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LAMP-B: A Fortran Program Set for the Lattice Cell Analysis by Collision Probability Method

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Program Abstract (in NEA DATA BANK format)

1. Name: LAMP-B
2. Computer for which the program is designed and others upon which it is operable: FACOM230-MODEL60/75
3. Nature of physical problem solved: LAMB-B solves an integral transport equation by the collision probability method for many variety of lattice cell geometries: spherical, plane and cylindrical lattice cell; square and hexagonal arrays of pin rods; annular clusters and square clusters. LAMP-B produces homogenized constants for multi and/or few group diffusion theory programs.
4. Method of solution: LAMP-B performs an exact numerical integration to obtain the collision probabilities.
5. Restrictions on the complexity of the problem: Not more than 68 group in the fast group calculation, and not more than 20 regions in the resonance integral calculation.
6. Typical running time: It varies with the number of energy groups and the selection of the geometry.
7. Unusual features of the program: Any or any combination of constituent subprograms can be used so that the partial use of this program is available.
8. Related and auxiliary programs
   SUPERTOG (ORNL-TM-2679) for updating the fast group constants
   GASKET (GA-7417) for thermal scattering law
9. Status:
10. References: LAMP-B: A Fortran Program Set for the Lattice Cell Analysis by Collision Probability Method, JAERI 1259.
11. Machine Requirement: 64K core memory is required. The installation of PDS (Partitioned Data Set) file management is recommended.
12. Programming language used: FACOM230 FORTRAN version D and GPL for PDS file management routine
13. Operating System or monitor under which the program is executed: FACOM 230M-V
14. Any other programming or operating information or restrictions: The subroutine 'READ' which manages to read/write the data from/into PDS file should be replaced by the corresponding routine for other machines.
15. Name and establishment of author: K. Tsuchihashi JAERI TOKAI Establishment

Tokai-mura Ibaraki-ken Japan

16. Material available: Magnetic tape containing source deck, library, and sample problem;

Keywords: Collision Probability, Lattice Cell, Multi-Group, Integral Transport, Cluster, Resonance Integral, Thermal Utilization, Fast Fission Effect, Heterogeneous

LAMP-B: 衝突確率法による格子計算のプログラム

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土 橋 敬 一 郎

1978年11月1日 受理

プログラムセット LAMP-B は衝突確率法によって原子炉の格子計算を行う．用いられる衝突確率は，板状，円柱状，球状の一次元格子，正方，六方のような燃料棒を一本含む格子，更に ATR で使用される円環状クラスタ，軽水炉で使用される正方クラスタのような複合格子を対象として，数値積分により正確に計算される．エネルギーの群構造は，高速中性子領域，共鳴中性子領域及び熱中性子領域のすべて，または，そのいずれかを選んで計算できる．一応高速炉中性子炉及び熱中性子炉用の断面積ライブラリーが用意されている．

このプログラムセットを構成する個々のプログラムの間で転送される情報は一旦ディスク，テープまたはカードに収容されるので個々のプログラムを単独に利用できることは特長の一つである．

このプログラムセットの出力である均質化された断面積は多群拡散プログラムに利用できる．

Contents

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1. General Description

A code LAMP-B is a set of computer programs for reactor lattice cell calculation for a wide range of reactor types by the collision probability method. We call the code as a 'set' because each constituent program can be used as a single program and hence the user is offered the choise of the program for the optional use of the data library, the geometry of the lattice cell, the scheme of the energy group collapsing or the energy range to be solved. In other words the thermal utilization factor, the resonance integral and the fast fission effect can be separately obtained.

As the informations to be used by the succeeding program are written on DISK, the user can keep them for reuse. Most of intermediate informations may be stored in PDS (Partitioned Data Set) which are also controled (stored, deleted, read, updated, ···,) by the CPS System of the FACOM computer. Almost all input data cards are read in a free format which remarkably reduces troubles in card punching.

This set has many options for the geometry of the lattice cell : slab, sphere, circular, square or hexagonal cylinder, two-dimensional square or hexagonal cylinder by PATH ; the assembly with circular arrayed pin rods by CLUP and with square arrayed pin rods by CLUP 77. The collision probabilities for these multi-region lattices are obtained by the numerical integration without any approximation. The PATH and CLUP programs have a option to calculate the directional collision probabilities to obtain the anisotropic diffusion coefficients. These collision probabilities are fed into the program PIJF where the linear equations for the neutron fluxes are solved by the succesive over relaxation technique.

The leakage effect to the spectrum is not taken care by PIJF, and therefore, when required, the dummy absorption DB^2 should be added to the macroscopic cross sections to obtain the few group constants weighted by the fluxes. The output of PIJF should be used only for making the homogenized multi-group cross sections of the lattice cell and then the energy group structure will be collapsed by any of the succeeding processes. To collapse the energy group structure the program EPISPC is prepared which smears the lattice and then solves the epithermal neutron spectrum by using DB^2s as the leakage effect. The one-dimensional multi-group diffusion program TUD works generally for this purpose.

The flux solving programs PIJF and TUD commonly require the macroscopic cross sections and the source distribution in such a format to economize the memory as WDSN[6], the English DNS program. The PIXEDT supplies the cross sections for the thermal energy region. The library for PIXEDT is made up by a English program PIXSE[7] which calculates the scattering matrix from the tabulated scattering law or by using the theoretical models such as the effective width model or the free gas model. The absorption and the fission vector are simultaneously added to the matrix which should have been made by THERMOSEC[9] ; one of the processing programs of ENDF/B format files or by PI2[8] for ENDF/A format files.

The FAXEDT works for the fast and epithermal energy regions as PIXEPT for the thermal. The library for FAXEDT which is compiled by FAXSE, contains GAM-I type data with the tabulation of the self-shielding factors depending on the σ_0, the admixture cross section, and temperature.

The resonance integrals obtained by using RICM[11] may replace the corresponding part in the library in the stage of FAXEDT for the homogeneous problem or in the stage of EPISPC for

the heterogeneous problem. In a general and detailed cell calculation for the thermal reactor lattices, one of the collision probability programs, PATH[2], CLUP[3] or CLUP77[4], should be used three times; for the fast energy region, for the resonance integral and for the thermal energy region. The total cross sections required for this stage are prepared by FAXEDT, PRERI and PIXEDT, respectively. In most cases of the thermal reactor lattices where the heterogeneity in the fast energy region has little effect the use of collision probability may be skipped. The final collision probabilities are fed into PIJF, RICM and PIJF, respectively, to calculate the space dependent spectrum. The one-group constants are available as a part of the edit of PIJF for the few group diffusion calculation. When the more detailed information is required for the fast energy region, the fluxes obtained by PIJF are fed into EPISPC together with the resonance integrals and then the multigroup constants are obtained. For the same requirement for the thermal energy region, the run of PIJF is followed by EDIT which calculates the homogenized multigroup constants.

2. Theory of Collision Probability

2.1 General theory

We shall describe how to calculate the collision probabilities for one-dimensional systems such as infinite slabs, infinite circular cylinders and spheres, and for two-dimensional cylindrical systems. We divide the system into several regions bounded by closed or open surfaces. Each region is assumed to be homogeneous with respect to its nuclear properties but different regions are not necessarily of different materials. In general we define the collision probability that a neutron emitted uniformly and isotropically in the i-th region will have its first collision in the j-th region as

$$P_{ij}=\frac{1}{4\pi V_i}\int_{V_i}d\boldsymbol{r}\int_{V_j}d\boldsymbol{r}'\frac{\Sigma(\boldsymbol{r}')}{R^2}\exp\left\{-\int_0^R\Sigma(s)ds\right\} \tag{2.1}$$

where R denotes the distance from the emitting point to the collision point. We have also another expression of the P_{ij} equivalent to Eq. (2.1):

$$P_{ij}=\frac{1}{4\pi V_i}\int_{V_i}d\boldsymbol{r}\int_{4\pi}d\boldsymbol{\Omega}\int_{R_{j^-}}^{R_{j^+}}dR\Sigma_j\exp\left\{-\int_0^R\Sigma(s)ds\right\}, \tag{2.2}$$

where R_{j^-} and R_{j^+} denote respectively the distances from point $\boldsymbol{r}$ to the inner and outer boundary of the j-th region along the line through the points $\boldsymbol{r}$ and $\boldsymbol{r}'$.

Similarly the directional probability P_{ijk} which is called as the 'Behrens term' by Benoist[12] is given by

$$P_{ijk}=\frac{1}{4\pi V_i}\int_{V_i}d\boldsymbol{r}\int_{4\pi}d\boldsymbol{\Omega}3\Omega_k{}^2\int_{R_{j^-}}^{R_{j^+}}dR\Sigma_j\exp\left\{-\int_0^R\Sigma(S)dS\right\}, \tag{2.3}$$

where k stands for direction, for example, the parallel or perpendicular direction to boundary plane in the case of slab lattices.

A region surrounded by a boundary surface is defined as a *zone*. The several *zones* which may be apart from each other, can be defined as a *region* by assigning a common region number to the *zones*. The *zone* is the fundamental spatial unit in LAMP. Therefore, when we consider the lattice cell system, the collision *region* j appears repeatedly in the neighbouring cells and even in a cell also. The correspondence between *regions* and *zones* is given by assigning the *region* number to each *zone* by the input and the geometrical disposition of the *zones* is set implicitly in the program. This dual configulation is introduced to save the computer time and storage. In the case where a neutron path enters into the *region* j more than once, a sum of such term as shown by Eq. (2.2) or (2.3) is required for calculating the collision probability.

When we apply the isotropic (white) condition on the outer surface S of the cell, the probability that a neutron emitted from the region i crosses S, P_{is}, is required.

$$P_{is}=\frac{1}{4\pi V_i}\int_{V_i}d\boldsymbol{r}\int dS\frac{1}{R_s{}^2}\exp\left\{-\int_0^{R_s}\Sigma(S)dS\right\} \tag{2.4}$$

or as an alternative expression

$$P_{is}=\frac{1}{4\pi V_i}\int_{V_i}d\boldsymbol{r}\int_{4\pi}d\boldsymbol{\Omega}\exp\left\{-\int_0^{R_s}\Sigma(S)dS\right\} \tag{2.5}$$

where R_s is a distance from the emitting point to the surface.

The isotropic boundary condition is frequently used for the lattice cell calculation not only

in the collision probability method but also in the S_n calculation. We shall describe its physical meaning and the application in the collision probability method.

We assume the system without source nor absorption where the neutron flux distribution is uniform and isotropic everywhere. Then we suppose a cell in the entire space surrounded by a surface S (it may be open) is divided into several regions. We consider the balance of the collision rate in the i-th region

$$\Sigma_i \phi V_i = \frac{\phi}{4} S G_i + \sum_{j=1}^{N} P_{ji} \Sigma_j \phi V_j \tag{2.6}$$

where the subscript i denotes that the quantity is assigned to the i-th region; and

Σ_i; The total macroscopic cross section

ϕ; the uniform scalar flux

V_i; the volume

S; the area of the surface

G_i; the probability that a neutron impinging on the surface has its first collision in the i-th region

P_{ji}; the probability that a neutron emitted in the j-th region has its first collision in the i-th region

The term on the left hand side (L. H. S.) denotes the collision rate in the i-th region. The first term on the right hand side (R. H. S.) denotes the contribution from the outside of the surface and the second term the contribution of the emission in each region inside of the surface. Using the reciprocity theorem:

$$\Sigma_i V_i P_{ij} = \Sigma_j V_j P_{ji}, \tag{2.7}$$

and the conservation theorem:

$$\sum_{j=1}^{N} P_{ij} + P_{is} = 1, \tag{2.8}$$

where P_{is} is the probability that a neutron emitted in the i-th region escapes from the outer surface S without suffering any collisions, we have

$$G_i = \frac{4V_i}{S} \Sigma_i P_{is}. \tag{2.9}$$

Then we define G_s as the probability that a neutron impinging from the outer surface into the inside of the surface escapes from the surface without suffering any collisions:

$$G_s = 1 - \sum_{i=1}^{N} G_i \tag{2.10}$$

When the cells are set in an array, we get the collision probabilities for the lattice cell by using the quantities for the isolated cell as follows:

$$P_{ij}(\text{lattice}) = P_{ij}(\text{isolated}) + P_{is}G_j + P_{is}G_sG_j + P_{is}G_s^2G_j + \cdots,$$

which can be rewritten as

$$P_{ij}(\text{lattice}) = P_{ij}(\text{isolated}) + P_{is}\frac{G_j}{1-G_s}. \tag{2.11}$$

2.2 Collision probabilities for slab lattice

In one-dimensional plane geometry shown in **Fig. 2.1**, we have

$$R = \left|\frac{x'-x}{\cos\theta}\right|,$$

$$d\boldsymbol{r} = dx,$$

$$d\boldsymbol{\Omega} = 2\pi \sin\theta \, d\theta.$$

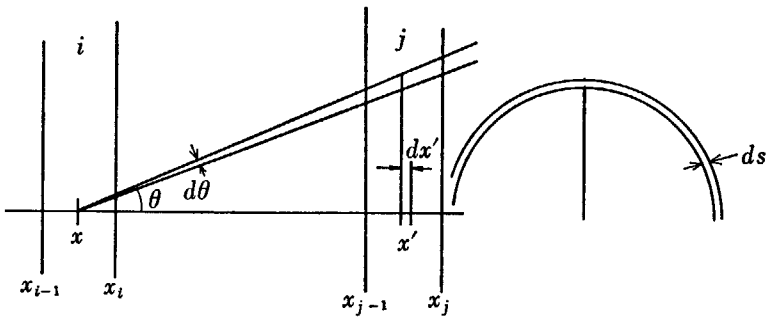


Fig. 2.1 Coordinate in slab geometry.

We assume that the system is divided into an array of slabs. The i-th slab has its left edge at x_i and its total cross section denoted by Σ_i. Then we have the P_{ij}'s as

$$P_{ij}=\frac{1}{2(x_i-x_{i-1})}\int_{x_{i-1}}^{x_i}dx\int_{x_{j-1}}^{x_j}dx'\int_0^{\frac{\pi}{2}}\Sigma(x')\frac{\sin\theta}{\cos\theta}\exp\left\{-\left|\frac{\int_x^{x'}\Sigma(t)dt}{\cos\theta}\right|\right\}d\theta, \tag{2.12}$$

for the case $x_i\leq x_{j-1}$, and the optical distance which appears in the exponential term in Eq. (2.12) is reduced to

$$\int_x^{x'}\Sigma(x'')dx'=\Sigma_i(x_i-x)+\Sigma_j(x'-x_{j-1})+\sum_{k=i+1}^{j-1}\lambda_k,$$

where $\lambda_k=\Sigma_k(x_k-x_{k-1})$.

Then we can carry out the integration over x and x', and we get

$$P_{ij}=\frac{1}{2\lambda_i}\int_0^{\frac{\pi}{2}}\sin\theta\cos\theta\,d\theta\left\{1-\exp\left(-\frac{\lambda_i}{\cos\theta}\right)\right\}*\left\{1-\exp\left(-\frac{\lambda_j}{\cos\theta}\right)\right\}*\exp\left\{-\sum_{k=i+1}^{j=1}\lambda_k/\cos\theta\right\}.$$

Now we introduce the E_{in} function defined by

$$E_{in}(x)=\int_0^1\mu^{n-1}d\mu\exp\left(-\frac{x}{\mu}\right).$$

We have the final form of P_{ij} for the case $x_i<x_j$ as follows

$$P_{ij}=\frac{1}{2\lambda_i}\{E_{i3}(\lambda_{ij})-E_{i3}(\lambda_{ij}+\lambda_i)-E_{i3}(\lambda_{ij}+\lambda_j)+E_{i3}(\lambda_{ij}+\lambda_i+\lambda_j)\}, \tag{2.13}$$

where

$$\lambda_{ij}=\sum_{k=i+1}^{j-1}\lambda_k, \qquad \text{for } x_i<x_j. \tag{2.14}$$

Next we shall consider the case where $x_i>x_j$, the optical distance is reduced to

$$\int_x^{x'}\Sigma(x'')dx''=\Sigma_i(x-x_{i-1})+\Sigma_j(x_j-x')+\sum_{k=j+1}^{i-1}\lambda_k.$$

By using the same procedure as $x_i<x_j$ we get the same expression as Eq. (2.13) except for the definition of λ_{ij}. In this case we have

$$\lambda_{ij}=\sum_{k=j+1}^{i-1}\lambda_k, \qquad \text{for } x_i>x_j. \tag{2.15}$$

In the last case where $x_i=x_j$, the optical distance in Eq. (2.12) is reduced to

$$\left|\int_x^{x'}\Sigma(x'')dx''\right|=\begin{cases}\Sigma_i(x'-x), & \text{for } x'>x.\\ \Sigma_i(x-x'), & \text{for } x'<x.\end{cases}$$

Integrating over x and x' gives the final form of P_{ii} as

$$P_{ii}=1-\frac{1}{\lambda_i}\{E_{i3}(0)-E_{i3}(\lambda_i)\}. \tag{2.16}$$

If the λ_i's are so small that the differences in Eq. (2.13) and in Eq. (2.16) can not be obtained accurately in the numerical calculation, we should use instead of Eqs. (2.13) and (2.16) the following differential forms:

$$P_{ij}=\frac{\lambda_j}{2}E_{i1}\left(\lambda_{ij}+\frac{\lambda_i}{2}+\frac{\lambda_j}{2}\right), \tag{2.17}$$

$$P_{ii}=\frac{\lambda_i}{2}E_{i1}\left(\frac{\lambda_i}{2}\right). \tag{2.18}$$

We, however, should take care of the property of the E_{i1} function which has the logarithmic singularity.

We consider now the lattice cell system where a unit cell is devided into N regions and the collision region j lies periodically. A sum of such terms as Eq. (2.13) gives

$$\begin{aligned}P_{ij}=\frac{1}{2\lambda_i}\sum_{l=0}\{&E_{i3}(\lambda_{ij}{}^{l1})-E_{i3}(\lambda_{ij}{}^{l1}+\lambda_i)-E_{i3}(\lambda_{ij}{}^{l1}+\lambda_j)+E_{i3}(\lambda_{ij}{}^{l1}+\lambda_i+\lambda_j)\\&+E_{i3}(\lambda_{ij}{}^{l2})-E_{i3}(\lambda_{ij}{}^{l2}+\lambda_i)-E_{i3}(\lambda_{ij}{}^{l2}+\lambda_j)+E_{i3}(\lambda_{ij}{}^{l2}+\lambda_i+\lambda_j)\},\end{aligned} \tag{2.19}$$

where

$$\begin{aligned}\lambda_{ij}{}^{l1}&=\sum_{k=i+1}^{j-1}\lambda_k+l*\sum_{k=1}^{N}\lambda_k,\\ \lambda_{ij}{}^{l2}&=\sum_{k=1}^{i-1}\lambda_k+\sum_{k=j+1}^{N}\lambda_k+l*\sum_{k=1}^{N}\lambda_k,\qquad \text{for } i<j.\end{aligned} \tag{2.20}$$

For the case where $j<i$, i and j in Eq. (2.20) must be replaced by j and i, respectively.

For the case $i=j$ we have

$$\begin{aligned}P_{ii}=1-\frac{1}{\lambda_i}&\left\{E_{i3}(0)-E_{i3}(\lambda_i)\right\}\\&+\frac{1}{\lambda_i}\sum_{l=0}\{E_{i3}(\lambda_{ii}{}^{l})-2E_{i3}(\lambda_{ii}{}^{l}+\lambda_i)+E_{i3}(\lambda_{ii}{}^{l}+2\lambda_i)\},\end{aligned} \tag{2.21}$$

where

$$\lambda_{ii}{}^{l}=(l+1)*\sum_{k=1}^{N}\lambda_k-\lambda_i. \tag{2.22}$$

When the lattice cell is arranged symmetrically, we would reduce the computing time into half, but the program is not prepared to take care of the symmetry. Therefore, the user must give the full geometry in input and allocate the same *region* number to the corresponding *zones*.

Now we consider the explicit form of the directional probability. For the perpendicular direction to the boundary plane, we have $3\cos^2\theta$ as $3\Omega_\perp{}^2$, c.f. **Fig. 2.1**, by which the integrand in Eq. (2.12) has to be multiplied. The similar procedure gives us the expression of $P_{ij\perp}$ corresponding to Eq. (2.13) as follows:

$$P_{ij\perp}=\frac{3}{2\lambda_i}\{E_{i5}(\lambda_{ij})-E_{i5}(\lambda_{ij}+\lambda_i)-E_{i5}(\lambda_{ij}+\lambda_j)+E_{i5}(\lambda_{ij}+\lambda_i+\lambda_j)\} \tag{2.23}$$

and corresponding to Eq. (2.16)

$$P_{ii\perp}=1-\frac{3}{\lambda_i}\{E_{i5}(0)-E_{i5}(\lambda_i)\} \tag{2.24}$$

For the parallel direction we can easily obtain the explicit form, but it is not neccessary because the following relation holds,

$$P_{ij}=\frac{1}{3}P_{ij\perp}+\frac{2}{3}P_{ij/\!/} \tag{2.25}$$

The relation comes from

$$1=\sum_k\Omega_k{}^2=\Omega_\perp{}^2+2\Omega_{/\!/}{}^2$$

So we can obtain $P_{ij/\!/}$ by subtraction.

At the end of this section we show the expression for the escape probability P_{is}:

$$P_{is}=\frac{1}{2\lambda_i}\{E_{i3}(\lambda_{is}{}^1)-E_{i3}(\lambda_{is}{}^1+\lambda_i)+E_{i3}(\lambda_{is}{}^2)-E_{i3}(\lambda_{is}{}^2+\lambda_i)\}, \tag{2.26}$$

where

$$\lambda_{is}{}^1=\sum_{k=1}^{i-1}\lambda_k,$$

$$\lambda_{is}{}^2=\sum_{k=i+1}^{N}\lambda_k.$$

2.3 Collision probabilities for annularly cylindrical lattice

We consider the infinitely long cylinder which is divided into several annular shells. The outer radius of the i-th shell is r_i. We suppose that a neutron emitted at the point P in the i-th shell has its first collision at the point Q in the j-th shell. The position of P is defined by only the distance from the cylindrical axis; r. The line PQ makes an angle θ with the vertical line. We define the point Q′ as the projection of the point Q on the horizontal cross section so that the line PQ′ makes an angle β with the line PO. The distance between P and Q′ is R. In the cylindrical coordinate system as shown in **Fig. 2.2 a**, we have the collision probability P_{ij} as,

$$P_{ij}=\frac{2}{V_i}\int_{r_{i-1}}^{r_i}rdr\int_0^{\pi}d\beta\int_0^{\frac{\pi}{2}}\sin\theta d\theta\int_{R_j^-}^{R_j^+}dR\frac{\Sigma_j}{\sin\theta}\exp\left\{-\frac{\int_0^R\Sigma(s)ds}{\sin\theta}\right\}, \tag{2.27}$$

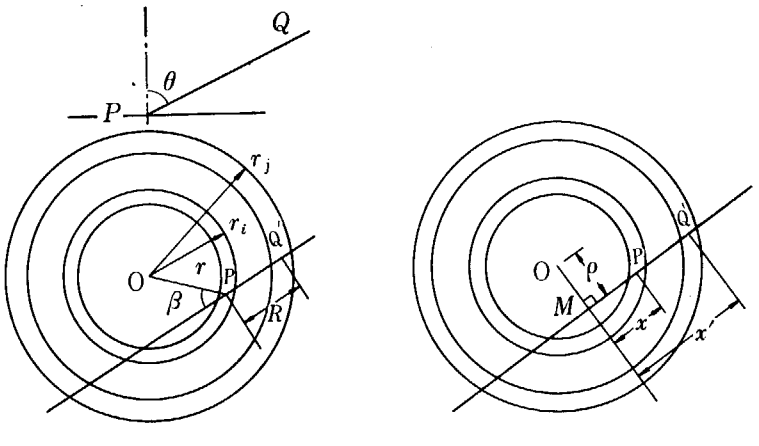
where $$V_i=\pi(r_i{}^2-r_{i-1}{}^2) \tag{2.28}$$

Then we transform the variables r, β, R into new ones ρ, x and x' as illustrated in **Fig. 2.2 b**. We define the perpendicular distance OM from O to the line PQ′ as ρ, the distance between P and M as x, and the distance between Q′ and M as x'. There are three relations among variables:

$$\begin{aligned}&r^2=\rho^2+x^2,\\&r\sin\beta=\rho,\\&R=x'-x.\end{aligned} \tag{2.29}$$

Using the relations we have the Jacobian

$$\frac{\partial(r,\beta,R)}{\partial(\rho,x,x')}=\frac{1}{r}. \tag{2.30}$$



a b

Fig. 2.2 Cylindrical coordinate.

Then we can rewrite Eq. (2.27) by new variables:

$$P_{ij}=\frac{2}{V_i}\int_0^{r_i}d\rho\int_0^{\frac{\pi}{2}}d\theta\int_{x_{i-1}}^{x_i}dx\int_{x_{j-1}}^{x_j}dx'\,\Sigma_j\left[\exp\left\{-\frac{\left|\int_x^{x'}\Sigma(x'')dx''\right|}{\sin\theta}\right\}+\exp\left\{-\frac{\left|\int_{-x}^{x'}\Sigma(x'')dx''\right|}{\sin\theta}\right\}\right], \tag{2.31}$$

where

$$x_i=\sqrt{r_i^2-\rho^2}, \qquad \text{for } r_i>\rho,$$
$$x_i=0, \qquad \text{for } r_i\leqq\rho,$$

and Σ_i denotes the total macroscopic cross section of the i-th shell.

For the case $r_j>r_i$, the optical distances which appears in the exponential terms in Eq. (2.31) are reduced to

$$\int_x^{x'}\Sigma(x'')dx''=\Sigma_i(x_i-x)+\Sigma_j(x'-x_{j-1})+\sum_{k=i+1}^{j-1}\lambda_k,$$
$$\int_{-x}^{x'}\Sigma(x'')dx''=\Sigma_i(x-x_{i-1})+\Sigma_j(x'-x_{j-1})+\sum_{k=1}^{i-1}\lambda_k+\sum_{k=1}^{j-1}\lambda_k,$$

where $\lambda_k=\Sigma_k(x_k-x_{k-1})$.

Then we can carry out the integration over x and x', and we get

$$P_{ij}=\frac{2}{\Sigma_i V_i}\int_0^{r_i}d\rho\int_0^{\frac{\pi}{2}}\sin\theta\,d\theta\left\{1-\exp\left(-\frac{\lambda_i}{\sin\theta}\right)\right\}*$$
$$*\left\{1-\exp\left(-\frac{\lambda_j}{\sin\theta}\right)\right\}*\left\{\exp\left(-\frac{\sum_{k=i+1}^{j-1}\lambda_k}{\sin\theta}\right)+\exp\left(-\frac{2\sum_{k=1}^{i-1}\lambda_k+\lambda_i+\sum_{k=i+1}^{j-1}\lambda_k}{\sin\theta}\right)\right\}.$$

Now we introduce K_{in} function defined by

$$K_{in}(x)=\int_0^{\frac{\pi}{2}}\sin^{n-1}\theta\,d\theta\,\exp\left(-\frac{x}{\sin\theta}\right).$$

We have the final form of P_{ij} for the case $r_i<r_j$, as follows:

$$P_{ij}=\frac{2}{\Sigma_i V_i}\int_0^{r_i}d\rho\,\{K_{i3}(\lambda_{ij1})+K_{i3}(\lambda_{ij2})-K_{i3}(\lambda_{ij1}+\lambda_i)-K_{i3}(\lambda_{ij2}+\lambda_i)$$
$$-K_{i3}(\lambda_{ij1}+\lambda_j)-K_{i3}(\lambda_{ij2}+\lambda_j)+K_{i3}(\lambda_{ij1}+\lambda_i+\lambda_j)+K_{i3}(\lambda_{ij2}+\lambda_i+\lambda_j)\}, \tag{2.33}$$

where

$$\left.\begin{aligned}\lambda_{ij1}&=\sum_{k=i+1}^{j-1}\lambda_k,\\ \lambda_{ij2}&=\sum_{k=1}^{i-1}\lambda_k+\sum_{k=1}^{j-1}\lambda_k,\end{aligned}\right\}\ \text{for } r_i<r_j. \tag{2.34}$$

Next we consider P_{ij} for the case $r_i>r_j$ where the optical distances differ from the case $r_i<r_j$ because Q is located inside of P so that $x'<x$. In this sense the symbol of the absolute value is required. We rewrite locally the optical distance as follows:

$$\left|\int_x^{x'}\Sigma(x'')dx''\right|=\int_{x'}^{x}\Sigma(x'')dx''=\Sigma_j(x_j-x')+\Sigma_i(x-x_{i-1})+\sum_{k=j+1}^{i-1}\lambda_k.$$

Integration over x and x' gives the final form of P_{ij} for the case $r_j<r_i$, which is exactly same expression as in Eq. (2.33) but λ_{ij1} in Eq. (2.34) must be replaced by

$$\lambda_{ij1}=\sum_{k=j+1}^{i-1}\lambda_k, \qquad \text{for } r_i>r_j. \tag{2.35}$$

We have not yet considered the case where the i-th shell coincide with the j-th shell. In this case the optical distances reduced to

$$\left|\int_x^{x'}\Sigma(x'')dx''\right|=\begin{cases}\Sigma_i(x'-x), & \text{for } x'>x,\\ \Sigma_i(x-x'), & \text{for } x'<x,\end{cases}$$

$$\int_{-x}^{x'}\Sigma(x'')dx''=\Sigma_i(x'-x_{i-1})+\Sigma_i(x-x_{i-1})+2\sum_{k=1}^{i-1}\lambda_k$$

In the integration over x' for the first term on R. H. S. of Eq. (2. 31), we must divide the range into (x_{i-1}, x) and (x, x_i) and then we have

$$P_{ii}=\frac{2}{V_i}\int_0^{r_{i-1}}d\rho\int_0^{\frac{\pi}{2}}d\theta\left[2\lambda_i\sin\theta-2\sin^2\theta\left\{1-\exp\left(-\frac{\lambda_i}{\sin\theta}\right)\right\}\right.$$
$$\left.+\left\{1-\exp\left(-\frac{\lambda_i}{\sin\theta}\right)\right\}^2\sin^2\theta\exp\left(-\frac{2\sum_{k=1}^{i-1}\lambda_k}{\sin\theta}\right)\right]$$
$$+\frac{2}{\Sigma_iV_i}\int_{r_{i-1}}^{r_i}d\rho\int_0^{\frac{\pi}{2}}d\theta\left[4\lambda_i\sin\theta-2\sin^2\theta\left\{1-\exp\left(-\frac{2\lambda_i}{\sin\theta}\right)\right\}\right.$$

By using K_{in} function we get the final form of P_{ij} as follows:

$$P_{ii}=\frac{2}{\Sigma_iV_i}\int_0^{r_{i-1}}d\rho[2\lambda_i-2K_{i3}(0)+2K_{i3}(\lambda_i)$$
$$+K_{i3}(\lambda_{ii})-2K_{i3}(\lambda_{ii}+\lambda_i)+K_{i3}(\lambda_{ii}+2\lambda_i]$$
$$+\frac{2}{\Sigma_iV_i}\int_{r_{i-1}}^{r_i}d\rho[4\lambda_i-2K_{i3}(0)+2K_{i3}(2\lambda_i)], \tag{2.36}$$

where

$$\lambda_{ii}=2\sum_{k=1}^{i-1}\lambda_k. \tag{2.37}$$

If the λ_i's are so small that the differences in the brackets of Eq. (2. 33) and (2. 36) can not be obtained accurately in numerical calculation, we should use instead of Eqs. (2. 33) and (2. 36) the following differential forms:

$$P_{ij}=\frac{2}{\Sigma_iV_i}\int_0^{r_i}d\rho\lambda_i\lambda_j\{K_{i1}(\lambda_{ij1})+K_{i1}(\lambda_{ij2})\}, \tag{2.38}$$

$$P_{ii}=\frac{1}{\Sigma_iV_i}\int_0^{r_{i-1}}d\rho\left\{\lambda_i^2K_{i1}\left(\frac{\lambda_i}{2}\right)+\lambda_i^2K_{i1}(\lambda_{ii})\right\}+\frac{4}{\Sigma_iV_i}\int_{r_{i-1}}^{r_i}d\rho\lambda_i^2K_{i1}(\lambda_i) \tag{2.39}$$

If we assume the cylindricalized cell with the perfect reflecting outer boundary, more terms like those in Eq. (2. 35) are required as follows:

$$K_{i3}(\lambda_{ij3})-K_{i3}(\lambda_{ij3}+\lambda_i)-K_{i3}(\lambda_{ij3}+\lambda_j)+K_{i3}(\lambda_{ij3}+\lambda_i+\lambda_j)$$
$$+K_{i3}(\lambda_{ij4})-K_{i3}(\lambda_{ij4}+\lambda_i)-K_{i3}(\lambda_{ij4}+\lambda_j)+K_{i3}(\lambda_{ij4}+\lambda_i+\lambda_j)$$
$$+\cdots\cdots$$
$$+K_{i3}(\lambda_{ijn})-K_{i3}(\lambda_{ijn}+\lambda_i)-K_{i3}(\lambda_{ijn}+\lambda_j)+K_{i3}(\lambda_{ijn}+\lambda_i+\lambda_j)$$
$$+\cdots\cdots$$

where

$$\lambda_{ij3}=\lambda_{ij1}+\lambda_j+2\sum_{k=j+1}^{N}\lambda_k$$
$$\lambda_{ij4}=\lambda_{ij2}+\lambda_j+2\sum_{k=j+1}^{N}\lambda_k$$
$$\lambda_{ijn}=\lambda_{ij(n-4)}+2\sum_{k=1}^{N}\lambda_k$$

As regards the directional probability in the cylindrical coordinate, we know for the axial direction $3\Omega_z^2=3\cos^2\theta$ and for the radial direction $3\Omega_r^2=(3/2)\sin^2\theta$. For the latter P_{ijr} is obtained by multiplying the integrand in Eq. (2. 27) by $3/2\sin^2\theta$. It is tedious and meaningless, however, to write here the whole expressions for each condition. It is enough for us to know only the fact that all the terms expressed by K_{in} function must be replaced by $3/2K_{i(n+2)}$. Similarly to the slab system the following relation holds:

$$P_{ij}=\frac{1}{3}P_{ijz}+\frac{2}{3}P_{ijr} \tag{2.40}$$

We know that the isotropic boundary condition brings more acculate result and is less time-consuming than the perfect reflecting boundary condition to obtain the flux distribution in the real cell by the calculation of the cylindricalized cell. In this case the probabilities, P_{is} that a neutron emitted in the i-th shell escapes from the outer boundary without suffering any collision are required. They are easily obtained as

$$P_{is}=\frac{2}{\Sigma_i V_i}\int_0^{r_i} d\rho\,\{K_{i3}(\lambda_{is1})+K_{i3}(\lambda_{is2})-K_{i3}(\lambda_{is1}+\lambda_i)-K_{i3}(\lambda_{is2}+\lambda_i)\}\,, \tag{2.41}$$

where

$$\left.\begin{aligned}\lambda_{is1}&=\sum_{k=i+1}^{N}\lambda_k,\\ \lambda_{is2}&=\sum_{k=1}^{i-1}\lambda_k+\sum_{k=1}^{N}\lambda_k.\end{aligned}\right\} \tag{2.42}$$

2.4 Collision probabilities for spherical system

A spherical system is divided into N spherical shells. We define the shell i that is bounded by two spherical surfaces of radii r_{i-1} and r_i. The shells are numbered by increasing order of r_i. In general a probability P_{ij} that a neutron emitted in the region i has its first collision in the region j is defined as

$$P_{ij}=\frac{1}{4\pi V_i}\int_{V_i} dV\int_{4\pi} d\Omega\int_{V_j} dR\Sigma_j \exp\left\{-\int_0^R \Sigma(s)ds\right\} \tag{2.43}$$

The integrand on R. H. S. of Eq. (2.43) is interpreted as follows by seeing **Fig. 2.3 a**. A neutron emitted at a point P in the region i moves toward the point Q which is in distance R from point P, has the exponential decay by the optical length $\int_0^R \Sigma(s)ds$ and suffers its collision at the layer of thickness dR in region j of the cross section Σ_j. In the spherically symmetric system the position of the point P is defined only by the distance r from the center C of the system. The position of point Q is defined at the distance R from the point P, and the lines PQ and PC make an angle θ (See **Fig. 2.3 a**). In this coordinate system,

$$dV=4\pi r^2 dr, \qquad 0\angle r\angle R_{\mathrm{MAX}},$$
$$d\Omega=2\pi\sin\theta d\theta, \qquad 0\leqq\theta\leqq\pi,$$

and Eq. (2.43) is rewritten by the triple integral:

$$P_{ij}=\frac{4\pi\Sigma_j}{V_i}\int_0^{R_{\mathrm{MAX}}} r^2dr\int_0^{\pi}\sin\theta d\theta\int_{V_j} dR\exp\left\{-\int_0^R\Sigma(s)ds\right\}. \tag{2.44}$$

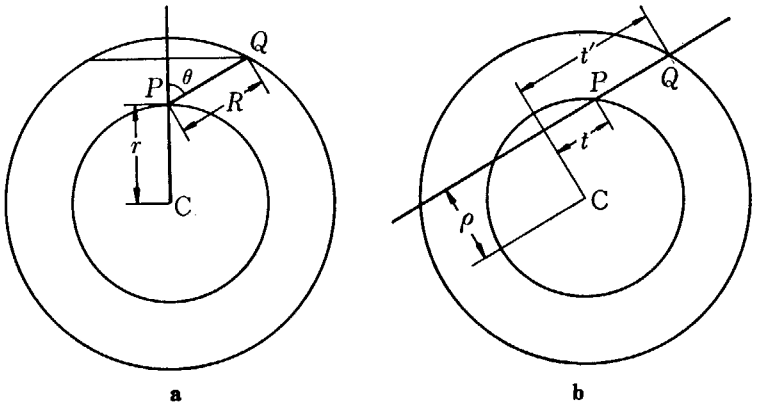


Fig. 2.3 Spherical coordinate.

To perform analytically the integration as far as possible, the coordinate shown in **Fig. 2.3 a** is transformed into the new coordinate shown in **Fig. 2.3 b** where the perpendicular length CM is ρ. The positions of points P and Q are defined by the distances t and t', respectively, from the point M. The following relations among variables are found:

$$\left.\begin{aligned} r^2&=t^2+\rho^2,\\ r\sin\theta&=\rho,\\ R&=t'-t. \end{aligned}\right\} \tag{2.45}$$

The Jacobian is then obtained as follows:

$$\frac{\partial(r,\theta,R)}{\partial(\rho,t,t')}=-\frac{1}{r}. \tag{2.46}$$

The probability is rewritten by the new variables as

$$P_{ij}=\frac{2\pi\Sigma_j}{V_i}\int_0^{R_{MAX}}\rho d\rho\int_{V_i}dt\int_{V_j}dt'\exp\left\{-\int_0^{t-t'}\Sigma(s)ds\right\} \tag{2.47}$$

As the nuclear cross section in each shell is uniform, we can integrate Eq. (2.47) over t and t'. Finally the shell-to-shell collision probabilities are given in the form of single integral (see **Fig. 2.4**):

$$\begin{aligned} P_{ij}=&\frac{2\pi}{\Sigma_i V_i}\int_0^{r_{i-1}}\rho d\rho\{1-\exp(-\lambda_i)\}\{1-\exp(-\lambda_j)\}\\ &*\left[\exp\left\{-\sum_{k=i+1}^{j-1}\lambda_k\right\}+\exp\left\{-2\sum_{k=1}^{i-1}\lambda_k-\lambda_i-\sum_{k=i+1}^{j-1}\lambda_k\right\}\right]\\ &+\frac{2\pi}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i}\rho d\rho\{1-\exp(-2\lambda_i)\}\{1-\exp(-\lambda_j)\}\exp\left\{-\sum_{k=i+1}^{j-1}\lambda_k\right\}, \qquad (2.48)\\ &\text{for } i<j, \end{aligned}$$

$$\begin{aligned} P_{ii}=&\frac{2\pi}{\Sigma_i V_i}\int_0^{r_{i-1}}\rho d\rho\left[\{\lambda_i-1+\exp(-\lambda_i)\}+\{1-\exp(-\lambda_i)\}^2*\exp\left(-2\sum_{k=1}^{i-1}\lambda_k\right)\right]\\ &+\frac{2\pi}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i}\rho d\rho\{2\lambda_i-1+\exp(-2\lambda_i)\}, \qquad (2.49)\\ &\text{for } i=j, \end{aligned}$$

where

$$\left.\begin{aligned} t_i&=\sqrt{r_i^2-\rho^2},\quad r_i\geqq\rho,\\ t_i&=0,\qquad\quad r_i<\rho, \end{aligned}\right\} \tag{2.50}$$

$$\lambda_i=\Sigma_i(t_i-t_{i-1}). \tag{2.51}$$

For $i>j$, the similar expression as Eq. (2.48) can easily be obtained but the reciprocity theorem gives P_{ij} directly from P_{ji}.

Now we have the escape probability P_{is} as

$$P_{is}=\frac{2\pi}{\Sigma_i V_i}\int_0^{r_{i-1}}\rho d\rho\{1-\exp(-\lambda_i)\}$$

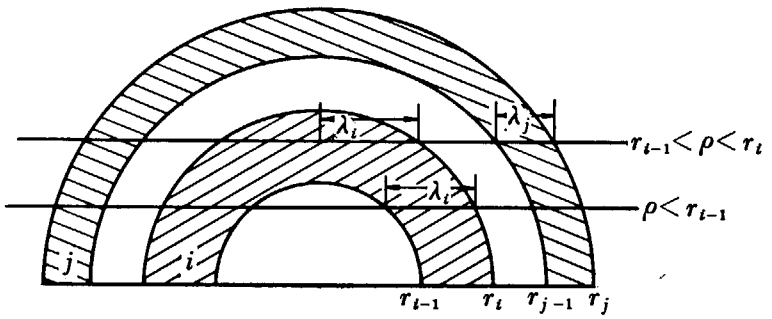


Fig. 2.4 Neutron paths in case $i<j$.

$$*\exp\left\{-\sum_{k=i+1}^{N}\lambda_k\right\}*\left\{1+\exp\left(-2\sum_{k=1}^{i-1}\lambda_k-\lambda_i\right)\right\}$$
$$+\frac{2\pi}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i}\rho d\rho\,\{1-\exp(-2\lambda_i)\}\exp\left(-\sum_{k=i+1}^{N}\lambda_k\right) \quad (2.52)$$

In our computer code, the integrands in Eqs. (2.48) and (2.49) with possible pairs of (i, j) are firstly calculated for a fixed ρ. Then, the integration over ρ is accomplished by changing the value ρ.

2.5 Collision probabilities for two-dimensional cylindrical lattice

In the cylindrical system with general shape of its cross section and of infinite height, the collision probability from a region i to another region j, P_{ij}, is expressed by the following Eq. (2.53) in the coordinate system of Fig. 2.5, assuming the flat flux in each region and an isotropic emission in the laboratory system.

$$P_{ij}=\left[\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\int_0^{\frac{\pi}{2}}\sin\theta\,d\theta\int_{AB}^{AC}dt\exp\left\{-\frac{\Sigma_i|AC-t|}{\sin\theta}\right\}\int_{AD}^{AE}dt'\frac{\Sigma_j}{\sin\theta}\right.$$
$$\left.\times\exp\left\{-\frac{\Sigma_j|t'-AD|}{\sin\theta}\right\}\exp\left\{-\frac{\left|\int_{AC}^{AD}\Sigma(s)ds\right|}{\sin\theta}\right\}\right]\Big/\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\int_0^{\frac{\pi}{2}}\sin\theta\,d\theta\int_{AB}^{AC}dt \quad (2.53)$$

In **Fig. 2.5** the line PQ′ defined by ρ and ϕ is the projection of the neutron path PQ on the horizontal plane. The points P and Q are, respectively, the source and collision positions. The point A is the origin of measures of t, t' and s. The points B, C, D and E are the points of intersection of the line PQ′ with the region boundaries. A restriction on the moving direction of a neutron is imposed so that a neutron moves only to the positive direction of t along the line PQ′.

If the line PQ′ enters region j more than once, a sum of Eq. (2.53) is required.

The self collision probability, P_{ii} is expressed by the following Eq. (2.54), where the point Q is in region i.

$$P_{ii}=\frac{\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\int_0^{\frac{\pi}{2}}\sin\theta d\theta\int_{AB}^{AC}dt\int_t^{AC}dt'\frac{\Sigma_i}{\sin\theta}\exp\left\{-\frac{\Sigma_i|t-t'|}{\sin\theta}\right\}}{\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\int_0^{\frac{\pi}{2}}\sin\theta d\theta\int_{AB}^{AC}dt} \quad (2.54)$$

If the line PQ′ reenters region i, a sum of such a term as Eq. (2.53) is required for obtaining P_{ii}.

The six-fold integrals of Eqs. (2.53) and (2.54) are reduced to the double integrals as follows:

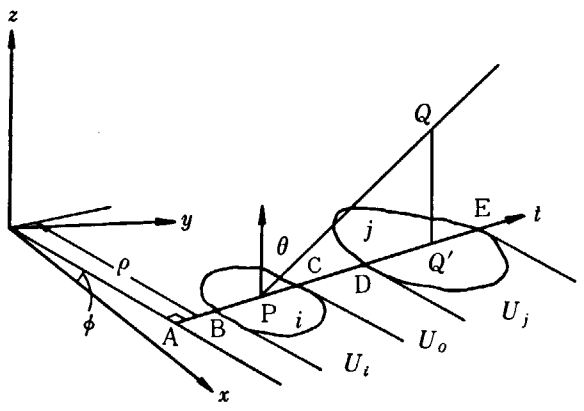


Fig. 2.5 Cylindrical coordinate system.

$$P_{ij}=\frac{\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\,[K_{i3}(U_0)-K_{i3}(U_0+U_i)-K_{i3}(U_0+U_j)+K_{i3}(U_0+U_i+U_j)]}{\pi\Sigma_i V_i}, \tag{2.55}$$

$$P_{ii}=\frac{\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\,[U_i-K_{i3}(0)+K_{i3}(U_i)]}{\pi\Sigma_i V_i}, \tag{2.56}$$

where U_i and U_j denote the optical path lengths (the physical path multiplied by the macroscopic total cross section), $U_i=BC*\Sigma_i$, and $U_j=DE*\Sigma_j$ and U_0 stands for the sum of optical path lengths between C and D; and K_{i3} is the third order Bickley function.

The escape probability P_{is} defined as a neutron emitted in region i escapes from the system without suffering collision, is expressed as

$$P_{is}=\frac{\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\,[K_{i3}(U_{is})-K_{i3}(U_{is}+U_i)]}{\pi\Sigma_i V_i}, \tag{2.57}$$

where U_{is} is the optical path length along the path from the edge of region i to the surface of the system.

As for the directional probability, similarly to the case of the circular cylinder, it is not necessary to write the whole components and hence a few samples are shown here:

$$P_{ij_x}=\frac{3\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\,[K_{i5}(U_0)-K_{i5}(U_0+U_i)-K_{i5}(U_0+U_j)+K_{i5}(U_0+U_i+U_j)]}{2\pi\Sigma_i V_i}, \tag{2.58}$$

$$P_{ii_x}=\frac{\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\,[2U_i-3K_{i5}(0)+3K_{i5}(U_i)]}{2\pi\Sigma_i V_i}. \tag{2.59}$$

Thus we have the double integration of the linear combination of K_{in} function as a final form of the collision probability for the two-dimensional cylindrical system. About the geometries taken into LAMP, they will be described later in detail.

Here we show the procedure how to get the probability from the double integration.

The integration over ρ are performed by the trapezoidal rule of uniform weight for the geometry of XY division. For the geometry of annular division the ρ range is divided by the radii which are the boundaries of regions and then in each subdivision the Gaussian quadrature is taken. The integration of ϕ is performed also by the trapezoidal rule and in some subprograms the subdivision is further devided by the Gaussian coordinate, which however is inefficient. Once ρ and ϕ are chosen, a line is drawn on the projected plane. The total number of lines directly affect the accuracy of the integration.

The geometrical informations required for the line are the table of the *region* identification numbers and the path lengths across each *region* along the line, with the number of *regions* in the base cell now considered and the total number of *regions* on the line. A subroutine GEOM is prepared for calculating the informations of a unit cell. By the successive use of the subroutine, the information for the cells other than the base cell is also obtained. They are stored for the multigroup calculation in a scratch unit for every line. The records are read and the P_{ij} integration is performed for the first set of cross sections, the scratch unit is than rewound and the records are again read for the second set of cross sections, and so forth. The content of the record is as follows,

L0, LLL, W, (T(L), II(L), L=1, LLL)

where L0=the number of *regions* in the base cell, LLL=the total number of *regions* on the line, W=the weight of the line, T(L)=the path length and II(L)=the *region* identification number. The source *region* is picked up sequentially from the first L0 *regions* in the table. If the L-th

region is set to the source *region* (the *region* identification number is II(L)), the collision *region* is picked up sequentially from the L-th *region* to the LLL-th *region* until the optical path length between the source and the collision exceeds the limited value (which is provisionally set to 7.0). Then the source *region* is set to the (L+1)-th *region*. If the vacuum boundary condition or the isotropic boundary condition is adopted, the table is cut off at LLL=L0, and P_{is} is culculated.

If a neutron path traverses a region which is empty or nearly empty, the integrands of Eqs. (2.55) and (2.56) tend to zero but P_{ij}/Σ_i which is used for the diffusion coefficient defined by BENOIST[12], has a finite non zero value, and the forms of the integrands bring the growth of the truncated error. To avoid this growth of error, the subroutine DELT integrates the symmetric elements Δ_{ij} defined by

$$\Delta_{ij}=\frac{1}{\pi}\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\frac{[K_{i3}(U_0)-K_{i3}(U_0+U_i)-K_{i3}(U_0+U_j)+K_{i3}(U_0+U_i+U_j)]}{\Sigma_i\Sigma_j}, \tag{2.60}$$

$$\Delta_{ii}=\frac{1}{\pi}\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\frac{[U_i-K_{i3}(0)+K_{i3}(U_i)]}{\Sigma_i\Sigma_i}, \tag{2.61}$$

$$\Delta_{is}=\frac{1}{\pi}\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\frac{[K_{i3}(U_{is})-K_{i3}(U_{is}+U_i)]}{\Sigma_i}. \tag{2.62}$$

The Δ_{ij} relates with P_{ij} as

$$P_{ij}=\frac{\Delta_{ij}\Sigma_j}{V_i}, \tag{2.63}$$

$$P_{is}=\frac{\Delta_{is}}{V_i}. \tag{2.64}$$

The reciprocity theorem

$$\Delta_{ij}=\Delta_{ji}$$

follows immediately from Eq. (2.60).

If a neutron path traverses an empty or a nearly empty region, the integrand of Δ_{ij} is reduced to the differential form of Bickley function as Eq. (2.66), Eq. (2.67), Eq. (2.68), and Δ_{ii} and Δ_{is} are reduced to the forms as Eqs. (2.69) and (2.70), respectively:

$$\Delta_{ij}=\frac{1}{\pi}\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\frac{T_j[K_{i2}(U_0)-K_{i2}(U_0+U_i)]}{\Sigma_i} \quad \text{for small } \Sigma_j \tag{2.66}$$

$$\Delta_{ij}=\frac{1}{\pi}\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi\frac{T_i[K_{i2}(U_0)-K_{i2}(U_0+U_j)]}{\Sigma_j} \quad \text{for small } \Sigma_i \tag{2.67}$$

$$\Delta_{ij}=\frac{1}{\pi}\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi T_iT_jK_{i1}(U_0) \quad \text{for small } \Sigma_i,\ \Sigma_j \tag{2.68}$$

$$\Delta_{ii}=\frac{1}{2\pi}\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi T_i^2K_{i1}(0) \quad \text{for small } \Sigma_i \tag{2.69}$$

$$\Delta_{is}=\frac{1}{\pi}\int_{-\infty}^{\infty}d\rho\int_0^{\pi}d\phi T_iK_{i2}(U_{is}) \quad \text{for small } \Sigma_i \tag{2.70}$$

where T_i and T_j are the path length across the region i and j. These forms are taken under the condition that U_i or U_j is less than 0.01. As a result, the forms are applied not only to an empty region but to the case where the path length T_i or T_j is small. This technique has little effect to reduce the number of lines required for a given accuracy for the case where the system contains a region, the area of which is only a small fraction of that of the whole system, because the region is crossed by relatively few of the lines defined by ρ and ϕ.

The modified collision probability, $P_{ij}^{\sharp}(P_{ij}^{\sharp}=P_{ij}/\Sigma_j)$ is prefered rather than the collision probability, P_{ij}, where an empty region appears. The modified probability is obtained from the following relation:

$$P_{ij}^{\sharp}=\frac{\Delta_{ij}}{V_i}. \tag{2.71}$$

After calculating $P_{ij}^{\sharp}$, to avoid the errors accompanied by the numerical integration, the adjustment is made so as to satisfy the conservation theorem as follows:

$$\chi_i\left\{\sum_{j=1}^{N} P_{ij}^{\sharp}\Sigma_j+P_{is}\right\}=1 \tag{2.72}$$

where χ_i is a normalization factor to satisfy Eq. (2.72). After finding χ_i, $P_{ij}^{\sharp}$ for $j=i\sim$NR are normalized. Such a procedure is repeated for $i=1\sim$N, and then, if required, $P_{ij}^{\sharp}$ is replaced by P_{ij}. It should be noted that if χ_i deviates from unity the reciprocity relation does not hold but it affects to the flux only slightly.

The numerical calculation of K_{in} functions has yet to be explained. Although some rational approximations are developed for the Bickley functions, they would be very time consuming because they have to be used so frequently as $10^6\sim10^7$ times. In LAMP the quadratic interpolation is performed numerically by using tables of **a**, **b**, **c**; the coefficients of three terms of the quadratic expression of the Bickley function. These tables list **a**, **b** and **c** as a function of x and n where

$$a_m=\frac{y_{m-1}-2y_{m-1/2}+y_m}{2\Delta x^2} \tag{2.73}$$

$$b_m=\frac{y_m-y_{m-1}}{\Delta x}-a_m(x_{m-1}+x_m) \tag{2.74}$$

$$c_m=y_{m-1}-b_m x_{m-1}-a_m {x_{m-1}}^2 \tag{2.75}$$

$$y_m=K_{in}(x_m) \tag{2.76}$$

$$\Delta x=x_m-x_{m-1} \tag{2.77}$$

The domain of the tables of a, b and c is as follows:

$\Delta x/x=0.01/(0.0,\ 0.48)$, 48 points,
$0.04/(0.48,\ 2.4)$, 48 points,
$0.1/(2.4,\ 9.6)$, 72 points,
$\infty/(9.6\quad)$, 1 points,

where $x>9.6$, $K_{in}(x)$ is assumed to be zero.

Thus the Bickley function is computed by performing twice the multiplication and twice the summations after the table look-up:

$$K_{in}(x)=(a_{nm}x+b_{nm})x+c_{nm} \tag{2.78}$$

where $x_{m-1}\leq x\leq x_m$

The table look up and the interpolation is performed in the routine itself to avoid the process of calling an external subroutine.

3. Iterative Solution of Linear Equations

The system under consideration is divided into N regions and the neutron energy range is divided into G groups. Let define the physical quantities considered as follows:

1. the volume of region i;

$$V_i=\int_{V_i} dV$$

2. the integral flux over the region i for the energy group g;

$$\phi_{ig}=\int_{V_i} dV\int_{\Delta E_g} dE\phi(\boldsymbol{r}, E)$$

3. the fixed source;

$$S_{ig}=\int_{V_i} dV\int_{\Delta E_g} dES(\boldsymbol{r}, E)$$

4. the collision probabilities from i to j for the group g

$$P_{ij}{}^g$$

5. the modified collision probability

$$P_{ij}{}^{g\sharp}=P_{ij}{}^g/\Sigma_j{}^g$$

6. H_{ig} the emission rate of the region i for the group g

7. Nuclear constants of the material m;

$\Sigma_m{}^g$ = total cross section,

$\nu\Sigma_m{}^g=\nu*$ fission cross section,

$\Sigma_n{}^g$ = absorption cross section,

$\Sigma_{sm}{}^{g\to g'}$ = scattering cross section from group g to group g',

$\Sigma_{slm}{}^g=\Sigma_s{}^{g\to G+1}+\chi_m{}^{G+1}\nu\Sigma_{fm}{}^g$

= slowing down cross section in which the superscript $G+1$ denotes the energy range below the lowest energy considered and $\chi_m{}^g$ stands for the fission yield to the group g.

Using the above definitions the equations to be solved is written as

$$\phi_{ig}=\sum_{j=1}^{N} P_{ji}{}^{g\sharp}H_{jg} \tag{3.1}$$

The emission rate for the fixed source problem is written as.

$$H_{ig}=S_{ig}+\sum_{g'=1}^{G}\Sigma_{sm}{}^{g'\to g}\phi_{ig'}+\chi_m{}^g\sum_{g'=1}^{G}\nu\Sigma_{fm}{}^{g'}\phi_{ig'}, \tag{3.2a}$$

where m denotes the material assigned to region i. For the eigenvalue problem,

$$H_{ig}=\sum_{g'=1}^{G}\Sigma_{sm}{}^{g'\to g}\phi_{ig'}+\frac{\chi_m{}^g}{\lambda}\sum_{g'=1}^{G}\nu\Sigma_{fm}{}^{g'}\phi_{ig'} \tag{3.2b}$$

The Eqs. (3.1) coupled with (3.2a) form inhomogeneous equations and those coupled with the Eqs. (3.2b) form homogeneous equations. In both problems the number of unknowns is $N*G$. The general matrix of the same rank consists of N^2G^2 elements, although the computer storage required for the above equations is at most $N^2G+MG^2<NG(N+G)$, [N^2G for the collision probabilities and MG^2 for scattering matrices, where M is the number of materials]. The size G^2 for the scattering matrix will be reduced if only down-scattering is considered or if only heavy elements compose the material.

The equations are solved iteratively using the method of Successive Over Relaxation (SOR) as in THERMOS[13], which is partly modified to deal with the eigenvalue problem. The procedure

is as followes.

Step 1. to set the initial guess of ϕ_{ig}

Step 2. to obtain the normalization factor B to normalize the source terms.

$$B=\sum_{g=1}^{G}\sum_{i=1}^{N}Q_{ig}\sum_{j=1}^{N}P_{ij}, \tag{3.3}$$

where

$$Q_{ig}=\begin{cases}S_{ig}, & \text{for the fixed source problem,}\\ \chi_m{}^g\sum_{g'}\nu\Sigma_{fm}{}^{g'}\phi_{ig'}, & \text{for the eigenvalue problem}\end{cases} \tag{3.4}$$

After obtaining B, Q_{ig}'s are divided by B. Then the number of the source which will have the next collision in the system considered is set to unity.

Step 3. to calculate H_{ig}'s according to Eqs. (3.2) and simultaneously the scaling factor C:

$$C=\sum_g\sum_j\{\Sigma^g\phi_{ig}-(H_{ig}-Q_{ig})\sum_j P_{ij}{}^g\}, \tag{3.5}$$

which is defined as the ratio of the removal reaction (absorption, slowing down and leakage) to the source which will have the next collision in the system. This factor must be unity in the converged state.

Step 4. to calculate the new fluxes $\phi^{(m+1/2)}$ according to Eq. (3.1) and simultaneously the weighted residual.

$$R=\{\sum_g\sum_i(\phi_{ig}{}^{(m+1/2)}/C-\phi_{ig}{}^{(m)})^2R_{ig}{}^2/\sum_g\sum_i R_{ig}{}^2\}^{1/2} \tag{3.6}$$

The root mean square (RMS) residual gill be used to estimate the converging slope as $\mu^{(m)}=R^{(m)}/R^{(m-1)}$. The weight reaction R_{ig} is chosen from Σ_{act}, $\nu\Sigma_f$, Σ and Σ_a. The superscript (m) is an iteration counter.

Step 5. to get the over-relaxation factor ω.

In the first L_e iterations the initial value of ω_0 will be used. At each iteration the value of $\omega_e=1/(1-\mu^{(m)})$ is tested. If all the value of ω_e in the last L_e iterations agree to within $100\varepsilon_e\%$ an extrapolation takes place using the most recent value of ω_e. The testing for a possible extrapolation is not allowed during the L_d iterations following the extrapolation. The value λ (the estimate of the eigenvalue of the matrix considered) is computed as $\lambda=(\mu^{(m)}-1+\omega_0)/\omega_0$ and a new ω is obtained as $\omega_1=2/(2-\lambda)$ to be used after the next iteration.

If the increase of the residual is detected during the iteration, a moderate value of ω is chosen as

$$\omega_2=(\omega_1 * f_{under})^{1/2} \tag{3.7}$$

Step 6. to obtain the new flux by the over relaxation:

$$\phi_{ig}{}^{(m+1)}=\phi_{ig}{}^{(m)}+\omega(\phi_{ig}{}^{(m+1/2)}/C-\phi_{ig}{}^{(m)}). \tag{3.8}$$

A small edit of the calculation is provided, such as the average cross sections and the diffusion coefficients (in Benoist model) for each group, and the energy integrals of reaction rates for each region and for each constituent material. Additional activation cross sections are acceptable in the edit stage for the reaction rate calculation.

The loop from Step 3. to Step 6. is repeated until the residual $R^{(m)}$ is less than ε or the number of the inner iterations amounts to L_{in}. In the eigenvalue problem the outer loop from Step 2. to Step 6. is repeated until the source normalization factor B converges or the iteration counter amounts to L_{out}. The quantities ε, ω_0, ε_e, f_{under}, L_e, L_d, L_{in}, and L_{out} are input numbers.

Specially for the one group problem each collision program (PATH, CLUP or CLUP 77) has a routine to solve the linear equations in itself by the Gauss-Jordan method.

4. Preparation of Nuclear Cross Sections

The collision probability programs require the macroscopic total cross section vector for each constituent material. In addition, the flux solving programs (PIJF, TUD) requires the complete set of macroscopic cross sections. They are fed into the program through input cards or disk in a fixed format described later. The resonance integral program RICM also requires the cross sections which are prepared by the exclusive program LTE.

4.1 Thermal cross-sections

The one dimensional data such as the absorption and the ν*fission cross sections are supplied by THERMOSEC[9] which takes the data from the ENDF/B[14] files, and by PI2[8] from the ENDF'A[15] files. They are fed into PIXSE originally an English code[7] which makes the library file for each nuclide. The scattering matrices for the principal moderators are obtained from the tabulation of scattering law in the ENDF/A format. Various theoretical models are provided such as effective width and the free gas models.

Here we give a brief explanation of processing the scattering matrix in PIXSE. The differential scattering cross section $\sigma(E\rightarrow E', \theta)$ is computed from the scattering law $S(\alpha, \beta)$. The relation is

$$\sigma(E\rightarrow E', \theta)=\frac{\sigma_p}{2kT}\sqrt{\frac{E'}{E}}e^{-\beta/2}S(\alpha, \beta) \tag{4.1}$$

where E and E' are respectively initial and final neutron energies, and θ is the angle of scattering.

The generalized energy transfer cross sections are obtained by Gaussian quadrature

$$\sigma_n(E\rightarrow E')=\int_{-1}^{1}\mu^n\sigma(E\rightarrow E', \mu)d\mu$$

where $\mu=\cos\theta$. The elastic component is obtained from

$$\sigma_n(E\rightarrow E)=\frac{\sigma_b}{2}\int_{-1}^{1}e^{-\alpha\lambda}\mu^n d\mu \tag{4.2}$$

where the parameter λ is the input number.

The group averaged cross sections are calculated by

$$\sigma_{ngg'}=\int_{E_1'}^{E_2'}dE'\int_{E_1}^{E_2}dE\sigma_n(E\rightarrow E')\phi(E/kT)\Big/\int_{E_2}^{E_1}\phi(E/kT)dE \tag{4.3}$$

where E_1 and E_2 are respectively upper and lower bounds of the g-th group and $\phi(E/kT)$ is a spectrum function used as the weight. The integrals over energy are evaluated also by Gaussian quadrature. As well known the collision probability method for the general geometry can not take account of the anisotropic scattering. Therefore the transport correction is available:

$$\sigma_{0gg}(\text{modified})=\sigma_{0gg}-\sum_{g'}\sigma_{1gg'} \tag{4.4}$$

The number of neutrons, S_g, scattered into the g-th group ($E_1\sim E_2$) from a $1/E$ spectrum above the upper limit of the highest group, E_m, can be calculated by

$$S_{ng}=\int_{E_1}^{E_2}dE\int_{E_m}^{\infty}\sigma_n(E'\rightarrow E)\frac{dE'}{E'}$$

The integrals are evaluated by Gaussian quadrature of the same order as that used in calculating $\sigma_{ngg'}$ in the range from E_m to ∞, which is split into equal logarithmic intervals $[\zeta^P E_m\sim\zeta^{P+1}E_m]$, where ζ is specified in the input and these integrals are summed up. The summation over P is

truncated when the p-th integral becomes less than a certain fraction (ε) of the source intensity of the highest group.

In the present thermal library of 30 groups from 0. to 0.625eV, the matrices of hydrogen, deuterium and graphite are available based upon the ENDF/A A0569, A0999 and A0560, respectively. All other matrices are based on the free gas scattering law. The thermal library is stored in PDS files in which a PDS member consist of a series of data for a nuclide. The macroscopic cross sections and the source distributions is composed by PIXEDT with use of the library.

4.2 Fast cross sections

The fast cross section library is compiled by FAXSE in a format similar to that of GAM-I[10]. It begins with the fission yield vectors. It is followed by a pack of data for the first nuclide, the second nuclide and so on. A pack of data for a nuclide consists of four physical records. The first record contains the identification number of the nuclide and some integer constants for the second record control. The second record contains the absorption vector, fission vector, ν vector, N-N matrix, etc. The third and the fourth are optionally provided for the tabulation of the self shielding factor due to resonance reactions as a function of σ_0 (the total cross section of admixtures per a resonant atom) and temperature T.

The process program to prepare the input of FAXSE is not contained in LAMP, but, SUPERTOG[16] and GROUCH[17] work for this purpose; the latter may be skipped because the geometry and temperature dependent cross sections due to resonances can be computed, case by case, by RICM to replace the corresponding part in the fast library.

We have two fast libraries, one for the fast reactors from the JAERI 70 group set[18] and the other a 68 group set for thermal reactors from the GAM-II library. The self-shielding factors for the former are tabulated simultaneously at the time of compiling the JAERI 70 group set.

The macroscopic cross sections will be composed by FAXEDT using these libraries. In the program the admixture cross section $\sigma_{0\mu}{}^g$ (the g-th group σ_0 of a constituent nuclide μ in a mixture) is given by

$$\sigma_{0\mu}{}^g=\frac{1}{N_\mu}\sum_{\mu'\neq\mu}N_{\mu'}\sigma_{t\mu'}{}^g+\frac{1}{N_\mu\hat{l}} \tag{4.5}$$

where N_μ is the atomic density of component μ: $\sigma_{t\mu}{}^g$ is the effective total cross section of component μ of g-th group and $\hat{l}$ is the mean chord length of the mixture lump:

$$\sigma_{t\mu}{}^g=\sigma_{NN\mu}{}^g+\sigma_{N2N\mu}{}^g+\sigma_{e\mu}{}^g+\sigma_{a\mu}{}^g, \tag{4.6}$$

where

$\sigma_{NN\mu}{}^g$	N-N inelastic	cross section.
$\sigma_{N2N\mu}{}^g$	N-$2N$ inelastic	
$\sigma_{e\mu}$	effective elastic	
$\sigma_{a\mu}$	effective absorption	

To interpolate between intermediate tabular values, we apply the same technique as used in EXPANDA[19] in which they assume that $f(\sigma_0, T)$ may be represented as a hyperbolic function of $\log(\sigma_0)$ and of T. For given values of σ_0 and T, we interpolate first $f_r(\sigma_0, T_1)$, $f_r(\sigma_0, T_2)$ and $f_r(\sigma_0, T_3)$ to obtain $f_r(\sigma_0, T)$ where r denotes a reaction among absorption, fission and elastic scattering.

After interpolating $f_{a\mu}{}^g$, $f_{f\mu}{}^g$ and $f_{e\mu}{}^g$ for required components in the mixture, we get effective cross sections as follows:

$$\sigma_{a\mu}{}^g=f_{a\mu}{}^g\sigma_{a\mu}{}^g \tag{4.7}$$

$$\sigma_{f\mu}{}^g=f_{f\mu}{}^g\sigma_{f\mu}{}^g \tag{4.8}$$

$$\sigma_{e\mu}{}^g=f_{e\mu}{}^g\sigma_{e\mu}{}^g \tag{4.9}$$

where

$$\sigma_{e\mu}^{g}=\sum_{g'}\sigma_{P_0}^{g\to g'}-\frac{1}{3}\sum_{g'}\sigma_{P_1}^{g\to g'}, \tag{4.10}$$

$$\sigma_{NN\mu}^{g}=\sum_{g'}\sigma_{NN\mu}^{g\to g'}, \tag{4.11}$$

$$\sigma_{N2N\mu}^{g}=\sum_{g'}\sigma_{N2N\mu}^{g\to g'}. \tag{4.12}$$

As an argument of interpolation depends implicitly on $f(\sigma_0, T)$, the above formulations are used iteratively.

After obtaining the converged microscopic cross sections we get the macroscopic ones for the coarse energy-group G as follows:

1. ν^* fission cross section:

$$\nu\Sigma_{fG}=\frac{\sum_{g\in G}W_g\sum_{\mu}N_\mu\nu\sigma_{f\mu}^{g}}{\sum_{g\in G}W_g}. \tag{4.13}$$

2. Transport cross section;

$$\Sigma_{trG}=\frac{\sum_{g\in G}W_g\sum_{\mu}N_\mu\sigma_{tr\mu}^{g}}{\sum_{g\in G}W_g}. \tag{4.14}$$

3. Fission source spectrum;

$$\chi_G=\frac{\sum_{g\in G}\sum_{\mu}N_\mu\chi_\mu^{g}F_\mu}{\sum_{\mu}N_\mu F_\mu}, \tag{4.15}$$

where

$$F_\mu=\sum_{g}\nu\sigma_{f\mu}^{g}\cdot W_g. \tag{4.16}$$

4. Diffusion coefficient;

$$D_G=\frac{\sum_{g\in G}W_g/(3\sum_{\mu}N_\mu\sigma_{tr\mu}^{g})}{\sum_{g\in G}W_g}, \tag{4.17}$$

5. Absorption cross section;

$$\Sigma_{aG}=\frac{\sum_{g\in G}W_g\sum_{\mu}N_\mu\sigma_{a\mu}^{g}}{\sum_{g\in G}W_g}. \tag{4.18}$$

6. Transfer cross section;

$$\Sigma_{G\to G'}=\frac{\sum_{g\in G}W_g\sum_{\mu}N_\mu\sum_{g'\in G'}\sigma_\mu^{g\to g'}}{\sum_{g\in G}W_g}, \tag{4.19}$$

where

$$\sigma_\mu^{g\to g'}=f_{e\mu}^{g}\left\{\sigma_{P_0\mu}^{g\to g'}-\frac{1}{3}\delta_{gg'}\sum_{g''}\sigma_{P_1\mu}^{g\to g''}\right\}+\sigma_{NN\mu}^{g\to g'}+2\sigma_{N2N\mu}^{g\to g''}. \tag{4.20}$$

7. Slowing-down cross section;

$$\Sigma_{slG}=\Sigma_{G\to NGR+1}+\left(1-\sum_{G'=1}^{NGR}x_{G'}\right)\nu\Sigma_{fG} \tag{4.21}$$

In these expressions the weight W_g is physically the flux and the input to be used for collapsing the energy groups.

4.3 Resonance integrals

The effective cross sections in the resonance energy-region are obtained from the resonance

integrals for resolved levels. A program RICM[11] is prepared to calculate them in a multi-region (up to 20 regions) lattice by using the collision probability method.

The program LTE which makes a pair with RICM supplies the numerical values of Doppler broadening functions, ψ and χ, for the fine energy points around each level. The LTE is thoroughly revised from the original version[11]. The ψ and χ are quickly obtained by interpolation from the tabulated form. The energy ranges of numerical integration of slowing-down source and the energy meshes are determined automatically to fit for any levels and for any nuclide. The level paramaters in the format of ENDF/B are fed into LTE.

The table of collision probabilities is prepared by any of the collision probability programs in LAMP. The tabulation must be performed for a wide range of cross sections of the resonant isotope, whereas the cross sections of moderating materials are kept constants. The auxiliary program PRERI provides the total cross sections to be used in the collision probability program for this purpose.

In the edit stage of RICM the additional group cross sections can be fed such as the smooth cross sections, the P wave component of the resonance integral and the tail of the levels lying under the lower limit of the energy group structure. The final results of RICM are stored in PDS file which is used to replace the data in the library.

4.4 I/O format of macroscopic multi-group cross sections

Now we describe the format of macroscopic cross sections commonly used by several programs in LAMP as I/O. One type of these cross sections is the total cross section and the other is a complete set of cross sections containing. $\nu*$fission, total, scattering matrix, etc. A series of data of each type for a mixture form a block. The data in a block are read or written at once. A block may be on cards or on PDS file (DISK)[1] but a series of blocks required for an execution are to be on the same device.

If a block is on PDS file, it has a name of eight characters. It is read or written in LAMP by the statement such as

$$\text{CALL} \quad \begin{matrix}\text{READ}\\ \text{WRITE}\end{matrix} \quad \text{(NAME, ARRAY, LENGTH)}$$

where NAME is a two words array storing the name of the block. ARRAY is an array where a series of data of the block are to be read or written. LENGTH means the length (the number of data) of the block. The name of the block must be specified before calling such a statement.

If a block of total cross sections on cards is read, the total cross sections must be arranged as beginning with group 1 (the highest energy-group), then group 2, ⋯, and finally the last group. They are read by the subroutine REAG[24] which reads the series of data in a free format into a one-dimensional array in the core storage.

If the complete set of cross sections is read from cards, the following layout must be specified for all energy groups in order beginning with group 1. For group g the layout is as follows:

Card 1 (2I6, 5E12.5)

LSS_g	1-6	Position of the self scatter cross section in the group vector
LGV_g	7-12	Length of the group vector
Σ_{actg}	13-24	Activation cross section for this group, which may be used for the convergence monitor
$\nu\Sigma_{fg}$	25-36	Fission cross section$*$average number of neutron produced by fission in this group
Σ_g	57-48	Total cross-sections. As $n-2n$ reaction is concerned, the apparent

conservation is not fulfilled

χ_g	49-60	The fraction of neutrons produced by fission of this material to this group
D_g	61-12	The diffusion coefficient which is used only in TUD optionally

Card 2 (6E12.5)

The group "vector" contained as follows to be punched 6 per card

Σ_a	The absorption cross section for this group
$\Sigma_s^{g\to g-n}$	
〃	
〃	
〃	
$\Sigma_s^{g\to g}$	the self scattering cross section
$\Sigma_s^{g\to g+1}$	
〃	
〃	
〃	
$\Sigma_s^{g\to g+m}$	

Here $\Sigma_s^{g\to g'}$ is the scattering cross section from group g to group g', the length of the group vector is $n+m+2$, and m and n may be chosen so that zero values for cross section other than Σ_a^g and $\Sigma_s^{g\to g}$ need not be punched.

The program determines $g_1=g-n$ and $g_2=g+m$ from the values of LSS_g and LGV_g for this group

$$g_1=g-n=g+2-LSS_g$$
$$g_2=g+m=LGV_g-LSS_g+g$$

Note that the data as $g_2>NG$ (=total number of energy groups) are allowed. They are considered to be the slowing down cross section from this group to the energy range below the lowest group.

If the complet set of cross sections is on PDS file, the arrangement of data is almost same as that on cards. Only one difference is that the integer numbers LSS and LGV have been converted into the floating point number which is converted into integer after reading into core storage. The array of group 1 is followed by that of group 2 and so on, all the arrays are packed into a block. The total length of the block can be found by calling the statement:

CALL GETLEN (NAME, LENGTH)

Once a block is read into core storage the name of eight characters loses its original meaning and, instead of it, the mixture which has been read at m-th turn is termed as *material* m in the program.

5. Description of Each Program and Specification of I/O

This chapter gives a brief description of each constituent program and specification of I/O. The hierarchy of the programs and the data flow will be described in Chap. 6. The constituent programs are classified into four groups by their functions.

Group a) Production of collision probabilities:

PATH; for the lattice cell of slab, sphere, circular square or hexagonal cylinder, two-dimensional square or hexagonal cylinder

CLUP; for the circular cylindrical assembly in which pin rods are arranged circulary

CLUP 77; for the assembly with squarely arrayed pin rods

One of the above three is optionally used depending on the geometry

Group b) Microscopic cross section libraries:

PIXSE; for thermal energy region

LTE; for resonance region

FAXSE; for epithermal and fast region

Each program should be used for a set of the energy group structure and for a version of the evaluated neutron data file.

Group c) Macroscopic cross sections:

PIXEDT; for thermal energy region

PRERI; for resonance integral calculation

FAXEDT; for epithermal and fast energy region

Group d) Neutron flux and average constants:

PIJF followed by EDIT; the solution of the simultaneous equations using the collision probabilities

TUD; the multi-group diffusion calculation

EPISPEC; the epithermal spectrum by using the homogenized cross sections

RICM; the resonance integrals by the collision probability method

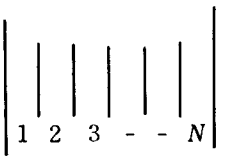
5.1 PATH; Collision probabilities for one-dimensional slab and spherical system and for cylindrical system containing a single rod

We restrict the geometries treated by PATH as shown by **Fig. 5.1 (a-g)**.

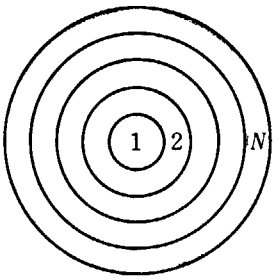
a. Slab; The slab system in **Fig. 5.1a** which is divided into N *zones* numbered from the left to right in the ascending order. The abscissas of the boundary surfaces must be at $N+1$ points beginning with 0.0. It may be an isolated system or a unit of periodic array. That is specified by the outer boundary condition for which three options are available. The term *vacuum* (equal to *black*) means the isolated system where no neutron comes from the outer surface. The term *isotropic* (equal to *white*) means that the same number of neutrons out-going comes back isotropically. The term *reflective* means that the outer surface works as mirror but it is to be noted that the PATH takes this option as *periodic*. In most cases the terms *reflective* and *periodic* mean the same. Only an exception occurs in the case of the unsymmetric slab lattice.

If the system is symmetric we can economize the calculation by giving the same *region* number to the corresponding two *zones* as illustrated below,

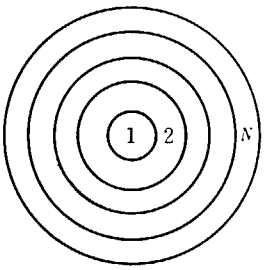
ZONE	1	2	3	⋯	N	⋯	$2N-1$	$2N$
REGION	N	$N-1$	$N-2$	⋯	1	⋯	$N-1$	N



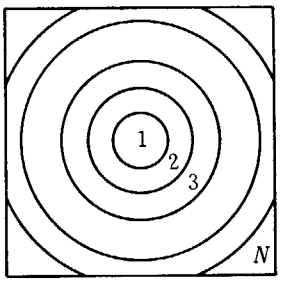
a. slab



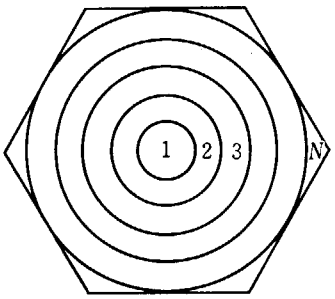
b. sphere



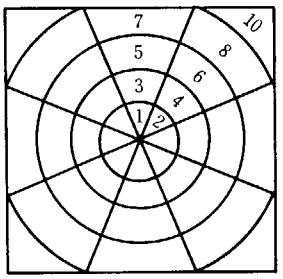
c. circular cylinder



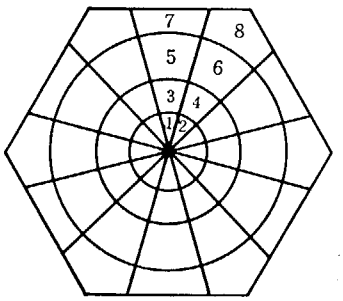
d. square cylinder



e. hexagonal cylinder



f. square cylinder with azimuthal division



g. hexagonal cylinder with azimuthal division

Fig. 5.1 Geometries treated by PATH.

The symmetric system of $2N$ *zones* is reduced to that of N *regions*.

b. Sphere; **Fig. 5.1b** shows the cross section of a spherical system. The system is divided by concentric spherical shells into N *zones* numbered from the central *zone* beginning with 1 to the outer *zones*. The radii dividing *zones* are to be given for $N+1$ points beginning with 0.0. In this geometry a *zone* should be in accord with a *region*.

c. Circular cylinder; **Fig. 5.1c** shows the cross section of a circularly cylindrical system. The system is divided by concentric annuli into N *zones* which should be numbered from the central *zone* to the outer *zones* in the ascending order. In this geometry a *zone* is in accord with a *region*. The radii dividing *zones* are to be given for $N+1$ points beginning with 0.0.

d. Square cylinder; **Fig. 5.1d** shows the cross section of a square cylinder of infinite length. The system is divided by concentric annuli into N *zones* which should be numbered from the central *zone* to the outer *zones* in the ascending order. The radii dividing zones are to be given for $N+1$ radii beginning with 0.0. But the last entry of radii has to be the half of lattice pitch. The radii before the last may be larger than the half pitch as illustrated in **Fig. 5.1d**, but they must be less than $\sqrt{2}$ times the half of lattice pitch.

e. Hexagonal; **Fig. 5.1e** shows the cross section of a hexagonal cylinder of infinite length. The system is divided by concentric annuli into N *zones* which should be numbered from the central *zone* to the outer *zones* in the ascending order. The radii dividing *zones* are to be at $N+1$ points beginning with 0.0. The last entry means the distance from the central axis to the outer surface (the half of lattice pitch). The radii before the last may be larger than the half pitch, but they most be less than $2/\sqrt{3}$ times the half pitch.

f. Square cylinder with azimuthal division; As shown in **Fig. 5.1f** a square cylinder is divided not only by the concentric annuli but also by four lines crossing the central axis. Each line makes an angle of 67.5° with a side line of the square. A ring is divided into eight pieces. As we assume the bilateral and diagonal symmetry, the pieces are classified into a side part and a corner part. If we divide the system by annuli into N rings, the total number of *zones* amounts to $2N$ in spite that some *zones* in a side part are out of the square in case where the inner radii of the zone exceeds $1/\cos 22.5°$ times the half pitch. When there are such ghost *zones* out of the square the number of *regions* should be less than that of *zones*. On the other hand in the annuli near the central axis the neutron fluxes in the side part and in the corner part are not so different each other so that both *zones* may be put into the same *region*.

Fig. 5.2 shows an example of the correspondence between *zones* and *regions*. For this case *region* numbers are assigned to each *zone* as follows:

ZONE	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
REGION	1	1	2	2	3	4	5	6	7	8	9	10	11	11	12	12

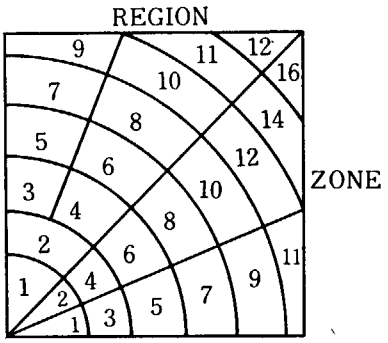


Fig. 5.2 Correspondence of zone to region.

g. Hexagonal cylinder with azimuthal division; As shown in **Fig. 5.1g** a hexgonal system is divided two-dimensionally by the six lines which cross each side by an angle of 75°. An annulus is also divided into a side part and a corner part. Other descriptions are the same as for the square case.

We now describe the procedures step by step:

1. Read and print input data.
2. Calculate volumes of each *zone*.
3. Assign the *zones* to *regions*.
4. Calculate the path tables (see page 35) for any geometry and store them into unit 12.
5. Read a path table, calculate the integrand of $\Delta_{ij}(\Delta_{ijk})$ integration and accumulate it in P_{ij} array for all possible pairs of (i, j) on the path table. Repeat this process for all path tables.
6. Transfer $\Delta_{ij}(\Delta_{ijk})$ into $P_{ij}^{\sharp}(P_{ijk}^{\sharp})$ and normalize them.
7. Transfer $P_{ij}^{\sharp}$ into P_{ij} if required.
8. $DR_{ik}=\sum_j P_{ijk}^{\sharp}$ if the directional probabilities are required.
9. PRINT, PUNCH and WRITE $P_{ij}(P_{ij}^{\sharp})$ and DR_i if required Repeat step 5 to 9 for all energy groups.
10. Calculate one-group flux distribution if required.

INPUT of PATH

BLOCK 0 (4H)

PROB 1-4	Program identification to be punched as 'PATH'

BLOCK 1 (72H)

TITLE 1-72	Problem identification

BLOCK 2

1 NR	Number of *regions*
2 NM	Number of *materials*
3 NG	Number of eneygy-groups
4 NGLAST	Number of energy groups already calculated in the previous calculation. The previous result is read from unit F20. (=0 otherwise)
5 IDRECT	Control of directional probability: =1, isotropic only =2, isotropic and radial for cylindrical geometry, or isotropic and perpendicular for plane geometry
6 IFORM	Output form of probabilities =0, P_{ij} (from V_i to V_j) =1, P_{ij}/Σ_j
7 ITYPE	Problem type: =0, P_{ij} only =1, fixed source calc. (only for NG=1) =2, k calculation (only for NG=1)
8 IEDPIJ	Output unit of probabilities: IEDPIJ=IPR+ICD+ITAPE IPR=1, print 0, skip ICD=2, card punch

0, skip

ITAPE=4, write in unit 21

0, skip

9 ITXEC — Input unit of total cross-sections
=0, card
=1, PDS file

10 NGR — Order of Gauss approximation for the radial integration (≦96)

11 NDA — Number of division of the angle for angular integration

12 NGA — Order of Gauss approximation for the angular integration (≦96)

Recommended value of NGR, NDA and NGA

	Slab	Cylinder	Sphere
NGR	1	8	8
NDA	1	2	1
NGA	2	4	1

13 IC — Coordinate and geometry of system

IG=100*ICOOD+10*ISH+IDIM:

ICOOD; coordinate
- =0, slab
- =1, cylinder
- =2, sphere

ISH; geometry of the lattice cells in the cylindrical coordinate (ignored if ICOOD≠1)
- =0, annular
- =1, square
- =2, hexagonal

IDIM; azimuthal division for square or hexagonal lattice
- =1, no division
- =2, divide

14 NX — Number of radial division

15 IBOUND — Outer boundary condition
- =0, isotropic reflection
- =1, perfect reflection
- =2, vacuum

16 NCELL — Number of lattice cells traversed by a neutron without suffering collisions (significant for IBOUND=1)

BLOCK 3

NREG(I), I=1, NZ — *Region* number assigned to the *zone* I

BLOCK 3 is required if NZ (=IDIM*NX)≠NR
If NZ is equal to NR, the *zone* I is assigned to *region* I.

BLOCK 4 (9A8) required if ITXEC=1

NAME(I) I=1, NM — Member name* in PDS file of the block of total cross sections of *material* I.

BLOCK 5

MAT(I) I=1, NR — *Material* numbers of the *region* I. The *material* number is numbered in order of reading in.

* a logical record is read/written with a member name in PDS file as a keyword in an indexed file.

BLOCK 6

RX(I),	Radii (RX(1) must be punched zero) For square or hexagonal lattice,
I=1, NX+1	RX(NX+1) must be the half of lattice pitch.

BLOCK 7 required only if ITXEC=0 and NG≠1

SIG(N, M)	Total cross section of *material* M.
N=1, NG,	A new card is necessary for each *material*
M=1, NM	

BLOCK 8 required if NG=1

SIGT(M)	Total	cross-sections for *material* M
SIGS(M)	Scattering	A new card is necessary for each *material*
SIGF(M),	Fission	
M=1, NM		

BLOCK 9 required if NG=1 and ITYPE=1

S(I), I=1, NR	Fixed source distribution

OUTPUT of PATH

See **OUTPUT of CLUP** in page 33

5.2 CLUP; Collision probabilities for the annular clustered assembly

The core of ATR (Advanced Thermal Reactor heavy water moderated, light water cooled, Pressure tube type) consists of the annular tubes assembly which contains the clustered pin rods in concentric rings. The program CLUP supplies the collision probabilities for such an assembly. **Figs. 5.3** and **5.4** show the sample lattice cells for CLUP. The geometry to be considered is as follows:

A. The system is divided into concentric annuli.

B. The annuli may include pin rods lying on a concentric circle. The rods in the inner

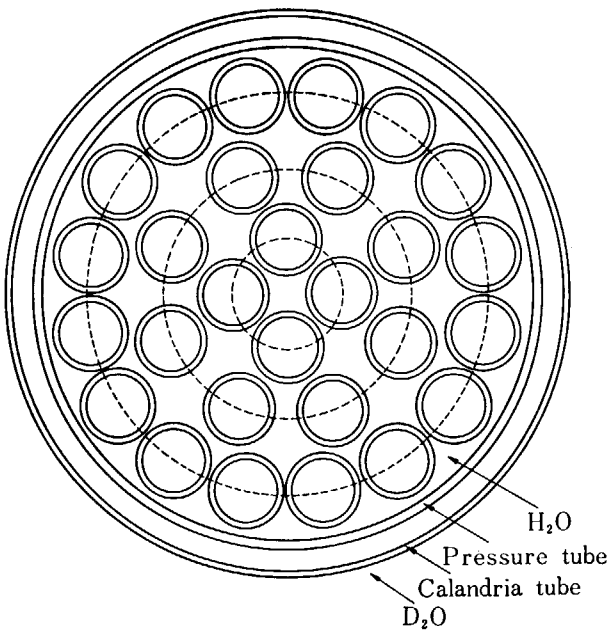


Fig. 5.3 Cross section of 28 rods cluster of ATR.

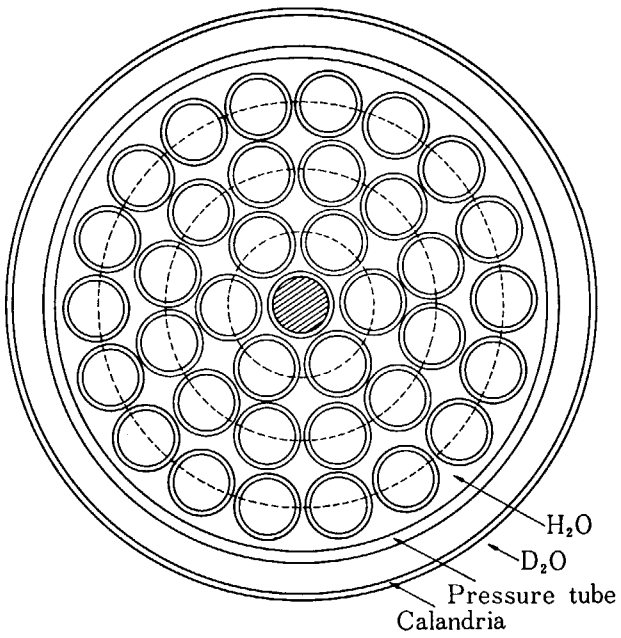


Fig. 5.4 Cross section of 36 rods cluster of SGHWR.

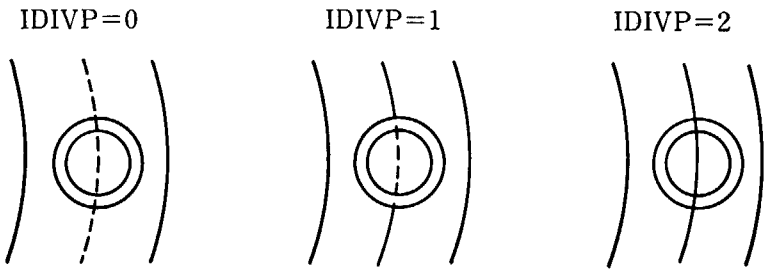


Fig. 5.5 The option of IDIVP.

most ring should be located at regular intervals. The system must be periodic around the central axis at every angle made by two adjacent rods to the central axis. For example, if the number of rods in the inner-most ring is 6, the period is $\pi/3$.

C. The pin rods included are to be in the same geometry. A pin rod may be divided into the concentric annuli.

D. The circle on which centers of rods locate becomes the boundary of *zones* by the optional use. If the input IDIVP is 0 it does not work. If IDIVP is 1 the circle divides only the moderator *zone* into the inner and outer annuli. If IDIVP is 2, each pin rod is also divided into an inner part and an outer part. The option of IDIVP is illustrated in **Fig. 5.5**.

E. To define the disposition of pin rods in rings, the following items are required:

1. Number of rings
2. Number of pin rods in each ring
3. Radii of rings
4. Azimuthal positions of each rods in each ring in radian

For the cases frequently used, some of typical geometries are installed in CLUP to save the input items 2 and 4. It works if the input number ISERI is specified to be as 4, 5 or 6 as shown in **TABLE 5.1**. A revised version of CLUP will remove the restrictions described in B and C and will divide annular moderator *zones* into sector *zones*.

TABLE 5.1 Specification of geometry in case ISERI 4, 5 or 6

ISERI	4	5	6
Number of rods in innermost ring	4	5	6
unit of period	$\pi/2$	$2\pi/5$	$\pi/3$
Number of rods in N-th ring	$2*2**N$	$5*N$	$6*N$
Interval of azimuthal angle between adjacent rods in N-th ring	$\pi/2**N$	$\frac{5}{2}\pi/N$	$\frac{\pi}{3}\Big/N$
Azimuthal angle of a rod in N-th ring	$\pi\left(\frac{3}{4}-\frac{1}{2**N}\right)$	$\pi\left(0.9-\frac{1}{5N}\right)$	$\pi\left(\frac{2}{3}-\frac{1}{6N}\right)$

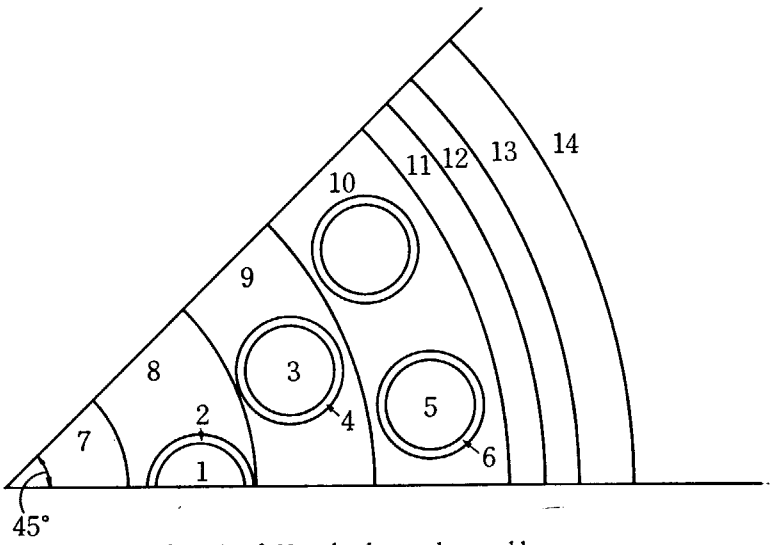


Fig. 5.6 Sample unit of 28 rods clustered assembly
(ISERI=4, IDIVP=0, NTPIN=28, NAPIN=3 and NDPIN=2)

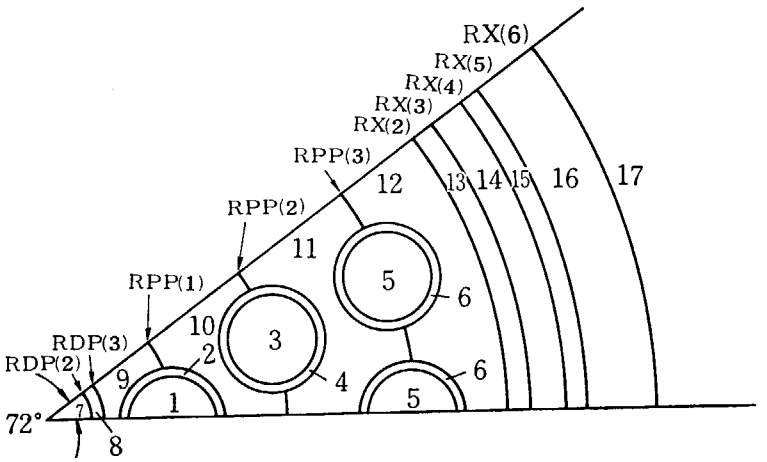


Fig. 5.7 Sample unit of 31 rods clustered assembly
(ISERI=5, IDIVP=1, NTPIN=31, NAPIN=3, NDPIN=2)

F. When the input item NTPIN is greater by one than the sum of the numbers of rods in each ring, it is assumed that one pin rod is located at the central axis of the assembly. In this case there is no need to specify the annular divisions for the center rod.

G. *Zone* numbers are assigned at first to pin rods in the innermost ring and then to those in the outer rings. After completed the numbering of pin rods in the outermost ring, the innermost annular *zone* is numbered and then the outer annular *zones*. When a rod in a ring is subdivided by annuli, and IDIVP is 0 or 1, *zones* must be numbered from

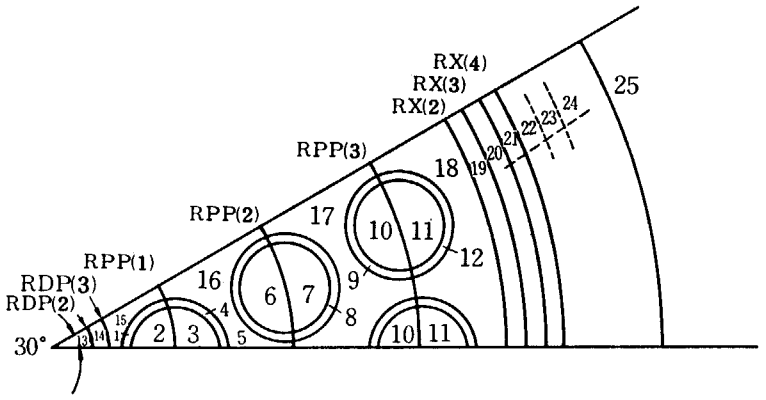


Fig. 5.8 Sample unit of 37 rods clustered assembly
(ISERI=6, IDIVP=2, NTPIN=37, NAPIN=3, NDPIN=2)

the inner of a rod to the outer of the rod. If IDIVP is 2, the rods are divided also by a circle on which the centres of the rods are placed, and *zones* must be numbered from the *zone* near to the center of the assembly. These are illustrated in **Figs. 5.6, 5.7** and **5.8**.

H. In CLUP the reflective condition is not prepared because the outer shape of the assembly is assumed to be circularly cylindrical where it has been found that the reflective condition does not work well than the isotropic one.

The procedure taken in CLUP is the same as that in PATH already described at page 26.

INPUT of CLUP

BLOCK 0 (4H)	
PROB 1-4	Program identification to be punched as 'CLUP'
BLOCK 1 (72H)	
TITLE 1-72	Problem identification
BLOCK 2	
1 NR	Number of *regions*
2 NM	Number of *materials*
3 NG	Number of energy-groups
4 NGLAST	Number of energy-groups already calculated in the previous calculation
5 IDRECT	Control of directional probability calculation:
	=1 isotropic only
	=2 isotropic and radial direction
6 IFORM	Output form of probability:
	=0 $P_{ij}(V_i \rightarrow V_j)$
	=1 P_{ij}/Σ_j
7 ITYPE	Problem type
	=0 P_{ij} only
	=1 fixed source calc. (NG=1)
8 IEDPIJ	Output unit of probabilities:
	IEDPIJ=IPR+ICD+ITAPE
	IPR=1, print
	0, skip

ICD=2, card punch
0, skip
ITAPE=4, write in unit 21
0, skip

9 ITXEC — Input unit of total cross sections
=0, card
=1, PDS file

10 NGR — Order of Gauss approximation for radial integration (=8)

11 NDA — Number of division of the range of angle (=2)

12 NGA — Order of Gauss approximation for angular integration (=4)

If any of NGR, NDA and NGA is punched zero, the values in brackets are used. The total number of NGR*NDA*NGA lines are drawn for the 'PATH table'.

13 NX — Number of annular division of the lattice cell

14 NDPIN — Number of annular division of a pin rod

15 NAPIN — Number of circular rings of pin rods

16 ISERI — Number of pin rods contained in the innermost ring
=4, 4, 8, 16 ..... rods
=5, 5, 10, 15 ..... rods
=6, 6, 12, 18 ..... rods
are contained in the first, second and third·····ring, respectively. If this item is neither 4 nor 5 nor 6 the number of pin rods is redefined by the data in BLOCK 5

17 NTPIN — Total number of pin rods in the unit lattice cell

18 IDIVP — Control of the division of pin rod *zones*:
=0 normal
=1 add RPP into RX
=2 divide pin rod by RPP's

19 NZ — Total number of *zones*

$$NZ=NAPIN*NDPINN*\left(\frac{IDIVP}{2}+1\right)+NX+NAPIN*\left(IDIVP-\frac{IDIVP}{2}\right)$$
$$+NDPIN*(NTPIN—Sum\ of\ No.\ of\ rods\ in\ rings)$$

The last term corresponds to the pin rod in the center

20 IBOUND — Control of outer boundary condition:
=0, isotropic
=1, not to be used
=2, vacuum

21 NRES — Not used

BLOCK 3 — required only if NZ≠NR

NREG(I) — *Region* number assigned to the *zone* I
I=1, NZ

The *zone* number is allocated as, at first, to the innermost pin rod ring as 1 then the second ring as 2. If NDPIN>1, the *zone* with the smaller mean distance from the center to the smaller *zone* number. Then the *zone* numbers of annular *zones* follow the numbers of pin rods rings.

BLOCK 4 (9A8) — required if ITXEC=1

NAME(I) I=1, NM	Member name in PDS file of the block of total cross sections of *material* I.
BLOCK 5	required only if ISERI>6
NPIN(I) I=1, NAPIN	The number of pin rods contained in the ring I.
BLOCK 6	
MAT(I), I=1, NR	The *material* number used in *region* I.
BLOCK 7	
RX(I), I=1, NX+1	Radii of annular division of lattice. If a pin rod is located at the center, the radii of the rod are not necessary. RX(1) must be punched zero.
BLOCK 8	
RDP(I), I=1, NDPIN+1	Radii of annular division of a pin rod. RDP(1) must be punched zero
BLOCK 9	
RPP(I), I=1, NAPIN	Radial positions of ring I from the center of the assembly
BLOCK 10	required only if ISERI>6
THETA(J, I), J=1, NPIN(I), I=1, NAPIN	Azimuthal position of the pin rod in ring I in ascending order in radian
BLOCK 11	required only if ITXEC=0 and NG≠1
SIG(N, M), N=1, NG, M=1, NM	Total cross sections. A new card is required for each *material*
BLOCK 11	required only if ITXEC=0 and NG=1
SIGT(M), SIGS(M), SIG(M), M=1, NM	Total Scattering Fission } cross sections for *material* M. A new card is required for each *material*.
BLOCK 12	required if NG=1 and if ITYPE=1
S(I), I=1, NR	Fixed source distribution

OUTPUT of CLUP

We describe the output of CLUP which is almost common among the collision probability programs.

A. LIST

1. Input data
2. Volumes for each zone
3. Number of paths drawn and the elapsed time for completing the path table
4. Ratios of numerically integrated volumes to analytic ones (CLUP 77 only)
   The formers are calculated as

$$V_{i\,(\text{numeric})}=\int\alpha\rho\int d\phi t_i$$

   where t_i is the cut of region i by a path
5. If required, $P_{ij}{}^{g}$ for $j=1$, NR, and $P_{is}{}^{g}$ for $i=1$, NR
6. If required, $P_{ijk}{}^{g}$ for $j=1$, NR, and $P_{isk}{}^{g}$ for $i=1$, NR

where k is perpendicular for plane and radial for cylindrical lattice.

Items 5 and 6 are repeated for $g=1$, NG

7. If required in case NG=1

$$\left.\begin{aligned} &\phi_i V_i \quad \text{for } i=1, \text{NR},\\ &\phi_i \quad \text{for } i=1, \text{NR},\\ &V_m=\sum_{i\in m} V_i\\ &\phi_m V_m=\sum_{i\in m}\phi_i\nu_i\\ &S_m V_m=\sum_{i\in m} S_i V_i\\ &\mathrm{A}_m=\sum_{i\in m}\Sigma_{am}\phi_i V_i \end{aligned}\right\} \text{for } m=1, \text{NM}$$

$$\overline{\nu\Sigma_f}=\sum_i \nu\Sigma_{fm}\phi_i\nu_i/\sum_i\phi_i V_i ,$$

$$\overline{\Sigma_a}=\sum_i \Sigma_{am}\phi_i\nu_i/\sum_i\phi_i V_i ,$$

$$\overline{\Sigma_t}=\sum_i \Sigma_{tm}\phi_i V_i/\sum_i\phi_i V_i ,$$

$$\overline{1/3\Sigma_t}=\sum_i \frac{\phi_i V_i}{3\Sigma_{tm}}\Big/\sum_i\phi_i V_i ,$$

$$\overline{D}=\sum_i\sum_i\phi_i V_i P_{ij}^{\sharp}/3\sum_i\phi_i V_i ,$$

$$\overline{D_k}=\sum_i\sum_i\phi_i V_i P_{ijk}^{\sharp}/3\sum_i\phi_i V_i .$$

B. PUNCH

1. TITLE
2. MAT
3. Volumes
4. P_{ij}^{g} $i=1$, NR, and $j=1$, NR
5. If the directional probabilities are required

$$DR_i^{g}=\sum_i P_{ijk}^{\sharp} \qquad i=1, \text{NR}$$

Item 4, 5 are repeated NG times

C. UNIT F21 (in binary)

1. LABELED COMMON /LAMPC/ 997 words
2. P_{ij}^{g} $i=1$, NR, $j=1$, NR
3. If the directional probabilities are required DR_i^{g} $i=1$, NR

Item 2 and 3 are repeated NG times

5.3 CLUP 77; Collision probabilities for square clustered assembly

The assembly of a light water moderated reactor consists of a finite array of square pitched rods. Moreover there are some irregularities which make uncertain the validity of the processes tried to homogenize the assembly, as performed in the two-dimensional diffusion calculation combined with the lattice cell calculation of an infinite array of rods. For example a cruciform controll rod, a poison curtain, Gd fuel rod and the channel box in the BWR type assembly, and poison rods, pin shape control rod and MD guide thimble in the PWR type assembly cause the irregularities. The program CLUP 77 is provided to solve irregularities by using collision probabilities for these complicated geometries. We offer the revised version of CLUP77 which differs from the original version[4] in the following points:

1. All input cards except the title card are read in "free format." The user's guide of

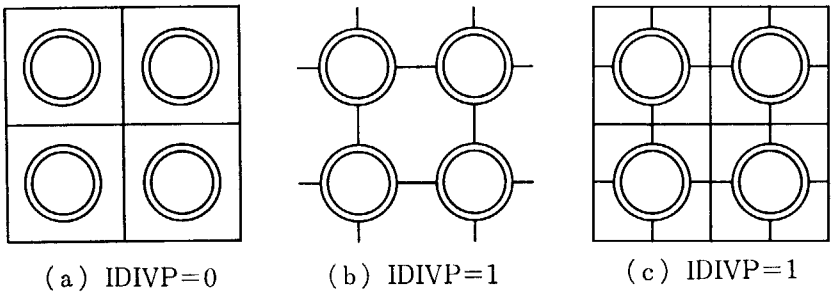


Fig. 5.9 Function of IDIVP.

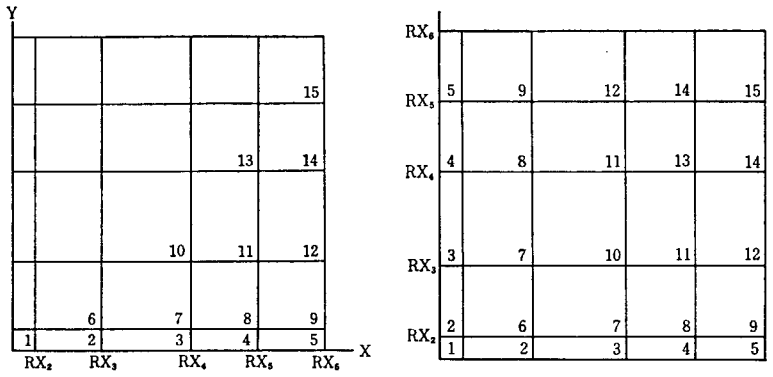


Fig. 5.10a Assignment of zone numbers for the case. NX=5 to a half quadrant of an assembly.

Fig. 5.10b Assignment of zone numbers to a remaining half quadrant by diagonal symmetry.

"The subroutine to read the date in free format" is prepared[24].

2. The order of control data is changed and a few options are added.
3. In one-group calculation, an option is added for the k-calculation as well as the fixed source calculation.
4. The moderator zones can be divided by the lines passing through the centers of the pin rods as shown in **Figs. 5.9b** and **5.9c**, while in the original version the division shown **Fig. 5.9a** is only allowed.
5. The restrictions on each interger number to specify the geometry are removed by adopting the variable dimension. Only one remained is the maximum number of *regions* which is 300.

We now describe the geometry considered. An assembly has a square cross section of infinite length and has the bilateral and diagonal symmetry. Therefore the geometry of the assembly is designed on the first half quadrant in the X-Y coordinate system and the two dimensional disposition of zones is determined by the position on only the abscissa. At first a quarter of the assembly is divided into NX$*$NX pillars with rectangular cross sections by straight lines x=RX(I+1), y=RX(I+1), I=1$\sim$NX (RX(1) is set identically zero).

A *zone* corresponds to a pillar. The *zone* numbers are allocated from right to left and from lower to upper, up to the NX$*$(NX+1)/2 rectangular pillars in the first half quadrant in X-Y coordinate system, as illustrated in **Fig. 5.10a**. The same numbers are assigned for the pillars in the remaining half quadrant according to the diagonal symmetry as illustrated in **Fig. 5.10b**. When an array (NAPIN$*$NAPIN) of the fuel rods (expressed by a nest of concentric annuli) is located in a quadrant, the position of annuli on the abscissa is assigned by RPP(I), I=1$\sim$NAPIN,

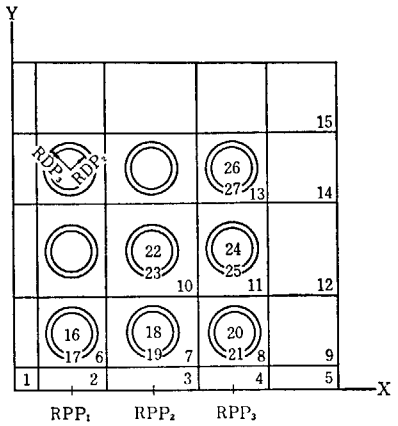


Fig. 5.11 Assignment of *zone* numbers for the case NX=5, NAPIN =3 and NDPIN=2.

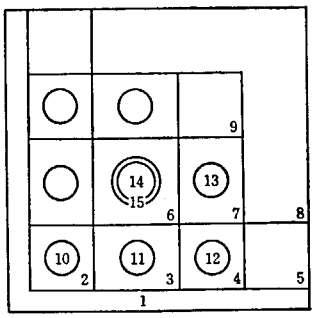


Fig. 5.12 *Region* numbers allocated to the assembly.

TABLE 5.2 Example of *region* number

ZONE	REGION	ZONE	REGION
1	1	15	8
2	1	16	10
3	1	17	2
4	1	18	11
5	1	19	2
6	2	20	12
7	3	21	2
8	4	22	14
9	5	23	15
10	6	24	13
11	7	25	7
12	8	26	9
13	9	27	9
14	8		

where NAPIN is the length of the array, and the radii of the annuli are assigned by RDP(I+1), I=1~NDPIN, where NDPIN is the number of division of a pin rod into annuli. A pin rod should be completely contained in one of the rectangular pillars, or the center of the pin rod should be at any of the cross points of x=RX(I) and y=RX(J). For the latter case, RX(I)'s on which the centers of pin rods are located are fed RPP's which are optionally used also as RX's when the input IDIVP is set to 1. In this case the input NX is changed after reading all the input of the program into NX+NAPIN or NX+NAPIN−1. The latter case occurs when RPP(1) is zero, i.e., the pin rods are located on the coordinate axes.

Following the maximum of *zone* numbers for rectangular pillars, NX(NX+1)/2, the *zone* numbers for annuli are allocated from inner to outer within a pin rod, and right to left, and then from lower to upper, up to the NDPIN*NAPIN*(NAPIN+1)/2 annuli as illustrated in **Fig. 5.11**. The total number of *zone* numbers is

$$\frac{NX*(NX+1)}{2}+NDPIN*\frac{NAPIN*(NAPIN+1)}{2}$$

The collision probabilities P_{ij}'s are calculated not for *zones* but for *regions* which consist of one or several *zones*. The correspondence of a *zone* to a *region* is assigned by a region number for each zone as illustrated in TABLE 5.2, for which example the two-dimensional disposition of *regions* is shown in **Fig. 5.12**. If the neighbouring *zones* are assigned by the **same** *region* number, the

boundary line between the *zones* disappears. In addition, a *region* number may be used in common for *zones* located at a distance. When assigning the region number to *zones* the number may be selected independently of the geometrical disposition of the *regions* but must not be vacant from 1 to NR, where NR is the total number of *regions* i.e., the dimension of the P_{ij} matrix.

INPUT of CLUP 77

BLOCK 0 (6H)

PROB 1-6 — Program identification to be punched as 'CLUP 77'

BLOCK 1

TITLE (72H) — The case identification

BLOCK 2

1 NR — Number of *regions*

2 NM — Number of *materials* for which cross sections are specified

3 NG — Number of energy-groups

4 IFORM — Definition of collision probability in the output:
=0, the normal probability P_{ij}
=1, the modified probability
$P_{ij}^{\sharp}$ $(=P_{ij}/\Sigma_j)$;
For the core capacity in PIJF stage if NR*NR*NG>37000, this item is internally set to be 1.

5 NGLAST — Continuation indicator:
Punch NGLAST if the output of the first NGLAST groups is already written in unit 20, and then to obtain the output of the remaining groups (0 otherwise).

6 IEDPIJ — Output unit indicator of P_{ij}:
IEDPIJ=IPR+ICD+ITAPE,
IPR=0, skip; =1, print,
ICD=0, skip; =2, card punch,
ITAPE=0, skip; =4, PDS file

7 NGR — Order of division for ρ integration; the larger value gives the finer integration, though more time consuming. A suggested value is "5"

8 NDA — Order of division for ϕ integration. A suggested value is "10".

9 ITYPE — Problem type indicator which is significant only for NG=1:
=0, fixed source calculation,
=1, k-calculation.

10 NX — Number of division of the side of a quardrant of a unit assembly to define the rectangular zones.

11 IBOUND — Outer boundary condition:
=0, vacuum,
=1, reflective.

12 NASS — significant if IBOUND=1
Number of assemblies traversed by a neutron which escapes from the base assembly.

13 NDPIN — Number of annular division of a pin rod.

14 NAPIN — Number of rods in an array in a quadrant of the base assembly

15 IDIVP — Rectangular *zone* division indicator; if not zero the abscissas RPP's play the role of RX's, where the formers indicate the position of pin rods and the latters divide the rectangular *zones*.

16 NMESH — Total *zone* numbers;

$$\text{NMESH}=\frac{\text{NX}'*(\text{NX}'+1)}{2}+\text{NDPIN}*\frac{\text{NAPIN}*(\text{NAPIN}+1)}{2}$$

NX′=NX, if IDIVP =0
NX′=NX+NAPIN, if IDIVP ≠0 and RPP(1) ≠0.0
NX′=NX+NAPIN−1, if IDIVP ≠0 and RPP(1) =0.0.

17 ITXEC — Input device of total cross section (=0, from card or =1, from PDS file)

18 MPTIND — If ≠0, skip to make the path table which is already in unit 12

BLOCK 3 — required only if NG=1

1 ITMAX — Maximum number of iterations allowed, L_{max} (=100)
2 ITBG — Minimum iteration before extrapolation, L_b (=5)
3 LCMX — Number of iterations tested for extrapolation, L_e (=5)
4 ITDM — Minimum delay between extrapolation, L_d (=5)
5 IPT — =0, monitor print for each iteration, =0, skip (=0).
6 EPS — Convergence precision ($=10^{-5}$)
7 RELC — Initial over-relaxation factor (=1.2)
8 EPSG — Extrapolation criterion (=0.01)
9 OVERX — Maximum extrapolation (=100.)
10 FACTOR — Under extrapolation factor (=1.0)

If ITMAX is punched zero, the values in parentheses are used

BLOCK 4
NRES(I), I=1, NMESH — *Region* number assigned to the *zone* I

BLOCK 5 (9A8) — required if ITXEC=1
NAME(I), I=1, NM — Member name in PDS file of the block of total cross sections of *material* I

BLOCK 6
MAT(I), I=1, NR — *Material* number assigned to the region I

BLOCK 7
RX(I), I=1, NX+1 — Abscissas of *zone* boundaries. [RX(1)=0.]

BLOCK 8
RDP(I), I=1, NDPIN+1 — Radii of annuli in a pin rod, required only if NAPIN≠0 [RDP(1)=0.]

BLOCK 9
RPP(I), I=1, NAPIN — Abscissas of the centers of pin rods, required only if NAPIN≠0

BLOCK 10
Q(I), I=1, NR — Fixed source density of each *region*, required only if NG=1 and ITYPE=0.

BLOCK 11

ϕ(I), I=1, NR — Initial flux distribution of each *region*, required only if NG=1.

BLOCK 12 — required if NG>1 and ITXEC=0

$\Sigma_{g,m}$, g=1, NG; m=1, NM — Total cross sections for each energy group beginning with g=1, first for the first *material*, then for the second *material* and so on. Begin on a new card for each *material*.

BLOCK 13 — required if NG=1

$\Sigma_t(m)$, $\Sigma_s(m)$, $\nu\Sigma_f(m)$; m=1, NM — The total cross section, the scattering cross section and the ν*fission cross section for the first *material*, then for the second *material* and so on. Begin on a new card for each *material*.

OUTPUT of CLUP 77

See OUTPUT of CLUP in page 33.

5.4 PIJF; Solution of the multi-group equation

The program PIJF is a Fortran program which solves the multigroup equations of the mothod of collision probabilities. Collision probabilities are not calculated but must be provided, together with all boundary conditions. It is a consequence of this requirement that the program can deal with any infinite or finite geometry for which the collision probabilities are given.

By using variable dimensioning, the program permits an optimum use of available storage. It is possible to treat fixed source or eigenvalue problems, under the assumption that both sources and scattering are isotropic.

Cross sections and source distributions are read from the PDS files or from cards. Collision probabilities are fed from unit 21 or from cards.

The method used in PIJF is already described in Chap. 3, therefore, we only describe here the usage of PIJF.

INPUT of PIJF

The input data is divided into seven sections:

A. Parameters
B. Material specification
C. Cross sections
D. Geometrical data
E. Flux guess (if any)
F. Fixed source (if any)
G. Additional activation cross section

A. Parameters

BLOCK 0 (4H)

PROB 1-4 — Program identification to be punched as 'PIJF'

BLOCK 1* (72H)

TITLE(I), I=1, 18 — This card is used for the identification purpose and may contain any information which to be print in the first line of the output print. When the input data are read from PDS file, the first four letters of TITLE(1) must be identified to the latter half of the member name of the fluxes to be read as an initial guess, It is also used in the output

fluxes for this case as 'FLUX****' where **** are the contents of TITLE(1). Moreover, the member name of the source block in PDS file is also labeled as 'SOUR****.'

BLOCK 2*

1 NR — Number of regions

2 NM — Number of materials for which cross sections are specified

3 NG — Number of energy groups

4 IDRECT — Control to read the directional probabilities:
=1, isotropic direction only,
=2, isotropic and radial directions for the cylindrical coordinate; isotropic and perpendicular directions for the plane geometry.

5 IFORM — Control to define the probabilities to be read:
=0, P_{ijg} ($V_i \rightarrow V_j$, in g-th group),
=1, modified probability
$P_{ijg}^{\sharp}=P_{ijg}/\Sigma_{jg}$.

6 ITPIJ — Control of input unit of the probabilities:
=0, card,
=1, unit 21 in binary mode.

BLOCK 3

1 NGS — Group number of the lowest energy-group of fixed sources having non-zero values. Set zero for eigenvalue problems.

2 NGK — Group number of the lowest energy-group in which neutrons from fission appears. Set zero if there is no fission source.

3 IFFG — Control of flux guess:
=0, uniform (no card required for flux guess),
=1, substituting fixed source to flux guess,
=2, NR*NG entries are required,
=3, Initial fluxes are read from PDS file.

4 IOFLUX — Control of flux output:
IOFLUX=ITF+ICDF+IPRF;
IPRF=1, print,
=0, skip,
ICDF=2, card punch,
=0, skip,
ITF=4, write on PDS file
=0, skip.

5 ICONV — Monitor of convergence:
Reaction rate of
=1, activation
=2, slowing down
=3, ν*fission
=4, collision
=5, absorption
} cross section
is used for the monitor of convergence.

6 ITXEC — Control of cross section input device:
=0, card,

=1, PDS file.

7 ITS — Control of fixed source input device:
=0, card,
=1, PDS file.

Any non zero value for the following items causes the corresponding quantities below to be printed.

8 IPTXEC — Cross sections
9 IPTV — Volumes
10 IPTPIJ — Collision probabilities
11 IPTFG — Flux guess
12 IPTS — Fixed sources

BLOCK 4 — This card sets iteration parameters:

1 ITMINN — Maximum number of inner iterations per an outer iteration (=100 for fixed source problems or =2 for eigenvalue problems)
2 ITMOUT — Maximum number of outer iterations (=1 for fixed source problem or =50 for eigenvalue problems)
3 ITBG — Minimum number of iterations before extrapolation 1_b(=5)
4 LCMX — Number of iterations for testing over-relaxation factor 1_e(=5)
5 ITDM — Minimum delay between extrapolation 1_d(=5)
6 IPT — Control of monitor print at each iteration:
If IPT>0, the record is printed (=0)
7 EOSI — Convergence criterion for the inner iteration ($=10^{-5}$)
8 EPSO — Convergence criterion for outer iteration ($=10^{-4}$)
9 IPSG — Extrapolation criterion (=0.05)
10 RELC — Initial over-relaxation factor (=1.2)
11 OVERX — Maximum extrapolation factor (=1.00)
12 FACTOR — Under extrapolation factor (=1.00)

If ITMINN is zero, the values in parentheses are used for the iteration parameters.

B. Material specification

BLOCK B.1 (9A8) — required only ITXEC=1

NAME(M), M=1, NM — Member name in PDS file of the complete set of cross sections of material M.

BLOCK B.2*

MAT(I), I=1, NR — The material number associated with each region from V_1 to V_{NR}. See the specification C the cross sections.

C. Cross sections

The complete NM sets of cross sections are read from cards if ITXEC=0 in the common format as described in **4.5.**

D. Geometrical data

BLOCK D.1*

VOL(I), I=1, NR — Volume of each region V_i.

BLOCK D.2 — required only if ITPIJ=0

P(J, I, g), J=1, NR, I=1, NR, g=1, NG — Collision probabilities, for each group in turn, beginning with group 1. Begin afresh on a new card for each I and for each group. Note that P(J, I, g)=P_{ij}^g or P_{ij}^g/Σ_j,

$P_{ij}{}^{g}$=Prob. ($V_i \rightarrow V_j$ in group g).

E. Flux guess

BLOCK E — required only if IFFG=2

ϕ_{ig}, i=1, NR, g=1, NGS — For each group in turn, beginning with group 1, give an estimate for the integral fluxes defined in page 16. Begin afresh on a new card for each group.

F. Fixed source

BLOCK F — required only if NGS≠0 and ITS=0

S_{ig}, i=1, NR g=1, NGS — For each group in turn, beginning with group 1, specify the fixed source density in each region.

Before entering to the iteration, the normalization is performed in a way that the total fixed source not escaping at least at the first collision is to be unity;

$$\sum_{g=1}^{NG}\sum_{i=1}^{NR} S_{ig}\left(1-\sum_{j=1}^{NR} P_{ij}{}^{g}\right)V_i=1\,.$$

G. Additional activation cross section

BLOCK G. 1

IRXEC — Specification of the sets of activation cross sections.
If IRXEC is positive, the cross sections are read in turn beginning with group 1 but if IRXEC is negative, they are read beginning with group NG. If IRXEC is zero, PIJF is terminated and the program identification card of the next step follows.

BLOCK G. 2

LABEL 1-32 — Comment of the set (32H)

XEC(g), g=1, NG or g=NG, 1 — Activation cross sections which may be microscopic or macroscopic

These cards are repeated IRXEC times.

Note*

BLOCK's marked by * are not required when the collision probability program (PATH, CLUP or CLUP77) and PIJF are sequentially used in LAMP.

OUTPUT of PIJF

A. LIST

1. Parameters
2. Material number
3. Cross sections (if IPRXEC. NE. 0)
4. Volume (if IPRV. NE. 0)
5. Collision prob. (if IPRP. NE. 0)
6. Flux guess (if IPRFG. NE. 0)
7. Source (if NGS. GT. 0. AND. IPTS. NE. 0)
8. Monitor of iteration (if IPT. GT. 0)
9. Information at the end of iteration
10. Flux*volume distribution
11. Flux energy distribution (homogenized)
12. Diffusion coefficient (homogenized)
13. Flux*volume spatial distribution (one group)
14. Flux spatial distribution (one group)

15. Flux*volume material by material
16. Integrated flux
17. Diffusion coefficient (one group)
18. Reaction rate
    activation, slowing down, ν*fission, total, absorption.
19. Additional reaction rate (if IRXEC. NE. 0)

B. Card in format (6E12. 5)
Integral flux distribution ϕ_{ig}

C. PDS file if ITF=4
1. Name 'CASE $ TITLE(1)'
   LABELED COMMON/LAMPC/ 997 words
2. Name 'FLUX $ TITLE(1)'
   FLUX*VOLUME, ϕ_{ig} NR*NG words

5. 5 EDIT; Auxiliary edit of PIJF

The program EDIT is offered to help the edit part of PIJF by supplying the following quantities.

1. Spatial distributions of reaction rates:
$$R_{i\mu}=\sum_g \sigma_\mu{}^g \phi_i{}^g / V_i ,$$
where the subscript μ denotes an activation reaction.
2. Multigroup isotropic and directional diffusion coefficients
$$D_g=\frac{\sum_i \phi_{ig} \sum_j P_{ij}{}^{g\sharp}}{3\sum_i \phi_{ig}},$$
$$D_{kg}=\frac{\sum_i \phi_{ig} \sum_j P_{ijk}{}^{g\sharp}}{3\sum_i \phi_{ig}},$$
where k denotes the directional component; the perpendicular or parallel to the boundary plane in the plane lattice, and the radial or axial in the cylindrical lattice.
3. Homogenized multigroup cross sections for each reaction
$$\Sigma^{g(\to g')}=\sum_i \Sigma_m{}^{g(\to g')} \phi_{ig} / \sum_i \phi_{ig},$$
The complete set of homogenized cross sections is formed in the common format. It is to be noted that the flux-volume averaged total cross section is not used here and hence in the set it is replaced by $1/3D^g$. If the directional diffusion coefficient is calculated, it is located at the position of the diffusion coefficient in the set. The difference between the averaged total cross section and $1/3D^g$ is canceled by giving the correction to the self-scattering cross section.

INPUT of EDIT

BLOCK 0 (4H)	Program identification
PROB 1-4	Card to be punched as 'EDIT'
BLOCK 1*	required only if this run of EDIT is not preceded by PIJF in this job step
1 TITLE 1-72	Columns 1 to 72 are printed as a title. For the identification in PDS file, the first four characters must be identified to that in the preceding run of PIJF

2 NR	Number of regions
3 NM	Number of sets of cross sections to be mixed
4 NG	Number of energy groups
5 IDRECT	Flag of directional collision probabilities =1, the calculation is skipped, =2, calculated.

BLOCK 2

1 IRPHI	Flag of input device for flux distribution: =1, cards, =2, PDS file,
2 IRP	Flag of input device for collision probabilities: =0, not to be read =1, card =2, unit 21 (binary)
3 NACT	Number of sets of activation cross sections to be read. If positive they are read in turn beginning with group 1 but if negative they are read in turn beginning with the lowest energy group NG.
4 IRXEC	Flag of input device for the complet sets of cross sections: =0, not to be read, =1, card, =2, PDS file.
5 IEDXEC	Flag of output device for the complete set of homogenized cross sections: =1(PRINT)+2(PUNCH)+4(PDS file)

BLOCK 3 (9A8)	required if IRXEC≠0, IRP=0 and IRPHI=1
NAME(M)	Member name of the sets of cross sections
M=1, NM	
BLOCK 4*	required if NM∗IRXEC≠0, IRPHI=1 and IRP≠2.
MAT(M)	Material number of material M
M=1, NM	
BLOCK 5*	required if IRPHI=1, IRP≠2 and NACT≠0
V(I), I=1, NR	Volume of region I
BLOCK 6	required if IRPHI=1
ϕ(I, N), I=1, NR, N=1, NG	Volume flux of region I for group N beginning afresh the card for each group
BLOCK 7	required if NACT≠0
NAME 1-72	Comment of the set of activation cross section
SIG(N)	
N=1, NG, if NACT>0,	
N=NG, 1, if NACT<0.	
	BLOCK 7 is to be repeated NACT time
BLOCK 8	required only if IRP=1
$P_{ij}^{g\sharp}$	Modified form of collision probabilities (P_{ij}^{g}/Σ_j^{g})
j=1, NR	Repeat NR∗NG times
BLOCK 9	required if IRP=1 and IDRECT=2
DR_{ig}	Directional components of diffusion coefficient; the perpendicular component for plane geometry or the radial for cylindrical geometry. These
i=1, NR	

cards may be prepared by any of collision probability program.
Repeat NG times

BLOCK 10 required if NM≠0 and IEDXEC≠0

The complete MN sets of cross sections in the common format in card.

Note * BLOCK's marked by * are not required when EDIT is called sequentially after PIJF

OUTPUT of EDIT

A. List
1. Parameters
2. Input data from card unit
3. Reaction rate (if NACT≠0) for each region
4. Isotropic diffusion coefficients (if IRP≠0), and if IDRECT≠0, the perpendicular and parallel components for plane, or radial and axial components for cylinder
5. Homogenized cross sections (if MOD (IEDXEC, 2) =1)

B. Card punch
1. Homogenized cross sections (if MOD (IEDXEC, 4) =2, 3)

C. PDS file
1. Homogenized cross sections (if IEDXEC. GT. 3) with the member name 'TMAT $ TITLE(1)' or 'FMAT $ TITLE(1)' of which the first four characters ('TMAT' or 'FMAT') follow those of the materials mixed.

5.6 PIXSE; Thermal microscopic library

The program PIXSE is derived from an English code PIXSE[7]. We divide this into the library making part and the edit part. We named the former as PIXSE and latter as PIXEDT. Since the theoretical method is the same and the brief description is already made in **4.1**, we show here only the usage.

INPUT of PIXSE

BLOCK 0 (5H)

PROB 1-5 Program identification to be punched as 'PIXSE'

BLOCK 1

1 NAME 1-4 The latter half of the member name (8H) for this case (isotope) labelled as 'TMIC $ NAME'. (4H)

2 N Number of energy groups (≤74)

3 NIN Unit of input device to read a file of $S(\alpha, \beta)$ in ENDF/A formt in EBCDIC.

4 ISAB Specification of the scattering law
−2, effective width model with $S(\alpha, \beta)$ calculated by the program
−1, gas model with $S(\alpha, \beta)$ calculated by the program
0, gas model using analytic formula for σ_0
1, tabulation of $S(\alpha, \beta)$ required as data
2, $S(\alpha, \beta)$ tabulation from the previous case

5 NG Order of Gauss approximation to be used for energy integrations in forming group cross sections. 3≤NG≤19. If NG=3, 5 or 7, Gauss weights are prepared in the program but for other values of NG they must be read as data BLOCK's 4 and 5.

6 NGA	Order of Gauss and approximation to be used for angular integration forming $\sigma(E\rightarrow E')$ from $\sigma(E\rightarrow E', \theta)$, $3\leq$NGA≤ 19. If NGA=3, 5 or 7, weights are in program but for other values of NGA, they are read as data BLOCK's 6 and 7, NGA is ignored if ISAB=0.
7 ISGMN	If $0\leq n=$ISGMN≤ 100, $\sigma_n=\int\mu^n\sigma(E\rightarrow E', \mu)d\mu$ is calculated. If $101\leq n \leq 104$, $\int P_{n-100}(\mu)*\sigma(E\rightarrow E', \mu)d\mu$ is calculated. If $n=-1$, σ_0 and σ_1 are both calculated, and the output consists of the σ_0 matrix with the total σ_1 cross section subtracted from the diagonal term.
8 INXTC	Indicator of control action after the present calculation : 1, calculate further cases, 0, end of PIXSE
9 IMOM	Ignored
10 ICASPU	Read ICASPU cross sections after forming the scattering matrix in the order absorption, $\nu*$fission, activation, elastic scattering corrections. If activation cross-sections are not read they are set to $1/\sqrt{(E_g+E_{g+1})/2}$ for each energy group
11 ISWT	Ignored if ISAB=0. Control of the interpolation of $S(\alpha, \beta)$ 1, interpolate S if IBAB>0, and read $S(\alpha, \beta)$ table, 0, interpolate $S\ e^{\beta/2}$ if ISAB>0, and read $S(\alpha, \beta)$ table, -1, interpolate $S\ e^{\beta/2}$ if ISAB>0, and read $S(\alpha, \beta)\ e^{\beta/2}$ ISWT is inefficient if ISAB<0
12 λ	Debye-waller coefficient. Set equal to 0.0 if not required
13 σ_f	Free atom scattering cross section
14 ζ	'Groups' with $E_2=\zeta E_1$ used for calculating the source and upscattering
15 T	Temperature (°K)
16 A	Mass of scatterer (in atomic unit)
17 ε	Set group transfer cross sections zero if the calculated values $<\varepsilon$. If $\varepsilon=-1.0$, PIXSE is terminated
18 y	Specification of the weighting function used in forming group average If $y>0$ Maxwell for $E/kT<y$ and $1/E$ for $E/kT\geqq y$, If $y=-1$, $\phi=1.0$, i.e., unit weighting.
19	not used
20 d	Reciprocal of the width in effective width model, $1/q$; (ignored unless ISAB=-2).
BLOCK 3 E(I), I=1, N+1	Group boundaries in eV in ascending order of energy.
BLOCK 4 WG(L), L=1, NG	Not required if NG=3, 5 or 7.
BLOCK 5 AGW(L), L=1, NG	Weights and ordinates (0, 1.0) for energy integration
BLOCK 6 WGA(L), L=1, NGA	Not required if NGA=3, 5 or 7.

BLOCK 7
AWB(L),
L=1, NGA — Weights and ordinates (−1, 1) for angular integration

BLOCK 8 — required only if ISAB=1.
Tabulated $S(\alpha, \beta)$ in ENDF/A format (BNL-8381)

Heading Cards — If NIN=5 (card reader), heading cards must be removed from input data

DCC2 card the number of β values,
DCC1 card with the first value of β,
values of α and then S,
DCC1 card with second value of β,
Values of S.
⋮
End card.

The program GASKET[20] supplies the scattering law in the required format.

If INXTC=1, BLOCK 1 card for the next case follows, otherwise PIXSE terminates.

The thermal library in PDS file

The PIXSE forms a block in PDS file in a following way

MEMBER NAME———'TMIC $ NAME'
Length N*(N+4)
Contents

$\sigma_{1\to1}\ \sigma_{1\to2}$ ····· $\sigma_{1\to N}$
$\sigma_{2\to1}\ \sigma_{2\to2}$ ····· $\sigma_{2\to N}$
⋮
$\sigma_{N\to1}\ \sigma_{N\to2}$ ····· $\sigma_{N\to N}$
$S_1,\ S_2$ ····· S_N
$\sigma_{a1}\ \sigma_{a2}$ ····· σ_{aN}
$\nu\sigma_{f1}\ \nu\sigma_{f2}$ ····· $\nu\sigma_{fN}$
$\sigma_{act1}\ \sigma_{act2}$ ····· σ_{actN}

5.7 PIXEDT; Thermal cross section and source distribution

PIXEDT is an edit program for the thermal energy region. It forms the macroscopic cross sections and the source distributions. The microscopic data must have been stored in PDS file as the thermal library.

INPUT of PIXEDT

BLOCK 0 (6H) — Program identification to be punched as 'PIXEDT'
PROB 1-6

BLOCK 1

1. MNOM 1-4 — Name of the mixture (*material*) to be formed. It will be used as the latter half of member name in PDS file as 'TMAT $ MNOM' for the complete set and as 'TTOT $ MNOM' for the total cross sections. (4H)
2. NISO — Number of isotopes to be mixed in this mixture.
3. IEDT — Signed number indicating the output device
=1(PRINT)+2(PUNCH)+4(PDS file)

If IEDT>0, after the final BLOCK 2, BLOCK 1 for the next mixture follows. If IEDT<0 the data for the source distribution follow. If IEDT=0 the end of PIXEDT.

BLOCK 2

1. ID — Isotope identification. The member name 'TMIC $ ID' is searched in PDS file. (4H)
2. WS — Atomic number density

Repeat BLOCK 2 NISO times.

BLOCK 3

1. NAME 1-4 — Case identification of the source distribution. These four characters are to be identified to the first four characters in TITLE in the input of the programs (PIJF or TUD) to use this source. The member name of the source distribution is 'SORC $ NAME' in PDS file. (4H)
2. NR — Number of regions
3. MA — Source shape indicator; if MA=0 source density is unity in each region, if MA>0 source density is read from cards, and if MA<0 it is read from PDS file.
4. IEDT — Signed number indicating the output device. /IEDT/=1(PRINT)+2(CARD)+4(PDS file)

BLOCK 4

NWS(I), I=1, NR — The mixture number of the source required for each region

BLOCK 5 — required if MA>0

S(L), I=1, NR — Source density

BLOCK 5′ — required if MA<0

SNAME 1-8 — Member name of the source density to be read from PDS file (8H)

If IEDT>0, BLOCK 3 follows for the next source but if IEDT<0, PIXEDT terminates.

5.8 FAXSE; Compilation of the fast neutron library

The program FAXSE compiles the nuclear data in fast energy region into the fast library as described in **4.2.** The contents of the library are listed in **TABLE 5.3.** In the library the arrangement of a scattering matrix is designed to economize the storage. As illustrated in **Fig. 5.13** the parameters LD and LA take an impotant role. In Record 4 they are ordered as$\cdots$, A_{11}, A_{21}, $\cdots A_{m1}$, $A_{22}\cdots A_{m+1,2}$, $\cdots\cdots$, A_{nn}, $A_{n+1,n}$, $\cdots\cdots A_{n+m-1,n}$, $\cdots\cdots$

TABLE 5.3 Contents of library tape

Record 1	NITL	No. of nuclides
	NFS	No. of fission spectra
	NNL	No. of energy groups
Record 2	(FSN(I), I=1, 12)	Description of fission spectrum
	AFSS(I), I=1, NNL	Fission spectrum
	Repeat Record 2 NFS time	
Record 3	BCDW(I), I=1, 4	Description of nuclide (16H)
	NID	I. D. no. of nuclide
	LTOT	Total length of data
	IWA	Absorption data check
	IWF	Fission data check

TABLE 5.3 Continued

	IWR	Shielding factor data check
	(LOL(I), LA(I), LD(I), I=1, 4)	Length of matrix, Width of across matrix, Length of down matrix, I=1 for N−N, I=2 for N−2N, I=3 for P_0, I=4 for P_1
Record 4	(ADUM(I), I=1, LTOT)	Microscopic cross section data
	If IWR≠0, Record 4 is followed by Record 5 and 6.	
Record 5	IFA, IFF, IFE	Absorption, Fission, Elastic } tabulation check
	LTEMP	Number of temperatures used
	LSF	Number of σ_0's used
	NLMIN	
	NLMAX	
	TEMP (1), TEMP (2), TEMP (3)	Temperature
	SIGE (1), SIGE (2), SIGE (3), SIGE (4), SIGE (5)	σ_0's
	((IFTAB(I, N), I=1, 4), N=1, NNL)	
Record 6	(((SFA(J, N, I), J=1, 5), N=1, NNL), I=1, 3)	
	(((SFF(J, N, I), J=1, 5), N=1, NNL), I=1, 3)	
	(((SFE(J, N, I), J=1, 5), N=1, NNL), I=1, 3)	

Repeat Record 3-6 NITL times.

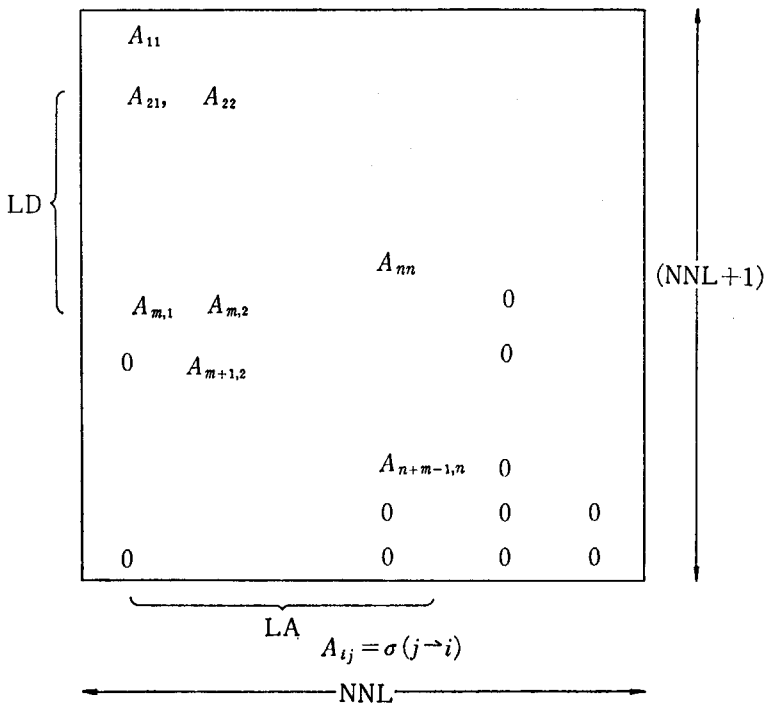


Fig. 5.13 Ordering of scattering matrix with LD(I)=$m-1$ and LA(I)=n

INPUT of FAXSE

BLOCK 0 (5H)

PROB 1-5 Program identification to be punched as 'FAXSE'

BLOCK 1

1. NAC	Number of isotopes to be corrected or added to the library
2. NISO	Number of isotopes already contained in the library
3. NSS	Indicator for fission source spectrum : =0, no change is made, =N, N-th source spectrum is corrected, =−N, N additional spectra is added to the library.
4. IPRINT	Print control : =0, no print, =1, print only new data, =2, print all data in the library.
5. NNL	Number of energy groups

BLOCK 2

FSN cols. 1-48	Description of the source spectrum being corrected or added (48H)
(AFSS(I), I=1, NNL)	Source spectrum for NNL groups. Repeat BLOCK 2 NSS times for each source to be added to the library.

BLOCK 3

(ID(J), J=1, NAC)	These entries give the identification numbers of the isotopes to be corrected or to be added. The ID's must be in ascending order. The following data of the isotopes are ordered according to ID numbers.

BLOCK 4

1. BCDW cols. 1-16	Description of the isotope. (16H)
2. NID	Identification number of the isotope to be identified to any of ID's in BLOCK 3. (integer)
3. LTOT	Total length of data record 4 LTOT=(IWA+2*IWF)*NNL+LOL(1)+LOL(2)+LOL(3)+LOL(4)
4. IWA	=0, no absorption tabulation, =1, absorption tabulation,
5. IWF	=0, no fission tabulation, =1, fission tabulation.
6. IWR	=0, no shielding factor tabulation, =1, shielding factor tabulation.
7, 8, 9 and 10. LOL(I), I=1, 4	Lengths of matrices I=1, for N−N matrix, I=2, for N−2N matrix, I=3, for elastic P_0 matrix, I=4, for elastic P_1 matrix.
11, 12, 13 and 14 LA(I), I=1, 4	The numbers of group in which each scattering process occurs.
15, 16, 17 and 18 LD(I), I=1, 4	The numbers of group to which down-scattering may occur (not counting self-scattering).

BLOCK 5

ADUM(I) I=1, LTOT	Contents of ADUM are composed of the following arrays.

NNL*IWA entries for absoption,
NNL*IWF entries fission,
NNL*IWF entries ν,
LOL(1) entries N−N array,
LOL(2) entries N−2N array,
LOL(3) entries P_0 array,
LOL(4) entries P_1 array.

Scattering arrays are formed as
(1 to 1), (1 to 2), (1 to 3), ·····
(1 to J), (2 to 2), (2 to 3), ·····
(2 to J), (2 to J+1), ·····etc.

The following BLOCK's are required only if IWR (in BLOCK 1)≠0

BLOCK 6

1. IFA Absorption
2. IFF Fission
3. IFE Elastic

If punched as 1, the self-shielding table of the corresponding reaction is read.

4. LSF Number of σ_0's in the table (≦5).
5. NLMIN The highest energy group in the table.
6. NLMAX The lowest energy group in the table.
7. LTEMP Number of temperatures in the table (≦3).

BLOCK 7 required only if LTEMP>1
(TEMP(I), I=1, LTEMP) Temperatures in the table.

BLOCK 8
(SIGZ(I), I=1, LSF) Values of σ_0's. The greatest must be read first

BLOCK 9 required if IFF≠0
(((LSFA(J, N, I), j=1, LSF), N=NLMIN, NLMAX), I=1, LTEMP) Shielding factors for absorption. Refresh a card at each energy group N.

BLOCK 10 required if IFA≠0
(((SFF(J, N, I), J=1, LSE), N=NLMIN, NLMAX), I=1, LTEMP) Shielding factors for fission

BLOCK 11 required if LFE≠0
(((SFE(J, N, I), J=1, LSF), N=NLMIN, NLMAX), I=1, LTEMP) Shielding factors for elastic scattering

Repeat cards BLOCK 4 to BLOCK 11, NAC times for each isotope of which new cross sections are supplied by input cards.

5.9 FAXEDT; Fast cross section and source distribution

FAXEDT is an edit program for the fast energy region. It forms the macroscopic cross sections and if required, the source distributions by using the microscopic data in the fast library. The final output as described in **4.4** is utilized in transport and diffusion calculations.

The heterogeneous effect to the self-shielding factor can be taken into account simply by the mean chord length (see **4.2**). When the effective cross sections are prepared by more rigorous

treatment, the data derived from the library may be replaced by them in forming the mixture cross sections.

The in-group scattering cross sections $\Sigma_{G\to G}$ of the mixture is modified by the first order transport correction for anisotropic scattering so that the isotropic transport equation can be solved.

In the tail of the energy transfer vector $\Sigma_{G\to G'}$, we put the slowing down cross section $\Sigma_{G\to NG+1}$ (corresponding to the reaction that collision takes place at group G and the energy of emitted neutron is lower than the limit of the lowest energy group NG). The fission spectrum of each isotope is mixed in the mixture by weighting by the fission rate. The collapsing of energy groups with or without weighting can be performed as illustrated in TABLE 5.4.

The FAXEDT may provide source distributions due to thermal fission. A source spectrum of a mixture is simply averaged as

$$\chi_m{}^g=\sum_\mu \chi_{\mu(\text{thermal})}{}^g N_\mu / \sum_\mu N_\mu$$

In addition, on specifying the mixture number and the weight of each region; WT_i, we get the fixed distributed source as $S_i{}^g = WT_i \chi_m{}^g$

TABLE 5.4 Energy group structure

Fine group		Coarse group	
NGLD	Group index (g)	Group index (G)	NGD (INPUT)
1	1	1	2
1	2		
2	3	2	3
2	4		
2	5		
3	6	3	2
3	7		
4	8	4	1
5	9	5	2
5	10		
⋮	⋮	⋮	⋮
NGR	NG−1	NGR	2
NGR	NG		
NGR+1	NG+1		
NGR+1	NG+2		
⋮	⋮		
NGR+1	NNL (=68)		
NGR+1	NNL		

INPUT of FAXEDT

BLOCK 0 (6H) Program identification to be punched as 'FAXEDT'

PROB 1-6

BLOCK 1

1. TITLE 1-48 Case identification printed as title. (48H)
2. NGR Number of coarse energy groups to form macroscopic cross sections

BLOCK 2

NGD(L) L=1, NGR	Number of fine groups in the L-th coarse group. The lowest energy group (=total number of fine groups) is the sum of NGD's which may less than NNL (number of groups in the library)

BLOCK 3

1. MNOM 1-4	Name of the mixture (*material*) to be formed. It is used as the latter half of member name in PDS file as 'FMAT $ MNOM' for the complete set and as 'FTOT $ MNOM' for the total cross sections. (4H).
2. IND	Indicator after forming mixture: =−2, last mixture and save source, =−1, last mixture but do not save source, =0, FAXEDT is terminated after reading this BLOCK, =1, form further mixtures but do not save source, =2, form further mixtures and save source.
3. NIS	Number of isotopes to be formed in the mixture
4. IWN	Indicator for weighting the fine groups to form coarse group constants: =0, WT(N)=1.0, N=1, NG, =1, read WT(N), =−1, use the same weights for the previous mixture.
5. IEDT	Indicator of output device for mixture cross sections =1(PRINT)+2(PUNDH)+4(PDS file)
6. XLBAR	Mean chord length used for the heterogeneous effect to the self-shielding factor (see **4. 2**)
7. TEMP	Temperature (°K) used for the shielding factor tabulation

BLOCK 4 required only if IWT=1

WT(N), N=1, NG	Weights for fine groups

BLOCK 5

1. NID	The identification number assigned to this isotope in the library (integer)
2. NFSS	If non zero, the NFSS-th fission spectrum is used for this isotope
3. NTSS	If non zero, the NTSS-th fission spectrum is used to form the source distribution due to thermal fission
4. NAMIN 5. NAMAX	If non zero, the effective microscopic absorption cross section from |NAMIN|-th fine group to NAMAX-th group is read from card (if NAMIN>0) or from PDS file (if NAMIN<0)
6. NFMIN 7. NFMAX	The same action for fission as NAMIN and NAMAX for absorption. If NAMAX=0 or NFMAX=0 they are set to NG by the program
8. WS	Atomic number density of the isotope in the mixture

BLOCK 6 required only if NAMIN>0

ADUM(I), I=NAMIN, NAMAX	Effective absorption cross sections

BLOCK 6′ required if NAMIN<0

NOMR 1-8	Member name of the effective absorption cross sections in PDS file. (8H)

BLOCK 7 required if NFMIN>0

ADUM(I) I=NNL+NFMIN, NNL+NFMAX	Effective fission cross sections

BLOCK 7′ required if NFMIN<0

NOMR 1-8 Member name of the effective fission cross sections in PDS file. (8H)

Repeat from BLOCK 5 to BLOCK 7′ NIS times. If IND>0, BLOCK 3 of the next mixture follows. The following cards specify the distributed source due to thermal fission which are used for multi-group transport or diffusion calculations.

BLOCK 8

1. NOMS 1-4 The latter half of member name for the source distribution to be stored in PDS file as 'SORC $ NOMS'. (4H)
2. NR Signed number of regions
3. KW The source shape indicator: if KW=0, the weight of source density is unity, but if KW>0 the weight is supplied for each region.
4. IEDTS Indicator for output device for the source
   =1(PRINT)+2(PUNCH)+4(PDS file)

BLOCK 9

(NWS(J), J=1, |NR|) The mixture number of J-th region.

The mixture number is numbered according to the order of forming the mixture.

BLOCK 10 required only if KW>0

(S(J), J=1, |NR|) Weights for the source density.

If NR is negative, repeat BLOCK 8 to BLOCK 10 for the further spatial source distribution. If NR is positive, FAXEDT is terminated.

5.10 PRERI; Total cross section for resonance integral calculation

PRERI is a small program to prepare the tabulation of macroscopic total cross sections which are used in any of the collision probability programs. The RICM, the resonance integral program requires the tabulation of collision probabilities which must cover the wide changes of cross section in the resonance region whereas the cross sections of moderator regions are assumed as energy-independent.

The length of table (limited less than or equal 15) is taken as the number of energy groups by the collision probability program. This program thus simplifies the preparation of the total cross sections.

INPUT of PRERI

BLOCK 0 (5H)

PROB 1-5 Program identification to be punched as 'PRERI'

BLOCK 1

1. LABEL 1-4 The latter half of member name of total cross sections for resonant material used as 'TRES $ LABEL' (4H)
2. NG Length of table ($\leqq$15)
3. NM Number of mixtures except resonant materials
4. NISO Number of isotopes to form a mixture ($\leqq$20)
5. NRES Indicator to calculate total cross sections for resonant material:
   =0, skip. LABEL of BLOCK 1 is not used,
   =1, calculate them.

BLOCK 2 required if NRES=1 and NG<15

Z(N), N=1, NG Σ_r(N)=Z(N), for Z$\leq$1.0,

$\Sigma_r(N)=1./(2.-Z(N)$, for $Z>1.0$,
where Σ_r=total cross section of resonant material

BLOCK 3
SIGP(M) Microscopic potential cross section of isotope M
M=1, NISO
BLOCK 4 required NM times
1. LABEL 1-4 The latter half of member name of the total cross sections for the mixture labeled as 'TRES $ LABEL'. (4H)
2. DN(M) Atomic number density of the isotope M to form this mixture
M=1, NISO

OUTPUT of PRERI

1. List of input data.
2. List of member names written in PDS files.
3. Total cross sections formed are written in PDS file.

5.11 LTE; Preparation of the resonance cross sections for RICM

The LTE is a modified version of the original LTE[11] which prepares pointwise resonance cross sections for RICM.

The modification is performed as follows.

1. The Doppler broadened shape functions $\psi(\xi, \chi)$ and $\chi(\xi, \chi)$, are interpolated by the subroutine QUICKW which is a part of RABBLE[21]. The table of ψ and χ is stored in the permanent file (read by $ DISKTO F18, J1480. WLIB).
2. Resonance level parameters are read in ENDF/B format.
3. P wave level can also be treated.
4. The energy range for numerical integration and the energy mesh are determined automatically.
5. Unresolved part is omitted.

INPUT of LTE

BLOCK 0 (3H)
PROB 1-3 Program identification to be punched as 'LTE'
BLOCK 1
TITLE 1-72 Problem identification
BLOCK 2
NOEL Number of elements (a set of the given angular-momentum L-state is counted as an element)
IPR If non zero, the interpolated values of Doppler broaded shape functions, ψ and χ, are printed.
ECUT Lower energy boundary
BLOCK 3
ELID(J) 1-8 Element identification (8H)
ELCT(I, J) 9-72 Description of J-th element (64H)
I=1, 16, J=1, NOEL BLOCK 3 is to be repeated here NOEL times
BLOCK 4
TEMP Temperature (°K)

R1, R2: Criterions Δx, the mesh width of the argument of Doppler functions. The width is adjusted as

$$R1 < \text{Max of } (\phi_{i-1}/\phi_i) < R2$$

S1: Initial value for adjusting Δx used as

$$\Delta x_{new} = \Delta x_{old} / [\text{Max of } (\phi_{i-1}/\phi_i)/(R1 * R2) * * S1/2]$$

where S1 is changed iteratively

S2: Parameter for selecting the number of points, NU, which is determined as $\sigma_0\phi_{NU-1} > S2 * \sigma_p > \sigma_0\phi_{NU}$

If any parameter in this card is punched as zero, the following value is set: TEMP=239., R1=1.8, R2=2.2, S1=1.0 and S2=0.1.

BLOCK 5 (E11.4)

ZA	1-11	(Z, A) designation for the element

BLOCK 6 (2E11.4)

EL	1-11	the lower limit of the resolved resonance range
EH	12-22	the upper limit of the range

BLOCK 7 (2E11.4, 22X, I11)

SPIN	1-11	the nuclear spin of the element
AP	12-22	the scattering radius
NLS	45-55	the number of sets of resonance parmeters given in this range

BLOCK 8 (E11.4, 11X, I11, 22X, I11)

AMASS	1-11	the ratio of the mass of a particular isotope to that of a neutron
LS	23-33	the value of L (neutron angular momentum number)
NRES	56-66	the number of resolved resonances for the given L-state

BLOCK 9 (6E11.4)

ER	1-11	the resonance energy
AJ	12-22	the floating-point value of J (spin of the resonance)
GT	23-33	the resonance total width Γ evaluated at the resonance
GN	34-44	the neutron width Γ_n
GG	45-55	the the radiation width Γ_γ
GF	56-66	the fission width Γ_f

BLOCK 9 must be repeated NRES times, and BLOCK's 8 and 9 NLS times. BLOCK's 4-9 must be repeated for the required elements.

NOTE

1. BLOCK 5—BLOCK 9 are in ENDF/B format.
2. Maximum of (ECUT in BLOCK 2 and EL in BLOCK 6) is used as the lower limit of the resonance range.
3. The array of Doppler broadened shape function for each level is calculated by the subroutine QUICKW which uses a six-point interpolation formula in $t_w(62\times62)$ tables of the real and imaginary part of $W(\xi, \chi)$, which should be prepared on File unit 18.
4. File unit 4 is a work area and output File unit 36 is used by RICM.

Limitation

NOEL ≤ 20
NRES ≤ 300
NLS ≤ 2
Number of meshes for ϕ and χ ≤ 150

5.12 RICM; Resonance integral

RICM is a modified version of the original RICM[11] which calculates the resonance absorption integral of a resonant isotope in a multiregion lattice.

The modifications performed are follows.

1. Up to 20 regions available (originally 5)
2. Collision probabilities are fed from the programs in LAMP, in which the geometries and boundary conditions are already included.
3. Unresolved part is omitted.
4. In final output, the group cross sections mixed with the smooth part which should be supplied by card, are now punched out or written in PDS file.

INPUT of RICM

BLOCK 0 (4H)

PROB 1-4 Program identification to be punched as 'RICM'

BLOCK 1

RSELID 1-8 Identification of the resonant isotope for which resonance calculation is to be done. This item is used for positioning the library file (8H)

PROBID 9-72 Problem identification which is printed on the top line of every page (64H)

BLOCK 2

JM Number of regions

NXR Number of sets for collision probability tabulation

LM Number of isotopes used in this case, inclusive of the resonant isotope

ICX Choice of cross section of the resonant isotope

ICX=ICX1+ICX2:

ICX1=1, with interference scattering

=0, without it

ICX2=2, with $\sqrt{E_r/E}$ factor

=0, without it

ICLV Specification of levels for which resonance calculation are to be done

ICLV=0, for all levels,

ICLV=1, for levels specified by BLOCK 8.

NPABC Specification of INPUT devices for collision probabilities and volumes

NPABC	P_{ij}	V
0	No	No
1	Cards	Cards
2	F21	F21

ICMG Specification of edit for multigroup cross section:

=0, no edit,

=−1, edit for the installed 42 groups beginning with the 25th to 66th group of 68 equal interval GAM-I groups,

=N edit for N groups specified by the following EMGMAX and DU array

ICOP Specification for printing the detailed information for each level:

ICOP=0, not required,
ICOP=1, required.

CL — $\hat{l}$ ($=2V_r/S$), half mean chord length of resonant region (in cm), which is multiplied by the total cross section of resonant region to be used as an argument of the interpolation of the collision probabilities. This item must be identified with that in PRERI which fixes the value as 1.0

XE — The effective escape cross section of the resonant region ($=S_{eff}/4V_r$) which is used for the calculation of the practical width

SA2200 — The 2200m value of the additional $1/v$ absorption. The RICM rejects the calculation of the level of the resonance energy lower than the cut off energy. The contribution of such a level to the effective cross section in the resonance range is considered by the optional use of this item

SF2200 — The 2200m value of the additional $1/v$ fission cross section

EMGMAX — The energy corresponding to the zero lethargy which is used if ICMG>0

BLOCK 3 — required if JM>1

ICRM(J) J=1, JM — ICRM(J)=0, J-th region is resonant,
ICRM(J)=1, J-th region is moderator.

BLOCK 4 — required if JM>1.

ICSD(J), J=1, JM — ICSD(J)=0, the flux in J-th region is calculated,
ICSD(J)=1, the flux J-th region is fixed as $1/E$.

BLOCK 5 — requird if JM>1 and NPABC=1

VOL(J), J=1, JM — Volumes of regions

BLOCK 6 — required if NXR≠0

X(N), N=1, NXR — The arguments of the tabulation of collision probabilities; $P_{ij}(X_n)$:
$X_n=\hat{l}*\Sigma_{tn}$, for $X_n\leq 1.$,
$X_n=2-1/(\hat{l}*\Sigma_{tn})$, for $X_n>1.$.

If NXR (in BLOCK 2) is punched zero, NXR is set to 15 and the installed values for X_n is used.

BLOCK 7 — required if NPABC=1

P(I, J, N), I=1, JM — Collision probabilities from J-th to I-th region

BLOCK 7 must be repeated JM*NXR times, changing first J from 1 to JM, then changing N from 1 to NXR.

BLOCK 8 — required if ICLV=1

ICL(N) N=1, NLVL — ICL(N)=0, calculation to be skipped for the N-th level,
ICL(N)=1, calculation is to be done
NLVL=the total number of levels of the element RSELID stored in the library file F36.

BLOCK 9 must be repeated for LM times. In addition the ELID(L) corresponding to the resonant isotope must be identified to RSELID.

BLOCK 9

ELID(L) 1-8 — Identification of the L-th isotope (8H)

ICSD(L) — Specification for the treatment of slowing down process
=0, exactly,
=1, NR approximation,
=2, WR approximation,

=3, NR ($\Delta \geq 5$), exactly ($5 > \Delta > 0.2$) and WR ($0.2 \geq \Delta$),
where $\Delta = (1-\alpha)E_r/\Gamma_p$.

AM(L) The atomic mass

SP(L) The microscopic potential scattering cross section, σ_p

ON(L, J), J=1, JM The atomic number density in the J-th region

BLOCK 10 required if ICOP=1

ICO(N) =0, no detailed information,

N=1, NLVL =1, print out E_r, E_1, ΔE, σ_0, ξ, etc. ……(O_1),

=2, 3 and 4, print out E_n, σ_{an}, σ_{sn}, σ_n, $\phi_j{}^n$ ……$(O_2)+(O_1)$

=5, 6 and 7, print out E_n, $\sigma_{an}\phi_j{}^n$ ……$(O_3)+(O_1)$,

=8 and 9, $(O_1)+(O_2)+(O_3)$.

The larger the number in each group the coarser the meshes for the print list.

BLOCK 11 required if ICMG>0

DU(I) the lethargy width of the group I

I=1, ICMG $\Delta U_i = U_i - U_{i-1}$

BLOCK 12 required if ICMG≠0

NGA Control for additional absorption cross section:

=0, no further additional absorption cross section,

=±N, additional absorption cross sections to be read from the N-th to the last (ICMG) group.

If N is positive, additional ones are read from card but if negative, those are from PDS file.

BLOCK 13 required if NAG>0

SIGA(I) The additional microscopic absorption cross sections of the resonant

I=NGA, ICMG isotope

BLOCK's 12 and 13 are repeated until NGA punched as zero is read in. Thus the accumulated (resonance integral part+1/v+additional) absorption cross section is prepared for groups from the final non-zero NGA-th to the ICMG-th group.

BLOCK 13′

XNOM 1-8 Member name of additional absorption cross sections to be read from PDS file (8H)

BLOCK 14

IEDT Indicator of output device for the group averaged absorption cross section:

if IEDT=0, print only,

IEDT=1, card punch,

IEDT=−1, PDS file.

BLOCK 15 required if IEDT=−1

XNOM 1-8 Member name of the group averaged cross section to be written in PDS file (8H)

The following BLOCK's from 16 to 19 are for the fission cross sections as BLOCK's 12 to 15 for the absorption. They are required when the resonance fission integral is processed.

BLOCK 16 corresponds to BLOCK 12

BLOCK 17 corresponds to BLOCK 13

BLOCK 17′ corresponds to BLOCK 13′

BLOCK 18 corresponds to BLOCK 14

BLOCK 19 corresponds to BLOCK 15

5.13 EPISPC; Smeared cross section in fast and epithermal energy region

The fast fission effect to a thermal reactor is prominent in the heterogeneous lattice cells. To calculate it simultaneously with the resonance shielding is very difficult due to the energy group structure. For the heterogeneous effect the division into a few ten's groups is enough but for resonance shielding calculations more than ten thousand's points are required. Fortunately the energy ranges where they have their effects are different from each other. The fast fission effect should be calculated in the energy range higher than the fission threshold of fertile material. We obtain the resonance integrals of the lattice but it is unreasonable to use them as the cross section of constituent materials for the transport equation to solve the spatial distribution in the lattice cell. The program EPISPC calculates the spectrum of homogenized system and the homogenized cross sections by considering the fast fission effect and the resonance shielding effects. Among the information in the fast fission region obtained by PIJF, only the flux distributions are applied in the EPISPC to smear the lattice but the energy spectrum of the fluxes is not used. The effective absorption and fission cross sections of the resonance region are used to replace the corresponding data in the fast library. The composition of the lattice must be specified again to read the fast library.

We now follow the steps.

1. Fluxes are read in by region and by coarse or fine energy-group.
2. Fluxes are arranged by material and by fine energy-group. If the input fluxes are of coarse groups, they are assumed as flat over the corresponding fine groups.
3. Material compositions are read.
4. The library is read by material and if the effective cross sections are prepared the corresponding data in the library are replaced by them.
5. The homogenized cross-section are composed as

$$\Sigma_{\text{homo}}^{g}=\sum_{m}\phi_{m}^{g}\Sigma_{m}^{g}/\sum_{m}\phi_{m}^{g}$$

where summation is performed over the material region which consists of the regions to which the same material number is assigned.

If the fluxes are not specified, they are assumed to be flat spatially.

6. The spectrum are calculated as

$$\phi_{g}=S_{g}/(\Sigma_{r}^{g}+B^{2}/3\Sigma_{tr}^{g}),$$

where the source

$$S_{g}=\sum_{g'=1}^{g-1}\phi_{g'}\Sigma_{s}^{g'\to g}+\chi^{g},$$

χ^{g}=the fission spectrum.

7. If the coarse group structure is given, the coarse group constants are obtained as

$$\Sigma_{\mu}^{G}=\sum_{g\in G}\Sigma_{\mu}^{g}\phi_{g}/\sum_{g\in G}\phi_{g},$$

where

Σ_{μ}^{G}=the coarse group constants for reaction μ,

Σ_{μ}^{g}=the fine group constants for reaction μ.

INPUT of EPISPC

BLOCK 0 (6H)

PROB 1-6 Program identification to be punched as 'EPISPC'

BLOCK 1

TITLE 1-72	Problem identification. The first four characters are to be identified with that of PIJF so that the fluxes can be read from PDS file. (72H)
NG	Number of fine groups, i.e., the NG-th group in the library is the lowest group
NGUP	The fluxes are specified unitl the first NGUP fine groups
NXI	The NXI-th fission spectrum in the library is used
BSQ	If non-zero, the bucklings used in the spectrum calculations to consider the leakage effect
BLOCK 2	required if the fluxes are supplied on the coarse NGRU groups from PIJF
NGD(L)	Number of fine groups in the L-th coarse group
L=1, NGRU	It is to be identified to BLOCK 2 in FAXEDT which is used for the PIJF calculation
BLOCK 3	
1 MNOM 1-4	Mixture name used only for printing (4H)
2	Not used
3 NIS	Number of isotopes to form the mixture
4	Not used
5	Not used
6 XLBAR	$\hat{l}$, mean chord length used for considering the heterogeneous effect to the shielding factor (see 4. 2)
7 TEMP	Temperature (°K) also used in the shielding factor tabulation
BLOCK 4	
1 NID	The identification number (integer) assigned to this isotope in the library
2	Not used
3	Not used
4 NAMIN 5 NAMAX 6 NFMIN 7 NFMAX 8 WS	The same action as used in BLOCK 5 of FAXEDT
BLOCK 5	required if NAMIN>0
ADUM(I)	Effective absorption cross section
I=NAMIN, NAMAX	
BLOCK 5′	required if NAMIN<0
NOMR 1-8	Member name of the effective absorption cross sections in PDS file (8H)
BLOCK 6	required if NFMIN>0
ADUM(I)	Effective fission cross section
I=NFMIN+NNL, NFMAX+NNL	
BLOCK 6′	required if NFMIN<0
NOMR 1-8	Member name of the effective fission cross sections in PDS file (8H)
	Repeat from BLOCK 3 to BLOCK 6′ NM times for all mixture (NM= number of mixtures used in the preceding PIJF calculation)
BLOCK 7	

XNOM 1-4	The latter half of member name of the set of cross sections to be written in PDS file with the name of 'FMAT $ XNOM'. (4H)
NGR	Number of coarse groups to form the homogenized cross sections
IEDXEC	Indicator of output device for the homogenized cross sections =1(PRINT)+2(PUNCH)+4(PDS file)

BLOCK 8

NGRF(L) L=1, NGR	Number of fine groups included in the L-th coarse group

5.14 TUD; One-dimensional diffusion program

The TUD is a Fortran program which solves the multi-group one-dimensional diffusion equations.

It is intended that the program should be used with data preparation and edit program. Consequently, such functions as the cross section mixing are not included and the edit routine has been reduced to minimum. For fast neutrons FAXEDT provides the effective macroscopic cross section from the GAM type data library, and for thermal neutrons, PIXEDT prepares the data. These two programs are for homogeneous systems. For system composed by the lattice cell, EPISPC and EDIT prepare the homogenized cross sections for fast and thermal neutrons, respectively. The edit program EDIT to process the result of the integral transport calculation is also useful to process the result of TUD.

The major features of the program are as follows.

1) By using variable dimensioning, the program permits the optimum use of available storage. On the 64K core machine, 35000 floating-point and 2000 integer words are allowed for storing all data blocks in core storage.
2) It is possible to treat fixed source as well as eigenvalue problems.
3) The choice of three geometries (plane, cylinder and sphere) is allowed.
4) It is possible to choose one of four possible sets of boundary conditions.
5) The cross sections are arranged to follow the format used in the collision probability programs in LAMP.
6) The numerical method used in iterations is based on the successive over-relaxation. For the fixed source problem, fluxes are renormalized so that the amount of absorption and leakage is equal to the source at each iteration. Such a treatment has resulted in a considerable improvement in the rate of converagence.

We consider here the following equations:

$$-D^g\nabla^2\phi^g(r)+\Sigma_r^g\phi^g(r)=Q^g(r), \qquad \text{(TUD. 1)}$$

$$Q^g(r)=\frac{1}{\lambda}\chi^g\sum_{g'}\nu\Sigma_f^{g'}\phi^{g'}(r)+\sum_{g'\neq g}\Sigma^{g'\to g}\phi^{g'}(r)+S^g(r) \qquad \text{(TUD. 2)}$$

where Σ_r^g=total removal from g-th group

$=D^gB^2+\Sigma^g-\Sigma_s^{g\to g}$,

Σ^g=totl cross section for the g-th group

$\nabla^2=\frac{d^2}{dr^2}+\frac{\rho}{r}\frac{d}{dr}$ ($\rho=0$ in plane geometry, $\rho=1$ in cylindrical geometry and $\rho=2$ in spherical geometry),

S^g=fixed source per unit volume for the g-th group,

λ=the eigenvalue for the eigenvalue problem or $=1$ for the fixed source problem.

A detailed derivation of the finite difference equations used in the program has been given

else where[22),23)]. We therefore describe only the result of the derivation.

If we denote by F_n the quantity $F(\gamma_n)$, then the multigroup difference equations become

$$-\alpha_n{}^g W_{n+1}{}^g+\beta_n{}^g W_n{}^g-\gamma_n{}^g W_{n-1}=\delta_n{}^g \qquad \text{(TUD. 3)}$$

where $W_n{}^g=\phi_n{}^g$ for $\rho=0$ and 1,

$W_n{}^g=r_n\phi_n{}^g$ for $\rho=2$,

$$\alpha_n{}^g=D^g(n+)\mu_{2n}, \qquad \text{(TUD. 4)}$$

$$\beta_n{}^g=D^g(n-)\mu_{1n}\sigma_{1n}+D^g(n+)\mu_{2n}\sigma_{2n}+\Sigma_r{}^g(n-)\tau_{1n}+\Sigma_r{}^g(n+)\tau_{2n}, \qquad \text{(TUD. 5)}$$

$$\gamma_n{}^g=D^g(n-)\mu_{1n}, \qquad \text{(TUD. 6)}$$

$$\delta_n{}^g=Q^g(n-)\tau_{1n}+Q^g(n+)\tau_{2n}, \qquad \text{(TUD. 7)}$$

$$\tau_{1n}=\frac{r_n-r_{n-1}}{2}\left(1-\frac{\rho(4-\rho)}{12}\frac{r_n-r_{n-1}}{r_n}\right), \qquad \text{(TUD. 8)}$$

$$\tau_{2n}=\frac{r_{n+1}-r_n}{2}\left(1+\frac{\rho(4-\rho)}{12}\frac{r_{n+1}-r_n}{r_n}\right), \qquad \text{(TUD. 9)}$$

$$\mu_{1n}=\frac{1}{r_n-r_{n-1}}\left(1-\frac{\rho(2-\rho)}{2}\frac{r_n-r_{n-1}}{r_n}\right), \qquad \text{(TUD. 10)}$$

$$\mu_{2n}=\frac{1}{r_{n+1}-r_n}\left(1+\frac{\rho(2-\rho)}{2}\frac{r_{n+1}-r_n}{r_n}\right), \qquad \text{(TUD, 11)}$$

$$\sigma_{1n}=1-\frac{\rho(\rho-1)}{2}\frac{r_n-r_{n-1}}{r_n}, \qquad \text{(TUD. 12)}$$

$$\sigma_{2n}=1+\frac{\rho(\rho-1)}{2}\frac{r_{n+1}-r_n}{r_n}, \qquad \text{(TUD. 13)}$$

The difference equations (TUD. 3) are derived for $N-1$ points of r_n, $n=1, 2, 3, \cdots\cdots, N-1$. The boundary conditions give two equations and thus we have $(N+1)$ linear equations for the $(N+1)$ unknown $W_n(n=0, 1, 2, \cdots\cdots, N)$.

At the origine of the system, we identically set the symmetric condition, which gives:

$$\mu_{10}=0.0,$$

$$\mu_{20}=(\rho+1)/r_1,$$

$$\tau_{10}=0.0,$$

$$\tau_{20}=\frac{1}{\rho+1}\left(\frac{r_1}{2}\right)^{\rho+1}, \quad \text{for} \quad \rho=0 \text{ and } 1$$

$$=\frac{1}{3}\left(\frac{r_1}{2}\right)^2, \quad \text{for} \quad \rho=2,$$

$$\sigma_{10}=0.0,$$

$$\sigma_{20}=1.0,$$

At the outer boundary of the system, the boundary conditions are written in the form:

$$D^g(N-)\left.\frac{d\phi^g}{dr}\right|_{r_N}+\omega^g\phi_N{}^g=0.$$

We take the simplest method of obtaining the corresponding difference equations to replace the derivative by a difference:

$$\alpha_N{}^g=S_N{}^g=0.0,$$

$$\mu_{1N}=\frac{1}{r_N-r_{N-1}}, \quad \text{for} \quad \rho=0 \text{ and } 1,$$

$$=\frac{1}{r_N-r_{N-1}}\left(\frac{r_N}{r_{N-1}}\right), \quad \text{for} \quad \rho=2,$$

$$\beta_N{}^g=\frac{D^g(N-)}{r_N-r_{N-1}}+\omega^g,$$

$$\gamma_N{}^g=D^g(N-)\mu_{1N}.$$

Upon providing these coefficients for the end points of the system, the boundary conditions are

replaced by the difference equations (TUD. 3).

The algorithm which has been used in most nuclear programs to solve the one-dimensional difference equations is given by the following steps:

A. Step by solve $\Gamma_n{}^g$ and $\Omega_n{}^g$ by using the following equations:

$$\Gamma_n{}^g=\beta_n{}^g-\gamma_n{}^g\delta_{n-1}{}^g/\Gamma_{n-1},$$
$$\Omega_n{}^g=(\delta_n{}^g+\gamma_n{}^g\Omega_{n-1}{}^g)/\Gamma_{n-1}{}^g,$$
$$n=1, 2, \cdots\cdots, N,$$

where

$$\Gamma_0{}^g=\beta_0{}^g \ , \ \Omega_0{}^g=\delta_0{}^g/\beta_0{}^g.$$

B. Solve then $W_{n-1}{}^g$ by

$$W_{n-1}{}^g=\Omega_{n-1}{}^g+\alpha_n{}^g W_n{}^g/\Gamma_{n-1}{}^g,$$

where

$$W_N{}^g=\Omega_N{}^g.$$

Note that for $\rho=2$, $\phi(r)$ varishes at the origine, but $r_1\phi(0)$ is given as $W_0{}^g$.

C. The procedures of iterations

The iterations procedures comprise two varieties. One is applied to the problem where fission source exists (NGKMAX>0), the other is to the problem without fission.

The former procedure is divided into three steps as follows:

1. 1) the eigenvalue problem,

normalize fission rate as

$$\int r^\rho P(r)dr=1$$

and set $Q_n{}^g=0$ for $n=1\sim$NNMAXR and $g=1\sim$NGMAX.

For the fixed source problem, set $Q_n{}^g=S_n{}^g$ $n=1\sim$NNMAXR and $g=1\sim$NGSMAX.

If the up-scattering source exists, add this source to $Q_n{}^g$ using the fluxes obtained in the previous iteration:

$$Q_n{}^g=Q_n{}^g+\sum_{g'=g+1}^{\text{NGMAX}}\Sigma_s{}^{g'\to g}\phi_n{}^{g'(m-1)}$$
$$n=1\sim\text{NNRMAX and } g=1\sim\text{NGMAX-1}$$

1. 2) After calculating the fission source to group g, solve the difference equation, obtain the fluxes $\phi_n{}^g$ and calculate the down-scattering sources to each succeeding groups is available. This step proceeds downward in energy from $g=1$ to $g=$NGMAX.

1. 3) Calculate power distribution,

$$P_n=\sum_g \nu\Sigma_f{}^g\phi_n{}^g$$

and correct it by S. O. R.:

$$P_n{}^{(m)}=P_n{}^{(m-1)}+\omega(P_n/\lambda^{(m)}-P_n{}^{(m-1)}),$$
for eigenvalue problem
$$P_n{}^{(m)}=P_n{}^{(m-1)}+\omega(P_n-P_n{}^{(m-1)}),$$
for fixed source problem,

where the superscript (m) denotes the iteration counter, and the renormalization factor is obtained as

$$\lambda^{(m)}=\int r^\rho P(r)\,dr$$

which is the total fission yield per unit source, i.e., the multiplication factor.

The iterative procedure terminates under the following conditions:

$$\left|\frac{\lambda^{(m)}-\lambda^{(m-1)}}{\lambda^{(m)}}\right|<\varepsilon,$$

$$\left[\int r^{\rho} dr \{P_{(r)}{}^{(m)} - P_{(r)}{}^{(m-1)}\}^2\right]^{1/2} < \varepsilon.$$

The other procedure for non-fission source problems (NGKMAX=0) is also divided into three steps as follows:

2. 1) Calculate the birth rate distribution $Q_n{}^g$ by summing the fixed source and the scattering source for all n and g, where the latter is computed by using the fluxes obtained in the last iteration.

$$Q_n{}^g = S_n{}^g + \sum_{g' \neq g} \Sigma_s{}^{g' \to g} \phi_n{}^{g'(m-1)}$$

2. 2) Solve the difference equations to obtain the fluxes $\phi_n{}^g$ for all n and g.

2. 3) Calculate the renormalization factor

$$F = \int r^{\rho} dr \{\sum_g \Sigma_r{}^g \phi^g(r)\} + \rho r_N{}^{\rho-1} \sum_g D^g \frac{d\phi^g(r)}{dr}\Big|_{r=r_N}$$

and correct fluxes by S. O. R.

$$\phi_n{}^{g(m)} = \phi_n{}^{g(m-1)} + \omega(\phi_n{}^g/F - \phi_n{}^{g(m-1)}).$$

The iterative procedure is terminated under the condition:

$$\left[\int r^{\rho} dr \sum_g \{\Sigma_r{}^g(\phi^g(r) - \phi^g(r)^{(m-1)})\}^2\right]^{1/2} < \varepsilon.$$

The calculation of the optimum overrelaxation factor is performed in same way as used in the program PIJF.

INPUT of TUD

The input data are divided into seven sections:

A) Control parameters,
B) Iteration parameters,
C) Geometric data,
D) Cross sections,
E) Initial guess of flux or fission rate (if any),
F) Fixed sources (if any),
G) Parameters for group reduction.

A. Control parameters

BLOCK A. 0 format (4H)

Columns 1 to 4 are punched as 'TUDb' as the program identification

BLOCK A. 1 format (72H)

(TITLE(I), I=1, 18) Columns 1 to 72 are printed as a title, and the first four characters are used as the latter half of the member name of sources and fluxes to be read in.

BLOCK A. 2

1	NRMAX	Number of regions
2	NMMAX	Number of different materials
3	NGMAX	Number of energy groups
4	NGSMAX	Number of the lowest group having a fixed source. Set zero if no fixed source.
5	NGKMAX	Number of the lowest energy-group having a fission source. Set zero if no fission source.
6	ITXEC	Unit of cross section input: '0', cross sections are read from card, '1', they are from PDS file.

7 ITS '0', fixed sources are read from card,
'1', they are from PDS file.

8 IG Geometry:
'0', for slab,
'1', for cylinder,
'2', for sphere.

9 IBOUND Outer boundary conditions:
'−1', for $\phi_N{}^g=0$

'0', for $\frac{d\phi^g}{dr}=0$

'1', for $D^g\frac{d\phi^g}{dr}+\omega^g\phi^g=0$

where
ω^g is computed by program as

$$\omega^g=\frac{1}{3*0.710466}*\left(1+0.710466\frac{3\rho*D^g}{2r_N}\right),$$

'2', for $D^g\frac{d\phi^g}{dr}+\omega^g\phi^g=0$, where ω is input

'3', for $D^g\frac{d\phi^g}{dr}+\omega^g\phi^g=0$, where ω^g is input.

10 IGUESS Initial guess of the the power distributions if NGKMAX (A.2)≠0, or the flux distributions if NGKMAX=0,
'0', uniform distributions are given by program,
'1', distributions are read by cards,
'−1', distributions are read from PDS file by the member name 'FLUX$TITLE(1)'.

11 ID Diffusion coefficient input option:
'0', diffusion coefficients are computed by program as
$D^g=0.33333(\Sigma_a{}^g+\sum_{g'}\Sigma_s{}^{g\to g'})$,
'1', they are computed by program as
$D^g=0.33333/\Sigma_t{}^g$,
where
$\Sigma_t{}^g$ is given in the field 5 of BLOCK D.1
'2', the program uses the value in field 7 of BLOCK D.1 as diffusion coefficient,
'3', diffusion coefficients are read by BLOCK D.3.

12 IBS Transverse bucklings by region:
'0', constant buckling
'1', region dependent values.

13 IPTXEC Cross section print
'0', skip,
'1', print.

14 IEDFLX Flux edit:
IEDFLX=IPRF+ICDF+ITPF
IPRF='0', skip,
'1', print,

ICDF='0', skip,
'2', punch,
ITPF='0', skip,
'4', write in PDS file with the member name 'FLUX$TITLE(1)'.

15 IPTS Fixed source print:
'0', skip,
'1', print.

B. Iteration parameters

BLOCK B

1 ITMAX Maximum number of iterations allowed [=100]
2 ITBG Minimum number of iteration before extrapolation [=5]
3 LCMX Number of overrelaxation factors tested ($\leq$20) [=5]
4 ITDM Minimum delay between extrapolations [=5]
5 IPT Monitor print of each iteration [=0]
=0, skip,
>0, print,
6 EPS Accuracy required for fission rate (NGKMAX$\neq$0) or the total absorption (NGKMAX=0) [=1.E−5]
7 EPSG Extrapolation criterion [=0.05]
8 RELC Initial overrelaxation factor [=1.2]
9 OVERX Maximum extrapolation factor [=100.0]
10 FACTOR Under extrapolation factor [=1.0]

If ITMAX is punched zero, the values in brackets are used as the iteration parameters, however, six integers and six floating-point numbers are to be read.

C. Geometric data

BLOCK C.1 (9A8) required if ITXEC$\neq$0

NAME(NM) Member name of the cross section set to be read from PDS file.
NM=1, NMMAX

BLOCK C.2

(NK(NR), NR=1, NRMAX)

NK(NR) is the number of intervals in region NR.

BLOCK C.3

(IK(NR), NR=1, NRMAX)

The material code numbers are specified for each region outward from the center of the system. The order of the material cross sections in the input data determines the material code number, i.e. the first material is the code number 1, the second is 2, etc.

BLOCK C.4

RK(NR), NR=1, NRMAX

The outer radius of region NR,

BLOCK C.5 required only if IBS=0

BSQ The transverse buckling B^2

BLOCK C.5′ required only if IBF$\neq$0

(BSQ(NR), NR=1, NRMAX)

The transverse buckling associated with each region.

BLOCK C.6 required only if IBOUND=2

EX The extrapolation constant of the outer boundary

BLOCK C. 6′ required only if IBOUND=3

(EX(NG), NG=1, NGMAX)

The extrapolation constants at the outer boundary associated with each group.

D. Cross sections

The cross sections for all energy groups of one material are followed by those of the next material, etc. The order of the material determines the material code number. All cross sections are macroscopic, and mixing of materials is not allowed. Although the format is the same as described in **4. 4** we rewrite here.

The following card layout is repeated for all energy-groups for each material starting with the highest group.

BLOCK D. 1 (2I6, 5E12. 5) required if ITXEC=0

LSS 1-6	Position of the self-scattering cross section in the group vector
LGV 7-12	Length of the group vector (see cards D. 2)
SIGACT 13-24	Activation cross section
SIGF 25-36	$\nu *$fission cross section for this group and material
SIGT 37-48	Total cross section for this group and material (required only if ID=1)
XI 49-60	The fraction of the total fission yield which forms the fission source into this group, which is required only fission sources are considered.
D 61-72	Diffusion coefficient for this group and material (required only if ID=2)

BLOCK D. 2 The first card is followed by a set of cards with format (6E12. 5) containing the group vector :

Σ_a (absorption), $\Sigma_s^{g\to g-n}$, $\Sigma_s^{g\to g-n+1}$, ……, $\Sigma_s^{g\to g}$ (self-scattering), $\Sigma_s^{g\to g+1}$, ……, $\Sigma^{g\to g+m}$

Where the scattering cross-sections for each energy-group represent the scattering out of this group to other groups, and m and n are chosen such that all non-zero scattering cross sections are included.

The program determines $g_1=g-n$ and $g_2=g+m$ from the value of LSS_g and LGV_g for this group

$$g_1=g-n=g+2-LSS_g,$$
$$g_2=g+m=LGV_g-LSS_g+g.$$

If $LSS_g=2$, then $g_1=g$ and there is no up-scattering, and if $LGV_g=LSS_g$, there is no down-scattering. If the down-scattering cross sections such as $g\to$NGMAX+1, $g\to$NGMAX+2, …… $g\to g+m-1$, $g\to g+m$ are read, the program treats them in a lump as the slowing-down cross sections. If ITXEC≠0, the cross section data mentioned above are read from PDS file.

BLOCK D. 3 required only if ID=3

((D(NG, NM), NG=1, NGMAX), NM=1, NMMAX))

Effective diffusion coefficients are read for all energy groups and for each material. Begin afresh on a new card for each material.

E. Initial guess of flux or fission rate

BLOCK E. 1 required if NGKMAX≠0 and IGUESS=1

(RRN(NN), NN=1, NNRMAX)

Fission rates for each spatial point from the origine to outward. At the region boundaries, both the inner and outer values are required, where

$$NNRMAX=\sum_{NR=1}^{NRMAX} NK(NR)+NRMAX$$

BLOCK E. 2 required if NGKMAX=0 and IGUESS=1

((FLUX(NN, NG), NN=0, NNMAX), NG=1, NGMAX)

For each group in turn, beginning with group 1, specify an estimate for the flux at space points 0 (the center) to NNMAX (the outer boundary), where

$$\text{NNMAX}=\sum_{\text{NR}=1}^{\text{NRMAX}} \text{NK(NR)}$$

F. Fixed souces

BLOCK F.1 required if NGSMAX≠0 and if ITS=0
(NGSMAX=0 for an eigenvalue problem)

((S(NN, NG), NN=1, NNRMAX), NG=1, NGSMAX)

For each group having a fixed source, specify the source per unit volume in intervals 1 to NNRMAX in that order. Note that at the region boundaries, both the inner and outer values are required. They may be calculated by using the program PIXSE or FAXSE.

If ITS≠0, fixed sources are read from PDS file by the member name of 'SOUC$TITLE(1)'.

G. Parameters for group reduction

BLOCK G.1

NGFEW Number of collapsed energy groups (≤10)
If the group collapsing is not required, the NGFEW must be zero, and BLOCK G.2 is not necessary.

BLOCK G.2 required if NGFEW≠0

(NGM(NG), NG=1, NGFEW)

The group indices of the lowest energy group in each collapsed (coarse) group, beginning with the first coarse group. The last NGM (NGFEW) must be equal to NGMAX. This group input is followed by the program identification card of any sub-program in LAMP, or by the JOB END card.

OUTPUT of TUD

The output list of TUD is self-explanatory which is initiated by the list of control parameters, then followed by geometical parameters. By using options, one optains the following lists:

a) The cross sections
b) The guess of fission rate or guess of flux
c) The monitor print of iterations
d) The information terminating the iteration:
iteration count, square root of residue and normalization factor (the multiplication factor)
e) The final flux distributions ϕ_n^g for $n=0\sim$NNMAX and for $\rho=1$ and 2, and $r_1\phi_0^g, r_1\phi_1^g, r_2\phi_2^g, \cdots r_n\phi_n^g, \cdots$ for $\rho=2$ for $g=1\sim$NGMAX.
f) The fission rate distributions
g) The integrated flux:

$$\int_{V_{\text{NR}}} r^\rho \phi^g(r)dr, \quad \text{for NR}=1\sim\text{NRMAX and } g=1\sim\text{NGMAX}$$

h) The collapsed group flux distributions:

$$\sum_{g\in G}\phi_N^g \quad \text{for } N=0\sim\text{NNMAX and } G=1\sim\text{NGFEW}$$

i) The group constants.
Activation cross sections;

$$\int_{V_{\text{NR}}} r^\rho dr \sum_{g\in G}\Sigma_{\text{act}}^g\phi^g(r)\Big/\int_{V_{\text{NR}}} r^\rho dr \sum_{g\in G}\phi^g(r),$$

ν* Fission cross section;

$$\int_{V_{\text{NR}}} r^\rho dr \sum_{g\in G}\nu\Sigma_f^g\phi^g(r)\Big/\int_{V_{\text{NR}}} r^\rho dr \sum_{g\in G}\phi^g(r),$$

Absorption cross sections:

$$\int_{V_{NR}} r^{\rho} dr \sum_{g\in G} \Sigma_a^g \phi^g(r) \Big/ \int_{V_{NR}} r^{\rho} dr \sum_{g\in G} \phi^g(r),$$

Down-scatter cross-section;

$$\int r^{\rho} dr \sum_{g\in G} \sum_{g'>G} \Sigma^{g\to g'} \phi^g(r) \Big/ \int_{V_{NR}} r^{\rho} dr \sum_{g\in G} \phi^g(r),$$

Self-scattering cross-sections;

$$\int_{V_{NR}} r^{\rho} dr \sum_{g\in G} \sum_{g'\in G} \Sigma_s^{g\to g'} \phi^g(r) \Big/ \int_{V_{NR}} r^{\rho} dr \sum_{g\in G} \phi^g(r),$$

UP-scattering cross-sections;

$$\int_{V_{NR}} r^{\rho} dr \sum_{g\in G} \sum_{g'<G} \Sigma_s^{g\to g'} \phi^g(r) \Big/ \int_{V_{NR}} r^{\rho} dr \sum_{g\in G} \phi^g(r),$$

Diffusion coefficients;

$$\int_{V_{NR}} r^{\rho} dr \sum_{g\in G} D^g \phi^g(r) \Big/ \int_{V_{NR}} r^{\rho} dr \sum_{g\in G} \phi^g(r),$$

Integral fluxes;

$$\int_{V_{NR}} r^{\rho} dr \sum_{g\in G} \phi^g(r)$$

For editing and giving the guess for further problems, TUD has an option to punch or/and write into PDS file as follows;

a. Flux distribution into PDS file
   name 'FLUX$TITLE(1)'
   ϕ_n^g (or $r_n\phi_n^g$ for the spherical geometry) (NNMAX+1)*NGMAX words

b. Flux distribution in card
   ϕ_n^g (or $r_n\phi_n^g$) (NNMAX+1)*NGMAX words,
   beginning with a new card for each energy group

5.15 Auxiliary programs

We now describe briefly the auxiliary programs which are not included in LAMP but are indispensable for updating the data libraries.

A. THERMOFILE[9]

This program is related to the following two programs. It retrieves all the data concerning the reactions in thermal energy region from ENDF/B and makes the thermal library in EBCDIC mode.

B. THERMOLIB[9]

This program works as converting the above-mentioned thermal library to that in BINARY mode. Only one execution of this program is required for a version of ENDF/B.

C. THERMOSEC[9]

This program generates the group averaged cross sections for a given temperature and for a given energy mesh. The scattering cross sections from the above-mentioned library are not advisable due to disagreement with those calculated from any of scattering models. Therefore we use this program to obtain only the capture, absorption, fission and ν*fission cross sections.

D. SMOOTH[25]

This program should be used to obtain the group averaged cross sections based on the FILE 3 in ENDF/B which is added to the cross sections calculated from the resonance integrals by using the FILE 2 of ENDF/B.

E. DCCLIB[8)]

This program is related to the following program, PI2. This compiles the thermal neutron data in ENDF/A format into its own library.

F. PI2[8)]

This program generates the group averaged cross sections by using the above-mentioned library.

6. Hierarchy of the Programs and Data Flow

We now show the flow of data from basic libraries to the final few group constants.

6.1 Preparation of data library

We use the files in the format of ENDF/A or ENDF/B as basic libraries. We can update our libraries via the process shown in **Fig. 6.1** and **6.2** respectively for the ENDF/A type and ENDF/B type files.

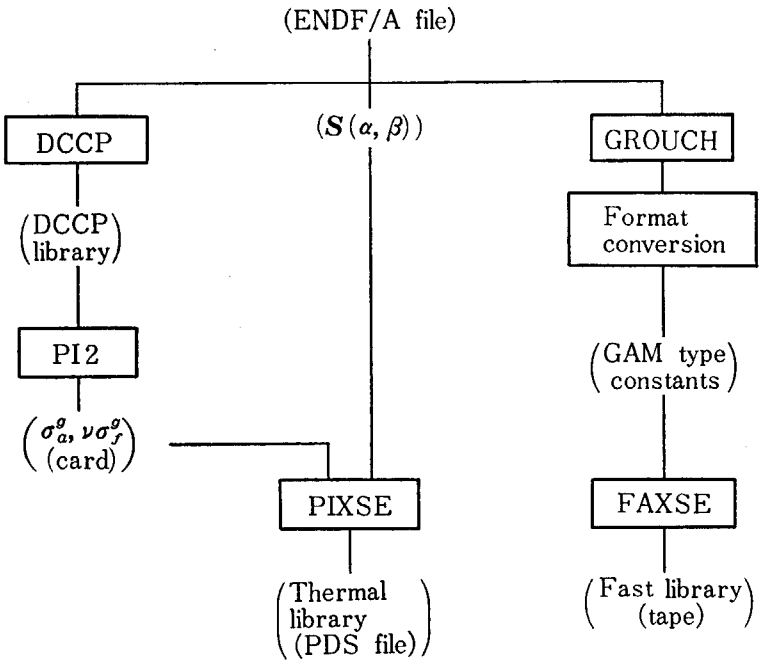


Fig. 6.1 Data library from the ENDF/A format file.

6.2 Cell calculation in thermal region

The first step for the cell calculation in the thermal energy region is to prepare the macroscopic cross sections and the source distributions. The macroscopic cross sections of constituent mixture in a cell are fed into any of the collision probability programs depending on the geometry of the cell (see **Fig. 6.3**) which calculates the collision probabilites and the directional probabilities.

The third step is to solve the linear equations by PIJF, to which the complete sets of cross sections and the source distributions are fed from PIXEDT via PDS file and the collision probabilities via unit 21. One group constants for the thermal region are obtained in this step.

For the calculation of the directional diffusion coefficients, the fouth step is required to run the EDIT to which the normal and directional collision probabilities are fed. The homogenized multigroup constants are optionally computed in this step which is followed by the multigroup core calculation in the thermal region by TUD.

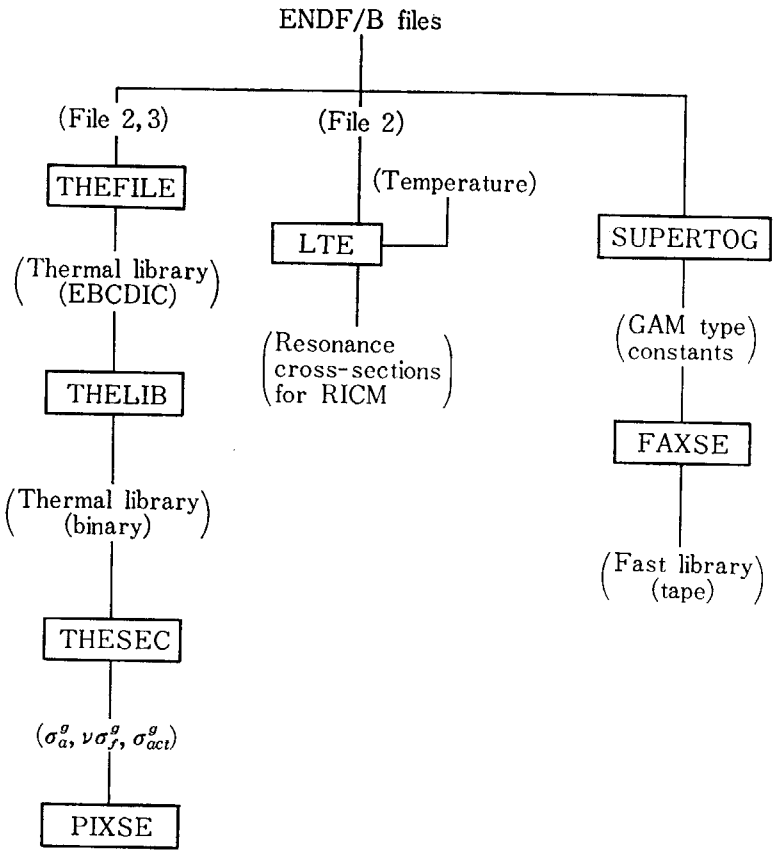


Fig. 6.2 Data library from the ENDF/B format file.

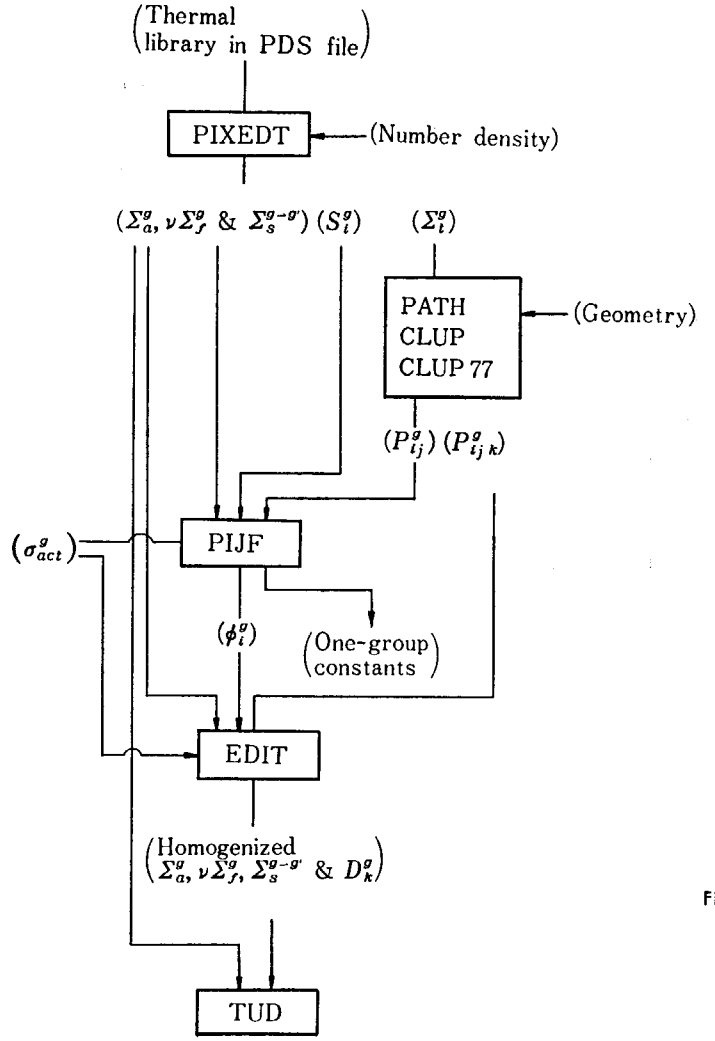


Fig. 6.3 Data flow for cell calculation in thermal region.

6. 3 Resonance integrals in the lattice cell

The PRERI prepares the table of the macroscopic total cross-sections of multigroup to the calculation of collision probabilities which are used in RICM as the table for the interpolation of P_{ij}.

Being given the level parameters and the temperature LTE computes ψ's and χ's as cross sections of resonant isotopes for each level and stores them in unit F36. In the edit stage of RICM, the additional group cross sections may be fed as shown in **Fig. 6.4**.

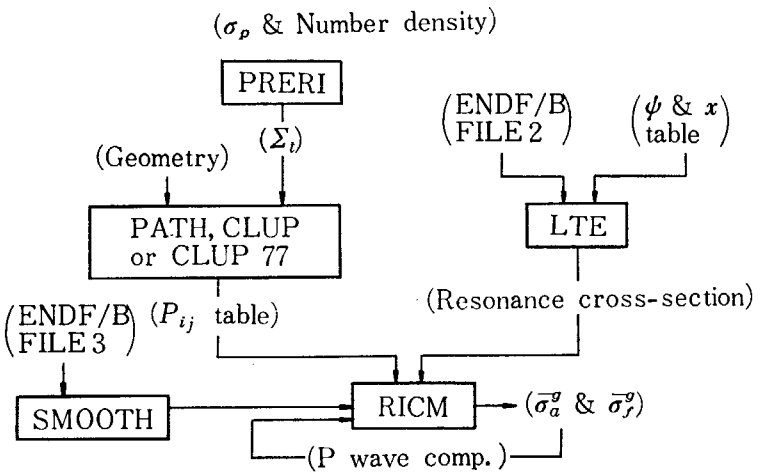


Fig. 6.4 Data flow for resonance integral calculations.

6. 4 Cell calculation in fast (and epithermal) region

In this energy region we have option to choose the combination of programs, the division

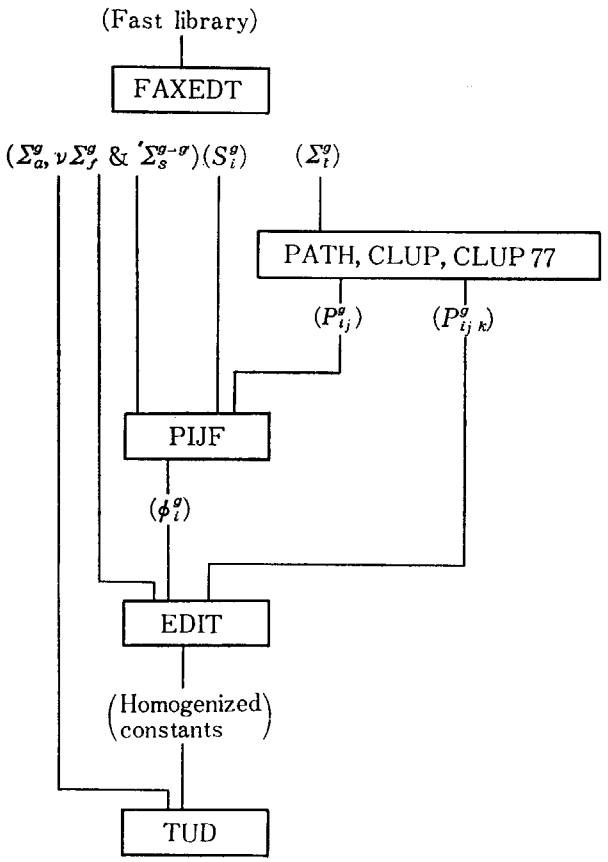


Fig. 6.5 Scheme I of the cell calculation in the fast energy region.

of energy range, the data library and the number of coarse groups. They depend on the reactor type and the accuracy wanted by user.

The first scheme shown in **Fig. 6.5** which coincides with the scheme in **Fig. 6.3** used in the thermal region, except for the use of the fast library is aimed at obtaining the anisotropic diffusion coefficients over the whole energy range. However, the resonance shielding should be considered in the stage to form the macroscopic cross sections. It is noted that we have not yet been on the theoretical ground on this process.

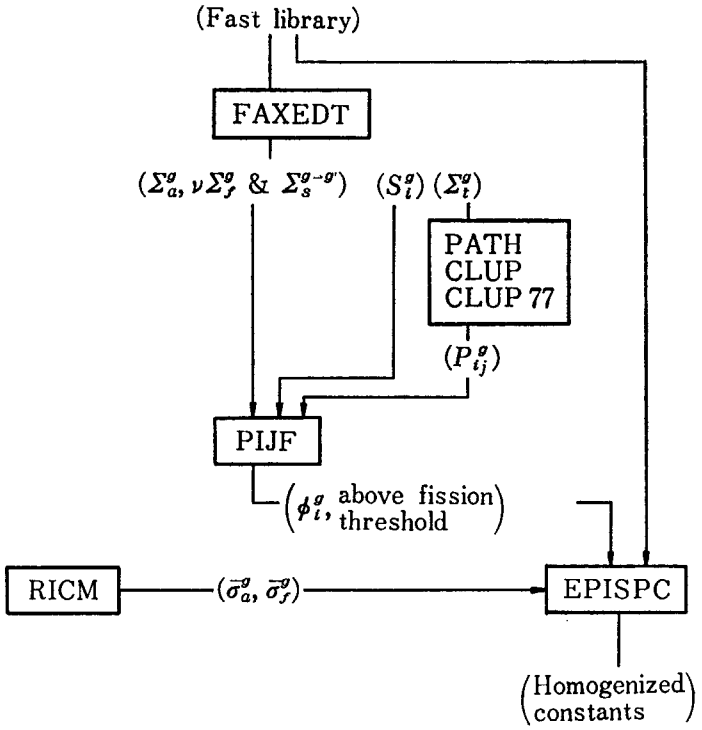


Fig. 6.6 Scheme II of the cell calculation in the first energy region for the thermal reactor.

Fig. 6.6 shows the scheme II which can be considered rigorously as the calculation of fission factor and the resonance shielding in the lattice cell of thermal reactors. These effects are separately calculated and coupled together in the step of EPISPC.

The process has a defect not to be able to consider the anisotropic diffusion coefficients, although the effect is not so much effective in this energy region for the usual type of thermal reactors except for the CO_2 cooled graphite moderated reactor. The reactor has a large coolant gap and large lattice pitch compared with the neutron free path so that the streaming effect

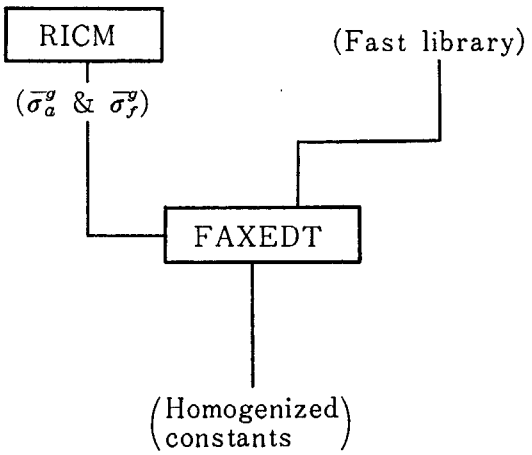


Fig. 6.7 Scheme III of the cell calculation in the first energy region for the thermal reactor.

should not be neglected.

Fig. 6.7 shows the scheme III which is simpler than the scheme II because of neglecting the heterogeneous effect in the fast fission. This process is useful for the homogeneous region.

6.5 Core calculation by one-dimensional multigroup diffusion theory

The scheme to collapse the energy group structure by using the simple buckling mode is included in the processes mentioned above. Here we show the process to obtain the few group region-wise constants for a finite system by the use of a one-dimensional multigroup diffusion program TUD which can take into account also up-scattering of neutrons.

We illustrate the process in **Fig. 6.8**. Suppose that we already have the multigroup constants for the epithermal and also for the thermal energy regions for each spatial region. The first step is to solve the eigenvalue problem by using the assumed values for thermal one group constants. Using the flux distributions for the epithermal region, we calculate the source distribution from the epithermal to the thermal region. The second step is to solve the inhomogeneous problem for the thermal energey region which results in the one-group thermal constants for each spatial region. Thus the one-group constants replace the corresponding values used in the first step for the following eigenvalue calculations. We know from the experience that the iterative procedure converges rapidly to the final values.

On the theoretical point of view, it is preferable to adopt one-step process by using a multigroup model for both the fast and thermal ranges. For this purpose, however, we must prepare the scattering cross sections from each epithermal group to all thermal groups.

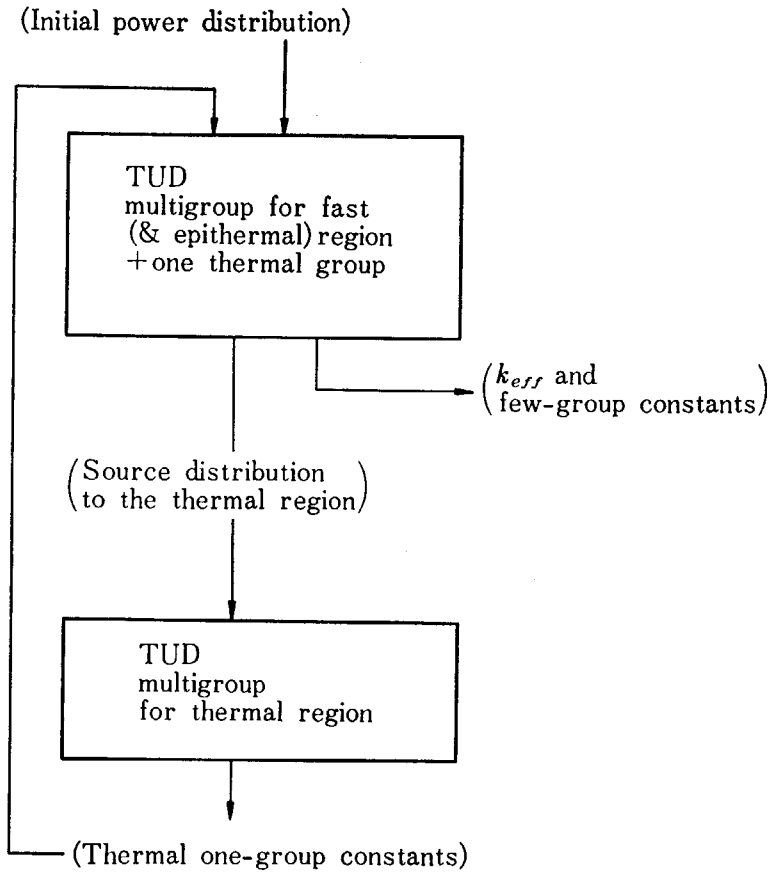


Fig. 6.8 The iterative process to obtain the region-wise few group constants.

7. Usage

We give here the informations about how to use LAMP, how to decompose for utilizing the part of LAMP and how to convert it written by the FACOM-Fortran to that of different machines.

7.1 Overlay structure

As shown in **Fig. 7.1** the hierarchy of programs in LAMP has a simple structure. At every program step the program identification card read by LAMPM causes the branching to the required subprogram.

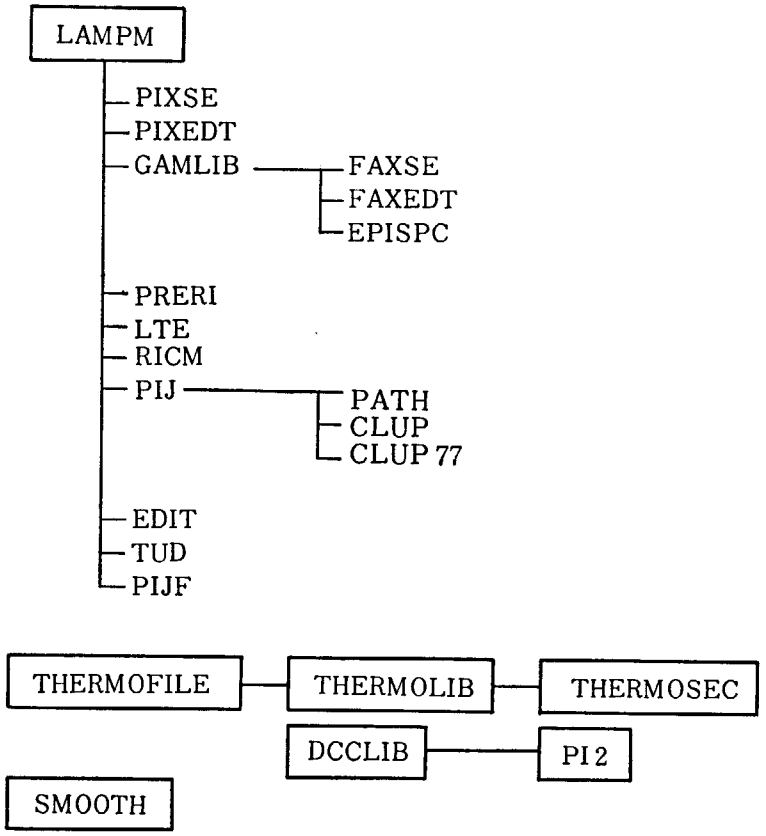


Fig. 7.1 Hierarchy on subprograms in LAMP.

7.2 Auxiliary units

TABLE 7.1 shows the auxiliary units used in LAMP to transfer the informations between subprograms. This table is useful in case where the user makes the FD statement for the auxiliary units. It shows the physical quantity and the file unit name to be read or written.

7.3 Usage of PDS file

We occasionally have some requirements to manage the data in PDS file produced by LAMP. The FACOM CPS (Conversational Programming System) is available to **condense** the file, to **extend**

TABLE 7.1 Auxiliary units

Prog. name	PDS file (USERPDS)	Classical file (F01-F99)
PIXSE	W-σ	R/W-SC (F04), R-SAB (F0N)
PIXEDT	R-σ, W-Σ_t, W-Σ, W-S	
FAXSE		R/W-SC (F04), R/W-σ (F08)
FAXEDT	R-σ, R-$\bar{\sigma}$, W-Σ_t, W-Σ, W-S	R-σ (F08)
EPISPC	R-$\bar{\sigma}$, W-Σ	R-σ (F08)
PRERI	W-Σ_t	
LTE		R/W-SC (F04), R-$\phi\chi$ (F18), W-RC (F36)
RICM	R/W-$\bar{\sigma}$	R-P (F21), R-RC (F36)
PATH	R-Σ_t	R/W-PT (F12), R-P (F20), W-P (F21)
CLUP	R-Σ_t	R/W-PT (F12), R-P (F20), W-P (F21)
CLUP 77	R-Σ_t	R/W-PT (F12), R-P (F20), W-P (F21)
PIJF	R-Σ, R-S, R/W-ϕ	R-P (F21)
EDIT	R-ϕ, R-Σ, W-Σ	R-P (F21)
TUD	R-Σ, R-S, R/W-ϕ	

R; Read
W; Write
σ; Micro library
Σ_t; Macro total
Σ; Macro complete
S; Source
$\bar{\sigma}$; Effective cross section
ϕ; Flux
SAB; S(α, β)
SC; Scratch
RC; Resonance cross section
P; Collision probabilities
PT; Path table
$\phi\chi$; ϕ and χ tabulation

the track, to **rename** the file and the member, and to **delete** any of them.

To read and write the contents of PDS file we have to use a GPL language subroutine, RWPDSF[1] which is called by a Fortran program as used in LAMP.

The specifications for the usage of the subroutine are as follows:

1) To create a PDS file we can use a job macro control card as '$DISKTN file unit name, file name, TRK=nn, RLSE='. The file unit name is normally fixed to 'USERPDS' by the subroutine. If the user needs to change the file unit name to process more than one files, set FILESW =1 in the labeled common /PDSPDS/ and assign another file unit name to FILEN (1)-FILEN (3) in the same common. The file name should be organized as 'Jnnnn. abcd' as to be managed by the FACOM CPS. After this execution the file is automatically catalogued.

2) To read or update the PDS file a job macro control card is formed as '$DISKTO file unit name, file name'.

3) Do not change the size and the order of the labeled common /PDSPDS/.

4) When a FILE OPEN ERROR encountered, ECODE in /PDSPSD/ is set to 1.

5) The file unit name is specified by 12 characters and the member name by 8 characters.

The subroutine consists of multi-entries used as follows, and it returns with the error messages ECODE and TEMP in the labeled common /PDSPDS/.

1. CALL READ (M, A, N)
   To read N words of the member M into A(1), ..., A(N), where N should be equal or less than L; the total number of words of the member M.
   If ECODE=3, the member M does not exist,
   ECODE=5, N>L
   ECODE=10, an error happens during reading.
2. CALL WRITE (M, A, N)
   To write A(1), ..., A(N) as the new member M in PDS file
   If ECODE=2, the member M already exists
   ECODE=9, an error happens during writing
3. Not used

4. Not used
5. CALL RENAME (M1, M2)
   To change the member name M1 to M2
   If ECODE=3, the member M1 does not exist
   =11, an error happens during renaming.
   In the last case
   If TEMP=1 and the member M2 already exist this instruction does not work (NO OPERATION)
   If TEMP=3, there in no room to add the DIRECTORY for the member M2, therefore increase the DIRECTORY area in FD statement.
6. CALL DELETE (M)
   To delete the member M
   If ECODE=3, the member M does not exist
   =11, Some error happens
   In the latter case, the member M does not exist in the file if TEMP=2.
7. CALL INFOR (M, L)
   To transfer the DIRECTORY of the member M into L(1), . . . , L(12)
   If ECODE=3, the member M does not exist.
8. CALL GETLEN (M, L) To transfer the length of the member M into L
   If ECODE=3, the member M does not exist.

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Appendix: Sample Input

Sample input #1 for the thermal constants of a graphite moderated 15 rods clustered assembly

```
*NO A111,P,0W,1T,5                                                    FUL00001
*GJOB [redacted],SMF=CLS                                              FUL00020
*DRUN LAMPCP,J1084,LAMPCF,SYSOUT=CLS                                  FUL00040
*DISKTO USERPDS,J1084,PDSFBLK                                         FUL00060
*DISK F12                                                             FUL00080
*DISKTN F21,J1084,PIJFBL2,TRK=200                                     FUL00100
*DATA                                                                 FUL00140
PIXEDT                                                                FUL00160
HELM  1  5                                                            FUL00161
0000  2,687E-5                                                        FUL00163
FUL4   4   5                                                          FUL00164
U238   1,956E-3                                                       FUL00170
U235   0,8150E-4                                                      FUL00171
0000   4,075E-3                                                       FUL00172
100C     0,083                                                        FUL00173
BRPN   2   5                                                          FUL00174
100C   0,08758                                                        FUL00180
0B10   8,626E-4                                                       FUL00181
GRAP   1   5                                                          FUL00182
100C   0,090225                                                       FUL00190
HGRA   1  -5                                                          FUL00191
100C   0,045                                                          FUL00200
FBLK    46 0  -5                                                      FUL00201
   12(1  2)  2(3  4)  2(4)  3(5)   13(4)   /                          FUL00210
CLUP                                                                  FUL00212
FBLK BLOCK 4%-ENRICHED U AND THREE B,P,                               FUL00520
 46   5   30   0   1   1   0   4   1   8  9  8   3                   FUL00540
 2   15   7   15   1   46   0   0                                     FUL00560
3 3 3 360/                                                            FUL00580
  3(-1) 3(1) 9(1)/                                                    FUL00600
   90 210 330/                                                        FUL00625
 15(1)                                                                FUL00660
TTOTHELMTTOTFUL4TTOTBRPNTTOTGRAPTTOTHGRA                              FUL00661
12(1  2)  2(3  4)  2(4)  3(5)  13(4)   /                             FUL00680
 0,0  2,5      14,7000  15,6986                        /              FUL00690
 12(0,0  0,9  1,8)  3(0,0  0,4  1,0)                                  FUL00701
6(6,5818)  6(11,4)  3(13,6)                    /                     FUL00720
  90 210 330 30 150 270                                               FUL00741
 0 5*60                                                               FUL00780
  30 150 270 /                                                        FUL00781
PIJF                                                                  FUL00782
FBLK    46-REGION FBLKBLOCK PIJF CAL,                                 RRR00700
46   5   30   1   1   1      /                                       RRR00710
30   0   0   5   5   1   1   0   1   0   0   0   /                   RRR00720
 100   1   5   5   5   0   1,0E-5   1,0E-4   0,05  1,2   1,0   1,0   /  RRR00730
TMATHELMTMATFUL4TMATBRPNTMATGRAPTMATHGRA          /                   RRR00740
0                                                                     RRR00750
EDIT                                                                  RRR00751
  2  2  0  2  5  /                                                    RRR00760
*JEND                                                                 RRR00770
                                                                      RRR00940
```

Sample input #2 for the resonance integrals of a graphite moderated 15 rods clustered assembly

```
*NO A111,P,0W,1T,2                                                    FUL00020
*GJOB [redacted],LAMPS,SMF=CLS                                        FUL00040
*DRUN LAMPCP,J1084,LAMPCF,SYSOUT=CLS                                  FUL00060
*DISKTO USERPDS,J1084,PDSPRCL                                         FUL00080
*DISK F12                                                             FUL00100
*DISKTN F21,J1084,PIJCLRI                                             FUL00140
*DISKTO F36,J1560,LTESHE                                              FUL00141
*DATA                                                                 FUL00160
PRERI                                                                 FUL00200
FUL4   15  5  3  1   /                                                FUL00210
   4,8  3,8  5,5          / CARBON OXYGEN BORON                      FUL00220
HGRA      0,045  0,0  0,0  /                                          FUL00230
GRAP      0,0902  0,0  0,0   /                                        FUL00240
BRPN     0,08758  0,0  0,0  /                                        FUL00250
F4RD     0,083   4,075E-3   0,0   /                                  FUL00260
HGAP    0,0  2,687E-5  0,0  /                                         FUL00270
CLUP                                                                  FUL00520
FUEL BLOCK 6%-ENRICHED U AND THREE B,P,                               FUL00540
  11   5   15   0   1   1   0   4   1   8   8   8   3   2           FUL00560
  15   7   15   1   46   0   0     /REPLACE FBRICM                    FUL00561
3 3 3 360/                                                            FUL00600
 3(-1) 12(1)                                                          FUL00601
 90  210  330 /                                                       FUL00624
6(1  2)  6(3  4)  2(5  6)  2(10)                                      FUL00625
 3(7)   6(8)   2(9)  10   2(9)   10   11   /                         FUL00626
 15(1)                                                                FUL00630
TRESHGAPTRESFUL4TRESBRPNTRESGRAPTRESHGRA                              FUL00680
 1  2  1  2  3  4 5  4(4)                          /                  FUL00690
 0,0  2,5      15,0  15,6986                       //                 FUL00701
 12(0,0  0,9  1,8)  3(0,0  0,4  1,0)               /                 FUL00720
6(6,5818)  6(11,4)  3(13,6)                      /                    FUL00741
  90 210 330 30 150 270                                               FUL00780
 0 5*60                                                               FUL00781
  30 150 270 /                                                        FUL00782
RICM                                                                  FUL01070
U238 S U RESONANCE INTEGRAL FOR U238 IN MK =3    RICMF                FUL01080
 11  0  5  3  0  3  -1  0  1,0  1,0  0,  0,  1,0   /                 FUL01090
 1  0  1  0  7(1)                                  /                 FUL01100
11(0)   /                                                             FUL01110
U235 S      0  235,  11,   2(0,0  0,815E-4)  7(0,0)   /U-235         FUL01120
U238 S U    0  238,  10,6  2(0,0  0,1956E-2)  7(0,0)      /U 235     FUL01130
OXYGEN      0   16,  3,8 2(2,687E-5  4,075E-3)  7(0,0)    /OXYGEN    FUL01140
CARBON      0   2,  4,8 2(0,0  0,083)  1(0,0876  0,0902)             FUL01150
0,045  4(0,0902)        /CARBON                                       FUL01160
BORON    0  10,  5,5  4(0,0)  0,004581  6(0,0)    /NATURAL BORO     FUL01170
9/ U238 A SMOOTH  33 TTO 66                                           FUL01180
 0,36656E 00 0,29132E 00 0,24301E 00 0,22381E 00 0,17665E 00 0,15881E 00FUL01190
 0,10581E 00 0,64679E-01 0,53716E-01 0,45230E-01 0,78702E-01 0,33963E-01FUL01200
 0,27436E-01 0,23194E-01 0,40420E-01 0,29197E-01 0,66098E-02 0,16648E-01FUL01210
 0,57450E-02 0,52423E-02 0,47836E-02 0,43650E-02 0,47778E-01 0,12008E-01FUL01220
 0,13207E-01 0,14972E-01 0,10796E 01 0,66074E 00 0,53030E 00 0,48827E 00FUL01230
 0,48293E 00 0,49834E 00 0,52817E 00 0,56992E 00                      FUL01240
 0     /16                                                            FUL01250
0 /1257                                                               FUL01255
 -1  /OUT IN PDS                                                      FUL01257
RICM                                                                  FUL01260
U235 S   RESONANCE INTEGRAL FOR U235  MK=3 J PAR, ENRCH,              FUL01270
 11  0  5  3  0  3  -1  0  1,0  1,0  0,  0,  1,0   /                 FUL01280
 1  0  1  0  7(1)                                  /                 FUL01290
 11(0)                                            /                 FUL01300
U238 S U    0  238,  10,6  2(0,0  0,1956E-2)  7(0,0)      /U 235     FUL01310
U235 S      0  235,  11,   2(0,0  0,815E-4)  7(0,0)   /U-235         FUL01320
OXYGEN      0   16,  3,8 2(2,687E-5  4,075E-3)  7(0,0)    /OXYGEN    FUL01330
CARBON      0   2,  4,8 2(0,0  0,083)  1(0,0876  0,0902)             FUL01340
0,045  4(0,0902)        /CARBON                                       FUL01350
BORON    0  10,  5,5  4(0,0)  0,004581  6(0,0)    /NATURAL BORO     FUL01360
25/ GAMMA U235                                                        FUL01370
-0,47195E 00-0,12065E 00-0,14724E 01-0,41268E 00-0,26081E 01 0,19276E 00FUL01380
-0,67772E 01-0,36046E 01-0,41575E 01-0,45923E 01 0,29201E 00 0,34416E 00FUL01390
-0,50346E 00-0,42452E 00-0,28368E 00 0,68774E 00 0,32258E 01 0,57237E 01FUL01400
25   /FISSION U235                                                    FUL01410
 0,95240E 00-0,63131E 00-0,84023E-01-0,29713E 00-0,22813E 01-0,11668E 01FUL01420
-0,12886E 01-0,10667E 02-0,61914E 01-0,18392E 01-0,13007E 01 0,33323E 00FUL01430
-0,31881E 00-0,97047E 00-0,41454E 01-0,14019E 02 0,35363E 02 0,56812E 02FUL01440
  0    /16                                                            FUL01450
25   /FISSION U235                                                    FUL01460
 0,95240E 00-0,63131E 00-0,84023E-01-0,29713E 00-0,22813E 01-0,11668E 01FUL01470
-0,12886E 01-0,10667E 02-0,61914E 01-0,18392E 01-0,13007E 01 0,33323E 00FUL01480
-0,31881E 00-0,97047E 00-0,41454E 01-0,14019E 02 0,35363E 02 0,56812E 02FUL01490
  0     /18                                                           FUL01500
  0 /                                                                 FUL01510
-1  /OUT IN PDS                                                       FUL01520
*JEND                                                                 FUL01530
```