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SP - 2000
Program for Calculating Fine Group Neutron Spectrum in Multi-region Cell and Effective Broad Group Constants

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Summary

The program SP-2000 is one of the 2000-series programs developed for utilizing AGLI fine group cross section library. This program calculates fine group neutron spectrum (about 2000 energy groups) in a multi-region cell by means of the collision probability method. The program also calculates homogenized broad group constants using the calculated neutron spectrum as a weighting function.

The program is written in FACOM-230/60 FORTRAN in order to take a part of DOYC computer software system developed for fast reactor analysis.

Computing time necessary for solving practical problems is shown to be reasonable and the results of neutron spectrum and heterogeneity effect on criticality obtained by the program are shown to well represent the measured ones.

SP-2000：多領域セル中の詳細群位置依存スペクトル及び実効少数群定数計算プログラム

日本原子力研究所　東海研究所

黒井英雄・東稔達三

1975年4月24日受理

要　旨

計算プログラム SP-2000 は，詳細群核断面積データライブラリ AGLI を入力として解析を行うために開発されたいわゆる 2000 シリーズプログラムの代表的な1つである．本プログラムは多領域セル中における位置依存詳細群（約 2000 エネルギー群）構造中性子スペクトルを衝突確率法を用いて求めると共に，求められた中性子スペクトルを重み函数として，均質化されたセル内実効少数群定数を計算するためのものである．本プログラムは FACOM-230/60 用 Fortran で作成されており，高速炉解析用計算システム DOYC に組みこまれ，多くの計算プログラムと有機的に結合されて使用される．本プログラムにより，妥当な計算時間内で中性子エネルギースペクトル，及び非均質効果の臨界性に与える効果を測定値とほぼ同程度の精度で予測し得ることが可能となった．

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1. Introduction

In order to satisfy the requirements of accurate analyses in the fast reactor physics, many efforts have been concentrated to develop codes to calculate the fine group neutron spectrum in a cell geometry. Such efforts have resulted in the computer codes MURAL[1], MC^2 [2] and ESE LEM-4[3]. However, the computer code MURAL is still not available for us, and in the other two codes, improvements in the following items are in need, (1) appropriate fine group cross section library which can be directly fed into the computer codes, (2) the method to deal with the capture resonance, especially for heavy nuclide.

Therefore, it was necessary to develop a new reliable computer code which can deal with the fine group neutron spectrum, especially for routine analyses of experimental data measured in fast critical assemblies.

The code SP-2000 is designed to be one of the most characteristic codes in the DOYC code system shown in **Fig. 1**. The SP-2000 calculates neutron spectrum in a multi-region cell by means of the collision probability method in about 2000 energy groups. Using this neutron spectrum as a weighting function, the code can provide cell-averaged macroscopic broad group cross sections up to 80 energy groups, which are used as cross section input to other codes provided in the DOYC system. This code also calculate various types of reaction rates such as region-dependent or cell-averaged values. The neutron leakage from the cell is dealt with in the fundamental mode approximation in a cell-averaged homogeneous medium.

All fine energy groups have the same lethargy width of 0.0085. All of the total, fission, capture and inelastic scattering cross sections are given for each fine group. A special provision for the histogram representation[4] has been incorporated in the code for dealing better with capture resonances of heavy nuclides.

The angular distribution of elastic scattering is represented by the Legendre polynomial, and the number of expansion terms up to seven can be dealt with in the code. Input data of the number of neutrons released per fission and the Legendre coefficients of elastic scattering should be given as a series of the Chebyshev polynomial with respect to the incident neutron lethargy. Using the expansion coefficients of Legendre polynomial, the group transfer cross section due to elastic scattering and the transport cross section are calculated for each fine group in the code. The group transfer cross sections due to inelastic scattering are also calculated for each fine group using a recurrence relation.

The cross section data read as input to the code are given from the following three cross section libraries which are stored on the disks or tapes.

1. AGLI-25/70 : The standard broad group cross section set (25 or 70 energy groups) including the so-called f-table,
2. AGLI library File-A : The fine group library of fission, capture and total cross section given by the histogram representation,
3. AGLI library File-B : The fine group library of inelastic scattering cross section, the number of neutrons per fission and Legendre expansion coefficients of the elastic scattering cross section.

Important calculated results are arranged in a proper form in order to utilize various supporting facilities provided for the DOYC computer code system which enables to display calculated results on a graphic plotter or to store them on a data pool disk for subsequent use.

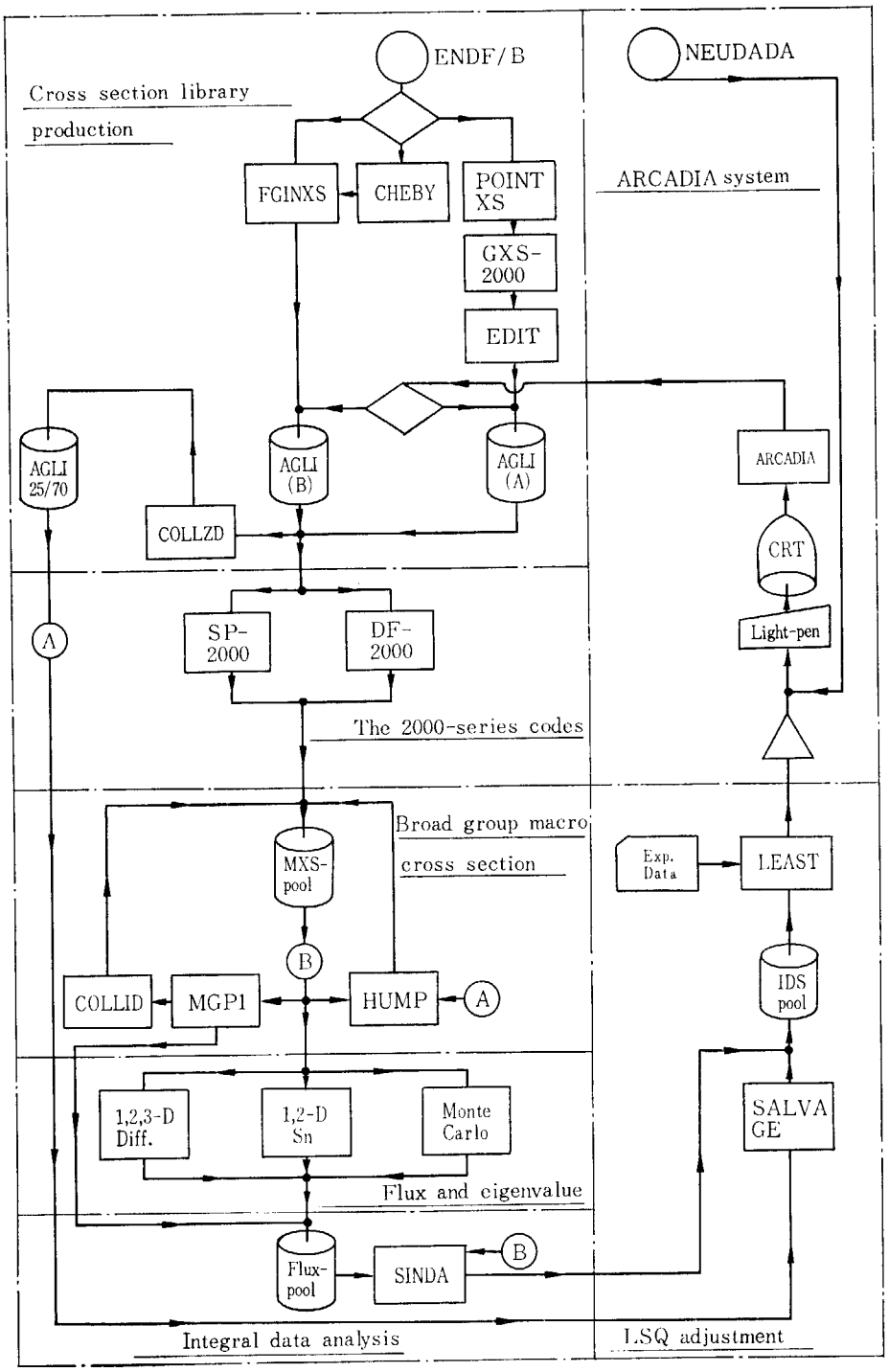

Fig. 1 Block diagram of DOYC system.

DF-2000 : Calculation of neutron flux and eigenvalue in one dimensional geometry in about 2000 energy group structure by means of the diffusion approximation, and collapsing the fine group cross section using the spacedependent fine group neutron spectrum

HUMP : Calculation of neutron spectrum and eigenvalue in multi-zone cell by means of collison probability method in a broad group structure less than 80 energy groups

MGP1 : Calculation of neutron flux and eigenvalue in one dimensional geometry by means of the P_1 approximation in about 80 energy groups structure

SINDA : Calculation of various integral data of sample reactivity worth, reaction rate, neutron lifetime, k_{eff} etc.

SALVAGE : Sensitivity analysis of cross section to various integral data

CHEBY : Chebyshev expansions of the Legendre coefficients of scattering angle and the average number of neutrons per fission, with respect to the incident neutron lethargy

FGINXS : Calculation of about 2000 energy group cross section of levelwise inelastic scattering cross sections and compilation of the AGLI/B

POINTXS : Production of point-wise cross section data

GXS-2000 : Calculation of histogram parameters in about 2000 energy group structure

EDIT : Compilation of the AGLI/A

COLLZD : Collapsing the fine group (2000 group) cross sections using weighting spectrum in bare homogeneous media, and production of the f-table used in a broad group cross section set

COLL1D : Collapsing group cross section using one dimensional spacedependent neutron spectrum as a weighting spectrum

SP-2000 : Calculation of about 2000 energy groups neutron spectrum in a multi-region cell by means of the collision probability method

LEAST : LSQ adjustment of group cross section

ARCADIA : On-line adjustment of the AGLI-library

AGLI (A) : AGLI fine group library/file A (1950 energy groups)

AGLI (B) : AGLI fine group library/file B (1950 energy groups)

AGLI-25/70 : Standard AGLI 70 and 25 group cross section set

MXS-pool : Data pool for macroscopic effective cross section (about 2000 TRK random access disk)

Flux-pool : Data pool for real and adjoint fluxes (about 4000 TRK random access disk)

IDS-pool : Data pool for various integral data and sensitivity functions (about 2000 TRK random access disk)

2. Preparative Calculation by Coarse Group

The fission source distribution in a multi-region cell and the material buckling of a medium are usually calculated by means of iterative procedures. It takes too much machine time to carry out these iterative procedures in fine group structure of 1950 energy groups for routine analyses of integral data. Therefore, the fission source distribution and the material buckling are calculated by a coarse group analysis (25 or 70 energy groups) before entering into a fine group analysis. These calculations are carried out using the standard group constants with the so-called f-table which are provided in the DOYC computer code system.

2. 1 Material Buckling

The material buckling of homogenized mixture of a cell is used for taking into account the neutron leakage. Using conventional notations, the buckling B^2 can be calculated to satisfy the following relation

$$\phi^i(\sigma_a{}^i+\sum_{j=1}^{IMAX}\sigma^{i\to j}+D^iB^2)=\chi^i+\sum_{j=1}^{IMAX}\sigma^{j\to i}\phi^j, \qquad (2\text{-}1)$$

$$\sum_i(\nu\sigma_f)^i\phi^i=1,$$

where i and j indicate energy group, and IMAX is the number of coarse energy groups used for the analysis

The method for obtaining B^2 in Eq. (2-1) is straightforward and the macroscopic cross section in Eq. (2-1) can be calculated in a conventional way using the standard group cross section with the f-table.

2. 2 Spatial Distribution of Fission Neutrons

The flux distribution in a multi-region cell is obtained by solving the integral transport equation in 25 or 70 energy group structure. Since detailed mathematical descriptions are given in the reference[5], only a brief description of the method used in SP-2000 is given here.

In a symbolic notation, the integral transport equation is written as[6]

$$\boldsymbol{F}=\boldsymbol{T}(\boldsymbol{S}+\boldsymbol{P}\boldsymbol{F}), \qquad (2\text{-}2)$$

where

$F_{n,i}$: volume integrated neutron flux of the i-th energy group at the n-th space mesh,

$T_{n,K,i}$: transport Kernel relating the neutron flux of the i-th energy group at the n-th space-mesh to a unit isotropic source of the i-th energy group at the K-th space-mesh,

$P_{K,j,i}$: the probability that a neutron of the j-th group will scatter into the i-th group at the K-th space mesh,

$S_{K,i}$: neutron source of the i-th group at the K-th space mesh.

The method for solving Eq. (2-2) used in the SP-2000 is the source iteration method in which one inserts a trial solution $\boldsymbol{F}^m$ into the right hand side of the equation and obtains an improved solution $\boldsymbol{F}^{m+1}$ on the left hand side,

$$\boldsymbol{F}^{m+1}=\boldsymbol{T}(\boldsymbol{S}+\boldsymbol{P}\boldsymbol{F}^m). \qquad (2\text{-}3)$$

If $\boldsymbol{A}$ denotes the absorption operator, the normalization condition is

$$\boldsymbol{AF}=1. \tag{2-4}$$

Using this normalization, the iteration equation is then,

$$\boldsymbol{F}^{m+1}=\frac{\boldsymbol{T}(\boldsymbol{S}+\boldsymbol{P}\boldsymbol{F}^{m})}{\boldsymbol{A}\boldsymbol{T}(\boldsymbol{S}+\boldsymbol{P}\boldsymbol{F}^{m})}. \tag{2-5}$$

Introducing an over-relaxation factor ω,

$$\boldsymbol{F}^{m+1}=\boldsymbol{F}^{m}+\omega\boldsymbol{R}^{m+1}, \tag{2-6}$$

where

$$\boldsymbol{R}^{m+1}=\frac{\boldsymbol{T}(\boldsymbol{S}+\boldsymbol{P}\boldsymbol{F}^{m})}{\boldsymbol{A}\boldsymbol{T}(\boldsymbol{S}+\boldsymbol{P}\boldsymbol{F}^{m})}-\boldsymbol{F}^{m}.$$

This iteration (2-6) is carried out until the following condition is satisfied; $|\boldsymbol{R}|\leq\varepsilon|\boldsymbol{F}|$, for each mesh point and energy group.

The method for obtaining the value of ω and the transport kernel used in SP-2000 are described in the reference[5].

The region dependent fission neutron sources $Q_f(l)$ are obtained by

$$Q_f(l)=\sum_{i=1}^{IMAX} F_{l,i}(\nu\sigma_f)_{l,i},$$

where $(\nu\sigma_f)_{l,i}$ denotes the fission cross section multiplied by the number of neutrons per fission of the i-th group in the region l.

The macroscopic cross sections in each region are straightforward in the off-resonance energy regions. In the resonance energy region, these are corrected using the f-table provided in the standard cross section set. The correction is made using the Bell-correction factor[7], in which a pseudo-potential scattering cross section per absorber atom σ_o in closely spaced lump is approximated by

$$\sigma_o=\sigma_p+\frac{S/(4V_0)}{1+(SV_0)/(4V_0V_1\Sigma_1)},$$

where

σ_p: potential scattering cross section per absorber,
S: lump surface area,
V_0: the total lump volume,
V_1: cell volume minus V_0,
Σ_1: macroscopic total cross section of environment material of lump.

As the result of this approximation, the macroscopic cross sections in the resonance energy region used for calculating spatial dependence of fission neutron source are somewhat under estimated.

3. Fine Group Cross Sections

Generation of macroscopic cross section necessary for computing fine group neutron spectrum is one of the most laborious procedures in usual computer codes of this kind. Since AGLI fine group cross section library is used as an input to the code SP-2000, provisions for the histogram representation of cross sections[4], for fine group inelastic scattering matrices and for Legendre and Chebyshev expansions of cross sections are necessary to generate macroscopic cross secteons.

3. 1 Effective Cross Section by Means of Histogram Representation

The histogram representation of a resonance structure proposed by J.L. Rowlands and J.D.

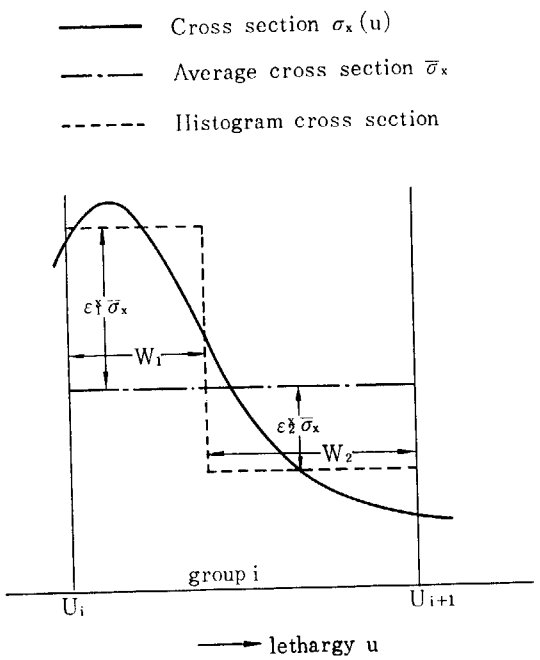

Fig. 2 Histogram representation of cross section $\sigma_x(u)$

Macdoougall[1] is applied to give self shielding factors within an energy group. A model of the twopoints histogram is shown in **Fig. 2**.

The total, absorption and fission cross sections are represented by the histogram. Various parameters in **Fig. 2** are determined so as to represent the self-shielding factor due to the histogram cross section as close as possible to that due to the cross section $\sigma_x(u)$ in **Fig. 2**. Detailed procedure to determine these parameters in **Fig. 2** is given in the reference[4].

The histogram representation makes the cell calculation by the collision probability method very simple as shown in this section. Since various quantities based on the two-points histogram representation are used in this section those definitions and notations are given first as follows ;

$\tilde{\sigma}_x(K,l,m)$: the self-shielded group cross section of the reaction type X of the material m in the region l at the K-th energy group,

$\sigma_x(m,E)$: the cross section of the type X of the material m at the energy E,

$\sigma_{tot}(m,E)$: the total cross section of the material m at the energy E,

$\bar{\sigma}_x(K,m)$: the group cross section of the type X or total of the material m at the K-th

$\bar{\sigma}_{tot}(K,m)$ energy group in infinitely diluted media,

$W_1(K,m)$: the histogram width for the cross section of the material m at the K-th

$W_2(K,m)$ energy group (see **Fig. 2**),

$\varepsilon_{x1}(K,m)$: the histogram height of cross section of the type X of the material m at the

$\varepsilon_{x2}(K,m)$ K-th energy group (see **Fig. 2**),

$\varepsilon_{t1}(K,m)$: the histogram height of the total cross section of the material m at the K-th

$\varepsilon_{t2}(K,m)$ energy group (see **Fig. 2**),

$\phi(l,E)$: the volume integrated neutron flux at the energy E in the region l,

$Q(l,E)$: the neutron source given at the energy E in the region l,

$P(l\rightarrow j,E)$: the collision probability relating the neutron collision in the j-th region to an isotropic neutron source of the energy E given in the l-th region,

$A(l,m)$: the atom density of the material m in the region l,

E_K, E_{K-1} : the upper and lower boundary energies of the K-th energy group,

J : the number of regions in a cell.

In a multi-region configuration, the effective group cross section $\tilde{\sigma}_x(K, l, m)$ is calculated relating to the neutron flux $\phi(l, E)$ as follows,

$$\tilde{\sigma}_x(K, l, m)=\int_{E_{k-1}}^{E_k}\sigma_x(m, E)\phi(l, E)dE\Big/\int_{E_{k-1}}^{E_k}\phi(l, E)dE \tag{3-1}$$

Using the collision probability method, the neutron flux in the multi-region cell is written as

$$\phi(l, E)=\sum_{j=1}^{J}Q(j, E)P(l\to j, E)/\Sigma_{tot}(j, E), \tag{3-2}$$

where $\Sigma_{tot}(j, E)$ is the total macroscopic cross section in the j-th region at the energy E,

Hence, combining Eqs. (3-1) and (3-2),

$$\tilde{\sigma}_x(K, l, m)=\int_{E_{k-1}}^{E_k}\sigma_x(m, E)\sum_{j=1}^{J}\frac{Q(j, E)P(l\to j, E)}{\Sigma_{tot}(j, E)}dE\Big/\int_{E_{k-1}}^{E_{kJ}}\sum_{j=1}^{J}\frac{Q(j, E)P(l\to j, E)}{\Sigma_{tot}(j, E)}dE. \tag{3-3}$$

The form given in Eq. (3-3) suggests that the method used for obtaining the self-shielded cross section in a homogeneous medium may be applied to a multi-region cell by making a correction to the collision probability.

Let us designate a pseudo-potential cross section σ_0 as defined in the ABBN set, which is obtained the sum of the total cross sections of all other materials than the material m. Thus the total macroscopic cross section in the region j is denoted by

$$\Sigma_{tot}(j, E)=A(j, m)[\sigma_0(j, m, E)+\sigma_{tot}(m, E)]. \tag{3-4}$$

When the source term $Q(j, E)$ is almost constant in the energy group K, $Q(j, E)$ can be replaced with $Q(j, K)$. Using the two point histogram representation of cross sections[4] as schematically shown in **Fig. 2**, Eq. (3-3) is rewritten as

$$\begin{aligned}\tilde{\sigma}_x(K, l, m)&=\frac{\sum_{j=1}^{J}\frac{Q(j, K)}{A(j, m)}\left[\int_{EM}^{E_k}\frac{\sigma_x P(l\to j, E)}{\sigma_0+\sigma_{tot}}dE+\int_{E_{k-1}}^{EM}\frac{\sigma_x P(l\to j, E)}{\sigma_0+\sigma_{tot}}dE\right]}{\sum_{j=1}^{J}\frac{Q(j, K)}{A(j, m)}\left[\int_{EM}^{E_k}\frac{P(l\to j, E)}{\sigma_0+\sigma_{tot}}dE+\int_{E_{k-1}}^{EM}\frac{P(l\to j, E)}{\sigma_0+\sigma_{tot}}dE\right]}\\&=\frac{\sum_{j=1}^{J}\frac{Q(j, K)}{A(j, m)}\sum_{n=1,2}\frac{W_n(K, m)\bar{\sigma}_x(K, m)(1+\varepsilon_{xn}(K, m))}{\sigma_0(K, j, m)+\bar{\sigma}_{tot}(K, m)(1+\varepsilon_{tn}(K, m))}P_n(l\to j, K, m)}{\sum_{j=1}^{J}\frac{Q(j, K)}{A(j, m)}\sum_{n=1,2}\frac{W_n(K, m)}{\sigma_0(K, j, m)+\bar{\sigma}_{tot}(K, m)(1+\varepsilon_{tn}(K, m))}P_n(l\to j, K, m)}.\end{aligned} \tag{3-5}$$

The collision probability $P_n(l\to j, K, m)$ in the histogram representation is obtained in the reference[4] as follows,

$$\begin{aligned}P_n(l\to j, K, m)=P(l\to j, K)\Bigg[1+&\frac{\varepsilon_{tn}(K, m)\bar{\sigma}_{tot}(K, m)}{\sigma_0(K, j, m)+\bar{\sigma}_{tot}(K, m)}\\&-\sum_{j=1}^{J}\frac{\varepsilon_{tn}(K, m)\bar{\sigma}_{tot}(K, m)P(l\to j, K)}{\sigma_0(K, i, m)+\bar{\sigma}_{tot}(K, m)}\Bigg],\end{aligned} \tag{3-6}$$

where $P(l\to j, K)$ is the collision probability calculated using unshielded cross sections $\bar{\sigma}_{tot}$ of all materials.

Since the second and third terms on the right hand side of Eq. (3-6) are correction terms due to the histogram representation and are considered to be small compared with unity, we assume the following relation between P_n and P instead of Eq. (3-6) in order to simplify the subsequent formulae to be introduced,

$$P_n(l\to j, K, m)\doteqdot\frac{\frac{\sigma_0(K, j, m)+\bar{\sigma}_{tot}(K, m)(1+\varepsilon_{tn}(K, m))}{\sigma_0(K, j, m)+\bar{\sigma}_{tot}(K, m)}}{1+\sum_{i=1}^{J}\frac{\varepsilon_{tn}(K, m)\bar{\sigma}_{tot}(K, m)}{\sigma_0(K, i, m)+\bar{\sigma}_{tot}(K, m)}P(l\to i, K)}\cdot P(l\to i, K). \tag{3-7}$$

In this approximation, the relation of $\sum_{j=1}^{J}P_n(l\to j, K, m)=1$ is still maintained.

Using Eq. (3-7), Eq. (3-5) can be rewritten as follows,

$$\tilde{\sigma}_x(K,l,m)=\bar{\sigma}_x(K,m)\left(1+\frac{\sum_{j=1}^{J}\frac{Q(j,K)}{A(j,m)}\sum_{n=1,2}\frac{\frac{W_n(K,m)\varepsilon_{xn}(K,m)}{\sigma_o(K,j,m)+\bar{\sigma}_{tot}(K,m)}P(l\to j,K)}{1+\sum_{i=1}^{J}\frac{\varepsilon_{tn}(K,m)\bar{\sigma}_{tot}(K,m)}{\sigma_o(K,i,m)+\bar{\sigma}_{tot}(K,m)}P(l\to i,K)}}{\sum_{j=1}^{J}\frac{Q(j,K)}{A(j,m)}\sum_{n=1,2}\frac{\frac{W_n(K,m)}{\sigma_o(K,j,m)+\bar{\sigma}_{tot}(K,m)}P(l\to j,K)}{1+\sum_{i=1}^{J}\frac{\varepsilon_{tn}(K,m)\bar{\sigma}_{tot}(K,m)}{\sigma_o(K,i,m)+\bar{\sigma}_{tot}(K,m)}P(l\to i,K)}}\right) \tag{3-8}$$

$$=\bar{\sigma}_x(K,m)\left(1+\frac{\sum_{n=1,2}\left(\frac{W_n(K,m)\varepsilon_{xn}(K,m)}{\sum_{i=1}^{J}\frac{\varepsilon_{tn}(K,m)\bar{\sigma}_{tot}(K,m)P(l\to i,K)}{\sigma_o(K,i,m)+\bar{\sigma}_{tot}(K,m)}+1}\right)}{\sum_{n=1,2}\left(\frac{W_n}{\sum_{i=1}^{J}\frac{\varepsilon_{tn}(K,m)\bar{\sigma}_{tot}(K,m)P(l\to i,K)}{\sigma_o(K,i,m)+\bar{\sigma}_{tot}(K,m)}+1}\right)}\right). \tag{3-9}$$

In the homogeneous case, put $J=1$ and $P=1$, then,

$$\tilde{\sigma}_x(K,1,m)=\bar{\sigma}_x(K,m)\left(\frac{\sum_{n=1,2}\frac{W_n\bar{\sigma}_x(K,m)(1+\varepsilon_{xn}(K,m))}{\sigma_o(K,1,m)+\bar{\sigma}_{tot}(K,m)(1+\varepsilon_{tn}(K,m))}}{\sum_{n=1,2}\frac{W_n}{\sigma_o(K,1,m)+\bar{\sigma}_{tot}(K,m)(1+\varepsilon_{tn}(K,m))}}\right), \tag{3-10}$$

which can be obtained also from the definition of the histogram representation.

In order to obtain σ_o in Eq. (3-9) or (3-10) accurately, an iteration procedure is recommended in the reference 4). In the code SP-2000, Ep. (3-9) or (3-10) is used, but no iteration procedure is incorporated for obtaining the value of σ_o because of the use of a fairly narrow group structure consisting of 1950 energy groups.

3. 2 Slowing Down Due to Elastic Scattering

It is assumed that the angular dependence of the elastic scattering cross section $\sigma(E_0,\mu)$ in the center-of-mass system is represented in the form

$$\sigma(E_0,\mu)=\sigma_s(E_0)\sum_{l=0}^{L}\frac{2l+1}{2}\cdot B_l(E_0)\cdot P_l(\mu), \tag{3-11}$$

where

$$\sigma_s(E_0)=\int_{-1}^{1}\sigma(E_0,\mu)d\mu,$$

μ: cosine of the scattering angle in the center of mass system,

$P_l(\mu)$: l-th Legendre polynomial,

$B_l(E_0)$: coefficient of expansion ($B_0(E_0)=1.$),

The fractional energy change per collision is related to μ,

$$\mu=\beta+\delta\frac{E}{E_0}, \tag{3-12}$$

where

$$\delta=\frac{(A+1)^2}{2A}$$

$$\beta=1-\delta,$$

A: mass number of target nucleus,

E_0: incident neutron energy,

E: scattered neutron energy.

Then, the probability $P_{er}[(\varepsilon_1,\varepsilon_2),(E_1,E_2)]$ that the neutrons suffering collision with energies between ε_1 and ε_2, are transferred into energies between E_1 and E_2 is given by,

$$P_{er}[(\varepsilon_1, \varepsilon_2), (E_1, E_2)] = \frac{1}{T}\int_{\varepsilon_2}^{\varepsilon_1} dE_0 \sigma_s(E_0) \sum_{l=0}^{L} \frac{2l+1}{2} B_l(E_0) \int_{\beta+\delta\frac{E_2}{E_0}}^{\beta+\delta\frac{E_1}{E_0}} P_l(\mu) d\mu, \tag{3-13}$$

where

$$T = \int_{\varepsilon_2}^{\varepsilon_1}\int_{-1}^{1} \sigma(E_0, \mu) d\mu dE_0. \tag{3-14}$$

If the energy ε_2 is very close to ε_1, the values of $\sigma_s(E_0)$ and $B_l(E_0)$ in the energy region between ε_1 and ε_2 can be assumed to be constant and equal to $\sigma_s(\bar{\varepsilon})$ and $B_l(\bar{\varepsilon})$ where $\bar{\varepsilon}=(\varepsilon_1+\varepsilon_2)/2$. Thus,

$$P_{er}[(\varepsilon_1, \varepsilon_2), (E_1, E_2)] = \sum_{l=0}^{L} \frac{2l+1}{2} B_l(\bar{\varepsilon}) \sum_{j=0}^{l} \frac{a(l)_j}{j+1} \delta^{j+1} \cdot \frac{2}{\varepsilon_1-\varepsilon_2}\left[\int_{\varepsilon_2}^{\varepsilon_1}\left(\alpha+\frac{E_1}{E_0}\right)^{j+1} dE_0 - \int_{\varepsilon_2}^{\varepsilon_1}\left(\alpha+\frac{E_2}{E_0}\right)^{j+1} dE_0\right], \tag{3-15}$$

where $\alpha=\beta/\delta$ and the l-th Legendre polynomial is replaced with $P_l(\mu)=\sum_{j=0}^{l} a(l)_j \mu^j$.

When a group structure with the equi-lethargy width of u, as shown in **Fig. 3**, is used the group transfer probability due to elastic scattering $P_{er}(1\to g)$ is expressed by

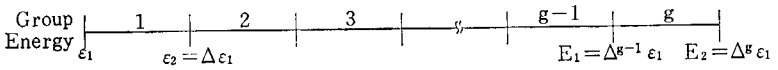

Fig. 3 Energy group structure.

$$P_{er}[1\to g] = P_{er}[(\varepsilon_1, \varepsilon_2), (E_1, E_2)] = \sum_{l=0}^{L} \frac{2l+1}{2} B_l(1) \sum_{j=0}^{l} \frac{a(l)_j}{(j+1)} \cdot \frac{\delta^{j+1}}{1-\Delta} \cdot H, \tag{3-16}$$

where Δ and H are obtained by performing the integrals in Eq. (3-15), as follows;

$$\begin{aligned}
H &= [Q(1, \Delta, g-1, j+1) - Q(1, \Delta, g, j+1)], \text{ for } 1<g<G-1,\\
&= [Q(1, \Delta^{G-1}/S, g-1, j+1) - R(1, \Delta^{G-1}/S, S, j+1)\\
&\quad + Q(\Delta^{G-1}/S, \Delta, g-1, j+1) - Q(\Delta^{G-1}/S, \Delta, g, j+1)], \text{ for } g=G-1,\\
&= [Q(\Delta^{G-1}/S, \Delta, g-1, j+1) - R(\Delta^{G-1}/S, \Delta, S, j+1)], \text{ for } g=G,\\
&= [R(1, \Delta, 1, j+1) - Q(1, \Delta, g, j+1)], \text{ for } g=1,\\
\Delta &= \exp(-u),\\
S &= \frac{(A-1)^2}{(A+1)^2},
\end{aligned}$$

G: the final energy group into which neutrons are scattered,

$$Q(f_1, f_2, g, N) \equiv \xi(N) \cdot (f_1-f_2) + \nu(N) \cdot \Delta^g \ln(f_1/f_2) - \sum_{i=2}^{N} \frac{\Delta^{(gi)} \eta(N)_i}{i-1}\left(\frac{1}{f_1^{i-1}} - \frac{1}{f_2^{i-1}}\right), \tag{3-17}$$

$$R(f_1, f_2, S, N) \equiv (f_1-f_2)(\alpha+S)^N, \tag{3-18}$$

and $\xi(N)$, $\nu(N)$ and $\eta(N)_i$ are the coefficients in the following relation,

$$(\alpha+Y)^N \equiv \xi(N) + \nu(N) \cdot Y + \sum_{i=2}^{N} \eta(N)_i Y^i.$$

As shown in Eq. (3-16), (3-17) and (3-18), if a group structure with equi-lethargy width is used, the group transfer probability depends on the group lethargy width and the target nucleus.

In the code SP-2000, therefore, the values of

$$\sum_{j=0}^{l} \frac{a(l)_j}{j+1} \frac{\delta^{j+1}}{1-\Delta} \cdot H$$

for each l and for each nuclide are given as built-in data. The maximum number of l provided in the current version of SP-2000 is 7.

3. 3 Slowing Down Due to Inelastic Scattering

The fine group inelastic scattering matrices are not provided on the AGLI/B, because the number of their elements is very vast, more than a few millions. The recurrence relation recommended in the reference[4] is used to calculate neutron sources due to inelastic scattering.

3. 3. 1 Group transfer cross section due to discrete level

Relations among the incident neutron energy E_i, the final neutron energy E_f, the excitation energy ω_n of the n-th level and the cosine of scattering angle in the center of mass system μ_c are given as follows[8)~10)],

$$\frac{E_f}{E_i}=\frac{2}{(A+1)^2}\frac{\mu_c}{W}+\frac{1}{(A+1)^2}\left(1+\frac{1}{W^2}\right), \tag{3-19}$$

where A is the mass number of the target nucleus and

$$W^2=\frac{1}{A^2-A(A+1)\omega_n/E_i}. \tag{3-20}$$

Thus the maximum and minimum energies of scattered neutron E_{max} and E_{min} are given by

$$E_{max}=\frac{A^2+1}{(A+1)^2}E_i-\frac{A\omega_n}{A+1}+\frac{2E_i}{(A+1)^2}\sqrt{A^2-A(A+1)\omega_n/E_i}, \tag{3-21}$$

$$E_{min}=\frac{A^2+1}{(A+1)^2}E_i-\frac{A\omega_n}{A+1}-\frac{2E_i}{(A+1)^2}\sqrt{A^2-A(A+1)\omega_n/E_i}. \tag{3-22}$$

Since the scattering is assumed to be spherically symmetric in the center of mass system in the code SP-2000, the energy transfer cross section from the energy E_i to E_f is expressed by

$$\sigma(E_i\rightarrow E_f) \quad =\sigma(E_i)\frac{1}{E_{max}-E_{min}}, \tag{3-23}$$

where

$$\sigma(E_i)=\int_{E_{min}}^{E_{max}}\sigma(E_i\rightarrow E_f)dE_f. \tag{3-24}$$

The energy transfer probability $P_{ir}(l\rightarrow j)$ from the l-th group to the j-th group is thus given by

$$P_{ir}(l\rightarrow j)=\frac{1-\exp(-\Delta u)}{1-\exp(-(m_l-K_l)\Delta u)}\exp(-(j_l-K_l)\Delta u), \tag{3-25}$$

where

$$K_l=\frac{\ln((E_l+E_{l-1})/2E_{max})}{\Delta u}, \tag{3-26}$$

$$m_l=\frac{\ln((E_l+E_{l-1})/2E_{min})}{\Delta u}, \tag{3-27}$$

$$j_l=\frac{\ln((E_l+E_{l-1})/2E_j)}{\Delta u}, \tag{3-28}$$

E_l : upper energy of the l-th group,

Δu : lethargy width of energy group.

The transfer probability to the energy group including the highest energy of scattered neutron E_{max} is

$$\frac{1-\exp(-\Delta u)}{1-\exp(-(m_i-k_i)\Delta u)}. \tag{3-29}$$

Hence based on the relation of Eq. (3-23) the transfer probability to the lower group can be easily obtained by multiplying the quantity (3-29) by a factor of $R=e^{-\Delta u}$ successively.

3. 3. 2 Group transfer cross section due to the continuum region of levels

For neutrons slowing down due to the continuum levels, the evaporation model gives

$$N(E_f)=b(E_i)E_f\exp(-E_f/T_{c,i}), \tag{3-30}$$

where

$N(E_f)$: the probability that the energy of the scattered neutron will be in a unit interval at E_f,

$T_{c,i}$: temperature of the residual nucleus,

$b(E_i)$: normalization factor.

As shown in the reference[4], the energy transfer probability $P_{ir}(l\to j)$ is represented by means of the recurrence procedure as follows,

$$P_{ir}(l\to j+1)=P_{ir}(l\to j)\frac{E_{j+1}(E_j-E_{j+1})}{E_j(E_{j-1}-E_j)}\exp((E_{j-1}-E_j)/T_{c,i}), \tag{3-31}$$

$$=P(l\to j)\cdot f_{i,j}, \tag{3-32}$$

where

$$f_{i,j}=\frac{1+(E_{j-1}-E_j)/2T_{c,i}}{1-(E_{j-1}-E_j)/2T_{c,i}}R^2. \tag{3-33}$$

The nuclear temperature used in the code SP-2000 is assumed to be

$$T_{c,i}=a\sqrt{E_i}, \tag{3-34}$$

where

$$a=2/(0.62\sqrt{A}). \tag{3-35}$$

3. 4 Data Given by Chebyshev Expansion

The number of neutrons per fission ν, the cosine of the scattering angle in the center of mass system μ_c and that in the laboratory system μ_l are expressed by the Chebyshev polynomial with respect to the lethargy. Using the Chebyshev polynomial T_n, an arbitrary function $\varphi(u)$ of the lethargy between $u=A$ and $u=B$ is expanded as

$$\varphi(u)\fallingdotseq\sum_{i=0}^{N}a_iT_i(x), \tag{3-36}$$

where

$$u=\frac{1}{2}[x\cdot(B-A)+(A+B)], \tag{3-37}$$

$$a_0=\frac{1}{N}\sum_{i=1}^{N}\varphi(u_i), \tag{3-38}$$

$$a_K=\frac{2}{N}\sum_{i=1}^{N}\varphi(u_i)T_K(x_i), \tag{3-39}$$

x_i : roots of $T_N(x)$, (3-40)

u_i : values of u corresponding to x_i in Eq. (3-37). (3-41)

Rewriting Eq. (3-36), the $\varphi(u)$ is denoted in the form of a power series of x,

$$\varphi(u)\fallingdotseq\sum_{i=0}^{N-1}b_ix^i, \tag{3-42}$$

where

$$x=[u-(A+B)/2]/[(B-A)/2]. \tag{3-43}$$

Using the values of b_i's given from the AGLI cross section library, ν, μ_l and μ_c in each fine energy group are calculated by Eq. (3-42) and (3-43), where the median value of the lethargy in each fine group is used in Eq. (3-43). In the outside of $A\sim B$, the constant values which are also given from the cross section library are used respectively for ν, μ_l and μ_c.

3. 5 Fission Neutron Spectrum

A simple Maxwellian distribution is assumed for obtaining fission neutron spectrum $\chi(E)$ as

$$\chi(E)=\frac{2}{\sqrt{\pi}}\left(\frac{E}{T}\right)^{1/2}\frac{1}{T}\exp\left(-\frac{E}{T}\right), \tag{3-44}$$

where T means the nuclear temperature. Then, in the multigroup representation, the fission neutron spectrum χ_g in the g-th group is given by

$$\chi_g=\int_{E_g}^{E_{g-1}}\chi(E)dE=\frac{2}{\sqrt{\pi}}\int_{E_g/T}^{E_{g-1}/T}\sqrt{X}\exp(-X)dX \tag{3-45}$$

$$=\frac{2}{\sqrt{\pi}}\left[\sqrt{E_g/T}\exp(-E_g/T)-\sqrt{E_{g-1}/T}\exp(-E_{g-1}/T)\right]$$
$$+\mathrm{erf}\left(\sqrt{E_{g-1}/T}\right)-\mathrm{erf}\left(\sqrt{E_g/T}\right).$$

4. Calculation of Neutron Flux

Using the notations given in the section 2. 2, the neutron flux in the i-th fine group in the region n is expressed as,

$$F_{n,i}=\sum_{K=1}^{N}T_{n,K,i}S_{K,i}+\sum_{K=1}^{N}T_{n,K,i}=\sum_{j=1}^{I}P_{k,j,i}F_{K,j}. \tag{4-1}$$

Ignoring up-scattering, the above equation is rewritten as

$$\sum_{K=1}^{N}\left[\delta_{n,K}-T_{n,K,i}P_{K,i,j}\right]F_{K,i}=\sum_{K=1}^{N}T_{n,K,i}Q_{K,i}, \tag{4-2}$$

where

$$\delta_{n,K}=1,\ \text{for}\ n=K,$$
$$=0,\ \text{for}\ n\neq K,$$

$$Q_{K,i}=S_{K,i}+\sum_{j=1}^{i-1}P_{K,j,i}F_{K,j}. \tag{4-3}$$

Since the fission source distribution has already been calculated in the coarse group model as mentioned in the section 2. 2, Eq. (4-2) can be solved successively from the first energy group. The elastic and inelastic scattering source relating to $P_{K,i,j}$ is calculated as shown in the sections 3. 2 and 3. 3. The method for calculating the collision probability $T_{n,K,j}$ used in the SP-2000 is the same as used in the subroutine PATH in the computer program LAMP[11].

In the SP-2000, the collision probability $T_{n,K,j}$ and the group transfer probability $P_{K,i,j}$ are calculated using the self-shielded fine group cross sections defined in the section 3. 1. The influence of neutron leakage on the neutron spectrum is taken into consideration by adding DB^2 to the macroscopic total and absorption cross sections, where D is the diffusion constant in an equivalent homogeneous medium of the cell.

The cell-averaged neutron spectrum defined by $\sum_{K=1}^{N}F_{K,i}/V$ where V is the cell volume, can be stored on a data disk and also can be plotted graphically by GPLOT1[12] as shown in **Fig. 4.**

5. Calculation of Macroscopic Broad Group Constants

Using the fine group neutron spectrum in a multi-region cell as a weighting function, the SP-2000 can also generate various sorts of effective macroscopic broad group constants which are used as cross section input to many computer codes for integral data analysis. The code also calculates various types of reaction rates.

5. 1 Effective Cell-Averaged Broad Group Constants

The broad group cell-averaged cross sections are obtained using space-dependent neutron spectrum as a weighting function. The total $\sigma_t(K)$, fission $\sigma_f(K)$, capture $\sigma_c(K)$, elastic $\sigma_e(K)$ and inelastic scattering $\sigma_{in}(K)$ cross sections at the K-th broad group are obtained using the relation of

$$\sigma_X(K)=\sum_{j\in K}\sum_{m=1}^{M}\sum_{l=1}^{L}\tilde{\sigma}_X(j,l,m)\cdot A(l,m)\cdot\phi(l,j)/\sum_{j\in K}\sum_{l=1}^{L}\phi(l,j), \tag{5-1}$$

where $\phi(l,j)$ describes the fine group neutron spectrum at the space mesh l and at the fine group j. Since the elastic scattering cross section is not stored on the cross section library AGLI, the cross section $\sigma_e(K)$ is obtained by subtracting the non-elastic scattering cross section from the total cross section.

The average number of neutrons per fission $\nu(K)$ at the K-th group is given by

$$\nu(K)=\frac{\sum_{j\in K}\sum_{l=1}^{L}\sum_{m=1}^{M}\nu(j,m)\cdot A(l,m)\cdot\tilde{\sigma}_f(j,l,m)\cdot\phi(l,j)}{\sum_{j\in K}\sum_{l=1}^{L}\sum_{m=1}^{M}A(l,m)\cdot\tilde{\sigma}_f(j,l,m)\cdot\phi(l,j)} \tag{5-2}$$

where $\nu(j,m)$ is given by the Chebyshev polynomial expansion as mentioned in the section 3. 4.

Using the probability $P_{ir}(j\to i)$ obtained in the section 3. 3, the broad group inelastic scattering matrices including a correction of the $(n,2n)$ reaction are calculated by

$$\sigma_{in}(K1\to K2)=\sum_{j\in K1}\sum_{l=1}^{L}\sum_{m=1}^{M}\sum_{i\in K2}\{P_c(j\to i)+\sum_{n=1}^{N}P_n(j\to i)\}\cdot\tilde{\sigma}_{in}(j,l,m)\cdot A(l,m)\cdot C(j,m)\cdot\phi(l,j)/\sum_{j\in K1}\sum_{l=1}^{L}\phi(l,j), \tag{5-3}$$

where

$\sigma_{in}(K1\to K2)$: broad group inelastic scattering matrix from the $K1$-th group to the $K2$-th group,

$P_c(j\to i)$: the fine group energy transfer probability due to the continuum region of levels,

$P_n(j\to i)$: the fine group energy transfer probability due to the n-th discrete level,

$C(j,m)$: correction term due to the $(n,2n)$ reaction of the m-th material at the j-th fine group.

The correction terms $C(j,m)$ are kept constant for the fine groups included in each group in the AGLI/70 standard group structure, and are given by the block data in SP-2000.

Using the probability $P_{er}(1\to g)$ obtained in the section 3. 2, the broad group elastic transfer matrices $\sigma_e(K1\to K2)$ are calculated by

$$\sigma_e(K1\to K2)=\sum_{j\in K1}\sum_{l}\sum_{m}\sum_{j+i-1\in K2}\tilde{\sigma}_e(j,l,m)\cdot P_{er}(1\to i,m)\cdot A(l,m)\cdot\phi(l,j)/\sum_{j\in K1}\sum_{l}\phi(l,j), \tag{5-4}$$

where

$\sigma_e(K1\to K2)$: broad group elastic scattering matrix from the $K1$-th group to the $K2$-th group,

$P_{er}(1\to i,m)$: transfer probability of the m-th material given in the section 3. 2.

The broad group transport cross section σ_{tr} relating to the diffusion constant is collapsed by

$$\sigma_{tr}(K)=\frac{\sum_{i\in K}\sum_{l}\phi(l,i)}{\sum_{i\in K}\sum_{m}\sum_{l}\{\tilde{\sigma}_t(i,l,m)-\tilde{\sigma}_e(i,l,m)\cdot\mu_l(i,m)\}A(l,m)\cdot\phi(l,i)/\sum_{l}\phi(l,i)}, \tag{5-5}$$

where $\mu_l(i,m)$ is given by the Chebyshev polynomial expansion as mentioned in the section 3. 4.

These broad group cross sections can be stored on the data pool provided in the DOYC system by a control option.

5. 2 Region-Dependent Broad Group Constants and Reaction Rates

For the reaction rate analysis, the fission and capture cross section are calculated for each material in each region according to the following formulae :

$$\sigma_f(K,l,m)=\sum_{i\in K}\tilde{\sigma}_f(i,l,m)\phi(l,m)/\sum_{i\in K}\sum_{l}\phi(l,i), \tag{5-6}$$

$$\sigma_c(K,l,m)=\sum_{i\in K}\tilde{\sigma}_c(i,l,m)\phi(l,m)/\sum_{i\in K}\sum_{l}\phi(l,i). \tag{5-7}$$

These broad group cross sections can be stored on the data pool provided in the DOYC system by a control option.

The following two types of the fission ($X=f$) and capture ($X=c$) reaction rates are calculated for all materials in the medium,

$$RC_x(m)=\sum_i\sum_l\tilde{\sigma}_x(i,l,m)\phi(l,i), \tag{5-8}$$

$$RF_x(m)=\sum_i\sum_l\tilde{\sigma}_x(i,l,m)\cdot A(l,m)\cdot\phi(l,i), \tag{5-9}$$

and the following quantity relating to the neutron fission yield is calculated,

$$RNF_f(m)=\sum_i\sum_l\tilde{\sigma}_f(i,l,m)\cdot\nu(i,m)\cdot\phi(l,i)\cdot A(l,m). \tag{5-10}$$

5. 3 Broad Group Fission Neutron Spectrum

The broad group fission neutron spectrum is recalculated in the SP-2000. The data of nuclear temperature used for fission spectrum calculation are tabulated as a function of the average number of neutrons per fission ν. The average number of neutrons per fission of the m-th material $\nu(m)$ is calculated in the SP-2000 by

$$\bar{\nu}(m)=\sum_l\sum_m\sum_i\tilde{\sigma}_f(i,l,m)\cdot\nu(i,m)\cdot\phi(l,i)\cdot A(l,m)/\sum_l\sum_m\sum_i\tilde{\sigma}_f(i,l,m)\cdot\phi(l,i)\cdot A(l,m). \tag{5-11}$$

Then, the nuclear temperature of the m-th material $\theta(m)$ at the above value of $\nu(m)$ is obtained by interpolation of the tabulated data given on the AGLI library, and the broad group fission neutron spectrum of the m-th material at the g-th group $\chi_g(m)$ is calculated using Eq. (3-45). Hence the effective broad group fission spectrum χ_g is obtained by

$$\bar{\chi}_g=\sum_m\chi_g(m)\cdot RF_f(m)/\sum_m RF_f(m). \tag{5-12}$$

This fission neutron spectrum can be stored on the data pool together with the cell-averaged cross sections.

6. Input

All of fine group cross section data are taken from the library AGLI stored on data disks or tapes. Since the 1950 energy group cross section library AGLI was reported elsewhere[4], only brief descriptions on the library are given in Appendix.

The code SP-2000 can be used not only as an element code in the DOYC system, but also as a stand-alone code. The card input instructions according to the DOYC general input form are given in the following table ;

Card input instruction

Card set No.	FORTRAN symbol	Format	Definition
1-1*	USER(2)	(2A4, I4)	User's name
	NPAY		User's pay roll number
1-2*	NCARD	(1I3)	The number of the following (1-3) cards
1-3*	IP(10)		
2-1	TLL(20)	(20A4)	Title card
2-2	DATI(3)	(3A4, E12.5, 6I6)	Data identification title on the MXS data pool**
	VOID		(1. -void fraction) in unit cell
	NREG		The number of regions in unit cell
	NMAT		The number of materials in unit cell
	NDATN		Data specification number of broad group cross sections, when the MXS pool is used
	NPLIO		=0, the MXS pool is not used =1, both cell-averaged and region dependent, broad group cross sections are stored on the MXS pool =2, only cell averaged ones are stored
2-3	MCODE(NMAT)	(12I6)	The code numbers of materials in unit cell
2-4	THIK(NREG)	(7E10.4)	Region thickness in cm
2-5	NMESH(NREG)	(12I6)	The number of space-mesh points in each region
2-6	TEP(NREG)	(7E10.4)	The absolute temperature in each region
2-7	AD(NMAT, 1)	(7E10.4)	Atom density in the 1st region
	AD(NMAT, 2)		Atom density in the 2nd region
	⋮		⋮
	AD(NMAT, NREG)		Atom density in the last region
3-1	TTL(20)	(20A4)	Comment card
3-2	NBOUND	(12I6)	=0, symmetric boundary condition =1, periodic boundary condition =2, for future use
	NGNO		The number of fine energy groups (=1950 when AGLI library is used)
	MCV		The number of Chebyshev expansion terms
	NPAS		For future use
	NBR		The number of broad energy groups
	OP2N		For future use
	IBCL		=0, the material buckling is calculated in the SP-2000, otherwise read from card input
3-3	NBRO(NBR)	(12I6)	The largest group number in each broad group
3-4	NTAP2		=8, the spatial fission source distribution is calculated in the SP-2000 =5, the distribution is given by card input
3-5	E1	(6E12.5)	The upper energy of the 1st fine group
	DELU		The lethargy width of each group
	TEMC		The nuclear temperature in MeV for fission spectrum calculation
	BUCL		The material buckling, when IBCL≠0
3-6	SPFS(NMESH)	(6E12.5)	The spatial distribution of fission source, if NTAP2 ≠5, skip (3-6)

* The DOYC system option control input

** Data pool disk provided in the DOYC system

7. Output

The line printer output, graphic output and disk/tape output are available as calculated results from the SP-2000.

The printer output includes;

(1) Summary of the card input data.
(2) Material buckling.
(3) Spatial distribution of fission source.
(4) Fine group space dependent neutron spectrum (on option).
(5) Cell-averaged neutron spectrum.
(6) Broad group structure.
(7) Broad group cell-averaged cross sections; total, fission, inelastic scattering, capture, elastic scattering, diffusion coefficient and fission yield ($\nu\sigma_f$).
(8) Broad group inelastic transfer cross sections.
(9) Elastic transfer cross sections.
(10) Average angle of elastic scattering in the laboratory system.
(11) Material-and-region dependent broad group fission and capture cross sections (on option).
(12) Information on fission; average number of neutrons per fission of each fissile nuclide, average nuclear temperature of each nuclide used for the calculation of fission spectrum.
(13) Broad group fission neutron spectrum.
(14) Fission and capture reaction rates of each nuclide.

An example of the print output is given in TABLE 1.

TABLE 1 Print output

```
MODULAR CODE SYSTEM      **** DOYC ****
MODULAR CODE SYSTEM      **** DOYC ****
MODULAR CODE SYSTEM      **** DOYC ****

USER NAME          *****.*****          PAY ROLL NO.    ******

ELEMENT CODE USED

HUMPTHUMP
SP-2000


RESULTS OBTAINED BY SP-2000



FCA VI-1 TEST ZONE HETERO , STANDARD (AGLI/ON=AGLI/0)
FCA VI-1 TEST ZONE HETERO , STANDARD (AGLI/ON=AGLI/0)
FCA VI-1 TEST ZONE HETERO , STANDARD (AGLI/ON=AGLI/0)

IDENT=NO.      VI-1HSTDON=0


  REGION= 7    NUCLIDE=11   BCUN-CON= 0   EN-GROUP=1950   CHB-EXP(ELAST)=12

CODE NO. AND DENSITY

                939          940          941          935          928          8            11           24           26           28           13
REGION= 1  0.0          0.0          0.0          0.0          0.0          0.0          0.0          1.655E-02 6.343E-02 6.288E-03 0.0
REGION= 2  0.0          0.0          0.0          0.0          0.0          0.0          1.817E-02 3.128E-03 1.113E-02 1.853E-03 0.0
REGION= 3  1.986E-02 1.772E-03 2.025E-04 0.0          0.0          0.0          0.0          5.802E-03 1.905E-02 2.321E-03 2.923E-03
REGION= 4  0.0          0.0          0.0          4.797E-05 2.185E-02 4.400E-02 0.0          0.0          0.0          0.0          0.0
REGION= 5  0.0          0.0          0.0          0.0          0.0          0.0          1.817E-02 3.128E-03 1.113E-02 1.853E-03 0.0
REGION= 6  0.0          0.0          0.0          3.840E-05 1.748E-02 4.806E-02 0.0          0.0          0.0          0.0          8.531E-03
REGION= 7  1.986E-02 1.772E-03 2.025E-04 0.0          0.0          0.0          0.0          5.802E-03 1.905E-02 2.321E-03 2.923E-03


             THICKNESS    MESH NO.
REGION= 1    0.3235E 00     1
REGION= 2    0.6350E 00     1
REGION= 3    0.1588E 00     1
REGION= 4    0.6350E 00     1
REGION= 5    0.6350E 00     1
REGION= 6    0.3969E 00     1
REGION= 7    0.7938E-01     1

** (N,2N) CORRECTION IS CONSIDERED ... OP2N = 0
```

** BUCKLING ... 0.26848E-02 (Material buckling of homogeneous medium, see Section-2-1.)

INITIAL ENERGY 0.10500E 08
DELTA-U 0.85000E-02
NUCLEAR TEMPERATU 0.14000E 01

(Spatial distribution of fission source, see Section-2-2.)

FISSION SOURCE DIST
0.0 0.0 0.6032E 00 0.8122E-01 0.0 0.4112E-01 0.3014E 00

AGLI-LIBRARY FILE-A AGLI/0 /JAN./1971/ NCHEB=12
TAPE NO.01112

AGLI-LIBRARY FILE-B AGLI/1 /APR./1972/
TAPE NO.01514

** R-FACTOR = 0.991536

** NPAS = 1 (PASSED GROUPS FOR CONTINUUM)

MAX.NO. OF ELASTIC TRANSFER
3 2 2 2 2 9 5 18 21 10 9 9 30 8 40 48 43 3 699 699

Fine group mesh-dependent neutron fluxes are printed out evergy fifty groups togetherwith those homogeneized neutron spectrum on option. An example of the print out from the 1901-group to the 1950-group is shown

	mesh point									
GPN	1	2	3	4	5	6	7	8	9	10
1901	0.157E-13	0.263E-13	0.737E-15	0.510E-13	0.521E-13	0.304E-13	0.464E-15			
1902	0.140E-13	0.234E-13	0.967E-15	0.469E-13	0.479E-13	0.279E-13	0.615E-15			
1903	0.125E-13	0.209E-13	0.121E-14	0.431E-13	0.440E-13	0.257E-13	0.779E-15			
1904	0.112E-13	0.188E-13	0.144E-14	0.396E-13	0.405E-13	0.236E-13	0.939E-15			
1905	0.101E-13	0.170E-13	0.165E-14	0.365E-13	0.373E-13	0.218E-13	0.108E-14			
1906	0.915E-14	0.155E-13	0.181E-14	0.337E-13	0.345E-13	0.201E-13	0.120E-14			
1907	0.836E-14	0.142E-13	0.194E-14	0.312E-13	0.319E-13	0.187E-13	0.129E-14			
1908	0.770E-14	0.132E-13	0.202E-14	0.289E-13	0.297E-13	0.174E-13	0.135E-14			
1909	0.714E-14	0.124E-13	0.208E-14	0.269E-13	0.276E-13	0.162E-13	0.139E-14			
1910	0.666E-14	0.116E-13	0.211E-14	0.250E-13	0.257E-13	0.151E-13	0.141E-14			
1911	0.625E-14	0.110E-13	0.212E-14	0.233E-13	0.240E-13	0.141E-13	0.141E-14			
1912	0.589E-14	0.104E-13	0.210E-14	0.218E-13	0.224E-13	0.132E-13	0.140E-14			
1913	0.558E-14	0.994E-14	0.208E-14	0.203E-13	0.210E-13	0.124E-13	0.138E-14			
1914	0.530E-14	0.950E-14	0.204E-14	0.190E-13	0.196E-13	0.116E-13	0.135E-14			
1915	0.505E-14	0.910E-14	0.199E-14	0.178E-13	0.184E-13	0.109E-13	0.131E-14			
1916	0.482E-14	0.872E-14	0.194E-14	0.167E-13	0.173E-13	0.103E-13	0.127E-14			
1917	0.461E-14	0.837E-14	0.189E-14	0.157E-13	0.162E-13	0.965E-14	0.123E-14			
1918	0.442E-14	0.803E-14	0.183E-14	0.147E-13	0.152E-13	0.908E-14	0.118E-14			
1919	0.423E-14	0.771E-14	0.176E-14	0.138E-13	0.143E-13	0.854E-14	0.113E-14			
1920	0.406E-14	0.740E-14	0.170E-14	0.130E-13	0.135E-13	0.804E-14	0.109E-14			
1921	0.389E-14	0.710E-14	0.163E-14	0.122E-13	0.126E-13	0.756E-14	0.104E-14			
1922	0.373E-14	0.681E-14	0.157E-14	0.115E-13	0.119E-13	0.711E-14	0.988E-15			
1923	0.358E-14	0.653E-14	0.150E-14	0.108E-13	0.112E-13	0.669E-14	0.940E-15			
1924	0.342E-14	0.626E-14	0.143E-14	0.101E-13	0.105E-13	0.628E-14	0.893E-15			
1925	0.328E-14	0.599E-14	0.137E-14	0.949E-14	0.986E-14	0.591E-14	0.847E-15			
1926	0.314E-14	0.573E-14	0.130E-14	0.891E-14	0.927E-14	0.555E-14	0.802E-15			
1927	0.300E-14	0.548E-14	0.124E-14	0.837E-14	0.871E-14	0.522E-14	0.759E-15			
1928	0.287E-14	0.523E-14	0.118E-14	0.786E-14	0.818E-14	0.490E-14	0.718E-15			
1929	0.274E-14	0.499E-14	0.112E-14	0.738E-14	0.769E-14	0.460E-14	0.678E-15			
1930	0.261E-14	0.476E-14	0.107E-14	0.693E-14	0.722E-14	0.433E-14	0.640E-15			
1931	0.249E-14	0.453E-14	0.101E-14	0.651E-14	0.679E-14	0.407E-14	0.604E-15			
1932	0.237E-14	0.432E-14	0.958E-15	0.612E-14	0.638E-14	0.382E-14	0.570E-15			
1933	0.226E-14	0.411E-14	0.907E-15	0.575E-14	0.600E-14	0.359E-14	0.537E-15			
1934	0.215E-14	0.390E-14	0.859E-15	0.541E-14	0.564E-14	0.338E-14	0.506E-15			
1935	0.204E-14	0.371E-14	0.813E-15	0.509E-14	0.530E-14	0.318E-14	0.477E-15			
1936	0.194E-14	0.352E-14	0.768E-15	0.478E-14	0.499E-14	0.299E-14	0.450E-15			
1937	0.184E-14	0.334E-14	0.726E-15	0.450E-14	0.469E-14	0.281E-14	0.423E-15			
1938	0.174E-14	0.316E-14	0.685E-15	0.423E-14	0.441E-14	0.264E-14	0.398E-15			
1939	0.165E-14	0.300E-14	0.647E-15	0.398E-14	0.415E-14	0.249E-14	0.375E-15			
1940	0.156E-14	0.284E-14	0.610E-15	0.374E-14	0.390E-14	0.234E-14	0.353E-15			
1941	0.148E-14	0.268E-14	0.575E-15	0.352E-14	0.367E-14	0.220E-14	0.331E-15			
1942	0.140E-14	0.254E-14	0.541E-15	0.331E-14	0.345E-14	0.207E-14	0.311E-15			
1943	0.132E-14	0.239E-14	0.509E-15	0.311E-14	0.324E-14	0.194E-14	0.292E-15			
1944	0.125E-14	0.226E-14	0.479E-15	0.292E-14	0.305E-14	0.182E-14	0.274E-15			
1945	0.118E-14	0.213E-14	0.450E-15	0.275E-14	0.286E-14	0.171E-14	0.257E-15			
1946	0.111E-14	0.201E-14	0.422E-15	0.258E-14	0.269E-14	0.161E-14	0.241E-15			
1947	0.105E-14	0.189E-14	0.396E-15	0.242E-14	0.254E-14	0.151E-14	0.226E-15			
1948	0.98?E-15	0.178E-14	0.371E-15	0.227E-14	0.237E-14	0.142E-14	0.211E-15			
1949	0.930E-15	0.167E-14	0.347E-15	0.213E-14	0.224E-14	0.133E-14	0.198E-15			
1950	0.93?E-14	0.101E-13	0.292E-14	0.183E-13	0.170E-13	0.116E-13	0.186E-14			

number of group

HOMOGENEIZED FLUX group →

0.617E-13	0.565E-13	0.517E-13	0.475E-13	0.436E-13	0.405E-13	0.376E-13	0.350E-13	0.327E-13	0.306E-13
0.287E-13	0.270E-13	0.254E-13	0.239E-13	0.226E-13	0.213E-13	0.201E-13	0.190E-13	0.180E-13	0.170E-13
0.161E-13	0.152E-13	0.144E-13	0.136E-13	0.128E-13	0.121E-13	0.114E-13	0.108E-13	0.102E-13	0.962E-14
0.908E-14	0.857E-14	0.808E-14	0.763E-14	0.720E-14	0.679E-14	0.640E-14	0.603E-14	0.569E-14	0.536E-14
0.505E-14	0.475E-14	0.447E-14	0.421E-14	0.396E-14	0.372E-14	0.350E-14	0.328E-14	0.308E-14	0.249E-13

RESULTS OBTAINED BY COLLAPSING

** COLLAPSING OF FINE-GROUP CROSS SECTIONS **

FCA VI-1 TEST ZONE HETERO , STANDARD (AGLI/GN-AGLI/0)

(Coarse group structure.)

GROUP CONDENSATION *** NRCAD GR= 26

LOWER ENERGY GROUP BOUNDARY

55	110	165	220	275	330	385	440	495	550
605	660	715	770	825	880	935	990	1045	1100
1180	1270	1360	1480	1600	1950				

(Lower boundary energy in ev.)

0.65790E 07	0.41222E 07	0.25828E 07	0.16183E 07	0.10140E 07	0.63532E 06
0.39807E 06	0.24942E 06	0.15628E 06	0.97918E 05	0.61352E 05	0.38441E 05
0.24086E 05	0.15091E 05	0.94558E 04	0.59247E 04	0.37122E 04	0.23260E 04
0.14574E 04	0.91314E 03	0.46261E 03	0.21527E 03	0.10017E 03	0.36121E 02
0.13025E 02	0.66490E 00				

GRP.	TOTAL	FISSION	INELA	CAPTURE	NU-SIGF	DIFFUSION	ELASTIC
1	1.5295E-01	1.1064E-02	4.3460E-02	5.1867E-03	4.0059E-02	4.1935E 00	9.3236E-02
2	1.7196E-01	7.4364E-03	5.1820E-02	1.8586E-03	2.4920E-02	3.4947E 00	1.1084E-01
3	1.8065E-01	7.3193E-03	4.5399E-02	5.4145E-04	2.2091E-02	2.9856E 00	1.2739E-01
4	1.7160E-01	7.1596E-03	3.9054E-02	4.5474E-04	2.0715E-02	2.7233E 00	1.2493E-01
5	1.9029E-01	4.1250E-03	3.0405E-02	8.3819E-04	1.2078E-02	2.2148E 00	1.5492E-01
6	2.1694E-01	2.8641E-03	2.0170E-02	1.3437E-03	8.4857E-03	1.9200E 00	1.9257E-01
7	2.4879E-01	2.5706E-03	1.4536E-02	1.2012E-03	7.5519E-03	1.7624E 00	2.3048E-01
8	2.2823E-01	2.4822E-03	1.1490E-02	1.2967E-03	7.2546E-03	1.7375E 00	2.1296E-01
9	2.5854E-01	2.4453E-03	9.2112E-03	1.6070E-03	7.1320E-03	1.4418E 00	2.4528E-01
10	2.6425E-01	2.4222E-03	6.4880E-03	2.0423E-03	7.0551E-03	1.3593E 00	2.5330E-01
11	2.8914E-01	2.3943E-03	2.8989E-03	2.6714E-03	6.9616E-03	1.2341E 00	2.8117E-01
12	2.9498E-01	2.3614E-03	6.3218E-04	3.5160E-03	6.8537E-03	1.1890E 00	2.8847E-01
13	3.9147E-01	2.4109E-03	5.3986E-04	4.5009E-03	6.9867E-03	9.9094E-01	3.8402E-01
14	3.0920E-01	2.7144E-03	4.8337E-04	5.4470E-03	7.8576E-03	1.1420E 00	3.0056E-01
15	3.6595E-01	3.0662E-03	3.4696E-04	6.9398E-03	8.8621E-03	9.4191E-01	3.5559E-01
16	5.0893E-01	3.2650E-03	2.8672E-05	7.9221E-03	1.0285E-02	6.8901E-01	4.9742E-01
17	5.2796E-01	4.0724E-03	0.0	9.3612E-03	1.1720E-02	6.4864E-01	5.1453E-01
18	1.3206E 00	4.6941E-03	0.0	1.1769E-02	1.3494E-02	3.2055E-01	1.3042E 00
19	4.6634E-01	5.9355E-03	0.0	1.2019E-02	1.7050E-02	7.5434E-01	4.4838E-01
20	3.7997E-01	7.4134E-03	0.0	1.4122E-02	2.1238E-02	8.9668E-01	3.5844E-01
21	3.8540E-01	1.0527E-02	0.0	1.5764E-02	3.0164E-02	8.8458E-01	3.5911E-01
22	4.0068E-01	1.5520E-02	0.0	1.6045E-02	4.4486E-02	8.5211E-01	3.6911E-01
23	4.3362E-01	1.9747E-02	0.0	2.2922E-02	5.6596E-02	8.2574E-01	3.9096E-01
24	4.3052E-01	1.8688E-02	0.0	2.3477E-02	1.1096E-01	8.2714E-01	3.6826E-01
25	3.9913E-01	1.5582E-02	0.0	3.7789E-02	4.4509E-02	8.8221E-01	3.4576E-01
26	4.5113E-01	4.8722E-02	0.0	4.5205E-02	1.3953E-01	7.8333E-01	3.5721E-01

ELASTIC REMOVAL $\sigma_{EL}(I-J)$

GROUP 1

0.8261E-01	0.1062E-01	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0			

(SUM -- 0.93236E-01)

GROUP 2

0.9945E-01	0.1139E-01	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0			

(SUM -- 0.11084E 00)

GROUP 3

0.1163E 00	0.1105E-01	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0			

(SUM -- 0.12739E 00)

GROUP 4

0.1104E 00	0.1456E-01	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0			

(SUM -- 0.12493E 00)

GROUP 5

0.1314E 00	0.2353E-01	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0			

(SUM -- 0.15492E 00)

ANGULAR DISTRIBUTION OF ELASTIC SCATTERING(LAB.) --- MUL(I,J) μ_{EL}

GROUP 1

0.25338E-01	0.59151E-03	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0				

(SUM -- 0.76763E-03)

GROUP 2

0.31281E-01	0.67400E-03	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0				

(SUM -- 0.33690E-02)

GROUP 3

0.36597E-01	-0.29009E-03	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0				

(SUM -- 0.80445E-02)

GROUP 4

0.31869E-01	-0.22892E-02	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0				

(SUM -- 0.10192E-01)

GROUP 5

0.30896E-01	-0.42904E-02	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0				

(SUM -- 0.95486E-02)

GROUP 6

0.45555E-01	-0.10634E-02	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0				

(SUM -- 0.20682E-01)

GROUP 7

0.58246E-01	-0.21134E-02	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0				

(SUM -- 0.24205E-01)

$\sigma_{IN}(i-j)$

INELASTIC REMOVAL

1--j									
0.8173E-03	0.4721E-02	0.8352E-02	0.1029E-01	0.9139E-02	0.6359E-02	0.3638E-02	0.1853E-02	0.8619E-03	0.3680E-03
0.1529E-03	0.6236E-04	0.2519E-04	0.9995E-05	0.3973E-05	0.1582E-05	0.6275E-06	0.2520E-06	0.3477E-07	0.5290E-08
0.4379E-08	0.7030E-09	0.0	0.0	0.0	0.0				
2--j									
0.1782E-02	0.7157E-02	0.1041E-01	0.1123E-01	0.9152E-02	0.5980E-02	0.3299E-02	0.1613E-02	0.7280E-03	0.3121E-03
0.1296E-03	0.5272E-04	0.2117E-04	0.8431E-05	0.3341E-05	0.1318E-05	0.5185E-06	0.5725E-07	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0				
3--j									
0.2718E-02	0.1172E-01	0.7679E-02	0.7834E-02	0.6824E-02	0.4284E-02	0.2306E-02	0.1102E-02	0.5027E-03	0.2167E-03
0.8739E-04	0.3410E-04	0.1424E-04	0.5652E-05	0.1762E-05	0.6723E-06	0.8897E-07	0.1118E-07	0.9253E-08	0.1485E-08
0.0	0.0	0.0	0.0	0.0	0.0				
4--j									
0.2620E-02	0.1319E-01	0.1047E-01	0.5255E-02	0.3669E-02	0.2076E-02	0.9525E-03	0.4388E-03	0.1915E-03	0.7980E-04
0.3247E-04	0.1282E-04	0.5068E-05	0.1964E-05	0.7692E-06	0.8707E-07	0.1548E-08	0.4060E-09	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0				
5--j									
0.5447E-02	0.7629E-02	0.8042E-02	0.4805E-02	0.2660E-02	0.1038E-02	0.4629E-03	0.2146E-03	0.6689E-04	0.2060E-04
0.7524E-05	0.1709E-05	0.5076E-06	0.1899E-06	0.2101E-07	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0				
0.1079E-01	0.5060E-02	0.1869E-02	0.1388E-02	0.6221E-03	0.2794E-03	0.8248E-04	0.4721E-04	0.2154E-04	0.7430E-05
0.1474E-05	0.7476E-07	0.1584E-07	0.1968E-07	0.1606E-07	0.1329E-07	0.2134E-08	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0				
0.1091E-01	0.2884E-02	0.3327E-03	0.2847E-03	0.7444E-04	0.2824E-04	0.1054E-04	0.4297E-05	0.1660E-05	0.5701E-06
0.2079E-06	0.1376E-07	0.6208E-09	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				
0.7375E-02	0.3872E-02	0.1638E-03	0.4685E-04	0.2146E-04	0.7443E-05	0.2834E-05	0.1045E-05	0.2018E-06	0.3528E-07
0.5625E-08	0.6246E-09	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				
0.4744E-02	0.4323E-02	0.9292E-04	0.3060E-04	0.1428E-04	0.4241E-05	0.1334E-05	0.5004E-07	0.1920E-07	0.7690E-08
0.8556E-09	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				
0.1763E-02	0.4027E-02	0.6872E-03	0.8440E-05	0.1346E-05	0.4016E-06	0.6346E-06	0.2593E-06	0.1253E-07	0.1497E-08
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0	0.0				
0.4341E-03	0.1407E-02	0.8923E-03	0.1472E-03	0.8526E-05	0.3515E-05	0.1579E-05	0.1450E-06	0.2508E-07	0.1256E-09
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

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CAPTURE,FISSION CROSS SECTION AND YIELD AT EACH MESH
    ( Material-mesh-dependent broad group cross section.)
 GROUP   1

 **MESH =  1
                material------>
   *CAPTURE
 0.0          0.0           0.0           0.0           0.0           0.0
 0.0          0.36809E-04   0.25984E-03   0.27790E-03   0.0

   *FISSION
 0.0          0.0           0.0           0.0           0.0           0.0
 0.0          0.0           0.0           0.0           0.0

   *YIELD
 0.0          0.0           0.0           0.0           0.0           0.0
 0.0          0.0           0.0           0.0           0.0

 **MESH =  2

   *CAPTURE
 0.0          0.0           0.0           0.0           0.0           0.0
 0.21783E-03  0.14547E-04   0.95261E-04   0.17093E-03   0.0

   *FISSION
 0.0          0.0           0.0           0.0           0.0           0.0
 0.0          0.0           0.0           0.0           0.0

   *YIELD
 0.0          0.0           0.0           0.0           0.0           0.0
 0.0          0.0           0.0           0.0           0.0

 **MESH =  3

   *CAPTURE
 0.12906E-05  0.17395E-05   0.27073E-07   0.0           0.0           0.0
 0.0          0.75322E-05   0.45391E-04   0.59531E-04   0.17874E-04

   *FISSION
 0.25927E-02  0.21281E-03   0.26109E-04   0.0           0.0           0.0
 0.0          0.0           0.0           0.0           0.0

   *YIELD
 0.10035E-01  0.80499E-03   0.10237E-03   0.0           0.0           0.0
 0.0          0.0           0.0           0.0           0.0

 **MESH =  4

   *CAPTURE
 0.0          0.0           0.0           0.23573E-07   0.43041E-04   0.19264E-02
 0.0          0.0           0.0           0.0           0.0

   *FISSION
 0.0          0.0           0.0           0.17129E-04   0.45343E-02   0.0
 0.0          0.0           0.0           0.0           0.0

   *YIELD
 0.0          0.0           0.0           0.59326E-04   0.15737E-01   0.0
 0.0          0.0           0.0           0.0           0.0

 **MESH =  5


INFORMATION ON FISSION

 ELEMENT    FISS.RATIO  NFIS. RATIO AVE.  NU.         NUCL. TEMP.

   939      0.8380E 00  0.8445E 00  0.2960E 01  0.1396E 01
   940      0.1997E-01  0.2095E-01  0.3081E 01  0.1368E 01
   941      0.1403E-01  0.1441E-01  0.3018E 01  0.1362E 01
   935      0.8271E-02  0.6981E-02  0.2479E 01  0.1308E 01
   928      0.1197E 00  0.1132E 00  0.2776E 01  0.1348E 01

AVERAGED GROUP FISS. SPECTRUM

 0.21962E-01  0.91342E-01  0.17896E 00  0.21366E 00  0.18508E 00  0.13056E 00  0.80837E-01  0.46092E-01  0.24951E-01  0.13074E-01
 0.67113E-02  0.34012E-02  0.17099E-02  0.85526E-03  0.42644E-03  0.21220E-03  0.10547E-03  0.52383E-04  0.61587E-05  0.0
 0.0          0.0          0.0          0.0          0.0          0.0


REACTION RATE ANALYSIS.

CELL TRAVERSE REACTION RATE PER AD=1.0
    1   939   FISS.= 0.16882E 01   CAOT.= 0.34317E 00
    2   940   FISS.= 0.43605E 00   CAOT.= 0.44106E 00
    3   941   FISS.= 0.27787E 01   CAOT.= 0.37166E 00
    4   935   FISS.= 0.17357E 01   CAOT.= 0.45135E 00
    5   928   FISS.= 0.55399E-01   CAOT.= 0.25966E 00
    6     8   FISS.= 0.0           CAOT.= 0.14807E-02
    7    11   FISS.= 0.0           CAOT.= 0.16734E-02
    8    24   FISS.= 0.0           CAOT.= 0.57852E-02
    9    26   FISS.= 0.0           CAOT.= 0.71728E-02
   10    28   FISS.= 0.0           CAOT.= 0.20708E-01
   11    13   FISS.= 0.0           CAOT.= 0.33980E-02

ACTUAL REACTION RATE IN CELL FOR ALL ATOM IN CELL

    1   939   FISS.= 0.28007E-02   CAOT.= 0.55113E-03
    2   940   FISS.= 0.66735E-04   CAOT.= 0.64440E-04
    3   941   FISS.= 0.46876E-04   CAOT.= 0.61908E-05
    4   935   FISS.= 0.27645E-04   CAOT.= 0.71919E-05
    5   928   FISS.= 0.40021E-03   CAOT.= 0.18486E-02
    6     8   FISS.= 0.0           CAOT.= 0.24435E-04
    7    11   FISS.= 0.0           CAOT.= 0.13222E-04
    8    24   FISS.= 0.0           CAOT.= 0.21698E-04
    9    26   FISS.= 0.0           CAOT.= 0.98283E-04
   10    28   FISS.= 0.0           CAOT.= 0.35561E-04
   11    13   FISS.= 0.0           CAOT.= 0.48630E-05
```

The graphic output and disk output are obtained according to the general procedure in the DOYC system. An example of graphic output of the cell-averaged neutron spectrum is shown in **Fig. 4**. The scale of graph can be chosen either as log-log or linear-log on option.

Various calculated results which will be used as input data to a subsequent computation can be stored on the data pool disk provided in the DOYC system on option. These are the fine and broad group neutron fluxes, all of broad group cross section data and fission neutron spectrum. The printer outputs of the data stored on the data pool are also available using a supporting device of the DOYC system.

8. Typical Results and Discussions

Typical numerical results for the neutron spectra and plate bunching effects on the criticality calculated by SP-2000 are discussed in this section.

Two calculated results for the fundamental mode neutron spectrum in the natural uranium blanket of FCA[13] are shown in **Fig. 4**. The solid line in the figure gives the result calculated by SP-2000 and the dotted line is obtained by the another code of similar type[14]. The neutron spectrum measured by a He-3 proportional counter is also given by circles in the figure. In **Fig. 5** the cell-averaged neutron spectrum in the core region of FCA V-3[13] is shown together with the spectrum measured at the core center using the Li-6 sandwich counter.

The plate bunching effects caused by the difference in the effective cross sections of different cells can be calculated by using SP-2000 in connection with a conventional eigenvalue code. The measured and calculated plate bunching effects[15] at FCA V-1 and V-2 and the corresponding plate configuration are summarized in **Figs. 6~9**. The effective thickness termed in the fig-

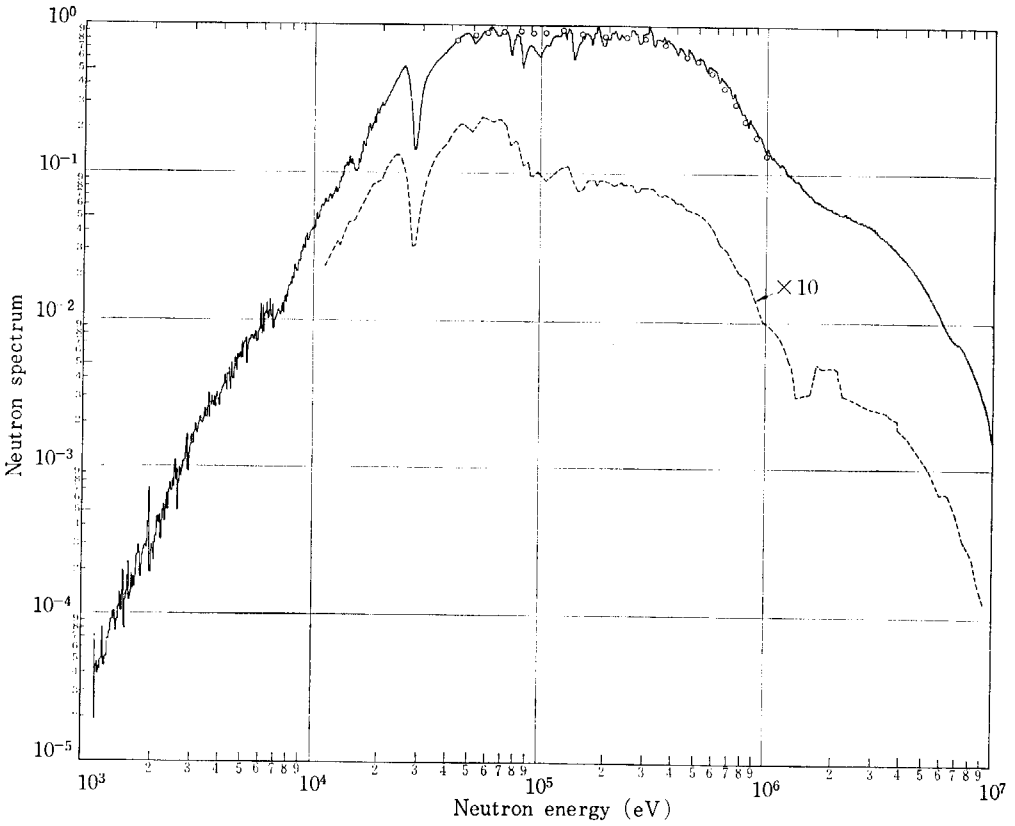

Fig. 4 Neutron energy spectrum in FCA natural uranium blanket.

ures is obtained by dividing the total thickness of fuel plate by the number of fuel-bunchings. The calculated values are obtained from the 24 broad group cell averaged cross sections produced by SP-2000.

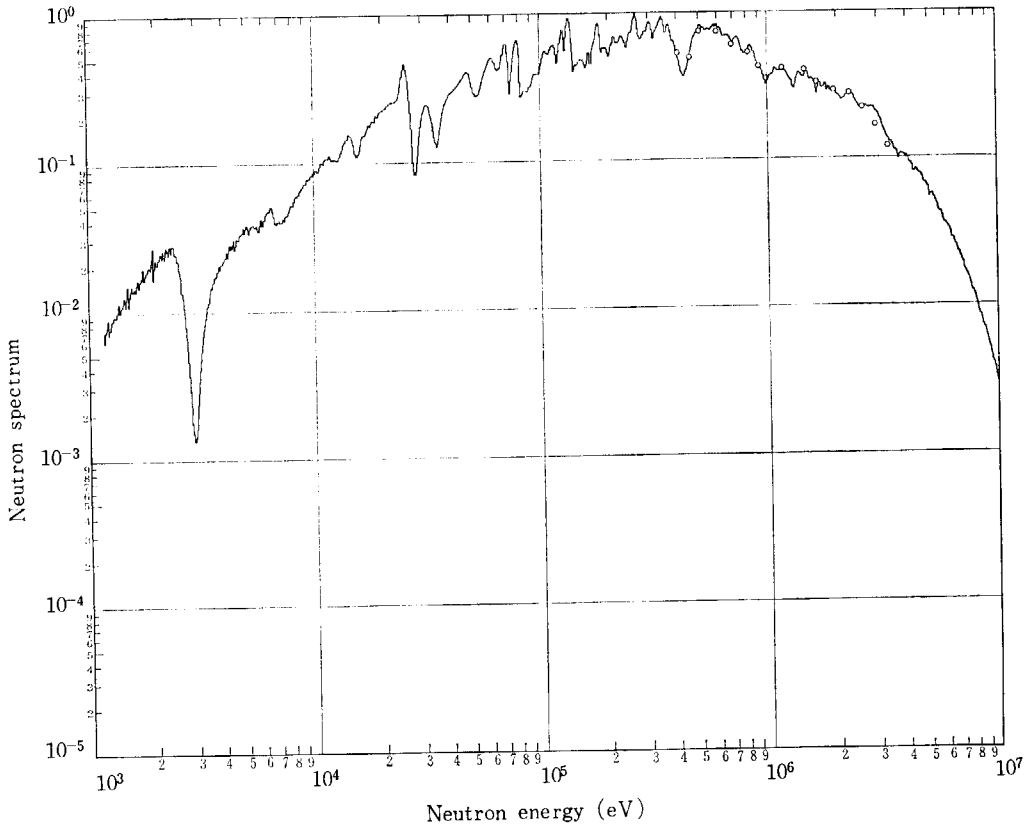

Fig. 5 Neutron energy spectrnm at core center of FCA V-3.

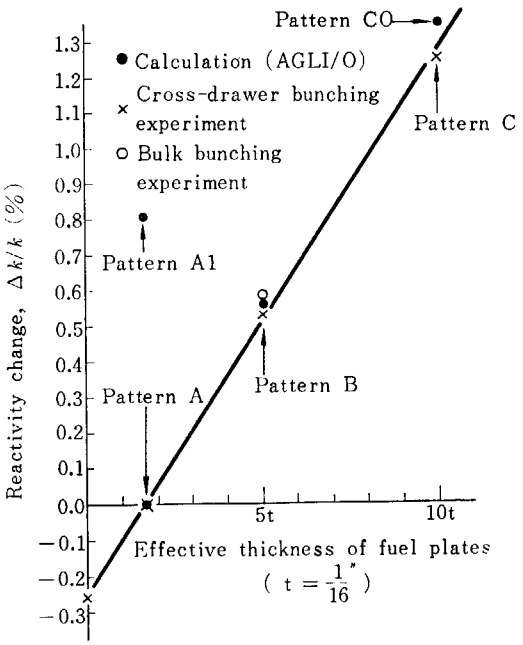

Fig. 6 Reactivity changes by different bunching patterns of FCA V-1 core.

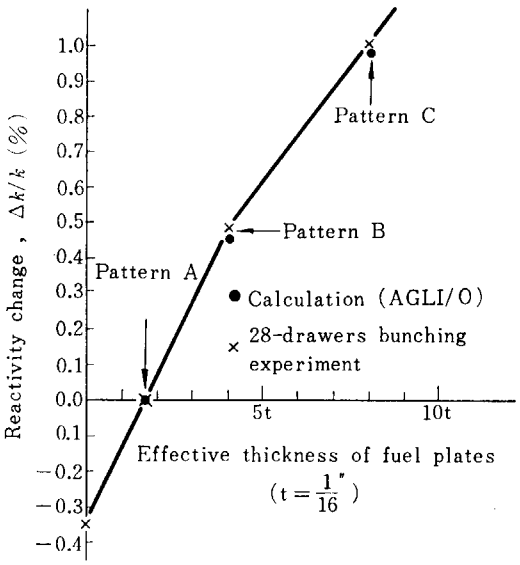

Fig. 7 Reactivity changes by different bunching patterns of FCA V-2 core.

The agreement between these measured and calculated results is satisfactory for both the neutron spectrum and the plate bunching effects.

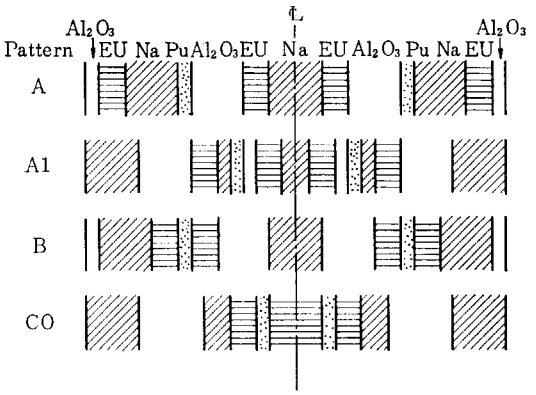

Fig. 8 Bunching patterns of FCA V-1 core.

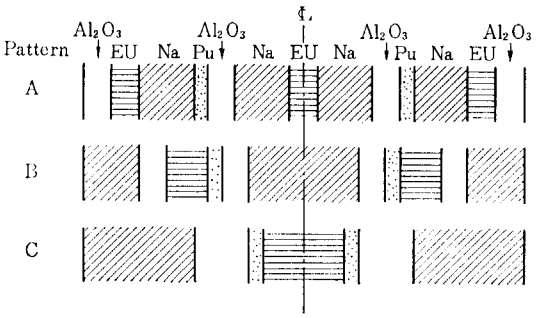

Fig. 9 Bunching patterns of FCA V-2 core.

The typical computer CPU time on FACOM-230/60 is about 30 min. (nearly equivalent to about 2 min. on IBM-360/195) to deal with a problem including the whole calculations of material buckling, spatial distribution of fission source, multi-region fine group spectrum and cell-averaged broad group cross section in a cell containing 10 materials and 12 space mesh points. For a homogeneous problem, the CPU time is about 5 min. on the same computer.

Since considerable part of above CPU time is spent in the calculation of inelastic scattering source in each region, it may be possible to shorten the CPU time significantly by modifying the source calculation routine. The option NPAS in the input data (3-2) will be used for this purpose in a future version of the SP-2000 to reduce the CPU time by assuming the same inelastic transfer probability over the neighbouring NPAS fine groups.

In the future version, further modifications are planned to shorten the compnter time of the calculation of collision probability in the cylindrical geometry and to introduce a new boundary condition with which one can analyze the reactivity worth of a sample material inserted in a reactor system.

Acknowledgements

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References

1) Rowlands J. L., Macdougall J. D., "The Use of Integral Measurements to Adjust Cross Sections and Predict Reactor Properties"; Proceedings of BNES International Conference on the Physics of Fast Reactor Operation and Design, London, (1969).
2) Toppel B. J. *et al.*, "MC²-A Code to Calculate Multigroup Cross Section"; ANL-7318, (1967).
3) Nakagawa M., Narita H., Katsuragi S., "Reactor Engineering Division Annual Report (April 1, 1973 to March 31, 1974)"; JAERI-M 5955, p. 43-44, (1975).
4) Kuroi H., Tone T., "Cross Section Data and Specification of AGLI/0 for Fast Reactor Analysis"; JAERI 1230, (1973).

5) Suzuki T., Ishiguro Y., "FASTOS Code"; JAERI-memo 1914, (1965).
6) Honeck H. C., "THERMOS"; BNL-5826, (1961).
7) Bell G. I., "A Simple Treatment for Effective Resonance Absorption Cross Section in Dense Lattice"; *Nucl. Sci. Eng.*, **5**, 138, (1959).
8) Tone T., Katsuragi S., "PROF GROUCH-G"; JAERI 1192, (1970).
9) Amaldi E., Handbuch der Physik, Vol. XXXVIII/2, (1959), Springer-Verlarg, Berline.
10) Parker K., "Neutron Cross Sections of U^{235} and U^{238} in the Energy Range 1 keV-15 MeV"; AW RE-0-79/63, (1964).
11) Tsuchihashi K., "Computer Program LAMP"; to be published.
12) Hasegawa A., "General Purpose Graph Producing Subroutine GPLOT 1"; JAERI-memo 4255, (1970).
13) Kaneko T., Kuroi H., Hirota J. "Use of Differential Neutron Spectrum Data in Reactor Physics"; Paper Presented for the 15-th EACRP-meeting, (1972).
14) Shirakata K., Mukaiyama T., Iijima T. "Equilibrium Neutron Spectrum in Natural Uranium"; 1972 Annual Meeting of JAES.
15) Tone T. *et al.*, "Analysis of Integral Data Measured at FCA V Assembly by Using AGLI Library"; Topical Meeting on Fast Reactor Physics, B-9, JAES, (1973).

Appendix : Organization of the 1950 Energy Groups AGLI Binary Data Library

Two data files of AGLI library, A and B, have been provided in binary mode. The organization of these binary data disks (or tapes) is described in this appendix.

1. Organization of File A

The first logical record on File A is composed of the title of the data disk, two values of σ_0 used for the histogram calculation, the highest energy of the data, the fine group lethargy width, the number of fine energy groups and the code number of each element on the file. Then the group cross sections and self-shielding correction factors are given by a bunch of 50 energy groups.

(a) 1st logical record

TITLE (15), S0, S 1, EM, DELU, MAXG, LCODE (20)

TITLE : Title of the file,
S0, S 1 : Value of σ_0 used for histogram calculation,
EM : The upper energy of the 1 st group,
DELU : Group lethargy width,
MAXG : The number of energy groups,
LCODE : Code numbers of elements stored on the data file.

(b) 2nd logical record

In the second logical record, the group cross sections of the first 50 energy groups of the first element are given as follows,

IGI : Group number, if flagged negative, no self-shielding corrections are given,
FX(K, 1) : $\bar{\sigma}_{tot}$,
FX(K, 2) : $\bar{\sigma}_{f}$,
FX(K, 3) : $\bar{\sigma}_{ab}$,
FX(K, 4) : $\bar{\sigma}_{e}$ (not given in AGLI/0),
FX(K, 5) : W_2,
FX(K, 6) : ε_1^t,
FX(K, 7) : ε_2^t,
FX(K, 8) : ε_2^a,
FX(K, 9) : ε_2^f,
FX(K, 10) : not used,
where K indicates an energy group.

(c) 3rd logical record

IGI and FX of the first 50 energy groups of the second element are given in the third record.

The similar records are repeated for all elements, then the data of the second 50 groups are stored in the same manner.

2. Organization of File B

The first logical record on File B contains the data for the angular distribution of elastic scattering which is expanded in a series of up to 13 terms of Chebyshev's polynomials. The second record is composed of data of the number of neutrons emitted per fission, which is expanded also in the Chebyshev polynomials (up to 13 terms), and the nuclear temperatures used for fission neutron spectra, together with the corresponding values of ν (the average number of neutrons per fission). In the third, fourth and fifth records, the data of inelastic scattering cross sections are stored.

(a) 1st logical record

TITLE (20), AWT (20), AEE (20), BBE (20), CNUE (7, 20, 13), CC (20), AAL (20), BBL (20), CNUL (7, 20, 13)

TITLE : Title of the file,
AWT : Atomic weight,
AAE : The lowest lethargy at which the Chebyshev expansion is applied for calculating μ_c of the elastic scattering in the center of mass system ; usually 0.0,
BBE : The highest lethargy of the interval where the above expansion is applied,
CNUE : Expansion coefficients of μ_c,
CC : Constant value of μ_l in the laboratory system,
AAL : The lowest lethargy at which the Chebyshev expansion is applied for μ_l of the elastic scattering in the laboratory system ; usually 0.0,
BBL : The highest lethargy of the interval where the above expansion is applied,
CNUL : Expansion coefficients of μ_l,

The dimensions given in the parentheses mean ; 20= the maximum number of nuclides, 13= the maximum number of terms in the Chebyshev expansion, 7= the maximum number of terms in the Legendre expansion.

(b) 2nd logical record

ACC (20), AS (20), BS (20), CVNU (20, 13), SITA (3, 20), CNU (3, 20)

ACC : Constant value of ν (the number of neutrons per fission),
AS : The lowest lethargy of the interval where the Chebyshev expansion is applied for ν, usually 0.0,
BS : The highest lethargy of the interval where the above expansion is applied,
CVNU : Expansion coefficients of ν,
SITA : Nuclear temperature used for fission neutron spectrum in MeV, corresponding to the value of ν given by CNU,
CNU : Value of ν at which the nuclear temperature is given,

The dimensions given in the parentheses mean ; 20= the maximum number of nuclides stored on the file, 3= the maximum number of different values of ν to which the corresponding nuclear temperature is given.

(c) 3rd logical record

MAXG, MING, NIDL (20), INIC (20), AC (20), ITHI (20), AQRTC (2000)

MAXG : The highest energy group above which the inelastic scattering does not occur ; usually 1,

MING : The lowest energy group below which the inelastic scattering does not occur,
NIDL : Indication for the presence of discrete levels,
INIC : Indication for the presence of the continuum,
AC : The value of a in Eq. (3-35),
ITHI : The lowest energy group below which the transfer cross section due to the continuum is ignored,
AQRTC : The value of $\sqrt{E}$ (E: upper energy of each fine group in MeV).

(d) 4 th logical record

QV (LEVT)

QV : Level energy (positive value),
LEVT : Aggregate number of discrete levels and the continuum for allelements in AG LI/0.

(e) 5 th logical record

EXECIN (LEVT, 50), FACTOR (20, 50)

EXECIN : Fine group cross sections due to discrete levels and the continuum,
FACTOR : The source due to the continuum into the self-group.

The fifth record is repeated for every fifty energy groups until the inelastic scattering cross section in the lowest group is stored.

The following code numbers are used in the AGLI

Code number of each nuclide

U-235	935
U-238	928
Pu-239	939
Pu-240	940
Pu-241	941
Fe	26
Cr	24
Ni	28
Mn	25
Mo	42
Pb	82
Cu	29
Al	13
Na	11
O	8
C	6
H	1
B-10	105
B-11	115