INDRA: A Program System for Calculating the Neutronics and Photonics Characteristics of a Fusion Reactor Blanket

R.T. Perry*
H. Gorenflo
W. Dänner

IPP 4/137 Januar 1976

MAX-PLANCK-INSTITUT FÜR PLASMAPHYSIK

8046 GARCHING BEI MÜNCHEN

MAX-PLANCK-INSTITUT FÜR PLASMAPHYSIK

GARCHING BEI MÜNCHEN

INDRA: A Program System for Calculating the Neutronics and Photonics Characteristics of a Fusion Reactor Blanket

R.T. Perry*
H. Gorenflo
W. Dänner

IPP 4/137 Januar 1976

* Present address:
Princeton Plasma Physics Laboratory
Princeton, New Jersey 08540
U.S.A.

Die nachstehende Arbeit wurde im Rahmen des Vertrages zwischen dem Max-Planck-Institut für Plasmaphysik und der Europäischen Atomgemeinschaft über die Zusammenarbeit auf dem Gebiete der Plasmaphysik durchgeführt.

IPP 4/137

INDRA: A Program System for Calculating the Neutronics and Photonics Characteristics of a Fusion Reactor Blanket

R.T. Perry
H. Gorenflo
W. Dänner

Januar 1976

Abstract

INDRA is a program system for calculating the neutronics and photonics characteristics of fusion reactor blankets. It incorporates a total of 19 different codes and 5 large data libraries. 10 of the codes are available from the code distribution organizations. Some of them, however, have been slightly modified in order to permit a convenient transfer of information from one program module to the next. The remaining 9 programs have been prepared by the authors to complete the system with respect to flexibility and to facilitate the handling of the results.

The present report should be considered as a users manual rather than a scientific publication, although one chapter is devoted to the theoretical background of transport calculations and the evaluation of heat production rates. The bigger part of the report, however, describes the logical structure of the program system and gives detailed information about each of the codes and libraries. Complete input descriptions and operational notes have been included for the user's convenience.

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

1. INTRODUCTION

The evaluation of the neutronics and photonics characteristics of a fusion reactor blanket represents the basis for any further technical investigation of this reactor component. The most essential physical quantities which can be gained by such calculations are: the neutron and gamma fluxes, the gamma source distribution, the tritium breeding ratio, and the heat production by neutron and gamma ray interaction with matter. The space and energy dependent neutron flux distribution may also be used to derive information about partial reaction rates the most vital of which are displacement and gas production rates necessary for radiation damage studies, and activation rates.

Since some years a number of computer programs and data libraries are available which allow to answer one or another of these questions separately, the program system INDRA* which is described in this report was developed to combine all necessary codes and libraries into a selfconsistent system. It contains transport codes, cross-section reduction codes, data retrieval programs, data handling codes, and codes which provide necessary input information such as kerma factors and gamma production cross-sections, most of which are readily available from the code distribution centers. To some of these codes, however, modifications have been applied in order to permit a direct transfer of information from one step of the analysis to the next.

A limited number of auxiliary programs was supplied by the authors. These programs represent either missing links in the procedure of the analysis and data preparation or programs which facilitate the handling of the final results.

---

* INDRA: Hindu God of Fire and Water

Large data libraries are also included which contain neutron cross-sections, gamma cross-sections, and gamma spectral data.

The entire code system is constructed in a modular manner. In this way it offers a high degree of transparency and flexibility to the user preserving the possibility of a quick exchange of single modules if necessary. It contains a total of approximately 14000 cards and at present makes use of five data libraries.

The present report should be regarded as a user manual rather than a scientific publication. Some physical background, however, is presented in chapter II. Chapter III gives an introduction to the logical structure of the entire system. In chapter IV a few remarks about original tapes are summarized which are of internal value only. Chapter V, finally, discusses the individual codes and libraries. Here a complete input description is presented together with a number of most valuable hints for optimum usage.

2. THEORETICAL BACKGROUND

2.1 The Importance of Fusion Neutrons

Thermonuclear reactions proceed through microscopic interactions. For fusion to occur, two nuclei must be brought into close proximity, where they fuse, forming a nucleus with increased binding energy, thus releasing energy. The conditions under which this process occurs, however, are far from simple and have been the subject of intense investigations during the last twenty years. The results of the investigations, although encouraging, have not yet demonstrated feasibility.

The elements which are candidates for such a process are limited to the lighter elements. In Figure 1 a plot of binding energy per nucleon as a function of the mass number of the element is shown. From Fig. 1, one may draw the conclusion that reactions of interest would involve the isotopes of hydrogen, helium, and lithium.

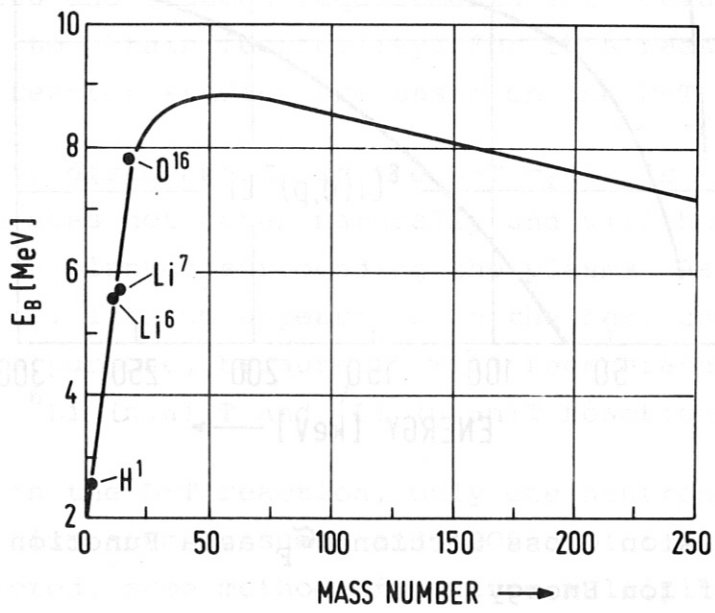


Fig. 1: Binding Energy E_B per Nucleon as a Function of Mass Number

Extremely high kinetic energies are required so that individual atoms may overcome nuclear coulombic forces and collide. Currently, the emphasis is on creating the conditions for fusion to occur, i.e., the production of a hot plasma and its containment within a magnetic bottle. A theoretical treatment of various devices and associated problems is treated elsewhere [1].

In Figure 2 a plot of fusion cross sections as a function of energy is given. On the basis of the

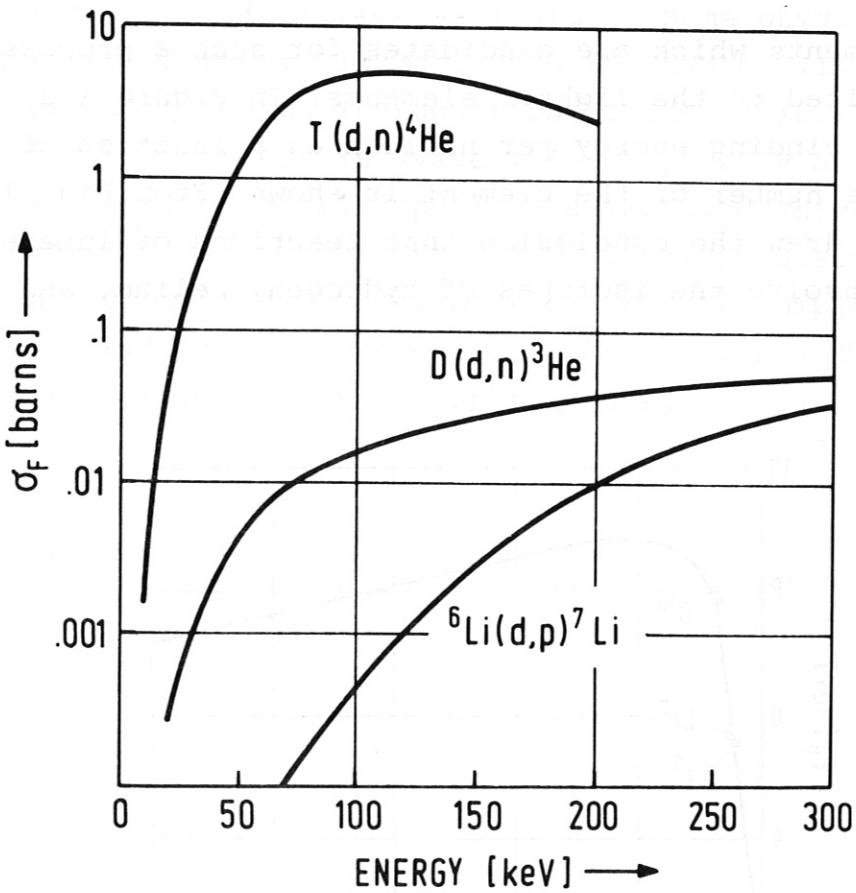


Fig. 2: Fusion Cross Section σ_F as a Function of Ion Energy

highest reaction cross section, the likely choices for the reaction are:

$$D + D \begin{cases} \rightarrow T + 1.01\ \text{MeV} + p + 3.03\ \text{MeV} \\ \rightarrow {}^3\text{He} + .82\ \text{MeV} + n + 2.45\ \text{MeV} \end{cases} \quad (1)$$

$$D + T \longrightarrow {}^4\text{He} + 3.52\ \text{MeV} + n + 14.06\ \text{MeV} \quad (2)$$

$$D + {}^3\text{He} \rightarrow {}^4\text{He} + 3.67\ \text{MeV} + p + 14.67\ \text{MeV} \quad (3)$$

Tritium and ^{3}He formed in the D-D reaction would be consumed in D-T and D-^{3}He reactions respectively if the ion temperatures and sufficient confinement times are maintained. This would result in an overall energy release of 21.6 MeV i.e.,

$$3D \longrightarrow {}^4\text{He} + n + p + 21.6\ \text{MeV} \quad (4)$$

Note in Figure 2, that the cross section for the D-T reaction is roughly 100 times greater than the D-D reaction cross section over the ion temperature range of interest thus requiring much smaller plasma densities and reduced requirements for containment devices to obtain feasibility. For this reason most fusion reactor studies are based on the D-T cycle.

The major disadvantage of the D-T cycle is that tritium does not occur naturally and will have to be bred in a blanket surrounding the plasma. As noted elsewhere, lithium appears to be the best choice for this purpose, because it will regenerate tritium through ^{6}Li (n,α) T and ^{7}Li (n,αn)T reactions.

Since, in the D-T reaction, only one neutron is formed per tritium atom consumed and 100% efficiency cannot be expected, some method of neutron multiplication must be provided. Fortunately, at the neutron energies

involved the cross sections for multiplication processes are large, and previous calculations [2, 3] indicate that the multiplication is sufficient to obtain tritium breeding ratios in excess of one. In Table I a sample of the possible neutron multiplication processes and the cross sections involved are given.

TABLE I Sample Nuclear Reaction Processes and Cross Sections that Produce Neutron Multiplication.

Reaction	Cross Section at 14. MeV
$^{7}Li(n,\alpha n)T$	.28 b
Fe(n,2n)Fe	.35 b
Mo(n,2n)Mo	1.4 b
$^{9}Be(n,2n)2^{4}He$	.5 b

In the D-D reaction some 66% of the energy appears with charged particles. For the D-T reaction this percentage is 20. Although the possibility exists of extracting this energy dierctly in the form of electricity, it is obvious that the energy carried by the neutrons will have to be deposited in the blanket, where it can be extracted and utilized. It is the energy carried by the neutrons that leads to the heat sources and resulting power profile in the blanket.

There is, however, a heat source other than that produced from neutron interactions. This heat source results from synchrotron radiation, bremsstrahlung, and charged particles from the plasma. The synchrotron radiation and bremsstrahlung are in the x-ray range or lower and their intensity is a function of ion temperature and magnetic

field strength. The heat generated is deposited in the skin of the vacuum wall and would lead to design difficulties. These processes have been treated elsewhere [4] and will not be discussed further.

Although the calculational methods employed in the program system are applicable to all reactor types and conceptual blankets, the emphasis is on blankets for the D-T cycle. As noted earlier, the D-T cycle is expected to be the cycle on which the first generation of reactors is based. However, at this stage of blanket development, the designs are rather independent of the reactor type. The current conceptual blanket designs could change drastically, should the D-D cycle be utilized in the future rather than the D-T cycle, since tritium breeding would not be required.

2.2 Neutron Transport

Within the plasma region, in a steady state reactor, neutrons are generated isotropically. For the D-T reactor, the majority of these neutrons carry 14.06 MeV, the remaining .1 to 1.%, depending on the ion temperature, carry 2.45 MeV and are the result of D-D reactions within the plasma. The transport of the neutrons through the blanket is described by the time independent Boltzman equation:

$$\bar{\Omega} \cdot \nabla \phi(\bar{r},\bar{\Omega},E) + \Sigma_T \phi(\bar{r},\bar{\Omega},E) =$$

$$S(\bar{r},\bar{\Omega},E) + \int_{E'}\int_{\Omega'} \Sigma(\bar{r},\bar{\Omega}' \rightarrow \Omega, E' \rightarrow E)\ \phi(\bar{r},\bar{\Omega},E)\ d\bar{\Omega}'\, dE' \qquad (5)$$

where $\phi(\bar{r},\bar{\Omega},E)$ is the vector flux at spatial location $\bar{r}$, in the direction $\bar{\Omega}$, and with energy E; $S(\bar{r},\bar{\Omega},E)$ is the

fixed source term; $\Sigma(\bar{r},\bar{\Omega}' \rightarrow \bar{\Omega},E' \rightarrow E)$ is the transfer cross section; and Σ_T is the total cross section.

Solutions to the Boltzman equation are well documented. Since the blankets may be described as cylinders, solutions in cylindrical geometry are applicable. In the solution considered here, the transfer cross sections are expanded in Legendre polynomials of the scattering angle

$$\Sigma(\bar{r},\bar{\Omega}' \rightarrow \bar{\Omega},\ E' \rightarrow E) = \sum_{\ell=0}^{\infty} \frac{2\ell+1}{4\pi} \Sigma_\ell(\bar{r},E' \rightarrow E)\ P_\ell(\bar{\Omega}' \cdot \bar{\Omega}) \qquad (6)$$

In cylindrical geometry the addition theorem is used to express $P_\ell(\bar{\Omega}' \cdot \bar{\Omega})$ as

$$P_\ell(\bar{\Omega}' \cdot \bar{\Omega}) = P_\ell(\mu)\ P_\ell(\mu') +$$

$$2 \sum_{m=1}^{\ell} \frac{(\ell-m)!}{(\ell+m)!} P_\ell^m(\mu')\ \cos\ [m(\Psi-\Psi')] \qquad (7)$$

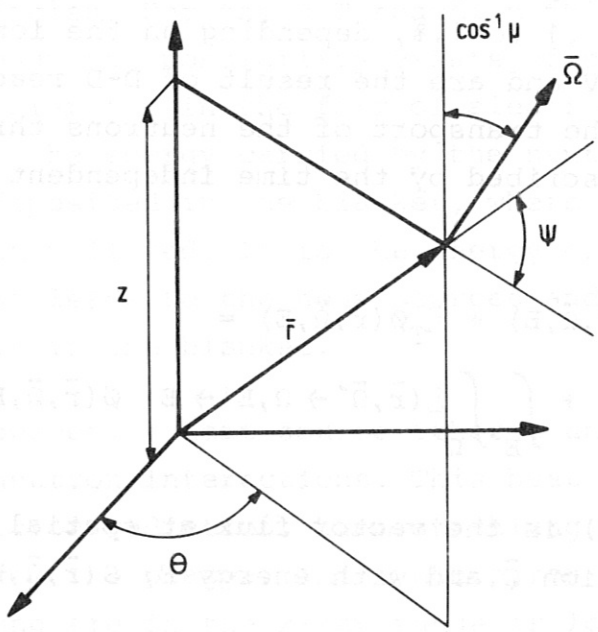


Fig. 3: Angular Coordinate System

The position of the angles Ψ and $\cos^{-1}(\mu)$ are indicated in Figure 3.

Substitution of Equation (6), containing the results of the addition theorem into Equation (5), yields the following equation:

$$\bar{\Omega}\cdot\nabla\phi(\bar{r},\bar{\Omega},E) + \Sigma_T(\bar{r},E)\ \phi(\bar{r},\bar{\Omega},E) = S(\bar{r},\bar{\Omega},E) + I_s(\bar{r},\bar{\Omega},E) \quad (8)$$

where the integral I_s is defined by:

$$I_s(\bar{r},\bar{\Omega},E) = \sum_{\ell=0}^{\infty} \frac{2\ell+1}{4\pi} \int_{E'} \Sigma_\ell(\bar{r},E'\rightarrow E)$$

$$x \left\{ P_\ell(\mu) \int_{\mu'} P_\ell(\mu')\ \phi(\bar{r},\mu',E)d\mu' + \right.$$

$$2 \sum_{m=1}^{\ell} \frac{(\ell-m)!}{(\ell+m)!} P_\ell^m(\mu) \int_0^{2\pi} \int_{-1}^{1} P_\ell(\mu')\ \cos[m(\psi-\psi')]\ x$$

$$\left. \phi(\bar{r},\bar{\Omega}',E')\ d\mu'\ d\psi \right\} dE' \quad (9)$$

Energy dependence of the flux is taken into account by dividing the energy range into small increments, ΔE_g, and integration of Equation (8) over these increments. The cross sections are group averaged and thus constant over the interval ΔE_g. Equation 8 may now be expressed in the form:

$$\bar{\Omega}\cdot\nabla\left[\int_{E_g}\phi(\bar{r},\bar{\Omega},E')\,dE'\right]+\Sigma_T(\bar{r},\Delta E_g)\int_{\Delta E_g}\phi(\bar{r},\bar{\Omega},E')dE'=$$

$$\int_{\Delta E_g}S(\bar{r},\Omega,E')\,dE'+\int_{\Delta E_g}I_s(\bar{r},\bar{\Omega},E')dE' \qquad (10)$$

Using the concept of group averaged cross sections over the energy interval ΔE, the integration over all other energies in the I_s term may be expressed as a summation.

$$I_s^g(\bar{r},\bar{\Omega})=\sum_{\ell=o}^{\infty}\frac{2\ell+1}{4\pi}\sum_h\Sigma_\ell^{h\to g}(\bar{r})\ x$$

$$\left\{P_\ell(\mu)\int P_\ell(\mu')\phi_g(r,\mu')d\mu'+2\sum_{m=1}^{\ell}\frac{(\ell-m)!}{(\ell+m)!}P_\ell^m(\mu)-\right.$$

$$\left.\int_{\psi'=o}^{2\pi}\int_{\mu'=-1}^{1}P_\ell(\mu')\cos[m(\psi-\psi')]\,\phi_h(r,\mu')\,d\mu'd\psi'\right\} \qquad (11)$$

Equation (10) may now be expressed in the form:

$$\bar{\Omega}\cdot\nabla\phi_g(\bar{r},\bar{\Omega})+\Sigma_T^g(\bar{r})\phi_g(\bar{r},\bar{\Omega})=S_g(\bar{r},\bar{\Omega})+I_s^g(\bar{r},\bar{\Omega}). \qquad (12)$$

In Equation (12), $\phi(\bar{r},\bar{\Omega})$, the vector flux, $\Sigma_T^g(\bar{r})$, the total cross section, and $S_g(\bar{r},\bar{\Omega})$, the fixed source term, have been averaged over the energy interval ΔE_g about E_g at spatial location $\bar{r}$, and in the direction $\bar{\Omega}$. $\Sigma_\ell^{h\to g}$ in equ. (11) is the Legendre polynomial expansion coefficient and represents the energy transfer cross section from the interval ΔE_h about E_h to the energy interval ΔE_g about E_g.

Equation (12) is the analytical form of the Boltzman equation for which a numerical solution is obtained by use of a transport code. The resultant vector fluxes obtained from the solution to the Boltzman equation are integrated over $\bar{\Omega}$ to obtain scalar fluxes:

$$\Phi_{ij} = \int_{\bar{\Omega}} \phi_{ij}(\bar{\Omega}) d\bar{\Omega} \tag{13}$$

where Φ_{ij} is the scaler flux and $\phi_{ij}(\bar{\Omega})$ is the vector flux at space point i and energy group j.

2.3 Gamma transport

Within the blanket, a space and energy dependent gamma source is produced from neutron interactions. These gamma sources arise from several processes, for example inelastic scattering and neutron reactions such as (n,p), (n,α), and radiative capture (n,γ).

One may define a gamma production cross section $\sigma_i(\bar{r},E'_n \rightarrow E_\gamma)$ as the probability that a neutron with energy E'_n at space point $\bar{r}$ produces a gamma ray with energy E_γ through process i. The fixed source term, $S(\bar{r},E)$, is the summation over all processes of the integral of the neutron energy flux, $\phi(\bar{r},E)$ times the gamma production cross section $\sigma_i(r,E'_n \rightarrow E_g)$.

$$S(\bar{r},E) = \sum_i \int_E \phi(\bar{r},E')\sigma_i(\bar{r},E'_n \rightarrow E_\gamma) dE' \tag{14}$$

The source term is assumed to be isotropic for calculational purposes.

The transport of the gamma rays through the blanket is described by the time independent Boltzman equation:

$$\bar{\Omega} \cdot \nabla \Psi(\bar{r},\bar{\Omega},E) + \sum_T \Psi(\bar{r},\bar{\Omega},E) = S(\bar{r},E)$$

$$+ \int_{E'}\int_{\Omega'} \sum(\bar{r},\bar{\Omega}' \rightarrow \bar{\Omega},E' \rightarrow E)\ \Psi(\bar{r},\bar{\Omega},E)\ d\bar{\Omega}'dE' \qquad (15)$$

where $\Psi(\bar{r},\bar{\Omega},E)$ is the vector gamma flux at spatial location $\bar{r}$, in the direction $\bar{\Omega}$, and with energy E; $S(\bar{r},\bar{\Omega},E)$ is the isotropic fixed source term; $\sum(\bar{r},\bar{\Omega}' \rightarrow \bar{\Omega},E)$ is the transfer cross section; and $\sum_T$ is the total cross section. This equation is identical in form to Equation (5), which describes the neutron transport in the blanket. Thus the solution to Equation (15) is identical to that obtained for Equation (5).

The solution to the Boltzman equation requires the use of group averaged total and transfer cross sections. The total cross section is the sum of the collision, pair-production and photoelectric cross sections. The pair production and photoelectric cross sections are obtained experimentally, and the collision and transfer cross sections may be calculated [5] by use of the Klein-Nishina approximation for unpolarized photons scattered from free electrons. The Klein-Nishina approximation is:

$$d\sigma = \frac{\chi^2}{2} \left(\frac{\nu'}{\nu}\right) \left(\frac{\nu}{\nu'} + \frac{\nu'}{\nu} - \sin^2\theta\right) d\Omega \qquad (16)$$

In Equation (15), χ is $e/m_o c^2$; $d\Omega$ equals $2\pi\sin\theta d\theta$; $d\sigma$ is the differential collision cross section; e is the charge of the electron; m_o is the rest mass of the electron; c is the velocity of light; ν is the incident photon frequency; ν' is the scattered photon frequency; and θ is the angle between incident photon direction and scattered photon direction.

Using the laws of conservation of energy and momentum, it may be shown that:

$$\frac{\nu'}{\nu} = 1/[1 + \beta(1-\cos\theta)] \qquad (17)$$

where β equals $h/m_o c^2$ and h is Planck's constant.

The use of Equations (16) and (17) and a given weighting flux, $\Psi(\nu)$, yields group averaged collision cross sections:

$$\sigma(\bar{\nu}) = \int_{\nu_2}^{\nu_1} \int_{o}^{\pi} \Psi(\nu)\sigma(\beta,\Theta)\ \sin\Theta d\Theta d\nu \Big/ \int_{\nu_2}^{\nu_1} \Psi(\nu)d\nu \qquad (18)$$

where $\sigma(\bar{\nu})$ is the group averaged collision cross section with frequency group limits of ν_1 and ν_2.

The differential scattering cross section for energy transfer, $d\sigma_s$, is:

$$d\sigma_s = (\nu'/\nu)d\sigma \qquad (19)$$

Substituation of Equation (15) into Equation (17) and eliminating Θ by use of Equation (16) gives:

$$d\sigma_s = \frac{\pi\chi^2\nu'}{\beta\nu^2}\left[\frac{\nu'}{\nu} + \frac{\nu}{\nu'} + \frac{2}{\beta}\left(\frac{\nu}{\nu'} - 1\right) + \frac{1}{\beta^2}\left(\frac{\nu}{\nu'} - 1\right)^2\right]d\nu' \qquad (20)$$

Group averaged transfer coefficients may be expressed in terms of Legendre polynomials by use of equation (18) and a given weighting flux $\Psi(\nu)$.

$$\sigma_{s\ell}^{\ell \to g} \pi\chi^2 \int_{\nu_1}^{\nu_2} \Psi(\nu) \int_{\nu_1}^{\nu_2} \frac{\nu'}{\beta\nu^2}\left[\frac{\nu'}{\nu} + \frac{\nu}{\nu'} - \frac{2}{\beta}\left(\frac{\nu}{\nu'} - 1\right) +$$

$$\frac{1}{\beta^2}\left(\frac{\nu}{\nu'} - 1\right)^2\right]\Gamma(\nu'\nu)\ P_\ell(\alpha)\ d\nu' d\nu \Big/ \int_{\nu_1}^{\nu_2} \Psi(\nu)d\nu$$

where $\Gamma = \begin{bmatrix} 1, & \frac{\nu}{1+2\beta} \le \nu' \le \nu \\ 0, & \text{otherwise} \end{bmatrix}$ and (21)

$P_\ell(\alpha)$ is the Legendre polynomial with the argument α equal to $(1 + \frac{1}{\beta} - \frac{\nu}{\beta\nu'})$. This equation expresses the energy transfer coefficients from group ℓ with limits ν_1 and ν_2 into group g with limits ν_1' and ν_2'.

A slight modification of σ_{so} is necessary since any photon with an energy above 1.02 MeV can produce two .51 MeV photons in a pair production event. Assuming that the two .51 MeV photons are produced at the same time and place where the pair production event occurs, then the modified cross section is:

$$\sigma_{so}(E,E')_{pp\ modified} = \sigma_{so}(E,E') + \sigma_{pp}\ 2\delta(E - .51\ MeV) \quad (22)$$

With the source term, total cross section and energy transfer coefficients now defined, Equation (12) may be used to obtain the gamma fluxes within the blanket.

2.4 Heat Production

The neutrons produced in the fusion reaction carry an amount of 80 % of the total energy released in the reaction. The majority of this kinetic energy will be deposited in the blanket in the form of heat. The remaining kinetic energy will be lost by neutron leakage from the blanket. It will be converted to heat in the subsequent regions, e.g. the shield or the magnets.

There are two ways on which that part of energy deposited in the blanket will be converted to heat. The first is a local deposition through scattering reactions, atom recoil, and charged particle production. The second is via the process of gamma production, -transport, and -absorption.

It is interesting to note that some neutron reactions are exothermic, thus releasing more energy than is carried into the reaction by the neutrons. An example of this is the $^6Li(n,\alpha)T$ reaction. The net results of the

exothermic reactions are that more heat is generated in the blanket than the total kinetic energy carried by the neutrons into the blanket. Lee [6] has shown that the blanket can be designed to capitalize on such reactions, but there is an unfortunate decrease in the tritium breeding ratio. Borgwaldt [3] calculated for each 14 MeV neutron that an average of 17 to 20 MeV was deposited.

To determine the amount of heat deposited in the blanket from a single charged particle reaction, each type of reaction must be analyzed independetly. For reactions which produce no gamma rays, the energy deposited may be calculated assuming that it is equal to the kinetic energy of the neutron plus the reaction mass difference. For reactions which produce gamma rays, the average energy carried off by the gamma rays must be subtracted as the gamma energy will be deposited elsewhere in the blanket. For reactions involving β decay, the energy carried off by neutrinos must be taken into account.

In neutron capture reactions, heat is deposited in the blanket due to the recoil of the nucleus following the emission of gamma rays. The energy given to the recoil nucleus may be determined from a momentum balance and may be expressed by the following relationship:

$$E_r = \frac{1}{A+1} E + \frac{1}{2} \frac{\left(Q + \frac{AE}{A+1}\right)^2}{M' c^2} \qquad (23)$$

where: E_r is the recoil energy; M' is the mass of the product nucleus; Q is the mass difference given up as photons; E is the energy of the incident neutron; and A is the mass of the primary nucleus.

Neutron scattering reactions also result in a transfer of energy. In this process the particle leaving the reaction site is identical with the incident particle,

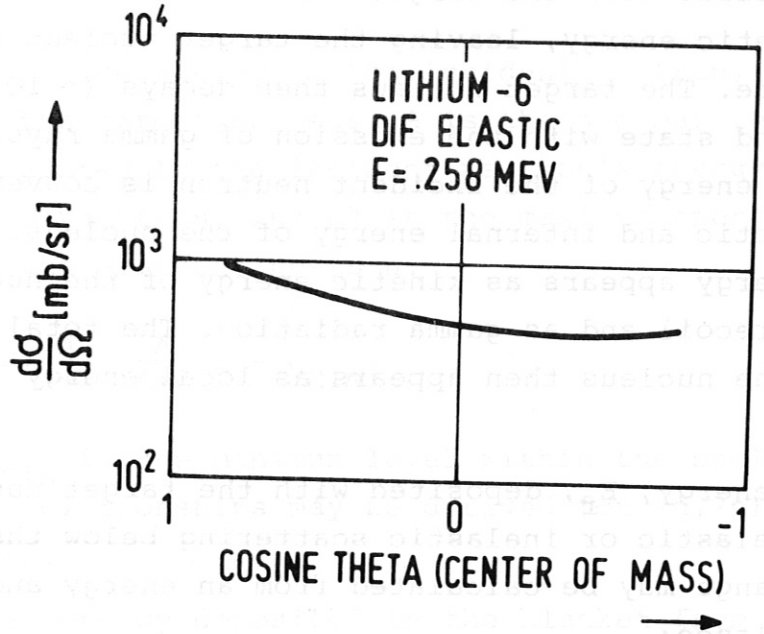
the net result being that some or all of the kinetic energy of the incident particle is deposited with the target nucleus. There are two types of scattering processes, elastic and inelastic.

In elastic scattering both momentum and kinetic energy are conserved, and there are again two types, resonance and potential scattering. In resonance scattering the energy of the incident neutron is such that a compound nucleus is formed close to one of its quantum states. In potential scattering, which occurs with neutrons having energy on either side of a resonance, the neutron is diffracted by the nuclear potential.

In both potential and resonance scattering, the net results are simply a transfer of kinetic energy from the neutron to the target nucleus which shows up as heat in the blanket. These processes may be regarded as "billiard ball" collisions and treated according to the laws of classical mechanics.

It can be shown that the kinetic energy transferred from a neutron to a stationary target is a function of the angle through which the neutron is scattered. For a given angle the fraction of the neutron energy transferred decreases as the mass of the target nuclides increases. For this reason the angular dependence of a scattered neutron is important not only in the transport calculations, but also in the calculation of the heat deposition of these neutrons.

At high energies the scatter is strongly peaked in the forward direction in the center of mass system. Below a few MeV, the scattering becomes almost isotropic, still with a preference towards the forward direction. The differential cross sections shown in Figure 4 illustrate this point. The scattering of a 14 MeV neutron by ^{6}Li is strongly peaked in the forward direction. However, the differential scattering cross section for .258 MeV neutrons in ^{6}Li has flattened considerably.



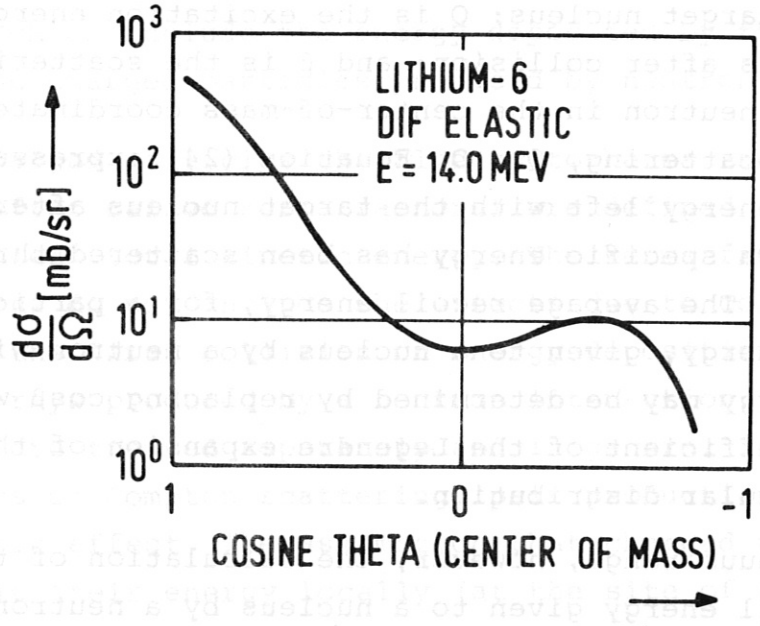


Fig. 4: Elastic Cross Sections

In inelastic scattering the incident neutron forms a compound nucleus with the target and is then expelled at a lower kinetic energy, leaving the target nucleus in an excited state. The target nucleus then decays ($\sim 10^{-15}$ sec) to its ground state with the emission of gamma rays. Thus part of the energy of the incident neutron is converted to both kinetic and internal energy of the nucleus. The internal energy appears as kinetic energy of the nucleus from gamma recoil and as gamma radiation. The total kinetic energy of the nucleus then appears as local energy deposition.

The recoil energy, E_r, deposited with the target nucleus for either elastic or inelastic scattering below the continuum range may be calculated from an energy and momentum balance:

$$E_r = \frac{2AE}{A+1}\left[1 - \frac{(A+1)Q}{2AE} - \sqrt{1 - \frac{(A+1)Q}{AE}}\cos\beta\right] \qquad (24)$$

where E is the energy of the incident neutron; A is the mass of the target nucleus; Q is the excitation energy of the nucleus after collision; and β is the scattering angle of the neutron in the center-of-mass coordinates. For elastic scattering, Q = 0. Equation (24) expresses the kinetic energy left with the target nucleus after a neutron with a specific energy has been scattered through some angle β. The average recoil energy, for a particular excitation energy, given to a nucleus by a neutron with a specific energy may be determined by replacing $\cos\beta$ with the first coefficient of the Legendre expansion of the neutron's angular distribution.

In the continuum range, however, the calculation of the average recoil energy given to a nucleus by a neutron with a specific energy becomes impossible for each excited state. The average recoil energy, E_r, calculated by Ritts [7] , of the nucleus is:

$$E_r = \frac{AE}{(A+1)^2} + \frac{E^{*2}/\Theta + 2E + 2\Theta(1-e^{E^*/\Theta})}{A(E^*/\Theta + 1 - e^{E^*/\Theta})} \quad (25)$$

where E is the energy of the incident neutron; A is the mass of the target nucleus; Θ is the nuclear temperature and is related to the incident neutron's energy and nuclear parameters; and E^* is the maximum energy available to the emergent neutron:

$$E^* = \frac{A}{A+1}\left(\frac{A}{A+1}E - Q_{MIN}\right) \quad (26)$$

where Q_{MIN} is the minimum level within the nucleus. Similar relationships may be derived for (n,2n) and (n,3n) reactions.

The total energy deposited in the blanket from the individual reactions may be determined by the use of fluence-to-kerma factors. Fluence is the time integrated flux amd kerma is an acronym for the kinetic energy released in materials. The kerma factors are placed in two classes, namely, gamma kerma, and local or neutron kerma which include the energy deposited by the neutrons and the charged particles produced by neutron interactions.

For charged particles the fluence-to-kerma factors (hereafter called kerma factors) are defined as the particles' total kinetic energy. The kerma factors for neutrons do not include the effects due to collisions after the first collision or energy deposition due to gamma rays produced by a neutron interaction. Gamma kerma are treated separately and in general include the effects of Compton scattering, pair production and photoelectric effect. The assumption that charged particles deposit their energy locally (at the site of the neutron interaction) is an approximation since some charged particles may travel far from the reaction site before all of their kinetic energy is deposited. However, in

a steady state case away from a boundary, the average energy deposition from all charged particles is such that this approximation may be made with little error.

The microscopic kerma factors may be found by summing over the energy releasing reactions for a particular nuclide

$$KF_i(E) = \sum_j \sigma_{ji}(E)\ E_j^*(E) \qquad (27)$$

where KF(E) is the kerma factor for element i; $\sigma_{ji}(E)$ is the microscopic cross section for reaction j; and $E_j^*(E)$ is the energy released by reaction j from a particle with energy E with element i.

In Figure 5 examples of kerma factors for two different elements are shown. These kerma factors were taken from reference [8] and the reactions included in their calculations are shown in Table II.

In ^{6}Li the exothermic (n,α) reaction makes a large contribution to the kerma factors. In fact, this reaction is one of the major contributors to the heat source in the blanket. It is evident from Figure 5 that there are large variations in the kerma for various elements, simply because of the variety of different reactions and magnitudes of the cross sections involved.

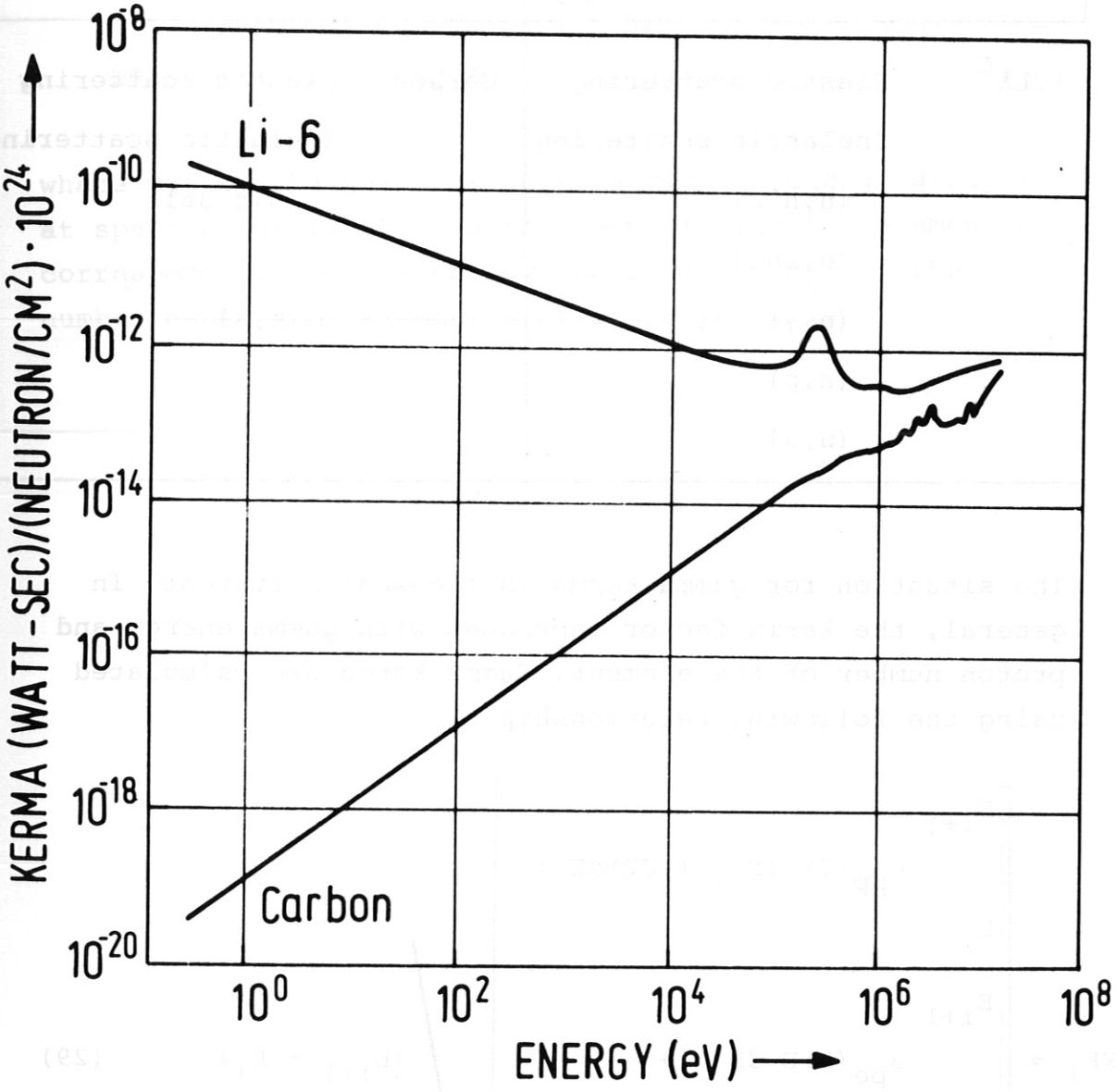


Fig. 5: Neutron Kerma Factors for C and ^{6}Li

Table II Contributions to Neutron Kerma Factors.

Element	Reaction	Element	Reaction
Li^6	Elastic scattering	Carbon	Elastic scattering
	Inelastic scattering		Inelastic scattering
	$(n,n'\alpha)$		$(n,n'3\alpha)$
	$(n,2n\alpha)$		
	(n,γ)		(n,α)
	(n,p)		
	(n,α)		

The situation for gamma kerma is somewhat different. In general, the kerma factor increases with gamma energy and proton number of the element. Gamma kerma are calculated using the following relationship

$$KF_i = \left[\int_{E_i}^{E_{i+1}} \sigma_{pp}(E)\ (E - 1.02)dE + \int_{E_i}^{E_{i+1}} \sigma_{pe}(E)E\ dE + \int_{E_i}^{E_{i+1}} \sigma_{ca}(E)E\ dE \right] \Big/ (E_{i+1} - E_i) \qquad (29)$$

where E_{i+1} and E_i are the energy boundaries of group i; σ_{pp} is the pair production cross section; σ_{pe} the photo effect cross section, and σ_{ca} is the Compton absorption cross section.

The space dependent heating rates, HR_i, in the blanket may now be calculated by use of the fluxes, and the respective group kerma factors from the following relationship:

$$HR_i = \sum_k \sum_\ell \phi_{\ell i} \, KF_{\ell k} \, A_{ki} \qquad (30)$$

where $\phi_{\ell i}$ is the gamma or neutron flux in energy group ℓ at space point i; $KF_{\ell k}$ is the kerma factor for element k corresponding to flux energy group ℓ; and A_{ki} is the number density of element k at space point i.

3. LOGICAL STRUCTURE OF INDRA

INDRA is a program system which calculates the neutronics and photonics characteristics of conceptual fusion reactor blankets. To use the program system, the user normally proceeds in the following manner:

At first the problem to be evaluated has to be defined. This includes the determination of the blanket geometry, its subdivision into regions and intervals, and its material composition expressed by the number densities of each element present in the single regions. An example for this procedure is shown in Figure 6 and Table III, respectively, for the "standard blanket" [9].

Using INDRA, the flow diagram of which is shown in Figure 7, the user has then to decide upon

a) the data libraries to be used
b) the transport code to be used
c) the energy group structure to be used.

3.1 Neutron transport calculation

For the neutron transport calculation two data libraries, DLC-2C and DLC-2D, are available. Both libraries contain 99-grop neutron transport cross-sections for a great number of elements. DLC-2D, which is the later version based on ENDF/B-III,also includes a set of thermal cross-sections. Besides using these libraries there is the possibility to create a new one by applying SUPERTOG-3 to ENDF/B-II or ENDF/B-III data. This should be done if the user does not agree with the weighting spectra applied to produce the existing DLC-2 data or in the case that an element needed is not contained in it.

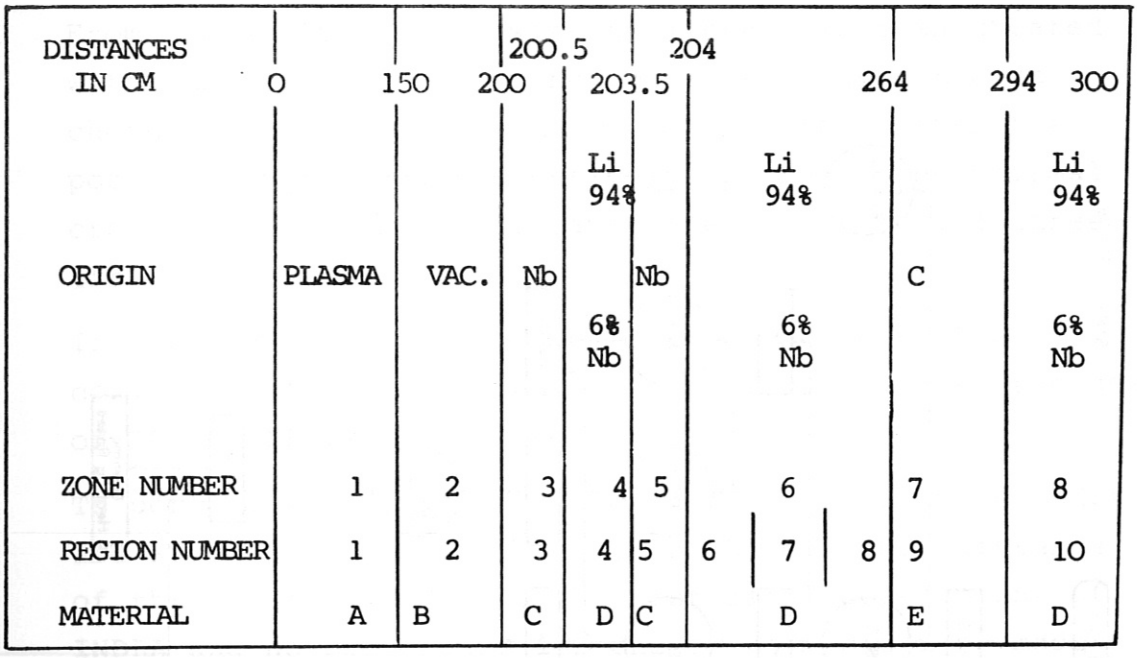


Fig. 6 STANDARD BLANKET GEOMETRY [9]

TABLE III: Atomic Densities for the Standard Blanket [9]

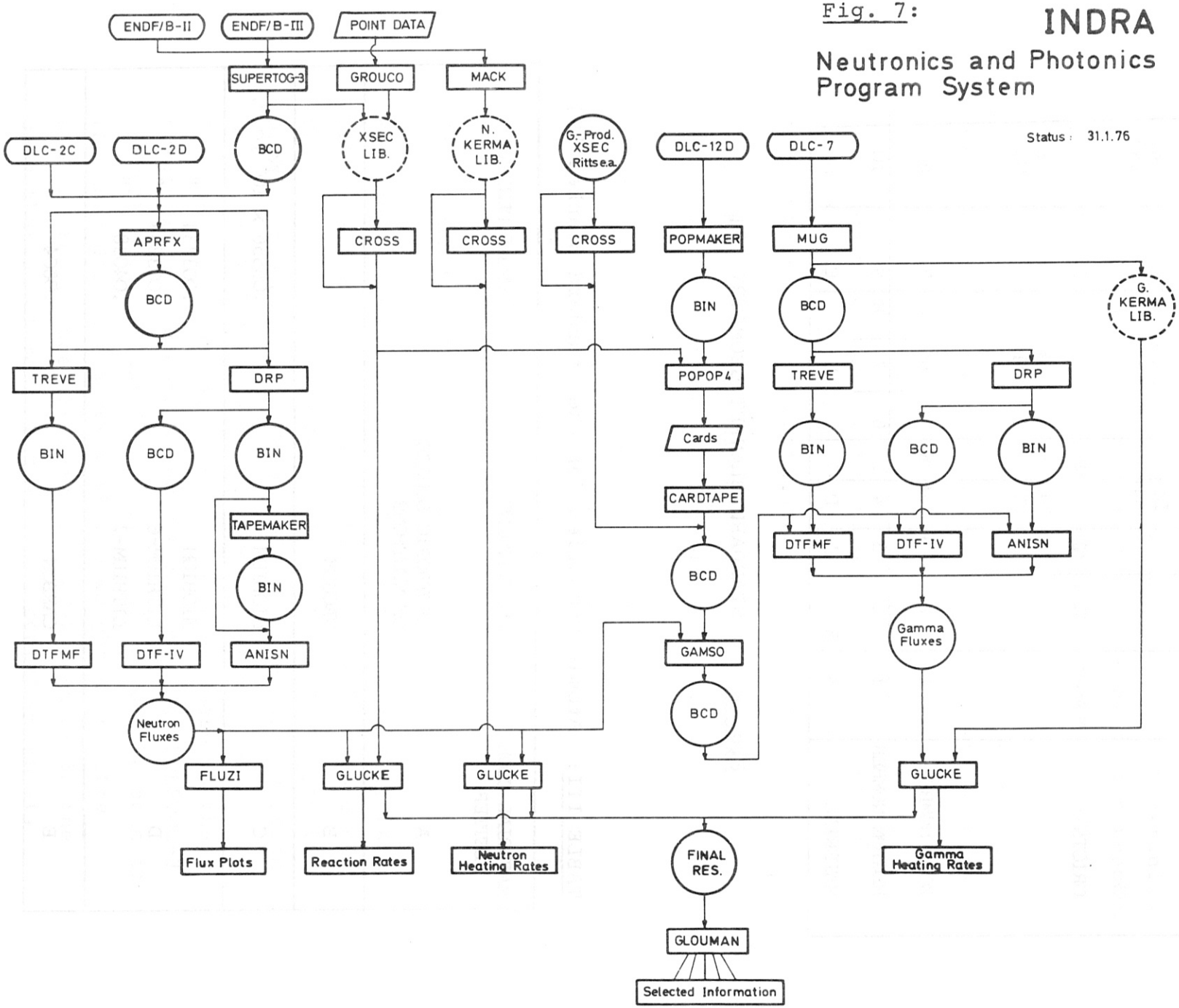
MATERIAL CODE LETTER	CONSTITUENT	NUMBER DENSITY
A	ISOTROPIC SOURCE OF NEUTRONS	
B	VACUUM	
C	NIOBIUM	.05556 x 10^{24}/c.c.
D	NIOBIUM	.003334 "
	LITHIUM-6	.003234 "
	LITHIUM-7	.04038 "
E	CARBON	.0804

Fig. 7:

INDRA

Neutronics and Photonics Program System

Status: 31.1.76



From these libraries a working library is then created with DRP or TREVE depending upon the transport code chosen to use. Both data retrieval programs offer the possibility to add, respectively to change the thermal cross-sections. Both programs have access to all three libraries simultaneously.

If less than 100 energy groups are to be used the collapsing code APRFX has to be applied prior to the use of DRP or TREVE.

To perform the transport calculation itself three codes are available: ANISN, DTF-IV, and DTFMF. The existence of three codes is due to the historical development of INDRA and no preference for one of them shall be made here. It should, however, be mentioned that DTFMF is a modified version of the original DTF-IV code which was especially prepared for fusion reactor blanket application. Its main advantage is the fact that it needs less core storage and is therefore applicable also on small computers. If the user has chosen ANISN as the transport code, he may use TAPEMAKER prior to the ANISN run to produce a group independent tape and so also conserve core storage.

If no partial reaction cross-sections are supplied to the transport code - which is the normal mode in the INDRA system - the result of the neutron transport calculation is the group and space dependent neutron flux distribution.

3.2 Gamma Source Calculation

Since up to now no coupled neutron/gamma cross-sections are available to INDRA,a separate procedure is necessary to produce the gamma source for input to the gamma transport calculation.

The system starts from the DLC-12C library to which the POPOP4-Tapemaker (POPMAKER) is applied for producing a binary library tape. From this library the user has to select the data sets he wants to use. By application of the POPOP4 code he then produces the gamma production cross-sections in form of card decks. In most of the cases the POPOP4 library contains only gamma spectral data. In this case the user has to add to the POPOP4 input neutron cross sections for the relevant elements in the appropriate group structure. These cross sections can be created from ENDF/B by SUPERTOG-3. INDRA provides the special possibility to store such cross sections once they are produced in a cross section library (XSEC-LIB). In general they are created in a 100 group structure to agree with they DLC-2 library. If for further use in POPOP4 or any other program they are needed in a different group structure they can be collapsed using the program CROSS.

The card decks produced by POPOP4 have then to be combined to a working library on tape. This is done by a small program called CARDTAPE. In this step data from other sources, for instance those published by J.J. Ritts et al. [10] may be added.

The working library and the neutron fluxes produced by the neutron transport calculation are finally input to the special program GAMSO which creates the fixed gamma source for input to the gamma transport calculation.

3.3 Gamma transport calculation

A library containing the gamma transport cross sections is created by MUG from the library DLC-7. The remainder of this procedure is equivalent to tne neutron transport calculation. All three transport codes have been provided to receive the fixed source from tape. The result is the gamma flux distribution.

3.4 Evaluation of final results

The results produced by the INDRA system up to this point are the neutron and gamma fluxes. To evaluate the finally interesting results a special program module called GLUCKE was provided which calculates reaction rates or heating rates depending on the input used.

Reaction rates like tritium production rates or helium and hydrogen production rates are calculated using the equivalent partial cross sections contained in XSEC-LIB. To create these cross sections SUPERTOG-3 can be applied to ENDF/B data. Besides this an auxiliary program called GROUCO is available which creates weighted group cross sections from pointwise data.

Neutron and gamma heating rates are also calculated by GLUCKE using neutron and gamma kerma factors. The neutron kerma factors are created by MACK from ENDF/B and stored in a special library (N-KERMA-LIB). If necessary, these kerma factors can also be collapsed to another group structure using CROSS. Gamma kerma factors are extracted from the MUG output and stored in a separate library (G-KERMA-LIB).

All for the blanket performance necessary information like tritium breeding ratio, energy multiplication factor and spatially dependent heating and reaction rates can be extracted from the GLUCKE output. There is, however, the additional possibility to store the most essential information on tape. A further program module called GLOUMAN was prepared which is able to search for special data sets, to draw information from a number of different data sets and to manipulate them in various ways. For example, it is possible to combine neutron and gamma heating results or to produce tables or diagrams for a comparative analysis of different runs.

A last auxiliary program called FLUZI was prepared to evaluate further information from the fluxes created by the transport codes. Its special purpose is to plot flux spectra at different space points or spatially dependent group fluxes. It can also be used to create zone averaged fluxes for use as a weighting function.

4. COMPUTER PROGRAM AND DATA TAPES

When tapes containing data files and codes are obtained from CPL or RSIC, they usually contain sample input and output along with the codes and data. The codes and data files are removed for use. The tapes are kept for future reference.

In the next few pages a listing of the files on some of these tapes are given should the need arise for their use. Note that the codes on these tapes have not been modified for use in the program system. The sample output on these tapes (RECORD LENGTH = 133) contain hollerith information. If the following small code is used to read and print sample output, it will be printed in the same form as if it were output from the code.

```
      DIMENSION C (133)
    3 READ (1,1,END = 2) C
      WRITE (6, 1) C
      GO TO 3
    2 STOP
    1 FORMAT (133A1)
      END
```

To read and punch codes (LRECL = 80) the following program may be used

```
      DIMENSION C (80)
    3 READ (1, 1, END = 2) C
      WRITE (7, 1) C
      WRITE (6, 4) C
      GO TO 3
    2 STOP
    1 FORMAT (80A1)
    4 FORMAT (1X, 80A1)
      END
```

4.1 POPOP4 Data Tape

FILE	RECORD LENGTH	NUMBER OF RECORDS	FILE CONTAINS
1	80	1847	POPOP4 - CODE
2	80	5	POPOP4 - IBM JCL
3	80	117	POPOP4 SAMPLE INPUT DATA
4	80	43	POPOP4 SAMPLE INPUT DATA - SPECTRA
5	133	881	POPOP4 SAMPLE OUTPUT
6	80	1870	POPOP4 LIBRARY TAPE MAKER - CODE
7	80	12	" " " - IBM JCL
8	80	5	" " " -SAMPLE INPUT
9	133	139	" " " -SAMPLE OUTPUT
10	80	429	SAMPLE COUPLING CODE
11	80	7	" " " - SAMPLE INPUT
12	133	57	" " " - SAMPLE OUTPUT
13	133	6621	139 DATA SET LIBRARY - CONTAINS DATA SETS FROM POPLIB
14	133	10822	243 DATA SET LIBRARY - POPLIB-DLC-12C
15	80	1867	POPOP4 LIBRARY TAPE MAKER - CODE
16	80	13	" " " IBM JCL
17	80	5	" " " SAMPLE INPUT DATA
18	80	7	" " " SAMPLE JCL
19	80	247	" " " INPUT FOR 243 DATA SET POPLIB
20	133	139	POPOP4 LIBRARY TAPE MAKER OUTPUT
21	133	2405	" " " "

The POPOP4 code (FILE 1) and sample problem (FILES 2 → 5) are described by W.E. Ford, III and D.H. Wallace [11]. The POPOP4 Library Tape Maker (FILE 6 and FILE 15) is described by the same authors in [11] and [12] . Note that the codes found in FILE 6 and FILE 15 are identical. The data set in FILE 13 is a short version of the data set in FILE 14. This data set is POPLIB and is described in [13] and [14].

The SAMPLE COUPLING CODE (FILE 10) may be used to produce coupled neutron-gamma cross section libraries. It is described in [12], page 142.

4.2 MUG Data Tape

FILE	RECORD LENGTH	NUMBER OF RECORDS	FILE CONTAINS
1	80	9079	ENDF/B PHOTON CROSS SECTIONS
2	80	1048	MUG CODE
3	80	7	MUG IBM JCL
4	80	40	MUG SAMPLE INPUT
5	80	68	MUG SAMPLE INPUT
6	133	247	MUG SAMPLE OUTPUT

The photon cross section library from ENDF/B files for input to MUG is found in FILE 1. This library is described in [15].
The MUG code is found in FILE 2.

4.3 DLC-2D 100 Group Neutron Cross Sections Tape

FILE	RECORD LENGTH	NUMBER OF RECORDS	FILE CONTAINS
1	80	40124	100 GROUP CROSS SECTIONS
2	80	244	DRP (RETRIEVAL PROG.)
3	80	724	APRFX (COLLAPSING CODE)
4	80	7	APRFX JCL
5	80	23	APRFX SAMPLE INPUT
6	133	869	APRFX SAMPLE OUTPUT

File 1 contains the 100 group neutron cross sections for input to transport codes. File 2 contains the data retrieval program DRP, and File 3 contains the cross section collapsing code APFRX. These codes and data sets are described in [16].

5. INDIVIDUAL LIBRARIES AND CODES

5.1 Libraries

Five large data libraries are incorporated in the program system. Each of these data sets is stored on magnetic tape. New versions may be obtained from CPL (Computer Program Library), Ispra,Italy,or RSIC (Radiation Shielding Information Center), Oak Ridge, U.S.A.

A description of the data sets follows: in each case the reader should refer to the referenced documents for a complete description of the data sets.

5.1.1. DLC-2 Neutron Cross Section Data

The DLC-2 Data Library Collection [16, 17] contains fine group neutron cross section data for use in transport calculations. Currently, with the program system, two different versions exist, DLC-2C [17] and DLC-2D [16].

The DLC-2C cross sections were created with SUPERTOG [18] using 1/E weighting from ENDF/B VERSION II data. The DLC-2D cross section set was created with SUPERTOG using fission spectrum weighting joined to a 1/E tail from ENDF/B VERSION III data.

The cross section sets include one dimensional reaction arrays, e.g. absorption, fission, total cross sections, and combined P_N scattering matrices for elastic, inelastic and (n,2n) reactions. The scattering matrices are expanded to the P_8 approximation. They reflect only downscatter and in the case of the DLC-2C library the 100th group is only a sink group containing no data for σ_a, $\nu\sigma_f$, σ_t or $\sigma_s 100 \rightarrow 100$.

The ID numbers for the various elements may be found in Tables IV and V. The title card for each cross section set and P expansion is in the following format:

TABLE IV

Contents of the DLC-2C Library

MAT.No.	ELEMENT	MAT.No.	ELEMENT
1001	H-1	1059	Na-23
1007	Be-9	1060	W-182
1009	B-10	1061	W-183
1013	O-16	1062	W-184
1014	Mg	1063	W-186
1015	Al-27	1083	Re-185
1016	Ti	1084	Re-187
1017	V	1085	Cu-63
1019	Mn-55	1086	Cu-65
1026	Xe-135	1087	Cu
1027	Sm-149	1088	He
1028	Eu-151	1102	U-235
1029	Eu-153	1103	U-238
1030	Gd	1104	Pu-239
1031	Dy-164	1105	Pu-240
1032	Lu-175	1106	Pu-241
1033	Lu-176	1111	Mo
1035	Ta-181	1112	Nb
1037	Au-197	1117	Th-232
1041	U-233	1120	H-2(D)
1043	U-234	1121	Cr
1046	U-236	1122	Fe
1048	Np-237	1123	Ni
1050	Pu-238	1140	C-12
1055	Pu-242	1005	Li-6
1056	Am-241	1006	Li-7
1057	Am-243	1012	N-14
1058	Cm-244		
		1005*	Li-6
		1006*	Li-7
		1012*	N-14

* The last three elements are found on file 2 of this library.

TABLE V

Contents of the DLC-2D Library

MAT.No.	ELEMENT	MAT.No.	ELEMENT
1148	H-1	1026	Xe-135
1120	H-2	1141	Cs-133
1088	He	1027	Sm-149
1146	He-3	1028	Eu-151
1115	Li-6	1029	Eu-153
1116	Li-7	1030	Gd
1154	Be-9	1031	Dy-164
1155	B-10	1032	Lu-175
1160	B-11	1033	Lu-176
1165	C-12	1126	Ta-181
1133	N-14	1127	Ta-182
1134	O-16	1060	W-182
1156	Na-23	1061	W-183
1014	Mg	1062	W-184
1135	Al-27	1063	W-186
1151	Si	1083	Re-185
1149	Cl	1084	Re-187
1150	K	1166	Au-197
1152	Ca	1136	Pb
1017	V	1117	Th-232
1121	Cr	1119	Pa-233
1019	Mn-55	1043	U-234
1180	Fe	1157	U-235
1118	Co-59	1158	U-238
1123	Ni	1050	Pu-238
1087	Cu	1159	Pu-239
1085	Cu-63	1105	Pu-240
1086	Cu-65	1106	Pu-241
1164	Nb	1161	Pu-242
1111	Mo	1056	Am-241
1138	Ag-107	1057	Am-243
1139	Ag-109	1162	Cm-244

9A4, A1, 10A4, A3. In each case the first word contains four dots, i.e., ..., and the 11th word contains the ID number.

Retrieval programs which use the library use the four dots to identify the card as a title card and then the ID number is compared to determine if the data set is the one desired.

The matrices are on the tape in the following manner, each P being a matrix:

$$P_0^1,\ P_1^1,\ P_2^1 \ldots P_8^1,\ P_0^2 \ldots P_8^2 \ldots P_1^n \ldots P_8^n$$

where the superscripts refer to an element and the subscripts to the P expansion.

Each matrix is composed of the following arrays:

$$\begin{array}{ccccccc}
\sigma_a^1 & \sigma_a^2 & \sigma_a^3 & \ldots & \sigma_a^g & \ldots & \sigma_a^{99} \quad 0 \\
\nu\sigma_f^1 & \nu\sigma_f^2 & \nu\sigma_f^3 & \ldots & \nu\sigma_f^g & & \nu\sigma_f^{99} \quad 0 \\
\sigma_T^1 & \sigma_T^2 & \sigma_T^3 & \ldots & \sigma_T^g & & \sigma_T^{99} \quad 0 \\
\sigma_s^{1\to1} & \sigma_s^{2\to2} & \sigma_s^{3\to3} & \ldots & \sigma_s^{g\to g} & & \sigma_s^{99\to99} \quad 0 \\
0 & \sigma_s^{1\to2} & \sigma_s^{2\to3} & \ldots & \sigma_s^{g-1\to g} & & \sigma_s^{98\to99} \quad \sigma_s^{99\to100} \\
0 & 0 & \sigma_s^{1-3} & \ldots & \vdots & & \vdots \quad \vdots \\
0 & 0 & 0 & \ldots & \sigma_s^{\ell\to g}\ (\ell<g) & & \vdots \quad \vdots \\
\vdots & \vdots & \vdots & \vdots & \vdots & & \\
0 & 0 & 0 & 0 & 0 & & \sigma_s^{1\to99} \quad \sigma_s^{2\to100} \\
0 & 0 & 0 & \ldots & 0 & & 0 \quad \sigma_s^{1\to100}
\end{array}$$

Note that when $\ell \leq 0$, the cross sections are zero. The P_1 and higher arrays are arranged in the same manner. Zeros, however, are substituted for σ_a $\nu\sigma_f$ and σ_T.

The arrays are stored on tape by column in card image format. Each column begins on a new card.

5.1.2 DLC-7 Evaluated Photon Interaction Library

The DLC-7 Data Library Collection [15] contains photon interaction cross sections from ENDF/B files. The cross sections have units of barns and cover an energy range from 1 keV to 100 keV. The cross sections included are: total, coherent scattering, incoherent scattering, pair production, and photoelectric for elements with Z from 1-83, 86, 90, 92, 94. These cross sections are used in the calculation of the gamma scattering matrices for use in transport calculations.

The contents of this library is shown in Table VI.

5.1.3 DLC-12 POPLIB - Secondary Gamma Ray Yield and Cross Section Data

The DLC-12 Data Library Collection [13, 14] contains the intensities of secondary gamma rays resulting from neutron-nucleus interactions or the secondary gamma ray production cross sections for neutron-nucleus interactions. The version currently contained in the program system is the DLC-12C Library.

The library contains the 243 data set shown in Table VII for various elements. Some of the reactions for which data appear are radiative capture, fission, inelastic scattering, (n,α) and (n, p). Often several data sets will be given for one particular reaction and in these cases one should refer to the literature for the most appropriate set.

TABLE VI

Contents of the DLC-7 Library

Z	Element	Z	Element	Z	Element	Z	Element
1	H	23	V	45	Rh	67	Ho
2	He	24	Cr	46	Pd	68	Er
3	Li	25	Mn	47	Ag	69	Tm
4	Be	26	Fe	48	Cd	70	Yo
5	B	27	Co	49	In	71	Lu
6	C	28	Ni	50	Sn	72	Hf
7	N	29	Cu	51	Sb	73	Ta
8	O	30	Zn	52	Te	74	W
9	F	31	Ga	53	I	75	Re
10	Ne	32	Ge	54	Xe	76	Os
11	Na	33	As	55	Cs	77	Ir
12	Mg	34	Se	56	Ba	78	Pt
13	Al	35	Br	57	La	79	Au
14	Si	36	Kr	58	Ce	80	Hg
15	P	37	Rb	59	Pr	81	Tl
16	S	38	Sr	60	Nd	82	Pb
17	Cl	39	Y	61	Pm	83	Bi
18	Ar	40	Zr	62	Sm	86	Rn
19	K	41	Nb	63	Eu	90	Th
20	Ca	42	Mo	64	Gd	92	U
21	Sc	43	Tc	65	Tb	94	Pu
22	Ti	44	Ru	66	Dy		

Z is identical with the material identification number.

POPOP4 LIBRARY (DLC-12) 8-4-71

HAVE JUST MADE A NEW POPOP4 BCD LIBRARY TAPE OF THE FOLLOWING 243 DATA SETS

NUMBER	DATA SET NUMBER	IDENTIFICATION		45 INTEGERS							
1	10101	HYDROGEN - (N,G) YIELDS FROM BLIZZARD, P.45 ONE DISCRETE PHOTON	01000101	3	0	1	0	1	2	1	1
2	12101	DEUTERIUM (N,G)TH YIELDS BLIZARD, P. 45	01020101	2	0	7	0	1	2	1	1
3	13101	DEUTERIUM (N,G)TH YIELDS PHYS. REV. 80, P. 918 ONE DISC PHOTON	01030101	3	0	1	0	1	2	1	1
4	30101	LITHIUM (N,G)TH YIELDS NUCLEAR DATA, VOL. 3, P. 386	03000101	3	0	7	0	1	2	1	1
5	30102	LITHIUM (N,G)TH YIELDS BLIZARD, P. 45	03000102	2	0	7	0	1	2	1	1
6	40101	BERYLLIUM (N,G)TH YIELDS FROM NUCLEAR DATA P.386 DISCRETE	04000101	3	0	7	0	1	2	1	1
7	40102	BERYLLIUM (N,G)TH YIELDS UNC-5139 P. 165	04000102	3	0	2	0	1	2	1	1
8	40103	BERYLLIUM (N,G)TH YIELDS BLIZARD, P. 45	04000103	2	0	7	0	1	2	1	1
9	40301	BERRYLLIUM DATA SET VOIDED 6-7-71	04000301	3	1	1	0	1	1	0	1
10	40401	BERRYLLIUM (N,2NG)YIELDS UNC-5140 P. 20	04000401	3	19	1	0	1	2	0	0
11	50102	BORON-10 (N,G)TH YIELDS BLIZARD, P. 45	05000102	2	0	7	0	1	2	1	1
12	50103	BORON (N,G)TH YIELDS ORPHAN, ET AL. GA-10248, P. 57	05000103	2	0	7	0	1	2	1	1
13	50201	BORON 10 (N,A,G,) YIELD FROM BLIZARD, P.46 DISCRETE 0.478 MEV G	05100201	3	0	1	0	1	2	1	0
14	60101	CARBON (N,G)TH YIELDS FROM NUCLEAR DATA, P. 389	06000101	3	0	4	0	1	2	1	1
15	60102	CARBON (N,G)TH YIELDS UNC-5099, P. 243	06000102	3	0	3	0	1	2	1	1
16	60103	CARBON (N,G)TH YIELDS BLIZARD, P. 45	06000103	2	0	7	0	1	2	1	1
17	60301	CARBON NO G-RAYS PER NEUT PROD REACTION UNC-5140 P. 22 G-GROUPS	06000301	1	23	12	0	1	2	0	1
18	60304	CARBON NO G-RAYS PER NEUT PROD REACTION UNC-5140 P. 22 DIS E-G	06000304	3	23	12	0	1	2	0	1
19	70101	NITROGEN (N,G)TH YIELDS UNC-5139 P. 21 MOTZ ET AL. (SEE P. 6)	07000101	1	0	19	0	1	2	1	1
20	70102	NITROGEN (N,G)TH YIELD ORNL-4382, P. 54 (MAERKER)	07000102	3	0	23	0	1	2	1	1
21	70301	NITROGEN NO. G-RAYS PER NEUT. PROD. REACTION UNC-5139 P. 22	07000301	1	33	18	0	1	2	0	1
22	70302	N INEL YIELD RITTS UNC-5139 + NARF-62-17T (USE WITH 07000303)	07000302	1	23	18	0	1	2	0	1
23	70303	N INEL YIELDS RITTS UNC-5139 + NARF-62-17T (USE WITH 07000302)	07000303	3	9	6	0	1	2	0	1
24	74103	NITROGEN-14 (N,G)TH YIELDS BLIZARD, P. 45	07140103	2	0	7	0	1	2	1	1
25	74401	NITROGEN-14 (N,2N)TH YIELD 2 (0.511 MEV) GAMMAS N-14 TO C-13	07140401	3	0	1	0	1	2	1	0
26	74901	N-14 X-SECT FOR INEL, N-P, N-A (2.02-EN-15.89) GA-8006 P.64	07140901	3	10	19	0	1	1	0	0
27	80201	OXYGEN (N,A,G)YIELDS UNC-5140 P.23	08000201	3	18	1	0	1	2	0	0
28	86301	OXYGEN 16 INEL YIELDS STRAKERS DATA	08160301	3	23	8	0	1	2	0	1
29	86302	OXYGEN INELASTIC YIELDS UNC-5139 P 48.	08160302	1	28	20	0	1	2	0	1
30	86601	OXYGEN (N,P,G)TH YIELDS N+O16=P+N16=O16+B+G	08160601	3	0	10	0	1	2	1	1
31	86901	O-16 X-SECT FOR INEL, N-P, N-A (5.43-EN-16.52) GA-8006 P.80	08160901	3	11	11	0	1	1	0	0
32	90101	FLUORINE (N,G)TH YIELDS NUCLEAR DATA VOL. 3, P. 397	09000101	3	0	61	1	1	2	1	1
33	90102	FLUORINE (N,G)TH YIELDS BLIZARD, P. 45	09000102	2	0	7	0	1	2	1	1
34	110101	SODIUM (N,G)TH YIELDS, ORNL-4382, P.124 (17 GAMMA GROUPS)	11000101	2	0	17	0	1	2	1	1
35	110102	SODIUM (N,G)TH YIELD ORNL-4382 (MAERKER) SPECTRUM E-G GT 1 MEV	11000102	3	0	53	0	1	2	1	1
36	110103	SODIUM (N,G)TH YIELDS BLIZARD, P. 45	11000103	2	0	7	0	1	2	1	1
37	113001	SODIUM G-RAY PROD X-SECTS OTHER EVENTS NDL-TR-89-II P. A67	11230001	2	7	27	0	1	1	0	1
38	113101	SODIUM (N,G)TH YIELDS NUCLEAR DATA VOL. 3, P. 402	11230101	3	0	85	0	1	2	1	1
39	113102	SODIUM-23 (N,G)TH YIELDS NUCLEAR DATA VOL. 3 P. 405 ORPHAN ETAL	11230102	3	0	82	0	1	2	1	1
40	113301	SODIUM-23 INEL G-RAY PROD X-SECTS NDL-TR-89-II PP.A45-56	11230301	3	107	29	0	1	1	0	1
41	120101	MAGNESIUM (N,G)TH YIELDS NUCLEAR DATA, VOL. 3, P. 40	12000101	3	0	30	0	1	2	1	1
42	120102	MAGNESIUM (N,G)TH YIELDS BLIZARD, P. 45	12000102	2	0	7	0	1	2	1	1
43	120301	MAGNESIUM INEL X-SECTS NDL-TR-89-III (DRAKE) PP. A 74-89	12000301	3	278	18	0	1	1	0	0
44	130101	ALUMINUM (N,G)TH YIELDS MAERKER TRANS ANS WINTER 68 P. 702	13000101	2	0	14	0	1	2	1	1
45	130102	ALUMINUM (N,G)TH YIELDS UNC-5139 P. 74 (TROUBETZKOY ET. AL.)	13000102	1	0	7	0	1	2	1	1
46	130103	ALUMINUM (N,G)TH YIELDS, ORNL-4382, P.74 (73 DISCRETE GAMMA EN)	13000103	3	0	73	0	1	2	1	1
47	130104	ALUMINUM (N,G)TH YIELD MAERKER RSIC NEWSLETTER 9-69	13000104	2	0	17	0	1	2	1	1
48	130105	ALUMINUM (N,G)TH YIELD HARDELL NUCLEAR PHYSICS A126 P. 397	13000105	3	0	100	0	1	2	1	1
49	130106	ALUMINUM (N,G)TH YIELDS, ORNL-4382, P.76 (14 GAMMA GROUPS)	13000106	2	0	14	0	1	2	1	1
50	130107	ALUMINUM (N,G)TH YIELDS BLIZARD, P. 45	13000107	2	0	7	0	1	2	1	1
51	130301	ALUMINUM NO. G-RAYS PER NEUT-PROD REACTION UNC-5139 P. 75	13000301	1	13	15	0	1	2	0	0

52	130302	ALUMINUM NO. G-RAYS PER NEUT-PROD REACTION UNC-5140 PP. 29-30	13000302	2	12	15	0	1	2	0	0
53	137101	ALUMINUM (N,G)TH YIELDS RASMUSSEN NUCLEAR DATA VOL. 3, P. 416	13270101	3	0	149	0	1	2	1	1
54	137102	ALUMINUM (N,G)TH YIELDS GROSHEV NUCLEAR DATA VOL. 3, P. 415	13270102	3	0	25	0	1	2	1	1
55	140001	SILICON G-RAY PROD X-SECTS CONTINUUM GA-8628 (DRAKE) TABLE A-10	14000001	2	18	35	0	1	1	0	1
56	140101	SILICON (N,G)TH YIELDS MAERKER, SCHMITT, AND NUCLEAR DATA	14000101	2	0	16	0	1	2	1	1
57	140102	SILICON (N,G)TH YIELDS NUCLEAR DATA VOL. 3, P. 420	14000102	3	0	31	0	1	2	1	1
58	140103	SILICON (N,G)TH YIELDS, ORNL-4382, P.151 (17 GAMMA GROUPS)	14000103	2	0	17	0	1	2	1	1
59	140104	SILICON (N,G)TH YIELDS UNC-5139 P. 100 (TROUBETZKOY)	14000104	1	0	12	0	1	2	1	1
60	140105	SILICON (N,G)TH YIELD ORNL-4382 (MAERKER) SPECTRUM E-G GT 1 MEV	14000105	3	0	44	0	1	2	1	1
61	140106	SILICON (N,G)TH YIELDS BLIZARD, P. 45	14000106	2	0	7	0	1	2	1	1
62	140201	SILICON (N,A) G-RAY PROD X-SECTS GA-8628 (DRAKE) TABLE A-8	14000201	3	34	7	0	1	1	0	1
63	140301	SILICON INEL SCATT G-RAY PROD X-SECTS GA-8628 (DRAKE) TABLE A-6	14000301	3	242	35	0	1	1	0	1
64	140302	SILICON INEL YIELDS UNC-5139 P. 101	14000302	1	13	15	0	1	2	0	1
65	140601	SILICON (N,P)G-RAY PROD X-SECTS GA-8628 (DRAKE) TABLE A-7	14000601	3	26	9	0	1	1	0	1
66	150102	PHOSPHORUS (N,G)TH YIELDS BLIZARD, P. 45	15000102	2	0	7	0	1	2	1	1
67	151101	PHOSPHORUS (N,G)TH YIELDS NUCLEAR DATA P. 426	15310101	3	0	138	0	1	2	1	1
68	160101	SULFUR (N,G)TH YIELDS NUCLEAR DATA VOL. 3, P. 430	16000101	3	0	33	0	1	2	1	1
69	160102	SULFUR (N,G)TH YIELDS, ORNL-4382, P.189 (17 GAMMA GROUPS)	16000102	2	0	17	0	1	2	1	1
70	160103	SULFUR (N,G)TH YIELDS, ORNL-4382, P.188 SPECTRUM E-G GT 1.2 MEV	16000103	3	0	47	0	1	2	1	1
71	160104	SULFUR (N,G)TH YIELDS BLIZARD, P. 45	16000104	2	0	7	0	1	2	1	1
72	170001	CHLORINE X-SECTS CONTINUOUS SPECTRUM NDL-TR-89-IV P. A-80	17000001	2	8	25	0	1	1	0	0
73	170101	CHLORINE (N,G)TH YIELDS NUCLEAR DATA VOL. 3 PP. 434-436 GROSHEV	17000101	3	0	65	0	1	2	1	1
74	170102	CHLORINE (N,G)TH YIELD ORNL-4382 (MAERKER) SPECTRUM E-G GT 1.0	17000102	3	0	89	0	1	2	1	1
75	170103	CHLORINE (N,G)TH YIELDS, ORNL-4382, P.132 (17 GAMMA GROUPS)	17000103	2	0	17	0	1	2	1	1
76	170104	CHLORINE (N,G)TH YIELDS BLIZARD, P. 45	17000104	2	0	7	0	1	2	1	1
77	170301	CHLORINE INEL G-RAY PROD X-SECTS NDL-TR-89-IV (DRAKE)	17000301	3	36	19	0	1	1	0	0
78	190001	POTASSIUM G-RAY PROD X-SECTS OTHER EVENTS NDL-TR-89-V P. 94	19000001	2	8	25	0	1	1	0	1
79	190101	POTASSIUM (N,G)TH YIELDS NUCLEAR DATA VOL. 3	19000101	3	0	188	0	1	2	1	1
80	190102	POTASSIUM (N,G)TH YIELDS, ORNL-4382, P.115 (17 GAMMA PROUPS)	19000102	2	0	17	0	1	2	1	1
81	190103	POTASSIUM (N,G)TH YIELD ORNL-4382 (MAERKER) SPECTRUM E-G GT 1	19000103	3	0	76	0	1	2	1	1
82	190104	POTASSIUM (N,G)TH YIELDS BLIZARD, P. 45	19000104	2	0	7	0	1	2	1	1
83	190301	POTASSIUM INEL G-RAY PROD X-SECTS NDL-TR-89-V PP. A63-64	19000301	3	42	6	0	1	1	0	1
84	190701	POTASSIUM G-RAY PROD X-SECTS CHARGED PART NDL-TR-89-V P. A-65	19000701	3	20	4	0	1	1	0	1
85	200001	CALCIUM G-RAY PROD X-SECTS OTHER EVENTS NDL-TR-89-VI P. A-92	20000001	2	10	27	0	1	1	0	0
86	200101	CALCIUM (N,G)TH YIELDS NUCLEAR DATA VOL. 3, P. 451	20000101	3	0	121	0	1	2	1	1
87	200102	CALCIUM (N,G)TH YIELDS, ORNL-4382, P.107 (17 GAMMA GROUPS)	20000102	2	0	17	0	1	2	1	1
88	200103	CALCIUM (N,G)TH YIELD ORNL-4382 (MAERKER) SPECTRUM E-G GT 1.06	20000103	3	0	52	0	1	2	1	1
89	200104	CALCIUM (N,G)TH YIELDS BLIZARD, P. 45	20000104	2	0	7	0	1	2	1	1
90	200301	CALCIUM G-RAY PROD X-SECTS INEL SCATT NDL-TR-89-VI P. A-73	20000301	3	70	14	0	1	1	0	0
91	200701	CALCIUM CHARGED PARTICLE X-SECTS, DRAKE NDL-TR-89-VI PA78-80	20000701	3	65	3	0	1	1	0	1
92	210101	SCANDIUM (N,G)TH YIELDS BLIZARD, P. 45	21000101	2	0	7	0	1	2	1	1
93	220101	TITANIUM (N,G)TH YIELDS NUCLEAR DATA, VOL. 3, P. 466	22000101	3	0	54	0	1	2	1	1
94	220102	TITANIUM (N,G)TH YIELD ORNL-4382 (MAERKER) SPECTRUM E-G GT 1.24	22000102	3	0	53	0	1	2	1	1
95	220103	TITANIUM (N,G)TH YIELDS, ORNL-4382, P.91 (17 GAMMA GROUPS)	22000103	2	0	17	0	1	2	1	1
96	220104	TITANIUM (N,G)TH YIELDS BLIZARD, P. 45	22000104	2	0	7	0	1	2	1	1
97	220301	TI INEL,G X-SECTS FOR DISCRETE G'S UNC-5002 USE WITH 22000302	22000301	3	58	4	0	1	1	0	0
98	220302	TI INEL,G X-SECTS FOR CONTINUUM UNC-5002 P.31 USE WITH 22000301	22000302	2	28	13	0	1	1	0	0
99	230101	VANADIUM (N,G)TH YIELDS BLIZARD, P. 45	23000101	2	0	7	0	1	2	1	1
100	240101	CHROMIUM (N,G)TH YIELDS NUCLEAR DATA VOL. 3, P. 480	24000101	3	0	30	0	1	2	1	1
101	240102	CHROMIUM (N,G)TH YIELDS UNC-5099, P. 220	24000102	2	0	7	0	1	2	1	1
102	240103	CHROMIUM (N,G)TH YIELDS BLIZARD, P. 45	24000103	2	0	7	0	1	2	1	1
103	240104	CHROMIUM (N,G)TH YIELDS CRPHAN, ET AL. GA-10248, P. 185 RESONLY	24000104	2	0	40	0	1	2	1	1
104	240301	CHROMIUM INEL YIELDS UNC-5140, P. 32	24000301	1	35	12	0	1	2	0	1
105	250101	MANGANESE (N,G)TH YIELDS BLIZARD, P. 45	25000101	2	0	7	0	1	2	1	1
106	255101	MANGANESE (N,G)TH YIELDS NUCLEAR DATA VOL. 3, P. 490	25550101	3	0	134	0	1	2	1	1
107	260101	IRON (N,G)TH YIELDS NUCLEAR DATA VOL. 3, P. 502	26000101	3	0	76	0	1	2	1	1
108	260102	IRON (N,G)TH YIELDS MAERKER TRANS ANS WINTER 68 P. 702	26000102	2	0	15	0	1	2	1	1
109	260103	IRON (N,G)TH YIELDS MAERKER RSIC NEWS LETTER 3-69	26000103	2	0	17	0	1	2	1	1
110	260104	IRON (N,G)TH YIELDS UNC-5139 P.129 (TROUBETZKOY AND GOLDSTEIN)	26000104	1	0	7	0	1	2	1	1
111	260105	IRON (N,G)TH YIELDS, ORNL-4382, P.67 (80 DISCRETE GAMMA EN)	26000105	3	0	80	0	1	2	1	1
112	260106	IRON (N,G)TH YIELDS, ORNL-4382, P.68 (16 GAMMA GROUPS)	26000106	2	0	16	0	1	2	1	1
113	260108	IRON (N,G)TH YIELDS BLIZARD, P. 45	26000108	2	0	7	0	1	2	1	1
114	260110	IRON(N,G) YIELDS JEW CAL. PRIVATE COMMUNICATION 11/24/70(5)	26000110	2	40	200	0	1	2	0	1
115	260301	IRON NO. G-RAYS PER NEUT PROD REACT UNC-5140 P. 35	26000301	1	53	10	0	1	2	0	1
116	260302	IRON NO. G-RAYS PER NEUT-PROD REACTION UNC-5139 PP. 130-131	26000302	1	52	10	0	1	2	0	0

Table VII (contd.)

117	266301	IRON INEL CROSS SECTIONS CRNL-4249, P. 19	26560301	3	15	36	0	1	1	0	0
118	269101	STAINLESS STEEL (N,G)TH YIELDS, ORNL-4382, P.168	26090101	3	0	161	0	1	2	1	1
119	269102	STAINLESS STEEL (N,G)TH YIELDS, ORNL-4382, P.170 (18 GAMMA GPS)	26090102	2	0	18	0	1	2	1	1
120	270101	COBALT (N,G)TH YIELDS BLIZARD, P. 45	27000101	2	0	7	0	1	2	1	1
121	279101	COBALT (N,G)TH YIELDS NUCLEAR DATA VOL. 3, P. 514	27590101	3	0	152	0	1	2	1	1
122	280101	NICKEL (N,G)TH YIELDS NUCLEAR DATA, VOL. 3, P. 524	28000101	3	0	52	0	1	2	1	1
123	280102	NICKEL NO. G-RAYS EMITTED PER ABSORPTION UNC-5140 P. 37	28000102	1	13	11	0	1	2	0	1
124	280103	NICKEL (N,G)TH YIELD ORNL-4382 (16 GAMMA GROUPS)	28000103	2	0	16	0	1	2	1	1
125	280104	NICKEL (N,G)TH YIELD ORNL-4382 (55 DISCRETE GAMMA ENERGIES)	28000104	3	0	55	0	1	2	1	1
126	280105	NICKEL (N,G)TH YIELDS BLIZARD, P. 45	28000105	2	0	7	0	1	2	1	1
127	280106	NICKEL (N,G)TH YIELDS ORPHAN, ET AL. GA-10248, P. 223 RES ONLY	28000106	2	0	37	0	1	2	1	1
128	280301	NICKEL NO. G-RAY EMITTED PER NEUT PROD REACT UNC-5140, P. 38	28000301	1	11	12	0	1	2	0	0
129	290103	COPPER (N,G)TH YIELDS MAERKER TRANS ANS WINTER 68 P. 702	29000103	2	0	14	0	1	2	1	1
130	290104	COPPER (N,G)TH YIELDS, ORNL-4382, P.81 (80 DISCRETE GAMMA EN)	29000104	3	0	80	0	1	2	1	1
131	290105	COPPER (N,G)TH YIELD MAERKER RSIC NEWSLETTER 9-69	29000105	2	0	17	0	1	2	1	1
132	290106	COPPER (N,G)TH YIELDS, ORNL-4382, P.83 (14 GAMMA GROUPS)	29000106	2	0	14	0	1	2	1	1
133	290107	COPPER (N,G)TH YIELDS RASMUSSEN IN NUCLEAR DATA, VOL. 3, P. 537	29000107	3	0	136	0	1	2	1	1
134	290108	COPPER (N,G)TH YIELDS BLIZARD, P. 45	29000108	2	0	7	0	1	2	1	1
135	300101	ZINC (N,G)TH YIELD ORNL-4382 (MAERKER) SPECTRUM E-G GT 1.06 MEV	30000101	3	0	84	0	1	2	1	1
136	300102	ZINC (N,G)TH YIELDS, ORNL-4382, P.98 (17 GAMMA GROUPS)	30000102	2	0	17	0	1	2	1	1
137	300103	ZINC (N,G)TH YIELDS BLIZARD, P. 45	30000103	2	0	7	0	1	2	1	1
138	310101	GALLIUM (N,G)TH YIELDS NUCLEAR DATA VOL. 3, P. 551	31000101	3	0	22	0	1	2	1	1
139	310102	GALLIUM (N,G)TH YIELDS BLIZARD, P. 45	31000102	2	0	7	0	1	2	1	1
140	330101	ARSENIC (N,G)TH YIELDS BLIZARD, P. 45	33000101	2	0	7	0	1	2	1	1
141	340101	SELENIUM (N,G)TH YIELDS BLIZARD, P. 45	34000101	2	0	7	0	1	2	1	1
142	350101	BROMINE (N,G)TH YIELDS BLIZARD, P. 45	35000101	2	0	7	0	1	2	1	1
143	380101	STRONTIUM (N,G)TH YIELDS BLIZARD, P. 45	38000101	2	0	7	0	1	2	1	1
144	390101	YTTRIUM (N,G) YIELDS BLIZARD, P. 45	39000101	2	0	7	0	1	2	1	1
145	400101	ZIRCONIUM (N,G)TH YIELDS NUCLEAR DATA, VOL. 3, P. 589	40000101	3	0	129	0	1	2	1	1
146	400102	ZIRCONIUM (N,G) YIELDS BLIZARD, P. 45	40000102	2	0	7	0	1	2	1	1
147	400301	ZIRCONIUM NO. G-RAYS PER NEUT PROD REACTION UNC-5140, P. 39	40000301	1	11	6	0	1	2	0	0
148	410101	NIOBIUM (N,G)TH YIELDS BLIZARD, P. 45	41000101	2	0	7	0	1	2	1	1
149	410102	NIOBIUM (N,G)TH YIELDS RASMUSSEN PRIV. COMM. 11/31/70	41000102	2	0	29	0	1	2	1	1
150	410103	NIOBIUM (N,G)TH YIELDS ORPHAN, ET AL. GA-10248, P. 338	41000103	2	0	33	0	1	2	1	1
151	410901	NB TOTAL SGRPXS EN.GT.4 MEV HOWERTON PRIV. COMM. 4-71	41000901	2	7	60	0	1	1	0	1
152	413301	NIOBIUM INEL YIELDS STEPHEN BLOW, PRIV COMM RE D. STEINER 5-71	41930301	2	93	30	0	1	2	0	1
153	414101	NIOBIUM DATA SET VOIDED 5-6-71	41940101	3	0	245	0	1	2	1	1
154	420101	MOLYBDENUM (N,G)TH YIELDS, NUCLEAR DATA, VOL. 3, P. 600	42000101	3	0	262	0	1	2	1	1
155	420102	MOLYBDENUM (N,G) YIELDS BLIZARD, P. 45	42000102	2	0	7	0	1	2	1	1
156	420103	MOLYBDENUM (N,G)TH YIELDS ORPHAN, ET AL. GA-10248, P. 349	42000103	2	0	38	0	1	2	1	1
157	450101	RHODIUM (N,G) YIELDS BLIZARD, P. 45	45000101	2	0	7	0	1	2	1	1
158	470101	SILVER (N,G) YIELDS BLIZARD, P. 45	47000101	2	0	7	0	1	2	1	1
159	480101	CADMIUM (N,G)TH YIELD RASMUSSEN NUCLEAR DATA V.5, P. 30-37	48000101	3	0	182	0	1	2	1	1
160	480102	CADMIUM (N,G)TH YIELDS BLIZARD, P. 45	48000102	2	0	7	0	1	2	1	1
161	490101	INDIUM (N,G) YIELDS BLIZARD, P. 46	49000101	2	0	7	0	1	2	1	1
162	500101	TIN (N,G) YIELDS BLIZARD, P. 46	50000101	2	0	7	0	1	2	1	1
163	500102	TIN(N,G) YIELDS NUCLEAR DATA, VOL.5, P.53 (RASMUSSEN)	50000102	3	0	231	0	1	2	1	1
164	510101	ANTIMONY (N,G) YIELDS BLIZARD, P. 46	51000101	2	0	7	0	1	2	1	1
165	520101	TELLURIUM (N,G) YIELDS BLIZARD, P. 46	52000101	2	0	7	0	1	2	1	1
166	530101	IODINE (N,G) YIELDS BLIZARD, P. 46	53000101	2	0	7	0	1	2	1	1
167	550101	CESIUM (N,G) YIELDS BLIZARD, P. 46	55000101	2	0	7	0	1	2	1	1
168	560101	BARIUM (N,G)TH YIELD MAERKER RSIC NEWSLETTER 9-69	56000101	2	0	17	0	1	2	1	1
169	560102	BARIUM (N,G)TH YIELDS, ORNL-4382, P.178 SPECTRUM FOR E-G GT 1	56000102	3	0	89	0	1	2	1	1
170	560103	BARIUM (N,G)TH YIELDS, ORNL-4382, P.179 (16 GAMMA GROUPS)	56000103	2	0	16	0	1	2	1	1
171	560104	BARIUM (N,G) YIELDS BLIZARD, P. 46	56000104	2	0	7	0	1	2	1	1
172	570101	LANTHANUM (N,G) YIELDS BLIZARD, P. 46	57000101	2	0	7	0	1	2	1	1
173	590101	PRASEODYMIUM (N,G) YIELDS BLIZARD, P. 46	59000101	2	0	7	0	1	2	1	1
174	600101	NEODYMIUM (N,G) YIELDS BLIZARD, P. 46	60000101	2	0	7	0	1	2	1	1
175	620101	SAMARIUM (N,G) YIELDS BLIZARD, P. 46	62000101	2	0	7	0	1	2	1	1
176	620102	SAMARIUM (N,G)TH YEILDS NUCLEAR DATA VOL. 5, PP. 122-128 RAS	62000102	3	0	213	1	1	2	1	1
177	630101	EUROPIUM (N,G)TH YIELDS FROM YOST CALCULATED 10/4/68	63000101	2	0	26	0	1	2	1	1
178	630102	EUROPIUM (N,G) YIELDS BLIZARD, P. 46	63000102	2	0	7	0	1	2	1	1
179	640101	GADOLINUM (N,G)TH YIELDS NUCLEAR DATA VOL. 5, P. 159, RASMUSSEN	64000101	3	0	163	0	1	2	1	1
180	640102	GADOLINIUM (N,G)TH YIELDS BLIZARD, P. 46	64000102	2	0	7	0	1	2	1	1
181	645101	GADOLINIUM (N,G)TH YIELDS NUCLEAR DATA (TO BE PUBLISHED)	64550101	3	0	126	0	1	2	1	1

182	645102	GD-155 (N,G)TH YIELDS NUCLEAR DATA, P. 161, GROSHEV'S DATA	64550102	3	0	71	0	1	2	1	1
183	647101	GADOLINIUM (N,G)TH YIELDS NUCLEAR DATA VOL. 5, P. 167, KNOWLES	64570101	3	0	116	0	1	2	1	1
184	650101	TERBIUM (N,G) YIELDS BLIZARD, P. 46	65000101	2	0	7	0	1	2	1	1
185	660101	DYSPROSIUM (N,G) YIELDS BLIZARD, P. 46	66000101	2	0	7	0	1	2	1	1
186	660102	DYSPROSIUM (N,G)TH YIELDS NUCLEAR DATA VOL. 5, PP. 184-187 RAS	66000102	3	0	169	1	1	2	1	1
187	670101	HOLMIUM (N,G) YIELDS BLIZARD, P. 46	67000101	2	0	7	0	1	2	1	1
188	680101	ERBIUM (N,G)TH YIELDS BLIZARD, P. 46	68000101	2	0	7	0	1	2	1	1
189	690101	THULIUM (N,G)TH YIELDS BLIZARD, P. 46	69000101	2	0	7	0	1	2	1	1
190	720101	HAFNIUM (N,G)TH YIELDS BLIZARD, P. 46	72000101	2	0	7	0	1	2	1	1
191	730101	TANTALUM (N,G)TH YIELDS FROM BLIZARD P. 46	73000101	2	0	5	0	1	2	1	1
192	730103	TANTALUM (N,G)TH YIELDS BLIZARD, P. 46	73000103	2	0	7	0	1	2	1	1
193	731101	TANTALUM-181 (N,G) YIELDS PENNYS CAL (HELMER) 4/69	73810101	2	28	25	0	1	2	0	1
194	731103	TA-181 (N,G)TH YIELDS RASMUSSEN IN NUCLEAR DATA, VOL. 5, P. 323	73810103	3	0	145	0	1	2	1	1
195	731107	TA-181 (N,G) YIELDS 0.0-EN-0.9 MEV, BS 5.0 EV, JEW 12/70	73810107	2	36	142	0	1	2	0	1
196	740103	TUNGSTEN (N,G) YIELDS GA-9121 ORPHAN P. 67	74000103	2	11	13	0	1	2	0	1
197	740104	TUNGSTEN (N,G)TH YIELDS NUCLEAR DATA, VOL. 3, P. 333	74000104	3	0	146	0	1	2	1	1
198	740109	W (N,G) YIELD PENNY PRIV COMM BART-ORPHAN 50KEV BINS 4/30/69	74000109	2	28	149	0	1	2	0	1
199	740111	TUNGSTEN (N,G)TH YIELDS UNC-5099, P. 145	74000111	1	0	12	0	1	2	1	1
200	740112	TUNGSTEN (N,G)TH YIELDS BLIZARD, P. 46	74000112	2	0	7	0	1	2	1	1
201	740114	W N,G YIELD JEW PRIVATE COMMUNICATION 10/8/70	74000114	2	32	165	0	1	2	0	1
202	740301	TUNGSTEN INEL YIELDS UNC-5140, P. 42	74000301	1	14	11	0	1	2	0	0
203	740901	TUNGSTEN TOTAL X-SECTS (3.5=E-N=11.5) CASA-2333, P. 91	74000901	2	5	14	0	1	1	0	0
204	750101	RHENIUM (N,G)TH YIELDS BLIZARD, P. 46	75000101	2	0	7	0	1	2	1	1
205	770101	IRIDIUM (N,G)TH YIELDS BLIZARD, P. 46	77000101	2	0	7	0	1	2	1	1
206	780101	PLATINUM (N,G)TH YIELDS BLIZARD, P. 46	78000101	2	0	7	0	1	2	1	1
207	780102	PLATINUM (N,G)TH YIELDS NUCLEAR DATA, VOL. 5 PP. 365-366 (RAS.)	78000102	3	0	190	0	1	2	1	1
208	790101	GOLD (N,G)TH YIELDS BLIZARD, P. 46	79000101	2	0	7	0	1	2	1	1
209	800101	MERCURY (N,G)TH YIELDS, NUCLEAR DATA (RASMUSSEN) VOL. 5 P. 384	80000101	3	0	115	0	1	2	1	1
210	800102	MERCURY (N,G)TH YIELDS BLIZARD, P. 46	80000102	2	0	7	0	1	2	1	1
211	810101	THALLIUM (N,G)TH YIELDS BLIZARD, P. 46	81000101	2	0	7	0	1	2	1	1
212	820102	LEAD (N,G)TH YIELD RASMUSSEN NUCLEAR DATA VOL. 5, P. 404	82000102	3	0	2	0	1	2	1	1
213	820104	LEAD (N,G)TH YIELDS BLIZARD, P. 46	82000104	2	0	7	0	1	2	1	1
214	820301	LEAD INEL YIELDS UNC-5099, P. 174	82000301	1	36	12	0	1	2	0	0
215	824101	LEAD-204 (N,G)TH YIELDS NUC DATA VOL 5 P. 405	82040101	3	0	40	0	1	2	1	1
216	826901	PB-206 ALL REACT,G XS FOR EN LT 3.2 ANN PHY 12 P. 485 MULTRY=2	82060901	3	12	7	0	1	2	0	0
217	827901	PB-207M X-SECT FOR EG=1.06 BULL AM PHY SOC 7 P. 334 MULTRY=2	82070901	3	12	1	0	1	2	0	0
218	828301	PB-208 INEL,G X-SECTS FOR E(N) LT 4.6 NUC PHY 44 P.256 MULTRY=2	82080301	3	7	3	0	1	2	0	0
219	830101	BISMUTH (N,G)TH YIELDS BLIZARD, P. 46	83000101	2	0	7	0	1	2	1	1
220	830102	BISMUTH (N,G)TH YIELDS NUCLEAR DATA, VOL. 5 P. 409 (MOTZ,JARC.)	83000102	3	0	7	0	1	2	1	1
221	900101	THORIUM (N,G)TH YIELDS BLIZARD, P. 46	90000101	2	0	7	0	1	2	1	1
222	924101	U-234 NUMBER OF GAMMA-RAYS EMITTED/ABSORPTION UNC-5099, P. 74	92340101	2	17	12	0	1	2	0	1
223	925101	U-235 (N,G)NON-FISSION YIELDS UNC-5140, P. 45	92350101	3	11	13	0	1	2	0	1
224	925301	U-235 NO. G-RAYS PER NEUT-PROD REACTION UNC-5140, TABLE 32	92350301	1	80	11	0	1	2	0	0
225	925801	U-235 (N,G)TH FIS YIELDS CLAIBORNE INTR-LAB CORRESP. 1/29/68	92350801	2	0	20	0	1	2	1	1
226	925802	U-235 PROMPT FISSION YIELD PEELLE 4-69 RSIC NEWS LETTER	92350802	2	0	10	0	1	2	1	1
227	925803	U-235 PROMPT FISSION YIELD 925805 .01-.3, 925802 .3-10.5 MEV	92350803	2	0	13	0	1	2	1	1
228	925804	U-235 (N,G)TH DELAYED FIS YIELDS (T GT 1 SEC) CLAIBORNE 1/29/68	92350804	2	0	20	0	1	2	1	1
229	925805	U-235 (N,G)TH PROMPT FIS YIELDS (T LT 1 SEC) CLAIBORNE 1/29/68	92350805	2	0	20	0	1	2	1	1
230	925806	U-235 PROMPT FISSION YIELD, UNC-5099, P. 55	92350806	2	0	12	0	1	2	1	1
231	926101	U-236 CAPTURE + FISSION YIELDS, UNC-5099, P. 86	92360101	2	10	12	0	1	2	0	1
232	928102	U-238 (N,G)TH YIELDS PHYSICAL REVIEW, VOL. 151, NO. 3, P. 1013	92380102	3	0	58	0	1	2	1	1
233	928109	U-238 (N,G)YIELD JW PRIV COMM (EXCIT.EN.COR) BREIT-WIGNER 4/70	92380109	2	38	117	0	1	2	0	1
234	928110	U-238 NO. G-RAYS PER ABS (CAPTURE + FISSION) UNC-5099, P. 28	92380110	1	9	12	0	1	2	0	1
235	928111	U-238 (N,G)TH YIELDS BLIZARD, P. 46	92380111	2	0	7	0	1	2	1	1
236	928112	U-238 (N,G)YIELD JW PRIV COMM B-W 8/7/70	92380112	2	37	117	0	1	2	0	1
237	928113	U-238 (N,G) YIELDS RESONANT NEUT CAPT, GA-10186 (GAPS FILLED)	92380113	2	21	16	0	1	2	0	1
238	928301	U-238 NO. G-RAYS EMITT PER NEUT PROD REACT UNC-5140, P. 54	92380301	1	79	11	0	1	2	0	0
239	928902	U-238 TOTAL SGRPXS EN=3.67-14.9 MEV HOWERTON PRIV COMM 4-71	92380902	2	7	55	0	1	1	0	1
240	930101	NEPTUNIUM (N,G)TH YIELDS BLIZARD, P. 46	93000101	2	0	7	0	1	2	1	1
241	937101	NEPTUMIUM (N,G)TH YIELD NUCLEAR DATA VOL. 5, P.421	93370101	3	0	6	0	1	2	1	1
242	940101	PU-240 NO. OF G-RAYS EMITTED PER ABSORPTION UNC-5099, P. 123	94400101	2	10	12	0	1	2	0	1
243	94C102	PLUTONIUM (N,G)TH YIELDS BLIZARD, P. 46	94000102	2	0	7	0	1	2	1	1

The library, however, is not complete for all elements of interest, and its application in some cases to fusion reactor blankets is sometimes questionable, as most of the data is for fission reactors.

5.1.4 ENDF/B Evaluated Nuclear Data File

The ENDF/B library [19] is a collection of documented data evaluations. The data includes partial and total point wise neutron and gamma cross sections.

The data is in a format that may be read by SUPERTOG [18] to produce neutron group cross sections in the necessary energy structure for use in transport calculations. It may also be read by MACK [20] to produce group kerma factors for use in GLUCKE.

The current version included in INDRA is ENDF/B-III. However, also ENDF/B-II is still available to the program system. The contents of ENDF/B-III is summarized in Table VIII.

TABLE VIII

Contents of the ENDF/B-III Library

Tape No.	Material	Mat. No.	No. of Records
301	He	1088	620
	Li-6	1115	749
	Li-7	1116	720
	H-2	1120	345
	N-14	1133	7789
	O-16	1134	4702
	He-3	1146	400
	H-1	1148	444
	Be-9	1154	779
	B-10	1155	529
	B-11	1160	924
	C-12	1165	2657
302	Mg	1014	653
	Al-27	1135	5246
	Cl	1149	4947
	Si	1151	6139
	Na-23	1156	3242
303	V	1017	891
	Mn-55	1019	2223
	Cu-63	1085	961
	Cu-65	1086	931
	Cu	1087	1081
	Co-59	1118	1989
	K	1150	4225
	Ca	1152	4724
304	Cr	1121	5105
	Ni	1123	5897
	Fe	1180	10133
305	Gd	1030	688
	W-182	1060	973
	W-183	1061	966
	W-184	1062	788

TABLE VIII (continued)

Tape No.	Material	Mat. No.	No. of Records
305	W-186	1063	729
	Re-185	1083	1360
	Re-187	1084	1383
	Mo	1111	867
	Ta-181	1126	811
	Ta-182	1127	748
	Pb	1136	3572
	Nb	1164	1078
	Au-197	1166	1473
306	Xe-135	1026	409
	Sm-149	1027	1197
	Eu-151	1028	1122
	Eu-152	1031	1180
	Dy-164	1031	1279
	Lu-175	1032	1080
	Lu-176	1033	1138
	U-233 (RSFP)	1042	104
	U-235 (RSFP)	1045	104
	Pu-239 (RSFP)	1052	104
	U-233 (SSFP)	1066	93
	U-233 (NSFP)	1067	89
	U-235 (SSFP)	1068	93
	U-235 (NSFP)	1069	89
	Pu-239 (SSFP)	1070	93
	Pu-239 (NSFP)	1071	89
	Ag-107	1138	848
	Ag-109	1139	790
	Cs-133	1141	887
307	Kr-83	1201	321
	Zr-95	1202	430
	NB-95	1203	429
	Mo-95	1204	321
	Mo-97	1205	313
	Mo-98	1206	274

TABLE VIII (continued)

Tape No.	Material	Mat. No	No. of Records
307	Mo-99	1207	391
	Mo-100	1208	271
	Ru-101	1210	322
	Ru-102	1211	422
	Ru-103	1212	308
	Ru-104	1213	421
	Ru-105	1214	429
	Ru-106	1215	423
	Rh-105	1217	422
	Pd-105	1218	335
	Pd-106	1219	313
	Pd-107	1220	448
	Pd-109	1221	395
	Cd-113	1223	326
	I-131	1224	432
	I-135	1225	429
	Xe-131	1226	325
	Xe-133	1227	319
	Cs-135	1229	351
	Cs-137	1230	385
	La-139	1231	320
	Ce-141	1232	397
	Pr-141	1233	279
	Pr-143	1234	401
	Nd-143	1235	286
	Nd-145	1236	300
	Nd-147	1237	402
	Pm-147	1238	297
	Pm-149	1240	394
	Pm-151	1241	407
	Sm-147	1242	291
	Sm-148	1243	387
	Sm-150	1244	393
	Sm-151	1245	311
	Sm-152	1246	370
	Sm-153	1247	439

TABLE VIII (continued)

Tape No.	Material	Mat. No.	No. of Records
307	Eu-154	1248	426
	Eu-155	1249	356
	Eu-156	1250	417
	Eu-157	1251	410
	Gd-155	1252	331
	Gd-157	1253	327
	Pm-148	1254	370
308	U-234	1043	873
	Pu-238	1050	934
	Am-241	1056	935
	Am-243	1057	524
	Pu-240	1105	1493
	Pu-241	1106	1852
	U-233	1110	2366
	Th-232	1117	1249
	Pa-233	1119	954
	U-235	1157	2995
	U-238	1158	2290
	Pu-239	1159	3069
	Pu-242	1161	958
	Cm-244	1162	1040

5.2 Input Description and Remarks for the Individual Documented Codes

This chapter is concerned with the input description of the individual codes which have already been documented elsewhere. Since this report should mainly be considered as a user manual frequently the input descriptions are directly taken from the equivalent publications. Some remarks and comments are included to facilitate their use.

5.2.1 FIDO Input Routine

The FIDO input routine is used by several programs in the program system for example, ANISN and POPOP4. In addition several of the new programs use the FIDO format.

The FIDO input routine was originally described in the user's manual for ANISN. The following instructions are reproduced from [11].

All data with the exception of the title cards are read using the same format. Each numerical data card is divided into six 12-digit fields which are in turn divided into three subfields as illustrated in the following figure. Only one data field is shown.

Column	1	2	3	4	5	6	7	8	9	10	11	12
	1st		2nd				3rd					Subfield

The first field is a two-digit integer; the second field contains either a $, *, R, I, T, S, F, A, E, Q, N, M, U, V, +, -, or a blank. The third field contains either a fixed or floating point number. The contents of the first two fields define the operation to be performed on the third field.

Blank fields are ignored. One can use any or all data fields on a card. For example, a group of blank cards sandwiched anywhere in a data array would be completely ignored.

Each data array is identified by a one-digit integer in the first subfield (right adjusted). There are both fixed and floating point arrays; a fixed point array is designated by a $ in the second subfield, a floating point array by an *.

The second subfield contains an operator which specifies the type of operation to be performed on the data. The possible operators are as follows:

Array Operators

$ indicates the beginning of an integer array. The first subfield contains a one- or two-digit number identifying the array.

* indicates the beginning of a floating point array. The first subfield identifies the array.

R indicates that the data in the third subfield are to be repeated the number of times specified in the first subfield.

I indicates linear interpolation between the data in the third subfield and the data in the following third subfield. The number in the first subfield gives the number of points to be placed equally spaced in the specified range.

T indicates termination of data reading for a block. A block can contain any number of arrays. Any data on a card after a T will be ignored.

S indicates skip. The first subfield defines the number of entries to be skipped. The third field can contain the first entry following the skips. A blank third subfield would be ignored.

F is used when it is desired to fill the remainder of a data array with the item given in the third subfield.

A is used to address a particular location in a data array. This location is specified in the third subfield; the first subfield is blank.

E may be used to end specifying data for an array. This option is particularly useful when it is desired to replace only some items in a particular array. The items in question are replaced, and the E prevents having to count and skip to the end of the array.

Q is used to repeat a sequence of numbers. The length of the sequence is defined in the third subfield.

N is used to repeat an inverted sequence of numbers. The length of the sequence is defined in the third subfield.

M is used to negate and repeat an inverted sequence. The length of the sequence is given in the third subfield.

U is used to replace the FIDO input format for an array. The array number is given in the first subfield. The format, written in normal FORTRAN, is specified on the card immediately following the card containing a U. The parentheses normally surrounding a format should be included.

V specifies that the array identified in the first subfield will be read according to the last variable format read in.

+ or - indicates exponentiation. The data in the third field are multiplied by $10^{\pm N}$ where N is an integer in the first subfield. This option allows one to specify a number in up to nine significant digits.
Integer data in the third subfield must be right adjusted.

Floating point data may be written with or without an exponent. If the decimal is omitted, it is assumed to be immediately to the left of the exponent field. If there is no exponent, the decimal point is assumed to be to the extreme right of the nine-column subfield.

Input Restrictions

The following restrictions must be observed when using the FIDO input format:

(1) Blank data fields are ignored.

(2) If the interpolation (I) is used, the next data field may not be either blank or an A entry.

(3) The third subfield of a data field containing a $ for an * may contain an integer, N. The next data entry is assumed to be the (N+1)th member of the array. Normally the third subfield is blank and is ignored.

(4) All data arrays must be filled with the correct number of entries. A data array is ended either by starting a new data array or by ending a data block.

5.2.2 SUPERTOG

SUPERTOG is a program which generates fine group parameters and scattering matrices from ENDF/B files. The output cross sections may be read by DRP to produce input cross sections for transport calculations. In addition SUPERTOG will produce partial cross sections for input to POPOP4 and GLUCKE.

The code is described in reference [18] from which the input and option description has been reproduced.

Input Description

Card No. 1 (4I5, 4E10.0)

Item	Cols.	Name	Description
1	1-5	INALL	0 = only card number 1 is read. 1 = all input cards are read.
2	6-10	MATNO	ENDF/B tape material number.
3	11-15	LORDER	Order of P-N for elastic scattering.
4	16-20	IREW	0 = no effect. 1 = ENDF/B tape is rewound.
5	21-30	SIGP	Nonresonance isotope potential scattering cross section per absorber atom, i.e., $\text{SIGP} = (\frac{\Sigma_P}{N} - 4\pi R^2)$, where Σ_P is the mixture macroscopic potential cross section, N is the resonance isotope number density, and $4\pi R^2$ is the resonance isotope potential scattering (SIGP = 10^8 for infinite dilution).
6	31-40	AJIN	Input value of J for unresolved resonance calculation.
7	41-50	RFACT	r-factor (for GAM update).
8	51-60	SFACT	s-factor (for GAM update).

Card No. 2 (915)

Item	Cols.	Name	Description
1	1-5	IDTAP	ENDF/B tape identification number.
2	6-10	MODE	1 = ENDF/B tape is binary mode. 2 = ENDF/B tape is BCD mode.
3	11-15	MCODE	1 = multigroup code is GAM-I. 2 = multigroup code is GAM-II.
4	16-20	MAXG	Number of multigroups.
5	21-25	IEU	1 = GAM-I 68-group structure. 2 = GAM-II 99-group structure. 3 = input energy group structure. 4 = input lethargy group structure.
6	26-30	IW	1 = weighting function is 1/E. 2 = weighting function is 1.0. 3 = weighting function is input. 4 = weighting function is 1/E joined to a fission spectrum. 5 = analytic 1/E.
7	31-35	ISPEC	0 = cross section averaging calculation. 1 = spectrum calculation.
8	36-40	IRES	Option for resolved resonance data. 0 = only the low energy resonance data is added to the smooth cross section. 1 = all resolved resonance data is added to the smooth cross section as infinite dilution cross sections. 2 = GAM-II OUTPUT, the $\ell = 0$ resonances are transferred and the $\ell = 1$ resonances are added to the smooth background.

Item	Cols.	Name	Description
9	41-45	IPUN	0 = no punched output. 1 = output in the GAM format. 2 = output in the ANISN format. 3 = output in both the GAM and ANISN formats. 4 = punch 1-D cross sections.

Card No. 3 (515)

Options desired for this SUPERTOG RUN.

1	1-5	LINK1	0 = no 1 = yes	Resonance calculation
2	6-10	LINK2	0 = no 1 = yes	Smooth cross sections
3	11-15	LINK3	0 = no 1 = yes	Elastic scattering matrices
4	16-20	LINK4	0 = no 1 = yes	Inelastic scattering matrix
5	21-23	LINK5	0 = no 1 = yes	(n,2n) scattering matrix

Card(s) No. 4

This is actually a card set and is necessary only if IW=3. The set consists of the desired weighting function as tabulated points plus the interpolation tables defining the interpolation scheme to be used with the tabulated points. The weighting function must be given from low to high in energy. The format of the card set is a standard ENDF/B TAB 1 record.

Card 4.1 (44X,2I11)

Item	Cols.	Name	Description
1	45-55	N1	Number of interpolation ranges.
2	56-66	N2	Number of weighting function points.

Card 4.2 - .... (6I11)

1	1-11	NBT(1)	Last point number in 1st interpolation range.
2	12-22	JNT(2)	Interpolation scheme for 1st range.
3	23-33	NBT(2)	Last point number in 2nd interpolation range.
4	34-44	JNT(2)	Interpolation scheme for 2nd range.
.			
.			
.			
etc.			
2*N1-1		NBT(N1)	Last point number in N1 interpolation range.
2*N1		JNT(N1)	Interpolation scheme for the N1 range.

Card 4.3 - ... (6E11.4)

1	1-11	BLOK3(1)	First energy point ($\leq$ lowest energy in group structure).
2	12-22	BLOK4(1)	Weight at this energy.
.			
.			
.			
etc. using N2/3 cards			
2*N2-1		BLOK3(N2)	Last energy point ($\geq$ highest energy in group structure).
2*N2		BLOK4(N2)	Weight at this energy.

Card(s) No. 5

This is actually a card set and is necessary only if IEU=3 or 4. The set consists of the desired group structure. If IEU=3, the set is the energy breakpoints of the structure given from low to high in energy. If IEU=4, the set is the lethargy breakpoints of the structure given from high to low in lethargy.

Card 5.1 (6E11.4)

Item	Cols.	Name	Description
1	1-11	XX(1)	Group breakpoint number 1.
2	12-22	XX(2)	Group breakpoint number 2.
.			
.			
.			
etc. using (MAXG+1)/6 cards			
.			
.			
.			
MAXG		XX(MAXG)	Group breakpoint number MAXG.
MAXG1		XX(MAXG+1)	Group breakpoint number MAXG+1.

NOTE: If IEU=3, XX is denoted as EGRP; if IEU=4, XX is denoted as UGRP.

Card 6 (4E12.5)

The purpose of this card is to read in σ_A, $\nu \times \sigma_F$, σ_t and $\sigma_{g\rightarrow g}$ for the thermal group. This card is required only if IPUN (Card 2) = 2 or 3.

1	1-12	SIGA	σ_A
2	13-24	NU-SIGF	$\nu \times \sigma_F$
3	25-36	SIGT	σ_t
4	37-48	SIG-GG	$\sigma_{g\rightarrow g}$

Option Descriptions

Card No. 1:

INALL: This option is designed to facilitate stacked cases where several materials are to be processed in the same way. Complete input is necessary only with the first case (INALL=1) and subsequent cases need only the first card (INALL=0).

IREW: This is to provide running efficiency by a single pass over the ENDF/B tape during a stack of cases. The first case should request a tape rewind (IREW=1) but subsequent cases should not (IREW=0). Materials should be requested in the order they appear on the ENDF/B tape.

SIGP: For infinite dilution cross section, a large value should be used for SIGP, e.g. 10^8.

AJIN: This variable is used to determine the unresolved data (J-state) which will be transferred to GAM-II. For ANISN type runs a value of 10.0 can be used. This insures that all J-states will be used to calculate contributions to the smooth cross sections. For GAM-II, the input values are a function of the material being processed; a table is given in WCAP-3845-2 (ENDF 133). In most cases the J-state with the smallest value of <D> (mean level spacing) was chosen.

RFACT, SFACT: These variables are used only for GAM-II and can be set = 0.0 for ANISN type runs. For GAM-II we use

RFACT = 0.0

SFACT = 5.0

Card No. 2:

MODE: The ENDF/B tape may be either in the binary or BCD mode. For completeness and running efficiency, it is recommended that the binary mode be used where possible.

MCODE: Use MCODE = 2 unless GAM-I decks are being produced.

MAXG: Limit of 150 in present version but can be easily increased to 300.

IEU: This option permits the standard GAM-I or GAM-II structures to be internally generated or allows the structure to be input in either energy or lethargy units.

IW: This option permits the weighting functions to be 1/E, 1.0, input or a combination of 1/E joined to a ^{235}U fission spectrum. Other built-in functions can be easily added in the future.

ISPEC: SUPERTOG will do either a general cross section averaging calculation or a spectrum calculation. Both cannot be done simultaneously. This option selects the calculation type. If both are desired, two cases must be run (and the tape rewound before the second case).

IRES: = 1 for ANISN
= 2 for GAM-II

Card No. 3:

A cross section averaging calculation consists of five segments:

LINK1 -- resolved and unreselved resonance treatment.
LINK2 -- smooth cross sections.
LINK3 -- elastic scattering matrices.
LINK4 -- inelastic scattering matrix.
LINK5 -- (n,2n) matrix.

Any or all of the above are available on option.

5.2.3 APRFX-I

This section describes the neutron cross section collapsing code APRFX-I, published by P.S. Richard [20]. Contributions to this code were made by the Nuclear Effects Laboratory, Ballistic Research Laboratories, Aberdeen Proving Ground, Maryland.

The code, APRFX, has been modified, in order that the data retrieval program, DRP, could read the output tape and thus create a binary tape for use with ANISN or place the cross sections in the proper format for use with DTF-IV.

This report is a repeat of sections from the original report, except in those cases where modifications affected the original. In addition extra information is provided which reflect the new capabilities of the code. The user is urged to read the original report also.

The cross section handling code, APRFX, is used to collapse and combine cross section sets for multigroup transport calculations. The program utilitizes the DLC-2 [16, 17] neutron cross section libraries or output from SUPERTOG [18] if the output is in the 100 group energy structure of the DLC-2 libraries.

Multigroup transport calculations with the 100 group sets generally require either special group independent cross section tapes or large core memory to accomodate the large cross section arrays. This requires a storage of 10300 locations for each element and P expansion. The size of a problem can be significantly reduced if cross section sets can be collapsed prior to execution. The APRFX code is designed to perform group collapsing for as many isotopes and Legendre expansion sets as desired.

APRFX collapses the fine group cross sections to a broad group structure according to a flux spectrum either input by the user or calculated by the code. The code will average the fine group cross sections to form either macroscopic or microscopic isotope cross section.

APRFX may also be used with cross section sets of less than 100 groups, for example, MUG output. The limitations are discussed in the section "Code Modifications".

In its present form the code will not read correctly the ID numbers of mixture cross section sets.

It is important to note that the DLC-2C library contains only a sink group for group 100. The values for σ_a, σ_T, σ_S and $\nu\sigma_f$ for group 100 may be entered in DRP. For use with APRFX, this library is considered to be 100 groups.

The methodology for collapsing is discussed in the original report.

Cross Section Library

The present version of the code utilizes the DLC-2 multigroup constant library which is in the ANISN-DOT format. The cross sections are arranged in the following way:

Position	Cross Section (Group g)
1	σ_a - absorption cross section
2	$\nu\sigma_f$ - (neutrons/fission) x fission cross section
3	σ_T - total cross section
4	$\sigma_{g\rightarrow g}$ - in group scattering cross section
5	$\sigma_{g-1\rightarrow g}$ - downscattering from group g-1 to group g
.	.
.	.
.	.
.	$\sigma_{1\rightarrow g}$ - downscattering from group 1 to group g

The code expects the cross sections to be in this order although as many activity cross sections as desired may be averaged if they are present in positions above σ_a. The remainder of the table for each group is filled with zero entries. The position of σ_T is input to the code and the number of activity cross sections can be adjusted by this position specification. Since upscattering is not included in the DLC-2 libraries it is not utilized in the code. The DLC-2 libraries do include a 100^{th} group (0.0 to 0.414 ev) as a sink group for adjusted thermal constants and the code will calculate the total transfer into this sink group so that only the quantities σ_a, $\nu\sigma_f$, and σ_T are required to complete the last group, which may be input into DRP.

Code Modifications

Possible changes to cross section collapsing codes such as APRFX-I include adaptation to a different cross section library (other than DLC-2B), modification of the

flux calculation, and options to edit or change the current library. Listed below are the basic types of modifications required to effect possible changes.

A. Adaptation of other cross section libraries

If the new library is in the ANISN-DOT format, no changes are needed except to the neutron velocities and energy group structure. This modification is not required to produce the correct averages of cross sections. It is necessary if a correct table of energy groups and neutron velocities are desired. This calculation is in SETUP, and neutron velocities for the fine groups are under the array name VX. This section is identified by comment cards in the SETUP subroutine listing. If the library is not in the ANISN-DOT format, the input routine RDIN must be adapted to arrange the cross section array in this format. RDIN reads in all cross sections under the array name XN2. Any set of instructions in RDIN which will input cross sections in the XN2 array in the order given above will allow calculation of the correctly averaged quantities. If larger libraries (> 100 energy groups) are to be used, the same modifications will be made accompanied by an increased DIMENSION specification to accomodate the larger arrays.

B. Flux Calculation

The flux spectrum may be either read in, or calculated by the subroutine INFLUX. If a more accurate calculation of the flux by the code is desired (P1 etc.) the INFLUX Subroutine may be replaced. The cross sections for the homogeneous mixture are in the XN1 array and the flux is contained in the APHI array. The inscattering and outscattering from each energy group are in the CTO and XN2 (temporary) arrays respectively.

C. Edit Options

If it is desired to perform any other function with the code such as editing, perturbing changing, or deleting cross sections, another subroutine may be added which is called from the main program. This subroutine will first call RDIN then proceed with user instructions to operate on the XN2 array which contains the cross sections. The problem can be either terminated or it may continue to perform the basic code operations with the modified cross section set.

Code Input

The input required for operation of the code consists of two types of data. The first type defines the control parameters and input source and flux data. The second section of input is the cross section data which may be on tape and/or cards. Values in parenthesis refer to the present DLC-2 library of cross sections. Each subsection of input data requires a new card section.

Section IA - Control Parameters

FORMAT (8I5, F10.5)

NFG - number of fine energy groups (100)

NCTL - length of cross section table (103)

IHG - position of total cross section in table (3)

NOBG - number of broad energy groups desired (NOBG $\leq$ 50)

NEL - number of isotopes to be read (1 $\leq$ NEL $\leq$ 20)

NUS - source options

= 0 - input source (100 group)

= M - unit source present in group M

= -1 - no source used

ITP - number of cross sections to be read from input tape ($\leq$ NEL)

ISPC - output options

= 0 - normal printed output only

= 1 - write cross sections on tape in ANISN-DOT format on tape N6

= 2 - option 0 and 1

= 3 - option 0, 1, 2 and also writes scaler fluxes, averaged sources and velocities on tape N6

= N - option 1 and print N groups of each cross section set (N $\geq$ 4)

TKN - buckling factor (B^2) for leakage correction term in flux calculation

Section IB - Group Structure

FORMAT (14I5)

number of fine energy groups in each broad group, NOBG entries. (The last broad group should contain only fine group 100 if using the DLC-2C libraries, since the 100^{th} group of these libraries are incomplete. Only the inscattering to this sink group is calculated.)

Section IC - Isotope Input Information

FORMAT (6X, A4, 4I10. 3F10.5)

NCID - DLC-2 isotope identification number. These are the DLC-2 I.D. numbers (Tables IV and V).

IP - highest order Legendre expansion of isotope cross sections to be averaged.

IAVE - Are isotope cross sections to be flux averaged?
= 0 - no
= 1 - yes

INFM - Is isotope to be utilized in flux calculation?
= 0 - no
= 1 - yes

NMAC - Is isotope included in a macroscopic mixture cross section set. (Each Legendre expansion cross section set is considered to be an additional mixture. Each NMAC should be consecutive to minimize storage of mixture cross section sets. If NMAC $\geq$ 0 the code will produce mixture cross section sets for all Legendre orders specified by IP, i.e. PO-PN)

L = mixture number
= NMAC + IP
= 0 - no
= N - yes - isotope is a component of mixture number N.

DIFM - density factor for isotope used in flux calculation (barn * atoms).

DAVE - density factor for isotope when averaged singly as a microscopic or macroscopic cross section set (barn * atoms) 1.0 for microscopic.

DF - density factor used for mixture cross section sets. The actual density used for the mixture is DF * DAVE, allowing different densities to be used for the same isotope in flux calculations, single isotopes, and mixtures.

Section ID - Source Specification (NUS = 0)

FORMAT (2A6, 5E12.3)

Source identification (U-235, Pu-239, etc. in first 12 columns of first card)

FORMAT (6E12.3)

Remainder of NFG entries for input source

Section IE - Input Flux (if all isotope INFM = 0)

FORMAT (6E12.3)

Input Flux(NFG entries)

Section 2 - Cross Section Data

The program allows the following variations in cross section input:

Card Input: Format - (6E12.3) or ANISN-DOT card image format including reapeats (I2, A1, A1, I5, A1, I2)

Tape Input - The code will search the input tape (N5) for ITP isotopes. If the total number of isotopes to be read is greater than ITP, the remaining isotopes are assumed to be on cards.

Comments:

The operation of the code requires no tapes other than input if fluxes are not calculated by the code and ISPC < 1. If the code calculates the flux spectrum, a

temporary tape (N8) is used to store data. If ISPC $\geq$ 1, cross sections are written on an auxiliary tape (N6). If ISPC = 3, fluxes and sources are also written on an auxiliary tape (N6). If ISPC = 3, fluxes and sources are also written on tape (N6).

N5 is logical unit 1 - Input cross sections:UNIT = TAPE --, DCB = (RECFM = FB, LRECL = 80, BLKSIZE = 3200).

N6 is logical unit 2 - Output cross sections:UNIT = TAPE .., DCB = (RECFM = FB, LRECL = 80, BLKSIZE = 3200).

N8 is logical unit 3 - Scratch disk:UNIT = SYSDA ---,DCB = (RECFM = VBS, BLKSIZE = 3604).

Code Output

The normal printed output includes the following information:

1. Edit of control parameters
2. Source listing (NUS $\geq$ 0)
3. Input flux listing (all INFM = 0)
4. Fine group lethargy and energy structure and neutron velocities
5. Broad group structure in terms of fine groups
6. Calculated fine group fluxes from diffusion solution (if $\sum$ INFM $\neq$ 0)
7. Flux weighted quantities

$$\phi_\ell = \sum_{i \text{ in } \ell} \phi_i \quad - \text{ flux weighting integrals}$$

$$X_\ell = \sum_{i \text{ in } \ell} X_i \quad - \text{ averaged source}$$

$$v_\ell \quad - \text{ averaged velocities}$$

8. Listing of isotopes to be spectrum weighted

9. Broad group averaged cross sections

   a. Macroscopic mixture of flux materials ($\sum$ INFM $\neq$ 0)
   b. Spectrum weighted cross sections for isotope (PO through IP)
   c. Listing of components in mixtures ($\sum$ NMAC $\neq$ 0)
   d. Mixture cross section sets (PO through IP)

10. Error Messages - the code checks for the following errors during operation.

   Error 1 - Flux weighting integral(s) are zero.

   Error 2 - Insufficient storage for all mixture cross section sets. (The number of cross section sets times the number of broad group cross sections ((NOBG X (NOBG + 3)) $\leq$ 10400)

   Error 3 - The final broad group contains more than the last fine group (100). Results in incorrect averages for σ_a, $\nu\sigma_f$, and σ_T.

The broad group cross sections may be obtained in the ANISN-DOT format and written on an auxiliary tape (N6) if ISPC = 2. A heading card precedes each cross section set giving the identification number and Legendre expansion order. For mixtures, the identification number is given by NMAC+N where N is the expansion order. The Legendre expansion order of a set of mixture cross sections can be found by refering to a summary table printed prior to the mixture sets. There is also printed a monitor of the cross sections read from the input tape (N5) and the intermediate storage operations on the temporary tape (N8). The monitor is intended to serve as a check on the tape reading and writing operations.

5.2.4 DRP - Data Retrieval Program

DRP is a cross section retrieval program which produces working libraries for use with DTF-IV or ANISN. The input library tapes are searched until the desired material numbers are found and these cross sections are output in the correct format for input to DTF-IV or ANISN. The library that is created may contain more cross section sets than is needed in a particular transport calculation.

The original code is described in [16, 17]. Most of the information contained in this section is taken from these two references. Modifications to the code are:

1. The code may be used with any group structure.
2. The thermal group may be entered or changed.
3. The control parameters are printed out.
4. The absorption, total, fission or scattering cross section may be printed or punched out for any desired element.

The input libraries are either:

1. SUPERTOG output
2. MUG output
3. DLC-2 neutron libraries
4. APRFX output cross sections (In its current form, however, it will not read mixture cross sections from APRFX).

Input Description

Section A. - Title Card and No. of Groups

FORMAT (I4, 17A4)

NGF, NTITLE(I)

NGP = No. of energy groups
NTITLE = Text describing the cross sections

Section B. - Control Parameters

FORMAT (1X, A4, 6I5, 3E12.5)

MATNO - material no.

LORDER - order of P-N desired

N - logical no. of input cross section tape

IREW - rewind option 1/0
- 1 - rewind
- 0 - do not rewind

MODE - mode option
- 0 - ANISN card image format
- 1 - edit of data
- 2 - write ANISN unformated tape
- 3 - DTF-IV card image format
- 4 - print - σ_a, $\nu\sigma_f$, σ_t or σ_s depending on value of NG
- 5 - print and punch σ_a, $\nu\sigma^f$, σ_t or σ_s depending on value of NG

NG - OPTION - not used if mode $\leq$ 3
- 1 - print and/or punch σ_a
- 2 - print and/or punch $\nu\sigma_f$
- 3 - print and/or punch σ_T
- 4 - print and/or punch σ_s
- enter 0 or blank if mode $\leq$ 3

IX - change thermal cross section; option 0/1
- 0 - option not used
  NOTE: IX may be left as blank and read as zero
- 1 - input SA, VSF, SS

SA,VSF,SS input only if IX = 1
- SA - thermal absorption cross section
- VSF - thermal fission cross section times ν i.e. $\nu\sigma_f$ ν is average number of neutrons per fission
- SS - thermal scattering cross section

REPEAT SECTION B DATA FOR EACH ELEMENT

Comments:

A) MATNO is the ENDF/B identification number for the DLC-2 libraries, SUPERTOG output, and APRFX output. Note, however, that the ID numbers change from version to version of ENDF/B data.

For MUG output, the ID numbers are the atomic numbers of the elements. If the rewind option is not used, the cross sections must be requested in the order on which they are on tape.

B) MODE = 0, output is written either on punched chards or on a magnetic tape in ANISN card image DLC-2 format on logical unit no. 7. For punched cards unit 7 must be defined as system punch. For magnetic tape, unit 7 must be defined with RECFM = FB, LRECL = 80, BLKSIZE = 3200.

MODE = 1, an edit of the cross section sets requested is written on logical unit no. 6 (assumed to be the standard output unit). The edit consists of printing the entire cross section set up through the P expansion requested for each material requested.

MODE = 2, an ANISN binary tape is written on unit 8. This is the mode normally used in preparing ANISN cross section sets. The magnetic tape must be defined on unit 8 with RECFM = VBS.

MODE = 3, output cross sections are written either on punched cards or on a magnetic tape in DTF-IV card image format on logical unit no. 7. For punched cards unit 7 must be defined as system punch. For magnetic tape unit 7 must be defined with RECFM = FB, LRECL = 80, BLKSIZE = 3200.

MODE = 4, an edit of either σ_a, $\nu\sigma_f$, σ_t or σ_s depending on NG for each element requested is written on logical unit no. 6 (logical unit no. 6 is assumed to be the standard output unit).

MODE = 5, same as MODE = 4 except that the cross sections requested are punched on cards on logical unit no. 7. Logical unit no. 7 must be defined as system punch.

C) The logical number, N, must be in the range 1 to 40, except N = 5, 6, 7, or 8 can not be used since they are required by the program. If the logical number is to be the same as for the previous material, it is not necessary to repeat this information on the input card.

The following units are used by DRP:

logical number	function
5	card input
6	standard output
7	punch (or magnetic tape)
8	ANISN unformatted tape
N = 1-4 or 9-40	input cross section tape(s)

Note that as many tapes as desired may be used as input. (Limited by the number of tape drive units available). Thus one output tape may be created from several different input libraries.

D) Note that by using the option to change thermal cross sections (IX = 1), the original thermal cross sections are replaced. If using the DLC-2C library, this option must be used since the thermal group contains only zeros.

E) The MODE = 3 option produces card image data suitable for input to the DTF-IV code. In essence, the DTF-IV format differs from the ANISN format as follows:

- ANISN cross section must have a $(2\ell+1)$ factor included; DTF-IV cross sections must not have the $(2\ell+1)$ factor.
- ANISN card image format allows repeats; DTF-IV card image format does not allow repeats.
- for input to DTF (not DTF-IV) use octal ID numbers from DRP printout.
  DTF is the code from which DTF-IV evolved. It will probably never be used.
- for input to DTF-IV use decimal ID numbers from DRP printout.

5.2.5 TAPEMAKER

This section was taken from the ANISN-documentation [21].

If it is desired to use ANISN for a computation involving several different elements for which a fine group, high order P expansion cross section library is available, it is possible that the resulting input cross section matrix will exceed the storage capacity of the computer.

This problem can be overcome by using a "group independent" cross section tape. When this tape is used, rather than storing the entire matrix, only the cross section data for a single group are stored in the memory while the calculations for that group are performed. Data for the next group are read from the group independent tape (replacing the data stored for the previous group) before calculations for that group are performed.

The Tapemaker routine produces the group independent tape from cross section data on cards and/or on ANISN library (binary) tape.

Input Description

1. LIM1 card - format (6X, I6)

   LIM1 dimension of DUMY in main program. (Tapemaker is most efficient when LIM1 and DUMY dimensions are as large as computer storage allows)

2. Integer parameters - format (12I6)

   IGM number of energy groups

   IHT position of σ_t in cross section table

   IHS position of σ_{gg} in cross section table

   IHM length of cross section table

   MS mixing table length

   MCR number of cross section sets to be read from cards

   MTP number of cross section sets to be read from tape

   MT number of mixtures** (Note change from ANISN definition.)

   ITH 0 - forward solution
   1 - adjoint solution

   IPRT 0 - do not print cross sections
   1 - print mixture cross sections
   2 - print all cross sections

3. Mixing table (ANISN format)

   10$, 11 $, 12* arrays (MS entries, each array)

   ANISN binary library tape ID numbers (ANISN format)

   13$ array with library ID numbers (MTP values, MTP > 0)

   NOTE: The data in this section followed by a T.

4. Cross sections from cards (note change from ANISN procedure)

   14* array followed by a T, one material at a time.

   NOTE: A 14* array followed by a T is required for each of the MCR materials (MCR > 0).

**NOTE: There are MCR + MTP + MT materials. Only the last MT materials (as mixed in the 10$, 11$, 12* mixing tables) are written on Logical 8.

Sample Problem

A sample problem is included to illustrate the use of the IBM 360 Tapemaker routine. A 27 group, P-2 expansion, set of cross section data (on cards) for hydrogen and oxygen are used to generate a group independent type containing a 27 group P-2 expansion, set of cross sections for water. The input data required for the sample problem are shown below.

```
     75000
   27   3     4    30    9    6      0   3      0     2
10$          117.0       9.1        117.0         9.1     117.0
   9.1
11$          3RO.0      4I1.0          6.1
12*          3RO.0      3RO.0668    3RO.0334
  T
14*
(Data for Hydrogen P-0, 810 values)
  T
14*
(Data for Hydrogen P-1, 810 values)
  T
14*
(Data for Hydrogen P-2, 810 values)
  T
14*
(Data for Oxygen P-0, 810 values)
  T
14*
(Data for Oxygen P-1, 810 values)
  T
14*
(Data for Oxygen P-2, 810 values)
  T
```

LOGICAL TAPE NUMBERS:

1 - scratch
2 - scratch
4 - scratch
8 - "group independent" tape
9 - ANISN binary library tape
5 - input
6 - output

The above data is for the 360 version of Tapemaker. For the 7090 version, LIM1 should be 23000 rather than 75000.

5.2.6 Transport Codes

5.2.6.1 General Remarks

Three different transport codes are available for use in the program system. They are: ANISN, DTF-IV and DTFMF.

ANISN is by far the most versatile and fastest code. In general it would be used for the transport problem. DTF-IV might possibly be used for a comparison problem since the flux convergence scheme is different from ANISN.

It is beyond the scope of this report to explain differences of the codes in detail. The user is referred to the literature as far as ANISN and DTF-IV are concerned [21 - 27]. Note that the versions presently incorporated in INDRA are described in [22] in the case of ANISN and [27] in the case of DTF-IV, respectively.

As compared to the original codes, DTFMF (DTF - Modified for Fusion) is a revision of the DTF-IV code which was done by one of the authors of this report. It will solve only a source type problem and is therefore not applicable to problems including fissionable materials. The advantage of this code is that the input data is much simpler than for the original code and that it requires less core storage. Since the modifications affect the essence of the original DTF-IV code very strongly, it will be treated in chapter 5.3 together with the other not yet documented programs.

Each of the codes will solve the gamma or neutron transport problem. In ANISN or DTF-IV, the data retrieval code, DRP, is required to produce the input cross section tape from one of the cross section libraries. The data retrieval program, TREVE (see Section 5.3.1), prepares the cross sections for use in DTFMF.

5.2.6.2 ANISN

ANISN is a discrete ordinates transport code that uses the S_N method to solve the Boltzman equation in plane, spherical, or cylindrical geometry. It may be used to solve the Boltzman equation for either a gamma or neutron source problem.

The code is described in [22] and the user should refer to this manual for a description and capabilities of the code.

Input to the code are group cross sections and parameters which describe the blanket. The cross sections are prepared for input from one of the data libraries by either DRP or TAPEMAKER which also give the cross section ID numbers.

The input data, with one exception, is exactly as defined in the users manual. The input requirements from the manual are repeated here for convenience.

Input Description

A. This section is intended as a quick in preparing problems for ANISN. More detailed information may be found in the manual. ANISN is variably dimensioned. To change the size of the problem, set the value of DUMY in the main program to the value desired. Set LIM1 equal to that value.

B. TITLE CARD

The title card may contain only 48 characters. Cols. 61-72 of the title card are used to enter a maximum execution time (ignored if zero. FORMAT (12A4, 12X, I12)

! See comment No. A !

C. PARAMETERS

15$ Integer parameters [36]

1. ID — problem ID number; if greater than 1,000,000, disadvantage factors will be computed by group for each material which appears in the calculation.
2. ITH — 0 - forward solution; 1 - adjoint solution
3. ISCT — maximum order of scatter found in any zone (P value)
4. ISN — order of angular quadrature (S value)
5. IGE — 1 - slab; 2 - cylinder; 3 - sphere
6. IBL — left boundary condition
   0 - vacuum (no reflection)
   1 - reflection
   2 - periodic
   3 - white/albedo

7. IBR — right boundary condition, same options as IBL
8. IZM — number of zones or regions (Same Material)
9. IM — number of mesh intervals
   ! see comment No. B. !
10. IEVT — eigenvalue type
   0 - fixed source
   1 - k calculation
   2 - α calculation
   3 - concentration search
   4 - zone width search
   5 - outer radius search
   6 - buckling search
11. IGM — number of energy groups
12. IHT — position of σ_{total} in cross section table
13. IHS — position of σ_{gg} (self-scatter in cross section table
14. IHM — length of cross section table
15. MS — cross section mixing table length (10$, 11$, 12*)
16. MCR — number of cross section sets to be read from cards (14*)
17. MTP — number of cross section sets to be read from tape (13$)
18. MT — total number of cross section sets (elements + mixtures)
19. IDFM
   0 - density factors (21*) not used
   1 - density factors used
20. IPVT
   0 - no effect
   1 - enter k_o as PV (16*)
   2 - enter α_o as PV
21. IQM
   0 - no effect
   1 - enter distributed source (17*)
   ! see comment No. A.!
22. IPM
   0 - no effect
   1 - enter shell source by group and angle (18*)
   IM - enter shell source by interval, group, and angle

23. IPP — interval number which contains shell source if IPM = 1; 0 otherwise

24. IIM — inner interation maximum

25. ID1
- 0 - no effect
- 1 - print angular flux
- 2 - punch scalar flux
- 3 - both 1 and 2

26. ID2
- 0 - no effect
- 1 - use specially prepared group independent cross section tape (contains MTP materials)
- 2 - use cross sections and fixed source from previous problem

27. ID3
- 0 - no effect
- N - compute N activities by zone where N is any positive integer

28. ID4
- 0 - no effect
- 1 - compute N activities by interval where N refers to ID3

29. ICM — outer iteration maximum

30. IDAT1
- 0 - all data in croe
- 1 - cross sections and fixed sources stored on tape
- 2 - fluxes and currents on tape also

31. IDAT2 — 0 - no effects

If IDAT2 is greater than zero, the first IDAT2 outer iterations will be executed according to the specifications in the 24$ array. In the 24$ array 0 indicates an S_n calculation, 1 indicates a diffusion calculation and 2 indicates an infinite homogeneous medium calculation. If convergence is not obtained after IDAT2 iterations, the problem continues using the S_n calculation for all groups until convergence is obtained or ICM is reached.

32. IFG
- 0 - no effect;
- 1 - execute cross section weighting;
- 2 - read cross sections weighted in previous ANISN case;
- 3 - read weighted cross sections on tape/ disk/etc. at completion of ANISN case; or
- 4 - both 2 and 3.

33. IFLU
- 0 - step model used when linear extrapolation yields negative flux (mixed mode)
- 1 - use linear model only
- 2 - use step model only
- 3 - weighted difference model
- 4 - weighted model used as negative flux fixup for linear model.

34. IFN
- 0 - enter fission guess (2*)
- 1 - enter flux guess (3*)
- 2 - use fluxes from previous case

35. IPRT
- 0 - print cross sections
- 1 - do not print cross sections

36. IXTR
- 0 - enter zero ; not used

16* Floating point parameters [14]

1. EV — first guess for eigenvalue (usually 0.0)
2. EVM — eigenvalue modifier
3. EPS — epsilon - accuracy desired
4. BF — buckling factor, normally 1.420892
5. DY — cylinder or plane height for buckling correction
6. DZ — plane depth for buckling correction
7. DFM1 — transverse dimension for void streaming correction
8. XNF — normalization factor (usually 1.)
9. PV — 0.0, k_o, or α_e according to IPVT = 0,1, or 2
10. RYF — λ_2 relaxation factor, normally 0.5
11. XLAL — point flux convergence criterion if entered greater than zero

12. XLAH upper limit for $|1.0 - \lambda_1|$ used in linear search
13. EQL eigenvalue change epsilon
14. XNPM new parameter modifier

NOTE: The above data is followed by a T.

D. CROSS SECTIONS (ID2 = 0)

13$ Library ID numbers [MTP] (MTP > 0) (Omit if using cards)

14* Cross sections [MCR x IGM x IHM] (MCR > 0)

NOTE: If entered, the above data is followed by a T.

E. FIXED SOURCE (IEVT = 0 and ID2 < 2)

17* Distributed source [IGM x IM] (IQM = 1)

! see comment No. A. !

18* Shell source [IGM x IPM x MM] (IPM > 0)

NOTE: If entered, the above data is followed by a T.

F. FLUX OR FISSION GUESS (IFN < 2)

2* Fission density [IM] (IFN = 0)

3* Flux guess [IGM x IM] (IFN = 1)

NOTE: If entered, the above data is followed by a T.

G. REMAINDER OF DATA

1* Fission spectrum [IGM]

! see comment no. C. !

4* Radii by interval boundary [IM + 1]

! see comment No. B. !

5* Velocities [IGM]

! see comment No. C. !

6* Angular quadrature weights [MM]*)

! see comment No. D. !

---

*) MM = ISN + 1 for plane or sphere

MM = (ISN x (ISN + 4))/4 for cylinder

7* Angular quadrature cosines [MM]
! see comment No. D. !
8$ Zone numbers by interval [IM]
9$ Material numbers by zone [IZM]
10$ Mixture numbers in mixing table [MS] (MS > 0)
11$ Component numbers in mixing table [MS] (MS > 0)
12* Number densities in mixing table [MS] (MS > 0)
19$ Order of scatter by zone [IZM] (ISCT > 0)
20* Radius modifiers by zone [IZM] (IEVT = 4)
21* Density factors by interval [IM] (IDFM = 1)
22$ Material numbers for activities [ID3] (ID3 > 0)
23$ Cross section table position for activities [ID3] (ID3 > 0)
24$ Calculation markers [IGM] (IDAT2 = 1)
25* Albedo by group - right boundary [IGM] (IBR = 3)
26* Albedo by group - left boundary [IGM] (IBL = 3)
27$ Few group parameters [5] (IFG = 1)

1. ICON — 0 - no effect
   1 - micro cross sections desired
   2 - macro cross sections desired (minus implies cell weighting)
2. IHTF — position of σ_{total} in weighted cross sections
3. IHSF — position of $\sigma_{g \to g}$ in weighted cross sections (minus implies upscatter removal)
4. IHMF — table length of weighted cross sections
5. IPUN — 0 - no effect
   1 - punch weighted cross sections

28$ Few group number for each multigroup [IGM] (IFG = 1)

NOTE: The above data is followed by a T.

Comments:

A. Title Card

The one exception to the input is that the fixed source may be read from tape (output from GAMSO). If this option is used, enter XXXX in the first four columns of the

TITLE CARD, and define logical unit No. 3 as the input tape. Do not enter a 17* or T on unit 5, the standard input unit, since the 17* and T are on tape. If XXXX is not in the first four columns, the program input is exactly as found in the manual [21, 22].

B. Mesh Size

Useful information in running a transport problem , i.e., mesh size, may be found in [23].

C. Fission Spectrum and Velocities

Note that the source type calculations used in fusion blanket transport problems do not require velocities or fission spectra. The input array in these codes may be filled with zeros for fission spectra and 1's for velocities.

D. Weights and Direction Cosines

Weights and direction cosines necessary for input to the code depend on the order of angular quadrature ISN and the type of geometry IGE. For cylindrical geometry, IGE = 2 and some orders of quadrature consistent parameter sets are given in table IX. For other parameters ISN and IGE complete sets can be found in [23].

TABLE IX

Weights and Direction Cosines for Cylindrical Geometry

	Direction Cosines (μ_m)	S_n		Weights (W_m)
1	-1.0000	S_2	1	0.0
2	-.57735		2	0.500
3	+.57735		3	0.500
1	-.9367418	S_4	1	0.0
2	-.8688903		2	.1666667
3	-.3500212		3	.1666667
4	+.3500212		4	.1666667
5	+.8688903		5	.1666667
6	-.4950046		6	0.0
7	-.3500212		7	.1666666
8	+.3500212		8	.1666666
1	-.9537974	S_6	1	0.0
2	-.6815076		2	.0880632
3	-.6815076		3	.0786035
4	-.2666355		4	.0880632
5	+.2666355		5	.0880632
6	+.6815076		6	.0786035
7	+.9261808		7	.0880632
8	-.7318110		8	0.0
9	-.6815076		9	.0786035
10	-.2666355		10	.0786035
11	+.2666355		11	.0786035
12	+.6815076		12	.0786035
13	-.3770795		13	0.0
14	-.2666355		14	.0880631
15	+.2666355		15	.0880631
1	-.975900	S_8	1	0.0
2	-.9511897		2	.0604938
3	-.7867958		3	.0453704
4	-.5773402		4	.0453704
5	-.2182179		5	.0604938
6	+.2182179		6	.0604938
7	+.5773503		7	.0453704
8	+.7867958		8	.0453704
9	+.9511897		9	.0604938
10	-.8164965		10	0.0
11	-.7867958		11	.0453704
12	-.5773503		12	.0462962
13	-.2182179		13	.0453704
14	+.2182179		14	.0453704
15	+.5773503		15	.0462962
16	+.7867958		16	.0453704

Table IX (continued)

	Direction Cosines (μ_m) S_n		Weights (W_m)
17	-.6172134	17	0.0
18	-.5773503	18	.0453704
19	-.2182179	19	.0453704
20	+.2182179	20	.0453704
21	+.5773503	21	.0453704
22	-.3086067	22	0.0
23	-.2182179	23	.0604938
24	+.2182179	24	.0604938

E. Flux output on tape

When it is desired that the calculated fluxes are output on tape, logical unit No. 7 is defined as tape unit.

F. Double T

Each ANISN/360 case must be followed by one extra card. This card may be blank or may contain the "double T" if desired. If a case terminates normally, this extra card is read and ignored. If a case is terminated by an error, the "double T" is searched for and if found, ANISN expects a title card and associated data to follow.

G. Essential Warning

ANISN should always be run in a mode which ignores floating point divide, overflow and underflow checks, and integer divide and overflow checks.

ANISN JCL

LOGICAL UNIT NO.

1. SCRATCH - always required
   UNIT = SYSDA. ----
   DCB = (RECFM = VBS, LRECL = 3620, BLKSIZE = 3624)

2. SCRATCH - Always required
   UNIT = SYSDA ----
   DCB = (RECFM = VBS, LRECL = 3620, BLKSIZE = 3624)

3. GAMSO Input - Required only if XXXX
   is in the title card.
   UNIT = TAPE ---
   DCB = (RECFM = FB, LRECL = 80, BLKSIZE = 3200)

4. Cross section library or group independent
   tape - DUMMY if neither used
   UNIT = TAPE ---
   DCB = (RECFM = VBS, BLKSIZE = 3624)

5. Standard input

6. Standard output

7. Flux output on tape
   UNIT = TAPE ----
   DCB = (RECFM = FB, LRECL = 80, BLKSIZE = 3200)
   Otherwise standard punch output

8. Flux scratch - normally dummy ;use if IDAT1 = 2
   UNIT = SYSDA

9. Flux scratch - normal dummy; use if INDAT1 = 2
   UNIT = SYSDA

5.2.6.3 DTF-IV

DTF-IV is a transport code which solves either the gamma or neutron transport problem. The code was documented by K.D. Lathrop [27].

No input description is given for this code in the present report because it is not a necessary part of the INDRA system. It has only been used a few times for comparison runs. Therefore the user must refer to the manual [27] for the theory and use of this code. The input data requirements are given starting on page 47.

The cross sections for input are prepared by the data retrieval program DRP and are read in on Logical Unit Number 10 in card image format. Thus for Unit 10

```
DCB = (RECFM = FB, LRECL = 80, BLKSIZE = 3200).
```

The fluxes are output on Logical Unit Number 7 (standard punch unit). If it is desired to put the fluxes on tape then Unit 7 must be defined as a tape unit, thus

```
UNIT = TAPE ----
DCB  = (RECFM = FB, LRECL = 80, BLKSIZE = 3200).
```

If it is desired to change the storage requirements of the code change the value of A in COMMON in the Main Program. Set I55 equal to the new dimension.

A small modification has been made in order that GAMSO output on tape may be read. If this option is desired enter a card with a "-1" in columns 2 and 3 in place of the gamma source. The gamma source will then be read on Logical Unit Number 3 in card image format. Thus for Unit 3

```
DCB = (RECFM = FB, LRECL = 80, BLKSIZE = 3200).
```

5.2.7 POPOP4 and POPMAKER

POPOP4 and the POPOP4 library tape maker are available from the computer code library - Ispra Italy. Only small modifications were necessary to incorporate the codes into the program system.

The purpose of POPOP4 is to convert gamma spectral data into gamma production cross sections. The gamma production cross sections are used in GAMSO to produce the gamma source for transport calculations.

The POPOP4 code requires as input gamma spectral data on punched cards or a binary data tape. The POPOP4 library tape maker will create the necessary binary data tape or the punched cards required using as input the BCD coded DLC-12 library (POPLIB).

The POPOP4 code also has the capability to convert existing gamma production cross section sets into the gamma and neutron energy structure of interest. These sets may be found in the literature or in some cases in the DLC-12 library.

In the case when only spectral data is found in the DLC-12 library the reaction cross sections must be input into POPOP4. These reaction cross sections may be obtained from ENDF/B files using SUPERTOG or literature sources.

The methodology used in the POPOP4 code is described in [11]. The input instructions are found on page 31 of this report but are repeated here for convenience. Note that the version of POPOP4 at IPP does not contain the POPOP4 tape maker. It is a separate code and the input instructions are found in [12], appendix E, page 150. These input instructions are also repeated here for convenience. Both codes use the FIDO input routine as described in section 5.2.1 of this report.

5.2.7.1 POPOP4 Library Tape Maker (Popmaker)

The input requirements have been reproduced from reference [12] page 150. The DD statements required for the various options have been reproduced also and follow this section.

POPOP4 library tape maker, a FORTRAN IV code, will make or update either a binary library tape or a BCD formatted library tape of secondary gamma-ray yield data sets. The two types of library tapes are required because (1) data sets can be input to POPOP4 from cards or from a binary library tape, and (2) BCD formatted (9-track) library tapes are used to pass data between installations. The code was written for use on an IBM/360 system and requires a core area of approximately 145,000 hexadecimal bytes.

POPOP4 library tape maker includes two distinct options. One option, POPBIN, makes or updates a binary library tape and the other option, POPBCD, makes or updates a BCD library tape. The first entry in the 1$ array (below) triggers either the POPBIN option or the POPBCD option. The capabilities and input for each of the options are described below.

5.2.7.1.1 POPOP4 Library Tape Maker - POPBIN Option

The capabilities of the POPBIN option are as follows:

1. Make a new binary library tape of data sets. The data sets can be input from cards or from a previously made BCD library tape.
2. Copy any or all data sets from an existing (old) binary library tape onto a new binary library tape.

3. Add data sets to the new binary library tape during the execution of capability 2 above. The data sets can be input from cards or from a previously made BCD library tape.

4. List (print) any or all the data sets on a binary library tape.

5. Punch any or all the data sets on a binary library tape in the format for use with POPOP4. (This capability is usually used to obtain data sets in order to modify data prior to use with POPOP4.)

The data sets are placed on the binary library tape in ascending order according to a six-digit data set identification number. The scheme used for the identification number is described in reference [12], page 40. The library ID numbers are listed in [13] and [28] and once more reproduced in Table VII of this report.

All numerical input are written in the FIDO format.

1. Title Card (format 18A4). This card is used to identify the tape being made, the data, etc.

2. 1ϕ Array (six integer parameters):
   LTRIG Must be zero.
   NADDRE The number of data sets to be added to the new POPOP4 binary library tape. If the data sets are to be input from cards, NADDRE is positive; if the data sets are to be read from a previously made POPOP4 BCD library tape, NADDRE is negative. The data sets to be added are identified in the 2ϕ array. (If no data sets are to be addes, NADDRE is zero).

NSUBRE — The number of data sets to be removed from the old binary or old BCD library tape when the new binary library tape is made. (The data sets are actually not removed from the old tape but are not transferred from the old to the new tape.) The data sets are identified in the 3$ array.

NREPRT — The number of data sets to be printed from the new binary tape (the tape identified with logical number 34). The data sets are identified in the 4$ array. Data sets with negative identification numbers in the 4$ array are also punched on cards.

NEWOLD — 0/1 no effect/copy the old binary library tape onto a new binary library tape. If NEWOLD equals one, NADDRE and NSUBRE must be zero; i.e., this is a special option to be used only for copying an existing tape.

NREA — The number of data sets on the old binary library tape.

End the 1$ array with a T.

3. 2$ Array (|NADDRE| entries):

If NADDRE = 0, do not enter a 2$ array.
If |NADDRE| > 0, enter the identification numbers of the data sets to be added to the new binary library tape. The entries must be in ascending order.

End the 2$ array with a T.

4. 3$ Array (NSUBRE entries):

If NSUBRE = 0, do not enter a 3$ array.
If NSUBRE > 0, enter the identification numbers of the data sets to be removed from the old binary or

old BCD library tape when the new binary library tape is made. The entries must be in ascending order.

End the 3$ array with a T.

5. 4$ Array (NREPRT entries):
   If NREPRT = 0, do not enter a 4$ array.
   If NREPRT > 0, enter the identification numbers of the data sets to be printed. If an identification number is negative, the data for the data set are punched in a form suitable for use in "Section F" of the POPOP4 input. The entries in this array are checked in the code to insure that the absolute values of the identification numbers are in ascending order.

   End the 4$ array with a T.

6. If NADDRE > 0, "Section F" data (see section 5.2.7.2) for POPOP4 are entered on cards for each of the NADDRE reactions to be added to the new binary library tape. The blocks of "Section F" data must be entered in order corresponding to the identification numbers in the 2$ array.
   If NADDRE = 0, no entry is made for this section of data.
   If NADDRE < 0, no entry is made for this section of data. The NADDRE reactions to be added to the new binary library tape are read from the previously made BCD library tape.

5.2.7.1.2 POPOP4 Library Tape Maker - POPBCD Option

The capabilities of the POPBCD option are as follows:

1. Make a new BCD library tape of data sets.
2. Copy any or all data sets from an existing (old) BCD library tape onto a new BCD library tape.
3. Add data sets to the new BCD library tape during the execution of capability 2 above.

4. List (print) any or all of the data sets on a BCD library tape.
5. Punch any or all the data sets on a BCD library tape in the format for use with POPOP4. (This capability is usually used to modify data in a data set prior to use with POPOP4.)

The data sets are placed on the BCD library tape in ascending order according to the six-digit data set identification numbers.

Input Description

1. Title Card (format 18A4). This card is used to identify the new BCD library tape, the date, etc.
2. 1$ Array (six integer parameters):

   LTRIG — Must be minus one (-1).

   NADDRE — The number of data sets to be added to the new BCD tape. The data sets are identified in the 2$ array.

   NSUBRE — The number of data sets to be removed from the old BCD tape. (The data sets are actually not removed from the old tape but are not transferred from the old to the new tape.) The data sets are identified in the 3$ array.

   NREPRT — The number of data sets to be printed from the new BCD tape (the tape identified by logical number 34). The data sets are identified in the 4 $ array. Data sets with negative identification numbers in the 4 $ array are also punched on cards.

   NEWOLD — 0/1 no effect/copy the old BCD library tape onto a new BCD library tape. If NEWOLD equals one, NADDRE and NSUBRE must be zero; i.e., this is a special option to be used only for copying an existing tape.

   NREA — The number of data sets on the old BCD tape.

   End the 1$ array with a T.

3. 2$ Array (NADDRE entries):

   If NADDRE = 0, do not enter a 2$ array.
   If NADDRE > 0, enter the identification numbers of the data sets to be added to the new BCD library tape. The entries must be in ascending order.

   End the 2$ array with a T.

4. 3$ Array (NSUBRE entries):

   If NSUBRE = 0, do not enter a 3$ array.
   If NSUBRE > 0, enter the identification numbers of the data sets to be removed from the old BCD library tape when the new BCD library tape is made. The entries must be in ascending order.

   End the 3$ array with a T.

5. 4$ Array (NREPRT entries):

   If NREPRT = 0, do not enter a 4$ array.
   If NREPRT > 0, enter the identification numbers of the data sets to be printed. If an identification number is negative, the data for the data set are punched in a form suitable for input to POPOP4. The entries in this array are checked to insure that the absolute values of the identification numbers are in ascending order.

   End the 4$ array with a T.

6. The POPOP4 "Section F" data (see section 5.2.7.2) are entered for each of the NADDRE data sets to be added. The blocks of Section F data must be entered in an order corresponding to the identification numbers in the 2$ array.

This concludes the input for POPOP4 library tape maker - POPBCD Option.

5.2.7.1.3 POPMAKER-JCL

The DD statements required for the various options of the POPOP4 library tape maker program (POPMAKER) are summarized in Tables X and XI.

TABLE X

DD Statements Required for Various POPOP4 Library Tape Maker Options*

OPTION						DD Statements Required (GO Step)
LTRIG	NADDRE	NSUBRE	NREPRT	NEWOLD	NREA	
0	>0	0	≤NT**	0	0	A,D
0	0	0	≤NT	1	>0	A,D,F
0	0	>0	≤NT	0	≥NSUBRE	A,D,F
0	>0	0	≤NT	0	>0	A,D,F
0	>0	>0	≤NT	0	>0	A,B,D,F
0	<0	0	≤NT	0	0	A,C,D
0	<0	0	≤NT	0	>0	A,C,D,F
0	<0	>0	≤NT	0	>0	A,B,C,D,F
0	0	0	≤NT	0	>0	D
-1	>0	0	≤NT	0	0	A,E
-1	0	0	≤NT	1	>0	A,E,G
-1	0	>0	≤NT	0	≥NSUBRE	A,E,G
-1	>0	0	≤NT	0	>0	A,E,G
-1	>0	>0	≤NT	0	>0	A,B,E,G
-1	0	0	≤NT	0	>0	E

* DD statements for card input, print, and punch devices are not listed.

** NT = NREA + NADDRE - NSUBRE.

TABLE XI

POPOP4 Library Tape Maker DD Statements for the GO Step

CARD ID	CONTROL CARD	DEVICE USE
A	//G .FT10F001 DD UNIT=2311,SPACE=(CYL,(5,1)), DCB=(RECFM=VBS, BLKSIZE=2624)	Scratch Unit
B	//G .FT20F001 DD UNIT=2311,SPACE=(CYL,(50,10,1), DCB=(RECFM=VBS,BLKSIZE=3624)	Scratch Unit
C	//G FT24F001 DD UNIT=TAPF, DISP=(OLD,PASS), LABEL=(1,SL), DCB=(RECFM=FB,LRECL=133, BLKSIZE=3990,VOL=SER=(tape identification)	Old BCD library tape for POPBIN option
D	//G .FT34F001 DD UNIT=TAPE, DISP=(OLD,PASS), LABEL=(1,SL), DCB=(RECFM=VBS, BLKSIZE=3604), VOL=SER=(tape identification)	New binary library tape for POPBIN option - or old binary library tape if NADDRE, NSUBRE, and NEWOLD equal zero and NREPRT greater than zero.
E	//G .FT34F001 DD UNIT=TAPE , DISP=(OLD,PASS), LABEL=(1,SL), DCB=(RECFM=FB,LRECL=133, BLKSIZE=3990, VOL=SER=(tape identification)	New BCD library tape for POPBCD option - or old BCD library tape if NADDRE, NSUBRE, and NEWOLD equal zero and NREPRT greater than zero.
F	//G .FT40F001 DD UNIT=TAPE , DISP=(OLD,PASS), LABEL=(1,SL), DCB=(RECFM=VBS, BLKSIZE=3604), VOL=SER=(tape identification)	Old binary library tape for POPBIN option
G	//G .FT40F001 DD UNIT=TAPE , DISP=(OLD,PASS), LABEL=(1,SL), DCB=(RECFM=FB,LRECL=133, BLKSIZE=3990, VOL=SER=(tape identification)	Old BCD library tape for POPBCD option

5.2.7.2 POPOP4

POPOP4 input instructions are reproduced from reference [11] page 31 ff. All numerical input data are written in FIDO format.

Input Description

A. Title Card (format 18A4). This card is used to identify the nuclide, all reactions included in the case, the group structure, etc. The information on this card is punched when POPOP4 is requested to punch the gamma-ray production cross sections. (It will be the first card of the punched cards.)

B. 1$ Array (integer parameters in FIDO format):

NMATLS The number of reactions which will be reduced to the desired group structure, e.g., (n;n',γ), (n;γ) non-fission, and (n;γ) fission would be three reactions; therefore, NMATLS would equal 3.

NFLT The number of reactions to be input from a library tape. The remaining reactions are input in card form (Section F below).

ING The number of neutron energy groups for the required gamma-ray production cross section array.

IGG The number of gamma energy groups for the required gamma-ray production cross section array.

LOOK LOOK = 0, only necessary information is printed.
LOOK = 1, a detailed check of the calculations is printed. (This option causes a vast amount of extraneous junk to be printed.)

MUTT MUTT = 0, no effect.
MUTT = 1, the final gamma-ray production cross sections for each reaction are added

and the resulting array printed. NMATLS must be less than or equal to 10.
MUTT = 1, same as for MUTT equal to -1, except the resulting array is also punched in FIDO format.

End the 1$ array with a T.

C. 2* Array (ING+1 entries):
The required neutron energy structure, i.e., the upper energy of each neutron group, and the bottom energy of the lowest neutron energy group. The upper energy of the highest energy group is entered first. Energies must be in electron volts (eV). Do not end the 2* array with a T.

D. 3* Array (IGG+1 entries):
The required gamma energy structure, i.e., the upper energy of each gamma group and the bottom energy of the lowest gamma energy group. The upper energy of the highest energy group is entered first. Energies must be in eV.

End the 3* array with a T.

E. 3$ Array (NFLT entries):
If NFLT = 0, do not enter a 3$ array.
If NFLT > 0, enter the identifying number of the reactions to be read from a POPOP4 library tape (Section F data), e.g., 928101 might be the identifying number for the (n;γ) reaction for a $_{92}U^{238}$ target. (Note that IPUNO (4$ array below) can be either 0 or 1. If a negative identification number is input, e.g., -928101, the value for IPUNO for the reaction on the library tape is changed to the other possible value.)

End the 3$ array with a T.

POPOP4 assumes that the Section F data for the first NFLT reactions come from a library tape. The requirement for Section G data depends on the value of MULTRY (4$ array below). Section G data must be input from cards.

F. If the reaction is read from a library tape, this section of data is omitted. For the last (NMATLS-NFLT) reactions, enter the following:

Title card identifying the nuclide, the reaction, the source of the cross sections or yields, etc. (format 20A4).

4$ Array (integer parameters in FIDO format):

LGGTYP How the gamma energies of the given yield or photon production cross section array will be entered in the 6* array:
LGGTYP = 1, the energies are "representative".
LGGTYP = 2, the energy boundaries of the gamma groups are entered.
LGGTYP = 3, discrete gamma energies are entered.

IGNG If KSEVEN (below) = 0, enter the number of neutron groups in the given yield or photon production cross section array.
If KSEVEN = 1, enter "0".

IGGG Number of gamma groups in the given yield or photon production cross section array.

IPUNO IPUNO = 0, no effect.
IPUNO = 1, the final photon production cross section array for the reaction is punched in FIDO format. If MUTT (1$ array) is one, IPUNO should normally be zero.

MULTGY MULTGY = 1, no effect.
MULTGY = 2, the entries for each neutron group in the given yield or photon production cross section array is multiplied by the corresponding entry in the 8* array.

MULTRY MULTRY = 1, no effect.
MULTRY = 2, the required yields or photon production cross sections for the reaction are multiplied by the entries in the 9* array. The values for the highest required neutron group are multiplied by the first entry in the 9* array, the values for the second highest required neutron group are multiplied by the second entry in the 9* array, etc.

KSEVEN KSEVEN = 0, no effect.
KSEVEN = 1, IGGG entries are made in the 7* array - one yield or photon production cross section for each given gamma group. POPOP4 assumes that the IGGG entries are applicable to each given neutron group.

KBINEN KBINEN = 0, no effect.
KBINEN = 1, the yields or photon production cross sections in the lowest given neutron group are multiplied by the respective gamma energy in the 5* array (if LGGTYP=3) or are multiplied by the respective midenergy of the gamma group (if LGGTYP≠3). These products are summed for the IGNG groups, and the result is printed following the reaction input printout.

End the 4$ array with a T.

5* Array
If KSEVEN = 1, do not enter a 5* array.
If KSEVEN (above) = 0, the 5* array is the upper energies of the neutron energy groups for the given yield or photon production cross section array and the bottom energy of the lowest neutron group. The first entry is the upper energy of the highest group. Energies must be in eV.
End the 5* array with a T.

6* Array
If LGGTYP = 1 (IGGG entries):
The representative gamma energies of the given yield or photon production cross section array. Energies must be in eV.

End the 6* array with a T.

If LGGTYP = 2 (IGGG+1 entries):
The upper gamma energies of the given yield or photon production cross section array and the bottom energy of the lowest group. The first entry is the upper energy of the highest gamma group. Energies must be in eV.

End the 6* array with a T.

If LGGTYP = 3 (IGGG entries):
The discrete gamma energies of the given yield or photon production cross section array are entered. The highest energy is entered first. Energies must be in eV.

End the 6* array with a T.

7* Array
If KSEVEN = 0 (IGNG X IGGG entries):
The given yields or photon production cross sections for the reaction. The values for the lowest given gamma group are entered first - one value for each given neutron group (highest neutron group first). The values are entered in progressive order from the lowest to the highest given gamma group.

If KSEVEN = 1 (IGGG entries):
The given yields or photon production cross sections for the reaction. The values for only one given neutron group are entered - lowest gamma group first. POPOP4 assumes that the IGGG entries are applicable to each given neutron group.

End the 7* array with a T.

8* Array (IGNG entries). Enter only if MULTGY = 2.

The given yields or photon production cross sections for the reaction are multiplied by the appropriate entry in this array. The given values (7* array) in the highest given neutron group are mutiplied by the first entry in this array. The given values (7* array) in the lowest given neutron group are multiplied by the IGNGth entry.

End the 8* array with a T.

G. 9* Array (ING entries). Enter only if MULTRY = 2.

The required yields or photon production cross sections for the reaction are multiplied by the appropriate entry in this array. The required values in the highest required neutron group are multiplied by the first entry in this array. The required values in the lowest required neutron group are multiplied by the INGth entry in this array. (This array is usually used for the neutron reaction cross sections.)

End the 9* array with a T.

This concludes the input for a reaction. If there are more reactions for the element,return to F above.

POPOP4 Control Cards

The control cards required for POPOP4 depend on the computing system and program option used. The conversion problem requires one scratch disk of about 4500 words. The DD statements shown reflect the fact that at IPP POPOP4 and the POPOP4 tape maker are two separate codes.

DD Statements Required for Various POPOP4 Options

Options	Conversion (C) and Tape Maker (T)	DD Statements Required
		GØ Step
NMATLS O,NFLT=O	C	A,C
NMATLS O,NFLT≠O	C	A,B

CARD ID	CONTROL CARD
A	//G.FT1OFOO1 DD UNIT=2311,SPACE=(CYL,(5,1)),
	DCB=(RECFM=VBS,BLKSIZE=36OO)
B	//G.FT34FOO1 ND UNIT=TAPE,DISP=(ØLD,PASS),
	LABEL=(1,SL),VØLUME=SER=(tape identification),
	DCB=(RECFM=VBS,BLKSIZE=36O4)
C	//G.FT34FOO1 DD DUMMY

Comments:

A) A message is printed out to the user to refer to "DUMY" in the main program and determine if enough storage is available. The amount of storage needed is calculated for each problem. If the amount of storage exceeds the value in "DUMY", increase its value and rerun the problem.

B) If the manual states tnat Logical Unit 44 is required, use Logical Unit 40 instead.

C) The results to be expected from the gamma source calculation and all subsequent steps of the photonics part of the INDRA system are only as good as the data used for input to POPOP4. Therefore, one always should refer to the original literature to determine if the data are applicable to fusion reactor calculations. An example of this type report is [28].

Other useful reports in helping define the capabilities and limitations of POPOP4 are [10, 29 - 32].

5.2.8 MUG (MULTIGROUP GAMMA CROSS SECTIONS)

MUG is a code which calculates multigroup photon cross sections, up to a maximum of 100 groups and transfer coefficients represented by a Legendre approximation up to P_{12}. In addition it calculates gamma kerma factors in the group structure desired.

The cross sections are used in gamma transport calculations and the kerma factors are used in gamma heating calculations.

The MUG code may be obtained from the Computer Code Library, Ispra Italy, and is described in [5]. The input cross sections are described in [15]. The user is urged to familiarize himself with these documents, as they describe the theory and cross section sets available.

There is no change to the input data as described in [5]. The output, however, has been changed. The cross sections are output on tape in a format that is suitable for input into DRP and the kerma factors are output on tape in a format suitable for input to GLUCKE.

In order that the cross sections are placed on tape in the proper order, each element must be entered as a separate case. That is, cross sections for as many elements as desired may be made in one run, but for each element separate input data must be prepared.

Although DRP has the capability to read more than one file, as a matter of convenience it is desirable to create in one run the cross sections for all elements expected to be used in a series of transport calculations. In this manner, they will all be contained on the same file.

The input described below was taken from Ref. [5]. It reflects the method of input for use with the program system when using the DLC-7 library.

Input Description

CARD 1. Title (18A4)

CARD 2. N,MF,LMAX,NUF,NSP,NPHI,T,LSTART,NTAPE (6I5,D10.4,2I5)

N = Number of energy levels (number of groups + 1), $\leq$ 101.

MF
1, enter flux
2, program supplies flat flux.
3, program supplies Einstein-Bose flux

LMAX = Highest L in P_L coefficients, $\leq$ 12.

NUF
0, energy dependent flux
1, photon number flux.
2, photon number flux, Compton only.

NSP = Number of intervals in outer integral (31 recommended) (100 maximum)

NPHI = Number of intervals in inner integral (51 recommended) (100 maximum)

T = Mean temperature (used if MF = 3).

*LSTART= FIRST L in P_L

NTAPE
0, cross sections are on cards.
1, cross sections from tape.

CARD(s) 3. Energy levels (6D12.6)
Enter N values ($\leq$ 101), in MeV, starting with the lowest energy.

CARD(s) 4. Weighting flux (used only if MF=1) (6D12.6)
Enter NPHI*(N-1) values.
The first NPHI values are for the lowest energy group, the next NPHI values are for the next higher energy group, etc.

CARD 5. NDS = no. of downscatters (I2)
Must be $\leq$(N-2).

CARD 6. LANO, SYMB, INDF, MX, LEMNO (I4, 2X, A4, 3I5)
LANO = Element Atomic number
SYMB = Element symbol
INDF = 0, No effect / 1, Read ENDF/B data
MX = 0, ENDF/B tape is BCD
1, ENDF/B tape is binary
LEMNO = 0, Element ID = LANO
0, Element ID = LEMNO

CARD 7. Blank Card
Repeat cards 1 - 7 for each element desired

NOTE: THE FOLLOWING VALUES ARE REQUIRED:

NUF = 1
LSTART = 0
NDS = N-2 (Not: NDS < N-2)
INDF = 1.
MX = 0
LEMNO = 0
NTAPE = 1

JCL for MUG

The input cross sections from the DLC-7 library are read on logical unit 9. The cross sections are in card image format, therefore for Unit 9,

DCB = (RECFM=FB,LRECL=80,BLKSIZE=3200)

The calculated cross sections are output on logical unit 7 in card image format; therefore for unit 7,

DCB = (RECFM=FB,LRECL=80,BLKSIZE=3200)

The kerma factors are output on logical unit 8 in card image format. Therefore for unit 8,

DCB = (RECFM=FB,LRECL=80,BLKSIZE=3200)

Comments:

The original code outputs the kerma factors in the scattering matrix in the position normally used for the absorption cross sections.

A subroutine "WEXEL" has been added which outputs the kerma factors on tape and replaces them in the scattering matrix with the absorption cross sections. WEXEL is called from NOFLUX. If NUF=0, WEXEL will not be called and no substition or kerma on tape will occur.

The subroutine WEXEL also converts the kerma factors to units of $\frac{(\text{WATT-SEC/ATOM})}{(\text{GAMMA/cm}^2)} \times 10^{24}$ so that the reaction rates will have the units of WATTS/cm^3.

The kerma factors are output with a title card with the format (9A4, A1, 10A4, A3.) The ID number is found in the 11th word. It is the atomic number of the element.

Note that if the last card of a set of kerma factors is not completely full, that the remainder of the card is filled with zeros. These zeros, however, are not used in any calculation.

The cross sections are output with a title card in the format 9A4, A1, 10A4, A3. Four dots are in the first four columns and the ID number is the 11th word. The

ID numbers are the atomic numbers of the element. The subroutine DTKP was changed so that each new energy group begins with a new card. These changes were necessary so that DRP can read the cross sections.

Input from the DLC-7 library requires that the atomic numbers (ID numbers) be input in increasing order. The output cross sections and kerma factors are also in increasing order of the atomic number.

5.2.9 MACK

MACK is a computer code which calculates neutron kerma factors from ENDF/B data files. These kerma factors are used in GLUCKE to evaluate volumetric heating rates due to neutron interactions. A complete description of MACK was given by its authors M.A. Abdou e.a. [33].

MACK published in 1973 replaces in the INDRA system the formerly used code AVKER [8] with the corresponding small library DLC-10. To the latter code a subroutine XKERM had been added, which calculated approximate kerma factors from data of DLC-2 type libraries for elements not contained in DLC-10.

The following input description of MACK was reproduced from reference [33].

Input Description

Card No. 0 (2I6)

This card is always the first card in the input data. It is required only once in a single run and is not to be repeated for each material.

Item	Cols.	Name	Description
1	1-6	IDTAP	ENDF/B tape identification number
2	7-12	MODE	Mode of ENDF/B data tape. 1 = binary tape 2 = BCD tape

The input data for each material is as follows.

Card No. 1 (18A4)

Title card.

Card No. 2 (I6,4A4,2X,2E12.4)

1	1-6	MATNO	ENDF/B tape material number.
2	7-22	MATHOL	Material name.
3	25-36	TEMTUR	$\leq$ 0 = no effect T = temperature (in ^{o}Kelvin) for Doppler broadening of resonance cross sections
4	37-48	SIGP	Non-resonance isotope potential scattering cross sections in barns per absorber atom.

Card No. 3 (12I6)

1	1-6	LINK1A	0 = no effect. 1 = process pointwise cross sections. (must be 1 if LINK2A = 1)
2	7-12	LINK1B	0 = no effect. 1 = calculate group cross sections.
3	13-18	LINK2A	0 = no effect. 1 = calculate pointwise kerma factors (must be 1 if LINK2B = 1)
4	19-24	LINK2B	0 = no effect 1 = calculate group kerma factors.
5	25-30	IPRT1A	0 = no effect 1 = print pointwise cross sections.

Item	Cols.	Name	Description
6	31-36	IPRT1B	0 = no effect.
			1 = print group cross sections.
7	37-42	IPRT2A	0 = no effect.
			1 = print pointwise kerma factors by reaction.
			2 = write pointwise kerma factors by reaction (and total) on tape.
			3 = both 1 and 2.
			4 = write only pointwise total kerma factors on tape.
8	43-48	IPRT2B	0 = no effect.
			1 = print group kerma factors.
9	49-54	IPUN1A	0 = no effect.
			1 = punch pointwise cross sections except inelastic levels.
			2 = punch pointwise cross sections for all reactions.
10	55-60	IPUN1B	0 = no effect
			1 = punch group reaction cross sections
11	61-66	IPUN2A	0 = no effect.
			1 = punch pointwise kerma factors by reaction (and total)
			2 = punch pointwise total kerma factor.
12	67-72	IPUN2B	0 = no effect.
			1 = punch group kerma factors.

Card No. 4 (12I6)

1	1-6	IRESON	0 = no effect.
			1 = calculate resonance cross sections if resonance parameters are present.
			2 = read resonance cross sections from tape (logical unit 8)

Item	Cols.	Name	Description
2	7-12	KEY	1 = process all reactions on ENDF/B tape for that material.
			2 = process only the reactions specified on Card No. 11.
3	13-18	IWXS	1 = flat weighting for group cross sections.
			2 = 1/E weighting for group cross sections.
4	19-21	IWKF	1 = flat weighting for group kerma factors.
			2 = 1/E weighting for group kerma factors.
			3 = weighting is input.
			4 = weighting is input * 1/E.

Card No. 5 (12I6)

1	1-6	INEP	1 = calculate energy point mesh.
			2 = read energy points.
			3 = use energy mesh from previous material.
2	7-12	NEP	number of energy points for the energy mesh (not used if INEP = 3)
3	12-18	NRANGE	number of energy ranges with equal lethargy intervals ($\leq$10). (Used only if INEP = 1).
4	19-24	IGRPS	number of energy groups (300). (not used if INEP = 3 and/or LINK1B and LINK2B are zero).
5	25-30	IGAM	1 = GAM-II 100-group structure.
			2 = input energy group structure.

Card No. 6 (12I6): This card is entered only if INEP = 1.

1	1-6	NIR(1)	number of energy points in range number 1
2	7-12	NIR(2)	number of energy points in range number 2.
.		.	
.		.	
.		.	
.		.	
NRANGE		NIR(NRANGE)	number of energy points in the last range.

Note: The user must ensure that NEP = 1 + $\sum_i$ NIR$_i$.

Card No. 7 (6E12.4)

This may be one or two cards and is entered only if INEP = 1.

Item	Cols.	Name	Description
1	1-12	ERB(1)	lowest energy for range number 1. (= lowest point for the energy mesh).
2	13-24	ERB(2)	lowest energy for range number 2. (= highest energy for range number 1.)
3	25-36	ERB(3)	lowest energy for range number 3.
.		.	
.		.	
.		.	
etc.		.	
.		.	
.		.	
NRANGE		ERB(NRANGE)	Lowest energy for range number NRANGE.
NRANGE + 1		ERB(NRANGE + 1)	highest energy for range number NRANGE. (= highest energy point for the energy mesh).

Card No. 8 (6E12.4)

This is actually a card set and consists of the points of the energy mesh. It is entered only if INEP = 2.

1	1-12	EP(1)	energy point number 1. (lowest energy).
2	13-24	EP(2)	energy point number 2.
3	25-36	EP(3)	energy point number 3.
.		.	
.		.	
.		.	
.		.	

etc. using NEP/6 cards

Item	Cols.	Name	Description
.		.	
.		.	
.		.	
.		.	
.		.	
NEP		EP(NEP)	energy point number NEP. (highest energy)

Card No. 9 (6E12.4)

This is a card set and is entered only if a) INEP = 1 or 2, and b) LINK1B and/or LINK2B is 1 and IGAM ≠ 1. It consists of the desired energy group structure.

1	1-12	EGRPB(1)	energy group breakpoint number 1. (lowest energy).
2	13-24	EGRPB(2)	energy group breakpoint number 2.
.		.	
.		.	
.		.	
.		.	
etc. using (NGRPS + 1)/6 cards			
.		.	
.		.	
.		.	
NGRPS		EGRPB(NGRPS)	energy group breakpoint number NGRPS.
NGRPS+1		EGRPB(NGRPS+1)	energy group breakpoint number NGRPS+1 (highest energy).

Card No. 10 (E12.4,4A4)

1	1-12	CONVF	conversion factor to convert kerma factors from electron volt*barn/atom to any other desired units.
2	13-28	UNTHOL	maximum of 16 alphanumeric characters describing the units of kerma factors after CONVF is used.

Card No. 11 (12I6)

This can be one or two cards and is entered only if KEY = 2 and consists of the number of reactions and the MACK number for each reaction desired.

Item	Cols.	Name	Description
1	1-6	NREAC	number of reactions to be processed for this material.
2	7-12	IR(1)	MACK number for the first reaction.
3	13-18	IR(2)	MACK number for the second reaction.
.		.	
.		.	
.		.	
.		.	
NREAC + 1		IR(NREAC)	MACK number for the last reaction.

Card No. 12 (6E12.4)

This is actually three cards and they are always required. The decay energies for the various reactions are entered on these three cards in units of electron volts. Decay energies for the reactions which are not required are not used and the corresponding fields can be left blank.

Item	Cols.	Name	Description	ENDF/B Reaction No. (MT)
I		E(I)	E decay for:	
1	1-12	DECG	(n,γ)	102
2	13-24	DECP	(n,P)	103
3	25-36	DECD	(n,d)	104
4	37-48	DECT	(n,t)	105
5	49-60	DECHE3	(n,He3)	106
6	61-72	DECA	(n,α)	107
7	1-12	DEC2A	(n,2α)	108
8	13-24	DEC3A	(n,3α)	109
9	25-36	DECN2N	(n,2n)	16
10	37-48	DNPA	(n,n')α	22
11	49-60	DNP3A	(n,n')3α	23
12	61-72	D2NA	(n,2n)α	24
13	1-12	DNPP	(n,n')P	28

Card No. 13

This set of cards is required if IWKF = 3 or 4 and it consists of the weighting function plus the interpolation scheme. The format of the card set is a standard ENDF/B

TAB 1 record (except for N1 and N2 on the first card). The weighting function must be tabulated from low to high energy.

Card 13.1 (2I11)

Item	Cols.	Name	Description
1	1-11	N1	Number of interpolation ranges.
2	12-22	N2	Number of weighting function points.

Card 13.2 (6I11)

1	1-11	NBT(1)	Last point number in first interpolation range.
2	12-22	JNT(1)	Interplation scheme for first range.
3	23-33	NBT(2)	Last point number in 2nd interpolation range.
4	34-44	JNT(2)	Interpolation scheme for second range.
.		.	
.		.	
.		.	
.		.	
etc.		.	
		.	
		.	
		.	
2*N1-1		NBT(N1)	Last point number in the N1 interplation range.
2*N1		JNT(N1)	Interpolation scheme for the N1 range

Card 13.3 (6E11.4)

1	1-11	X(1)	First energy point ($\leq$lowest energy or the energy point mesh).
2	12-22	Y(1)	Weight at this energy.
etc. (using N2/3 cards)			
2*N2-1		X(N2)	Last energy point ($\geq$highest energy in the energy point mesh).
2*N2		Y(N2)	Weight at this energy.

Comments:
Reference [33] gives detailed information about the input data, starting on page 37. Some of the most essential comments are summarized below.

A: SIGP on card no. 2 is treated as in SUPERTOG. For the usual case of infinite dilution one should use e.g. SIGP = 10^8.

B: Energy Mesh Construction (cards 5 to 9)

This set of cards flexibly specifies the desired energy mesh and group structure. It should be noted that all energies in the input are always entered in units of electron volts and in the order of increasing energy.

If INEP = 3, the energy mesh and group structure, if any, are used from the previous material and cards 6 through 9 should not be entered.

If INEP = 2, cards no. 6 and no. 7 should not be entered and the energy mesh is entered on the next card (no. 8) in the order of increasing energy (FORMAT (6E12.4)). The number of entries must be equal to NEP.

The energy mesh is generated by the code at NEP points if INEP is specified as 1. Cards no. 6 and no. 7 are needed in this case and they specify the characteristics of the energy mesh. The total energy range is divided into a number of energy ranges given by the entry NRANGE. The energy breakpoints for these ranges are entered in the array ERB which has NRANGE + 1 entries. Each range, i, is divided into a number of intervals, NIR_i, of equal lethargy width. The energies of the boundaries for these intervals define the points for the energy mesh.

It is recommended to the user to have at first a look to the cross sections before constructing an appropriate energy mesh for running the code. In the case of the INDRA system this possibility exists by studying the SUPERTOG-output.

C: CONVF on card no. 10 should be set to CONVF = $1.602 \cdot 10^{-19}$ to gain the kerma factors in units acceptable by GLUCKE.

D: Card No. 11

In the usual case all reactions should be taken into account for the kerma factor calculation. If a user wants to process only single reaction types - the only case in which this card has to be entered - he should take the reaction type number from reference [33], table 2.

E: Card No. 12

Decay energies for various elements and reactions can be found in reference [34].

Operational Note

No changes have been made up to now to the code in order to move the final kerma factors to a magnetic tape which can be read by GLUCKE. To do this the tape output would have to be converted to a FIDO format. Each set of data must further be preceeded by a ID card of following format:

(9A4, A1, 10A4, A3)

The ID number is supplied by the user and is right adjusted in the 11th word (columns 38 → 41). The remainder of the space may be used for text to describe the set.

5.3 Input Description and Remarks for the Individual Not Yet Documented Codes

There are three types of codes supplied by the authors.

The first type to which we want to count the transport code DTFMF and the retrieval program TREVE are codes not necessarily needed by the INDRA system because there is at least one alternative set of already published programs namely ANISN and DRP. The existance of these additional programs is due to the historical development of INDRA.

The second type comprises a number of auxiliary codes which the present version of INDRA needs to work: CARDTAPE, GAMSO, GLUCKE, CROSS, and GROUCO. These are codes being either necessary links in the blanket analysis procedure or, as is GLUCKE, a code for evaluating the final results from the flux spectra.

The third type is a set of two auxiliary programs, FLUZI and GLOUMAN, which have been prepared for a convenient handling of the final results including the possibility of graphical display.

5.3.1 TREVE

TREVE is a data retrieval program for use with DTFMF. It will read and place on tape those cross sections of interest from large data libraries such as the DLC-2C and DLC-2D neutron cross section libraries. It will also read the cross section tapes output from MUG, APFRX and SUPERTOG.

When desired it will enter or change the thermal cross sections.

The output will list the record numbers of the cross section sets on tape. These record numbers are used for input to DTFMF to locate the cross section sets of interest on tape. Note that a record number must be entered into DTFMF for each P-expansion of an element. DTFMF will skip over those sets not used, thus as large a library as desired may be created.

Input Description

Card 1: FORMAT (20A4)

Title Card

Card 2: FORMAT (4I5)

Control Parameters

NE - Number of elements

NP - Maximum P expansion of new cross section tape

NG - Number of energy groups

NGT - Enter new thermal cross sections = 99/100 - YES/NO

Card(s) 3: FORMAT (A4, A4, A4, 3F10.5)

Identification card and thermal cross sections (if entered). One card for each element to be placed on the new tape.

AUMBER element ID number on old tape

AN(1), AN(2) - Eight spaces used to print element name or other information

SA Thermal abosrption cross section

VSF Thermal fission cross section * number of neutrons per fission, i.e. $\nu\sigma_f$.

SS Thermal scattering cross section.

Job Control Language

UNIT 1 - Output cross section tape
DCB=(RECFM=VSB,BLKSIZE=1000,BUFNO=1)

UNIT 2 - Input cross section tape
DCB=(RECFM=FB,LRECL=80,BLKSIZE=3200)

A scratch unit must be provided for each P-expansion greater than zero beginning with unit 8. Thus, for example if a P_3 cross section set was being created three scratch units would be needed.

```
//G .FT08F001  DD UNIT = SYSDA,
               SPACE = (20,(21400,1)),
               DCB = (RECFM=VSB,BLKSIZE=7200)
//G .FT09F001  DD UNIT = SYSDA,
               SPACE = (20,(21400),1)),
               DCB = (RECFM = VSB,BLKSIZE=7200)
//G .FT10F001  DD UNIT = SYSDA,
               SPACE = (20,(21400,1)),
               DCB = (RECFM=VSB,BLKSIZE=7200)
```

Comments:

The thermal cross sections may be entered or changed depending on the value NGT given. If NGT = 100, it is not necessary to enter SA, VSF or SS. They may be left blank.

In the case of the DLC-2C library note that it is a 100 group library with zeros for the thermal group. For this library it is always necessary to enter SA, VSF and SS.

For other libraries if NGT =99 the values of the thermal cross sections will be changed to the values of SA, VSF and SS. The total cross section will be given the value of SA + SS.

5.3.2 DTF MF (DTV-IV Modified for Fusion)

DTF MF is a revision of the DTF-IV discrete ordinates transport code described in reference [27].

Since the revisions made did not affect the integration procedure of the DTF-IV code, the user should refer to the above users manual for the theory. The criteria for

the weights and directions, inner convergence criterion and negative flux fixup remain the same. However, the subroutines used for activity calculations, upscatter, adjoint calculations, search calculations, and fission source calculation were removed.

This results in a code with simplified input which may be used with large cross section libraries and which is most suitable for computers with a small core storage capability.

The resulting code will only solve source type calculations, i.e.: a fusion reactor blanket with constant neutron or gamma source. Note that in independent source calculations, convergence is complete as soon as a converged inner solution is obtained. The DTF MF code then requires only one outer iteration. Therefore both EPS and EPSA should be used and IIM made large.

The cross section input routine was rewritten in order to mix the cross sections in temporary storage and then to place the mixed cross sections on disks. The input routine will only read cross sections that have been placed on tape by the data retrieval program TREVE. The cross sections are read in unformatted.

The fluxes are output unformatted. GAMSO input, if used, is read in in card image format.

The size of the storage required may be changed by changing the dimension of A in the main program. Set I1 equal to the new value.

Input Description

SECTION A

Card 1: FORMAT (20A4)

Title Card

Card 2, 3: FORMAT (12I6)

Control Parameters Integers

ID	Problem Identification Number
ISCT	Scattering (O/N=Isotropic/Nth Order Anisotropic)
ISN	Quadrature Order-2,4,6,etc. (N in SN-PN-1=DPN/2-1)
IGE	Geometry (1/2/3=Plane/Cylinder/Sphere)
IBL	Left Boundary Condition (O/1/2=Vacuum/ Reflective/Periodic)
IBR	Right Boundary Condition (Same)
IZM	Number of Zones
IM	Number of Intervals
IGM	Number of Groups
MTP	Number of P_o cross section sets of the read in. (Note - this is the number of elements in blanket)
IIM	Inner Iteration Maximum (per Group)
ID1	Print Angular Flux (O/1=No/Yes)
ID2	Number of Cross Sections per Element to Print
ID3	Print currents (O/1= No/Yes)

Card(s) 4: FORMAT (12I6)

Cross section record numbers from TREVE output - (ISCT+1)*(MTP) entries required.

These numbers identify the cross section sets to be read in. The numbers must be in increasing order. One number per P expansion per element must be entered.

Card 5: FORMAT (20A4)

Title Card Identifying Element

Card(s) 6: FORMAT (6E12.6)

Density by Zone for Element; IZM entries

REPEAT CARD 5to6 FOR EACH ELEMENT

Card 8: FORMAT (6E12.6)

Control Parameters

EPS	Overall Convergence Precision
EPSA	Pointwise Convergence Criterion (Not Used if Zero)
BF	Buckling Factor (Normally $\pi/3$)
DY	Buckling Total Height, cm (Slabs, Cylinders)
DZ	Buckling Total Depth, cm (Slabs Only)
XNF	Normalization Factor (Normally Unity, Not Used if Zero)

SECTION B DATA

All section B data are read by the subroutines REAI (integers) or REAG (floating point). The subroutines use a 6 (I1, I2, I9) or a 6 (I1, I2, E9.4) format respectively.

In REAI and REAG, if I1 = 0, only the word in the nine-digit format is read; if I1 = 1, the nine digit word is repeated I2 times; if I1 = 2, I2 linear interpolants are placed between the nine digit word and the next such word; and if I1 = 3, reading is terminated. Thus, all data read by REAG and REAI must be followed by a three in the first column after the last word.

Card(s) 9: FORMAT 6(I1,I2,E9.4)

Radii

(IM+1) entries - begin with zero.

Card(s) 10: FORMAT 6(I1, I2, E9.4)

Zone Numbers

IZM entries - No. of intervals per zone

Card(s) 11: FORMAT 6(I1, I2, E9.4)

Material Numbers

IZM entries - enter the numbers 1,2,3,4,···· (IZM-1), IZM

For those zones in which a anisotropic scattering calculation is desired, the entry would be negative i.e. -1, -2. Normally all numbers are negative.

Card(s) 12: FORMAT 6(I1, I2, E9.4)

Weights

Card(s) 13: FORMAT 6(I1, I2, E9.4)

Directions

Card(s) 14: FORMAT 6(I1, I2, E9.4)

Fluxes

IM * IGM entries - normally all zeros.

Card(s) 15: FORMAT 6(I1, I2, E9.4)

Source

IM * IGM entries - if desired the gamma source calculated by GAMSO may be tape input. If so enter a card with a "-1" in columns 2 to 3 here.

Job Control Language

Unit 1 - Cross section input - TAPE
DCB=(RECFM=VSB,BLKSIZE=1000,BUFNO=1)

Unit 2 - Flux output - TAPE
DCB=(RECFM=VSB,BLKSIZE=1000,BUFNO=1)

Unit 3 = GAMSO input - if not used - DUMMY
DCB=(RECFM=FB,LRECL=80,BLKSIZE=3200)

For each P-expansion, a scratch disk must be provided, beginning with unit 8. For example, a P_2 problem would require 3 disks; one for P_o cross sections, one for P_1 and one for P_2. Thus for this case

```
//G .FT08F001 DD UNIT=SYSDA,SPACE=(20,(21400,1)),
              DCB=(RECFM=VSB,BLKSIZE=7200)
//G .FT09F001 DD UNIT=SYSDA,SPACE=(20,(21400,1)),
              DCB=(RECFM=VSB,BLKSIZE=7200)
//G .FT10F001 DD UNIT=SYSDA,SPACE=(20,(21400,1)),
              DCB=(RECFM=VSB,BLKSIZE=7200)
```

5.3.3 CARDTAPE

The gamma production cross section sets calculated and punched out on cards by POPOP4 and those sets found in the literature, e.g. RITTS set, must be placed on tape prior to use in GAMSO.

Each set of cross sections must be preceded by an ID card supplied by the user. The ID card must be in the following format

```
(9A4, A1, 10A4, A3)
```

Column 1 → 4 must contain dots, i.e., ····, and columns 38 → 41 must contain a right adjusted ID number. The rest of the space in the card may be used to describe the set.

The cards are placed on tape in a card image format, thus the JCL for the output unit will contain

```
DCB=(RECFM=FB,LRECL=80,BLKSIZE=3200)
```

With the exception of the ID card supplied by the user, the punched output from POPOP4 is in the correct format. It is necessary, however, to remove the extranious ID cards punched by POPOP4 from the cross section set before adding the ID card supplied by the user to the front of the set.

When using cross section sets found in the literature the cross sections are keypunched onto cards in the FIDO format, i.e., 6(3X, E9.5).

Beginning with neutron energy group 1, gamma energy group 1; the cross sections are keypunched until all neutron energy groups for gamma group 1 have been entered. These are followed by gamma energy group 2. The process is repeated until the set is complete. Do not begin a new gamma group on a new card.

Utility programs may be used to place the cross sections on tape. The following program may also be used.

```
Program CARDTAPE
      DIMENSION  CARD (80)
  3   READ (5, 1, END = 2) CARD
  1   FORMAT (80 A1)
      WRITE (1, 1) CARD
      GO TO 3
  2   END FILE 1
      STOP
      END
```

Unit 1 is the output tape. All cross section sets are in the same file, as they must be.

The ID numbers given to the cross section sets are at the discussion of the user.

Note that it is quite easy to modify this program to meet the needs of the user, for example, to print out the ID card or to update the library. To update the

library one would read the cross sections from the tape into core, read in the new sets and output them onto tape.

It is also not necessary that the cross section sets all have the same energy group structure or that the tape contain only those elements that are used in one calculation as GAMSO will read only those sets requested. Thus a library may be made which contains many elements and energy group structures. The only limitation being that each set have a different ID number.

5.3.4 GAMSO (GAMMA SOURCE)

Within the blanket, a space and energy dependent gamma source is produced from neutron interactions. These gamma sources arise from several processes, for example: from neutron inelastic scattering and neutron reactions such as (n,p), (n,α) and radiative capture (n,γ).

GAMSO calculates in finite form the gamma source matrix, S_{i_j}, for input into a transport code for a gamma transport calculation. S_{i_j} is the gamma source in gammas/cm^3-sec at space point i and gamma energy group j. The subscript, i, varies over all space points in the blanket, and the subscript, j, varies over all gamma energy groups used in the gamma transport calculation.

Input into the code are the scalar neutron fluxes from a neutron transport calculation, the gamma production cross sections from POPOP4 or literature sources, and parameters which describe the blanket.

Each point in the source matrix, S_{i_j}, is calculated according to the following relationship:

$$S_{ij} = \sum_{K=1}^{\text{No. ELEMENTS}} \sum_{\ell=1}^{\text{No. NEUTRON ENERGY GROUPS}} \rho_{ki} \phi_{\ell,i} \sigma_{k,\ell \rightarrow j} \quad (31)$$

where:

$\rho_{k,i}$ is the number density of element K at space point i,

$\phi_{\ell,i}$ is the scalar neutron flux in neutron energy group ℓ at space point i,

$\sigma_{k,\ell \rightarrow j}$ is the gamma production cross section for element k which produces gamma rays in gamma energy group j from a reaction with neutrons in neutron energy group ℓ.

Input Description

SECTION A - Title and Control Parameters

Card 1: FORMAT (20A4)

Title

Card 2: FORMAT (11I4)

IM - No. Intervals
IGM - No. of Neutron Groups
IZM - No. of Zones
NE - No. of Elements in the Blanket
NZS - Zone No. Blanket Starts
NPF - Flux Print Option (O/N)
O - Do not print flux
N - Print flux at intervals of N
NGG - No. of Gamma Groups
NPSP - Cross section print Option (O/N)
O - Do not print cross sections
N - Print cross sections for gamma energy group intervals of N

NPS - Source Print Option (-N/O/M)
-N - Print source by element and total at space intervals of |-N|
O - Do not print source
M - Print total source at space intervals of M

NPAS - Punch Source Option (O/1)
O - Do not punch source
1 - punch source

NTC - Flux Source (-1/O/1/5)
-1 - DTF MF
O - DTF IV
1 - ANISN
5 - CARDS

SECTION B - Flux on Cards

Card(s) 3: FIDO FORMAT

Enter only if NTC = 5
Enter a title card followed by fluxes punched out by ANISN in FIDO format.

SECTION C - Zone Definitions

Card(s) 4: FORMAT (18I4)

NZONE(J) - IZM entries - space interval each zone starts. Note: NZONE(1) = 1

SECTION D - Element Title Card and Density

Card 5: FORMAT (A4, 17A4)

TITLE(I) The first four spaces of the title card contains the ID number of the element's gamma production cross sections on tape. The remainder 17A4 spaces is for text to describe the element.

Card(s) 6: FORMAT (6E12.6)

DENSITY (I) IZM-entries

Element density for each zone.

REPEAT SECTION D DATA FOR EACH ELEMENT -

THAT IS NE TIMES

Job Control Language

A. The output scalar fluxes on tape from DTF-IV, DTF MF, or ANISN may be used as input to GAMSO. They are all read on unit 1.

The fluxes from DTF MF are unformatted, thus for unit 1 when these fluxes are used:

DCB=(RECFM=VSB,BLKSIZE=7200)

The fluxes from DTF-IV and ANISN are formatted. Thus for unit 1, when these fluxes are used:

DCB=(RECFM=FB,LRECL=80,BLKSIZE=3200)

Punched fluxes from ANISN may also be used.

B. The program as written is designed to read the gamma production cross sections from tape. They are read on unit 2. The cross sections are formatted, thus for unit 2:

DCB=(RECFM=FB,LRECL=80,BLKSIZE=3200)

The cross section tape may contain more cross section sets than is used in any one problem. However, the sets must be read in the order in which they appear on tape. The program will skip over those sets not requested.

C. The gamma source matrix is output on tape on unit 7. It is formatted output, therefore for unit 7,

DCB=(RECFM=FB,LRECL=80,BLKSIZE=3200)

Comments:

A. Preparation of the cross section tape is described in the program CARDTAPE, section 5.3.3.

B. Normally, each cross section set will have all reactions, which produce gamma rays, summed for a given element. It it occurs that for a given element two or more cross section sets are used, then the control parameter, NE,must be the number of elements in the blanket. For each set TITLE(I) and DENSITY(I) must be input.

C. The output tape may be read directly by the modified version of the ANISN and DTF-IV code. It may also be read by DTF MF.

If a modified version of the ANISN code is not available, unit 7 may be defined as system punch and the matrix output on cards. These cards may be read by ANISN. DTF IV or DTF MF cannot read these cards unless a small modification is made in the programs.

D. GAMSO is variably dimensioned. If more or less storage space is required, change the dimension of STORE in the common statement in the main program. Set I55 equal to the new dimension.

5.3.5 GLUCKE

GLUCKE is a code which calculates reaction rates and the total number of reactions within a blanket. The reaction cross sections and scalar neutron or gamma fluxes are input into the code.

A reaction rate, R, is any process which may be described by the following relationship:

$$R = \phi \sigma \varrho \tag{32}$$

where: ϕ is a scalar flux,
σ is a reaction cross section, and
ϱ is a number density.

The reaction rates normally have the units of reactions/cm^3-sec. Some of the processes which may be described by reaction rates are:

1. Heating rates in watts/cm^3 if the reaction cross sections are kerma factors.
2. Tritium production rates in ^{T} atoms/cm^3-sec if the reaction cross sections are L,^{6}(n,α)T and L,^{7}(n,nα)T cross sections.
3. Atom displacement rates in dpa/cm^3-sec if the reaction cross sections are displacement cross sections.

The calculation of other reaction rates is possible.

The total number of reactions per unit time, $\bar{R}$, for a given volume is:

$$\bar{R} = RV \tag{33}$$

where R is the reaction rate and
V is the given volume.

If, for example, V is the volume of a fusion reactor blanket, then $\bar{R}$ would be the reactions/sec in the blanket.

If $\bar{R}$ is divided by the source strength S (the number of neutrons/sec from the D-T reaction), then the number of reactions per source neutron is obtained.

Note that if $\bar{R}$ is the number of tritium atoms produced per second then $\bar{R}/S$ is the tritium breeding ratio.

GLUCKE will calculate the reaction rate at each space point for each element according to the following ship

$$R_{iK} = \sum_{j=1}^{\text{No.ENERGY GP.}} \phi_{ji} \sigma_{Kj} \varrho_{K\,i} \tag{34}$$

where R_{iK} is the reaction rate for element K at space point i,

ϕ_{ji} is the scalar flux at space point i in energy group j,

σ_{Kj} is the reaction cross section for element K in energy group j, and

ϱ_{Ki} is the number density of element K at space point i.

GLUCKE also calculates the reaction per source neutron at each space point, i.e.

$$R^{s}_{iK} = R_{iK}/S \tag{35}$$

where R^{s}_{iK} is the reaction rate per source neutron at space point i and for element K

R_{iK} is the reaction rate at space point i and for element K

S is the neutron source strength.

GLUCKE then calculates the total number of reactions in each interval volume for each element:

$$\bar{R}_{iK} = R_{iK}\, V_i \tag{36}$$

where $\bar{R}_{iK}$ is the number of reactions in the interval i for element K

R_{iK} is the reaction rate in interval i for element K, and

V_i is the volume of interval i.

The total number of reactions in each interval is divided by the source strength to calculate the number of reactions per unit source:

$$\bar{R}^{s}_{iK} = \bar{R}_{iK}/S \tag{37}$$

where $\bar{R}^{s}_{iK}$ is the number of reactions per unit source in volume element i for element K,

$\bar{R}_{iK}$ is the number of reactions per volume element i for element K and

S is the neutron source strength.

For each element the terms $\bar{R}^{s}_{iK}$ and $\bar{R}_{iK}$ are summed over the zones and over all the blanket. This results in the total number of reactions for element K in each zone and in the blanket and the number of reactions per unit source in each zone and the blanket for each element

that is $$_{z}\bar{R}_{K} = \sum_{\substack{\text{all i} \\ \text{in ZONE} \\ Z}} \bar{R}_{iK} \tag{38}$$

and $$\bar{R}_{K} = \sum_{\substack{\text{all Z} \\ \text{in the} \\ \text{blanket}}} {}_{Z}\bar{R}_{K} \tag{39}$$

where $_{z}\bar{R}_{K}$ is the total number of reactions for element K in zone Z

$\bar{R}_{K}$ is the total number of reactions in the blanket for element K, and

$\bar{R}_{iK}$ is the total number of reactions in volume element i for element K.

Likewise:

$$ {}_z\bar{R}^s_K = \sum_{\substack{i \\ \text{in Zone } z}} \bar{R}^s_{Ki} \tag{40} $$

$$ \bar{R}^s_K = \sum_{\substack{\text{all } z \\ \text{in the} \\ \text{blanket}}} {}_z\bar{R}^s_K \tag{41} $$

where ${}_z\bar{R}^s_K$ is the total reactions per unit source in zone z and for element K

$\bar{R}^s_K$ is the total reactions per unit source in the blanket for element K

$\bar{R}^s_{Ki}$ is the total reactions per unit source for element K in volume element i.

To calculate the total reactions in each volume element, zone and blanket $\bar{R}_{Ki}$, ${}_z\bar{R}_K$ and $\bar{R}_K$ is summed over all elements in the blanket.

$$ \bar{R}_i = \sum_{\text{all } K} \bar{R}_{Ki} \tag{42} $$

$$ {}_z\bar{R} = \sum_{\text{all } K} {}_z\bar{R}_K \tag{43} $$

$$ \bar{R} = \sum_{\text{all } K} \bar{R}_K \tag{44} $$

where $\bar{R}_i$ is the number of reactions in volume element i for all elements

${}_z\bar{R}$ is the number of reactions in zone z for all elements

$\bar{R}$ is the total number of reactions in the blanket for all elements.

The terms $\bar{R}_i$, $_z\bar{R}$, and $\bar{R}$ may be divided by the neutron source strength to calculate the number of reactions in volume interval i, zone z and in the total blanket.

Note that $\bar{R}/S$ would be the breeding ratio if the reaction cross sections are the $L,^6(n,\alpha)T$ and $L,^7(n,n\alpha)T$ cross sections. $\bar{R}/S$ would be the heat released per fusion neutron if the reaction cross sections are kerma factors.

The volume elements V_i are calculated by the following relationship:

For cylindrical geomtry

$$V_i = (r_{i+1}^2 - r_i^2)\pi \tag{45}$$

For plane geometry

$$V_i = (r_{i+1} - r_i) \tag{46}$$

where r_{i+1} and r_i are the boundaries of interval i.

Note then that the source strength, S, must be the source strength of one cm length of blanket. This value is output by the transport code.

The total number of reactions calculated is for one cm length of blanket.

The code also calculates the breeding ratio due to branching if the value of the D-D source strength is entered. It would only be entered if breeding ratios are being calculated. Normally it is insignificant.

Most of the results calculated by GLUCKE can be moved to magnetic tape for further disposition. Some details of this procedure are summarized in section 5.3.9.

Input Description

SECTION A - Title and Control Parameters

Card 0: FORMAT(2I6)

IDEX,IDRT (see comment D.)

Card 1: FORMAT (18A4)

Title Card

Card 2: FORMAT (8I6, 2E12.6)

NE - Number of elements for which reaction rates are calculated

NEG - Number of energy groups

NIN - Number of intervals

NZ - Number of zones

NTC - Origin of flux (-1/0/1/5)

-1 - DT FMF (unit 1)

0 - DTF IV (unit 1)

1 - ANISN (unit 1)

5 - CARDS (unit 5)

JG1 - Reaction rates are calculated

JG1 - for groups JG1 through JG2 (normally: JG1 = 1, JG2 = NEG)

IE - Geometry (1/2)

1 - Plane

2 - Cylinder

DTRR - Total source

(Normally: total 14 MeV Neutron Source, n/sec)

DDRR - Total 2.5 MeV neutron source form D-D reaction, n/sec. Not used if zero entered.

SECTION B - Intervals per Zone and Radii

Card(s) 3: FORMAT (20I4)

NIZ(I) - Intervals per zone; NZ entries. This array contains the no. of intervals per zone starting with zone 1.

Note: $\sum_{I=1}^{NZ} NIZ(I) = NIN$

Card(s) 4: FORMAT (6E12.6)

R(I) - Radii - (NIN+1) entries
Radii boundaries of intervals
Note: R(1) = 0.0
These radii may be transferred from ANISN or DTF-IV input.

SECTION C - Element Title Card, Density, and Reaction Cross Sections

Card 5: FORMAT (A4, 17A4)

TITLE(I) - The first four spaces of this title card is checked for an ID number. If they are blank, the reaction cross sections are assumed to follow the densities. If they contain an ID No., the reaction cross sections are assumed to be on tape, on unit 2; the tape is searched until the reaction cross section ID No. is found.

The remaining 17A4 spaces of the title card is used for text to describe the cross sections and element.

Card(s) 6: FORMAT (6E12.6)

DEN(I) - Element density per zone NZ entries.
This array contains the density in each zone of the element described in the title card, beginning with zone 1.

Card(s) 7: FORMAT (6E12.6)

XSEC(I) - Reaction cross sections on cards
NEG entries.
Enter only if title (1) is blank.
This array contains the reaction cross section described in title.

NEG entries must be made even if JG1≠1 and/or JG2≠NEG.

REPEAT SECTION C DATA FOR EACH ELEMENT, THAT IS, NE TIMES.

SECTION D - Fluxes on Cards

Card 8: FORMAT (20A4)

TITLEQ(I) - Title card for fluxes

Card(s) 9: FORMAT (6(3X,F9.0))

FLUX(I,J) - Neutron or gamma fluxes.
Enter by interval
Beginning with interval 1 of energy group 1. When nine entries have been made, start with new card for the next energy group.
Note: This is the format used when the fluxes are punched by ANISN.

Job Control Parameters

A. Input fluxes on tape have to be placed on logical unit 1, fixed blocked.

B. Input reaction cross sections on tape have to be placed on logical unit 2, fixed blocked.

C. GLUCKE also provides the possibility to move the final results on tape for further preparation. For this procedure logical unit 3 has been defined. Use here:

DCB=(RECFM=VBS,LRECL=3600)

Comments:

A. Reaction cross sections may be read from both tape and cards in any calculation.

B. The ID numbers of the reaction cross sections must be in the order in which they are found on tape.

C. By changing the dimension of ST in the main program the storage available will be changed.
SET IST = Dimension of ST.

D. Definition of IDEX and IDRT:
These identification numbers have been introduced to identify the data sets written on tape by GLUCKE. They are not used by the program itself. The definition should be as follows:

IDEX : Problem identification number.
It specifies the blanket arrangement calculated.

IDRT : Reaction type identification number.
It specifies the kind of data in the data set. At present four numbers have been specified:

= 1 Heating rates from neutron kerma factors
= 2 Heating rates from gamma kerma factors
= 3 Tritium production rates
= 4 Other reactions rates

E. A modification has been made to the code to aid in transfer of cross sections from tape to the AMOS system installed at the IPP computer device.

If the TITLE(I) card has four pluses (++++) in columns 1 → 4 the cross sections will be read on unit 5, however, each set must still have an ID card with the ID number in columns 38 → 41.

5.3.6 CROSS

CROSS is used to reduce cross sections from fine to broad group structures. It is especially useful to collapse cross section sets found in the literature. CROSS will only reduce those cross sections that are essentially composed of one dimensional arrays such as kerma factors or gamma production cross section sets. It will not for example reduce scattering matrices.

The boundaries of a new group do not necessarily have to lie on the boundary of an old group. Any new group structure desired may be used. In those cases where the boundaries of the old and new group do not lie on each other, various schemes are used to modify the weighting flux. These schemes are based on ratios of energy differences between the old and new group boundaries.

When the boundaries of a new group fall on some boundary of an old group, the collapsing is straightforward, i.e.:

$$\sigma_\ell = \sum_{\substack{i \\ \text{in} \\ \ell}} \phi_i \sigma_i \Big/ \sum_{\substack{i \\ \text{in} \\ \ell}} \phi_i \qquad (47)$$

where: σ_ℓ is the broad group cross section,
ϕ_i is the weighting function in fine group i, and
σ_i is the fine group cross section in group i.

Any desired number of weighting functions may be input and CROSS will produce a reduced set of cross sections corresponding to each set of weighting functions.

Input Description

SECTION A - Title Card and Control Parameter

Card 1: FORMAT (20A4)

Title Card

Card 2: FORMAT (3I4)

NO - No. of old groups
NN - No. of new groups
NG - No. of arrays

SECTION B - Energy Group Structures

Card(s) 3: FORMAT (6E12.6)

EL(I) - Old energy group boundaries (NO+1)entries

Card(s) 4: FORMAT (6E12.6)

E(I) - New energy group boundaries (NN+1) entries

SECTION C - Cross Sections

Card 5: FORMAT (20A4)

TITLE2(I) - Title card identifying cross sections

Card(s) 6: FORMAT (6(I2, 1X, F9.0)

M(I),XSECT(I,NG) - Cross section array
(NO * NG entries)
Enter NO cross sections
for NG=1 in a F9.0 format.
If NG > 1 start on a new card
after each NO entries.
Note: M(I) is a control parameter
which allows repeats. If two or
more cross section in a row are
equal, set M(I) = to the number
of equal cross sections and enter
the cross section only once. The
M(I) cross sections are counted in the NO

total of each array. It is not necessary to use this option. If not used, M(I) may be left blank.
Punched output from POPOP4 may be read with this program.

SECTION D - Weighting Functions

Card 7: FORMAT 20A4

TITLE - Title card describing weighting functions

Card(s) 8: FORMAT (6E12.6)

FLUX(I) - "NO" entries weighting functions

SECTION D DATA MAY BE REPEATED FOR AS MANY SETS OF WEIGHTING FUNCTIONS AS DESIRED.

Comments:

A. All input to CROSS is by cards and output is also by punched cards on unit 7.

B. When using CROSS with gamma production cross sections note that only the neutron group structure may be changed, but not the gamma group structure. This is illustrated by Figure 8.

C. New energy group boundaries do not necessarily have to lie on a boundary of an old energy group. It is also not necessary that $NO \geq NN$.
However; $EL(1) \geq (E(1)$ and
$EL(NO+1) \leq E(NN+1)$

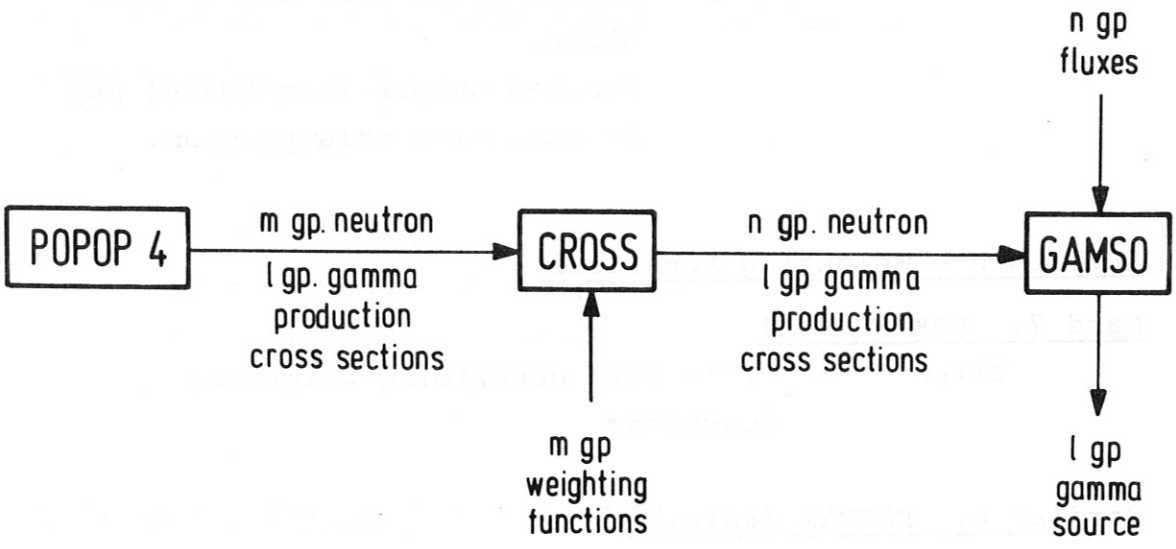


Fig. 8: POPOP4 Used with CROSS

5.3.7 GROUCO

This is a program which transmutes pointwise cross section data to group constants.

If SUPERTOG is applied to create a library for use in neutron transport calculations most of the interesting cross sections for use in GLUCKE are created simultaneously. The number of reactions output by SUPERTOG printout, however, is restricted to the most common reaction types. For instance, not printed are the cross sections of the $^{7}Li(n,n\alpha)T$ reaction which is needed for calculating the tritium breeding ratio.

In this case, or if pointwise data are available from other literature sources (even ENDF/B) GROUCO can be used to create a set of group constants for use in GLUCKE.

Input Description

Card 1: FORMAT (6I12)

NGP - number of energy groups
NX - number of cross section data points
NI - number of interpolation intervals in one energy group
NO - lowest energy group for which cross section is not zero
INT - interpolation scheme (see comment B)
IW - option for weighting function (see comment C)

Card(s) 2: FORMAT (6E12.5)

E - energy group boundaries
NGP+1 entries
only needed if (NGP < 99) which means that the fixed GAM-II energy group structure is not be applied.

Card(s) 3: FORMAT (6E12.5)

EX,SX - cross section data points, characterized by subsequent pairs of energy,cross section data (NS entries).

Card 4: FORMAT (6E12.5)

S(NGP+1) - thermal cross section

Comments:

A. In its present form the program is not applicable to all kinds of cross section data. Limitation exists especially in the number of weighting functions available (see comment C), and in the fact that only one interpolation scheme can be applied to the entire energy range.

B. Five interpolation schemes are available:

INT = 1 : σ constant in E
= 2 : σ linear in E
= 3 : σ linear in ℓnE
= 4 : $\ell n\sigma$ linear in E
= 5 : $\ell n\sigma$ linear in ℓnE

where σ denote the cross section and E the energy.

C. Two built in weighting functions can be used:

IW = 1 - weighting function is 1/E
IW = 2 - weighting function is fission spectrum joined to a 1/E tail

The input of specially prepared weighting functions is not possible in the present version.

5.3.8 FLUZI (FLUX REDUCED BY ZONE FOR INPUT)

FLUZI calculates volume averaged fluxes for cross section weighting or analysis. Input to the code is the group scalar fluxes from a transport calculation and control parameters.

The zone structure chosen for the averages is arbitrary. For example, the entire blanket may be considered as one zone for the purpose of this code. However, the zone structure chosen for averaging would probably correspond to the various blanket structural zones.

The following relationship is used to obtain the volume averaged fluxes:

$$\bar{\phi}_{iK} = \sum_{\substack{j \\ \text{in} \\ K}} (r_{j+1}^2 - r_j^2)\, \phi_{ij} \Big/ \sum_{\substack{j \\ \text{in} \\ K}} (r_{j+1}^2 - r_j^2) \qquad (48)$$

where $\bar{\phi}_{iK}$ is the volume weighted flux in zone K and energy group i;
$\bar{\phi}_{ij}$ is the scalaer flux at space point j for energy group i; and r_j and r_{j+1} are the radius boundaries of space point j.

Note that the relationship is valid only for cylindrical geometries.

In addition FLUZI will sum fluxes over various energy groups and produce flux plots.

An example of FLUZI used with APRRX is shown in Fig. 9.

The fluxes are input on unit 1 in card image format for ANISN and DTF-IV fluxes and unformatted for DTF MF fluxes.

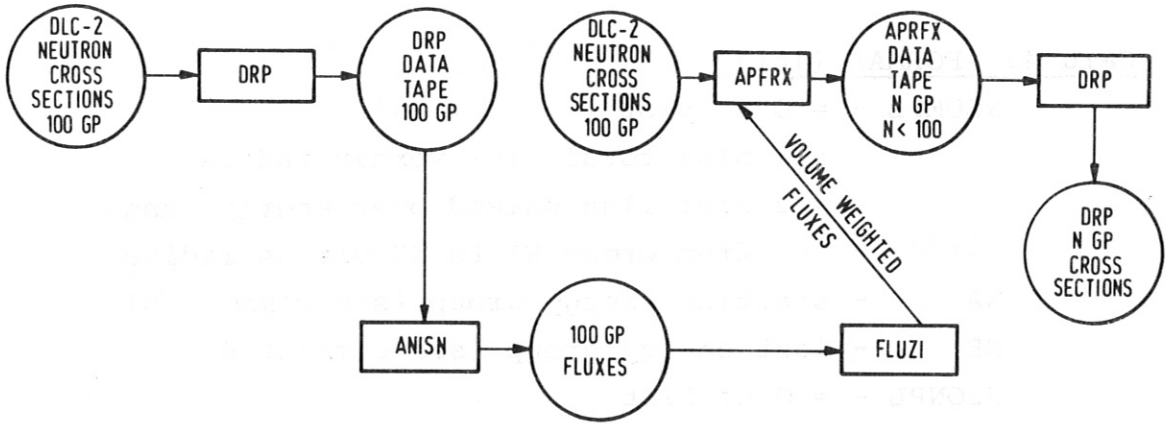


Fig. 9: Use of FLUZI with APRFX

Input Description

Card 1: FORMAT (18A4)

Title card

SECTION A - Plot-Control

Card(s) 2: FORMAT (18I4)

NIPLOT - number of intervals for which flux spectra are to be plotted
= 0: no plot

NIPL - NIPLOT entries
indices of the intervals

Card(s) 3: FORMAT (18I4)

NGPLOT - number of energy groups for which the space dependent group flux is to be plotted
= 0: no plot

NGPL - NGPLOT entries
energy group numbers

Card 4: FORMAT (18I4)

NSUMPL - = 0 no plot
= 1 plot total flux versus radius
= 2 plot flux summed over energy range from group NA to NE versus radius

NA - starting energy group (see comment B)

NE - last energy group (see comment B)

NZONPL - = 0 no plot
= 1 if averaged flux spectrum for each zone is to be plotted

NFLUPL - = 0 no plot
= 1 produce a three-dimensional plot

IA - starting interval, only used if NFLUPL = 1

IE - last interval, only used if NFLUPL = 1

NINTPL - = 0 no plot
= 1 plot total flux versus interval index (see comment C)

SECTION B - Flux Reducing and Control Information

Card 5: FORMAT (18I4)

NG - number of energy groups
NI - number of intervals
NZO - number of old zones
NZN - number of new zones
NTC - origin of flux data
= -1 fluxes from DTFMF
= 0 " " DTF-IV
= 1 " " ANISN
= 5 " " cards
NPC - reduced flux output on cards?
(0/1) = (no/yes)

Card(s) 6: FORMAT (18I4)

N1O - number of intervals per zone for original problem (NZO entries)

Card(s) 7: FORMAT (18I4)

N1N - number of intervals per zone for reduced problem (NZN entries)

Card(s) 8: FORMAT (6E12.5)

R - Radii (NI+1 entries)

Card(s) 9: FORMAT (6E12.5)

E - Energy group boundaries (NG+1 entries)

Comments:

A. The tape containing the input fluxes has to be placed on logical unit 1. This is valid only for NTC = 0 or $\pm$ 1.

B. The option NSUMPL = 2 permits to plot the total flux above a certain energy, for instance $\phi >$ 0.1 MeV ($\rightarrow$ NA = 1; NE = 50 in the GAM-II structure)

C. The option NINTPL = 1 permits to plot the total flux versus the index of intervals irrespective if the actual radii corresponding to these indices. This is a first mean to judge about the appropriate choice of the mesh spacing.

5.3.9. GLOUMAN

In doing systematic blanket investigations with the INDRA program system a lot of final information is produced. This is, partly due to the fact that at least four GLUCKE runs are necessary to evaluate neutron and gamma heating rates, tritium production rates, and, if desired, other reaction rates for one problem.

To store all this information GLUCKE has been provided with an output on tape to which all results are moved which are necessary for further studies. Each data set contains the following information:

Block 1: Titlecard

BPKT	=	'....' identification of titlecard
IDEX		problem identification number
IDRT		reaction type identification number

Block 2: Blanket Description and Geometry

LE	number of elements
NIN	number of intervals
NINP	number of radii
NZ	number of zones
TITELP	description of contents
R	radii
IZ	number of intervals per zone
VI	volumes of intervals
VZ	volumes of zones
VT	total volume

C. The option NINTPL = 1 permits to plot the total flux versus the index of intervals irrespective if the actual radii corresponding to these indices. This is a first mean to judge about the appropriate choice of the mesh spacing.

5.3.9. GLOUMAN

In doing systematic blanket investigations with the INDRA program system a lot of final information is produced. This is, partly due to the fact that at least four GLUCKE runs are necessary to evaluate neutron and gamma heating rates, tritium production rates, and, if desired, other reaction rates for one problem.

To store all this information GLUCKE has been provided with an output on tape to which all results are moved which are necessary for further studies. Each data set contains the following information:

Block 1: Titlecard

BPKT	=	'....' identification of titlecard
IDEX		problem identification number
IDRT		reaction type identification number

Block 2: Blanket Description and Geometry

LE	number of elements
NIN	number of intervals
NINP	number of radii
NZ	number of zones
TITELP	description of contents
R	radii
IZ	number of intervals per zone
VI	volumes of intervals
VZ	volumes of zones
VT	total volume

Block 3: Reaction rates by element

L element sequence number

TITEL description of element

Q(I,L,K) reaction rates for element L and all intervals I

The index K denotes:

K=1 reaction rate (M=1)*

K=2 reaction rate per source neutron (M=2)*

K=3 total number of reactions per interval (M=3)*

K=4 total number of reactions per interval per source neutron (M=4)*

Q56(N,L,K) zone integrated reaction rates for element L and all zones N.

The index K denotes:

K=1 total number of reaction per zone (M=5)*

K=2 total number of reactions per zone per source neutron (M=6)*

Q7(L) total number of reactions for element L in the whole arrangement (M=7)*

Q8(L) total number of reactions per source neutron for element L in the whole arrangement (M=8)*

* For the definition and application of M see description of the option code, paragraph E.

Block 3 is repeated for each element L, i.e. LE times. Behind this the same block is repeated once more containing the sum over all L elements. This block is identified by L = 0.

To write these data sets on tape no special order has to be observed. Each data set can be found by searching for the title card containing the correct identification of the set.

The program GLOUMAN (GLUCKE Output Manipulator) now is able to read this tape and to work upon the data sets in prescribed ways. To do this a set of input instructions to the program has to be defined and to be arranged in a logical sequence.

Each input instruction consists of one single input card containing an alphameric "option code" and a parameter string. The "option code" tells the program what to do. The parameter string tells it to which array this instruction shall be applied.

One single input instruction is of the following format:

FORMAT (A4, 17I4)

It is referred to by the program by a one dimensional array named KENN. Therefore:

KENN(1) = "option code"
KENN(I) = parameter if I $\geq$ 1

At present nine different options are available each of which needs a definite parameter string. The parameters needed are summarized in Table XII.

Description of the "option code"

A. The option "LIST"

The purpose of this option is to present a list of the tape contents. It needs only one parameter, KENN(2),which is either zero or equal to the problem identification number IDEX.

KENN(2) = 0	all title cards are printed including those for each element.
KENN(2) = IDEX	only the title cards for the problem identified by IDEX are printed.

In addition to this also the record numbers are presented. This is of value for the case that any data set must be removed.

B. The option "READ"

The purpose of this option is to read one data set from the tape. Two parameters are needed to identify the data set:

KENN(2) = IDEX	the problem identification number
KENN(3) = IDRT	the reaction type identification number

This option has to preceed any of the further described options.

C. The option "SAVE"

The purpose of this option is to save a part of the data set in an auxiliary storage for further disposition. This option has to be applied if more than one data set shall be handled. It needs only one parameter, KENN(2) which is

KENN(2) = L	the element sequence number within the data set read in.

The information stored corresponds to one of the blocks no. 3. Note: There is no possibility to save more than one block at once even if a second "SAVE" command follows the first one. In this case the second command would overwrite the information saved by the first command.

D. The option "SUM"

The purpose of this option is to sum up equal quantities of different block no. 3 - records within a single data set. The application of this command is indicated, for instance, in the case that equal reaction rates

in different elements shall be added if more than one reaction type was evaluated by a single GLUCKE run (IDRT = 4). The summation is applied to all quantities contained in block 3.

The parameters needed for input are the element sequence numbers L_i. Therefore:

KENN(2) = L_1
KENN(3) = L_2
.
.
.
KENN(N) = L_{N-1}

The result of this summation is stored in an auxiliary array.

E. The option "ADD"

In contrast to the option "SUM" this option permits the addition of equal quantities contained in different data sets. A typical case for application is, for instance, adding gamma heating rates and neutron heating rates to obtain total heating rates.

It is obvious that this option can only be applied to data sets characterized by the same problem identification number IDEX. Additionally, the option works only correctly, if it is preceeded by at least the following three commands:

```
READ   (read the first data set)
SAVE   (store the desired information)
READ   (read the second data set)
ADD    (add to the stored information the
        equivalent information of the second
        data set)
```

Two parameters are needed for input:

KENN(2) = L element sequence number
KENN(3) = M denotes that array of block 2 to which the addition shall be applied.

For convenience the definition of the arrays shall be repeated in detail:

M = 1 reaction rates per interval
M = 2 reaction rates per interval per source neutron
M = 3 total number of reactions per interval
M = 4 total number of reactions per interval per source neutron
M = 5 total number of reactions per zone
M = 6 total number of reactions per zone per source neutron
M = 7 total number of reactions for all zones
M = 8 total number of reactions for all zones per source neutrons

(The array M = 8, for instance, contains the tritium breeding ratio.)

F. The option "INTG"

"INTG" stands for "integrate". The purpose of this option is to perform a step-by-step integration of any block 3 array over the intervals. Actually the program does a summation rather than an integration. This option is, for instance, a useful tool if one wants to follow the growth of tritium breeding with increasing blanket depth.

The parameter string for input is the same as for the option "ADD":

KENN(2) = L element sequence number
KENN(3) = M characterizes the array to which the commanded is to be applied.

The option "INTG" can be used just behind a "READ" command. It can, however, also be applied behind an "ADD" command, but it is not possible to add integrated quantities.

G. The option "PRNT"

This option is used to print information. Three parameters KENN(2-4) permit to select any information desired from one data set to be printed.

As is shown in Table XII, KENN(2) refers to the element sequence number L, KENN(3) to the block-3-array number M, and KENN(4) to the zone number N.

The following variations are possible:

KENN(2) = L print all block-3-arrays for element L (L = 0 means "results for all elements")

< 0 print the block-2 and all block-3 arrays for all elements L.

KENN(3) $\leq$ 0 all block-3-arrays (M = 1 to 8) for the element(s) specified by KENN(2) are printed.

0 < KENN(3) $\leq$ 4 only the arrays M = 1 to M = 4 for the element(s) specified by KENN(2) are printed

KENN(3) $\geq$ 5 only the arrays M = 5 to M = 8 for the element(s) specified by KENN(2) are printed

KENN(4) $\leq$ 0 the total arrays specified by KENN(2) and KENN(3) are printed

KENN(4) = N print output is restricted to zone number N.

TABLE XII

Input Instructions for GLOUMAN

KENN(1)	KENN(2)	KENN(3)	KENN(4)	KENN(5)	KENN(6)
LIST	IDEX				
READ	IDEX	IDRT			
SAVE	L				
SUM	L_1	L_2	L_3		
INTG	L	M			
ADD	L	M			
PRNT	L	M	N		
PLØT	page	L	M	N1 start	N2
PLØG	page	L	M	N1 start	N2

L refers to an element within block 3

M refers to an array within block 3

N refers to a zone

The flexibility in the output selection provided by this option is of use only in the case that the "PRNT" option is requested directly behind a "READ" option. If the "PRNT" option with a zero parameter string (PRNT 0 0 0) is requested just behind one of the operational options, e.g. "ADD" or "INTG", only the results of that operation are printed.

H. The options "PLOT" and "PLOG"

These options offer the possibility for a graphical display of the results. The difference between the two is that "PLOG" provides a logarithmic ordinate scale while with "PLOT" both axes are linearly scaled. The parameter string necessary for input is as follows:

KENN(2)	=	0	plot on a new page
	≠	0	plot on the existing page (the second command can be requested up to 9 times subsequently)
KENN(3)	=	L	element sequence number
KENN(4)	=	M	identification of the block-3-array to be plotted. Only M = 1 to M = 4 is allowed.
	=	0	the results of a precedding operation, e.g. "ADD" or "INTG" are plotted.
KENN(5)	=	N1	N1 and N2 denote the first and last zone for which the plot-output shall be made.
KENN(6)	=	N2	

In every case of graphical display the quantities are plotted versus the "blanket thickness" starting with X = 0 at R = RW, the first wall radius. Each "PLOT" or "PLOG" option requested with KENN(2) = 0 has to be followed by a card containing the text which appears at the bottom of the diagram.

If the "PLOT" or "PLOG" option is requested behind an operational command the results of the operation are plotted. In this case both KENN(3) and KENN(4) have to be zero.

The following input example shall illustrate the capability of the program GLOUMAN:

```
      //G.SYSIN DD *
   1  LIST
   2  READ 103   3
   3  PLOG   0   0   1   0   3
   4  LOG10(Q1,L=0) FUER IDEX/IDRT = 103/3 $
   5  INTG   0   4
   6  PLOT   0   0   0   3   3
   7  INTG(Q4,L=0)  FUER IDEX/IDRT = 103/3
   8  PRNT
   9  READ 103   1
  10  PRNT   0   1   0
  11  PLOG   0   0   1   3   6
  12  LOG10(Q1,L=0) FUER IDEX/IDRTT = 103/1, 103/2 UND "ADD"
  13  SAVE   0
  14  READ 103   2
  15  PRNT   0   1   0
  16  PLOG   1   0   1   3   0
  17  ADD    0   1
  18  PRNT
  19  PLOG   1   0   0   0   6
```

The single lines read as follows:

1) Make a complete list of all data sets present on the tape.
2) Read the data set with problem number (KENN(2) = IDEX =103) and reaction type number (KENN(3) = IDRT = 3) (tritium production).

3) Make a logarithmic plot on a new page (KENN(2) = O) showing the reaction rates [$cm^{-3}s^{-1}$], (KENN(4) = M = 1) summed over all elements (KENN(3) = L = O) (^{6}Li + ^{7}Li), depending on the blanket thickness up to zone 3 (KENN(5) = N1 = O; KENN(6) = N2 = 3).

4) Put this text on the plot page.

5) Integrate the reaction rates per interval and source neutron (KENN(3) = M = 4) summed over all elements (KENN(3) = L = O).

6) Plot in linear scale on the new page (KENN(2) = O) the results of this integration (KENN(3) = KENN(4) = O) only for zone 3 (KENN(5) = N1 = KENN(6) = N2 = 3).

7) Put this text on the new plot page.

8) Print the results of the integration.

9) Now read for the same problem number (KENN(2) = IDEX=103) the neutron heating rates (KENN(3) = IDRT = 1).

10) Print from the summary results (KENN(2) = L = O) the volumetric heating rates (KENN(3) = M = 1) [W/cm^{3}] for all zones (KENN(4) = N = O).

11) Make a logarithmic plot on a new page (KENN(2) = O) of this array (KENN(3) = L = O; KENN(4) = M = 1) including the zones 3 to 6 (KENN(5) = N1 = 3; KENN(6) = N2 = 6).

12) Put this text on the new plot page.

13) Save the summary results (KENN(2) = L = O) for further disposition.

14) Now read for the same problem (KENN(2) = IDEX = 103) the gamma heating rates (KENN(3) = IDRT = 2).

15) Print again from the summary results (KENN(2) = L = O) the volumetric heating rates (KENN(3) = M = 1) for all zones (KENN(4) = N = O).

16) Plot on the last page just these results (KENN(3) = L = O, KENN(4) = M = 1), but only those of zone 3 (KENN(5) = N1 = 3; KENN(6) = O).

17) Now add the volumetric heating rates (KENN(3) = M = 1) of the summary table (KENN(2) = L = 0) interval by interval to the equivalent array saved above.

18) Print the results of this operation (total heating rates).

19) Plot these results (KENN(3) = KENN(4) = 0) on the last page (KENN(2) = 1) for all 6 zones (KENN(5) = N1 = 0; KENN(6) = N2 = 6).

ACKNOWLEDGEMENTS

The authors are very much indebted to Prof. W.H. Köhler (Interatom) who had stimulated this work some years ago. They gratefully acknowledge the useful discussions and practical assistance received from H. Krug and W.J. Weber (Labor für Reaktorregelung und Anlagensicherung, Garching) during the entire period of work. They further want to express their gratitude to J. Cox and E. Fisser (IPP) for their support during the phase of implementing and testing the codes on the IPP computer device. Finally they would like to thank Miss J. Pfister for the excellent preparation of the pictures and Miss R. Schubert for her patience in typing the manuscript.

REFERENCES

[1] D.J. Rose and M. Clark, Plasmas and Controlled Fusion, MIT Press, Combridge, Mass., (1961).

[2] D. Steiner, "Neutronics Calculations and Cost Estimates for Fusion Reactor Blanket Assemblies," ORNL-TM-2360 (1968).

[3] H. Borgwaldt, W. Köhler, and Schroeter, "Neutronic and Thermal Design Aspects of Thermonuclear Fusion Reactor Blankets, " Plasma Physics and Controlled Nuclear Fusion Research, IAES, Vienna, pp. 457-468, (1969).

[4] R.G. Mills, "Synchrotron Radiation from Fusion Reactors",MATT-658,(1969).

[5] J.R. Knight, et al., "MUG, A Program for Generating Multigroup Photon Cross Sections," CTC-17 (1970).

[6] J.D. Lee,"Tritium Breeding and Energy Generation in Liquid Lithium Blankets", Proceedings of Culham Conference on Nuclear Fusion Reactors, pp. 471-482, (1969).

[7] J.J. Ritts, et al., "Calculation of Neutron Fluence-to-Kerma Factors for the Human Body", Nuclear Applications and Technology, 7, 1, pp. 89-99,(1969).

[8] M. Solomito, et al., "AVKER Neutron Kerma Response Function Data and Retrieval Program", DLC-10, (1969).

[9] D. Steiner, Analysis of a Bench-Mark Calculation of Tritium Breeding in a Fusion Reactor Blanket: The United States Contribution. ORNL-TM-4177 (1973).

[10] G.J. Ritts, M. Solomito, D. Steiner, Kerma Factors and Secondary Gamma-Ray Sources for some Elements of Interest in Thermonuclear Blanket Assemblies. ORNL-TM-2564 (1970).

[11] W.E. Ford III, D.H. Wallace, POPOP4: A Code for Converting Gamma-Ray Spectra to Secondary Gamma-Ray Production Cross Sections. CTC-12 (1969).

[12] W.E. Ford III, D.H. Wallace: The Use and "Testing" of Al, Fe, Ni, Cu, and Pb Secondary Gamma-Ray Production Data Sets from the POPOP4 Library. CTC-20 (1970).

[13] RSIC : POPLIB: A Compendinon of Neutron-Induced Secondary Gamma-Ray Yield and Cross Section Data. DLC-12 (1970).

[14] W.E. Ford III, The POPOP4 Library of Neutron-Induced Secondary Gamma-Ray Yield and Cross Section Data. CTC-42 (1970).

[15] RSIC : HPICE: Evaluated Photon Interaction Library, ENDF/B File 23 Format. DLC-7 (1971).

[16] RSIC : 100 G: 100-Group Neutron Cross Section Data Based on ENDF/B. DLC-2 (1972).

[17] RSIC : 99 G: 99-Group Neutron Cross Section Data Based on ENDF/B. DLC-2 (1971).

[18] R.Q. Wright, N.M. Greene, J.L. Lucins, C.W. Craven, jr., "SUPERTOG, A Program to Generate Fine Group Constants and P_N Scattering Matrices from ENDF/B", ORNL-TM-2679 (1969).

[19] H.C. Honeck, "ENDF/B Specifications for an Evaluated Nuclear Data File for Reactor Applications", BNL 50066 (1966/67).

[20] P.S. Pickard, Neutron Cross Section Collapsing Code - APRFX-I. AMXRD-BNL (12-70) (1970).

[21] ENEA-Computer Programme Library: CPL ANISN Package (received 1974).

[22] W.W. Engle, A Users Manual for ANISN, a one Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering. K-1693 (1967).

[23] Westinghouse Astronuclear Laboratory, One-Dimensional, Discrete Ordinates Transport Technique - a Description of the ANISN-W Code. WANL-PR-(LL)-034, Vol. 4 (1970).

[24] K.D. Lathrop, Discrete Ordinates Methods for the Numerical Solution of the Transport Equation. Reactor Technology, Vol. 15 (1972).

[25] B.G. Carlson, DTF Users Manual, Vol. 1 and Vol. 2, UNC Phys./MATH 3321 (1963/64).

[26] W.W. Engle et al., DTF-II, a One-Dimensional Neutron Transport Program. NAA-SR-10951 (1966).

[27] K.D. Lathrop: DTF-IV, a FORTRAN Program for Solving the Multigroup Transport Equation with Anisotropic Scattering . LA-3373 (1965).

[28] J. Celnik et al., Gamma Spectral Data for Shielding and Heating Calculations. NASA-CR-54794.

[29] W.E. Ford et al., The Testing of ^{238}U Secondary Gamma-Ray Production Data Sets from the POPOP4 Library. ORNL-TM-3482.

[30] RSIC: CASK-40 Group Coupled Neutron and Gamma-Ray Cross Section Data. DLC-23.

[31] RSIC: AMPX01 - 126 Group Coupled Neutron and Gamma-Ray Transport Cross Section Data Generated by AMPX01. DLC-27.

[32] RSIC: Coupled Neutron and Gamma-Ray Cross Section Sets for Fusion Reactor Calculations. DLC-28; ORNL-TM-4277.

[33] M.A. Abdou, C.W. Maynard, R.Q. Wright, MACK - A Computer Program to Calculate Neutron Energy Release Parameters (Fluence-to-Kerma Factors) and Multigroup Neutron Reaction Cross Sections from Nuclear Data in ENDF Format. ORNL-TM-3994 (1973).

[34] M.A. Abdou, R.W. Roussin, MACKLIB-100-Group Neutron Fluence-to-Kerma Factors and Reaction Cross Sections Generated by the MACK Computer Program from Data in ENDF Format. ORNL-TM-3995 (1974).