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EPSILON-An IBM-7090 Code for Computing Fast Fission Effects in Lattice by Collision Probability Method

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Summary

The fast fission effects in the multi-region lattices of cylindrical rods and slabs are calculated, using the collision probability method for the spatial problem and the multi-group technique. The content of the EPSILON code for calculation of the fast fission effects by the IBM-7090 is described in this report.

The EPSILON code has a maximum number of 5 regions in the lattices, including a maximum number of 2 fuel regions, a maximum number of 100 energy groups, and a maximum number of 10 elements in one region. The computing time is about 3 minutes when the number of energy groups is 18, the region number is 5 and the element number in the cylindrical rod lattice is 10.

In this report, the calculation method for the fast fission effects is first given, then, the collision probabilities used in the EPSILON code are discussed, and the content of the code is last described. In the Appendix, the EPSILON-LT code for making a library tape of the EPSILON is described.

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衝突確率法による格子系の高速分裂効果の IBM-7090 用コード-EPSILON

要　旨

板状，円柱状多領域格子系での高速分裂効果は空間的には衝突確率法を，また多組組わけ法を用いて計算される．この報告ではこの高速分裂効果を計算するための IBM-7090 用コード，EPSILON，の内容を示す．

EPSILON では燃料領域として最大 2 領域がとれるのを含めて，格子系では領域が 5 領域まで，エネルギー組数の最大は 100，1 領域中の元素の種類の最大は 10 である．エネルギー組数 18，領域数 5，格子系の元素数 10 の円柱格子系の計算での所要時間は約 3 分である．

この報告では，初めに高速分裂効果の計算の方法，次に EPSILON コードに用いられている衝突確率について論じ，最後にこのコードの内容を述べた．付録には，このコードのライブラリーテープを作成するコード，EPSILON-LT について示した．

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Contents

目　　次

1. Introduction

In the accurate calculation of fast fission effects in lattice, it is well known to use the Monte Carlo method. However the Monte Carlo method requires much computing time to get higher accuracy. While one of the authors has derived an exact expression for collision probability in actual multi-region cylindrical lattices[1], and it is thus possible to calculate the fast fission effects in lattice economically and accurately by using the collision probability method. Since we completed an IBM-7090 code, EPSILON, for computing this problem, we will represent its content here.

In this program the following assumptions are made,

(i) Scattering of neutrons is isotropic in the laboratory system.
(ii) Angular distribution of neutron source is isotropic.
(iii) Lattice has less than 5 regions. The collision probabilities in each region are derived from the flat flux approximation.
(iv) All fissionable elements have same fission epectrum by either fast or thermal neutrons.
(v) Fast fission neutron cycle has a constant period even if fast fission occurs at any energy level, but it is remarkably faster than thermal fission neutron cycle.
(vi) A spatial distribution of virgin neutrons born by thermal neutrons is given either in uniform shape or as a function of square of radius only for case where a region which contains fissionable elements is located at the center of the lattice.

2. Calculating method of fast fission effect

2.1 Regions and energy groups

A system considered here is a slab lattice or a cylindrical rod lattice, and all neutrons in the system have fast neutron energy. The lattice is divided up into a number of identical unit cells, and all regions in the unit cell have a homogeneous medium. Whole range of energy from the upper limit of fission spectrum to any given lowest energy should be divided up into as wide intervals as cross sections and neutron flux are not so steeply changed.

Nomanclatures for energy grouping and region are shown in schema of **Fig. 1**. The cross sections, neutron flux and so on are averaging over the i-th group and k-th region shown in **Fig. 1** as follows,

Averaged macroscopic cross section (except for scattering cross section);

$$\Sigma_k^i=\int_{u^{i-1}}^{u^i}du\int_{x_k}^{x_{k+1}}dx\,\Sigma(x,\ u)\ \Big/\ \int_{u^{i-1}}^{u^i}du\int_{x_k}^{x_{k+1}}dx, \tag{2·1}$$

where $\Sigma(x,\ u)$; cross section at lethargy u and x.

Averaged neutron flux;

$$\Phi_k^i=\int_{u^{i-1}}^{u^i}du\int_{x_k}^{x_{k+1}}dx\,\Phi(x,\ u)\ \Big/\ \int_{u^{i-1}}^{u^i}du\int_{x_k}^{x_{k+1}}dx, \tag{2·2}$$

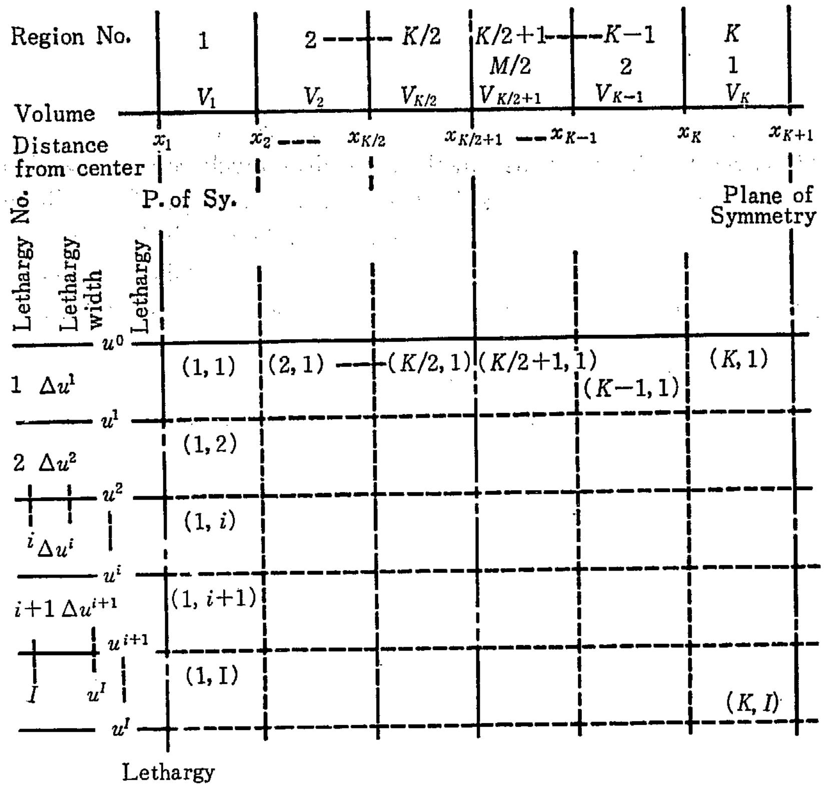
where $\Phi(x,\ u)$; neutron flux for the same lethargy and region as $\Sigma(x,\ u)$.

Averaged scattering cross section;

$$\Sigma_{sk}^{j\to i}=\int_{u^{i-1}}^{u^i}du\left[\int_{u^{j-1}}^{u^j}du'\int_{x_k}^{x_{k+1}}dx\,\Sigma_s(x,\ u'\to u)\Big/\int_{u^{j-1}}^{u^j}du'\int_{x_k}^{x_{k+1}}dx\right]\Big/\int_{u^{i-1}}^{u^i}dx, \tag{2·1}'$$

Averaged fission spectrum;

$$X^i=\int_{u^{i-1}}^{u^i}du\,X(u)\ \Big/\ \int_{u^{i-1}}^{u^i}du \tag{2·3}$$



i; Maximum No. of lethargy
k; Maximum No. of region

Fig. 1 Schema for Numbering of Energy Group and Region

where $\int_{u_{\max}}^{0} X(u)\,du=1$, and $X(u)=Ce^{-E/A}\sinh\sqrt{BE}$

with E in MeV, and A, B and C are constants.

2.2 Calculating method for reaction rates

From the neutron balance in the k-th region and the i-th energy group, the total collision rate of neutrons R_{ki} is given by

$$R_{ki}=\sum_{l=1}^{k}\left[\sum_{j=1}^{i} C_{l\to k,\, j\to i}+SF_{l\to k,\, i}+ST_{l\to k,\, i}\right], \tag{2·4}$$

where $SF_{l\to k,\, i}$ is a reaction rate that fast fission neutrons born in the (l, i)-th region and energy group will have their first collision in the (k, i)-th ones, $ST_{l\to k,\, i}$ is a rate that thermal fission neutrons born in the (l, i)-th ones will have their first collision in the (k, i)-th ones, and $C_{l\to k,\, j\to i}$ is a rate that neutrons scattered from the (l, j)-th ones into the (l, i)-th ones will have their first collision in the (k, i)-th ones. These quantities can be calculated by introducing the scattering, total, and fission cross section $\Sigma_{sk}^{j\to i}$, Σ_{Tk}^{i}, $(\nu\Sigma_f)_k^i$ and the first flight collision probabilities, P_{lk}^i and $\bar{P}_{lk}^i$, with the flat and thermal neutron flux distributions, respectively, that neutrons born in the l-th region will have first collision in the k-th region as follows,

$$\begin{aligned} R_{kj}&=\Sigma_{Tk}^{i}\,\Phi_k^i\,V_k\Delta u^i\,, \\ ST_{l\to k,\, i}&=\bar{P}_{lk}^i\,V_l\,S_l\,X^i\,\Delta u^i\,, \\ SF_{l\to k,\, i}&=P_{lk}^i\,V_l\Big(\sum_{j=1}^{I}(\nu\Sigma_f)_l^j\,\Phi_l^j\,\Delta u^j\Big)X^i\,\Delta u^i\,, \end{aligned} \tag{2·5}$$

and

$$C_{lk,\, j\to i}=P_{lk}^j V_l\,\Sigma_{sl}^{j\to i}\,\Phi_l^j\,\Delta u^j\,.$$

Substituting equ. (2·5) into equ. (2·4), we can calculate Φ_k^i and the detail derivation will be presented later. If the fast neutron flux Φ_k^i is given, the fast fission effect ε, the fast fission rate to thermal fission δ and contributions of the component elements to ε or δ can be easily derived. The fast fission effect is defined by the following two ways,

$$\varepsilon=\frac{\text{neutrons born by thermal and fast fission}}{\text{neutrons born only by thermal fission}} \tag{2·6}$$

$$=\frac{\text{neutron density slowed down below threshold energy}}{\text{neutrons born by thermal fission}} \tag{2·7}$$

It is considered from the neutron balance that equ. (2·6) is equivalent to equ. (2·7). Here a numerical result of ε is derived from both the definitions. By use of Φ_l^j, an expression of ε can be given by

$$\varepsilon=\frac{S+\sum_{j=1}^{I}\sum_{l=1}^{K}[(\nu\Sigma_f)_l^j-\Sigma_{al}^{j}]\Phi_l^j V_l \Delta u^j}{S} \tag{2·8}$$

with

$$S=\sum_{j=1}^{I}\sum_{l=1}^{K} V_l S_l X^j \Delta u^j , \tag{2·9}$$

and Σ_{al}^{j} is the absorption cross section of the l–th region in the j–th energy group.

Since a definition of δ is a ratio of fast fission rate to thermal fission rate, we give

$$\delta=\frac{\sum_{l=1}^{K}\sum_{j=1}^{I} V_l \Sigma_{fl}^{i} \Phi_l^j \Delta u^j}{S/\nu_T} , \tag{2·10}$$

where ν_T is the number of neutrons per thermal fission. A contribution of atom m in the l–th region to the fast fission is given as

$$\varepsilon_{lm}=1+\sum_{j=1}^{I} [\{(\nu\sigma_f{}^{m})_l{}^j-\sigma_{fl}{}^{mj}-\sigma_{al}{}^{mj}\} N_l{}^m \Phi_l{}^j \Delta u^j]/S , \tag{2·11}$$

where $N_l{}^m$ is the number of nucleus per cm^3 of the atom m in the l–th region and $\sigma_l{}^{mj}$ is a microscopic cross section of the atom m in the (l, j)–th region and group. A contribution of the same to the fission rate is similarly given as

$$\delta_{lm}=\sum_{j=1}^{I} [V_l \sigma_{fl}{}^{mj} N_l{}^m \Phi_l{}^j \Delta u^j] \Big/ (S/\nu_T) . \tag{2·12}$$

Finally the averaged macroscopic cross section $\overline{\Sigma}_k$ over the k–th region is obtained as follows,

$$\overline{\Sigma}_k=\sum_{j=1}^{I}(\Sigma_k^j \Phi_k^j) \Big/ \sum_{j=1}^{I} \Phi_k^j \Delta u^j . \tag{2·13}$$

2.3 Calculation of averaged slowing down cross section

First we represent calculating procedures to matrix of slowing down cross sections by elastic scattering.

(a) According to the assumption (i) and equ. (2·3), a balance equation

$$\int_{u^{i-1}}^{u^i}\int_{u^{j-1}}^{u^j}\Sigma_{\text{el m}}(u'\to u)\Phi_{\text{m}}(u')du'du=\int_{u^{i-1}}^{u^i}\int_{u^{j-1}}^{u^j}\frac{e^{-(u-u')}}{1-\alpha_{\text{m}}}\Sigma_{\text{el m}}(u')\Phi_{\text{m}}(u')du'du$$

is changed to

$$\left(\Sigma_{\text{el m}}^{j\to i}\Delta u^i\right)\Phi_{\text{m}}{}^j \Delta u^j=\Sigma_{\text{el m}}^{j}\Phi_{\text{m}}{}^j\frac{1}{1-\alpha_{\text{m}}}(e^{u^j}-e^{-u^{j-1}})(e^{-u^{i-1}}-e^{-u^i}) . \tag{2·14}$$

Then any element of (j, i) matrix of the slowing down cross sections by elastic scattering of neutrons is shown as

$$\sum_{\mathrm{el\,m}}^{j\to i} \Delta u^i = \beta_{\mathrm{m}}[(e^{-(u^{i-1}-u^j)} - e^{-(u^{i-1}-u^{j-1})}) - (e^{-(u^i-u^j)} - e^{-(u^i-u^{j-1})})] \ , \ i \neq j$$

$$= \beta_{\mathrm{m}}[\Delta u^j - (e^{-(u^i-u^j)} - e^{-(u^i-u^{j-1})})] \ , \ i=j \qquad (2\cdot15)$$

where $\beta_{\mathrm{m}} = \sum_{\mathrm{el\,m}}^{j} \Big/ (1-\alpha_{\mathrm{m}})\Delta u^j$ (2·15)′

and $\alpha_{\mathrm{m}} = (A_{\mathrm{m}}-1)^2/(A_{\mathrm{m}}+1)^2$ (2·15)″

A_{m} is the Atomic mass number of the atom m amd $\sum_{\mathrm{el\,m}}^{j}$ is its elastic scattering cross section averaged in the j-th energy group.

(b) For case of $u^j+\Delta \leqq u^i$ and $u^{j-1}+\Delta \geqq u^{i-1}$,

where $\Delta = -\ln \alpha_{\mathrm{m}}$, $\sum_{\mathrm{el\,m}}^{j\to i}$ is

$$\sum_{\mathrm{el\,m}}^{j\to i} \Delta u^i = \beta_{\mathrm{m}}[(e^{-(u^{i-1}-u^j)} - e^{-(u^{i-1}-u^{j-1})}) - e^{-\Delta}\Delta u^j] \ . \qquad (2\cdot16)$$

(c) For case of $u^j+\Delta \leqq u^i$ and $u^{j-1}+\Delta \leqq u^{i-1}$, $\sum_{\mathrm{el\,m}}^{j\to i}$ is

$$\sum_{\mathrm{el\,m}}^{j\to i} \Delta u^i = \beta_{\mathrm{m}}[(e^{-(u^{i-1}-u^j)} - e^{-\Delta}) - e^{-\Delta}(u^j+\Delta-u^{i-1})] \ . \qquad (2\cdot17)$$

(d) For case of $u^j+\Delta \geqq u^i$ and $u^{j-1}+\Delta \leqq u^i$, $\sum_{\mathrm{el\,m}}^{j\to i}$ is

$$\sum_{\mathrm{el\,m}}^{j\to i} \Delta u^i = \beta_{\mathrm{m}}[e^{-\Delta}(u^{j-1}+\Delta-u^i) - (e^{-(u^i-u^j)} - e^{-\Delta})] \ . \qquad (2\cdot18)$$

(e) For case of $u^j+\Delta \geqq u^i$ and $u^{j-1}+\Delta \leqq u^{i-1}$, $\sum_{\mathrm{el\,m}}^{j\to i}$ is

$$\sum_{\mathrm{el\,m}}^{j\to i} \Delta u^i = \beta_{\mathrm{m}}[(e^{-(u^{i-1}-u^j)} - e^{-(u^i-u^j)}) - e^{-\Delta}\Delta u^i] \ . \qquad (2\cdot19)$$

The relation between lethargies as mentioned above is shown in schema of **Fig. 2**.

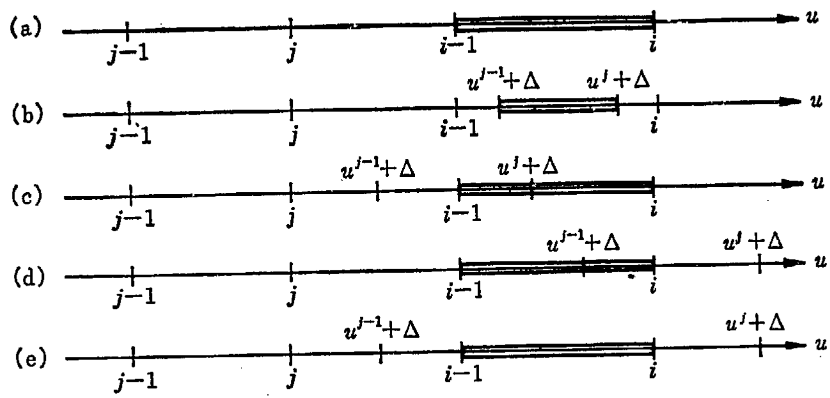


Fig. 2 Schema for slowing Down by Elastical Scattering

A similar matrix of slowing down cross sections by inelastic scattering is also calculated. In lower energy range are given the excited energy Q MeV of an compound nucleus and the inelastic scattering cross section to make the compound nucleus with Q, and, in higer energy range, the energy distribution of neutrons scattered inelastically is in accordance with the Maxwell statistical distribution. Under these assumptions, the matrix is calculated.

2.4 Calculation of neutron flux

Equ. (2·4) leads to simultaneous equations with K unknowns for the i-th energy group, from which vector representation i is derived as follows,

$$\vec{\boldsymbol{R}}_i=\sum_{j=1}^{i}\vec{\boldsymbol{C}}_{j\to i}+\vec{\boldsymbol{SF}}_i+\vec{\boldsymbol{ST}}_i\ , \tag{2·20}$$

where

$$\vec{\boldsymbol{R}}_i=[R_i]\vec{\Phi}^i=\Delta u^i\begin{pmatrix}\Sigma_{T1}^{i}V_1 & 0 & 0\cdots\cdots 0\\ 0 & \Sigma_{T2}^{i}V_2 & 0\cdots\cdots 0\\ \vdots & \vdots & \vdots\\ 0 & 0 & \Sigma_{TK}^{i}V_K\end{pmatrix}\begin{pmatrix}\Phi_1^i\\ \Phi_2^i\\ \vdots\\ \Phi_K^i\end{pmatrix} \tag{2·21}$$

$$\vec{\boldsymbol{C}}_{j\to i}=[C_{j\to i}]\vec{\Phi}^i=\Delta u^j\begin{pmatrix}P_{11}^i\Sigma_{s1}^{j\to i}V_1 & P_{21}^i\Sigma_{s2}^{j\to i}V_2\cdots\cdots P_{K1}^i\Sigma_{sK}^{j\to i}V_K\\ P_{12}^i\Sigma_{s1}^{j\to i}V_1 & P_{22}^i\Sigma_{s2}^{j\to i}V_2\cdots\cdots P_{K2}^i\Sigma_{sK}^{j\to i}V_K\\ \vdots & \vdots\qquad\qquad\qquad\vdots\\ P_{1K}^i\Sigma_{s1}^{j\to i}V_1 & P_{2K}^i\Sigma_{s2}^{j\to i}V_2\cdots\cdots P_{KK}^i\Sigma_{sK}^{j\to i}V_K\end{pmatrix}\begin{pmatrix}\Phi_1^j\\ \Phi_2^j\\ \vdots\\ \Phi_K^j\end{pmatrix} \tag{2·22}$$

$$\vec{\boldsymbol{SF}}_i=[C_i]\vec{\boldsymbol{SF}}^i=\begin{pmatrix}P_{11}^iV_1 & P_{21}^iV_2\cdots\cdots\cdots\cdots P_{K1}^iV_K\\ P_{12}^iV_1 & P_{22}^iV_2\cdots\cdots\cdots\cdots P_{K2}^iV_K\\ \vdots & \vdots\qquad\qquad\qquad\vdots\\ P_{1K}^iV_1 & P_{2K}^iV_2\cdots\cdots\cdots\cdots P_{KK}^iV_K\end{pmatrix}\begin{pmatrix}SF_1^i\\ SF_2^i\\ \vdots\\ SF_K^i\end{pmatrix} \tag{2·23}$$

$$SF_k^i=\sum_{j=1}^{I}[(\upsilon\Sigma_f)_k{}^j\,\Phi_k^j\,\Delta u^j]X^i\,\Delta u^i \tag{2·23$'$}$$

$$\vec{\boldsymbol{ST}}_i=[\bar{C}_i]\vec{ST}_i=\begin{pmatrix}\bar{P}_{11}^iV_1 & \bar{P}_{21}^iV_2\cdots\cdots\cdots\cdots\bar{P}_{K1}^iV_K\\ \bar{P}_{12}^iV_1 & \bar{P}_{22}^iV_2\cdots\cdots\cdots\cdots\bar{P}_{K2}^iV_K\\ \vdots & \vdots\qquad\qquad\qquad\vdots\\ \bar{P}_{1K}^iV_1 & \bar{P}_{2K}^iV_2\cdots\cdots\cdots\cdots\bar{P}_{KK}^iV_K\end{pmatrix}\begin{pmatrix}ST_1^i\\ ST_2^i\\ \vdots\\ ST_K^i\end{pmatrix} \tag{2·24}$$

$$ST_k^i=S_kX^i\,\Delta u^i\ . \tag{2·24$'$}$$

For solving equ. (2·20), we can use the following two methods;

(1) $\vec{\boldsymbol{R}}_i-\sum_{j=1}^{i}\vec{\boldsymbol{C}}_{j\to i}=\vec{\boldsymbol{SF}}_i+\vec{\boldsymbol{ST}}_i$ is derived from equ. (2·20), and we then calculate $\vec{\Phi}^i$ from inversed matrix of the left hand in the above equation, and (2) $\vec{\Phi}$ is divided up into two contributions due to $\vec{\boldsymbol{SF}}_i$ and $\vec{\boldsymbol{ST}}_i$ which will be separately calculated, and afterwards it is superposed from these contributions. The former is not used because the matrix inversion becomes complicate, and the latter is used here, where $\bar{\Phi}_k^i$ is the flux of neutrons which are born in the k-th region and the i-th energy group from the fast fission source, and $\hat{\Phi}_k^i$ is the same born from the thermal fission source. These neutron fluxes are calculated from the following equations,

$$\vec{\hat{\Phi}}^i;\ [R_i]\vec{\hat{\Phi}}^i=\sum_{j=1}^{i}[C_{j\to i}]\Phi^i+[\hat{\bar{C}}_i]\vec{ST}^i \tag{2·25}$$

$$\vec{\bar{\Phi}}^i;\ [R_i]\vec{\bar{\Phi}}^i=\sum_{j=1}^{i}[C_{j\to i}]\vec{\bar{\Phi}}^i+[C_i]\vec{SF}^i \tag{2·25$'$}$$

where $\vec{\hat{\Phi}}^i$ and $\vec{\bar{\Phi}}^i$ are K dimensional colume vectors which have $\hat{\Phi}_k^i$ and $\bar{\Phi}_k^i$ as the k-th element, respectively.

Next we use the following iteration method for the calculation of $\vec{\bar{\Phi}}^i$; first $\vec{\hat{\Phi}}^i$ is easily calculated by equ. (2·25) as $\vec{ST}^i$ is given, and $(\vec{SF}^i)_0$ is derived from $\vec{\hat{\Phi}}^i$ by equ. (2·23)′ and leads to a first result of $(\vec{\bar{\Phi}}^i)_0$ by equ. (2·25)′. Next a second result of $(\vec{\bar{\Phi}}^i)_1$ is similary given from the above results of $(\vec{\bar{\Phi}}^i)_0$. Thus, after n times of the interation, $\vec{\phi}^i$ is

$$\vec{\Phi}^i=\vec{\hat{\Phi}}^i+(\vec{\bar{\Phi}}^i)_0+(\vec{\bar{\Phi}}^i)_1+\cdots\cdots\cdots\cdots\cdots\cdots+(\vec{\bar{\Phi}}^i)_n\ . \tag{2·26}$$

(i) Detail of iteration method for the calculation of the neutron flux

For the case where there are two regions with fission source in lattices, the iteration method as mentioned above will be presented in detail. Since only the terms of ST_l are given at beginning of the iteration, first results of the fast fission source $(SF_l)_0$ are derived from

$$\begin{pmatrix} SF_1 \\ SF_2 \end{pmatrix}_{(0)}=\begin{pmatrix} \bar{S}_{11} & \bar{S}_{12} \\ \bar{S}_{21} & \bar{S}_{22} \end{pmatrix}\begin{pmatrix} ST_1 \\ ST_2 \end{pmatrix}\ , \tag{2·27}$$

where $\bar{S}_{lk}$; the source of fast fission neutrons in the l-th region due to the unit fission source of thermal neutrons in the k-th region, and ST_l; thermal fission source in the l-th region,

$$ST_l^i=ST_l\,X^i\Delta u^i\ . \tag{2·27)′$$

When fast neutron sources of SF_1 and SF_2 are in the first and second regions at $(n-1)$-th iteration time, respectively, the sames at next n-th time are represented as

$$\begin{pmatrix} SF_1 \\ SF_2 \end{pmatrix}_{(n)}=\begin{pmatrix} S_{11} & S_{12} \\ S_{21} & S_{22} \end{pmatrix}\begin{pmatrix} SF_1 \\ SF_2 \end{pmatrix}_{(n-1)} \tag{2·28}$$

where S_{lk}; the similar source from the fast fission to $\bar{S}_{lk}$, and SF_l; fast fission source in the l-th region,

$$SF_l^i=SF_l\,X^i\,\Delta u^i. \tag{2·28)′$$

From combination of equs. (2·27) and (2·28), all neutron sources in this system are given as summation of the following two sources,

$$\text{Thermal fission source}=\begin{pmatrix} ST_1 \\ ST_2 \end{pmatrix}\ ,\ \text{and} \tag{2·29}$$

$$\begin{aligned}\text{Fast fission source}&=\begin{pmatrix} SF_1 \\ SF_2 \end{pmatrix}=\begin{pmatrix} SF_1 \\ SF_2 \end{pmatrix}_{(0)}+\begin{pmatrix} SF_1 \\ SF_2 \end{pmatrix}_{(1)}+\cdots\cdots\\ &\quad+\begin{pmatrix} SF_1 \\ SF_2 \end{pmatrix}_{(n)}+\cdots\cdots\\ &=\left[1-\begin{pmatrix} S_{11} & S_{12} \\ S_{21} & S_{22} \end{pmatrix}\right]^{-1}\begin{pmatrix} \bar{S}_{11} & \bar{S}_{12} \\ \bar{S}_{21} & \bar{S}_{22} \end{pmatrix}\begin{pmatrix} ST_1 \\ ST_2 \end{pmatrix}\ . \end{aligned}\tag{2·30}$$

Thus, since the neutron sources were given, the final result for total neutron flux $\vec{\Phi}^i$ is given by

$$\vec{\Phi}_i=ST_1\cdot{}_1\vec{\hat{\Phi}}^i+ST_2\cdot{}_2\vec{\hat{\Phi}}^i+SF_1\cdot{}_1\vec{\bar{\Phi}}^i+SF_2\cdot{}_2\vec{\bar{\Phi}}^i\ , \tag{2·31}$$

where ${}_l\bar{\Phi}_k^i$; the neutron flux at the i-th energy group in the k-th region due to unit source by fast fission in the l-th region,

${}_l\hat{\Phi}_k^i$; the similar neutron flux from the thermal fission to ${}_l\bar{\Phi}_k^i$,

${}_l\vec{\bar{\Phi}}^i$; the K dimensional colume vector with the element of ${}_l\bar{\Phi}_k^i$, and

${}_l\vec{\hat{\Phi}}^i$; the same with the one of ${}_l\hat{\Phi}_k^i$,

(ii) Derivations of S_{lk}, $\bar{S}_{lk}$ and ${}_l\vec{\bar{\Phi}}^i$, ${}_l\vec{\hat{\Phi}}^i$

(a) Case where there is only fast fission source in the l-th region (S_{lk})

For this case, equ. (2·25)′ is changed to

$$\{[R_i]-[C_{i\to i}]\}_l\vec{\bar{\Phi}}^i=\sum_{j=1}^{i-1}[C_{j\to i}]_l\vec{\bar{\Phi}}^j+[C_i]\vec{SF}^i$$

where equ. (2·28)′ is used as $SF_kV_k=1$ at $k=l$ and $SF_kV_k=0$ at $k\neq l$. Namely, as $i=1$,

$$_l\vec{\bar{\Phi}}^1=\{[R_1]-[C_{1\to 1}]\}^{-1}[C_1]\vec{SF}_1\ , \tag{2·32}$$

and, as $i=n$,

$$_l\vec{\bar{\Phi}}^n=\{[R_n]-[C_{n\to n}]\}^{-1}\{\sum_{j=1}^{n-1}[C_{j\to n}]_l\vec{\bar{\Phi}}^j+[C_n]\vec{SF}^n\}\ . \tag{2·32}'$$

Using these results of $_l\vec{\bar{\Phi}}^i$, S_{lk} is given by

$$S_{lk}=\sum_{j=1}^{I}[(\nu\Sigma_f)^j{}_k\,{}_l\vec{\Phi}_k{}^j\Delta u^j]V_l/V_k\ . \tag{2·33}$$

(b) Case where there is only thermal fission source in the l-th region ($\bar{S}_{lk}$)

For this case, derivation of $\bar{S}_{lk}$ is quite similar to the above case, except for use of equ. (2·25) and (2·27)′.

Then $\bar{S}_{lk}$ is

$$\bar{S}_{lk}=\sum_{j=1}^{I}[(\nu\Sigma_f)_k{}^j\,{}_l\hat{\Phi}_k{}^j\Delta u^j]V_l/V_k\ . \tag{2·34}$$

2.5 Calculation of fast multiplication factor

A fast multiplication factor λ is defined as decreasing rate of the neutron flux per one generation in the fast fission system when the external neutron source, i. e. the thermal fission source, disappears.

Thus λ is derived from equ. (2·28) as follows,

$$\begin{pmatrix}S_{11} & S_{12}\\ S_{21} & S_{22}\end{pmatrix}\begin{pmatrix}ST_1\\ ST_2\end{pmatrix}=\lambda\begin{pmatrix}ST_1\\ ST_2\end{pmatrix}\ . \tag{2·35}$$

Thence an eigen value is calculated by solving the determinant of

$$\begin{vmatrix}S_{11}-\lambda & S_{12}\\ S_{21} & S_{22}-\lambda\end{vmatrix}=0\ , \tag{2·36}$$

i. e. λ is equal to

$$\lambda=\frac{1}{2}\left[(S_{11}+S_{21})+\sqrt{(S_{11}+S_{22})^2-4(S_{11}S_{22}-S_{12}S_{21})}\right]. \tag{2·37}$$

3. Collision probabilities

3.1 Slab lattice

A general formulation of the collision probability for an infinite fuel slab array consists of an infinite series of the En-function. However, when optical chord lengths in the slab lattice become small in the calculation of fast neutron behavior, convergence of the series becomes worse and the numerical value of the En-function also needs to become fairly accurate. Here we calculate the probability by integrating the expression directly but not by expanding it into such series. Thus it is considered to be able to keep an accuracy of the numerical results of the probability.

The collision probability averaged over the energy range between the lethargies u^i and u^{i-1} is given by use of a weighting function $\Sigma_{s(u)}\Phi_{(u)}$ as follows,

$$P_{kl}{}^i=\frac{1}{V_k\Sigma_{Tk}(u^i)}\int_0^1 u\boldsymbol{K}_{kl}(\mu,u^i)[1-\Gamma_k(u^i)][1-\Gamma_l(u^i)]d\mu\ , \tag{3·1}$$

where

$$\boldsymbol{K}_{kl}(\mu)=K_{kl}(\mu)+\overline{K}_{kl}(\mu)+K_{M-k+1,l}(\mu)+\overline{K}_{M-k+1,l}(\mu) \tag{3·2}$$

$$\left.\begin{aligned}\boldsymbol{K}_{kl}(\mu)&=\frac{\Gamma_1\cdot\Gamma_2\cdots\cdots\Gamma_{k-1}\cdot\Gamma_{l+1}\cdot\Gamma_{l+2}\cdots\cdots\Gamma_M}{1-\Gamma}\ ;\ k\leqq l\\ &=\frac{\Gamma_{l+1}\cdot\Gamma_{l+2}\cdots\cdots\cdots\cdots\cdots\cdots\Gamma_{k-2}\cdot\Gamma_{k-1}}{1-\Gamma}\ ;\ k\geqq l-1\end{aligned}\right\} \tag{3·3}$$

$$\left.\begin{aligned}K_{kl}(\mu)&=\frac{\Gamma_{k+1}\cdot\Gamma_{k+2}\cdots\cdots\cdots\cdots\cdots\cdots\Gamma_{l-2}\cdot\Gamma_{l-1}}{1-\Gamma}\ ;\ k\leqq l-1\\ &=\frac{\Gamma_1\cdot\Gamma_2\cdots\cdots\Gamma_{l-1}\cdot\Gamma_{k+1}\cdot\Gamma_{k+2}\cdots\cdots\Gamma_M}{1-\Gamma}\ ;\ k\geqq l\end{aligned}\right\} \tag{3·4}$$

and

$$\Gamma_l=\exp\left[-\frac{\Sigma_{Tk}(u^i)}{|\mu|}(x_{l+1}-x_l)\right]\ . \tag{3·5}$$

3.2 Cylindrical rod lattice

An exact expression of the collision probability for a cylindrical rod lattice has been derived by the author[1]. The result is

$$P_{kl}{}^i=P_{kl}{}^{*i}+2W\sum_{n=1}^{\infty}\sum_{m=0}^{\infty}P^i{}_{klnm}\quad ;\quad k<l \tag{3·6}$$

with $W=2$ is for a rectangular lattice and $W=3$ for a hexagonal lattice, where $P_{kl}{}^*$ is the collision probability that neutrons born uniformly and isotropically in the k-th region of any unit cell in the lattice will have their next collision in the l-th region of the same unit cell at lethargy u^i, and is shown as

$$\begin{aligned}P_{kl}{}^{*i}=\frac{x_k}{\Sigma_{Tk}V_k}\int_0^{\pi/2}\sin^2\theta\,d\theta\int_{-1}^{1}dv\Big\{&e^{-a_{k+1}^{l-1}}(1-e^{-2a_1{}^k})\\ &-e^{-a_k^{l-1}}(1-e^{-2a_1{}^{k-1}})-e^{-a_{k+1}^{l}}(1-e^{-2a_1{}^k})-e^{-a_k^{l}}(1-e^{-2a_1{}^{k-1}})\Big\}\end{aligned} \tag{3·7}$$

with

$$a_n^m=x_k\sum_{q=n}^{m}\Sigma_{Tq}(u^i)\left[\sqrt{(x_q/x_k)^2-v^2}-\sqrt{(x_{q-1}/x_k)^2-v^2}\right]\cos\theta\ . \tag{3·8}$$

And $P_{klnm}{}^i$ is the collision probability that neutrons born uniformly and isotropically in the k-th region of the unit cell will have their next collision in the l-th region of the nm-th cell which is located in the n-th row and the m-th line from the unit cell in which the neutrons are born at lethargy u^i, but the expression is not represented here because of its complexity.

Since it needs long computing time to calculate $P_{kl}{}^i$ for all i by use of equ. (3·6), an approximate equation by the equivalent unit cell method with the isotropic reflecting condition at the cylindricalized cell boundary is used for the calculation of all energy groups, and then the calculated results are corrected by using the accurate values which are calculated for a part of all the groups by use of equ. (3·6). This cylindrical cell approximation has been obtained independently by H. Kiesewetter[2] and E. M. Pennington[3]. The expression derived by them is here extended to a multi-region lattice as follows,

$$P_{kl}{}^i=P_{kl}{}^{*i}+P_{kp}{}^{*i}\cdot G_{pl}{}^i\Big/\sum_{m=1}^{m}G_{pm}{}^i \tag{3·9}$$

with

$$G_{pm}{}^i=\frac{4\,\Sigma_{Tm}(u^i)V_m}{S_M}P_{mq}{}^{*i}\ , \tag{3·10}$$

where p shows the index of the probability that neutrons born in the unit cell escape from the cylindricalized cell boundary. Since it has been particularly represented by the author[4] that even equ. (3·9) is in good agreement with equ. (3·6), we can get a more accurate value by the above mentioned procedure than that by use of equ. (3·9).

For case where the fuel region is located at the center of the unit cell, we calculate the collision probability due to thermal fission source, a spatial distribution of which is represented as as the following function of square radius x^2,

$$S(x)=1+Bx^2 \quad \text{with constant B.} \tag{3·11}$$

Although an exact expression of this probability can be derived, since equ. (3·11) does not strictly represent the thermal fission distribution and the exact expression would be speculated to be extremely complicated, the probability is approximately derived as follows; first we calculate a ratio $S^i(u^i)$ of the collision probability $SP_{1l}{}^{*i}(u^i)$ that neutrons from the center region will have their next collision in the i-th one in the same unit cell having the spatial distribution of equ. (3·11) to the collision probability $P_{1l}{}^{*i}(u^i)$ with the flat source distribution, and next the collision probability with the source distribution of equ. (3·11) in the lattice is obtained as follows,

$$\bar{P}_{1l}{}^i(u^i)=S^i(u^i)P_{1l}{}^i(n^i) \ . \tag{3·12}$$

Then $SP_{1l}{}^{*i}(u^i)$ is defined as

$$SP_{1l}{}^{*i}(u^i)=\frac{\int\Omega\int_{V_1}S(x)P_{1l}(x,\ \theta,\ \phi)d\nu\, d\Omega}{4\pi\int_{V_1}S(x)d\nu}\ , \tag{3·13}$$

where $P_{1l}(x,\ \theta,\ \phi)$ is the collision probability that neutrons born at x will have their next collision in the l-th region of the same cell.

After some algebraic calculations, $SP^*{}_{1l}(u^i)$ is

$$SP_{1l}{}^{*i}(u^i)=\frac{1}{2\sum_{T1}x_1}(Q_{l-1}-Q_l) \tag{3·14}$$

with

$$Q_l=\frac{2}{\pi(1+Bx_1{}^2/2)}\left[(1+Bx_1{}^2)M_0-\frac{2B}{\sum_{T1}}M_1+\frac{2B}{\sum_{T1}{}^2}M_2\right]\ , \tag{3·15}$$

where

$$M_0=\int_{-1}^{1}d\nu[K_{i3}(d_1)-K_{i3}(d_1+2d_2)] \tag{3·16a}$$

$$M_1=\int_{-1}^{1}d\nu\, d_2[K_{i4}(d_1)+K_{i4}(d_1+2d_2)] \tag{3·16b}$$

$$M_2=\int_{-1}^{1}d\nu[K_{i5}(d_1)-K_{i5}(d_1+2d_2)] \tag{3·16c}$$

$$d_1=x_1\sum_{q=2}^{l}\ \sum_{Tq}(u^i)\left[\sqrt{(x_q/x_1)^2-\nu^2}-\sqrt{(x_{q-1}/x_1)^2-\nu^2}\right]$$

and

$$d_2=\sum_{T1}x_1\sqrt{1-\nu^2}$$

When $\sum_{1x1}$ is smaller than 0. 055, the following equation derived from expansion of M_1 and M_2 should be used instead of equ. (3·15),

$$Q_l=\frac{2}{\Pi(1+Bx_1{}^2/2)}\left[(1+Bx_1{}^2)M_0-\frac{4}{3}\frac{B}{\sum_{T1}{}^2}M_3\right] \tag{3·17}$$

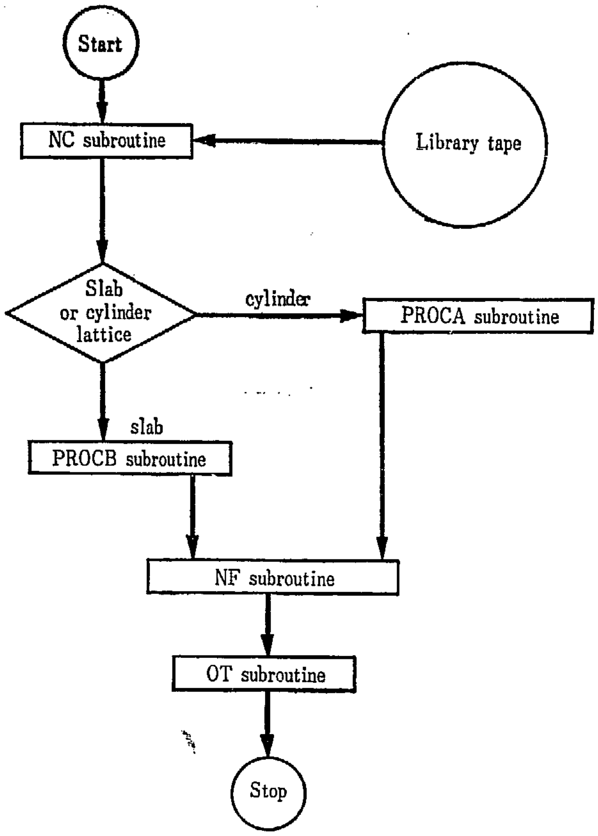
with

$$M_3=\int_{-1}^{1}d\nu\, d_2{}^3[K_{i2}(d_1)-d_2\,K_{i1}(d_1)]\ . \tag{3·18}$$

4. Description of EPSILON code

4.1 Flow of calculation

The flow of calculation in EPSIRON code is shown in Fig. 3.



NC subroutine; Macroscopic cross sections are calculated from microscopic cross sections in the library tape.

PRφCB subroutine; Collision probabilities in a slab lattice are calculated by use of equs. (3·1)~(3·5)

PRCA subroutine; Collision probabilities in an actual cylindrical rod lattice are calculated by use of equs. (3·6)~(3·18).

NF subroutine; Fast neutron flux distribution is calculated by the method represented in the chapter 2-4, and fast multiplication factor is calculated by use of eqn (2-37).

OT subroutine; Fast fission effect, ratio of fast fission rate of thermal fission rate, fast fission rates, slowing down rates and absorption rates are calculated.

Fig. 3 Block Chart of EPSIRON Code

4.2 Input and output

(i) Input

From input cards;

Choice of lattice configuration	MG=1 slab =2 square =3 hexagonal
Maximum region number	MR≦5
Maximum energy group number	ML≦100
Choice of fission spectrum	MS indicates no. of F. S. in library
Indication of the fuel region	IS
Spatial distribution of neutron source	STF(l)
Thickness of region	RAD(l)cm
Indication of composition in each region	MC(l)
Composition number	$I \leqq 10$
Indication of element	A(I, J) the J-th element in the I-th composition
Density of the element of the composition	AN(I, J) nucleus/cm^3
Number of element of the composition	$J \leqq 5$
Control of output print	$ISS1$-$ISS6$

From library tape;

Total, elastic, inelastic and fission cross section and fission number	$\sigma(i)$ σ_T, σ_s, σ_m, σ_f, $\nu\sigma_f$
Matrix of scattering cross section	$\sigma_s(j \to i)$
Lethargy width	$\Delta u(i)$
Fission spectrum	$X(i)$
Atomic number of element	$A(j)$

(ii) Input format

(a) Title card

Col. No.	format		Content
1	I 1	IIS 1	Neutron flux print=0; do, =1; do not
2	I 1	IIS 2	Cross sections of region print=0; do, =1; do not
3	I 1	IIS 3	Probabilities print=0; do,=1; do not
4	I 1	IIS 4	Scattering matrix of region print=0; do, =1; do not
5	I 1	IIS 5	Cross sections of element print=0; do, =1; do not
6	I 1	IIS 6	Scattering matrix of element print=0; do, =1; do not
7—66	A 6		≦06 characters of title

(b) Control card

1	I 1		=1 indicates the control card
2— 5	I 4	MG	=1; slab, =2; square, =3; hexagonal
6—10	I 5	MR	≤5, maximum region number
11—20	E 10.5	BL	=B in equ. (3·11)
21—25	I 5	ML	≤100, maximum energy group number
26—30	I 5	MS	Spectrum no. of library
31—45	—	—	Blank
46—50	I 5	I	Region no. of fuel region
51—60	E 10.5	STF(I)	Neutron source strength of the I-th region, n/cm^3
61—65	—	—	Blank
66—70	I 5	I	Same as col. 46—50
71—80	E 10.5	STF(I)	Same as col. 51—60

(c) Region card

4 regions/one card

1	I 1		=2 indicates the region card
2— 5	I 4	±J	≦5, fuel region; -J, moderator; J
6—10	I 5	MC(J)	≦10, =0; air gap
11—20	E 10.5	RAD(J)	Thickness of the J-th region
21—25	I 5	±J	Same as col. 2—02
26—30	I 5	MC(J)	Same as col. 2—02
31—40	E 10.5	RAD(J)	Same as col. 2—02

When region no. ≦ 4, one card needs and when region no. =5, two cards need.

(d) Composition card

Col. No.	format		Content
1	I 1		=3 indicates the composition card
2—80			Blank

4 composition card/one card

1— 2	I 2	M	≦10 M=MC(J)
3— 4	I 2	N	≦10 N-th element of the M-th composition
5—10	A 6	A(N, M)	N-th element name of the M-th composition
11—20	E 10.5	AN(N, M)	Atomic number density (10^{24}/cm^3) of the above
21—22	I 2	M	Same as col. 1—20
23—24	I 2	N	Same as col. 1—20
25—30	A 6	A(N, M)	Same as col. 1—20
31—40	E 10.5	AN(N, M)	Same as col. 1—20

When composition no. ≤4, one card needs, and when composition 4≤ no. ≤8, two cards need.

(e) **End card** ………… **Blank card**

(iii) Output

1st page; Input control, multiplication factor λ, fast fission factor ε, primary generated fast neutron source, secondary generated fast neutron source.

2nd page; Fast fission ratios δ_{ml}, fast fission effects $(\varepsilon-1)_{ml}$ absorption rates, slowing down rates (or fast fission rates).

3rd page; Cross sections, neutron fluxes, collisions probabilities, matrix of scattering cross sections.

4.3 Restricted conditions

Maximum energy group number ML$\leqq$100

Maximum region number MR$\leqq$5

Lattice configuration; slab, square and hexagonal cylindrical rod lattices.

Fission spectrum number MS$\leqq$no. of F. S. in library.

Number of fuel region SR$\leqq$2

Element number in region I$\leqq$10

4.4 Code manual and notice

(i) Tape usage

Logical No.	Actual No.	Content
1	B1	System tape
2	B2	Scratch tape
3	B3	Chain (1, 3)
4	A4	Chain (2, 4)
5	A2	Input tape
6	A3	Output tape
7	B4	Binary tape
8	B1	Group constants for each element
9	A5	Library tape
10	B5	Collision probability
11	A6	Group constants
12	B6	Group constants for flux calculation

(ii) Sense light usage

Chain	Subroutine	Sense Light			
		1	2	3	4
1	Main	on	off	off	off
	RD	on	on	off	off
	NC	on	off	on	off
	RLT	on	off	off	on
2	Main	off	off	off	off
	PROCA or PROCB	off	on	off	off
	NF	off	off	on	off
3	Main	on	on	on	off
	OT	on	on	on	off
	WD	on	on	off	on

(iii) Notice

(a) It is better that the elements in each region are prepared according to the order of the atomic number, if possible.

(b) Since the computing time for the calculation of macroscopic cross sections is more consumed, if one uses this code for a parametric calculation, one should notice the input data arragement.

(c) If one obtains a reaction rate of any other elements by using the calculated fast neutron flux, one may add the calculating routine only to the OT subroutine and recomiple it. But the number of the added reaction rate must be less than 4.

4.5 Computing time

Problem;

Nnmber of Energy Groups =18
Number of Regions =5
Number of Elements in an Unit Cell =10

Computing Time; 3 min.

{ 2.5 min. by calculation of cross sections
{ 0.5 min. by remained calculation

4.6 Sample problem

Lattice; Slightly enriched U-H_2O moderated hexagonal lattice with Al cladding.

Library; Data from H. Rief[5]

PROBLEM Sample problem WRITTEN BY DATE PAGE OF

INPUT DATE A | COMMENT

```
1   5    10   15   20   25   30   35   40   45   50   55   60   65   70  |73 75  80
   111 CASE 210  1.0 ENRICHED U-L.W., E=6MEV-0.1MEV,RAT=1-1  ,T=20
1   3    3 0.0        18     1                   1 1.0
2   1    1 0.4915       2    2 0.07112      3    3 0.1845
3
 1 1  U238 0.04685   1 2  U2350.0004685  2 1 AL    0.06020   3 1  H    0.06682
 3 2  O 16 0.03341
```

Fig. 4

References

1) Y. Fukai; *J. of Nuclear Energy*, **17**, 115 (1964).
2) H. Kiesewetter; *Kernerergie*, **6**, 106 (1963).
3) E.M. Pennington; *Nucle. Scie. Engng.*, **19**, 215 (1964).
4) Y. Fukai; *Nukleonik*, **7**, 144 (1965).
5) H. Rief; BNL-646 (T-206) Jan. (1961).

APPENDIX EPSILON-LT, for making library tape of EPSILON code

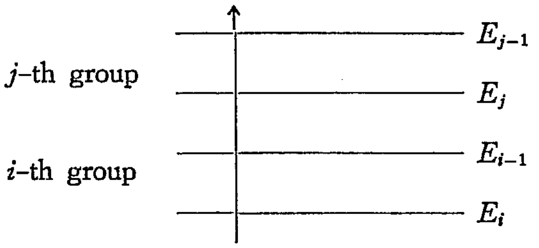
1. Calculating method

(i) Matrix of elastic scattering cross sections

They are calculated by use of equs. (2·14)-(2·19) in text.

(ii) Matrix of inelastic scattering cross sections

(a) Case where Q value is given



When Q is the excited energy of an compound nucleus in MeV, σ_Q the inelasticscattering cross section to make the compound nucleus with Q, and $\Delta E_j = E_{j-1} - E_j$, $\Delta E_i = E_{i-1} - E_i$, $QE_{j-1} = E_{j-1} - Q$, we obtain

$$\left.\begin{aligned} \sigma_{in,Q}^{j\to i} &= \sigma_Q \frac{\Delta E_i}{\Delta E_j} \quad ; QE_{j-1} \geqq E_{i-1},\ QE_j \leqq E_i \\ &= \sigma_Q \frac{E_{i-1} - QE_j}{\Delta E_j} \quad ; QE_{j-1} \geqq E_{i-1},\ QE_j \geqq E_i \\ &= \sigma_Q ;\ QE_{j-1} \leqq E_{i-1}, QE_j \geqq E_i \\ &= \sigma_Q \frac{QE_j - E_i}{\Delta E_j} \quad ; QE_{j-1} \leqq E_{i-1},\ QE_j \leqq E_i \end{aligned}\right\} \qquad (A\cdot 1)$$

and

$$\sigma_{in}^{j\to i} = \sum_{Q=1}^{MQ} \sigma_{in,Q}^{j\to i} , \qquad (A\cdot 2)$$

where MQ is the total level number to permit the j-th group.

(b) Case where statistical energy distribution is given

Assuming the Maxwell distribution as the statistical one, the energy spectrum $N(E)$ is given as

$$N(E) = \beta E e^{-E/\theta} \quad \text{with} \quad \theta = kT = k\alpha\sqrt{E_0} , \qquad (A\cdot 3)$$

where E is neutron energy after inelastically scattering, E_0 the same before scattering, k Boltzmann's constant, α the constant and β the normalization constant as

$$\int_{\text{all}E} N(E)dE = 1.$$

Using $E_0 = (E_{j-1} + E_j)/2$, the scattering cross section from the j-th group to the i-th group is given as

$$\sigma_{in}^{j\to i} = \beta\theta[(E_i+\theta)e^{-E_i/\theta} - (E_{i-1}+\theta)e^{-E_{i+1}/\theta}]\sigma_{in}^{j} , \qquad (A\cdot 4)$$

where σ_{in}^{j} is the inelastic scattering cross section of the j-th energy group.

(iii) Fission spectrum

$$X(E_i) = \int_{E_i}^{E_{i-1}} Ce^{-E/A} \sinh\sqrt{BE}dE , \qquad (A\cdot 5)$$

where A and B are the constant, C is the normalization constant and E is in MeV.

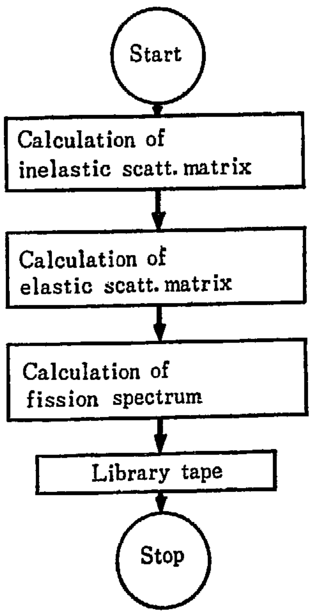


Fig. 5

2. Flow of calculation (Fig. 5)

3. Input and output

(i) Input format

(a) Title card

Col. No.	format	Content
1—72	A6	≦72 characters of title

(b) Cotrol card

1— 5	I 5	L	Indication of library tape making procedure, =0; partial change, =1; whole change
6—10	I 5	NGRP	Maximum numder of energy groups of input data
11—15	I 5	NFE	Element number of input data=number of element card
16—20	I 5	NFSS	Number of input fission spectrum
21—25	I 5	MNGIN	Maximum energy group number of elastic scattering matrix
26—30	I 5	MNGRP	The same of library to be made
31—40	E 10. 5	EG(1)	Maximum energy of neutron spectrum

(c) Element card One card/one element

1— 4	—	—	Blank
5—10	A5	FLN	Element name
11—12	I 2	IA	Indication to discriminate same elements which have different values, in library
13—14	I 2	IB	Atomic number
15—17	I 3	IC	Mass number
18—20	I 3	MGIN	Group number of inelastic scatt. matrix.
21—74	A6	ELNϕC	≦54 characters, about source data of the element, reference etc.

(d) Cross section card

A—card One card/one element

1— 6	A6	ELN	Element name
7— 9	I 3	*I	Number of fission spectrum
10			Blank
11—20	E 10. 5	*EMAX	The highest energy of fission spectrum
21—30	E 10. 5	*EMIN	The lowest energy of the above
31—40	E 10. 5	*AS	Constant of fission spectrum formula
41—50	E 10. 5	*BS	
51—60	E 10. 5	*CS	$N(E)=(CS)\exp(-E/(AS))\sinh\sqrt{(BS)\cdot E}$
61—70	E 10. 5	ν_0	Fission neutron number of ELNϕelement
71—80	E 10. 5	α	$=d\nu/dE$; $\nu(E)=\nu_0+\alpha E$

* must be blank when they are unnecessary.

B—card NGRP cards/one element

Col. No.	format		Content
1— 6			Blank
7— 9	I 3	J	J-th energy group number
10			Blank
11—20	E 10. 5	EG(J)*	The lowest energy of J-th group (Mev)

21—30	E 10. 5	SA(J)	Absorption cross section of the above (cm^2)
31—40	E 10. 5	SS(J)	Elastic scatt. cross section of 〃 (cm^2)
41—50	E 10. 5	SIN(J)	Inelastic scatt. cross section of 〃 (cm^2)
51—60	E 10. 5	SF(J)	Fission cross section of 〃 (cm^2)

* must be blank if this information has already been put.

(e) Scattering cross section matrix card—type I

C—card one card/one element

1— 6	A5	SIGIN	Element name
7— 9	I 3	K1	Indication of k-th energy group card
10	I 1	K2	Indication of $(K2+1)$-th card in k-th group
11—20	E 10. 5	σ_{in}(I)	Inelastic scattering cross section from K-th energy group to I-th group (or I-th level with Q MeV)
21—30	E 10. 5	σ_{in}(I+1)	I=K1+K2×7
⋮	⋮	⋮	
71—80	E 10. 5	σ_{in}(I+6)	

About K1;

+K1:
- $\sum_{I=1}^{NGIN} \sigma_{in}(I) \neq 0$; $\sigma_{in}(I)$ is matrix of inelastic scattering cross section.
- $\sum_{I=1}^{NGIN} \sigma_{in}(I) = 0$; use of $\sigma_{in}(I-1)$ in the upper energy group.

−K1:
- $\sum_{I=1}^{NGIN} \sigma_{in}(I) \neq 0$; calculated by the method shown in 1.-(ii)-(a).
- $\sum_{I=1}^{NGIN} \sigma_{in}(I) = 0$; calculated by the method shown in 1.-(ii)-(b).

D—card necessary number of cards

Col. No.	format		Content
1— 6			Blank
7— 9	I 3	K1	Same as C—card
10	I 1	K2	
11—20	E 10. 5	σ_{in}(I)	
⋮	⋮	⋮	
71—80	E 10. 5	σ_{in}(I+6)	

(f) Scattering cross section matrix card—type II

E—card one card/one element

1— 6	A6	TQ	Any characters
7—10			Blank
11—20	E 10. 5	AT	Value of α in equ. (A-3)

F—card necessary number of cards

1— 6			Blank
7— 9	I 3	M1	Level numbers of compound nucleus
10	I 1	M2	Order number of F—card
11—20	E 10. 5	QL(M)	M-th level energy
21—30	E 10. 5	QL(M+1)	$(M+1)$-th level energy; $M=M2\times 7+1$
⋮	⋮	⋮	
71—80	E 10. 5	QL(M+6)	

Ml must be written only on the first card of F—cards.

(g) Source card

G—card one card/one element

1— 6	A6	SOURCE	Identification of card
7— 9	I 3	N1	Numbering of fission spectrum
10	I 1	N2	Order number of card, from $N=N_2\times 7+1$
11—20	E 10. 5	X(N)	N-th value of N1-th spectrum
21—30	⋮	X(N+1)	$(N+1)$-th 〃 } $N=N_2\times 7+1$
⋮	⋮	⋮	
71—80	E 10. 5	X(N+6)	$(N+6)$-th 〃

H—card necessary number of cards

Col. No.	format		Content
1— 6	I		Blank
7— 9	I 3	N1	............or blank
10	I 1	N2	Same as G—card
11—20	E 10. 5	X(N)	
21—30	⋮	X(N+1)	
⋮	⋮	⋮	
71—80	E 10. 5	X(N+6)	

(ii) Arrangement of input cards

(a) Title card
(b) Control card
(c) Element card
(d) Cross section card
(e) Scatt. cross section matrix card-I } 1st element
(f) Scatt. cross section matrix card-II
one blank card
(d) ....
(e) .... } 2nd element
(f) ....
one blank card
⋮ ⋮ ⋮
(g) Source card } 1st spectrum
one blank card
(g) .... } 2nd spectrum
one blank card
⋮ ⋮ ⋮
Last card (one blank card)

(iii) Output

1. One library tape (No. B6)
2. Off line output print

list of elements in the library
Energy group and fission spectrum
1st element; various cross section vs. energy

$(\sigma_{el}, \sigma_c, \sigma_{in}, \sigma_f, \sigma_t, \nu, \mu\sigma_s)$

matrix of total scatt. cross sections

2nd element; same as above

⋮ ⋮

4. Restricted condisions

(i) Fission spectrum to be calculated in LT code is only of Watt type, and other type must be read as input data.

(ii) Dependence of $\upsilon(E)$ on energy is linear.

(iii) Matrix of inelastic scattering cross sections to be calculated here is the one shown in chapter 1. -(ii) in the Appendix.

(iv) Matrix of elastic scattering cross sections is calculated by assuming isotropic scattering in

PROBLEM Sample problem WRITTEN BY DATE PAGE OF

INPUT DATA

1–10	11–20	21–30	31–40	41–50	51–60	61–70	71–72 / COMMENT 73–80
THOSE INPUT DATA ARE USED FOR CALCULATION OF EPSILON IN BNL-645 BY RIEF							
1 18	1 1	18 50	6.0				
U238	92238 18	U-238 BNL	-645				
U238 1	12.0	0.0	0.965	2.29	0.4527	2.384	0.14
1	5.75	0.0	3.8	2.74	0.70		
2	5.25	0.0	4.14	2.71	0.59		
3	4.75	0.0	4.27	2.70	0.58		
4	4.25	0.008	4.47	2.68	0.58		
5	3.75	0.0125	4.37	2.68	0.58		
6	3.25	0.016	4.33	2.70	0.58		
7	2.75	0.021	4.2	2.80	0.58		
8	2.40	0.027	3.8	2.99	0.58		
9	2.15	0.037	3.40	3.0	0.58		
10	1.90	0.041	3.50	3.0	0.57		
11	1.65	0.054	4.0	2.998	0.50		
12	1.40	0.074	5.06	2.54	0.278		
13	1.15	0.105	4.8	2.2	0.079		
14	0.95	0.138	5.35	2.108	0.026		
15	0.65	0.145	5.2	2.00	0.006		
16	0.40	0.13	7.03	1.87	0.001		

Fig. 6

INPUT DATE

1–10	11–20	21–30	31–40	41–50	51–60	61–70	71–72 / COMMENT 73–80
17	0.25	0.14	8.53	0.53	0.0		
18	0.1	0.21	10.81	0.40	0.0		
U238 -1							
-2							
-3							
-4							
-5							
-6							
-7							
-8							
-9							
-10	0.59	0.35	0.15	0.088	0.51	0.44	0.1
	10.77						
11							
-12	0.65	0.34	0.15	0.28	0.46	0.46	0.20
-13	0.88	0.63	0.088	0.51			
14							
-15	1.45	0.29	0.13				
16							
-17	0.53						
-18	0.40						
U238	0.19						
8	0.044	0.146	0.3	0.73	0.98	1.06	1.24
	11.4						
		blank card	1枚				

Fig. 7

the center-of-mass system.

(v) It is impossibl to change a part of the library type which has already been made.

5. Computing time

For case of 7 elements with 18 energy groups of elastic and inelastic scattering matrices and one fission spectrum, it takes 42 sec. Then computing time of one element with 18 groups is less than 6 sec.

6. Sample problem

Library tape of ^{238}U cross section data is made from data of H. RIEF[5].

CASE 210 1.0 ENRICHED U-L.W. ,E=6MEV-0.1MEV,RAT=1-1 ,T=20 ****** INPUT DATA AND MAIN OUTPUT ******

REGION NUMBER	1	2	3
COMPOSITION	FUEL REGION	MODERATOR REGION	MODERATOR REGION
	U235 0.4695E-03	AL 0.6020E-01	H 0.6682E-01
	U238 0.4685E-01	0.	O 16 0.3341E-01
NEUTRON SOURCE	0.1000E 01	0	0
VOLUME OF REGION	0.7599E 00	0.[illegible]55E-00	0.7589E 00
WIDTH OF REGION	0.4915E-00	0.7112E-01	0.1845E-00

GEOMETRY ------------------ HEX. CYLINDER

NUMBER OF ENERGY GROUP ----- 18

MULTIPLICATION FACTOR ------ 0.1582E-00

EPSILON ----(6.000MEV-0.1000MEV)---- 0.1069E 01 ----(SLOW DOWN BELOW 0.1000E-00MEV/ THERMAL FISSION)

0.1069E 01 ----(TOTAL FISSION / THERMAL FISSION

PRIMARY GENERATED NEUTRONS --- 0.1582E-00 ---(GENERATED FAST NEUTRONS BY NORMALIZED THERMAL FISSION NEUTRONS)
SECONDARY GENERATED NEUTRONS-- 0.2507E-01 ---(GENERATED FAST NEUTRONS BY PRIMARY GENERATED NEUTRONS)

CASE 210 1.0 ENRICHED U-L.W.,E=6MEV-0.1MEV,RAT=1-1 ,T=20 ****** INPUT DATA AND MAIN OUTPUT ******

THE RATIO (SECONDARY NEUTRON/PRIMARY NEUTRON) OF EACH ELE. ---- 0.1675E-00 --(OVER-ALL UNIT CELL)--

	1	2	3
******ELEMENTS	U235 0.1413E-01	AL 0.	H 0.
	U238 0.1739E-00	0.	O 16 0.
REGION TOTAL	0.1880E-00	0.	0.

1-EPSILON (FAST FISSION N./THERMAL FISSION N.) OF EACH ELE ---- 0.69612E-01 --(OVER-ALL UNIT CELL)--

	1	2	3
******ELEMENTS	U235 0.8268E-02	AL -0.1219E-03	H -0.
	U238 0.6244E-01	0.	O 16-0.7792E-03
REGION TOTAL	0.7071E-01	-0.1219E-03	-0.7792E-03

THE FAST NEUTRON CAPTURE(/THERMAL NEUTRON S.) OF EACH ELEM ---- 0.11817E-00 --(OVER-ALL UNIT CELL)--

	1	2	3
******ELEMENTS	U235 0.5864E-02	AL 0.1219E-03	H 0.
	U238 0.1114E-00	0.	O 16 0.7792E-03
REGION TOTAL	0.1173E-00	0.1219E-03	0.7792E-03

SLOWING DOWN NEUTRONS BELOW TD-E (BY ELASTIC OR INELA SCATT)---- 0.10243E 01 --(OVER-ALL UNIT CELL)--

	1	2	3
******ELEMENTS	U235 0.2308E-02	AL 0.1939E-03	H 0.9816E 00
	U238 0.4024E-01	0.	O 16-0.1145E-07
REGION TOTAL	0.4255E-01	0.1939E-03	0.9816E 00

CASE 210 1.0 ENRICHED U-L.W.,E=6MEV-0.1MEV,RAT=1-1 ,T=20 ****** ENERGY-SPACE DISTRIBUTION OF NEUTRON FLUX ******

REG. NO.--- 1 2 3 0 0 REG. NO.--- 1 2 3

ENERGY GROUP	1	2	3
1	0.7576E-01	0.7110E-01	0.7029E-01
2	0.1045E-00	0.9868E-01	0.9724E-01
3	0.1598E-00	0.1524E-00	0.1502E-00
4	0.2307E-00	0.2205E-00	0.2175E-00
5	0.2933E-00	0.2783E-00	0.2728E-00
6	0.4224E-00	0.4034E-00	0.3956E-00
7	0.6639E 00	0.6475E 00	0.6335E 00
8	0.7368E 00	0.7037E 00	0.6972E 00
9	0.8825E 00	0.8512E 00	0.8427E 00
10	0.1053E 01	0.1011E 01	0.9846E 00
11	0.1283E 01	0.1231E 01	0.1199E 01
12	0.1819E 01	0.1769E 01	0.1711E 01
13	0.2185E 01	0.2091E 01	0.2029E 01
14	0.2785E 01	0.2666E 01	0.2587E 01
15	0.4604E 01	0.4449E 01	0.4346E 01
16	0.5436E 01	0.5238E 01	0.5111E 01
17	0.7269E 01	0.6991E 01	0.6815E 01
18	0.9423E 01	0.9275E 01	0.9181E 01

CASE 210 1.0 ENRICHED U-L.W.,E=6MEV-0.1MEV,RAT=1-1 ,T=20 ****** NEUTRON FLUX DISTRIBUTION BY NORMALIZED THERMAL N. SOURCE

REG. NO.--- 1 2 3 0 0 REG. NO.--- 1 2 3

ENERGY GROUP	1	2	3
1	0.8404E-01	0.7886E-01	0.7796E-01
2	0.1159E-00	0.1095E-00	0.1079E-00
3	0.1773E-00	0.1691E-00	0.1666E-00
4	0.2560E-00	0.2446E-00	0.2413E-00
5	0.3253E-00	0.3087E-00	0.3025E-00
6	0.4686E-00	0.4475E-00	0.4388E-00
7	0.7364E 00	0.7183E 00	0.7027E 00
8	0.8173E 00	0.7806E 00	0.7733E 00
9	0.9789E 00	0.9442E 00	0.9346E 00
10	0.1168E 01	0.1121E 01	0.1092E 01
11	0.1423E 01	0.1365E 01	0.1330E 01
12	0.2013E 01	0.1962E 01	0.1898E 01
13	0.2424E 01	0.2320E 01	0.2251E 01
14	0.3039E 01	0.2957E 01	0.2870E 01
15	0.5107E 01	0.4935E 01	0.4821E 01
16	0.6030E 01	0.5810E 01	0.5669E 01
17	0.8064E 01	0.7757E 01	0.7560E 01
18	0.1045E 02	0.1029E 02	0.1018E 02

CASE 210 1.0 ENRICHED U-L.W.,E=6MEV-0.1MEV,RAT=1-1 ,T=20 ****** NEUTRON FLUX DISTRIBUTION BY NORMALIZED SECONDARY N SOURCE

REG. NO.--- 1 2 3 0 0 REG. NO.--- 1 2 3

ENERGY GROUP	1	2	3
1	0.8404E-01	0.7886E-01	0.7796E-01
2	0.1159E-00	0.1095E-00	0.1079E-00
3	0.1772E-00	0.1691E-00	0.1666E-00
4	0.2560E-00	0.2446E-00	0.241[illegible]E-00
5	0.3253E-00	0.3087E-00	0.3023E-00
6	0.4686E-00	0.4475E-00	0.4338E-00
7	0.7264E 00	0.7182E 00	0.7027E 00
8	0.8173E 00	0.7806E 00	0.7733E 00
9	0.9739E 00	0.9442E 00	0.9236E 00
10	0.1168E 01	0.1121E 01	0.1092E 01
11	0.1423E 01	0.1365E 01	0.1330E 01
12	0.2018E 01	0.1962E 01	0.1898E 01
13	0.2424E 01	0.2320E 01	0.2251E 01
14	0.3089E 01	0.2957E 01	0.2870E 01
15	0.5107E 01	0.4935E 01	0.4821E 01
16	0.6030E 01	0.5810E 01	0.5669E 01
17	0.8063E 01	0.7755E 01	0.7560E 01
18	0.1045E 02	0.1029E 02	0.1018E 02

CASE 210 1.0 ENRICHED U-L.W.,E=6MEV-0.1MEV,RAT=1-1 ,T=20 ****** MACROSCOPIC TOTAL CROSS SECTION ******

REG. NO.--- 1 2 3 0 0 REG. NO.--- 1 2 3

ENERGY GROUP	1	2	3
1	0.3424E-00	0.1282E-00	0.1340E-00
2	0.3519E-00	0.1300E-00	0.1500E-00
3	0.3572E-00	0.1338E-00	0.1497E-00
4	0.3661E-00	0.1388E-00	0.1547E-00
5	0.3618E-00	0.1579E-00	0.2095E-00
6	0.3609E-00	0.1629E-00	0.2392E-00
7	0.3598E-00	0.1704E-00	0.1998E-00
8	0.3501E-00	0.1834E-00	0.2031E-00
9	0.3320E-00	0.1903E-00	0.2135E-00
10	0.3364E-00	0.1798E-00	0.2613E-00
11	0.3566E-00	0.1834E-00	0.3077E-00
12	0.3754E-00	0.1850E-00	0.3084E-00
13	0.3394E-00	0.2183E-00	0.3705E-00
14	0.3602E-00	0.1695E-00	0.4501E-00
15	0.3476E-00	0.2408E-00	0.4273E-00
16	0.4266E-00	0.2396E-00	0.6165E 00
17	0.4352E-00	0.2101E-00	0.6656E 00
18	0.5401E 00	0.3203E-00	0.8587E 00

CASE 210 1.0 ENRICHED U-L.W.,E=6MEV-0.1MEV,RAT=1-1 ,T=20 ****** MACROSCOPIC SCATTERING CROSS SECTION ******

REG. NO.--- 1 2 3 0 0 REG. NO.--- 1 2 3

ENERGY GROUP	1	2	3
1	0.3089E-00	0.1253E-00	0.1280E-00
2	0.3237E-00	0.1284E-00	0.1480E-00
3	0.3294E-00	0.1326E-00	0.1473E-00
4	0.3381E-00	0.1380E-00	0.1527E-00
5	0.3334E-00	0.1575E-00	0.2082E-00
6	0.3324E-00	0.1627E-00	0.2392E-00
7	0.3210E-00	0.1704E-00	0.1998E-00
8	0.3210E-00	0.1834E-00	0.2031E-00
9	0.3025E-00	0.1903E-00	0.2135E-00
10	0.3072E-00	0.1798E-00	0.2613E-00
11	0.3301E-00	0.1834E-00	0.3077E-00
12	0.3593E-00	0.1850E-00	0.3084E-00
13	0.3302E-00	0.2183E-00	0.3705E-00
14	0.3519E-00	0.1695E-00	0.4501E-00
15	0.3299E-00	0.2408E-00	0.4273E-00
16	0.4198E-00	0.2396E-00	0.6165E 00
17	0.4279E-00	0.2101E-00	0.6656E 00
18	0.5294E 00	0.3203E-00	0.8587E 00

CASE 210 1.0 ENRICHED U-L.W.,E=6MEV-0.1MEV,RAT=1-1 ,T=20 ****** MACROSCOPIC FISSION CROSS SECTION ******

REG. NO.--- 1 2 3 0 0 REG. NO.--- 1 2 3

ENERGY GROUP	1	2	3
1	0.3350E-01	0.	0.
2	0.2818E-01	0.	0.
3	0.2771E-01	0.	0.
4	0.2772E-01	0.	0.
5	0.2774E-01	0.	0.
6	0.2776E-01	0.	0.
7	0.2778E-01	0.	0.
8	0.2778E-01	0.	0.
9	0.2779E-01	0.	0.
10	0.2731E-01	0.	0.
11	0.2404E-01	0.	0.
12	0.1[illegible]6[illegible]E-01	0.	0.
13	0.4297E-02	0.	0.
14	0.1757E-02	0.	0.
15	0.84[illegible]E-03	0.	0.
16	0.6[illegible]25E-03	0.	0.
17	0.6[illegible]25E-03	0.	0.
18	0.7496E-03	0.	0.

CASE 210 1.0 ENRICHED U-L.W. ,E=6MEV-0.1MEV,RAT=1-1 ,T=20 ****** MACROSCOPIC FISSION NUMBER CROSS SECTION ******

ENERGY GROUP	REG. NO.--- 1	2	3	0	0
1	0.1075E-00	0.	0.		
2	0.8895E-01	0.	0.		
3	0.8553E-01	0.	0.		
4	0.8761E-01	0.	0		
5	0.8172E-01	0.	0.		
6	0.7984E-01	0.	0.		
7	0.7797E-01	0.	0.		
8	0.7631E-01	0.	0.		
9	0.7514E-01	0.	0.		
10	0.7292E-01	0.	0.		
11	0.6333E-01	0.	0.		
12	0.3547E-01	0.	0.		
13	0.1104E-01	0.	0.		
14	0.4490E-02	0.	0.		
15	0.2153E-02	0.	0.		
16	0.1606E-02	0.	0.		
17	0.1592E-02	0.	0.		
18	0.1871E-02	0.	0.		

REG. NO.--- 1 2 3

ENERGY GROUP

CASE 210 1.0 ENRICHED U-L.W. ,E=6MEV-0.1MEV,RAT=1-1 ,T=20 ****** MACROSCOPIC SLOWING DOWN TO OUT OF ENERGY RANGE ******

ENERGY GROUP	REG. NO.--- 1	2	3	0	0
1	0.1337E-01	0.2299E-02	0.1479E-02		
2	0.1798E-01	0.2675E-02	0.1848E-02		
3	0.1506E-01	0.2730E-02	0.2180E-02		
4	0.1623E-01	0.1863E-08	0.2601E-02		
5	0.1789E-01	0.2226E-04	0.3178E-02		
6	0.2001E-01	0.5388E-08	0.3921E-02		
7	0.2237E-01	0.1995E-05	0.5115E-02		
8	0.2801E-01	0.3725E-08	0.6498E-02		
9	0.3033E-01	-0.5388E-08	0.7792E-02		
10	0.2057E-03	0.3725E-08	0.9417E-02		
11	0.2139E-03	-0.5388E-08	0.1160E-01		
12	0.2244E-03	-0.1863E-08	0.1493E-01		
13	0.2307E-03	-0.1118E-07	0.1945E-01		
14	0.2579E-03	0.1618E-0[illegible]	0.2686E-01		
15	0.2976E-03	0.	0.4262E-01		
16	0.3450E-03	-0.1863E-08	0.7786E-01		
17	0.4092E-03	-0.1863E-06	0.1675E-00		
18	0.5701E-02	-0.1118E-07	0.4490E-00		

REG. NO.--- 1

ENERGY GROUP

CASE 210 1.0 ENRICHED U-L.W. ,E=6MEV-0.1MEV,RAT=1-1 ,T=20 ****** COLLISION PROBABILITY ******

PAGE 1

FLAT FLUX APP. ---P(REG.,REG ,ENERGY)---

E.G.	REG. NO. \ REG. NO.---	1	2	3	0	0
1	1	0.6801E 00	0.7788E-01	0.2460E-00	0.	0.
	2	0.6257E 00	0.9668E-01	0.2676E-00	0	0.
	3	0.6286E 00	0.7948E-01	0.2919E-00	0.	0.
2	1	0.6676E 00	0.7122E-01	0.2611E-00	0.	0.
	2	0.6211E 00	0.9405E-01	0.2849E-00	0	0.
	3	0.6125E 00	0.7662E-01	0.3109E-00	0.	0.
3	1	0.6700E 00	0.7241E-01	0.2576E-00	0.	0.
	2	0.6230E 00	0.9573E-01	0.2813E-00	0	0.
	3	0.6145E 00	0.7800E-01	0.3075E-00	0.	0.
4	1	0.6686E 00	0.7296E-01	0.2584E-00	0.	0.
	2	0.6204E 00	0.9666E-01	0.2830E-00	0	0.
	3	0.6116E 00	0.7876E-01	0.3097E-00	0	0.
5	1	0.6098E 00	0.7516E-01	0.3149E-00	0.	0.
	2	0.5564E 00	0.9983E-01	0.3438E-00	0	0.
	3	0.5437E 00	0.8042E-01	0.3759E-00	0.	0.
6	1	0.5943E 00	0.7411E-01	0.2411E-00	0.	0.
	2	0.5291E 00	0.9860E-01	0.3723E-00	0	0.
	3	0.5147E 00	0.7867E-01	0.4067E-00	0.	0.
7	1	0.6131E 00	0.8231E-01	0.3046E-00	0.	0.
	2	0.5601E 00	0.1077E-00	0.3322E-00	0.	0.
	3	0.5483E 00	0.8790E-01	0.3638E-00	0.	0.
8	1	0.6003E 00	0.8895E-01	0.2108E-00	0.	0.
	2	0.5470E 00	0.1151E-00	0.3379E-00	0.	0.
	3	0.5355E 00	0.9468E-01	0.3698E-00	0.	0.
9	1	0.5772E 00	0.9726E-01	0.3296E-00	0.	0.
	2	0.5243E 00	0.1194E-00	0.3562E-00	0.	0.
	3	0.5125E 00	0.9853E-01	0.3890E-00	0.	0.
10	1	0.5478E 00	0.8164E-01	0.3706E-00	0.	0.
	2	0.4922E-00	0.1066E-00	0.4012E-00	0.	0.
	3	0.4772E-00	0.8567E-01	0.4372E-00	0.	0.
11	1	0.5323E 00	0.7542E-01	0.3923E-00	0.	0.
	2	0.4725E-00	0.1006E-00	0.4269E-00	0.	0.
	3	0.4546E-00	0.7896E-01	0.4664E-00	0.	0.
12	1	0.5445E 00	0.7384E-01	0.3817E-00	0.	0.
	2	0.4828E-00	0.9953E-01	0.4176E-00	0.	0.
	3	0.4646E-00	0.7774E-01	0.4577E-00	0.	0.

E.G.	REG. NO. \ REG. NO.---	1	2	3
13	1	0.4806E-00	0.8363E-01	0.4358E-00
	2	0.4189E-00	0.1102E-00	0.4709E-00
	3	0.3991E-00	0.8610E-01	0.5148E 00
14	1	0.4681E-00	0.5883E-01	0.4731E-00
	2	0.4028E-00	0.8229E-01	0.5149E 00
	3	0.3786E-00	0.6019E-01	0.5612E 00
15	1	0.4585E-00	0.8460E-01	0.4569E-00
	2	0.3935E-00	0.1123E-00	0.4942E-00
	3	0.3716E-00	0.8641E-01	0.5420E 00
16	1	0.4479E-00	0.6423E-01	0.4879E-00
	2	0.3685E-00	0.9237E-01	0.5391E 00
	3	0.3376E-00	0.6502E-01	0.5973E 00
17	1	0.4427E-00	0.5423E-01	0.5030E 00
	2	0.3619E-00	0.8052E-01	0.5575E 00
	3	0.3289E-00	0.5462E-01	0.6165E 00
18	1	0.4479E-00	0.6400E-01	0.4881E-00
	2	0.3478E-00	0.9784E-01	0.5544E 00
	3	0.3069E-00	0.6417E-01	0.6289E 00
19	1	0.	0.	0.
	2	0.	0.	0.
	3	0.	0.	0.
20	1	0.	0.	0.
	2	0.	0.	0.
	3	0.	0.	0.
21	1	0.	0.	0.
	2	0.	0.	0.
	3	0.	0.	0.
22	1	0.	0.	0.
	2	0.	0.	0.
	3	0.	0.	0.
23	1	0.	0.	0.
	2	0.	0.	0.
	3	0.	0.	0.
24	1	0.	0.	0.
	2	0.	0.	0.
	3	0.	0.	0.

CASE 210 1.0 ENRICHED U-L.W. ,E=6MEV-0.1MEV,PAT=1-1 ,T=20 ****** COLLISION PROBABILITY ******

BY THERMAL N. S. 1+BL*R**2 --PS(R,R,E)-- BL= 0.

REG. NO.--- 1 2 0 0 1 REG. NO.--- 2

E.G.	REG. NO.					
1	1	0.6891E 00	0.7738E-01	0.2460E-00	0.	0.
	2	0.6757E 00	0.9668E-01	0.2676E-00	0.	0.
	3	0.6286E 00	0.7948E-01	0.2919E-00	0.	0.
2	1	0.6676E 00	0.7122E-01	0.2611E-00	0	0.
	2	0.6211E 00	0.9407E-01	0.2849E-00	0.	0.
	3	0.6125E 00	0.7662E-01	0.3109E-00	0	0.
3	1	0.6700E 00	0.7241E-01	0.2576E-00	0	0.
	2	0.6220E 00	0.9575E-01	0.2815E-00	0.	0.
	3	0.6145E 00	0.7800E-01	0.3075E-00	0.	0.
4	1	0.6685E 00	0.7296E-01	0.2584E-00	0	0.
	2	0.6204E 00	0.9666E-01	0.2810E-00	0.	0.
	3	0.6116E 00	0.7876E-01	0.3097E-00	0	0
5	1	0.6098E 00	0.7576E-01	0.3147E-00	0.	0.
	2	0.5364E 00	0.9985E-01	0.3458E-00	0.	0.
	3	0.5437E 00	0.8042E-01	0.3759E-00	0.	0.
6	1	0.5848E 00	0.7413E-01	0.3411E-00	0.	0.
	2	0.5291E 00	0.9860E-01	0.3727E-00	0.	0.
	3	0.5147E 00	0.7867E-01	0.4067E-00	0	0.
7	1	0.6[illegible]E 00	0.8231E-01	0.3046E-00	0.	0.
	2	0.5603E 00	0.1077E-00	0.3322E-00	0.	0.
	3	0.548[illegible]E 00	0.8790E-01	0.3638E-00	0.	0.
8	1	0.6003E 00	0.8895E-01	0.3108E-00	0.	0.
	2	0.5470E 00	0.1151E-00	0.3379E-00	0.	0.
	3	0.5[illegible]E 00	0.9468E-01	0.3698E-00	0	0.
9	1	0.5772E 00	0.9126E-01	0.3296E-00	0.	0.
	2	0.5243E 00	0.1194E-00	0.3562E-00	0.	0.
	3	0.5125E 00	0.9853E-01	0.3890E-00	0.	0.
10	1	0.5478E 00	0.8164E-01	0.3706E-00	0.	0.
	2	0.4922E-00	0.1066E-00	0.4017E-00	0.	0.
	3	0.4772E-00	0.8567E-01	0.4372E-00	0.	0.
11	1	0.5727E 00	0.7542E-01	0.3929E-00	0.	0.
	2	0.4725E-00	0.1006E-00	0.4269E-00	0.	0.
	3	0.4546E-00	0.7896E-01	0.4664E-00	0.	0.
12	1	0.5445E 00	0.7784E-01	0.3917E-00	0.	0.
	2	0.4828E-00	0.9953E-01	0.4176E-00	0.	0.
	3	0.4646E-00	0.7774E-01	0.4577E-00	0	0.

E.G.	REG NO			
13	1	0.4806E-00	0.8363E-01	0.4358E-00
	2	0.4189E-00	0.1102E-00	0.4709E-00
	3	0.3991E-00	0.8610E-01	0.5148E 00
14	1	0.4681E-00	0.588[illegible]E-01	0.47[illegible]E-00
	2	0.4028E-00	0.9229E-01	0.5149E 00
	3	0.3786E-00	0.6019E-01	0.5612E 00
15	1	0.4585E-00	0.8460E-01	0.4569E-00
	2	0.3925E-00	0.1125E-00	0.4942E-00
	3	0.3716E-00	0.8641E-01	0.5420E 00
16	1	0.4479E-00	0.6421E-01	0.4879E-00
	2	0.3685E-00	0.9237E-01	0.5391E 00
	3	0.3376E-00	0.6502E-01	0.5973E 00
17	1	0.4427E-00	0.5423E-01	0.5020E 00
	2	0.3619E-00	0.8052E-01	0.5575E 00
	3	0.3289E-00	0.5462E-01	0.6165E 00
18	1	0.4479E-00	0.6400E-01	0.4881E-00
	2	0.3478E-00	0.9784E-01	0.5544E 00
	3	0.3069E-00	0.6417E-01	0.6289E 00
19	1	0.	0.	0.
	2	0.	0.	0.
	3	0	0.	0.
20	1	0	0.	0.
	2	0.	0.	0
	3	0.	0.	
21	1	0.	0.	0.
	2	0.	0	0.
	3	0.	0.	0.
22	1	0.	0.	0.
	2	0.	0	0.
	3	0.	0.	0.
23	1	0.	0.	0.
	2	0.	0.	0.
	3	0.	0	0.
24	1	0.	0.	0.
	2	0.	0.	0.
	3	0.	0.	0.

THOSE INPUT DATAS ARE USED FOR CALCULATION OF EPSILON IN BNL-645 BY RIEF NGRP= 50 NFE= 7 NFSS= 1 MNGIN= 50 JNGRP=1275

ELEMENT	ELE.I.D	ATOMIC.NO	MASS NO.	INELA-M	COMENT FOR DATA
U238	-0	92	238	18	U-238 BNL-645

THOSE INPUT DATAS ARE USED FOR CALCULATION OF EPSILON IN BNL-645 BY RIEF

BUNDARY ENERGY OF GROUP ---MEV---

6.000000	5.750000	5.250000	4.750000	4.250000	3.750000	3.250000	2.750000	2.400000	2.150000
1.900000	1.650000	1.400000	1.150000	0.900000	0.650000	0.400000	0.250000	0.100000	0.
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

ENERGY INTERVAL OF ENERGY GROUP

0.250000	0.500000	0.500000	0.500000	0.500000	0.500000	0.500000	0.350000	0.250000	0.250000
0.250000	0.250000	0.250000	0.250000	0.250000	0.250000	0.150000	0.150000	0.	0.
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.250000

FISSION SPECTRUM --- NORMALIZED BY INT S*E =1

U238 0000000000000000000000000000000000000000000000000000000000000000

0.020137	0.026462	0.037640	0.053078	0.074082	0.102128	0.138644	0.176846	0.207885	0.235976
0.265326	0.294766	0.322334	0.344789	0.356627	0.347745	0.313571	0.252475	0.138598	0.
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

THOSE INPUT DATAS ARE USED FOR CALCULATION OF EPSILON IN BNL-645 BY RIEF

U238 NGRP= 18

	ELA-SCATTER	CAPTURE	INELA-SCATTER	FISSION	TOTAL	NUMBER/FISSION	ANISOTROPY
1	0.38000E 01	0.	0.27400E 01	0.70000E 00	0.	0.32065E 01	0.
2	0.41400E 01	0.	0.27100E 01	0.59000E 00	0.	0.31540E 01	0.
3	0.42700E 01	0.	0.27000E 01	0.58000E 00	0.	0.30840E 01	0.
4	0.44700E 01	0.80000E-02	0.26800E 01	0.58000E 00	0.	0.30140E 01	0.
5	0.43700E 01	0.12500E-01	0.26800E 01	0.58000E 00	0.	0.29440E 01	0.
6	0.43300E 01	0.16000E-01	0.27000E 01	0.58000E 00	0.	0.28740E 01	0.
7	0.42000E 01	0.21000E-01	0.28000E 01	0.58000E 00	0.	0.28040E 01	0.
8	0.38000E 01	0.27000E-01	0.29900E 01	0.58000E 00	0.	0.27445E 01	0.
9	0.34000E 01	0.37000E-01	0.30000E 01	0.58000E 00	0.	0.27025E 01	0.
10	0.35000E 01	0.41000E-01	0.30000E 01	0.57000E 00	0.	0.26675E 01	0.
11	0.40000E 01	0.54000E-01	0.29980E 01	0.50000E 00	0.	0.26325E 01	0.
12	0.50600E 01	0.74000E-01	0.25400E 01	0.27800E-00	0.	0.25975E 01	0.
13	0.48000E 01	0.10500E-00	0.22000E 01	0.79000E-01	0.	0.25625E 01	0.
14	0.53500E 01	0.13800E-00	0.21080E 01	0.26000E-01	0.	0.25275E 01	0.
15	0.52000E 01	0.14500E-00	0.20000E 01	0.60000E-02	0.	0.24925E 01	0.
16	0.70300E 01	0.13000E-00	0.18700E 01	1.00000E-03	0.	0.24575E 01	0.
17	0.85300E 01	0.14000E-00	0.53000E 00	0.	0.	0.24295E 01	0.
18	0.10810E 02	0.21000E-00	0.40000E-00	0.	0.	0.24085E 01	0.

THOSE INPUT DATAS ARE USED FOR CALCULATION OF EPSILON IN BNL-645 BY RIEF

SCATTERING MATRIX FOR U238

GROUP NO.										
1	0.38001E 01	0.	0.24072E-07	0.64756E-03	0.17243E-02	0.45334E-02	0.11729E-01	0.29709E-01	0.44103E-01	0.52959E-01
	0.81072E-01	0.12220E-00	0.18048E-00	0.25930E-00	0.35776E-00	0.46409E-00	0.53724E 00	0.30845E-00		
2	0.41402E 01	0.	0.47682E-03	0.13167E-02	0.5899E-02	0.76317E-02	0.25300E-01	0.38681E-01	0.47442E-01	0.73959E-01
	0.11752E-00	0.17074E-00	0.24780E-00	0.35115E-00	0.46557E-00	0.54656E 00	0.31822E-00	0.		
3	0.42703E 01	0.	0.88556E-03	0.25503E-02	0.72270E-02	0.20050E-01	0.32042E-01	0.40569E-01	0.64998E-01	0.10253E-00
	0.15849E-00	0.23829E-00	0.34423E-00	0.46697E-00	0.56544E 00	0.33650E-00	0.	0.		
4	0.44706E 01	0.	0.16998E-02	0.51331E-02	0.15175E-01	0.25528E-01	0.33536E-01	0.55466E-01	0.90321E-01	0.14412E-00
	0.22367E-00	0.[illegible]3257E-00	0.46699E-00	0.58351E 00	0.35604E-00	0.	0.	0.		
5	0.43711E 01	0.	0.34[illegible]42E-02	0.10944E-01	0.13558E-01	0.26835E-01	0.46084E-01	0.77916E-01	0.12908E-00	0.20800E-00
	0.32200E-00	0.46802E-00	0.60691E 00	0.38139E-00	0	0.	0.	0.		
6	0.43321E 01	0.	0.73718E-02	0.14167E-01	0.20487E-01	0.36809E-01	0.65122E-01	0.11289E-00	0.19034E-00	0.30829E-00
	0.46878E-00	0.67577E 00	0.41[illegible]95E-00	0.	0	0.	0.	0.		
7	0.42046E 01	0.	0.96503E-02	0.14887E-01	0.28301E-01	0.52968E-01	0.97134E-01	0.17324E-00	0.29680E-00	0.47728E-00
	0.68433E 00	0.46553E-00	0.	0.	0	0.	0.	0.		
8	0.38065E 01	0.	0.10743E-01	0.21690E-01	0.43115E-01	0.83970E-01	0.15904E-00	0.28934E-00	0.49402E-00	0.75176E 00
	0.53587E 00	0.	0.	0.	0.	0.	0.	0.		
9	0.34076E 01	0.	0.16180E-01	0.33890E-01	0.69546E-01	0.13879E-00	0.26602E-00	0.47847E-00	0.76673E 00	0.56909E 00
	0.	0.	0.	0.	0	0.	0.	0.		
10	0.41322E 01	0.42853E-00	0.37065E-01	0.12184E-00	0.80814E 00	0.50994E 00	0.46231E-00	0.	0.	0.
	0.	0.	0.	0.	0.	0.	0.	0.		
11	0.46318E 01	0.42824E-00	0.37040E-01	0.12176E-00	0.80760E 00	0.50960E 00	0.46200E-00	0.	0.	0.
	0.	0.	0.	0.	0	0.	0.	0.		
12	0.57370E 01	0.43296E-00	0.52400E-01	0.29440E-00	0.78080E 00	0.23040E-00	0.72000E-01	0.	0.	0.
	0.	0.	0.	0.	0.	0.	0.	0.		
13	0.58303E 01	0.61909E 00	0.60949E-01	0.48968E-00	0.	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.	0.	0.		
14	0.63372E 01	0.59320E 00	0.58400E-01	0.30600E-00	0.16320E-00	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.	0.	0.		
15	0.66069E 01	0.56530E 00	0.27807E-01	0.	0.	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.	0.	0.		
16	0.83454E 01	0.47656E-00	0.78000E-01	0.	0.	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.	0.	0.		
17	0.89045E 01	0.15547E-00	0.	0.	0.	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.	0.	0.		
18	0.11093E 02	0.11733E-00	0.	0.	0.	0.	0.	0.	0.	0.
	0.	0.	0.	0.	0.	0.	0.	0.		