 TO 1.57500E 01
95	1.444981E 00 TO 1.125352E 00	1.57500E 01 TO 1.60000E 01
96	1.125352E 00 TO 8.764252E-01	1.60000E 01 TO 1.62500E 01
97	8.764252E-01 TO 6.825607E-01	1.62500E 01 TO 1.65000E 01
98	6.825607E-01 TO 5.315788E-01	1.65000E 01 TO 1.67500E 01
99	5.315788E-01 TO 4.139940E-01	1.67500E 01 TO 1.70000E 01

APPENDIX E

Structure of the Data Tapes

Tables XIII and XIV of this appendix show the structure of the two data tapes used in New Barnyard. Table XV lists explanations for some parts shown in Tables XIII and XIV. If one desires to use either of these data tapes in a new computer code, he should study these tables carefully.

TABLE XIII

Structure of the GGC-4 Data Tape

Record	Number of Words	Variable*	Description
1	3	NTID(1) = NBT	Tape identification number.
		NTID(2) = NEP	Number of fast energy boundaries (groups + 1). (Note #1)
		NTID(3) = NGT	Number of thermal energy points. (Note # 2)
2	90	AT(1-90)	Tape description (5 lines of 72 characters each).
3	1	LAD = NNOT	Number of nuclides on tape.
	Record 4 is repeated for each nuclide (number of records 4 = LAD)		
4	21	ESS(1-18)= DAD (1-18)	Nuclide description.
		ESS(19-21)= DAD (19-21)	Nuclide I.D. number, number of resolved resonances, number of unresolved resonances.
5	4	LEN(1)	Resonance data. (Note #3)
		LEN(2)	Resonance data. (Note #3)
		LEN(3)	Resonance data. (Note #3)
		LBS = LEN(4)	Resonance data. (Note #3)
6	LBS (max. = 5101)		Resonance data. (Note #3)
7	NELT = NEP + 2 NEP + 1)	TTT(1-NELT)	Fast energy group boundaries (NEP+1 values), fast lethargy boundaries (NEP+1 values), fast lethargy intervals (NEP values).
8	NGT	TTT(1-NGT)	Thermal energy points. (Note #2)

* The variable names that appear in the GGC-4 Source deck are listed in this column. This tape is written in binary and no Format statements are required when "reading in" these variables.

Table XIII (Contd')

Record	Number of Words	Variable	Description
9	1	NSP	Number of fission spectra on tape.
	Record 10 is repeated for each fission spectrum (number of records 10 = NSP).		
10	LNR=17+NEP	ESS (1-18)	Description of fission source spectrum.
		ESS (19-LNR)	Fission source spectrum for each fast energy group. (Note #4)
11	1	MRESN	Number of nuclides with resonance data.
	Records 12 and 13 are repeated for each resonance nuclide (number of records 12 and 13 = MRESN).		
12	3	SS(13)	Nuclide I.D. number. (Note #5)
		SS(14)	Resonance data.(Note #3)
		SS(15)	Resonance data. (Note #3)
13	LT=6*SS(14)+ SS(15) + 9	TT(1-LT)	Resonance data for nuclide.
	Records 14 through 25 are repeated for each nuclide.		
14	37	ESS(1-18)	Nuclide description.
		SS(13)	Nuclide I.D. number
		SS(14)	Resonance data. (Note #3)
		SS(15)	Resonance data. (Note #3)
		SS(16)	Length of P0 array (same length for P1, P2, and P3 arrays). (Note #5)
		SS(17)	Number of groups scattered from P0 scattering. (Note #6)
		SS(18)	Number of groups scattered into P0 scattering. (Note #7)

Table XIII (Contd')

Record	Number of Words	Variable	Description
		SS(19-21)	For inelastic array, same as SS(16-18).
		SS(22-24)	For n, 2n array, same as SS(16-18). (Note #8)
		SS(25)	Fission index: non-zero= fissionable nuclide.
		SS(26)=NX	Number of 1-D arrays (absorption, fission, etc.)
		SS(27-29)	For total scatter array, same as SS(16-18).
		SS(30)	Number of records for this nuclide.
		SS(31)	Mass number of nuclides.
	If number of 1-D arrays [SS(26)] is zero, record 15 not present.		
15	IMX=(6+NEP)NX	TTT(1-IMX)	One-dimensional cross-section arrays in form: type number (1=absorption, 2=fission, etc.), description (5 words), cross-section for each energy group. (Note #9)
	Records 16 through 19 not present if length of PO array SS(16) is zero.		
16	IDK=SS(16)	TTT(1-IDK)	PO scattering array in form: $\sigma 1\rightarrow 1$, $\sigma 1\rightarrow 2$, $\sigma 1\rightarrow 3$..., $\sigma 2\rightarrow 2$, $\sigma 2\rightarrow 3$,..., $\sigma 3\rightarrow 3$, $\sigma 3\rightarrow 4$, ... (Note #10)
17	IDK	TT(1-IDK)	P1 scattering array (same form as PO. (Note #11)
18	IDK	TT(1-IDK)	P2 scattering array (same form as PO). (Note #11)
19	IDK	TT(1-IDK)	P3 scattering array (same form as PO). (Note #11)

Table XIII (Contd')

Record	Number of Words	Variable	Description
	Record 20 not present if length of inelastic array SS(19) is zero.		
20	LIT=SS(19)	TTT(1-LIT)	Inelastic scattering array (same form as P0). (Note #12)
	Record 21 not present if length of n, 2n array [SS(22)] is zero.		
21	LIT=SS(22)	TTT(1-LIT)	n, 2n scattering array (same form as P0). (Note #12)
22	NEV=Number of groups	BST(1-NEV)	σtotal for each group. (Note #13)
	Record 23 not present if length of total scatter array [SS(27)] is zero.		
23	LAT=SS(27)	TTT(1-LAT)	Total scattering array (same form as P0). (Note #14 and 15)
	Records 24 and 25 not present if P0 and P1 arrays omitted [SS(16)=0.]		
24	NEV	SCT(1-NEV)	σ Scatter for each group. (Note 15)
25	NEV	SPP(1-NEV)	σP1 scatter for each group. (Note 17)
26	---	---	End-of-file mark.

TABLE **XIV**

Structure of the ENDF/B Data Tapes

Record	Number of Words	Variable*	Description
1	21	A(I),I=1,21 Format (9A4,A1, 10A4,A3)	Description of the P0 data that follows (Note #18)
2	12	M(K),TEMP(K), K=1,6 Format (6(I2,1X,E9.0))	The absorption, fission, total, and P0 scattering transfer microscopic cross sections (Note #19)
	Record 2 is repeated until the entire P0 array is established.		
3	21	A(I),I=1,21 Format (9A4,A1, 10A4, A3)	Description of the P1 data that follows.
4	12	M(K),TEMP(K) K=1,6 Format (6(I2,1X,E9.0))	The P1 elastic scattering transfer microscopic cross sections (Note #20)
	Record 4 is repeated until the entire P1 array is established.		
5	21	A(I),I=1,21 Format (9A4,A1, 10A4,A3)	Description of the P2 data that follows.
6	12	M(K),TEMP(K) K=1,6 Format (6(I2, 1X,E9.0))	The P2 elastic scattering transfer microscopic cross section (Note #11)
	Record 6 is repeated until the entire P2 array is established.		
7	21	A(I),I=1,21 Format (9A4,A1, 10A4, A3)	Description of the P3 data that follows
8	12	M(K),TEMP(K), K=1,6 Format (6(I2,1X,E9.0))	The P3 elastic scattering transfer microscopic cross sections.

* The variable names that appear in the ENDF/B source deck are listed in this column. This tape is written in BCD and requires format statements when "reading in" these variables.

TABLE XIV (Contd')

Record	Number of Words	Variable	Description
9	21	A(I), I=1,21 Format (9A4, A1, 10A4, A3)	Description of the P4 data that follows
10	12	M(K),TEMP(X), K = 1, 6 Format (6 (I2 , 1X, E9.0))	The P4 elastic scattering transfer microscopic cross section
	Record 10 is repeated until the entire P4 array is established.		
11	21	A(I), I=1, 21 Format (9A4, A1, 10A4, A3)	Description of the P5 data that follows
12	12	M(K),TEMP(K), K=1,6 Format (6(I2 ,1X, E9.0))	The P5 elastic scattering transfer microscopic cross sections
	Record 12 is repeated until the entire P5 array is established.		
13	21	A(I),I=1,21 Format (9A4,A1,10A4,A3)	Description of the P6 data that follows
14	12	M(K),TEMP(K), K=1,6 Format (6(I2, 1X, E9.0))	The P6 elastic scattering transfer microscopic cross sections
	Record 14 is repeated until the entire P6 array is established.		
15	21	A(I),I=1,21 Format (9A4,A1,10A4,A3)	Description of the P7 data that follows
16	12	M(K), TEMP(K) K=1,6 Format (6(I2,1X,E9.0))	The P7 elastic scattering transfer microscopic cross section
	Record 16 is repeated until the entire P7 array is established.		
17	21	A(I),I=1,21 Format (9A4,A1,10A4,A3)	Description of the P8 data that follows

TABLE XIV(Contd')

Record	Number of Words	Variable	Description
18	12	M(K),TEMP(K), K=1,6 Format (6(I2,1X,E9.0))	The P8 elastic scattering transfer microscopic cross sections

Record 18 is repeated until the entire P8 array is established.

Records 1 through 18 are repeated for each nuclide.

TABLE XV

Comments About the Data Tapes

Note Number	Comments
1	NEP is equal to one more than the number of fine groups. This number is 100 for a 99 group data tape.
2	This information is for a thermal spectrum calculation and is not used in the GGC-4 source deck.
3	The resonance data included on this tape is not used; therefore, no effort is made to describe it.
4	The fission spectrum for each fine group is given as the fractional number of fission neutrons born in that group, ${}_i$. Thus, $\sum_{i=1}^{99} \chi_i = 1.000.$
5	SS(16) is the total length of the P0 elastic transfer array. The P1, P2, and P3 arrays are the same length as that for P0.
6	SS(17) is the number of incident energy groups, and is equal to 99.
7	SS(18) is the maximum number of secondary energy groups for which elastic transfer cross sections are given. This number includes the in-group term.
8	SS(19) through SS(24) are values similar to those given for elastic scattering. Note: no P1, P2, or P3 arrays are given for inelastic or (n,2n) scattering.

TABLE XV (Contd')

Note Number	Comments
9	The I-D arrays have been given a numeric identification:

I.D. Number	Description of Reaction
1	Absorption (the fission cross section plus any other neutron removing reaction).
2	Fission cross section.
3	Nu, ν, the average number of fission neutrons produced per fission event (includes delayed neutrons).
4	(n, γ) radiative capture
5	(n, p)
6	(n, α)
7	(n, d)
8	(n, t)
9	(n, n)p
10	(n, n)d
11	(n, n)t

(The I-D arrays are given in sequential order.)

10 The elastic P0 scattering cross sections are given as a continuous array. The order of giving the cross sections is $\sigma^{elas}_{1\to1}$, $\sigma^{elas}_{1\to2}$, $\sigma^{elas}_{1\to3}$, ..., $\sigma^{elas}_{1\to[1+SS(18)-1]}$, $\sigma^{elas}_{2\to2}$, $\sigma^{elas}_{2\to3}$, ..., $\sigma^{elas}_{2\to[2+SS(18)-1]}$, $\sigma^{elas}_{3\to3}$, $\sigma^{elas}_{3\to4}$, ..., $\sigma^{elas}_{3\to[3+SS(18)-1]}$, ...

TABLE XV (Contd')

Note Number	Comments
	$\sigma^{elas}_{N\rightarrow N}$, $\sigma^{elas}_{N\rightarrow N+1}$, ..., $\sigma^{elas}_{N\rightarrow[N + SS(18)-1]}$, ... These cross sections are given in units of barns. There are SS(16) total terms in this array. The elastic P1, P2, P3 arrays are given in exactly the same manner.
11	Recall PN = (2N+1)*σSN(i→j); therefore, P1 = 3*σs1(i→j), P2 = 5*σs2 (i→j), etc. The values listed on the tape are the PN values and not the σ_{sn} (i→j) values.
12	The inelastic and (n, 2n) scattering arrays are given in the same manner as the P0 elastic arrays. However, the number of groups scattered is generally taken to be 100 for these reactions, thus for the inelastic scattering array, the cross sections are given as: $\sigma^{inelas}_{1\rightarrow 1}$, $\sigma^{inelas}_{1\rightarrow 2}$, ... $\sigma^{inelas}_{1\rightarrow 100}$, $\sigma^{inelas}_{2\rightarrow 2}$, $\sigma^{inelas}_{2\rightarrow 3}$, ... $\sigma^{inelas}_{2\rightarrow 100}$, $\sigma^{inelas}_{3\rightarrow 3}$, $\sigma^{inelas}_{3\rightarrow 4}$, ..., $\sigma^{inelas}_{3\rightarrow 100}$, ... $\sigma^{inelas}_{SS(20)\rightarrow SS(20)}$, ... $\sigma^{inelas}_{SS(20)\rightarrow 100}$. For a 99 group library tape there is a low energy "dump" group (#100). This group represents the cross sections scattered to all energies below the low energy boundary of group 99 (0.414 ev).
13	BST(I), σ total for group i is calculated by $$\sigma^{total}_{i} = \sigma^{abs}_{i} + \sum_{j=i}^{j=[i + SS(18)-1]} \sigma^{P0(elastic)}_{j(i\rightarrow j)} + \sum_{j=1}^{j=100} \sigma^{inelas}_{j(i\rightarrow j)} + 2* \sum_{j=1}^{j=100} \sigma^{n,2n}_{j(i\rightarrow j)}$$

TABLE XV (Contd')

Note Number	Comments
14	SS(27) is the total size of the total transfer array. Since the data tape has 99 fine groups, this number will generally be 5049.
15	The total scattering array for scattering from i→j is obtained by $\sigma_{i\to j}^{\text{total scattering}} = \sigma_{i\to j}^{\text{P0 (elas)}} + \sigma_{i\to j}^{\text{inelas}} + 2*\sigma_{i\to j}^{n,2n}$
16	σ Scatter for group i is calculated by $\sigma_i^{\text{scatter}} = \sum_{j=i}^{j=[i+SS(18)-1]} \sigma_{j(i\to j)}^{\text{P0(elastic)}}$
17	σ P1 scatter for group i is calculated by $\sigma_i^{\text{P1 scatter}} = \sum_{j=i}^{j=[i+SS(18)-1]} \sigma_{j(i\to j)}^{\text{P1(elastic)}}$
18	An example of this description is ....MATERIAL NUMBER 1012 100 GROUPS P0
19	These 99 group cross sections are given as $\sigma_1^{ABS}, \nu\sigma_1^{\text{fission}}, \sigma_1^{\text{total}}, \sigma 1\to 1$, 99R $\sigma_2^{ABS}, \nu\sigma_2^{\text{fission}}, \sigma_2^{\text{total}}, \sigma 2\to 2, \sigma 1\to 2$, 98R ⋮ $\sigma_g^{ABS}, \nu\sigma_g^{\text{fission}}, \sigma_g^{\text{total}}, \sigma g\to g, \sigma g-1\to g,$ $\sigma g-2\to g \cdots$ ⋮

TABLE XV (Contd')

Note Number	Comments
	$\sigma_{99}^{ABS}, \nu\sigma_{99}^{Fission}, \sigma_{99}^{Total}, \sigma_{99\rightarrow99},$
	$\sigma_{98\rightarrow99}\cdots$ 3R, 0. , $\sigma_{99\rightarrow100}, \sigma_{98\rightarrow100},\ldots$
	The R denotes zero and the number in front of the R indicates the number of cross section values that are equal to zero.
20	These cross sections are given as
	3R, $3\sigma_{1\rightarrow1}$, 99R
	$\vdots$
	3R, $3\sigma_{g\rightarrow g}$, $3\sigma_{g-1\rightarrow g}$, $3\sigma_{g-2\rightarrow g}$, ...
	$\vdots$
	3R, $3\sigma_{99\rightarrow99}$, $3\sigma_{98\rightarrow99}$, ...
	3R, 0. , $3\sigma_{99\rightarrow100}$, $3\sigma_{98\rightarrow100}$, ...

APPENDIX F

99 Group Fission Source Spectrums

99 group fission source spectrums are presented here for the following nuclides:

(1) U-233

(2) U-235

(3) PU-239

(4) PU-241

(5) CF-252

This data was taken from pages 42-46 of GA-4265, "GAM-11, A B_3 code for the calculation of Fast-Neutron Spectra and Associated Multi Group Constants" by G. D. Joanou and J. S. Rudek. Each source spectrum has been normalized to 1.

SPECTRUM NUMBER 1 U-233 FISSION SOURCE

GROUP	SOURCE	GROUP	SOURCE	GROUP	SOURCE	GROUP	SOURCE
1	3.51793996-5	26	3.92637998-2	51	3.54065996-3	76	0.
2	9.84442985-5	27	3.66024998-2	52	2.46145996-3	77	0.
3	2.44521996-4	28	3.38495997-2	53	1.70685999-3	78	0.
4	5.45474994-4	29	3.10797998-2	54	1.18125999-3	79	0.
5	1.10456999-3	30	2.83532998-2	55	8.16252995-4	80	0.
6	2.04995999-3	31	2.57167995-2	56	5.63359994-4	81	0.
7	3.51742998-3	32	2.32045999-2	57	3.88456997-4	82	0.
8	5.62406993-3	33	2.08402997-2	58	0.	83	0.
9	8.43932986-3	34	1.86385998-2	59	0.	84	0.
10	1.19615999-2	35	1.66066998-2	60	0.	85	0.
11	1.61068998-2	36	1.47461998-2	61	0.	86	0.
12	2.07130998-2	37	1.30542000-2	62	0.	87	0.
13	2.55585998-2	38	1.15246999-2	63	0.	88	0.
14	3.03897998-2	39	1.01492000-2	64	0.	89	0.
15	3.49524999-2	40	8.91796982-3	65	0.	90	0.
16	3.90193996-2	41	7.82039994-3	66	0.	91	0.
17	4.24106991-2	42	6.84550995-3	67	0.	92	0.
18	4.50054997-2	43	5.98236996-3	68	0.	93	0.
19	4.67448997-2	44	5.22037995-3	69	0.	94	0.
20	4.76267993-2	45	4.54939991-3	70	0.	95	0.
21	4.76962996-2	46	3.95991996-3	71	0.	96	0.
22	4.70342994-2	47	3.44308996-3	72	0.	97	0.
23	4.57442999-2	48	2.99078995-3	73	0.	98	0.
24	4.39418995-2	49	2.59561998-3	74	0.	99	0.
25	4.17442995-2	50	5.07637995-3	75	0.		

SPECTRUM NUMBER 2 U-235 FISSION SOURCE

GROUP	SOURCE	GROUP	SOURCE	GROUP	SOURCE	GROUP	SOURCE
1	4.12165993-5	26	3.88628998-2	51	3.45782998-3	76	0.
2	1.13324998-4	27	3.61786994-2	52	2.40337995-3	77	0.
3	2.77032998-4	28	3.34158996-2	53	1.66631998-3	78	0.
4	6.09151995-4	29	3.06469998-2	54	1.15305997-3	79	0.
5	1.21751998-3	30	2.79301000-2	55	7.96686995-4	80	0.
6	2.23310998-3	31	2.53095999-2	56	5.49813992-4	81	0.
7	3.79103997-3	32	2.28181997-2	57	3.79094997-4	82	0.
8	6.00337994-3	33	2.04778996-2	58	0.	83	0.
9	8.93036997-3	34	1.83020997-2	59	0.	84	0.
10	1.25582999-2	35	1.62968998-2	60	0.	85	0.
11	1.67905997-2	36	1.44630998-2	61	0.	86	0.
12	2.14540997-2	37	1.27971998-2	62	0.	87	0.
13	2.63196996-2	38	1.12925999-2	63	0.	88	0.
14	3.11312997-2	39	9.94079983-3	64	0.	89	0.
15	3.56363997-2	40	8.73163986-3	65	0.	90	0.
16	3.96132994-2	41	7.65444994-3	66	0.	91	0.
17	4.28905994-2	42	6.69822997-3	67	0.	92	0.
18	4.53566992-2	43	5.85206997-3	68	0.	93	0.
19	4.69619995-2	44	5.10540992-3	69	0.	94	0.
20	4.77124995-2	45	4.44821995-3	70	0.	95	0.
21	4.76600993-2	46	3.87105995-3	71	0.	96	0.
22	4.68901992-2	47	3.36521995-3	72	0.	97	0.
23	4.55092996-2	48	2.92266998-3	73	0.	98	0.
24	4.36339992-2	49	2.53611997-3	74	0.	99	0.
25	4.13814992-2	50	4.95893997-3	75	0.		

SPECTRUM NUMBER 3 PU-239 FISSION SOURCE

GROUP	SOURCE	GROUP	SOURCE	GROUP	SOURCE	GROUP	SOURCE
1	5.06228995-5	26	3.85750997-2	51	3.45138997-3	76	0.
2	1.35049999-4	27	3.59134993-2	52	2.39935997-3	77	0.
3	3.21494997-4	28	3.31758994-2	53	1.66377999-3	78	0.
4	6.90658998-4	29	3.04330999-2	54	1.15143999-3	79	0.
5	1.35262997-3	30	2.77418998-2	55	7.95642996-4	80	0.
6	2.43729997-3	31	2.51460999-2	56	5.49133992-4	81	0.
7	4.07442993-3	32	2.26773998-2	57	3.78647995-4	82	0.
8	6.36674994-3	33	2.03576997-2	58	0.	83	0.
9	9.36279988-3	34	1.82001998-2	59	0.	84	0.
10	1.30374999-2	35	1.62110999-2	60	0.	85	0.
11	1.72354999-2	36	1.43911998-2	61	0.	86	0.
12	2.19297996-2	37	1.27372998-2	62	0.	87	0.
13	2.67424998-2	38	1.12428999-2	63	0.	88	0.
14	3.14731997-2	39	9.89958990-3	64	0.	89	0.
15	3.58783996-2	40	8.69761980-3	65	0.	90	0.
16	3.97464994-2	41	7.62640995-3	66	0.	91	0.
17	4.29155995-2	42	6.67514992-3	67	0.	92	0.
18	4.52825995-2	43	5.83310992-3	68	0.	93	0.
19	4.68035996-2	44	5.08984995-3	69	0.	94	0.
20	4.74875993-2	45	4.43543988-3	70	0.	95	0.
21	4.73875999-2	46	3.86057997-3	71	0.	96	0.
22	4.65880990-2	47	3.35660997-3	72	0.	97	0.
23	4.51937997-2	48	2.91559997-3	73	0.	98	0.
24	4.33182991-2	49	2.53030998-3	74	0.	99	0.
25	4.10759997-2	50	4.94851995-3	75	0.		

SPECTRUM NUMBER 4 PU-241 FISSION SOURCE

GROUP	SOURCE	GROUP	SOURCE	GROUP	SOURCE	GROUP	SOURCE
1	5.94814992-5	26	3.79503995-2	51	3.29129997-3	76	0.
2	1.56710999-4	27	3.52316996-2	52	2.28679997-3	77	0.
3	3.68644997-4	28	3.24607998-2	53	1.58502999-3	78	0.
4	7.83022994-4	29	2.97030998-2	54	1.09656999-3	79	0.
5	1.51705998-3	30	2.70178998-2	55	7.57524997-4	80	0.
6	2.70567995-3	31	2.44393995-2	56	5.22716993-4	81	0.
7	4.47914994-3	32	2.19984999-2	57	3.60373995-4	82	0.
8	6.93450993-3	33	1.97139999-2	58	0.	83	0.
9	1.01081999-2	34	1.75965998-2	59	0.	84	0.
10	1.39581999-2	35	1.56505999-2	60	0.	85	0.
11	1.83597998-2	36	1.38750999-2	61	0.	86	0.
12	2.31180999-2	37	1.22653998-2	62	0.	87	0.
13	2.79914999-2	38	1.08142999-2	63	0.	88	0.
14	3.27218997-2	39	9.51253986-3	64	0.	89	0.
15	3.70650998-2	40	8.34978986-3	65	0.	90	0.
16	4.08152997-2	41	7.31521994-3	66	0.	91	0.
17	4.38211995-2	42	6.39783996-3	67	0.	92	0.
18	4.59928995-2	43	5.58634999-3	68	0.	93	0.
19	4.73007995-2	44	4.87183994-3	69	0.	94	0.
20	4.77678996-2	45	4.24299991-3	70	0.	95	0.
21	4.74587995-2	46	3.69111997-3	71	0.	96	0.
22	4.64675999-2	47	3.20772997-3	72	0.	97	0.
23	4.49050993-2	48	2.78506997-3	73	0.	98	0.
24	4.28889996-2	49	2.41606998-3	74	0.	99	0.
25	4.05347997-2	50	4.72235996-3	75	0.		

SPECTRUM NUMBER 5 CF-252 FISSION SOURCE

GROUP	SOURCE	GROUP	SOURCE	GROUP	SOURCE	GROUP	SOURCE
1	2.11560997-4	26	3.42097998-2	51	2.68588996-3	76	0.
2	4.81211996-4	27	3.14282998-2	52	1.86336999-3	77	0.
3	9.91409993-4	28	2.86850998-2	53	1.29004999-3	78	0.
4	1.86835998-3	29	2.60282999-2	54	8.91687989-4	79	0.
5	3.24963999-3	30	2.34938997-2	55	6.15560991-4	80	0.
6	5.25856996-3	31	2.11062998-2	56	4.24525994-4	81	0.
7	7.97483987-3	32	1.88811998-2	57	2.92553997-4	82	0.
8	1.14091998-2	33	1.68263997-2	58	0.	83	0.
9	1.54893999-2	34	1.49438998-2	59	0.	84	0.
10	2.00627998-2	35	1.32312998-2	60	0.	85	0.
11	2.49124998-2	36	1.16824998-2	61	0.	86	0.
12	2.97849998-2	37	1.02892998-2	62	0.	87	0.
13	3.44215998-2	38	9.04196990-3	63	0.	88	0.
14	3.85871997-2	39	7.92976987-3	64	0.	89	0.
15	4.20929998-2	40	6.94171995-3	65	0.	90	0.
16	4.48085994-2	41	6.06630995-3	66	0.	91	0.
17	4.66665995-2	42	5.29432994-3	67	0.	92	0.
18	4.76579994-2	43	4.61403996-3	68	0.	93	0.
19	4.78235990-2	44	4.01632988-3	69	0.	94	0.
20	4.72410995-2	45	3.49224997-3	70	0.	95	0.
21	4.60134995-2	46	3.03357998-3	71	0.	96	0.
22	4.42563993-2	47	2.63281998-3	72	0.	97	0.
23	4.20884997-2	48	2.28316996-3	73	0.	98	0.
24	3.96239999-2	49	1.97853997-3	74	0.	99	0.
25	3.69672999-2	50	3.86108997-3	75	0.		

APPENDIX G

Sample Problems

This appendix is devoted to exhibiting New 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 appendix:

(1) Water: 5 group set using GGC-4 source deck (P0 to P3 cross sections included)

(2) Magnesium: 4 group set using ENDF/B source deck. (P0 to P4 cross sections included.)

For both of these problems a U-235 source spectrum was used. Each problem has a printout of the input data followed by the cross section output. The output cross section values are identified and they need no explanation here. The printout of the input data is followed by the cross section output. The output cross section values are identified and they need no explanation here. The printout of the input data is included only for convenience of the user. The actual source decks do not print out the data used.

```
THE INPUT DATA FOR THIS PROBLEM IS ...
WATER--5 GROUP SET USING GGC-4 SOURCE DECK
  5  2  1
 10 22 47 75 99
   1.0000000 0.668000E-01
   8.0200000 0.334000E-01

0.412166E-040.113325E-030.277033E-030.609152E-030.121752E-020.223311E-02
0.379104E-020.600338E-020.893037E-020.125583E-010.167906E-010.214541E-01
0.263197E-010.311313E-010.356364E-010.396133E-010.428906E-010.453567E-01
0.469620E-010.477125E-010.476601E-010.468902E-010.455093E-010.436340E-01
0.413815E-010.388629E-010.361787E-010.334159E-010.306470E-010.279301E-01
0.253096E-010.228182E-010.204779E-010.183021E-010.162969E-010.144631E-01
0.127972E-010.112926E-010.994080E-020.873164E-020.765445E-020.669823E-02
0.585207E-020.510541E-020.444822E-020.387106E-020.336522E-020.292267E-02
0.253612E-020.495894E-020.345783E-020.240338E-020.166632E-020.115306E-02
0.796687E-030.549814E-030.379095E-030.            0.            0.
0.            0.            0.            0.            0.            0.
0.            0.            0.            0.            0.            0.
0.            0.            0.            0.            0.            0.
0.            0.            0.            0.            0.            0.
0.            0.            0.            0.            0.            0.
0.            0.            0.            0.            0.            0.
0.            0.            0.
```

WATER--5 GROUP SET USING GGC-4 SOURCE DECK

FAST DATA TAPE NUMBER = 31568

TAPE DESCRIPTION.....

TAPE I.D. IS 31568.0

THIS TAPE IS GGC4 VERSION OF GGC2 304.0

CONTENTS OF FAST DATA TAPE NUMBER 31568

NUCLIDE NO.	NUCLIDE DESCRIPTION
1.0000000	HYDROGEN
1.2000000	DEUTERIUM
2.0000000	HELIUM
3.0062000	LITHIUM-6 ENDF/B DATA AUGUST 1967
3.0072000	LITHIUM-7 ENDF/B DATA AUGUST 1967
4.0000000	BERYLLIUM GA-5905
5.0000000	BORON NATURAL
5.0100000	BORON 10
6.0200000	CARBON ENDF/B DATA JULY 1967
7.0000000	NITROGEN
8.0200000	OXYGEN ENDF/B OCTOBER 1967
11.0000000	SODIUM
12.0000000	MAGNESIUM
13.0000000	ALUMINUM GA-5884
14.0000000	SILICON
16.0000000	SULFUR
20.0000000	CALCIUM
22.0000000	TITANIUM
24.0000000	CHROMIUM
25.0000000	MANGANESE
26.0000000	IRON
27.0000000	COBALT
28.0000000	NICKEL
29.0000000	COPPER
42.0000000	MOLYBDENUM
48.0000000	CADMIUM
74.0000000	TUNGSTEN GA-5885
74.1799994	TUNGSTEN 180
74.1819992	TUNGSTEN 182 GA-5885
74.1820993	TUNGSTEN 182 RESONANCE GA-5885
74.1829996	TUNGSTEN 183 GA-5885
74.1830997	TUNGSTEN 183 RESONANCE GA-5885
74.1839991	TUNGSTEN 184 GA-5885
74.1840992	TUNGSTEN 184 RESONANCE GA-5885
74.1859999	TUNGSTEN 186 GA-5885
74.1860991	TUNGSTEN 186 RESONANCE GA-5885
82.0000000	LEAD
92.2334995	URANIUM 233 ENDF/B JANUARY 1 1968
92.2349997	URANIUM 235 NASA REPORT
92.2351999	URANIUM 235 KAPL ENDF/B DATA FEB 1967
92.2379999	URANIUM 238 NASA REPORT JAN 1965
92.2380991	URANIUM 238 RESONANCE NASA REPORT JAN 1965
92.2381992	URANIUM 238 ENDF/B DATA JULY 1967
92.2382994	URANIUM 238 RESONANCE ENDF/B DATA JULY 1967
94.2411995	PLUTONIUM 241 ENDF/B DATA JANUARY 1967

WATER--5 GROUP SET USING GGC-4 SOURCE DECK

FINE GROUP STRUCTURE

GROUP	ENERGY INTERVAL(E.V.)			LETHARGY INTERVAL		
1	1.491825E 07	TO	1.349859E 07	-4.00000E-01	TO	-3.00000E-01
2	1.349859E 07	TO	1.221403E 07	-3.00000E-01	TO	-2.00000E-01
3	1.221403E 07	TO	1.105171E 07	-2.00000E-01	TO	-1.00000E-01
4	1.105171E 07	TO	1.000000E 07	-1.00000E-01	TO	0.
5	1.000000E 07	TO	9.048374E 06	0.	TO	1.00000E-01
6	9.048374E 06	TO	8.187308E 06	1.00000E-01	TO	2.00000E-01
7	8.187308E 06	TO	7.408182E 06	2.00000E-01	TO	3.00000E-01
8	7.408182E 06	TO	6.703200E 06	3.00000E-01	TO	4.00000E-01
9	6.703200E 06	TO	6.065307E 06	4.00000E-01	TO	5.00000E-01
10	6.065307E 06	TO	5.488116E 06	5.00000E-01	TO	6.00000E-01
11	5.488116E 06	TO	4.965853E 06	6.00000E-01	TO	7.00000E-01
12	4.965853E 06	TO	4.493290E 06	7.00000E-01	TO	8.00000E-01
13	4.493290E 06	TO	4.065697E 06	8.00000E-01	TO	9.00000E-01
14	4.065697E 06	TO	3.678794E 06	9.00000E-01	TO	1.00000E 00
15	3.678794E 06	TO	3.328711E 06	1.00000E 00	TO	1.10000E 00
16	3.328711E 06	TO	3.011942E 06	1.10000E 00	TO	1.20000E 00
17	3.011942E 06	TO	2.725318E 06	1.20000E 00	TO	1.30000E 00
18	2.725318E 06	TO	2.465970E 06	1.30000E 00	TO	1.40000E 00
19	2.465970E 06	TO	2.231302E 06	1.40000E 00	TO	1.50000E 00
20	2.231302E 06	TO	2.018965E 06	1.50000E 00	TO	1.60000E 00
21	2.018965E 06	TO	1.826835E 06	1.60000E 00	TO	1.70000E 00
22	1.826835E 06	TO	1.652989E 06	1.70000E 00	TO	1.80000E 00
23	1.652989E 06	TO	1.495686E 06	1.80000E 00	TO	1.90000E 00
24	1.495686E 06	TO	1.353353E 06	1.90000E 00	TO	2.00000E 00
25	1.353353E 06	TO	1.224564E 06	2.00000E 00	TO	2.10000E 00
26	1.224564E 06	TO	1.108032E 06	2.10000E 00	TO	2.20000E 00
27	1.108032E 06	TO	1.002589E 06	2.20000E 00	TO	2.30000E 00
28	1.002589E 06	TO	9.071796E 05	2.30000E 00	TO	2.40000E 00
29	9.071796E 05	TO	8.208501E 05	2.40000E 00	TO	2.50000E 00
30	8.208501E 05	TO	7.427359E 05	2.50000E 00	TO	2.60000E 00
31	7.427359E 05	TO	6.720552E 05	2.60000E 00	TO	2.70000E 00
32	6.720552E 05	TO	6.081007E 05	2.70000E 00	TO	2.80000E 00
33	6.081007E 05	TO	5.502323E 05	2.80000E 00	TO	2.90000E 00
34	5.502323E 05	TO	4.978708E 05	2.90000E 00	TO	3.00000E 00
35	4.978708E 05	TO	4.504921E 05	3.00000E 00	TO	3.10000E 00
36	4.504921E 05	TO	4.076221E 05	3.10000E 00	TO	3.20000E 00
37	4.076221E 05	TO	3.688317E 05	3.20000E 00	TO	3.30000E 00
38	3.688317E 05	TO	3.337327E 05	3.30000E 00	TO	3.40000E 00
39	3.337327E 05	TO	3.019739E 05	3.40000E 00	TO	3.50000E 00
40	3.019739E 05	TO	2.732373E 05	3.50000E 00	TO	3.60000E 00
41	2.732373E 05	TO	2.472353E 05	3.60000E 00	TO	3.70000E 00
42	2.472353E 05	TO	2.237078E 05	3.70000E 00	TO	3.80000E 00
43	2.237078E 05	TO	2.024192E 05	3.80000E 00	TO	3.90000E 00
44	2.024192E 05	TO	1.831564E 05	3.90000E 00	TO	4.00000E 00
45	1.831564E 05	TO	1.657268E 05	4.00000E 00	TO	4.10000E 00
46	1.657268E 05	TO	1.499558E 05	4.10000E 00	TO	4.20000E 00
47	1.499558E 05	TO	1.356856E 05	4.20000E 00	TO	4.30000E 00
48	1.356856E 05	TO	1.227734E 05	4.30000E 00	TO	4.40000E 00
49	1.227734E 05	TO	1.110900E 05	4.40000E 00	TO	4.50000E 00
50	1.110900E 05	TO	8.651698E 04	4.50000E 00	TO	4.75000E 00

WATER--5 GROUP SET USING GGC-4 SOURCE DECK

FINE GROUP STRUCTURE

GROUP	ENERGY INTERVAL(E.V.)			LETHARGY INTERVAL		
51	8.651698E 04	TO	6.737949E 04	4.75000E 00	TO	5.00000E 00
52	6.737949E 04	TO	5.247520E 04	5.00000E 00	TO	5.25000E 00
53	5.247520E 04	TO	4.086773E 04	5.25000E 00	TO	5.50000E 00
54	4.086773E 04	TO	3.182782E 04	5.50000E 00	TO	5.75000E 00
55	3.182782E 04	TO	2.478753E 04	5.75000E 00	TO	6.00000E 00
56	2.478753E 04	TO	1.930455E 04	6.00000E 00	TO	6.25000E 00
57	1.930455E 04	TO	1.503440E 04	6.25000E 00	TO	6.50000E 00
58	1.503440E 04	TO	1.170880E 04	6.50000E 00	TO	6.75000E 00
59	1.170880E 04	TO	9.118823E 03	6.75000E 00	TO	7.00000E 00
60	9.118823E 03	TO	7.101746E 03	7.00000E 00	TO	7.25000E 00
61	7.101746E 03	TO	5.530846E 03	7.25000E 00	TO	7.50000E 00
62	5.530846E 03	TO	4.307427E 03	7.50000E 00	TO	7.75000E 00
63	4.307427E 03	TO	3.354627E 03	7.75000E 00	TO	8.00000E 00
64	3.354627E 03	TO	2.612587E 03	8.00000E 00	TO	8.25000E 00
65	2.612587E 03	TO	2.034684E 03	8.25000E 00	TO	8.50000E 00
66	2.034684E 03	TO	1.584614E 03	8.50000E 00	TO	8.75000E 00
67	1.584614E 03	TO	1.234098E 03	8.75000E 00	TO	9.00000E 00
68	1.234098E 03	TO	9.611169E 02	9.00000E 00	TO	9.25000E 00
69	9.611169E 02	TO	7.485186E 02	9.25000E 00	TO	9.50000E 00
70	7.485186E 02	TO	5.829468E 02	9.50000E 00	TO	9.75000E 00
71	5.829468E 02	TO	4.539995E 02	9.75000E 00	TO	1.00000E 01
72	4.539995E 02	TO	3.535751E 02	1.00000E 01	TO	1.02500E 01
73	3.535751E 02	TO	2.753646E 02	1.02500E 01	TO	1.05000E 01
74	2.753646E 02	TO	2.144542E 02	1.05000E 01	TO	1.07500E 01
75	2.144542E 02	TO	1.670171E 02	1.07500E 01	TO	1.10000E 01
76	1.670171E 02	TO	1.300730E 02	1.10000E 01	TO	1.12500E 01
77	1.300730E 02	TO	1.013010E 02	1.12500E 01	TO	1.15000E 01
78	1.013010E 02	TO	7.889328E 01	1.15000E 01	TO	1.17500E 01
79	7.889328E 01	TO	6.144214E 01	1.17500E 01	TO	1.20000E 01
80	6.144214E 01	TO	4.785119E 01	1.20000E 01	TO	1.22500E 01
81	4.785119E 01	TO	3.726654E 01	1.22500E 01	TO	1.25000E 01
82	3.726654E 01	TO	2.902321E 01	1.25000E 01	TO	1.27500E 01
83	2.902321E 01	TO	2.260330E 01	1.27500E 01	TO	1.30000E 01
84	2.260330E 01	TO	1.760347E 01	1.30000E 01	TO	1.32500E 01
85	1.760347E 01	TO	1.370960E 01	1.32500E 01	TO	1.35000E 01
86	1.370960E 01	TO	1.067704E 01	1.35000E 01	TO	1.37500E 01
87	1.067704E 01	TO	8.315290E 00	1.37500E 01	TO	1.40000E 01
88	8.315290E 00	TO	6.475955E 00	1.40000E 01	TO	1.42500E 01
89	6.475955E 00	TO	5.043478E 00	1.42500E 01	TO	1.45000E 01
90	5.043478E 00	TO	3.927865E 00	1.45000E 01	TO	1.47500E 01
91	3.927865E 00	TO	3.059024E 00	1.47500E 01	TO	1.50000E 01
92	3.059024E 00	TO	2.382370E 00	1.50000E 01	TO	1.52500E 01
93	2.382370E 00	TO	1.855392E 00	1.52500E 01	TO	1.55000E 01
94	1.855392E 00	TO	1.444981E 00	1.55000E 01	TO	1.57500E 01
95	1.444981E 00	TO	1.125352E 00	1.57500E 01	TO	1.60000E 01
96	1.125352E 00	TO	8.764252E-01	1.60000E 01	TO	1.62500E 01
97	8.764252E-01	TO	6.825607E-01	1.62500E 01	TO	1.65000E 01
98	6.825607E-01	TO	5.315788E-01	1.65000E 01	TO	1.67500E 01
99	5.315788E-01	TO	4.139940E-01	1.67500E 01	TO	1.70000E 01

GNE/PHYS 69-8

WATER--5 GROUP SET USING GGC-4 SOURCE DECK

BROAD GROUP STRUCTURE

BROAD GROUP	ENERGY INTERVAL(E.V.)
1	1.491825E 07 TO 5.488116E 06
2	5.488116E 06 TO 1.652989E 06
3	1.652989E 06 TO 1.356856E 05
4	1.356856E 05 TO 1.670171E 02
5	1.670171E 02 TO 4.139940E-01

```
NUCLIDE NO. 1.0000000 HAS 1-D ARRAY

NUCLIDE NO. 1.0000000 HAS P-0,P-1,P-2,AND P-3 ARRAY

NUCLIDE NO. 1.0000000 DOES NOT HAVE INELASTIC ARRAY

NUCLIDE NO. 1.0000000 DOES NOT HAVE N-2N ARRAY

NUCLIDE NO. 1.0000000 HAS TOTAL ISO. SCATTER ARRAY
```

NUCLIDE NO. 8.0200000 HAS 1-D ARRAY

NUCLIDE NO. 8.0200000 HAS P-0,P-1,P-2,AND P-3 ARRAY

NUCLIDE NO. 8.0200000 HAS INELASTIC ARRAY

NUCLIDE NO. 8.0200000 DOES NOT HAVE N-2N ARRAY

NUCLIDE NO. 8.0200000 HAS TOTAL ISO. SCATTER ARRAY

WATER--5 GROUP SET USING GGC-4 SOURCE DECK

GROUP	FLUX
1	5.232288E-04
2	1.462861E-03
3	3.336402E-03
4	7.471954E-03
5	1.472939E-02
6	2.565449E-02
7	4.339071E-02
8	6.701126E-02
9	1.016520E-01
10	1.316190E-01
11	1.797270E-01
12	2.133987E-01
13	2.608718E-01
14	2.672685E-01
15	3.020317E-01
16	3.786525E-01
17	4.193974E-01
18	4.496019E-01
19	4.579706E-01
20	4.147970E-01
21	3.911531E-01
22	4.168634E-01
23	4.063766E-01
24	4.044704E-01
25	3.698691E-01
26	3.695762E-01
27	2.976449E-01
28	2.994022E-01
29	3.531688E-01
30	3.666530E-01
31	3.153495E-01
32	2.894981E-01
33	2.707462E-01
34	2.501992E-01
35	2.040345E-01
36	1.881376E-01
37	2.065525E-01
38	2.201407E-01
39	2.071412E-01
40	1.930225E-01
41	1.811432E-01
42	1.707197E-01
43	1.622364E-01
44	1.553621E-01
45	1.495198E-01
46	1.452956E-01
47	1.361418E-01
48	1.304334E-01
49	1.252771E-01
50	2.931348E-01

WATER--5 GROUP SET USING GGC-4 SOURCE DECK

GROUP	FLUX
51	2.709523E-01
52	2.519898E-01
53	2.367955E-01
54	2.252021E-01
55	2.151739E-01
56	2.083026E-01
57	2.031521E-01
58	1.984380E-01
59	1.947144E-01
60	1.922773E-01
61	1.905074E-01
62	1.892594E-01
63	1.886703E-01
64	1.882078E-01
65	1.876260E-01
66	1.869519E-01
67	1.862143E-01
68	1.856203E-01
69	1.849905E-01
70	1.844139E-01
71	1.839628E-01
72	1.836022E-01
73	1.833264E-01
74	1.831110E-01
75	1.829322E-01
76	1.827969E-01
77	1.826802E-01
78	1.824064E-01
79	1.820083E-01
80	1.816798E-01
81	1.814170E-01
82	1.812050E-01
83	1.810321E-01
84	1.808792E-01
85	1.807562E-01
86	1.806410E-01
87	1.801672E-01
88	1.789832E-01
89	1.779842E-01
90	1.771898E-01
91	1.765532E-01
92	1.760275E-01
93	1.755907E-01
94	1.752122E-01
95	1.748724E-01
96	1.742999E-01
97	1.719690E-01
98	1.684121E-01
99	1.658085E-01

WATER--5 GROUP SET USING GGC-4 SOURCE DECK

BROAD GROUP AVERAGED MACROSCOPIC CROSS SECTIONS

GROUP	SIGMA ABSORPTION	SIGMA TOTAL	SIGMA FISSION
1	2.167554E-03	1.250808E-01	0.
2	3.959128E-04	2.155311E-01	0.
3	0.	5.218956E-01	0.
4	6.715012E-05	1.317152E 00	0.
5	1.691048E-03	1.508743E 00	0.

WATER--5 GROUP SET USING GGC-4 SOURCE DECK

GROUP	NU*SIGMA FISSION
1	0.
2	0.
3	0.
4	0.
5	0.

WATER--5 GROUP SET USING GGC-4 SOURCE DECK

BROAD GROUP AVERAGED MACROSCOPIC CROSS SECTIONS

GROUP	P-0	INELASTIC	N-2N	TOTAL SCATTER
FROM 1 TO 1	3.491491E-02	2.022698E-05	0.	3.493514E-02
FROM 1 TO 2	6.291986E-02	1.346142E-03	0.	6.426600E-02
FROM 1 TO 3	2.011283E-02	1.779472E-03	0.	2.189231E-02
FROM 1 TO 4	1.796387E-03	2.125040E-05	0.	1.817637E-03
FROM 1 TO 5	2.208431E-06	4.846245E-10	0.	2.208916E-06
FROM 2 TO 2	1.013809E-01	0.	0.	1.013809E-01
FROM 2 TO 3	1.050091E-01	0.	0.	1.050091E-01
FROM 2 TO 4	8.734505E-03	0.	0.	8.734505E-03
FROM 2 TO 5	1.073797E-05	0.	0.	1.073797E-05
FROM 3 TO 3	3.730391E-01	0.	0.	3.730391E-01
FROM 3 TO 4	1.486773E-01	0.	0.	1.486773E-01
FROM 3 TO 5	1.788487E-04	0.	0.	1.788487E-04
FROM 4 TO 4	1.137157E 00	0.	0.	1.137157E 00
FROM 4 TO 5	1.794877E-01	0.	0.	1.794877E-01
FROM 5 TO 5	1.275828E 00	0.	0.	1.275828E 00

WATER--5 GROUP SET USING GGC-4 SOURCE DECK

BROAD GROUP AVERAGED MACROSCOPIC CROSS SECTIONS

	P-1	P-2	P-3
FROM 1 TO 1	6.652408E-02	8.839789E-02	9.911263E-02
FROM 1 TO 2	1.029138E-01	8.854647E-02	-5.693567E-03
FROM 1 TO 3	2.168826E-02	-2.922178E-02	-5.571292E-02
FROM 1 TO 4	5.223204E-04	-4.348100E-03	-1.789059E-03
FROM 1 TO 5	2.255498E-08	-5.520861E-06	-7.894038E-08
FROM 2 TO 2	1.802595E-01	2.221573E-01	1.886510E-01
FROM 2 TO 3	1.669813E-01	4.189492E-02	-1.495191E-01
FROM 2 TO 4	4.203339E-03	-1.990882E-02	-1.381823E-02
FROM 2 TO 5	1.815097E-07	-2.684201E-05	-6.352366E-07
FROM 3 TO 3	6.211619E-01	6.275355E-01	2.882293E-01
FROM 3 TO 4	2.037582E-01	-6.702652E-02	-2.798133E-01
FROM 3 TO 5	8.942934E-06	-4.466725E-04	-3.127661E-05
FROM 4 TO 4	2.166618E 00	1.604800E 00	3.325324E-01
FROM 4 TO 5	2.356694E-01	-1.098725E-01	-3.323999E-01
FROM 5 TO 5	2.474773E 00	1.869701E 00	4.282221E-01

WATER--5 GROUP SET USING GGC-4 SOURCE DECK

GROUP	AVERAGE COS(THETA)	DIFF. COEFF.	SIGMA TRANSPORT
1	6.3474E-01	3.2392E 00	1.0291E-01
2	5.9268E-01	2.1444E 00	1.5544E-01
3	5.5505E-01	1.0587E 00	3.1484E-01
4	6.3510E-01	5.6027E-01	5.9495E-01
5	6.4658E-01	4.8746E-01	6.8382E-01

```
THE INPUT DATA FOR THIS PROBLEM IS ...
MAGNESIUM--4 GROUP SET USING ENDF/B SOURCE DECK
  4  1  1
 13 27 54 71
 1014 4 2 12341    0.3900E-00
0.412166E-040.113325E-030.277033E-030.609152E-030.121752E-020.223311E-02
0.379104E-020.600338E-020.893037E-020.125583E-010.167906E-010.214541E-01
0.263197E-010.311313E-010.356364E-010.396133E-010.428906E-010.453567E-01
0.469620E-010.477125E-010.476601E-010.468902E-010.455093E-010.436340E-01
0.413815E-010.388629E-010.361787E-010.334159E-010.306470E-010.279301E-01
0.253096E-010.228182E-010.204779E-010.183021E-010.162969E-010.144631E-01
0.127972E-010.112926E-010.994080E-020.873164E-020.765445E-020.669823E-02
0.585207E-020.510541E-020.444822E-020.387106E-020.336522E-020.292267E-02
0.253612E-020.495894E-020.345783E-020.240338E-020.166632E-020.115306E-02
0.796687E-030.549814E-030.379095E-030.           0.           0.
0.           0.           0.           0.           0.           0.
0.           0.           0.           0.           0.           0.
0.           0.           0.           0.           0.           0.
0.           0.           0.           0.           0.           0.
0.           0.           0.           0.           0.           0.
0.           0.           0.           0.           0.           0.
0.           0.           0.
```

```
FOLLOWING INFO. INDICATES WHETHER OR NOT YOU HAVE   THE DATA FOR THE RIGHT
..............      MATERIAL NUMBER  1014        100  GROUPS      P0    NUCLIDES
 THE ID NUMBER FOR MATERIAL 1014   P0  IS   100
..............      MATERIAL NUMBER  1014        100  GROUPS      P1
 THE ID NUMBER FOR MATERIAL 1014   P1  IS   101
..............      MATERIAL NUMBER  1014        100  GROUPS      P2
 THE ID NUMBER FOR MATERIAL 1014   P2  IS   102
..............      MATERIAL NUMBER  1014        100  GROUPS      P3
 THE ID NUMBER FOR MATERIAL 1014   P3  IS   103
..............      MATERIAL NUMBER  1014        100  GROUPS      P4
 THE ID NUMBER FOR MATERIAL 1014   P4  IS   104
..............      MATERIAL NUMBER  1014        100  GROUPS      P5
..............      MATERIAL NUMBER  1014        100  GROUPS      P6
..............      MATERIAL NUMBER  1014        100  GROUPS      P7
..............      MATERIAL NUMBER  1014        100  GROUPS      P8
```

MAGNESIUM--4 GROUP SET USING ENDF/B SOURCE DECK

GROUP	FLUX
1	9.556081E-05
2	2.715339E-04
3	6.604603E-04
4	1.462057E-03
5	2.962775E-03
6	5.448673E-03
7	1.000503E-02
8	1.650977E-02
9	2.829213E-02
10	3.612760E-02
11	4.907892E-02
12	6.523898E-02
13	9.182791E-02
14	1.420293E-01
15	1.433325E-01
16	1.865966E-01
17	2.042672E-01
18	2.262544E-01
19	2.943707E-01
20	2.939286E-01
21	3.973576E-01
22	4.364193E-01
23	7.150182E-01
24	5.582233E-01
25	6.289372E-01
26	8.627642E-01
27	1.048663E 00
28	1.053914E 00
29	6.233043E-01
30	8.198278E-01
31	7.346589E-01
32	9.804919E-01
33	8.603694E-01
34	8.654983E-01
35	5.711158E-01
36	4.032375E-01
37	6.436866E-01
38	5.168559E-01
39	3.627552E-01
40	2.668761E-01
41	2.763459E-01
42	3.789079E-01
43	4.893963E-01
44	5.830240E-01
45	6.528828E-01
46	6.925256E-01
47	7.096412E-01
48	6.885358E-01
49	6.048491E-01
50	3.642916E-01

MAGNESIUM--4 GROUP SET USING ENDF/B SOURCE DECK

GROUP	FLUX
51	9.726620E-01
52	1.831392E 00
53	1.956725E 00
54	1.904309E 00
55	1.771586E 00
56	9.586921E-01
57	1.704383E 00
58	1.948184E 00
59	1.917449E 00
60	1.926907E 00
61	1.934948E 00
62	1.941029E 00
63	1.945448E 00
64	1.948694E 00
65	1.950924E 00
66	1.952301E 00
67	1.952993E 00
68	1.953340E 00
69	1.953515E 00
70	1.953345E 00
71	1.952657E 00
72	1.951796E 00
73	1.950762E 00
74	1.949555E 00
75	1.948349E 00
76	1.946971E 00
77	1.945078E 00
78	1.943015E 00
79	1.940611E 00
80	1.937868E 00
81	1.934787E 00
82	1.931369E 00
83	1.927447E 00
84	1.923024E 00
85	1.918102E 00
86	1.912519E 00
87	1.906110E 00
88	1.904762E 00
89	1.904762E 00
90	1.902049E 00
91	1.897636E 00
92	1.891695E 00
93	1.884401E 00
94	1.878294E 00
95	1.861902E 00
96	1.842328E 00
97	1.820671E 00
98	1.796367E 00
99	1.769217E 00

MAGNESIUM--4 GROUP SET USING ENDF/B SOURCE DECK

BROAD GROUP AVERAGED MACROSCOPIC CROSS SECTIONS

GROUP	SIGMA ABSORPTION	SIGMA TOTAL	NU*SIGMA FISSION
1	1.389300E-02	7.408909E-01	0.
2	3.125178E-03	1.061269E 00	0.
3	4.178245E-03	2.170218E 00	0.
4	8.041124E-04	1.405475E 00	0.

MAGNESIUM--4 GROUP SET USING ENDF/B SOURCE DECK

BROAD GROUP AVERAGED MACROSCOPIC CROSS SECTIONS

GROUP	P-0	P-1	P-2	P-3
FROM 1 TO 1	4.150959E-01	6.905661E-01	7.391705E-01	5.947326E-01
FROM 1 TO 2	2.912752E-01	-4.153767E-02	2.015781E-02	-3.518052E-02
FROM 1 TO 3	2.062810E-02	0.	0.	0.
FROM 1 TO 4	0.	0.	0.	0.
FROM 2 TO 2	9.506294E-01	9.948039E-01	7.249279E-01	2.160423E-01
FROM 2 TO 3	1.075208E-01	-2.140035E-02	-1.247154E-02	-5.033118E-02
FROM 2 TO 4	2.890327E-05	0.	0.	0.
FROM 3 TO 3	2.123368E 00	9.266635E-01	9.475510E-01	1.486644E-01
FROM 3 TO 4	4.249124E-02	-4.851247E-02	7.776200E-03	-1.838741E-03
FROM 4 TO 4	1.377493E 00	4.843637E-02	8.710037E-03	9.448551E-04

MAGNESIUM--4 GROUP SET USING ENDF/B SOURCE DECK

GROUP	AVERAGE COS(THETA)	DIFF. COEFF.	SIGMA TRANSPORT
1	5.5454E-01	6.5270E-01	5.1070E-01
2	3.4882E-01	4.5683E-01	7.2967E-01
3	1.4547E-01	1.7908E-01	1.8613E 00
4	1.1721E-02	2.3992E-01	1.3893E 00

MAGNESIUM--4 GROUP SET USING ENDF/B SOURCE DECK

BROAD GROUP AVERAGED MACROSCOPIC CROSS SECTIONS

GROUP	P-4	P-5	P-6	P-7
FROM 1 TO 1	4.238755E-01	0.	0.	0.
FROM 1 TO 2	4.189399E-03	0.	0.	0.
FROM 1 TO 3	0.	0.	0.	0.
FROM 1 TO 4	0.	0.	0.	0.
FROM 2 TO 2	6.132632E-02	0.	0.	0.
FROM 2 TO 3	-1.042430E-02	0.	0.	0.
FROM 2 TO 4	0.	0.	0.	0.
FROM 3 TO 3	1.382124E-02	0.	0.	0.
FROM 3 TO 4	7.583821E-05	0.	0.	0.
FROM 4 TO 4	3.794491E-04	0.	0.	0.

VITA

Bruce D. Green, was born on 16 August 1942, in Portland, Oregon, the son of Sammuel L. Green and Mary J. Green. He graduated from Blackfoot High School, Blackfoot, Idaho, in May 1960 and attended Seattle University, Idaho State University, and then Kansas State University, from which he received the degree of Bachelor of Science in Nuclear Engineering on 15 June 1967. After attending Officer Training School he was commissioned a Second Lieutenant in the USAF on 21 August 1967, and was then assigned to the Air Force Institute of Technology, Resident School of Engineering.

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AFIT Thesis

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First Lieutenant USAF

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11. SUPPLEMENTARY NOTES	12. SPONSORING MILITARY ACTIVITY

13. ABSTRACT

New Barnyard, a moments code, was written to calculate multigroup neutron macroscopic cross sections. Total, absorption, fission, and scattering transfer (group to group) cross sections can be calculated. The transport cross section, diffusion coefficient, and the average cosine of the scattering angle can also be calculated for each group. The energy range of these cross sections extends from 14.918 MeV to .4139 eV. Two versions of the moments code were written so that two different data sources could be used. P0 through P8 elastic scattering transfer cross sections can be calculated for 20 broad groups.

DD FORM 1 JAN 64 1473

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14. KEY WORDS	LINK A		LINK B		LINK C	
	ROLE	WT	ROLE	WT	ROLE	WT
Neutron Cross Section Code						

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It is highly desirable that the abstract of classified reports be unclassified. Each paragraph of the abstract shall end with an indication of the military security classification of the information in the paragraph, represented as *(TS)*, *(S)*, *(C)*, or *(U)*.

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