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An IBM-7044 FORTRAN IV Code for Calculation of the Neutron Spectrum for Square and Hexagonal Lattices

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Code GRAFA
An IBM-7044 FORTRAN IV Code for calculation of the Neutror Spectrum for Square and Hexagonal Lattices

Abstract

This report is a description of the GRAFA code written in Fortran IV for the IBM 7044. The code computes the scalar thermal neutron spectrum as a function of spatial regions in a square and hexagonal lattice by using the first flight collision probabilities for these geometries.* The scattering kernel data calculated by various models are input from the library tape.

Aug. 1965

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コード GRAFA
正方および六方格子系の等方中性子束スペクトルに対する IBM-7044 FORTRAN IV コード

要　　旨

この報告は IBM-7044 の FORTRAN IV で書かれた GRAFA コードのマニュアルである．GRAFA コードは正方および六方格子系の等方中性子束スペクトル分布を各空間領域の函数として積分型輸送方程式に基礎をおいた衝突確率法を用いて計算する．計算に用いられる熱中性子散乱核は各種模型を用いて計算したデーターの入ったライブラリーテープから読まれる．

1965 年 8 月

日本原子力研究所，東海研究所

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* It is recommended to refer the reference paper (1), when this manual is read.

Contents

目　　次

1. Intoroduction

The code GRAFA performs the calculation of the thermal neutron spectrum in a square and hexagonal lattice systcm by using the first flight collision probability method.

The system consists regions, namely fuel, moderator and coolant. The regions are divided to subregions and the first flight collision probabilities (F. F. C. P) for these subregions are calculated for each energy group. The simultaneous equation for the fluxes in the subregions are solved by over-relaxation iteration method.

The simultaneous equation for neutron spectra $\phi_j(E)$ is

$$\sum_i p_{ij}(E)\left[\int_0^{E_c}\Sigma_{si}(E'\to E)\phi_i(E')dE+S_i(E)\right]V_i=\Sigma_j(E)\phi_j(E)V_j \qquad (1)$$

where $P_{ij}(E)$ is the first flight collision probabilities from i-th region to j-th region at energy E, $\Sigma_{si}(E'\to E)$ is the isotropic scattering kernel in j-th region from energy E' to energy E, $\phi_i(E')$ is the isotropic neutron flux in the j-th region at energy E', $S_j(E)$ is the neutron source at energy E in the j-th region, which is slowed down from the energy region above the cutoff energy E_c, $\Sigma_j(E)$ is the total cross section of material in i-th region at energy E, and V_i is the volume of the i-th region. The integration with respect to energy is replaced by the summation over multigroup energies.

The code consists three parts (chain (1), chain (2), chain (3)). In chain (1), the macroscopic scattering kernel for the material in the subregions is obtained by multipling the atomic density in the region with the scattering kernel of the atoms, where the scattering kernel is read from library tape. The neutron sources for the thermal energy range are calculated by the slowing down model or are read from cards. The total cross section of material in the subregion is calculated and used for the calculation of the F. F. C. P. in chain (2). Most of the data such as the radius of the subregion, the energies in the multigroup calculation, the masses of the atoms and the number of subregions, etc., are read in this chain.

In chain (2), the F. F. C. P., P_{ij}, for these subregions in the equation (1) is calculated.

In chain (3), the simultaneous equation (1) for the fluxes in the energy groups and subregions, containing the macroscopic scattering kernel and the F. F. C. P., is solved by the iteration method with an over-relaxation, and calculation of the activity is carried out. Although this code is limited to the case of 30 energy groups and 10 subregions, the large number of core memory locations available will permit the enlargement of these dimensions.

2. Theory

See reference (1), JAERI Roport 1072, "The First Flight Collision Probability in the Square and Hexagonal Lattice Systems" and reference (2), The Thermal Neutron Spectrum in a Heterogeneous Reactor.

3. Description of the code

3.1 General

This code is consists of three parts, chain (1), chain (2) and chain (3).

In chain (1), the macroscopic neutron scattering kernel for the material (mixture of

isotopes) in the subregions is obtained by multiplying the isotope density in the region with the neutron scattering kernel by the isotope, the scattering kernel is read from the library tape which is described later. The neutron sources for the thermal energy range are calculated by the slowing-down model or the source data is read from cards. The total cross sections of the material in the subregions are calculated and are used for the calculation of the F. F. C. P. in chain (2). Most of data, such as the radius of the subregions, the energies in the multigroup calculation the masses of the isotopes and the number of subregions, etc., are read in this chain.

The neutron scattering kernel is read from the library tape, which is installed in tape No. 10. Both up and down scattering kernels are stored in the library tape with the title (TITLE).

To read the scattering data, the program reads RNAME from card input and compares it with the TITLE (for example KERNEL H 2O 255), and when both are coincident, the data are read, When it reads the data, the order of data must be the same as the order of data stored in the tape. The tape shown in table contains both the P_0 scattering data and the P_1 scattering data because this libiary can be used in the code FIRST 2. FIRST 2 calculates the anisotropic flux together with the isotropic flux in a cylindrical cell by using the P_0 and P_1 scattering kernels.

Since this code is restricted to the case of isotropic scattering, the other library for the transport scattering kernel defined by HONECK as follows.

$$\Sigma_{tr}(E'\to E)=\Sigma_{s0}(E'\to E)-\Sigma_{s1}(E)\,\delta(E'-E)$$

is provided for calculating the neutron spectra in anisotropic scattering media.

Contents in library tape for scattering kernel

KERNEL OF	H	ON TAPE, TEMPERATURE=	255
KERNEL OF	H	ON TAPE, TEMPERATURE=	451
KERNEL OF	H2O	ON TAPE, TEMPERATURE=	255
KERNEL OF	H2O	ON TAPE, TEMPERATURE=	451
KERNEL OF	D	ON TAPE, TEMPERATURE=	255
KERNEL OF	D	ON TAPE, TEMPERATURE=	451
KERNEL OF	D 3 6	ON TAPE, TEMPERATURE=	255
KERNEL OF	D2O	ON TAPE, TEMPERATURE=	255
KERNEL OF	D2O	ON TAPE, TEMPERATURE=	451
KERNEL OF	C	ON TAPE, TEMPERATURE=	255
KERNEL OF	C	ON TAPE, TEMPERATURE=	451
KERNEL OF	O	ON TAPE, TEMPERATURE=	255
KERNEL OF	O	ON TAPE, TEMPERATURE=	451
KERNEL OF	H	ON TAPE, TEMPERATURE=	255
KERNEL OF	H	ON TAPE, TEMPERATURE=	451
KERNEL OF	H2O	ON TAPE, TEMPERATURE=	255
KERNEL OF	H2O	ON TAPE, TEMPERATURE=	451
KERNEL OF	D	ON TAPE, TEMPERATURE=	255
KERNEL OF	D	ON TAPE, TEMPERATURE=	451
KERNEL OF	D 3 6	ON TAPE, TEMPERATURE=	255
KERNEL OF	D2O	ON TAPE, TEMPERATURE=	255
KERNEL OF	D2O	ON TAPE, TEMPERATURE=	451
KERNEL OF	C	ON TAPE, TEMPERATURE=	255
KERNEL OF	C	ON TAPE, TEMPERATURE=	451
KERNEL OF	O	ON TAPE, TEMPERATURE=	255
KERNEL OF	O	ON TAPE, TEMPERATURE=	451

In chain 2, the first flight collision probability for the square or hexagonal lattice is calculated, the expressions $P_{i,j}(i \neq j)$ are

$$P_{ij}=\frac{4}{\pi^2}\frac{1}{\Sigma_i(a_i^2-a_{i-1}^2)}\int_0^{\pi/2}d\alpha\int_0^{a_i}dr[F_{0s}(r,\alpha)+F_{1s}(r,\alpha)] \tag{2}$$

for the square lattice, and

$$P_{ij}=\frac{6}{\pi^2}\frac{1}{\Sigma_i(a_i^2-a_{i-1}^2)}\int_0^{\pi/3}d\alpha\int_0^{a_i}dr_i[F_{0h}(r,\alpha)+F_{1h}(r,\alpha)] \tag{3}$$

for the hexagonal lattice, where the suffixes s and h are the values for the square and hexagonal lattices, respectively. And a_i is the outer radius of i-th region, Σ_i is the macroscopic scattering in i-th region, and $F_0(r,\alpha)$ and $F_0(r,\alpha)$ are defined as follows:

$$\begin{aligned}F_0(r,\alpha)=&\sum_{l=1}^{2}\{K_{i3}(U_{ij}^1+b_{li})-K_{i3}(U_{ij}^1+b_{li}+u_i^1)\\&-K_{i3}(U_{ij}^1+b_{li}+u_i^1)+K_{i3}(U_{ij}^1+b_{li}+u_i^1)\}\end{aligned} \tag{4}$$

$$\begin{aligned}F_1(r,\alpha)=&\sum_{q=2}^{\infty}\sum_{l=1}^{2}\sum_{m=1}^{2}\{K_{i3}(U_{ij}^q+b_{li}+c_{mi}^q)-K_{i3}(U_{ij}^q+b_{li}+c_{mi}^q+u_i^1)\\&-K_{i3}(U_{ij}^q+b_{li}+c_{mi}^q+v_j^q)+K_{i3}(U_{ij}^q+b_{li}+c_{mj}^q+u_i^1+u_j^q)\}\end{aligned} \tag{5}$$

where $K_{in}(x)$ is the Bickerley function defined by $K_{in}(x)=\int_0^{\pi/2}e^{-x/\sin\theta}(\sin\theta)^{n+1}d\theta$ and

$$\left.\begin{aligned}&u_k^q=\Sigma_k y_x^q a_0\\&u_k^q=\Sigma_0 y_0^q a_0\\&b_{1i}=0 \qquad b_{2i}=2\sum_{k=1}^{i-1}u_k^1+u_i^1\\&c_{1j}^q=0 \qquad c_{2j}^q=2\sum_{k=1}^{j-1}u_k^q+u_j^q\\&U_{ij}^p=U_{ij}^{(q-1)}+d_j^q \qquad q\geqslant 2\\&U_{ij}^1=\sum_{k=i+1}^{j-1}u_k^1\\&d_i^q=\sum_{k=j}^{N}u_k^{(q-1)}+\sum_{k=j+1}^{N}u_k^q+u_0^{(q-1)}\end{aligned}\right\} \tag{6}$$

a_0y_k and $a_0y_0/2$ are the half lengths of the neutron path through point (r,α) in the first cell in the k-th and outermost subregions.

The calculation of the F. F. C. P. for the lattice systems is easily carried out, if the lengths of the neutron path in the subregions are obtained. Reference 1 shows how the lengths in the subregions are calculated for the square and hexagonal lattice.

From neutron transport theory, it is easily shown that the neutron conservation law is expressed by

$$\sum_j P_{ij}=1 \tag{7}$$

and the reciprocity relation exists between P_{ij} and P_{ji} as follows.

$$V_i\Sigma_i P_{ij}=V_j\Sigma_j P_{ji} \tag{8}$$

Thus, the numerical calculation for P_{ij} is carried out for the case of $i<j$, and the values of P_{ji} are usually obtained from the reciprocity relation of (8). Furthermore, the F. F. C. P. from the i-th region to the i-th region itself can be obtained from the conservation of (7). But if the values of P_{ii} is obtained in this way, the numerical error due to replacing the integral of equations (2) and (3) by the summation is accumulated in the value of P_{ii}. Thus, the calculation of P_{ii} is carried out by the following formula. The F. F. C. P. P_{ii}'s also have the same expression as the P_{ij} in equation (2) and (3), but the expression $F_0(r,\alpha)$ is replaced by the following $F_0'(r,\alpha)$

$$\begin{aligned}F_0'(r,\alpha)=&2[u_i^1-K_{i3}(0)+K_{i3}(u_i^1)]\\&+[K_{i3}(b_{2l}-u_i^1)-2K_{i3}(b_{2l})+K_{i3}(b_{2l}+u_i^1)]\end{aligned} \tag{9}$$

and the suffix j in expression $F_1(r,\alpha)$ is replaced by the suffix i.

In order that the P_{ij} satisfy both conditions of (7) and (8), the P_{ij} are normalized

in the following way. At first, P_{1j} are calculated for every j, and are normalized so as to satisfy equation (7). Then the values of P_{j1} are obtained by the reciprocity relation (8). Next, the P_{2j} are calculated for $j \geq 2$, and are normalized together with P_{21}. These procedures are continued until the end.

To save calculation time, the program is made to calculate the F. F. C. P. in some energy groups from interporation of the values of F. F. C. P. in the other energy groups calculated by the above formulation. This method is not a bad approximation in the case when the energy change of the total cross section in energy is smooth.

In chain 3, the simultaneous equation for the flux containing the scattering kernel, the absorption cross section and the F. F. C. P. is solved by the iteration method with an over relaxation, which is an improvement of simple normalized method. Although several acceralation methods are recommended in the iteration method, the acceralation paramete is not usually determined uniquely. And if the parameter is not chosen properly, the iterated fluxes might oscillate about the exact fluxes. Theirfore, care must be taken in choosing the parameter.

In the code, the Maxwellian distribution with the same temperatures as the physical temperatures of the subregion material is taken as the first approximation of the neutron spectrum. Convergence of the iterated fluxes is discerned as follows: the maximum absolute difference between the latest flux and the previous iteration fiux must be smaller than the maximum neutron flux multiplied by some convergence parameter, EPSIRO.

The criterion for the convergence of the iterated fluxes is determined by comparing with the maximum value of the fluxes in the outermost subregion. so if the absorption in the fuel region is very large and thus the flux is small, a small value must be taken for the EPSIRO. If an accurate value of the epithermal flux, which is very small, is required, a small value must also be taken for the EPSIRO.

Other criteria for the convergence of fluxes may also be taken. For example, the relative error value for the flux in each subregion and in each energy group should be small. But the time required for calculation using these criterion is larger than the case of the above criterion. The maximum value of difference of the iterated flux is edited later to indicate the error.

The over relaxation iterated method is carried out in the following way,

$$\phi^{(n+1)}(E) = \phi^{(n-1)}(E) + \mathrm{OVERR} \times (C^{(n)}\phi^{(n)}(E) - \phi^{(n-1)}(E)) \tag{10}$$

where $\phi^{(n)}(E)$ is the flux obtained after the n-th iteration, $\phi'^{(n+1)}(E)$ is the flux used in the $(n+1)$ iteration, $C^{(n)}$ is the normalization factor defined as

$$C^{(n)} = \frac{\sum_i \int_0^{E_c} S_i(E)\,dE}{\sum_i \int_0^{E_c} \phi_i^{(n)}(E)\,\Sigma_{ai}(E)\,dE}. \tag{11}$$

When OVERR is equal to 1,

$$\phi'^{(n+1)}(E) = C^{(n)}\phi^{(n)}(E) \tag{12}$$

which, therefore, is a usual normalized iteration method. EPSIRO defined in the above is used for the criterion for convergence, as follows:

$$|C^{(n)}\phi_i^{(n)}(E_l) - \phi_i^{(n-1)}(E_l)| < \mathrm{EPSIRO} \times \phi_{IR}^{(n)}(E_{\max F}) \tag{13}$$

for any i-th subregion and l-th energy group.

Furthermere, the calculation of activity can be carried out by using the calculated neutron spectra to analize activation experiments in this chain (3). The cross sections of activation or fission for isotopes (numbers<10) are read from cards. This can be used for calculating the reactor constants, for example the diffusion cross section.

3.2 Input date

Input data used in this code are explained in the order of their appearance.

(I) Input in Chain 1.

1 IMN, IMNN (FORMAT (10 I 3))

IMN is the number of materials, whose scattering kernels are read from the library tape. Where the same material with different temperature is counted as different material. The maximum of IMN is limited to 6. IMNN=IMN+1.

2 IR, IE, IM, IRM (FORMAT (10 I 3))

IR is the number of subregions, which is limited to 10. IE is the number of energy groups, which is limited to 30. IM is the number of isotopes composing the materials. The same isotope of a different temperature is counted as a different isotope. The IM is limited to 11. IRM is the number of materials, for example, fuel, aluminium and H_2O. The number is limited to 6.

3 (NDASH (I), I=1, IR) FORMAT (10 I 3)

The NDASH is used in numbering the material as NDASH (I) etc., which is put in the I-th subregions. An example is illustrated in the following.

	Fuel	Fuel	Fuel	Fuel	Al	H_2O	H_2O	H_2O	H_2O	H_2O
I	1	2	3	4	5	6	7	8	9	10
NDASH (I)	1	1	1	1	2	3	3	3	3	3

4 (R (I), I=1, IR) FORMAT (1 P 5 E 11.4)

R (I) is the outer radius of the i-th subregion in cm.

5 (E (I), I=1, IE) FORMAT (1 P 5 E 11.4)

E (I) is the middle energy of the i-th energy group in eV.

6 (DISA (I), I=1, IR) FORMAT (1 P 5 E 11.4)

DISA is used for taking into account the spatial distribution of slowing down sources from the epithermal energy range above the cut-off energy Ec. The flux distribution in the epithermal energy region is substituted into the DISA (I). If the flux distribution is flat, all the values of DISA (I) are substituted by 1.0.

7 ((QNU (I, J), I=1, IM), J=1, IRM) FORMAT (1 P 5 E 11.4)

QNU (I, J) is used in making the macroscopic scattering kernel of the material, it is the product of the density (N) of the isotopes composing the material and the free scattering cross section (σ_{free}) of the composite isotopes. That is, the value of Σ_{free} ($=\sigma_{free}N$) is input.

8 (TEMP (I), I=1, IR) FORMAT (1 P 5 E 11.4)

In chain 3, the iteration method is used for solving the simultaneous equation. The Maxwellian distribution with the physical temperatures of the materials in the each subregion is used as the first approximation of the spectrum in each subregion TEMP is a physical temperature in eV.

9 (QMASS (I), I=1, IR) FORMAT (1 P 5 E 11.4)

QMASS is the ratio of isotope mass to neutron mass and is used in the calculation of the thermal neutron source spectrum. If it equals zero, the thermal neutron source spectrum for this isotope is read from the card data in subroutine SAUCE. These cards are read after Input No. 13.

10 IABB FORMAT (10 I 3)

This is the option parameter concerning calculation of the absorption cross section. When IABB=0, the $1/v$ functions are calculated in the program and are used as the $1/v$ absorption cross section. When IABB$\neq$0 the absorption cross sections are read from the card data in subroutine ABSORP.

11 (S (I, J), J=1, NUME) in subroutine ABSORP FORMAT (1 P 5 E 11.4)

The S (I, J) is the absorption cross section divided by the free scattering cross section. NUME is equivalent to IE. When there is an isotope with a non $1/v$ absorption corss section, the other $1/v$ absorption cross sections are read in this subroutine. Thus, the number of the data (I) is IM.

12 (ABB (I), I=1, IM) FORMAT (1 P 5 E 11.4)

ABB (I) is the ratio of the absorption cross section at 0.0255 eV, σ_a (0.025 eV), to the free scattering cross section σ_i for the i-th isotope, and is used for calculation of the $1/v$ absorption cross section.

13 (FACTOR, I=1, IMN) FORMAT (1 P 5 11.4)

FACTOR is used in the calculation of scattering kernels, because the values of $((1+1/M)^2/4\sigma_i)\cdot\sigma_s(E'\to E)$ are stored in the library tape as scattering kernel data. The values of FACTOR corresponds to the M. The values for H and D in H_2O and D_2O of Brown and St-Johns model should be used as 1.9 and 3.6 respectively.

14 (RNAME (I), I=1, IM) FORMAT (12 A 6)

The scattering kernel data in this code are stored in a library tape with their own names. The RNAME is used for calling the data from the library tape, which is put in tape unit No. 10. The RNAME's are shown in TABLE I.

15 (SOCMA (I), I=1, IE) in subroutine SAUCE FORMAT (1 P 5 E 11.4)

The SOCMA (I) is the thermal neutron source spectrum, which is calculated as the epithermal neutron flux equals to unity at cut off energy. The cut off energy is the sum of DELTAE (I).

By comparing RNAME with the title in the library tape, the scattening kernels SIGMA (J, I) are read in the machine.

(II) Input in chain 2.

1 ISSTEP, IQMAX, IKIND, LREAD, ILSI, ISTEPS FORMAT (10 I3)

ISSTEP is the number of divisions of α in the integration with respect to α in eqs (2) and (3), $\leqslant 10$.

IQMAX is the maximun number of cells (q) which are expected to contribute to F. F. C. P.. IKIND is the parameter which chooses the lattice geometry, that is, IKIND=0 for square lattice, $\neq 0$ for hexagonal lattice.

LREAD is the option parameter which is used for editing the F. F. C. P. by using the formalism. If LREAD is unity, the values of F. F. C. P. in energy group of number 1, 6, 11, 16, 21, 26, and 30 are edited, and if LREAD is two, the values of energy groups 6, 16 and 26 are edited in the case shown in the loter.

ILSI is the interval of energy groups, for which the F. F. C. P.'s are calculated, and is restricted to the number which can divide the number of energy groups IE. For example, in the case of IE=30, the numbers of 1, 2, 3, 5, 6, 10, 15, 30 are allowed, and if numbers 5 is chosen in this case, the F. F. C. P. in energy groups number 1, 6, 11, 16, 21, 26, and 30 are calculated by the formalism, and the F. F. C. P. in the other groups are obtained as linear interporation of the values obtained in above.

ISTEPS is the number concerning to integration with respect to r in eqs (2) and (3). The details are explained in the appendix.

2 (MDASH (I), I=1, N) FORMAT (10 I 3)

MDASH are the numbers concerning production of the tables of $y_0(r_q, \alpha)$, $r_{q+1}(r_q, \alpha)$. The details are explained in the appendix.

(III) Input in chain 3.

1 ITMAX, MAXF, OVERR, EPSIRO FORMAT (2I 3, 1P 10E 11.4)
ITMAX is the maximum number of iterations which is allowed in the calculation. MAXF is the number of the energy group where the neutron spectrum ϕ(MAXF, IR) in the outermost subregion is expected to have a maximum value. OVERR is a parameter used in the over relaxation. EPSIRO is a value used for the criterion for convergence.
2 L in subroutine ACTIVA, FORMAT (I2)
L is the number of isotopes whose activation or fission cross sections are read.
3 (SIGMAA (I), I=1, IE) in subroutine ACTIVA, FORMAT (1P 5E 11.4)
SIGMAA is the activation or fission cross section.

3.3 Out put data

Formats for output are written to clarify the input data and results in table form.
1 The parameter used in the calculation and the dimensions of the lattice are edited, the values of IR, IE, IRM, IM, IMN, R(I), NDASH (I), TEMP (I), DISA (I) are written out on page 1.
2 The macroscopic free cross section QNU and the values of QMASS and FACTOR, which are input data, are edited on page 2.
3 The absorption cross sections of composite material are edited together with the middle energies (E) and energy widths (DELTA) of energy group on page 4.
4 The total cross sections are edited in the same way as the absorption cross section on page 4.
5 The thermal neutron source spectrum obtained by slowing down due to the composite material is edited on page 5. Where the source is calculated from epithermal flux normalized at cut off energy using the values of QMASS.
6 The input data in chain 2 IKIND, ISSTEP, IQMAX, MDASH (I) are edited on page 6.
7 The calculated values of F. F. C. P. P(I, J) (I→J) in chain 2 are edited together with the total cross sections and the number of energy group (ILS) on subsequent pages.
8 The input data in chain 3 ITMAX, MAXF, OVERR, EPSIRO are edited. And the neutron flux spectra in each subregion are edited as a function of energy together with RESIDU ,which expresses the accuracy of the calculated spectrum by the iteration method, and the number of iterations is edited.
9 In the next page, the neutron flux spectra calculated by one more iteration are edited together with the values of RESIDU and the number of iterations to estimate the convergence of the iterated results.
10 In the last page, the activities calculated by the neutron spectra in subregions are put out together with the activation cross sections.

3.4 Subroutines

1 SUBROUTINE SAUCE (E, DELTA, QMAS, SOCMA, IE) in chain 1 prepares the thermal neutron source spectrum due to scattering from the composite isotope. When QMAS equals zero, the spectrum is input from cards, and when QMAS does not equal zero, the spectrum is calculated using this mass. The format of the data is (1P 5E 11.4).
2 SUBROUTINE ABSORP (E, SA, S, NUMS, NUME, J) in chain 1 prepares the absorption cross section for the composite isotopes. When J is less than zero, the data of absorption cross section are input from cards, and in the other cases the absorption cross sections are calculated as $1/v$ absorption. The format of the

data is (1 P 5 E 11.4)

3 SUBROUTINE HOLLER (A, B) in chain 1 is used in reading the scattering kernel data from tape 10.

4 SUBROUTINE CCC (TEMP 1, TEMP 2, TEMP 3, TEMP 4, QK 3, QK 3 D, X) in chain 2 calculates the Bickerley functions of order 3 for arguments TEMP 1~ TEMP 4, K_{i3}(TEMP 1)~K_{i3}(TEMP 4) and do the calculation of

$$x=x+K_{i3}(\text{TEMP 1})-K_{i3}(\text{TEMP 2})-K_{i3}(\text{TEMP 3})+K_{i3}(\text{TEMP 4}).$$

5 SUBROURINE TAB (A, B, QK3, QK3D) in chain 2 is used to look up the table by linear interpolation, where the values of $A=K_{i3}(B)$ are obtained.

6 SUBROUTINE BIC (I, X) in chain 2 calculates the Bickerley function $K_{i3}(x)$.

7 SUBORUTINE ACTIVA (PHI, IE, IR, DELTA, L) in chain 3 is used for calculating the activation values or the reactor constant values averaged by the neutron spectra obtained.

4. Example

To make the above explanation clear, the input and output data for the case of uranium and light water hexagonal lattice are shown. The geometrical dimensions and the data of composition are given in the out put data (page 1, 2).

5. Appendix

Determination of r_q. As mentioned above, tables for $r^{q+1}(r^q, a)$ and $y^q(r^q, a)$ are made for reducing the calculation time.

The values of r^q, which are in i-th subregion, are determined in the following way. The angle $\beta_i=\arccos(a_{i-1}/a_i)$ is divided by MDASH (I), and $r_{q+\text{MDASH(I)}}$, $r_{q+\text{MDASH(I)}-1}$,..... and, r_q are calculated respectively by

$$r_{q+\text{MDASH(I)}}=a_i,$$

$$r_{q+\text{MDASH(I)}-1}=a_i\cos\left(\frac{\beta_i}{\text{MDASH(I)}}\right)$$

$$\vdots$$

$$r_{q+1}=a_i\cos\left(\frac{(\text{MDASH(I)}-1)\beta_i}{\text{MDASH(I)}}\right)$$

where $q=\sum_{I=1}^{i-1}\text{MDASH(I)}+1.$ (See figure).

The maximum number of q is limited to 30 in this code, so the values of MDASH (I) are restricted as follows,

$$\sum_{I=1}^{IR}\text{MDASH(I)}<29$$

If we use the number of r_q obtained as above in integration with respect to r in the expression for F. F. C. P., the calculation takes a long time. Thus, the r_q values in the integration of F. F. C. P. $P_{i,j}$ from the i-th region are chosen in the following way.

At first, the angle $\pi/2$ is divided by the number of ICSTEPS, and the values of r'_s are calculated by

$$r'_{s1}=0$$

$$r'_{s2}=a_i\sin\delta$$

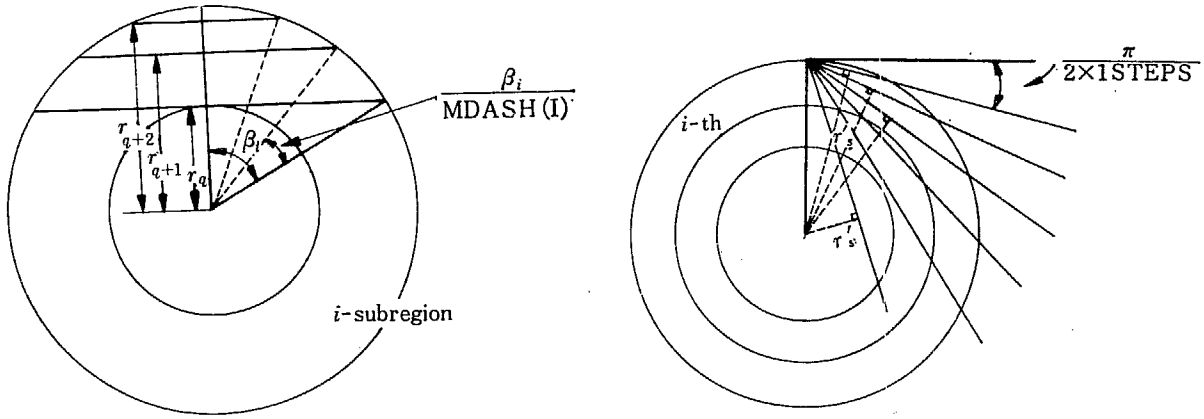
$$r'_{s3}=a_i\sin 2\delta$$

.....

where $\delta=\frac{\pi}{2\times \mathrm{ISTEPS}}$.

Next, the values of r_q used in the integration are chosen as the nearest values from r'_{sn}. Thus, the number of r_q used in the integration is equal to ISTEPS and the calculation time becomes short.

When the neutron path goes to the next cell, the r^{q+1} value obtained is not usually the same as the value determined for r_q in table. So, the value of r_q in the next cell is chosen as the nearest r_q value in the table from the r^{q+1}.



Acknowledgement

The authors wish to acknowlege the valuable discussion they have had with Dr. Y. FUKAI. They also wish to express their appreciation to Mr. T. SUZUKI for giving crucial assistance in debugging the code.

6. Reference

1. HIROSHI TAKAHASHI, TAKASHI NAKAYAMA
"The First Flight Collision Probability in the Square and Hexagonal Lattice Systems" JAERI Report 1072 Nov. 1964
2. HIROSHI TAKAHASHI
"The Thermal Neutron Spectum in a Heterogeneous Reactor" *Nucl. Sci. Eng.*, 5, 338 (1959)
3. HIROSHI TAKAHASHI
"Resonance Escape probabilities in Circular Cylindrical Cell Systems" *J. Nucl. Energy*, Part A, 2, 26 (1960)
4. HENRY HONECK
"THERMOS," A Thermalization Transport Theory Code for Reactor Lattice Calculations. BNL-5826 (1961)
5. FIRST 1 Code, EURATOM Report to be published
6. FIRST 2 Code.

Code description

1 Name of Code. GRAFA
2 Computer. IBM-7044 FORTRAN IV, 32 K, 3 tapes
3 Nature of physical problem solved. Computes the scalar thermal neutron spectum as function of spatial regions in square and hexagonal lattice. The scattering kernel model is arbitrary.
4 Method of solution. The neutron transport is solved by using the first flight collision probability method based on the integral transport equation. The scattering kernel is input from the library tape, the simultaneous equation is solved by the iteration method with over-relaxation.
5 Restriction. Isotropic neutron scattering. Mesh size limitation; 30 energy groups, 10 spatial regions, 6 mixtures (assigned arbitrarily to the spatial regions) composed of 11 isotopes.
6 Typical running time. The following calculation times occur for the case of 30 energy groups and 10 subregions (The calculation of the first flight collision probability is performed in 6 groups, and the F. F. C. P. s. in the other energy groups are obtained by interpolation): about seven minutes for the closely packed water lattice and 1.5 minutes for the loose lattice of D_2O moderator.
7 Usual features. The hexagonal and square lattices, which are too complicated to solve by the differential transport equation for example Pn method and Sn method are dealt with. The formulation used in calculation of F. F. C. P. is exact. Arbitrary space and energy mesh and arbitrary energy groups for calculating F. F. C. P..
8 Related and auxiliary Program. Similar to the FIRST 1 code and FIRST 2 code. Reference (5) and (6).
9 Status. Fully tested.
10 References 1 H. TAKAHASHI and T. NAKAYAMA: The first flight collision probability in the Square and Hexagonal Lattice Systems JAERI-Report 1072 Nov. 1964. 2 H. TAKAHASHI. *Nuclear Science and Engineering* 5 338 (1959)
11 Machine requirements IBM-7044 and 3 tapes.
12 Programming language used. FORTRAN IV
13 Monitor No. Version 8. modification 1
14 Other restriction and establishment. Need library tape for input.
15 Name of authors. HIROSHI TAKAHASHI, TAKASHI NAKYAMA, TSUNEO TSUTSUI, Japan Atomic Energy Research Institute
16 Material available. FORTRAN deck and library tape.

```
  2  3                                                                    IMN, IMNN
 10 30  5  3                                                             IR, IE, IM, IRM
  1  1  1  1  2  3  3  3  3  3                                           NDSH(10)
+1.5875E-01+2.2469E-01+2.7496E-01+3.1750E-01+4.0130E-01                 R(10)
+4.7983E-01+5.5835E-01+6.3688E-01+7.1540E-01+7.1540E-01
+1.0000E-00+8.4000E-01+7.0000E-01+5.8000E-01+4.8500E-01
+4.0000E-01+3.3000E-01+2.7800E-01+2.3100E-01+1.9200E-01
+1.6000E-01+1.3300E-01+1.1100E-01+9.2000E-02+7.6000E-02                 E(30)
+6.3500E-02+5.3000E-02+4.4000E-02+3.6600E-02+3.0400E-02
+2.5200E-02+2.1000E-02+1.7500E-02+1.4500E-02+1.2000E-02
+1.0000E-02+8.0000E-03+5.6000E-03+3.5000E-03+2.0000E-03
+1.0000F+00+1.0000E+00+1.0000E+00+1.0000E+00+1.0000E+00                 DISA(10)
+1.0000E+00+1.0000E+00+1.0000E+00+1.0000E+00+1.0000E+00
+0.0000E+00+0.0000E+00+0.0000E+00+5.5400E-03+3.9267E-01                 QNU(5, 3)
+0.0000E+00+0.0000E+00+8.4350E-02+0.0000E+00+0.0000E+00
+1.3646E+00+1.2573E-01+0.0000E+00+0.0000E+00+0.0000E+00
+2.5500E-02+2.5500E-02+2.5500E-02+2.5500E-02+2.5500E-02                 TEMP(10)
+2.5500E-02+2.5500E-02+2.5500E-02+2.5500E-02+2.5500E-02
+0.0000E-00+1.6000E+01+2.7000E+01+2.3500E+02+2.3800E+02                 QMASS(5)
  1                                                                     IABB
+1.0000E+00+1.6000E+01+2.7000E+01+2.3500E+02+2.3800 +02                 FACTOR(10)
  H20   255      O   255     AL   255  U235   255  U238   255  RNAME(5)
+1.1100E+00+1.1100E+00+1.1100E+00+1.1100E+00+1.1100E+00
+1.1100E+00+1.1100E+00+1.1100E+00+1.1100E+00+1.1050E+00
+1.1000E+00+1.0900E+00+1.0800E+00+1.0600E+00+1.0350E+00                 S(1, 30)
+1.0100E+00+9.8000E-01+9.4000E-01+9.1000E-01+8.7000E-01                 in SUBROUTINE
+8.3500E-01+7.9000E-01+7.5000E-01+7.0000E-01+6.8000E-01                   ABSORP
+6.5000E-01+6.1000E-01+5.5500E-01+4.9000E-01+4.3000E-01
+2.5000E-03+2.7800E-03+3.0000E-03+3.3000E-03+3.6000E-03
+4.0000E-03+4.3800E-03+4.7800E-03+5.2000E-03+5.7000E-03
+6.3000E-03+6.8500E-03+7.6000E-03+8.3000E-03+9.1500E-03                 S(2, 30)
+1.0000E-02+1.1000E-02+1.2000E-02+1.3200E-02+1.4600E-02
+1.6000E-02+1.7500E-02+1.9000E-02+2.1000E-02+2.3000E-02
+2.5400E-02+2.8400E-02+3.4000E-02+4.3000E-02+5.7000E-02
+2.6000E-02+2.8500E-02+3.1200E-02+3.4000E-02+3.7000E-02                 S(3, 30)
+4.1000E-02+4.6000E-02+4.9500E-02+5.4000E-02+5.9000E-02
+6.5000E-02+7.0500E-02+7.8000E-02+8.5000E-02+9.3500E-02
+1.0300E-01+1.1300E-01+1.2300E-01+1.3500E-01+1.4900E-01                 S(4, 30)
+1.6400E-01+1.7900E-01+1.9400E-01+2.1400E-01+2.3400E-01
+2.5900E-01+2.8900E-01+3.4500E-01+4.3500E-01+5.7500E-01
+4.9000E+00+5.6000E+00+6.4000E+00+7.8000E+00+9.7000E+00
+1.3500E+01+2.0000E+01+2.3000E+01+2.1500E+01+2.1200E+01
+2.2500E+01+2.4300E+01+2.7400E+01+3.1000E+01+3.5000E+01                 S(5, 30)
+3.9000E+01+4.3500E+01+4.8500E+01+5.4200E+01+6.1000E+01
+6.8500E+01+7.5500E+01+8.2300E+01+9.1000E+01+1.0000E+02
+1.1000E+02+1.2400E+02+1.5000E+02+1.8800E+02+2.5000E+02
+5.2000E-02+5.7000E-02+6.2400E-02+6.8000E-02+7.4000E-02                 SOCMA(30)
+8.2000E-02+9.2000E-02+9.9000E-02+1.0800E-01+1.1800E-01                 in SUBROUTINE.
+1.3000E-01+1.4100E-01+1.5600E-01+1.7000E-01+1.8700E-01                   SAUCE
+2.0600E-01+2.2600E-01+2.4600E-01+2.7000E-01+2.9800E-01
+3.2800E-01+3.5800E-01+3.8800E-01+4.2800E-01+4.6800E-01
+5.1800E-01+5.7800E-01+6.9000E-01+8.7000E-01+1.1500E-00
  6 10  1  1  5  6                                                      ISSTEP, IQMAX, IKIND, LREAD, ILSI, ISTEPS
  6  4  4  4  2  2  2  2  2  1                                          MDASH(10)
100 21+1.0000E+00+1.0000E-05                                           ITMAX, MAXF,  OVERR, EPSIRO
```

L in SUBROUTINE ACTIVE

```
+2.0000E+02+2.3300E+02+2.7300E+02+3.2000E+02+3.7500E+02
+4.4100E+02+5.2500E+02+6.1200E+02+7.3800E+02+8.7000E+02
+1.0000E+03+1.1400E+03+1.2700E+03+1.4050E+03+1.5500E+03
+1.7300E+03+1.8700E+03+2.0500E+03+2.2600E+03+2.4900E+03
+2.7100E+03+2.9700E+03+3.2300E+03+3.5800E+03+3.9000E+03
+4.2800E+03+4.8000E+03+5.7000E+03+7.2100E+03+9.6500E+03
+6.0000E+01+5.4500E+01+5.6000E+01+6.6000E+01+8.2000E+01
+1.1000E+02+1.6500E+02+1.8900E+02+1.7700E+02+1.7500E+02
+1.8500E+02+2.0700E+02+2.3300E+02+2.6400E+02+3.0000E+02
+3.3500E+02+3.7500E+02+4.2000E+02+4.7000E+02+5.2000E+02
+5.7500E+02+6.4000E+02+7.1000E+02+7.8500E+02+8.8000E+02
+9.6000E+02+1.0650E+03+1.2500E+03+1.5500E+03+2.2000E+03
+3.6000E+01+4.6000E+01+6.4000E+01+1.1500E+02+2.2000E+02
+5.5000E+02+2.1000E+03+3.1200E+03+1.2700E+03+7.4000E+02
+5.4200E+02+4.6300E+02+4.4750E+02+4.4800E+02+4.6000E+02
+4.9000E+02+5.2000E+02+5.4500E+02+5.9800E+02+6.4000E+02
+7.0200E+02+7.7200E+02+8.4000E+02+9.2000E+02+1.0100E+03
+1.1000E+03+1.2250E+03+1.4500E+03+1.8300E+03+2.4200E+03
+2.8000E+00+2.5200E+00+2.8000E+00+3.0300E+00+3.3200E+00
+3.6000E+00+4.0000E+00+4.4000E+00+4.8500E+00+5.4000E+00
+5.8400E+00+6.4000E+00+7.2000E+00+7.8200E+00+8.0400E+00
+9.2000E+00+1.0000E+01+1.0800E+00+1.1750E+01+1.2800E+01
+1.4000E+01+1.5200E+01+1.6500E+01+1.8000E+01+1.9500E+01
+2.1100E+01+2.3400E+01+2.7500E+01+3.4000E+01+4.3400E+01
+1.2500E+01+2.1000E+01+3.8000E+01+7.2000E+01+1.2700E+02
```

SIGMAA(30)

SIGMAA(30)

SIGMAA(30)

SIGMAA(30)

Code GRAFA Input I

```
I N P U T

   PARAMETERS
      PROBLEM NUMBER
      NUMBER OF SUBREGION      (IR).....10
      NUMBER OF ENERGY GROUPS (IE).....30
      NUMBER OF MATERIALS      (IRM)..... 3
      NUMBER OF ELEMENTS       (IM)..... 5
      NUMBER OF ELEMENTS WHOSE SCATTERING
      KERNELS ARE READ FROM DATA (IMN)..... 2
```

NUMBER OF SUBREGION	RADIUS (R)	NUMBER OF MATERIAL (NDASH)	TEMPERATURE (TEMP)	EPITHERMAL FLAX DISTRIBUTION (DISA)
1	1.5875E-01	1	2.5500E-02	1.0000E 00
2	2.2469E-01	1	2.5500E-02	1.0000E 00
3	2.7496E-01	1	2.5500E-02	1.0000E 00
4	3.1750E-01	1	2.5500E-02	1.0000E 00
5	4.0130E-01	2	2.5500E-02	1.0000E 00
6	4.7983E-01	3	2.5500E-02	1.0000E 00
7	5.5835E-01	3	2.5500E-02	1.0000E 00
8	6.3688E-01	3	2.5500E-02	1.0000E 00
9	7.1540E-01	3	2.5500E-02	1.0000E 00
10	7.1540E-01	3	2.5500E-02	1.0000E 00

Code GRAFA Output I

INPUT OF CHAIN (1)

FREE CROSS-SECTION OF ELEMENTS (QNU)

NUMBER OF MATERIAL	NUMBER OF ELEMENT 1	2	3	4	5	6
	H20 255	O 255	AL			
1	0.	0.	0.	5.5400E-03	3.9267E-01	
2	0.	0.	8.4350E-02	0.	0.	
3	1.3646E 00	1.2573E-01	0.	0.	0.	

OTHER CONSTANTS

	NUMBER OF ELEMENT 1	2	3	4	5	6
MASS OF ELEMENT (QMASS)	0.	1.6000E 01	2.7000E 01	2.3500E 02	2.3800E 02	
EFFECTIVE MASS (FACTOR)	1.0000E 00	1.6000E 01				

Code GRAFA Output II

NUMBER OF ENERGY GROUPS	MIDDLE ENERGY (E)	ENERGY WIDTH (DELTA)	ABSORPTION CROSS-SECTION QNKERN(1E1) 1	2	3	4	5	6
1	1.0000E 00	1.6000E-01	4.7565E-02	2.1931E-03	3.4115E-03			
2	8.4000E-01	1.5000E-01	5.3406E-02	2.4040E-03	3.7936E-03			
3	7.0000E-01	1.3000E-01	5.9959E-02	2.6317E-03	4.0938E-03			
4	5.8000E-01	1.0750E-01	6.9914E-02	2.8679E-03	4.5032E-03			
5	4.8500E-01	9.0000E-02	8.2796E-02	3.1209E-03	4.9126E-03			
6	4.0000E-01	7.7500E-02	1.0699E-01	3.4583E-03	5.4584E-03			
7	3.3000E-01	6.1000E-02	1.4693E-01	3.8801E-03	5.9769E-03			
8	2.7800E-01	4.9500E-02	1.6629E-01	4.1753E-03	6.5228E-03			
9	2.3100E-01	4.3000E-02	1.6152E-01	4.5549E-03	7.0959E-03			
10	1.9200E-01	3.5500E-02	1.6378E-01	4.9766E-03	7.7782E-03			
11	1.6000E-01	2.9500E-02	1.7570E-01	5.4828E-03	8.5970E-03			
12	1.3300E-01	2.4500E-02	1.8999E-01	5.9467E-03	9.3475E-03			
13	1.1100E-01	2.0500E-02	2.1305E-01	6.5793E-03	1.0371E-02			
14	9.2000E-02	1.7500E-02	2.3849E-01	7.1697E-03	1.1326E-02			
15	7.6000E-02	1.4250E-02	2.6733E-01	7.8867E-03	1.2486E-02			
16	6.3500E-02	1.1500E-02	2.9695E-01	8.6880E-03	1.3646E-02			
17	5.3000E-02	9.7500E-03	3.2973E-01	9.5315E-03	1.5011E-02			
18	4.4000E-02	8.2000E-03	3.6529E-01	1.0375E-02	1.6375E-02			
19	3.6600E-02	6.8000E-03	4.0629E-01	1.1387E-02	1.8013E-02			
20	3.0400E-02	5.7000E-03	4.5496E-01	1.2568E-02	1.9923E-02			
21	2.5200E-02	4.7000E-03	5.0829E-01	1.3833E-02	2.1834E-02			
22	2.1000E-02	3.8500E-03	5.5885E-01	1.5099E-02	2.3881E-02			
23	1.7500E-02	3.2500E-03	6.0830E-01	1.6364E-02	2.5927E-02			
24	1.4500E-02	2.7500E-03	6.7220E-01	1.8051E-02	2.8657E-02			
25	1.2000E-02	2.2500E-03	7.3777E-01	1.9738E-02	3.1386E-02			
26	1.0000E-02	2.0000E-03	8.1280E-01	2.1847E-02	3.4661E-02			
27	8.0000E-03	2.2000E-03	9.1392E-01	2.4377E-02	3.8755E-02			
28	5.6000E-03	2.2500E-03	1.1019E 00	2.9101E-02	4.6396E-02			
29	3.5000E-03	1.8000E-03	1.3831E 00	3.6692E-02	5.8678E-02			
30	2.0000E-03	2.7500E-03	1.8366E 00	4.8501E-02	7.7782E-02			

Code GRAFA Output III

NUMBER OF ENERGY GROUPS	TOTAL CROSS-SECTIONS QNKERN(IE2) 1	2	3	4	5	6
1	4.4577E-01	8.6543E-02	1.5543E 00			
2	4.5162E-01	8.6754E-02	1.5748E 00			
3	4.5817E-01	8.6982E-02	1.5944E 00			
4	4.6812E-01	8.7218E-02	1.6280E 00			
5	4.8101E-01	8.7471E-02	1.6521E 00			
6	5.0520E-01	8.7808E-02	1.7546E 00			
7	5.4514E-01	8.8230E-02	1.8177E 00			
8	5.6450E-01	8.8525E-02	1.8558E 00			
9	5.5973E-01	8.8905E-02	1.9191E 00			
10	5.6199E-01	8.9327E-02	2.0977E 00			
11	5.7391E-01	8.9833E-02	2.1707E 00			
12	5.8820E-01	9.0297E-02	2.2506E 00			
13	6.1126E-01	9.0929E-02	2.3177E 00			
14	6.3670E-01	9.1520E-02	2.4325E 00			
15	6.6554E-01	9.2237E-02	2.5163E 00			
16	6.9516E-01	9.3038E-02	2.5531E 00			
17	7.2794E-01	9.3882E-02	2.6434E 00			
18	7.6350E-01	9.4725E-02	2.8336E 00			
19	8.0450E-01	9.5737E-02	3.0809E 00			
20	8.5317E-01	9.6918E-02	3.3398E 00			
21	9.0650E-01	9.8183E-02	3.6012E 00			
22	9.5706E-01	9.9449E-02	3.8485E 00			
23	1.0065E 00	1.0071E-01	4.0836E 00			
24	1.0704E 00	1.0240E-01	4.3212E 00			
25	1.1360E 00	1.0409E-01	4.5571E 00			
26	1.2110E 00	1.0620E-01	4.7795E 00			
27	1.3121E 00	1.0873E-01	5.0447E 00			
28	1.5002E 00	1.1345E-01	5.5173E 00			
29	1.7814E 00	1.2104E-01	6.3327E 00			
30	2.2348E 00	1.3285E-01	7.5654E 00			

Code GRAFA Output IV

NUMBER OF ENERGY GROUPS	NEUTRON SAUCE SPECTRA QNKERN(IE3) 1	2	3	4	5	6
1	0.	3.8993E-02	1.5984E 00			
2	0.	0.	1.5147E 00			
3	0.	0.	1.5147E 00			
4	0.	0.	1.5147E 00			
5	0.	0.	1.5147E 00			
6	0.	0.	1.5147E 00			
7	0.	0.	1.5147E 00			
8	0.	0.	1.5147E 00			
9	0.	0.	1.5147E 00			
10	0.	0.	1.5079E 00			
11	0.	0.	1.5011E 00			
12	0.	0.	1.4874E 00			
13	0.	0.	1.4738E 00			
14	0.	0.	1.4465E 00			
15	0.	0.	1.4124E 00			
16	0.	0.	1.3782E 00			
17	0.	0.	1.3373E 00			
18	0.	0.	1.2827E 00			
19	0.	0.	1.2418E 00			
20	0.	0.	1.1872E 00			
21	0.	0.	1.1394E 00			
22	0.	0.	1.0780E 00			
23	0.	0.	1.0235E 00			
24	0.	0.	9.5522E-01			
25	0.	0.	9.2793E-01			
26	0.	0.	8.8699E-01			
27	0.	0.	8.3241E-01			
28	0.	0.	7.5735E-01			
29	0.	0.	6.6865E-01			
30	0.	0.	5.8678E-01			

Code GRAFA Output V

I N P U T A N D O U T P U T O F C H A I N (2)

HEXAGONAL LATTICE GEOMETRY (IKIND)

THE NUMBER OF DIVISIONS IN THE INTEGRATION WITH RESPECT TO PHI IS 6 (ISSTEP)

THE MAXIMUM NUMBER OF CELL WHICH IS EXPECTED TO CONTRIBUTE TO F.F.C.P. 10 (IQMAX)

THE NUMBERS OF DIVISION IN EACH SUBREGION(MDASH)

NO. OF SUB.	1	2	3	4	5	6	7	8	9	10
NO. OF DIV.	6	4	4	4	2	2	2	2	2	1

ILS= 1

TOTAL CROSS SECTION
4.4577E-01 4.4577E-01 4.4577E-01 4.4577E-01 8.6543E-02 1.5543E 00
1.5543E 00 1.5543E 00 1.5543E 00 1.5543E 00

P(I,J)

1	1	0.92778E-01	1	2	0.48174E-01	1	3	0.38666E-01	1	4	0.27923E-01	1	5	0.14522E-01
1	6	0.19166E-00	1	7	0.17340E-00	1	8	0.17128E-00	1	9	0.15258E-00	1	10	0.89018E-01
2	1	0.48017E-01	2	2	0.79886E-01	2	3	0.35884E-01	2	4	0.37866E-01	2	5	0.10584E-01
2	6	0.19291E-00	2	7	0.17739E-00	2	8	0.17015E-00	2	9	0.16014E-00	2	10	0.87179E-01
3	1	0.38795E-01	3	2	0.36122E-01	3	3	0.67797E-01	3	4	0.27274E-01	3	5	0.14370E-01
3	6	0.20172E-00	3	7	0.17773E-00	3	8	0.17572E-00	3	9	0.16823E-00	3	10	0.92251E-01
4	1	0.27921E-01	4	2	0.37987E-01	4	3	0.27181E-01	4	4	0.69783E-01	4	5	0.10655E-01
4	6	0.21175E-00	4	7	0.19147E-00	4	8	0.17478E-00	4	9	0.15634E-00	4	10	0.92128E-01
5	1	0.31295E-01	5	2	0.22885E-01	5	3	0.30865E-01	5	4	0.22964E-01	5	5	0.43993E-01
5	6	0.23221E-00	5	7	0.19947E-00	5	8	0.17189E-00	5	9	0.15437E-00	5	10	0.90054E-01
6	1	0.20020E-01	6	2	0.20216E-01	6	3	0.21001E-01	6	4	0.22121E-01	6	5	0.11255E-01
6	6	0.29743E-00	6	7	0.20254E-00	6	8	0.17090E-00	6	9	0.15380E-00	6	10	0.80725E-01
7	1	0.15375E-01	7	2	0.15780E-01	7	3	0.15706E-01	7	4	0.16978E-01	7	5	0.82069E-02
7	6	0.17192E-00	7	7	0.28954E-00	7	8	0.20900E-00	7	9	0.17214E-00	7	10	0.85352E-01
8	1	0.13190E-01	8	2	0.13145E-01	8	3	0.13486E-01	8	4	0.13460E-01	8	5	0.61421E-02
8	6	0.12599E-00	8	7	0.18151E-00	8	8	0.31126E-00	8	9	0.21762E-00	8	10	0.10419E-00
9	1	0.10386E-01	9	2	0.10937E-01	9	3	0.11413E-01	9	4	0.10643E-01	9	5	0.48761E-02
9	6	0.10022E-00	9	7	0.13216E-00	9	8	0.19237E-00	9	9	0.33401E-00	9	10	0.19298E-00
10	1	0.12246E-01	10	2	0.12033E-01	10	3	0.12649E-01	10	4	0.12675E-01	10	5	0.57488E-02
10	6	0.10632E-00	10	7	0.13243E-00	10	8	0.18614E-00	10	9	0.39062E-00	10	10	0.12974E-00

I N P U T A N D O U T P U T O F C H A I N (3)

MAXIMUM NUMBER OF ITERATION (IITMAX) 100

NUMBER OF ENERGY GROUP IN WHICH THE FLUX IS EXPECTED TO BE MAXIMUM (MAXF) 21

OVER RELAXATION (OVERR) 1.0000E 00

CONVERGENCE CRITERION (EPSIRO) 1.0000E-05

THE NEUTRON SPECTRA

ACCURACY OF THE NEUTRON SPECTRA (RESIDU) 2.1744E-04

NUMBER OF ITERATION (ITNOM) 39

ENERGY GROUPS	NUMBER OF SUBREGION (PHI) 1	2	3	4	5	6	7	8	9	10
1	1.7822E 00	1.7810E 00	1.7909E 00	1.7900E 00	1.8238E 00	1.8081E 00	1.8106E 00	1.8116E 00	1.8131E 00	1.8123E 00
2	2.1974E 00	2.2003E 00	2.2108E 00	2.2128E 00	2.2046E 00	2.2377E 00	2.2447E 00	2.2487E 00	2.2523E 00	2.2503E 00
3	2.7181E 00	2.7224E 00	2.7374E 00	2.7391E 00	2.7059E 00	2.7790E 00	2.7891E 00	2.7950E 00	2.8004E 00	2.7978E 00
4	3.2598E 00	3.2659E 00	3.2873E 00	3.2882E 00	3.2440E 00	3.3452E 00	3.3597E 00	3.3682E 00	3.3759E 00	3.3727E 00
5	3.8918E 00	3.9004E 00	3.9310E 00	3.9302E 00	3.8939E 00	4.0600E 00	4.0809E 00	4.0934E 00	4.1045E 00	4.1005E 00
6	4.6654E 00	4.6782E 00	4.7248E 00	4.7210E 00	4.8022E 00	4.8481E 00	4.8797E 00	4.8986E 00	4.9154E 00	4.9102E 00
7	5.4424E 00	5.4621E 00	5.5304E 00	5.5287E 00	5.9538E 00	6.0555E 00	6.1069E 00	6.1374E 00	6.1640E 00	6.1566E 00
8	6.4811E 00	6.5081E 00	6.5952E 00	6.5992E 00	7.2032E 00	7.4783E 00	7.5509E 00	7.5940E 00	7.6305E 00	7.6221E 00
9	8.5213E 00	8.5579E 00	8.6677E 00	8.6811E 00	9.1536E 00	9.5875E 00	9.6858E 00	9.7447E 00	9.7935E 00	9.7850E 00
10	1.5238E 01	1.5308E 01	1.5501E 01	1.5541E 01	1.6014E 01	1.6044E 01	1.6226E 01	1.6338E 01	1.6428E 01	1.6418E 01
11	2.2372E 01	2.2487E 01	2.2777E 01	2.2864E 01	2.3388E 01	2.3747E 01	2.4055E 01	2.4247E 01	2.4399E 01	2.4391E 01
12	3.8639E 01	3.8861E 01	3.9399E 01	3.9557E 01	4.0187E 01	4.0521E 01	4.1121E 01	4.1504E 01	4.1800E 01	4.1794E 01
13	6.0452E 01	6.0851E 01	6.1777E 01	6.2045E 01	6.3390E 01	6.4078E 01	6.5167E 01	6.5867E 01	6.6399E 01	6.6405E 01
14	9.2152E 01	9.2843E 01	9.4390E 01	9.4830E 01	9.7695E 01	9.7126E 01	9.9016E 01	1.0024E 02	1.0116E 02	1.0120E 02
15	1.2970E 02	1.3080E 02	1.3319E 02	1.3385E 02	1.3933E 02	1.3849E 02	1.4153E 02	1.4351E 02	1.4496E 02	1.4506E 02
16	1.6370E 02	1.6527E 02	1.6854E 02	1.6944E 02	1.7787E 02	1.8047E 02	1.8481E 02	1.8761E 02	1.8962E 02	1.8981E 02
17	1.9091E 02	1.9301E 02	1.9726E 02	1.9867E 02	2.0938E 02	2.1877E 02	2.2462E 02	2.2834E 02	2.3092E 02	2.3126E 02
18	2.1538E 02	2.1806E 02	2.2341E 02	2.2546E 02	2.3827E 02	2.4847E 02	2.5594E 02	2.6064E 02	2.6381E 02	2.6436E 02
19	2.3121E 02	2.3446E 02	2.4085E 02	2.4364E 02	2.5815E 02	2.6641E 02	2.7549E 02	2.8116E 02	2.8489E 02	2.8570E 02
20	2.3489E 02	2.3859E 02	2.4583E 02	2.4934E 02	2.6519E 02	2.7341E 02	2.8400E 02	2.9056E 02	2.9479E 02	2.9588E 02
21	2.2810E 02	2.3211E 02	2.3991E 02	2.4407E 02	2.5968E 02	2.7058E 02	2.8241E 02	2.8972E 02	2.9431E 02	2.9568E 02
22	2.1804E 02	2.2226E 02	2.3068E 02	2.3520E 02	2.5276E 02	2.6003E 02	2.7277E 02	2.8054E 02	2.8539E 02	2.8689E 02
23	2.0276E 02	2.0705E 02	2.1582E 02	2.2053E 02	2.3812E 02	2.4395E 02	2.5722E 02	2.6523E 02	2.7019E 02	2.7179E 02
24	1.8118E 02	1.8538E 02	1.9416E 02	1.9888E 02	2.1708E 02	2.2348E 02	2.3697E 02	2.4503E 02	2.4998E 02	2.5163E 02
25	1.5865E 02	1.6265E 02	1.7122E 02	1.7580E 02	1.9325E 02	2.0097E 02	2.1435E 02	2.2227E 02	2.2708E 02	2.2873E 02
26	1.3586E 02	1.3959E 02	1.4775E 02	1.5207E 02	1.6817E 02	1.7878E 02	1.9182E 02	1.9947E 02	2.0406E 02	2.0569E 02
27	1.0837E 02	1.1234E 02	1.2081E 02	1.2568E 02	1.3767E 02	1.5189E 02	1.6484E 02	1.7 17E 02	1.7640E 02	1.7793E 02
28	7.4324E 01	7.7874E 01	8.5335E 01	8.9869E 01	1.0074E 02	1.1433E 02	1.2581E 02	1.3212E 02	1.3562E 02	1.3690E 02
29	4.5000E 01	4.7748E 01	5.3460E 01	5.7074E 01	6.6200E 01	7.5964E 01	8.4846E 01	8.9612E 01	9.2158E 01	9.3105E 01
30	2.3762E 01	2.5594E 01	2.9373E 01	3.1843E 01	3.9007E 01	4.5211E 01	5.1247E 01	5.4405E 01	5.6031E 01	5.6645E 01

THE NEUTRON SPECTRA

ACCURACY OF THE NEUTRON SPECTRA (RESIDU) 2.1744E-04

NUMBER OF ITERATION (ITNOM) 40

ENERGY GROUPS	NUMBER OF SUBREGION (PHI) 1	2	3	4	5	6	7	8	9	10
1	1.7822E 00	1.7810E 00	1.7909E 00	1.7900E 00	1.8238E 00	1.8081E 00	1.8106E 00	1.8116E 00	1.8131E 00	1.8123E 00
2	2.1974E 00	2.2003E 00	2.2108E 00	2.2128E 00	2.2046E 00	2.2377E 00	2.2447E 00	2.2487E 00	2.2523E 00	2.2503E 00
3	2.7181E 00	2.7224E 00	2.7374E 00	2.7391E 00	2.7059E 00	2.7790E 00	2.7891E 00	2.7950E 00	2.8004E 00	2.7978E 00
4	3.2598E 00	3.2659E 00	3.2873E 00	3.2882E 00	3.2440E 00	3.3452E 00	3.3597E 00	3.3682E 00	3.3759E 00	3.3727E 00
5	3.8918E 00	3.9004E 00	3.9310E 00	3.9302E 00	3.8939E 00	4.0600E 00	4.0809E 00	4.0934E 00	4.1045E 00	4.1005E 00
6	4.6654E 00	4.6782E 00	4.7248E 00	4.7210E 00	4.8022E 00	4.8481E 00	4.8797E 00	4.8986E 00	4.9154E 00	4.9102E 00
7	5.4424E 00	5.4621E 00	5.5304E 00	5.5287E 00	5.9538E 00	6.0555E 00	6.1069E 00	6.1374E 00	6.1640E 00	6.1566E 00
8	6.4811E 00	6.5081E 00	6.5952E 00	6.5992E 00	7.2032E 00	7.4783E 00	7.5509E 00	7.5940E 00	7.6305E 00	7.6221E 00
9	8.5213E 00	8.5579E 00	8.6677E 00	8.6811E 00	9.1536E 00	9.5875E 00	9.6858E 00	9.7447E 00	9.7935E 00	9.7850E 00
10	1.5238E 01	1.5308E 01	1.5501E 01	1.5541E 01	1.6014E 01	1.6044E 01	1.6226E 01	1.6338E 01	1.6428E 01	1.6418E 01
11	2.2372E 01	2.2487E 01	2.2777E 01	2.2864E 01	2.3388E 01	2.3747E 01	2.4055E 01	2.4247E 01	2.4399E 01	2.4391E 01
12	3.8639E 01	3.8861E 01	3.9399E 01	3.9557E 01	4.0187E 01	4.0521E 01	4.1121E 01	4.1504E 01	4.1800E 01	4.1794E 01
13	6.0452E 01	6.0851E 01	6.1777E 01	6.2045E 01	6.3390E 01	6.4078E 01	6.5167E 01	6.5867E 01	6.6399E 01	6.6405E 01
14	9.2152E 01	9.2843E 01	9.4390E 01	9.4830E 01	9.7695E 01	9.7126E 01	9.9016E 01	1.0024E 02	1.0116E 02	1.0120E 02
15	1.2970E 02	1.3080E 02	1.3319E 02	1.3385E 02	1.3933E 02	1.3849E 02	1.4153E 02	1.4351E 02	1.4496E 02	1.4506E 02
16	1.6370E 02	1.6527E 02	1.6854E 02	1.6944E 02	1.7787E 02	1.8047E 02	1.8481E 02	1.8761E 02	1.8962E 02	1.8981E 02
17	1.9091E 02	1.9301E 02	1.9726E 02	1.9867E 02	2.0938E 02	2.1877E 02	2.2462E 02	2.2834E 02	2.3092E 02	2.3126E 02
18	2.1538E 02	2.1806E 02	2.2341E 02	2.2546E 02	2.3827E 02	2.4847E 02	2.5594E 02	2.6064E 02	2.6381E 02	2.6436E 02
19	2.3121E 02	2.3446E 02	2.4085E 02	2.4364E 02	2.5815E 02	2.6641E 02	2.7549E 02	2.8116E 02	2.8489E 02	2.8570E 02
20	2.3489E 02	2.3859E 02	2.4583E 02	2.4934E 02	2.6519E 02	2.7341E 02	2.8400E 02	2.9056E 02	2.9479E 02	2.9588E 02
21	2.2810E 02	2.3211E 02	2.3991E 02	2.4407E 02	2.5968E 02	2.7058E 02	2.8241E 02	2.8972E 02	2.9431E 02	2.9568E 02
22	2.1804E 02	2.2226E 02	2.3068E 02	2.3520E 02	2.5276E 02	2.6003E 02	2.7277E 02	2.8054E 02	2.8539E 02	2.8689E 02
23	2.0276E 02	2.0705E 02	2.1582E 02	2.2053E 02	2.3812E 02	2.4395E 02	2.5722E 02	2.6523E 02	2.7019E 02	2.7179E 02
24	1.8118E 02	1.8538E 02	1.9416E 02	1.9888E 02	2.1708E 02	2.2348E 02	2.3697E 02	2.4503E 02	2.4998E 02	2.5163E 02
25	1.5865E 02	1.6265E 02	1.7122E 02	1.7580E 02	1.9325E 02	2.0097E 02	2.1435E 02	2.2227E 02	2.2708E 02	2.2873E 02
26	1.3586E 02	1.3959E 02	1.4775E 02	1.5207E 02	1.6817E 02	1.7878E 02	1.9182E 02	1.9947E 02	2.0406E 02	2.0569E 02
27	1.0837E 02	1.1234E 02	1.2081E 02	1.2568E 02	1.3767E 02	1.5189E 02	1.6484E 02	1.7217E 02	1.7640E 02	1.7793E 02
28	7.4324E 01	7.7874E 01	8.5335E 01	8.9869E 01	1.0074E 02	1.1433E 02	1.2581E 02	1.3212E 02	1.3562E 02	1.3690E 02
29	4.5000E 01	4.7748E 01	5.3460E 01	5.7074E 01	6.6200E 01	7.5964E 01	8.4846E 01	8.9612E 01	9.2158E 01	9.3105E 01
30	2.3762E 01	2.5594E 01	2.9373E 01	3.1843E 01	3.9007E 01	4.5211E 01	5.1247E 01	5.4405E 01	5.6031E 01	5.6645E 01

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I N P U T  A N D  O U T P U T  O F  S U B R O U T I N E  A C T I V A T I O N

   NUMBER OF ACTIVATION MATERIALS (L)..... 4

   ENERGY GROUP-----CROSS-SECTION


    NUMBER OF ACTIVATION MATERIAL      (SIGMAA)
        2.0000E 02  2.3300E 02  2.7300E 02  3.2000E 02  3.7500E 02  4.4100E 02  5.2300E 02  6.1200E 02  7.3800E 02  8.7000E 02
        1.0000E 03  1.1400E 03  1.2700E 03  1.4050E 03  1.5500E 03  1.7000E 03  1.8700E 03  2.0500E 03  2.2600E 03  2.4900E 03
        2.7100E 03  2.9700E 03  3.2300E 03  3.5800E 03  3.9000E 03  4.2800E 03  4.8000E 03  5.7000E 03  7.2100E 03  9.6500E 03

NUMBER OF              N U M B E R  O F  S U B R E G I O N      (ACT)
ACTIV. MAT. 1          2           3           4           5           6           7           8           9           10
   1    4.2554E 04  4.3263E 04  4.4731E 04  4.5443E 04  4.8415E 04  5.0218E 04  5.2366E 04  5.3650E 04  5.4458E 04  5.4663E 04


    NUMBER OF ACTIVATION MATERIAL      (SIGMAA)
        6.0000E 01  5.4500E 01  5.6000E 01  6.6000E 01  8.2000E 01  1.1000E 02  1.6500E 02  1.8900E 02  1.7700E 02  1.7500E 02
        1.8500E 02  2.0700E 02  2.3300E 02  2.6400E 02  3.0000E 02  3.3500E 02  3.7500E 02  4.2000E 02  4.7000E 02  5.2000E 02
        5.7500E 02  6.4000E 02  7.1000E 02  7.8500E 02  8.8000E 02  9.6000E 02  1.0650E 03  1.2500E 03  1.5500E 03  2.2000E 03

NUMBER OF              N U M B E R  O F  S U B R E G I O N      (ACT)
ACTIV. MAT. 1          2           3           4           5           6           7           8           9           10
   2    8.8339E 03  8.9846E 03  9.2970E 03  9.4498E 03  1.0083E 04  1.0470E 04  1.0929E 04  1.1202E 04  1.1373E 04  1.1418E 04


    NUMBER OF ACTIVATION MATERIAL      (SIGMAA)
        3.6000E 01  4.6000E 01  6.4000E 01  1.1500E 02  2.2000E 02  5.5000E 02  2.1000E 03  3.1200E 03  1.2700E 03  7.4000E 02
        5.4200E 02  4.6300E 02  4.4700E 02  4.4800E 02  4.6000E 02  4.9000E 02  5.2000E 02  5.4500E 02  5.9800E 02  6.4000E 02
        7.0200E 02  7.7200E 02  8.4000E 02  9.2000E 02  1.0100E 03  1.1000E 03  1.2250E 03  1.4500E 03  1.8300E 03  2.4200E 03

NUMBER OF              N U M B E R  O F  S U B R E G I O N      (ACT)
ACTIV. MAT. 1          2           3           4           5           6           7           8           9           10
   3    1.4413E 04  1.4613E 04  1.5036E 04  1.5227E 04  1.6195E 04  1.6747E 04  1.7344E 04  1.7703E 04  1.7933E 04  1.7983E 04


    NUMBER OF ACTIVATION MATERIAL      (SIGMAA)
        2.3000E 00  2.5200E 00  2.8000E 00  3.0300E 00  3.3200E 00  3.6000E 00  4.0000E 00  4.4000E 00  4.8500E 00  5.4000E 00
        5.8400E 00  6.4000E 00  7.0000E 00  7.6200E 00  8.0400E 00  9.2000E 00  1.0000E 01  1.0800E 00  1.1750E 01  1.2800E 01
        1.4000E 01  1.5200E 01  1.6500E 01  1.8000E 01  1.9500E 01  2.1100E 01  2.3400E 01  2.7500E 01  3.4000E 01  4.3400E 01
                                                                                                                        16.34

NUMBER OF              N U M B E R  O F  S U B R E G I O N      (ACT)
ACTIV. MAT. 1          2           3           4           5           6           7           8           9           10
   4    2.0870E 02  2.1209E 02  2.1915E 02  2.2256E 02  2.3678E 02  2.4514E 02  2.5543E 02  2.6159E 02  2.6547E 02  2.6644E 02

END-OF-DATA ENCOUNTERED ON SYSTEM INPUT FILE.
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