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SRAC : JAERI Thermal Reactor Standard Code System for Reactor Design and Analysis

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Page 46 line 21 (6(i3,E9.0)) replaced by (6(I3,E9.0))

Page 150 line 17 Equation VI.2.2-4) replaced by

$$\sigma_1(E\to E') = \frac{3\sigma_s(E)}{(1-\alpha)E}\left\{\frac{A+1}{2}(E'/E)^{1/2} - \frac{A-1}{2}(E/E')^{1/2}\right\} \qquad (VI.2.2\text{-}4)$$

Page 191 line 9 from bottom % replaced by θ

Page 145 line 6 Subsection VI.1.6 replaced by

VI.1.6 Spline Interpolation for Resonance Self-Shielding Factors

The multigroup constant set prepared for SRAC-FASTLIB has the tables, $f(\sigma_0,T)$, of resonance self-shielding factors which are given as function of temperature T and effective background cross section σ_0 of Bondarenko scheme. In reactor calculation, the effective microscopic cross sections for each material zone are calculated from the infinitely dilute cross sections and the self-shielding factors interpolated for the effective temperature and background cross sections using $f(\sigma_0,T)$ -tables.

As for the interpolation schemes of $f(\sigma_0,T)$ -tables, many methods have been studied(Ref.(61)). For example,in the 1-D diffusion code EXPANDA-4(Ref.(59)) a rational function is used for both the σ_0 - and T-interpolations, and the Kidman's schemes(Ref.(60)) used in the 1-D diffusion code IDX are a Lagrange three-point interpolation on the (lnT,f) tabular points and the four-parameter interpolation formula for the $f(\sigma_0)$ -table. Furthermore, there are several methods described in Ref.(61). Among these schemes, however, the superiority of the accuracy and computing cost is not compared. At the present time, the f-tables can numerically be calculated with high accuracy by the use of a processing code such as TIMS-1. Hence the error caused by interpolating the f-tables should be as small as possible.

In Ref.(21), various interpolation methods has been compared in connection with computing cost and accuracy. Moreover, the effects of the difference among various interpolations on integral quantities in fast reactor have been studied.As the results, the cubic spline method offers a very practical means with good accuracy and short computing time, as to the interpolation for the 'table look-up' method of resonance self-shielding factors. It was shown that the errors caused by the differences among the various interpolation methods produce non-negligible effects on effective multiplication factor, control rod reactivity worth and Doppler effect, because the effects are comparable to the goal accuracy requested from design study for a large fast breeder reactor. Furthermore, the difference produces a slightly anxious effect on reaction rate distribution in blanket region (Ref.(61)).

The cubic spline interpolation is adopted in the SRAC code system.

SRAC : JAERI Thermal Reactor Standard Code System for Reactor Design and Analysis

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The SRAC (Standard Reactor Analysis Code) is a code system for nuclear reactor analysis and design. It is composed of neutron cross section libraries and auxiliary processing codes, neutron spectrum routines, a variety of transport , 1-, 2- and 3-D diffusion routines, dynamic parameters and cell burn-up routines.

By making the best use of the individual code function in the SRAC system, the user can select either the exact method for an accurate estimate of reactor characteristics or the economical method aiming at a shorter computer time, depending on the purpose of study. The user can select cell or core calculation ; fixed source or eigenvalue problem ; transport (collision probability or Sn) theory or diffusion theory. Moreover, smearing and collapsing of macroscopic cross sections are separately done by the user's selection. And a special attention is paid for double heterogeneity . Various techniques are employed to access the data storage and to optimize the internal data transfer.

Benchmark calculations using the SRAC system have been made extensively for the Keff values of various types of critical assemblies (light water, heavy water and graphite moderated systems, and fast reactor systems). The calculated results show good prediction for the experimental Keff values.

Keywords: Cell Calculation, Core Calculation, Neutronic Calculation, Group Constants, Thermal Reactor, Collision Probability Method, Sn Transport, Diffusion Code, Burn-up, Benchmark Calculation , LWR, MTR, VHTR, ATR, Resonance Absorption, Bondarenko, Double Heterogeneity

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SRAC：核設計と解析のための原研熱中性子炉標準コードシステム

日本原子力研究所東海研究所原子炉工学部

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1982年9月24日受理

要　　旨

SRACは熱中性子炉の核設計と解析のためのコードシステムである．このシステムは中性子断面積ライブラリーとそのための処理コード，中性子スペクトルの計算ルーチン及び種々の輸送コード，1，2，3次元拡散ルーチンや動特性パラメータ，格子燃焼ルーチンから成っている．

SRACの個々のコードの最適な利用によって，その目的に従って炉特性を精度良く予測する正確な方法，或いは計算時間の短い経済的な方法を選ぶことができる．オプションにより非斉次問題又は固有問題，衝突確率法やSN法のような輸送理論又は拡散理論を選ぶことができる．二重非均質性への配慮から断面積の空間平均と縮約は別々に行うことができる．データの収納や内部データの引き渡しにも種々のテクニックが用いられている．

SRACを用いたベンチマーク計算がいろいろの臨界集合体で行われ，計算結果は実験値のKeffと良い一致を示している．

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*** I. S. L.

Program Abstract in NEA DATA BANK format

1. Name : SRAC

2. Computer for which the program is designed and others upon which it is possible : FACOM M-200

3. Nature of physical problem solved : Cell calculation including burn-up where care is paid to solve double heterogeneity even in resonance integral, and core calculation. Applicable to any type of thermal reactor.

4. Method of solution : Collision probability method , 1D and 2D Sn for cell calculation; 1D, 2D, and 3D diffusion for core calculation.

5. Restriction on the complexity of the problem : Not more than two different mixtures in a cell which have resolved resonance levels

6. Typical running time : It varies by the number of energy group, geometry option, and with or without burn-up calculation.

7. Unusual features of the program : PDS files are used to transfer and keep the libraries, macroscopic cross sections, neutron fluxes, etc..

8. Related and auxiliary programs : PROF-GROUCH GII to update the fast neutron library. TIMS to update the unresolved resonance data in the library. SRACTLIB to update the thermal neutron library. PDSEDT and PDSEDGRP to control and edit of PDS files.

9. Status : Under testing

10. References : JAERI-1285

11. Machine requirement : 768 KB core memory for normal work area of 60,000 words. Plotter facility

12. Programming language used : FACOM Fortran IV

13. Operating system or monitor under which the program is executed : FACOM OS IV

14. Any other programming or operating information or restrictions : A PDS file (Partitioned Data Set) is used with undefined record format. A member contains an array of binary data. An assembler routine RWPDSF enables read, write, rename, and delete member by Fortran statements

15. Name and establishment of author : K. Tsuchihashi et al, JAERI Tokai establishment Tokai-mura Japan

16. Material available : the present report

Foreword

A number of computer codes have been introduced from foreign countries and also developed at Japan Atomic Energy Research Institute(JAERI) for thermal reactor analysis. Unfortunately, these codes have been used separately at the individual sections or laboratories on the various usages of methods adopting different nuclear data libraries. This situation has been occasionally introduced some troublesome problems at safety licensing stage, for instance. Hence, the necessity of standard analysis code system has been persisted for a detailed estimate of reactor characteristics, safety analysis, fuel cycle strategy etc.. On the other hand, the standardization under state of the art has been thought to be possible at the present state where much knowledge has been accumulated both on experimental and theoretical fields.

The JAERI thermal reactor standard code committee was started on July 1978 at The Tokai Establishment for the purpose of standardizing the data and method used in thermal reactor design study. Three working groups were established under the committee and, since then, have played an active part in standardizing the data and method that can be commonly used with high reliability in the design and analysis of a variety of thermal reactors . This standardization was thought of being essential to safety analysis of thermal reactors, detailed design of VHTR, reconstruction plan of research reactors, establishment of nuclear energy strategy etc.. Here, in setting forward the standardization, the intent was to enhance the development efficiency of the code system and to generalize the code system under a new viewpoint based on the technological level of this age and under an unified control of the committee.

The standard reactor analysis code (SRAC) system has been developed during the four years of active work under one of the working groups, the nuclear design working group. Benchmark calculations using the SRAC system have been made extensively for the Keff values of various types of critical assemblies (light water, heavy water and graphite moderated systems, and fast reactor systems). The SRAC system has been successfully applied to the works for the reconstruction plan of The Japan Research Reactor-3 (JRR-3) and for the reduction of fuel enrichment of JAERI research reactors and Japan Material Testing Reactor (JMTR). Its application has been also made for the reconstruction plan of Semi-Homogeneous Experiment (SHE) assembly for Very High Temperature Gas-Cooled Reactor (VHTR) development.

The extension work of the SRAC system continues for fuel cycle calculation. Moreover, the nuclear design working group makes a plan to amalgamate the SRAC system with the fruits of other two working groups.

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I. General Descriptions

The SRAC system is designed so as to have a unique feature by positively incorporating the data and methods which have been developed and verified in JAERI. Furthermore, this system is designed to allow a variety of usages described as follows :

1) The fundamental group constant library has been produced mainly from the ENDF/B-IV nuclear data file (Ref.(1)) with the energy group structure of 107 groups (45 groups for thermal and 74 for fast energy ranges, respectively, with 12 overlapping groups). The library holds the Bondarenko type table (Ref.(2)) for resonance shielding factors and keeps, additionally, the resonance parameters on the energy range between 130.7 to 0.4 eV for the optional use of intermediate resonance approximation (Refs.(3, 4)) or the rigorous calculations using the collision probability method (Ref.(5)).

The thermal group-to-group transfer matrices are compiled on the eleven fixed-temperatures. The macroscopic cross sections used in cell calculations are generated on any of the given temperatures. The group constants for core calculations can be obtained on arbitrary temperatures by the linear interpolation of the resulting effective cross sections.

2) The user must construct his own library from the fundamental library for the desired nuclei and temperatures with his own energy group structure, where the thermal cut off energy can be chosen from the overlapping energy range between the thermal and fast energy regions.

3) The SRAC system can solve the fixed source problem in a partial energy range, which is mainly used for cell calculation, and can also solve the eigenvalue problem in the whole energy range which appears in cell and/or core calculation.

4) A variety of the transport codes are available for cell calculations (collision probability method with 14 types of geometries (Ref.(6)), 1D-SN (Ref.(7)), 2D-SN (Ref.(8)). The one space-point solution by P_1 or B_1 approximation (Ref.(9)) is also available after smearing the cross sections.

5) The 1-, 2- and 3-D diffusion codes (Ref.(10)) in addition to above mentioned transport codes are available for normal and adjoint flux calculations.

6) Smearing and/or collapsing of macroscopic cross sections is done separately by the user's selection.

7) Double heterogeneity such as the grain effect in the HTGR fuel rod or the pin rods in a channel box can be treated by utilizing smeared cross sections obtained from homogenizing the microscopic heterogeneity into the macroscopic heterogeneity.

8) The several PDS (Partitioned Data Set) files (Ref.(11)) are used for the data storage. The built-in FACOM utility program is used for the file control (LIST DIRECTORY, DELETE, CONDENSE, COPY, RENAME), and a few service programs are ready to read/write the contents of these files.

By making the best use of the specific character of the SRAC system, the user can select either the exact method for accuracy assessment or the economical method aiming at a shorter computer time, depending on the purpose of study.

In this report, descriptions are given at first on an outline of the SRAC system. Successively described are instructions to the user about input data requirements for the SRAC system and its auxiliary codes , user information needed in use of the SRAC system, structure of I/O file, formulations used in the methods adopted, dictionaries and sample I/O.

I.1 Calculational Scheme

```
STEP NO.   START A CASE
             |
1            +---READ OPTION CONTROL (ANY CASE)
             |   and FILE CONTROL (FIRST CASE ONLY)
2            +---COMPOSE USER FAST LIBRARY (FIRST CASE ONLY)
3            +---COMPOSE USER THERMAL LIBRARY (FIRST CASE ONLY)
4            +---READ GEOMETRY FOR COLLISION PROB. METHOD
5            +---READ GEOMETRY FOR SN TRANSPORT
5'           +---READ GEOMETRY FOR DIFFUSION THEORY
6            +---READ MIXTURE COMPOSITION
7            +---COMPOSE MACRO X-SECTION OF FAST ENERGY RANGE
8            +---COMPOSE MACRO X-SECTION OF THERMAL ENERGY RANGE
10           +---MODIFY RESONANCE X-SECTION BY IRA
11           +---CALC FAST FLUX AS FIXED SOURCE PROBLEM
12           +---UPDATE USER RESONANCE X-SECTION LIBRARY
13           +---MODIFY RESONANCE X-SECTION BY ULTRA- FINE
             |   RESONANCE INTEGRAL BY COLLISION PROB. METHOD
14           +---MIX FAST X-SECTION
15           +---CALC THERMAL FLUX AS FIXED SOURCE PROBLEM
16           +---MIX THERMAL X-SECTION
17           +---SOLVE ONE POINT SPECTRUM BY P1-B1 APPROXIMATION
18           +---CONDENSE FEW GROUP X-SECTION
19           +---CONCATENATE FAST AND THERMAL X-SECTION
20           +---CELL BURN-UP CALCULATION
             |
             |
21           +---CALC FLUX AS EIGENVALUE PROBLEM
22           +---MIX X-SECTION
23           +---CONDENSE FEW GROUP X-SECTION
24           +---PREPARE X-SECTION FOR EXTERNAL CITATION CODE
             |
             END OF A CASE
```

Fig.I-1 SRAC FLOW DIAGRAM

In **Fig.I-1**, a flow diagram of the SRAC is shown by step the number of which appears on the output of SYSPRINT file to follow the flow. All steps except step 1 is optionally called by the option control entered in on the top of a case. This diagram and the function of each step will be detailed in **IV.3** the program flow diagram.

Here we shall explain a flow assuming a typical example.

Specification of the example

Cell calculation for a pin rod lattice cell
in fixed source problem
by collision probability method
where resonance integral by IRA approximation
Core calculation for simplified core of a homogenized active core
surrounded by reflector
in eigenvalue problem
in multigroup energy group structure
by one-dimensional diffusion theory calculation
Condense of macroscopic cross sections for external CITATION

The following steps are traced to execute the above example as;

Step 1 read the option and file control
Step 2,3 prepare the user fast and thermal libraries by specifying the energy group structure, nuclides, temperatures
Step 4 read the geometry of the pin rod cell for collision method
Step 6 read the geometry of the core for one-dimensional diffusion calculation
Step 6' read the edit control for the macroscopic cross sections in format of CITATION
Step 10 modify cross sections in resolved resonance energy range by IRA approximation
Step 11 solve linear equations in the fast energy range by collision probability method
Step 14 homogenize cross sections using the spatial distribution of neutron flux obtained by cell calculation
Step 15 solve linear equations in the thermal energy range by collision probability method
Step 16 homogenize cross sections using the spatial distribution of neutron flux obtained by cell calculation
Step 17 calculate Kinf,Keff by a point model
Step 19 concatenate fast and thermal cross sections into a set for succeeding core calculation
Step 20 solve an eigenvalue problem for a core calculation by one-dimensional diffusion routine
Step 23 condense cross sections into those of few group structure using neutron fluxes obtained in the above core calculation
Step 24 store macroscopic cross sections in catalogued file for external use of CITATION

I.2 Data Libraries

There are two kinds of nuclear group constants libraries : One is named 'Public'(P) library, which is the fundamental one in the SRAC system, and the other is 'User'(U) library. The U-library is used as the user's own library after the group constants in the P-library are collapsed into the proper group number for the necessary number of nuclei. All the calculations can be started from the U-library once after it is created. The addition of new nuclides to the U- library can be readily made.

The present P-library was produced by processing the nuclear data files ENDF/B-III (Ref.(12)) and IV (Ref.(1)). The nuclides cited from these files are listed in Dictionary VII.3. Some evaluation works (Ref .(13)) were also made for Hf isotopes, which were not included in

the ENDF/B-IV. The P-library is planned to be replaced by a library based on the JENDL-2B nuclear data file (Ref.(14)).

I.2.1 Energy Group Structure

The energy group structure of the P-library is shown in Dictionary VII.4, where the energy range from 10 MeV to 1.8554 eV is divided by the lethargy width of 0.25 into 62 groups, that from 1.8554 to 0.41399 eV by the lethargy width of 0.125 into 12 groups and the energy region below 0.41399 eV is divided with the equal velocity width of 270m/sec into 33 groups. The structure corresponds to those widely adopted in the world for thermal and fast reactor analysis, hence has a convenience when comparison is made among the cross section libraries and/or the calculated results.

The P-library is classified into the library corresponding to the fast neutron energy region (En $\geqq$ 0.41399 eV) and that to the thermal (En $\leqq$ 3.9279 eV), with the overlapping region (3.9279 eV $\geqq$ En $\geqq$ 0.41399 eV). This overlapping region is provided for the optional choice of the thermal cut off energy to take account of the thermal neutron spectrum shift due to higher temperature or to leave the option for the treatment of the first resonance level of ^{240}Pu .

I.2.2 Group Constants in Fast Neutron Energy Region

The fast neutron energy region is defined as the range from 0.41399 eV to 10 MeV, which includes all the energy ranges except for the thermal neutron energy range described in Sect.I.3. The group constants in this energy region are arranged in the form of the cross section set of the Bondarenko type (Ref.(2)), that is, the self- shielding factors are given for scattering, removal, capture, fission and transport cross sections, which are tabulated by temperature T and admixed cross section σ_0 . This library carries also the standard spectrum for collapsing the P-library into the U-library. The angular dependence of elastic scattering is taken into consideration up to the P_1 component, for which the shielding factor is presently not prepared. The present library was produced by use of PROF GROUCH-GII code (Ref.(15)) and TIMS-1 code (Ref.(16)).

I.2.3 Resonance Parameters and MCROSS Library

The effective resonance cross sections in the second resonance range defined in Sect.I.3 are, on simple treatment, calculated by the combined use of the intermediate resonance approximation (IRA) and the table-look-up method of the resonance shielding factors, while those in the first resonance range are always obtained by the method based on the narrow resonance approximation (NR), which is widely used in fast reactor analysis. The resonance parameters needed for the IRA are accommodated in the fast group constants library. In Dictionary VII.3 , the assignment IRES = 1 is given to the nuclides having the parameters.

The SRAC system provides another option, by which the effective resonance cross sections in the second resonance region are calculated with the collision probability method using the ultrafine energy group structure of 4600 points (Dictionary VII.4). The resonance cross

section library needed for the ultrafine group calculation is given, for every temperature used, by the MCROSS-2 code (Ref.(17)) and stored in the PDS files (See Sect.I.5) . Here in general a multilevel formalism is used for the cross section representation (Ref.(18)), and the necessary multilevel parameters are also prepared for the energy range($\leqq$ 130.07 eV) in the fast group constant library.

I.2.4 Group Constants in Thermal Energy Region

The thermal neutron scattering library consists of the matrices with fixed dimension of 45 energy groups. For moderator materials, the scattering matrices were constructed from the scattering law $S(\alpha,\beta)$ and tabulated on the discrete values of temperature. Especially for the moderators for which the chemical binding effects are important, the coherent elastic scattering cross section calculated by the HEXSCAT code (Ref.(19)) was added.

The self-shielding factors were prepared for the fission and capture cross sections of the nuclides whose resonance levels exist in the thermal neutron energy region. For some of such nuclides, a representative matrix without temperature dependence was assigned for all the temperatures used in the tabulation of the thermal library, because of their smaller contributions to neutron energy transfer in practical reactor calculation. Here, the weighting spectrum used for collapsing the P-library into the U-library was assumed to be (the Maxwellian distributions corresponding to the above temperatures + 1/E spectrum), commonly for all the nuclides concerned.

I.2.5 Nuclear Data for Depletion Calculation

Several fission product models (Refs.(20 - 24)) for BWR lattice calculation have been proposed, which consist of a few explicit nuclides and several pseudo groups representing the residual absorption. In the SRAC system, three models are available, depending on the purpose of burnup calculations.

The first model of Garrison and Roos (Ref.(20)) consists of two explicit nuclides (Xe-135 and Sm-149) and three pseudo groups (rapidly saturating, slowly saturating and non-saturating fission products). This model is very useful for economical calculations of cell burn-up. The group cross sections were produced from ENDF/B-II for three pseudo fission products and from the ENDF/B-IV (Ref.(1)) for two explicit nuclides, respectively. The nuclear data of ENDF/B-IV are used for half-lives and fission yields for these nuclides.

Secondly, a detailed model by Iijima et al.(Ref.(22)) is available to predict the burnup reactivity change with high precision. This model consists of 45 explicit nuclides and one pseudo group. The nuclear data of ENDF/B-IV were used for half-life and fission yield data. The group constants of the explicit nuclides were mostly produced from ENDF/B-IV, except for Cd-113,Xe-133,Cs-134 and pseudo fission product.

Thirdly, the VSOP-chain model (Ref.(24)) developed in KFA Jeulich is available. In this model, four branches can be selected, which consist of 27, 32, 37 and 42 explicit nuclides, respectively. The group cross sections for these nuclides were produced from ENDF/B-IV. For the

fission product yields, the data recommended in the VSOP scheme are used.

I.3 Definition of Energy Range

The method producing the cross section library, the methodology used and the way to divide spatial meshes in the SRAC system are in general changeable depending on the energy range. Here the energy range is defined as follows :

i) Fission Energy Range

This range corresponds to the fast energy region higher than the threshold energy of fertile nuclei , where the weighting spectrum used for producing the P-library is assumed to be fission spectrum. The energy averaged spectrum in each group is used as the standard one for the P-library collapsing into the U-library. For lower enriched fuel rods with larger radius, fast fission effect and its heterogeneity effect are important in this energy range.

ii) Smooth Energy Range

Since the fluctuations of the various reaction cross section are rather small in the energy range below about 1 MeV, the neutron energy spectrum is smooth, hence the spatial distribution can be assumed to be flat. Though there happen to be some small variations in the neutron spectrum due to the resonance scattering of light and medium weight nuclides, this effect is not so important in thermal reactor. The weighting spectrum is assumed to be $1/(E\Sigma_t(E))$, as well as in the following two energy ranges. The energy averaged spectrum is again used as the standard one.

iii) The First Resonance Range

Below about 50 KeV, fine structure appears in the neutron spectrum due to isolated and/or statistical resonance levels of heavy isotopes ,and the Doppler effect must be taken into account. For heavy resonant nuclides, an exact calculation is made for the resonance shielding- factor production using TIMS-1 code (Ref.(16)). There is however no special difference in programming between the smooth and resonance energy ranges in the SRAC system.

iv) The Second Resonance Range

This energy range corresponds to be lower resonance energy region where are many sharp and strong resonance levels of fission and fertile isotopes. A special attention must be paid for this range in thermal reactor analyses, because most of resonance absorption occur in these strong resonances. The resonance shielding-factors for heavy resonant isotopes are obtained again by the exact calculation. The upper energy boundary of this range is fixed to be 130.07 eV(u = 11.5), while the lower energy boundary is selected by the user from one of the group energy boundaries of the SRAC library between 3.9279 and 0.41399 eV, depending on the problem under study. The lower boundary is located in the overlapping region previously mentioned. Here, for a simple calculation the intermediate resonance approximation is used for

calculating the effective resonance cross sections, and the ultrafine spectrum can be also calculated by use of the collision probability method when higher accuracy is needed.

v) Thermal Neutron Energy Range

Since the thermal scattering matrices in the library are prepared only for the given temperatures, the calculations of the thermal neutron spectrum are restricted in the temperature that can be treated, hence some interpolations would be needed for a detailed calculation of temperature coefficients. The scattering up to the second resonance range can be taken into consideration only with the few group calculation after the homogenized cross sections are obtained.

I.4 Optional Use of Transport Codes and Their Usage

Several kinds of optional paths can be available for neutron transport and/or diffusion calculations. The path based on the collision probability method can treat the 14 types of geometries shown in **Fig.II.3-1** . The SN path adopts the ANISN code (Ref.(7)) for 1D calculation and the TWOTRAN code (Ref.(8)) for 2D, respectively. On the other hand, the diffusion code CITATION (Ref.(10)) is generally used for the diffusion-calculation path, though the 1D code TUD (Ref.(6)) is additionally prepared for the sole purpose of 1D calculation. Any selection from these pathes is possible for each energy range except for the second resonance energy range in cell calculation. The ultrafine group calculation based on the intermediate resonance approximation (IRA) of the resonance absorption in the second resonance range always uses the collision probability method.

The neutron spectrum calculation for smearing and/or collapsing of macroscopic cross sections can be made by choice of a proper sequence of the paths. Moreover, the P_1 or B_1 approximation based on the fundamental mode assumption is available for the multigroup collapse into few group after smearing the multigroup cross sections.

Particularly for cell calculations, various space regions and meshes are defined to enhance the calculation accuracy or to save the computer time, as the needs of the case demand.

i) Sub-region

This is the fundamental region that is formed by the lines or circles used to define the geometry under consideration for the collision probability program.

ii) T-region

Each 'T'-region consists of a few sub-regions when the neutron flux distributions in the sub-regions are the same due to geometrical symmetry or when those in the adjacent sub-regions can be assumed to be the same due to the smallness of their cross sections or of their volumes. This 'T'-region is used for the calculation of the spatial fine structure of the neutron flux in the thermal energy range.

iii) R-region

In some cases of the neutron flux calculations in the fission or resonance energy range, it is not always necessary to divide the geometry into so many meshes as in the thermal energy range. Such being the case, several T-regions are collected to form the R- region.

iv) **X-region**

This region formed by gathering some of the R-regions is used for calculation and output of the average cross sections. For usual cases, one X-region corresponds to one unit cell. On the other hand, for the special cases such as the spatial dependence of the effective cross sections is needed to be considered, each of the X- region is recommended to make one-to-one correspondence to each R- region, that is, the same geometry is used both for the X- and R-region representation. Moreover, some of the R-regions may not be included in any of the X-regions for the case where the concept of 'super cell' is used.

v) **M-region**

This M-region is formed by collectively gathering some R-regions which have the same composition. On the calculation of the background cross section σ_0 based on the narrow resonance approximation (NRA) or IRA, the collision probability needed is calculated on the geometry represented by the M-region.

I.5 Data Storage in PDS File

The storage and search of the variety of data is carried out by using the PDS (Partitioned Data Set) files (Ref.(11)), where the Member Name of data is designated by 8 letters of 36 alphanumeric characters (figures 0-9 and Roman letters A-Z). The items of the 8 letters are described in Chap. V. The built-in FACOM utility program is used for file control, and a few service programs are ready to read/write the contents of these files. The built-in FACOM utility programs can be also used for file control, and a few service programs are ready to read/write the contents of these files.

II Input Data Requirements for SRAC

All input data for the SRAC code system except for the few routines are read in a free format by the REAG routine which accepts a series of data prepared by EBCDIC characters in any form, length, occurrence on any column on the fixed logical record. This function permits the user punching the data without notifying where the data column is on a card image record and facilitates deletion and insertion of a part of data in a sequential array.

How to use the free format

1) Columns 1 to 72 are used as data field.

2) A word (integer or floating number) is separated by a blank, a comma, or sign codes '+' '-' from the next word.

3) A floating number may be entered by F type or E type; the latter needs the exponent code 'E' at the beginning of exponent.

4) A word must be completed on a card image record.

5) One or more blank columns between sign code and digit code are neglected.

6) One or more blank columns between the last digit of mantissa and the succeeding exponent code character 'E' are neglected.

7) Any character including blank entered in A type variable is taken as data. Concerning A type variables, there is no free format. All A type variables which appear in the input are designed to be entered starting at the first column of the record.

8) An integer before the code '(' is taken as the repetition number of a data or a series of data which are enclosed by '(' and ')'.

9) An integer before '*' is taken as the number of accumulations , and the data value after '*' is taken as increment to the previous data.

10)The character '/' is taken as the termination code character. If the termination code character is encountered, a check whether or not the array length meets with one required by the program. However the '/' on the new record after entering required data on the previous record causes error because the read-in is finished on the previous record without the termination code, then such '/' is read on the head of next read-in.

11)The character '$' is taken as the end-of-record code character. If the entries are not yet finished at this code, the remaining data follow on the next record.

12)If encountered the end-of-record code or the termination code character, the remaining columns on a card image record are released for comment.

13)When both of an integer array and a floating point number array are read in a BLOCK, repetition or accumulation function must close within each type of array.

Examples

```
A record ' 1 0.0002  3. E-3  -.4E3 $ COMMENT '  is accepted as
         ' 1.0  2.0E-3 3.0E-3  -4.0E+2          '

A record ' 5(2.) 2*1                 $ COMMENT '  is accepted as
         ' 2.0 2.0 2.0 2.0 2.0 3.0 4.0          '

A record ' 2+2-3  E1+1  2(5 6)                    '  is accepted as
         ' 2.0 2.0 -30.0 1.0 5. 6. 5. 6.         '
```

Although data type (A, I, or E) of variable or array in the following description is not always mentioned, the user can recognize A-type data by finding Hollerith count after the variable name as (4H), and concerning numerical data the user can discriminate I-type or E-type by the first character of the variable name whether if it is one of characters I through N. For any type of numerical data the REAG routine reads the data as floating type, then converts into integer type if required so that the user has not be so careful about data type.

The term BLOCK appearing in the descriptions denotes one or a series of data required by a Fortran read statement which may be entered on any number of cards. The use of the termination code '/' is recommended to have suitable message if the data length is mismatched. The data length required in BLOCK is shown as /20/ or /NRR/.

II.1 General Control and Energy Structure Specifications

BLOCK 1

CASENAME(4H) Case identification which is used as the former half of MEMBER names of the spatially averaged macroscopic cross sections in the MACRO and MACROWRK files, and of the region-wise neutron flux distributions in the FLUX file. Enter blank CASENAME after the last case to terminate the SRAC execution.

BLOCK 2

TITLE(72H) Case description

BLOCK 3 Integers to specify the options /20/

IC1 Indicator to call the collision probability routine in any usage
=0 skip
=1 call

IC2 Selection of the routines to solve the fixed source problems in each of the fast and thermal energy ranges
=0 none of routines is used
=1 the collision probability method is used
=2 the ANISN (one dimensional SN) is used
=3 the TWOTRAN (two dimensional SN) is used
=4 the TUD (one dimensional diffusion) is used
=5 the CITATION (multi-dimensional diffusion) is used

IC3 Selection of the process to get the Dancoff correction

factors which are used in the two steps; first for the heterogeneous effect on the admixture cross sections in the interpolation of resonance shielding factors; second for the IRA(Intermediate Resonance Approximation) of absorption calculations of resolved resonance levels
=0 the input value in the mixture specification is used
=1 the collision probability method is used
=2 the empirical formula is used (not yet installed in SRAC)

IC4 Indicator for the calculation of the thermal energy range
=0 the thermal range is excluded (for FBR calculation)
=1 the thermal range is included

IC5 Selection of the process for resonance absorption in the resonance II range
=0 the table look-up method
=1 the IRA (Intermediate Resonance Approximation)
=2 the PEACO (ultra fine group calculation by the collision probability method
Restrictions : The IRA routine works for unique resonant R-region. The PEACO does not work for more than two resonant mixtures in a cell.

IC6 Indicator to get the flux-volume averaged cross sections for each of the energy ranges
=0 skip the averaging process
=1 call the averaging process specified by IC7 following

IC7 Selection of the process to get the spatial distributions of fluxes for each of the energy ranges. This selection is prepared for computer time saving by avoiding the calculation of spatially flat and nearly asymptotic neutron spectrum forseen by the experienced user

Energy Range	*	Fast Fission	Smooth	Resonance I	Resonance II	Thermal
Upper	*	10 MeV	.82 MeV	67.4KeV	130. eV	thl cut
Lower	*	.82 MeV	67.4KeV	130. eV	thl cut	10^{-5} eV
=0	*	ViWg	ViWg	ViWg	ViWg	calc
=1	*	calc	ViWg	ViWg	ViWg	calc
=2	*	calc	calc	ViWg	ViWg	calc
=3	*	calc	calc	calc	ViWg	calc
=4	*	calc	calc	calc	calc	calc

where 'ViWg' denotes the fluxes are assumed as the product of the asymptotic spectrum installed in the library and the volume of the mixture. 'calc' denotes the routine specified by IC2 will be used to calculate the flux distribution. If the fine resonance calculation is selected by IC5=2, the above indication for the resonance II range is taken as 'calc'. Note that in case where IC7=1,2 or 3 is specified, the condensed group structure can not override the energy boundaries of the energy ranges which are fixed in the library; i.e. the group

boundaries of the condensed energy structure have to coincide with the boundaries of the ranges if they are the boundaries of 'calc' and 'ViWg'.

IC8 Indicator to call MCROSS routine to create or to update the microscopic resonance neutron file
=0 skip
=1 call
If IC8=1, the user has to prepare his own file as DD=MCROSS to write the data.

IC9 Indicator to call HOMOSP routine to calculate the one point (bare) reactor neutron spectrum and K-infinitive
=0 skip
=+-1 call and P1 approximation
=+-2 call and B1 approximation
If negative value is read, P0 spectrum is used to collapse the homogenized cross sections, otherwise the spectrum of infinite cell is used.
Note: The geometrical buckling given in BLOCK4 is used in the leakage term.

IC10 Indicator to call CONDENSE routine to collapse the energy structure of the macroscopic cross sections in the MACROWRK file to put into the MACRO file before the eigenvalue calculation.
=0 skip
=1 call

IC11 Indicator to read the geometric information of the case
=0 read in the new geometry
=1 skip reading and use the same as the previous case

IC12 Selection of the routine for eigenvalue calculation
=0 end of the case ; go to the next case
=1 the collision probability method is used
=2 the ANISN (one dimensional SN) is used
=3 the TWOTRAN (two dimensional SN) is used
=4 the TUD (one dimensional diffusion) is used
=5 the CITATION (multi-dimensional diffusion) is used

IC13 Indicator to call CONDENSE to collapse the energy structure of the macroscopic cross sections in the MACROWRK file to put into the MACRO file after the eigenvalue calculation
=0 skip
=1 call

IC14 Indicator to write the macroscopic cross sections in the format of the CITATION into the PS file FT31F001 for the separate use of the CITATION. The energy structure of the cross section is decided by whether or not the routine CONDENSE was called in the case.

IC15 Selection of the process to compose (or define) the microscopic total cross sections in the resonance energy range
=1 interpolation of the self-shielding factor tabulation

of the total cross sections which originate from the inverse average of the total cross sections with fine structures. The in-group scattering cross sections are adjusted to meet the neutron balance.

$$\sigma_{t,g} = \int_{\Delta E_g} \varphi(E)dE / \int_{\Delta E_g} (\varphi(E)/\sigma(E))dE$$

=2 summation of all partial reactions

$$\sigma_{t,g} = \sum_{reaction,r} \sigma_{r,g}$$

IC16 Indicator how to form the macroscopic transport (collision) cross sections which are used in the isotropic transport routine

=0 the extended transport approximation (see Sect.VI.3))

$$\Sigma_{tr,g} = \Sigma_{0,g} - \sum_{g'=1}^{G} \Sigma_{1,g-g'}$$

however, in the resonance shielding calculation

$$\Sigma_{tr,g} = \Sigma_{0,g}$$

is assumed

=1 P1 approximation of the multi-group calculation of the homogeneous media (see Sect.VI.3)

$$\Sigma_{tr,g} = \Sigma_{0,g} - \sum_{g'} \Sigma_{1,g'-g} J_{g'} / J_g$$

=2 B1 approximation of the multi-group calculation of the homogeneous media (see Sect.VI.3)

$$\Sigma_{tr,g} = \Sigma_{0,g} - \sum_{g'} \Sigma_{1,g'-g} J_{g'} / J_g$$

=3 average by current component of the result of anisotropic SN calculation (not yet installed)

$$\Sigma_{tr,g} = \Sigma_{0,g} - \sum_{g'} \Sigma_{1,g'-g} J_{g'} / J_g$$

Note : For any material strongly absorbing or almost empty, do not use the options other than IC16=0, or the spectrum calculation will fail to converge.

IC17 Indicator of the cell averaged diffusion coefficients into D1 position of macroscopic cross section format.

=1 inverse of the spatially averaged transport cross

section. The values by this option are set to D2 unconditionally unless the third option of this item is taken. For the homogeneous mixture this option is unconditionally taken and the values are written in the place of D1 and D2.

$$D_g = 1/3\Sigma_{tr,g}$$

=2 the isotropic components of Behrens' term of the Benoist model (Ref(35)) which are written into D1, whereas the values by the previous option survive in D2 positions in the macroscopic format (see Sect. V.4).

$$D_g = \{\sum_i \varphi_{ig} \sum_j \frac{P_{ij}}{\Sigma_{tr,j}}\} / 3\sum_i \varphi_{ig}$$

=3 the anisotropic components of the Behrens' term of the Benoist model (Ref(35)). The radial components in the cylindrical coordinate, the perpendicular components in the plane geometry are written into D1; the axial components in the cylindrical, the parallel components are written into D2 positions in the macroscopic format.

$$D_{kg} = \{\sum_i \varphi_{ig} \sum_j \frac{P_{ijk}}{\Sigma_{tr,j}}\} / 3\sum_i \varphi_{ig}$$

Note. If the option IC17=3 is not used, the data filled in the position of D2 are the values made as if the options IC16=0 and IC17=1 are specified.

IC18 Indicator to call activation calculation (not yet available)

IC19 Print control in the routines for forming the macroscopic cross sections (MACROF,MACROT,P1B1,HOMOSP,IRA, PEACO)
=0 the most brief edit
=1 or more ; the larger value prints the more fine information

IC20 Indicator to call the burn-up routines (called just after fixed source problem where all mixtures used in the problem are to be specified with nuclide composition)

BLOCK 4 /1/

BSQ Geometrical buckling(cm^{-2}) used in P1 or B1 approximation in one point spectrum calculation

Following BLOCKs from 5 through 10 are required only in the first case in a job step.

BLOCK 5 Data set specification for main PDS library files /4/

DATASET (8H)

IF any alphameric characters except eight blanks punched to the file, it means the file will be used in the job step, otherwise eight blank characters signify no use of the file.

1st for the public fast library
2nd for the public thermal library
3rd for the user fast library
4th for the user thermal library

BLOCK 6 Specification for energy group structures /4/

NEF Number of the fast neutron groups in the user fast library

NET Number of the thermal neutron groups in the user thermal library; punch 0, if IC4 = 0 in BLOCK 3

NERF Number of the fast few groups if the CONDENSE routine is called, otherwise punch 0

NERT Number of the thermal few groups if the CONDENSE routine is called, otherwise punch 0

BLOCK 7 /NEF/ required if DATASET(1) is specified in BLOCK 5

NEGF Number of the public fast groups in a user fast group

BLOCK 8 /NET/ required if DATASET(2) is specified in BLOCK 5

NEGT Number of the public thermal groups in a user thermal group

BLOCK 9 /NERF/ required if non zero NERF is specified

NECF Number of the user fast groups in a condensed fast group

BLOCK 10 /NERT/ required if non zero NERT is specified

NECT Number of the user thermal groups in a condensed thermal group

Note 1 As BLOCK 6 through BLOCK 10 is required only in the first case, the energy group structure is kept during the job step. If the CONDENSE routine is called in any of succeeding cases, the condense information NEGF, NEGT ,NECF's, NECT's have to be specified. To feed these entries, any of condense indicator IC10 or IC13 is to be non-zero.

Note 2 The following relations must hold to avoid overlapped group assignment.

Summation of NEGF(I) and NEGT(I) = 107
58 < Summation of NEGF(I) < 75
Summation of NECF(I) = NEF
Summation of NECT(I) = NET

II.2 User's Microscopic Cross Section Libraries

The following data are required only in the first case and when the public library files are specified

BLOCK 1

IDENT (8H)

Nuclide identification expressed by eight characters, composed of six tags as 'XZZMCIIT' of the nuclide to be added to the libraries. If calculation for many temperature points are required in a series of cases, the same IDENT's with different T-tag must be repeated. When the identical nuclide which is already in the library is required, no action except warning message is taken.

X-tag the tag used internally to specify physical quantities, then any character is accepted

ZZ-tag the chemical symbol of the nuclide (see Dictionary VII.1). For a nuclide denoted by one character such as H for hydrogen , the character '0' is added to complete the tag as 'H0'.

M-tag the last digit of the mass number is used to discriminate the isotopes as 'PU9' for PU-239. A few exceptional cases occur to discriminate the excited states of some fission produced isotopes as PM-148-G and PM-148-M where the special characters are used. All available combinations of ZZMC are listed in Dictionary VII.3

C-tag the chemical compound status tag to select the proper thermal scattering law (see Dictionary VII.2)

II-tag punch '00' always in this step. This tag is used to identify the effective microscopic cross sections associated to the mixture in the proper case because the same BLOCK organization is applied in the mixture specification step.

T-tag the temperature indicator effective to the thermal library (see Dictionaries VII.3 and VII.6)

BLOCK 1 is repeated until a blank card is entered. The total number in an execution must be less than 100 for storage reason. Any double entry of the same nuclide and temperature causes only warning message.

II.3 Collision Probability Method

The input of this section is required if the combination of non zero IC1 and zero IC11 is specified in BLOCK 3 of Sect.II.1.

BLOCK 1 Control integers /18/

IGT Geometry type (see **Fig.II.3-1 a-m**)

=1 one dimensional sphere
=2 one dimensional slab
=3 one dimensional circular cylinder
=4 square cylinder divided by concentric annuli
=5 square cylinder of type 4 further divided by the radiating lines of equal azimuthal angle of 22.5 degree pitch
=6 hexagonal cylinder divided by concentric annuli
=7 hexagonal cylinder of type 6 further divided by the radiating lines of equal azimuthal angle of 15 degree pitch
=8 octant symmetric square pillar divided by X-Y coordinates
=9 type 8 which further contains square arrays of pin rods
=10 type 3 which further contains annular arrays of pin rods
=11 type 10 further divided by radiating lines which may contain unoverlapping pin rod array without angular period
=12 type 6 further divided by radiating lines without angular period
=13 rectangular pillar divided by X-Y coordinates
=14 hexagonal cylinder divided by concentric hexagons

NZ — Total number of sub-regions (geometrical sub-division)

NR — Total number of T-regions (for thermal neutron calculations). A T-region is composed of one or more sub-regions

NRR — Total number of R-regions (for resonance neutron calculations). A R-region is composed of one or more T-regions.

NXR — Total number of X-regions (for smeared cross sections).A X-region is composed of one or more R-regions.

IBOUND — Outer boundary condition of the cell calculation
=0 isotropic (white) reflection
=1 periodic (mirror) reflection, but for IGT=2 (one dimensional slab) or IGT=13 (X-Y two dimensional pillar) periodic condition is taken. In the latter case any symmetric geometry has to be specified explicitly.

=2 isolated (black)
=3 fixed boundary source (not yet installed)
=-1 60 degree rotational (applicable only for IGT=12)

NX — Number of mesh intervals for X or R division

NY — Number of mesh intervals for Y or angular division

NTPIN — Total number of pin rods (effective for IGT=10,11, or 12, calculated internally for IGT=9)

NAPIN — Number of pin rods in an array (for IGT=9),or number of circles on which the pin rods are located (for IGT=10)

NCELL — Minimum number of lattice cells traced by a neutron path (effective only for IBOUND=1). This item is used to cut

off a neutron path without knowing the actual optical path length. Recommended value to this item is NCELL = 2 for the geometry of assembly type i.e. IGT= 8,9, or 12, or NCELL = 5 for the geometry of pin rod type i.e. IGT= 2,4,5 6, or 7. The larger value causes the longer computer time. The user should not be afraid of the short cut of a path by insufficient number of NCELL while a certain amount of neutrons reach the end of the path and loose the contribution . It will be recovered by the later process of normalization of collision probabilities.

IEDPIJ =0 skip print
=1 print collision probabilities

NGR Order of Gaussian integration for the numerical radial integration , ineffective for one dimensional slab (IGT=2). Recommended value is from 6 to 10 . Note that the computer time for the integration of Pij is proportional to this item. For the geometries IGT=8,9, and 13, the Gaussian integration is replaced by the trapezoidal rule.

NDA Number of division of the range IBETM (described below) degree for the numerical angular integration of Pij, significant for IGT =4,5,6,7,8,9,10,11,12,13, and 14. Recommended value of NDA is order of IBETM/2.

Total amount of NX*NGR*NDA neutron paths are traced for the two-dimensional integration. After storing the path information and before the actual time-consuming integration, the ratios of the numerically integrated volumes to the exact ones are printed out. The deviations of the ratios from unity (should be less than a few percent) predict the accuracy of the integration. The user should adjust the values of NGR and NDA so as to be accurate but not time-consuming.

NDPIN Number of annular division of a pin rod, effective for IGT= 9,10,11, or 12

IDIVP Control of sub-division by RPP's, used for IGT=9,10,11, or 12

=0 RPP's indicate the radial positions of pin rods
=1 RPP's also play the role of RX's i.e. positions of annular division
=2 RPP's divide also the pin rod regions into inner and outer regions, (inefficient for IGT=9)

IBETM Range of angular integration in degree. Punch =45 in octant symmetric geometry, =30 in hexagonal symmetry, inefficient for one dimensional geometry.

IPLOT =0 skip plotting
=1 call plotter routine for geometric mapping

BLOCK 2 /7/ integers and /6/ floating point numbers ; parameters for the iterative solution of linear equations for neutron fluxes; the values in < > shows defaulted values used when ITMINN=0 is specified.

IEDIT	Edit control =0 suppress print plus 1 print reaction balance and flux distribution plus 2 print macroscopic cross sections plus 4 print collision probabilities plus 8 print fixed source distribution
ITMINN	Maximum number of inner iterations per an outer iteration <100> for the fixed source problem in each energy range < 4> for the eigenvalue problem
ITMOUT	Maximum number of outer iterations for the eigenvalue problem < 50 >
ITBG	Minimum number of iterations before extrapolation < 5 >
LCMX	Number of iterations for testing over-relaxation factor < 5 >
ITDM	Minimum delay between extrapolation < 5 >
IPT	Control of monitor print at each iteration < 0 > =0 suppress print =1 print record
EPSI	Convergence criterion for inner iterations < .0001 >
EPSO	Convergence criterion for outer iterations < .001 >
EPSG	Extrapolation criterion < .05 >
RELC	Initial over-relaxation factor < 1.2 >
OVERX	Maximum extrapolation < 100. >
FACTOR	Under extrapolation factor < 1.0 >
BLOCK 3	/NZ/ required if NR < NZ
NREG	T-region number by sub-region
BLOCK 4	/NR/ required if NRR < NR
IRR	R-region number by T-region
BLOCK 5	/NRR/ required if NXR < NRR
IXR	X-region number by R-region
BLOCK 6	/NRR/
MAR	Material number by R-region ; sequential order in the mixture specification is used as material number
BLOCK 7	/NAPIN/ required only if IGT=10
NPIN	Number of pin rods on each ring

BLOCK 8	/NX+1/
RX	X-abscissae or radii ; RX(1)=0
BLOCK 9	/NY/ required if IGT=11 or 12 and if NY > 0
TY	Theta's
BLOCK 9'	/NY+1/ required if IGT=13 and if NY>1
TY	Y-abscissae ; TY(1)=0
BLOCK 10	/NAPIN/ required if IGT=9 or 10
RPP	X-position of pin rods for IGT=9. Radius of the rings on which pin rods are located for IGT=10.
BLOCK 10'	/NTPIN/ required if IGT=11, or 12
RPP	Radial position of each pin rod for IGT=11 or 12
BLOCK 11	/NTPIN/ required if IGT=10,11, or 12
THETA	Angular position of each pin rod in degree
BLOCK 12	/NDPIN+1/ required if IGT=9 or 10
RDP	Radii for annular sub-division in a pin rod ; where RDP(1)=0. The radii are common through all pin rods.
BLOCK 12'	/(NDPIN+1)*NTPIN/ required if IGT=11, or 12
RDP	Radii for annular sub-division of each pin rod ; where RDP(1,J)=0.
BLOCK 13	Plotter control integers /3/ required if non zero value of IPLOT in BLOCK 1 is specified
IG	Signed integer to specify the combination of required region map ;the integer is made of the summation of following integers corresponding to the kind of map. =0 none plus 1 sub-region plus 2 T-region plus 4 R-region plus 8 Material number plus 16 X-region Positive value indicates printing of assignment of region numbers in the figure ,and negative value requires only figure.
ISCALE	Indicator of the scale of figures =1 A figure of diameter of 20 cm in a screen =2 Two figures of each diameter of 15 cm in a screen =3 Five figures of each diameter of 10 cm in a screen

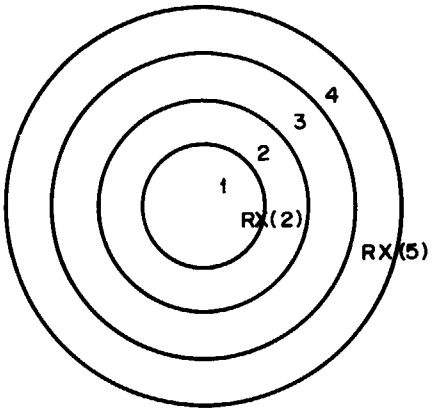
ICONT Continuation indicator
=0 followed by the next BLOCK 13
=1 the last plotting

Examples of BLOCK 13

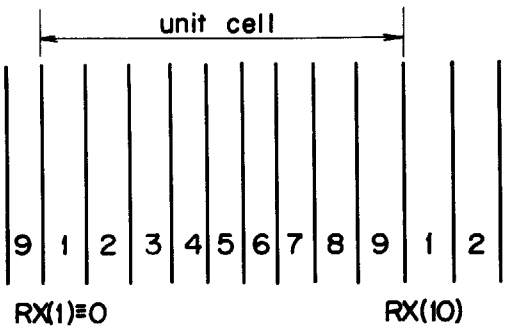
1) RECORD (31 1 0) signifies that each of Sub-,T-,R-, and X- region map and Material number map is figured in five screens with assignment of region numbers , and continued.

2) RECORD (-1 1 1) signifies that only Sub-region boundaries are figured , and return to main routine.

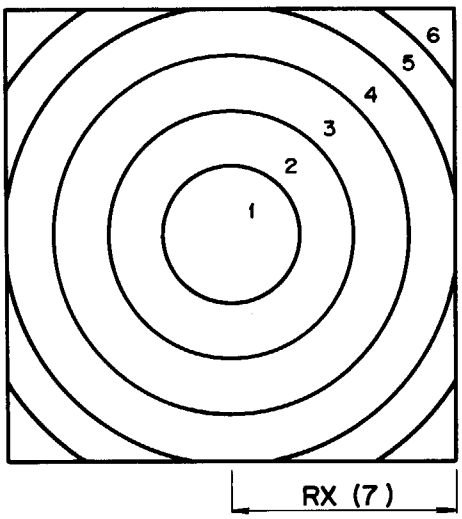
a. Type 1 (Sphere) and Type 3 (Circular cylinder)



b. Type 2 (Infiniteslab)



c. Type 4 (Square cylinder)



d. Type 5 (Two dimensional square cylinder)

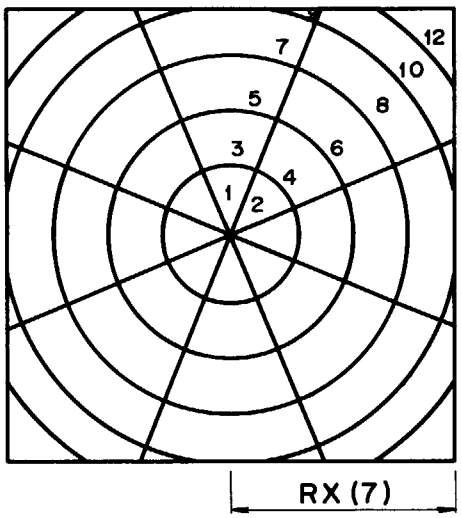
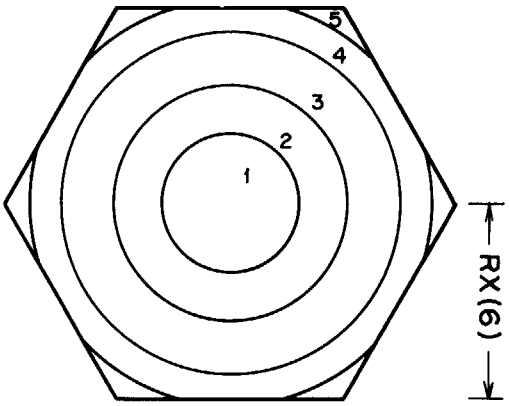
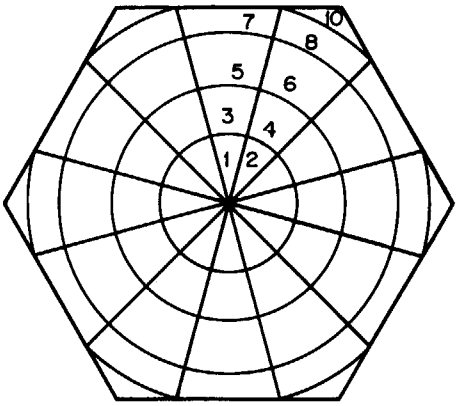


Fig.II.3-1 Geometries and sub-region numbers

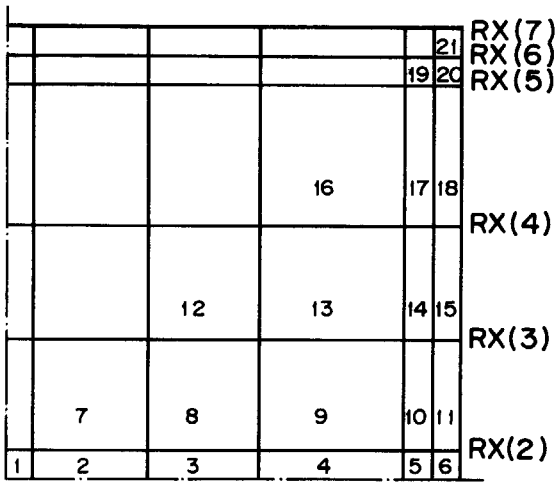
e . Type 6 (Hexagonal cylinder)



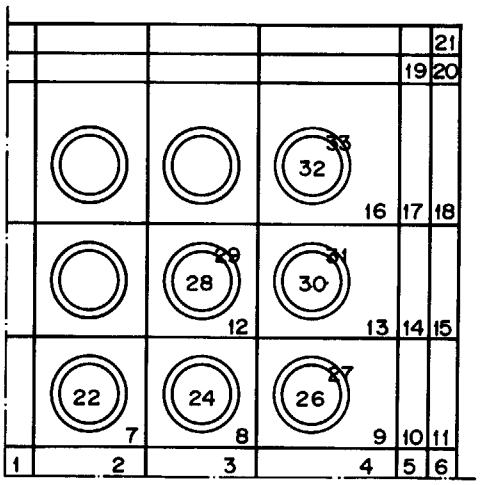
f. Type 7 (Two dimensional hexagonal cylinder)



g . Type 8 (Square assembly)



h. Type 9 (Square assembly with pinrods)



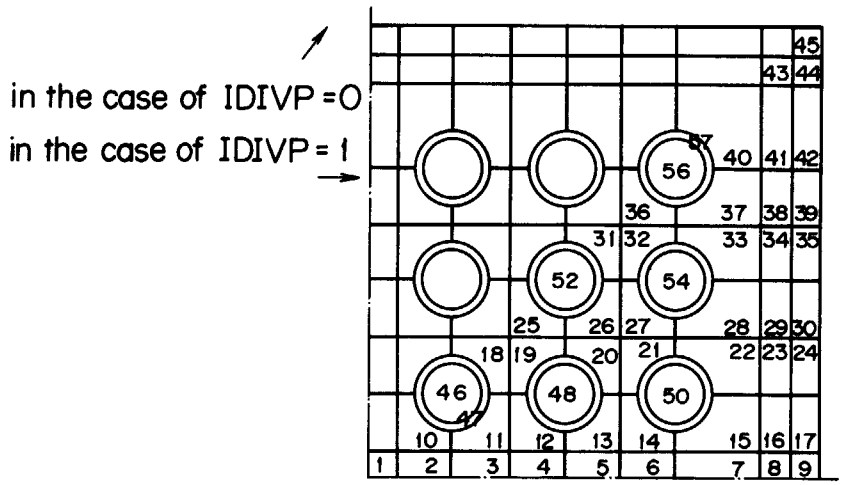
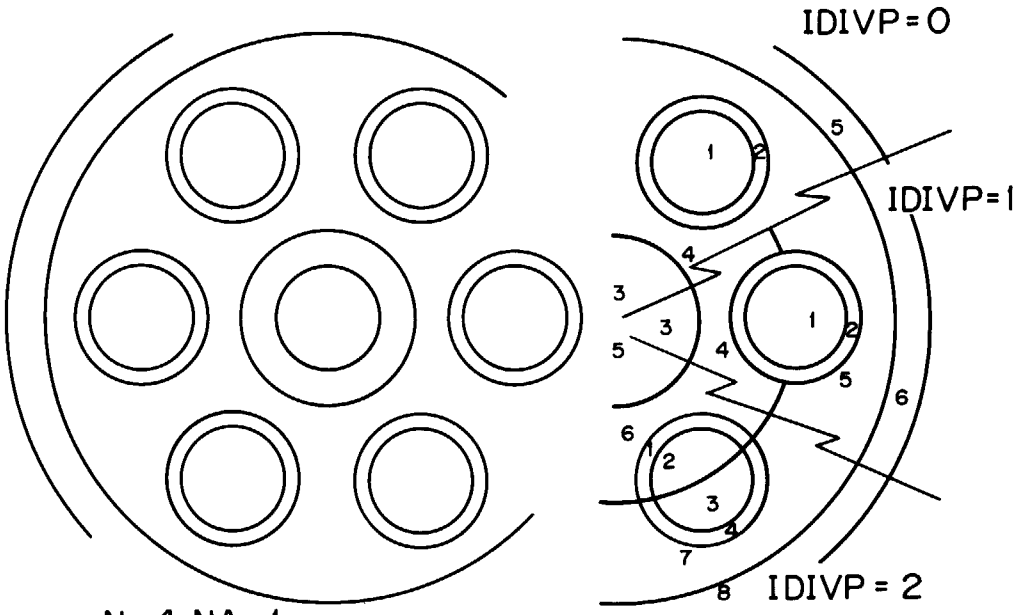


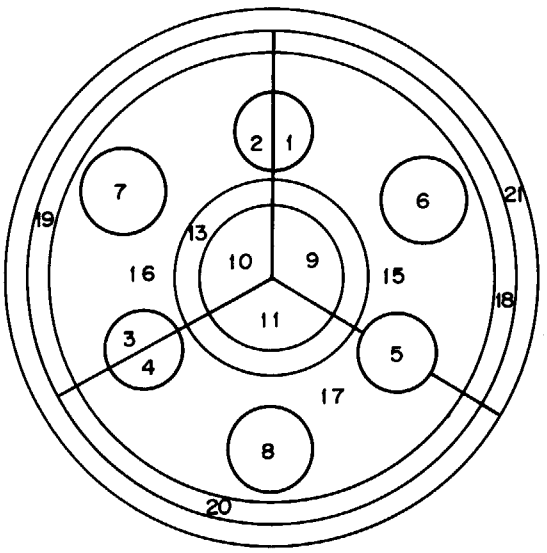
Fig.II.3-1 (continued)

i. Type 10 (Annular assembly with regular arrays of pin rods)



case N=4, NA=1
NPIN (1) = 6

j. Type 11 (Annular assembly with asymmetric pin rods)



case N = 5, M = 3
NP = 6

k. Type 12 (Hexagonal assembly with asymmetric pin rods)

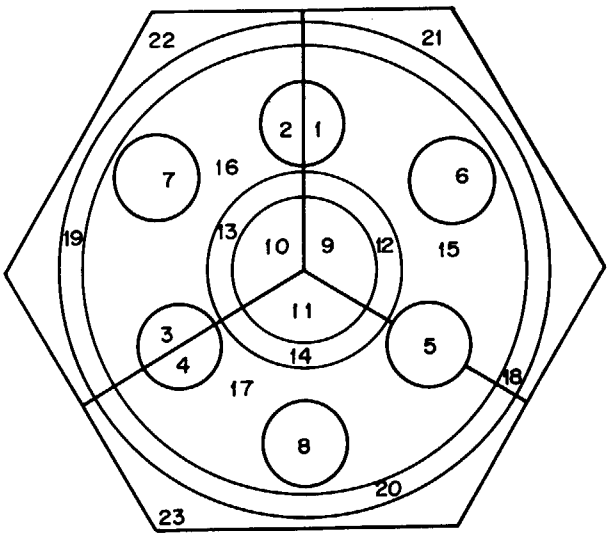
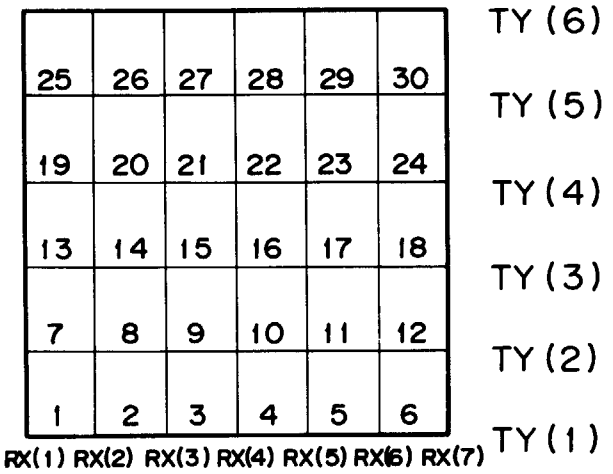


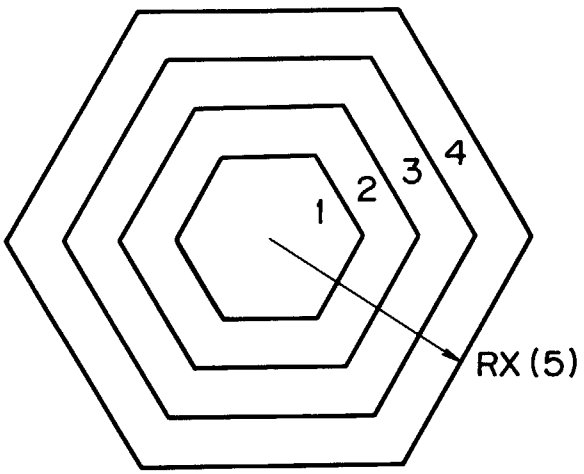
Fig. II.3-1 (continued)

l. Type 13 (XY two dimensional infinite pillar)



Case NX = 6 NY = 5

m. Type 14 (Multi-layer of hexagon)



Case NX = 4

Fig.II.3-1 (continued)

II.4 ANISN ; One dimensional SN Transport

The one dimensional SN routine ANISN can be used in either cell calculation or core calculation. The original input format (Ref(7)) is replaced by our free format described in the top of II, however the original keycodes corresponding to the array names and their orders are still kept to facilitate the reference to the original manual. The way of reading by indicating the array code of three characters and then entering the array beginning on the next logical record is not changed.

As the cross sections are prepared after reading the input of this section, several control integers relating to cross sections are automatically set at that step, then they are indicated here as internally set. And also the user will find that several functions are suppressed due to the restricted use of ANISN in the SRAC.

BLOCK 15$ Integer parameters /36/

ID
Problem ID number. If ID is greater than 1000000, disadvantage factors will be computed by group for each material which appears in the calculation.

ITH
=0 forward solution
=1 adjoint solution

ISCT
Maximum order of scatter found in any zone
=0 P0 component
=1 P1 components

ISN
Order of angular quadrature (even integer only, 2/4/6.. ; S2/S4/S6...)

IGE
Geometry
=1 slab
=2 cylinder
=3 sphere

IBL
Left boundary condition
=0 vacuum (no reflection)
=1 reflection
=2 periodic (angular flux leaving left boundary reenters in the right boundary)
=3 white / albedo (some fraction returns isotropically)

IBR
Right boundary condition, same as IBL

IZM
Number of zones or regions

IM
Number of mesh intervals

IEVT
Eigenvalue type
=0 fixed source
=1 K calculation
=2 alpha search

IGM
Number of energy group, required if BLOCKs 24$,25*,26* are read, otherwise punch =0

IHT	Position of total cross section in cross section table (internally set)
IHS	Position of self-scatter cross section in cross section table (internally set)
IHM	Length of cross section table (internally set)
MS	Cross section mixing table length (mixing function is suppressed) punch =0
MCR	Number of cross section sets to be read from cards in the original format. (suppressed) punch =0
MTP	Number of cross section sets to be read from tape (internally set)
MT	Total number of cross section sets (internally set)
IDFM	Control for density factors =0 density factors (21*) not used =1 density factors used
IPVT	Indicator of parameter EV, effective if IEVT=2 =0 no effect =1 EV as k =2 EV as alpha
IQM	=0 no distributed source in eigenvalue problem =1 distributed source formed in SRAC =2 distributed source read in execution time of ANISN
IPM	=0 no shell source =1 shell source IGT*MM =N shell source N*IGT*MM
IPP	=0 (suppressed)
IIM	Inner iteration maximum (suggested value, IIM = 35)
ID1	=0 no effect =1 print angular flux
ID2	=0 (suppressed)
ID3	=0 no effect =N compute N activity by zone
ID4	=0 no effect =1 compute activity by interval
ICM	Outer iteration maximum (suggested value for K calculation ICM=50) if fixed source ICM=1
IDAT1	=0 all data in core (other options suppressed)
IDAT2	=0 no effect

	=1 execute diffusion solution for specified group (24$)
IFG	=0 no effect (other options suppressed)
IFLU	=0 step model used when linear extrapolation yields negative flux (mixed mode) (suggested option) =1 use linear model only =2 use step model only
IFN	Control for initial guess; punch =0 =0 enter fission guess (internally set if eigenvalue) =1 enter flux guess (internally set if fixed source) =2 use fluxes from previous case (suppressed)
IPRT	=0 print cross sections =1 do not print cross sections
IXTR	=0 calculate P1 scattering constants (Legendre coefficients,suggested option) =1 read P1 constants from card (34*)
BLOCK 16*	Floating point parameters /14/
EV	Initial guess for eigenvalue
EVM	Eigenvalue modifier for first eigenvalue change.
EPS	Epsilon - accuracy desired (suggested value, EPS=0.0001)
BF	Buckling factor, normally 1.420892
DY	Cylinder or plane height for buckling correction (may include extrapolation length)
DZ	Plane depth for buckling correction
DFM1	Transverse dimension for void streaming correction
XNF	Normalization factor (If XNF=0.0, no normalization is done), Suggested value, XNF=0.
PV	=0.0, or = α_0 if IPVT=2.
RYF	Relaxation factor (suggested value = 0.5)
XLAL	Point flux convergence criterion if punched greater than zero (suggested value = 2.0*EPS)
XLAH	Upper limit for |1.0-lambda 1| used in linear search, normally =0.05
EQL	Eigen value change option, 0.001< EQL < 3* EPS
XNPM	New parameter modifier, normally, =0.75
BLOCK 00T	Termination of control numbers
BLOCK 04*	Radii by interval boundary /IM+1/

BLOCK 06* Angular quadrature weights /MM/
MM = ISN + 1 for plane or sphere
MM = ISN * (ISN + 4)/4 for cylinder

For ISN = 2, 4, 6, 8, 12, and 16, built-in constants are prepared; then no entry required.

BLOCK 07* Angular quadrature cosines /MM/

For ISN = 2, 4, 6, 8, 12, and 16, built-in constants are prepared; then no entry required.

BLOCK 08$ Zone numbers by interval /IM/

BLOCK 09$ Material numbers by zone /IZM/
(The function not to consider transverse buckling if negative number punched is suppressed)

BLOCK 19$ Order of scatter by zone /IZM/

BLOCK 21* Density factors by interval /IM/, if IDFM=1

BLOCK 22$ Material numbers for activities /ID3/ if ID3 > 0

BLOCK 23$ Cross section table position for activities /ID3/ if ID3 > 0

BLOCK 24$ Diffusion,infinite homogeneous medium calculation marks /IGM/ if IDAT2 > 0

BLOCK 25* Albedo by group - right boundary /IGM/ if IBR = 3

BLOCK 26* Albedo by group - left boundary /IGM/ if IBL = 3

BLOCK 27$ X-regions by zone to indicate the region where the cross sections are averaged by flux volume /IZM/

BLOCK 34* Pl scatter constants /JT*MM/ if IXTR = 1 , where
JT = ISCT for plane or sphere
JT = ISCT * (ISCT + 4)/4 for cylinder

BLOCK 00T Termination card to indicate the end of ANISN input

II.5 TWOTRAN : Two dimensional SN transport

The two dimensional SN routine can be used in either cell calculation or core calculation. We split the original code into two steps ; the first to read, check, and store the input; the second to execute SN calculation. The second step is called either in fixed source problems (once in fast neutron range, once in thermal neutron range) or in eigenvalue problem.

The original input format (Ref(8)) is replaced by our free format described in the top of Chapt.II. As the cross sections are prepared in later steps following this input step, the user will find that several items in the original TWOTRAN are automatized or omitted in the input.

BLOCK 1

NTITLE Number of job title card

BLOCK 2

TITLE (72H) Job title or job description, repeat NTITLE times

BLOCK 3 Control integers /42/

ITH Type of calculation to be performed
=0 direct
=1 adjoint

ISCT Order of scattering calculation. There are NM=(ISCT+1)(ISCT+2)/2 spherical harmonic flux components computed. There are not used to compute a scattering source unless some zone material identification number is negative. See IDCS below.
=0 isotropic
=N N-th anisotropic (N=1 only acceptable)

ISN SN order. Even integer only. If negative, quadrature coefficients are taken from interface file SNCONS. Otherwise (for ISN= from 2 through 16) built-in constants are used.

IGM Number of energy groups (internally set)

IM Number of rebalance coarse mesh intervals in the i-direction. See IMC and XRAD below.

JM Number of rebalance coarse mesh in the j-direction. See entries JMC and YRAD below.

IBL Left boundary condition
=0 vacuum
=1 reflective

IBR Right boundary condition
=0 vacuum
=1 reflective
=2 white

IBB Bottom boundary condition
=0 vacuum
=1 reflective
=2 white
=3 periodic

IBT Top boundary condition
=0 vacuum
=1 reflective
=2 white
=3 periodic

IEVT Eigenvalue type
=0 inhomogeneous source
=1 K calculation
=2 time absorption alpha search
=3 nuclide concentration search (suppressed)
=4 zone thickness (delta) search (suppressed)

ISTART Input flux guess and starting option
=-5 an entire scalar flux guess from FT33F001
=0 no flux guess required, but a fission guess (unity in fissionable zone) is automatically supplied
=6 a problem restart dump is read from unit FT33F001

Other options except the aboves are omitted

MT Total number of cross section blocks including anisotropic cross sections

MTPS Number of input material set from the interface file ISOTXS supplied by SRAC

MCR Number of input materials from the code dependent input file ; punch =0 (suppressed)

MS Number of mixture instructions ; punch =0 (suppressed)

IHT Row of total cross section in the cross section format ; punch =0 (internally set)

IHS Row of within-group scattering cross section in the cross section format ; punch =0 (internally set)

IHM Total number of rows in the cross section format ;
=0 (internally set)

IQOPT Option for input inhomogeneous source ; Qgijn ,
for g=1,IGM, i=1,IT, j=1,JT, n=1,NMQ
where NMQ = (IQAN+1)(IQAN+2)/2
=0 no input required; for the fixed source problem in fast neutron energy range fission neutron spectrum of spatially flat and isotropic source will be supplied automatically, for the problem in thermal neutron energy range spatially flat slowing-down source in each zone will be supplied automatically.
=1 enter an energy spectrum for each angular component n

of NMQ , spatially flat source is assumed.
=2 enter an energy spectrum and NMQ sets of spatial shape.
=3 enter first a spectrum and a shape IT*JT, then repeat for n.
=4 enter first a spectrum and an i-direction shape, and a j-direction shape, then repeat for n.
=5 the entire source is read from unit FT32F001

IQAN Order of anisotropy of inhomogeneous distributed source
=0 isotropic
=N N-th anisotropic

IQR Right boundary source to be specified as input.
=0 no source
=1 read source

IQT Bottom boundary source to be specified as input
=0 no source
=1 read source

IQB Top boundary source to be specified as input
=0 no source
=1 read source

IPVT Specification of PV; parametric eigenvalue.
=0 none
=1 K
=2 alpha

IITL Maximum number of inner iterations allowed per group

IXM I-direction zone thickness modifier (suppressed)

IYM J-direction zone thickness modifier (suppressed)

ITLIM Time limit in cpus, if an integer of seconds is punched, a restart dump is taken after this number of seconds and the job is terminated.

IGEOM Geometry
=1 X-Y
=2 R-Z
=3 R-Theta

IEDOPT Edit options.
=0 none
=1 macroscopic edit
=2 (suppressed)
=3 option 1 plus a zone relative power density edit

If IEDOPT is -1, or -3, an edit will be performed immediately if all necessary data is present.
If IEDOPT =-5, direct access to the program output section is provided, say to create an output interface file from a final dump.

ISDF Indicator to density factor input
=0 no
=1 yes

I1 Full input flux print suppression indicator
=0 no
=1 yes

I2 Final flux print indicator
=0 all
=1 isotropic
=2 none

I3 Cross section print indicator
=0 all
=1 mixed
=2 none

I4 Final fission print indicator
=0 yes
=1 no

I5 Source input print indicator
=0 all
=1 input
=2 normalized
=3 none

I6 Indicator to prepare and print coarse-mesh balance tables. The tables are for the rebalance mesh when the rebalance mesh and material mesh are different. Caution : The preparation of these tables requires an additional outer iteration after problem convergence.
=0 yes
=1 no

IANG Edit for angular fluxes. The preparation of angular fluxes requires an additional outer iteration after problem convergence as well as additional storage.
=-1 print and store
=0 none
=1 store

IMC Number of material coarse-mesh intervals in the i-th direction. When this value is non-zero the rebalance coarse-mesh as given by items IM and JM is not the same as the material coarse-mesh. The material coarse-mesh is the same as the mesh upon which all edits are done. When edits are requested and IMC is not zero, angular fluxes must be stored. See entries IDCS, XM, IHXC, and XRAD below.
Enter =0 (suppressed)

JMC Number of material coarse-mesh intervals in the j-th direction.
Enter =0 (suppressed)

IFO Interface file output is created. Total (angle-integrated) flux and SNCONS files are always created. Angular flux

file is created only if IANG is not zero.
=0 no
=1 yes

BLOCK 4 Control floating point data /10/

EV Eigenvalue guess. It is satisfactory to punch =0.0

EVM Eigenvalue modifier used only if IEVT > 1.

PV Parametric value of K for subcritical or supercritical systems or 1/v absorption.

XLAL Lower limit for eigenvalue searchs.

XLAH Search lambda upper limit.

XLAX Search lambda convergence precision for second and subsequent values of the eigenvalue.

EPS Convergence precision.

NORM Normalization factor. Total number of particles in system normalized to this number if it is nonzero. No normalization if NORM is zero.

POD Parameter oscillation dumper used in eigenvalue search.

BHGT Buckling height (not used in macroscopic input as in SRAC system).

The number of entries for certain data such as BLOCK 15 of IDCS is dependent on whether the rebalance mesh and the material coarse mesh are the same. We use the parameter MESH to indicate the difference. We take MESH.EQ.1 when IMC.GT.0 and JMC.GT.0, otherwise MESH.EQ.0.

BLOCK 5 /IM/

IHX Integer defining the number of fine mesh i-intervals in each coarse-mesh k-interval. This vector is used for both the rebalance mesh and material mesh .

BLOCK 6 /JM/

IHY Integer defining the number of fine mesh j-intervals in each coarse-mesh l-interval. This vector is used for both the rebalance mesh and material mesh.

BLOCK 7 / see option IQOPT /
Q IQOPT=0 none
=1 NQM set of IGM
=2 IGM groups of NMQ blocks of IT*JT
=3 NMQ sets of IGM+IT*JT
=4 NMQ sets of IGM+IT+JT
=5 entire source from FT32

BLOCK 8 /JT*MM/ where JT is total number of fine mesh in i-direction, equal to the sum of IHX vector , and MM is

total number of angular meshs; MM = ISN*(ISN+2)/8

QR1 Right boundary source (flux) in the in-down directions. Do not enter unless IQR = 1

BLOCK 9 /JT*MM/

QR2 Right boundary source (flux) in the in-up directions. Do not enter unless IQR = 1

BLOCK 10 /IT*MM/ where IT is total number of fine meshs in j-direction, equal to the sum of IHY vector.

QB1 Bottom boundary source (flux) in the in-up directions. Do not enter unless IQB = 1

BLOCK 11 /IT*MM/

QB2 Bottom boundary source (flux) in the out -up directions. Do not enter unless IQB = 1

BLOCK 12 /IT*MM/

QT1 Bottom boundary source (flux) in the in-down directions. Do not enter unless IQT = 1

BLOCK 13 /IT*MM/

QT2 Bottom boundary source (flux) in the out -down directions. Do not enter unless IQT = 1

BLOCK 14 /K/

XRAD Coarse k-mesh boundaries. Must form increasing sequense. When MESH.EQ.0, K = IM+1, and entries are used for both the rebalance and material mesh.

BLOCK 15 /L/

YRAD Coarse l-mesh boundaries. Must form increasing sequense. When MESH.EQ.0, L = JM+1, and entries are used for both the rebalance and material mesh.

BLOCK 16 /IC/

IDCS Cross section zone identification numbers. Number of entries IC = IM*JM . These numbers assign a cross section block to each zone defined by the material coarse mesh. If these numbers are negative, an anisotropic scattering source is calculated in the zone; but the numbers need not be negative when ISCT > 0

BLOCK 17 /IT/

XDF Radial fine-mesh density factors. Do not enter if ISDF.EQ. 0

BLOCK 18 /JT/

YDF — Axial fine-mesh density factors. Do not enter if ISDF.EQ.0

The effective cross sections at fine mesh point(I,J) are cross sections for the point defined by IDCS array multiplied by the factor XDF(I)*YDF(J).

BLOCK 19 /2/

NEDS — Integer defining number of edits to be performed. Do not enter unless 0 < |IEDOPT| < 5

MN — Integer defining number of microscopic activities to be computed. (suppressed)

BLOCK 20 /2/ required if NEDS > 0

NZ — Integer of edit zones.

NORMZ — The zone to which the power density is normalized. (NORMZ is not used unless IEDOPT = 3 or 4)

BLOCK 21 /IZ/

NEDZ — Integers defining which edit zone each coarse mesh material zone is in. Number of entries, IZ, is IM*JM. Caution: The edit blocks beginning with NZ,NORMZ must be repeated NEDS times. Do not enter unless 0<|IEDOPT|<5

BLOCK 22 /IC/

IXZ — X-region numbers by zone. Number of entries IC is IM*JM.

***** Caution ****

The distributed and boundary sources; BLOCKs 7, 8, 9, 10, 11, 12, and 13 are read in the execution step of TWOTRAN; they must be positioned after the material specifications .

II.6 TUD : One dimensional diffusion

The following data are required if IC2=4 or IC12=4 is entered in BLOCK 3 in Sect.II.1.

BLOCK 1 Control integers /9/

NRMAX Number of regions

IG Geometry indicator
=0 slab
=1 cylinder
=2 sphere

IBOUND Outer boundary condition
=-1 zero flux at the outer boundary
=0 reflective
=1 zero flux in the extrapolated distance (calculated)
=2 zero flux in the constant extrapolated distance (input number XLAMD)
Note that the boundary condition at r = 0 is always set as reflective.

IGUESS Initial flux guess control
=-1 read from FT33F001
=0 uniform
=n read from FTOnF001

IPTXEC Print control for cross section
=0 skip print
=1 print

ITFLUX Print control for fine flux distribution
=0 skip print
=1 print
=4 write into FT33F001 by the statement WRITE(33)((F(I,G),I=1,NNMAX+1),G=1,NGMAX)
=5 action 1 and 4

IPTS Print control for fixed source distribution
=0 skip print
=1 print

IDOPT Selection of diffusion coefficients
=1 use D1 in the macroscopic cross section organization
=2 use D2 in the macroscopic cross section organization

NXR Number of X-regions

BLOCK 2 Integer parameters for iteration /6/

ITMIN Maximum number of inner iterations in thermal neutron energy range per power iteration. If entered =0 ,the values in < > are substituted for the succeeding six integers.

< 100 > for the fixed source problem

< 10 > for the eigenvalue problem

ITMOUT Maxmum number of power iterations < 25 >

ITBG Minimum number of inner iterations before extrapolation < 5 >

LCMX Number of inner iterations for testing over-relaxation factor < 5 >

ITDM Minimum delay between extrapolation < 5 >

IPT Control for monitor print < 0 >
=0 skip print
=1 monitor print at each inner iteration

BLOCK 3 Floating point parameters /6/

EPSI Convergence criterion for inner iteration.
If entered =0.0 , the values in < > are substituted for succeeding six floating numbers < 0.001 >

EPSO Convergence criterion for outer iteration < .0001 >

EPSG Extrapolation criterion < 0.05 >

RELC Initial over-relaxation factor < 1.2 >

OVERX Maximum extrapolation factor < 100. >

FACTOR Under extrapolation factor < 1.0 >

BLOCK 4 /NRMAX/

NK Number of mesh intervals by region

BLOCK 5 /NRMAX/

IK Material number by region ; the material number is assigned by the order of the material appears in the mixture specification.

BLOCK 6 /NRMAX/

IXR X-region number by region

BLOCK 7 /NRMAX/

RK Outer radii by region

BLOCK 8 /1/

BSQ Transverse buckling in cm-2

BLOCK 9 /1/ required if IBOUND=2

XLAMD Extrapolation distance in cm

II.7 CITATION : Multi-dimensional Diffusion

The input for the CITATION routine keeps the original input format so that the same input data except the leading BLOCK and final two BLOCKs can be used in separate execution of the original CITATION code (Ref(10)).

BLOCK 1 Control integers required by SRAC /3/

NM Number of materials used in this routine (i.e., count of materials read in Section 008)

NXR Number of X-regions used for cross section edit, punch 0 if it is not required.

ID Option to select diffusion coefficient
= 1 select D1 in the SRAC macroscopic format
= 2 select D2

Following data starting at Title card, Section 001, 003,,,,,through Section 099, and a blank card have the same organization as the original CITATION.

Title Card (18A4/18A4)
Each individual case has two title cards at the beginning.

Section 000: (omitted)
****

Section 001: General Control
****

CARD 1: 001

CARD 2: Control Options (24I3)

NGC1 Depletion option: suppressed. Enter 0

NGC2 Restart option: to exercise a restart, a complete set of data must be supplied for the Section 001. Options available are exercised with the following input values; when .NE.0, a restart disk properly written from earlier machine run must be prepared (see NGC3 below and check control card requirement):
=0 no restart.
=-1 continue a previous case.
>0 restart a depletion calculation (suppressed)

NGC3 Option to write data on logical device 13 to permit restart. Exercised if >0

NGC4 -

NGC5 Save macroscopic cross sections (suppressed)

NGC6 Option to write neutron flux map on I/O logical device 9 if > 0.

NGC7 Option to write power density map on I/O logical device 32 if > 0.

NGC8 Option to write point neutron source (space-energy) on logical device 17 (see GLIM 5 on CArd 6 of 001)

NGC9 -

NGC10 Type of eigenvalue problem (internally set to either of following types depending on the step where the routine called)
=-5 Fixed source
=0 Effective multiplication factor calculation

NGC11 Search option (suppressed)

NGC12 Adjoint indicator. Enter =1 if an adjoint calculation is required.

NGC13 Option to input the adjoint flux (suppressed)

NGC14 -

NGC15 Termination option (applied only to the flux iteration calculation
=0 Terminate calculation and proceed as if converged if machine time or iteration count is exceeded (see card 3 and card 4 below).
=1 If limits are exceeded, terminate calculation and proceed as if converged only if the iterative process is converging
=2 If limits are exceeded, terminate calculation

NGC16 -

NGC17 -

NGC18 Residue calculation option
If = 0, values of the multiplication factor and the relative macroscopic absorption cross section, each of which minimizes the sum of squares of the residues of the point neutron balance equations, are obtained after termination of each eigenvalue problem.
If < 0, this calculation is not done.

NGC19 Macroscopic cross section option
If > 0, only macroscopic cross sections input in section 008 will be used. (punch 1 always, because there is no support for microscopic input in the SRAC)

NGC20 -

NGC21 Coolant flow direction (suppressed)

NGC22 Calculate and print delayed neutron data (suppressed)

CARD 3: Edit Options (24I3)

In the description below, the option will exercised if the input number is > 0.

IEDG1	Print iteration data each mesh sweep.
IEDG2	Print final nuclide densities (suppressed)
IEDG3	Print macroscopic group-to-group transfer cross sections
IEDG4	Print macroscopic reaction rate cross sections
IEDG5	Print gross neutron balance over system by group
IEDG6	Print gross neutron balance by zone by group
IEDG7	Print gross reaction rates in individual nuclides (suppressed)
IEDG8	Print reaction rates in individual nuclides (suppressed)
IEDG9	Print zone average flux values by group (IEDG6=0)
IEDG10	Print point flux values by group
IEDG11	-
IEDG12	Print zone average power densities
IEDG13	Print relative power density traverses through peak
IEDG14	Print point power densities
IEDG15	Print point cumulative heat deposited in coolant (suppressed)
IEDG16	Print point neutron densities summed over energy
IEDG17	Nuclide number; print point neutron absorption rates in this nuclide (suppressed)
IEDG18	-
IEDG19	Print space point maps of damage rates (suppressed)

CARD 4: General iteration count and machine time limits (24I3)

The first numbers on this card are the iteration count limits for the various loop calculations. Problems are terminated when the iteration count reaches the limit and the calculation proceeds as per NGC15 (see CARD2). For a statics problem (no depletion or dynamics) only ITMX1, ITMX19, and ITMX21 are applied.

ITMX1	Any initial eigenvalue problem (statics problem or initialization) <200>
ITMX2	All other eigenvalue problems <100>

ITMX3 Any time step depletion loop calculation (suppressed)

ITMX4 Any repeat time step loop (suppressed)

ITMX5 Any repeat cycle loop (suppressed)

ITMX6 –
ITMX7 –
ITMX8 –
ITMX9 –
ITMx10 –
ITMx11 –
ITMx12 –
ITMX13 –
ITMX14 –
ITMX15 –
ITMX16 –
ITMX17 –
ITMX18 –

These numbers are the machine time limits (min) for the various loop calculations and also total computer time limit; generally calculations continue if time is exceeded as if convergence criteria had been satisfied.

ITMX19 The initial eigenvalue problem <60>

ITMX20 All other eigenvalue problem <30>

ITMX21 Any reactivity loop (suppressed)

ITMX22 Any repeat time step loop (suppressed)

ITMX23 Any repeat cycle loop (suppressed)

ITMX24 Total machine time <120>

Built-in numbers are shown in < >; these are replaced by zero input data.

CARD 5: General restraints (6E12.5)

Any calculation will be terminated if the following restraints are not met, excluding GLIM5.

GLIM1 Maximum multiplication factor <1.5>

GLIM2 Minimum multiplication factor <0.5>

GLIM3 Maximum search nuclide density (suppressed)

GLIM4 –

GLIM5 Factor applied to neutron productions for generating a fixed source file, see NGC8 <1.0>

GLIM6 Multiplication factor to be satisfied if a search is to made (suppressed)

Section 002: Depletion history description (suppressed)
****

Section 003: Description of the neutron flux problem
****

CARD 1: 003

CARD 2: General description (24I3)

NUAC1 Type of flux approximation
=0 Finite-difference diffusion theory
=1 -
=2 -
=3 -
=4 -

NUAC2 Initialization of the flux for the first problem, applicable only for a restart calculation, NGC2.NE.0
=0 use available flux, multiplication factor and acceleration parameters from the previous problem
=1 use only flux from a previous calculation
=2 use built-in initialization procedure. For continuation of a statics problem (NGC2 < 0), this number should be 0

NUAC3 -

NUAC4 -

NUAC5 Geometry option
=1 One-dimensional slab (X)
=2 One-dimensional cylinder (R)
=3 One-dimensional sphere (S)
=4 -
=5 -
=6 Two-dimensional slab (X,Y)
=7 Two-dimensional cylinder (R,Z)
=8 Two-dimensional circle (Theta,R)
=9 Two-dimensional hexagonal (H)
=10 Two-dimensional triangular (T)
=11 Three-dimensional slab (X,Y,Z)
=12 Three-dimensional cylinder (Theta,R,Z)
=13 Three-dimensional hexagonal (H,Z)
=14 Three-dimensional triangular (T,Z)

NUAC6 -

NUAC7 -

NUAC8 Indicator of two-dimensional diagonal symmetry (on plates if 3-D); if > 0, there is symmetry about the diagonal starting at the upper lefthand corner and there are the same number of rows and columns; if < 0, there is inverted diagonal symmetry. Set to 0 if NUAC11= -1

NUAC9 Indicator of two-dimensional symmetry along column slices

for 3-D problems only, see option above

NUAC10 –

NUAC11 Left boundary condition (always required)
=-1 periodic (implemented for geometries given by NUAC5 = 1, 6, 8, 11 and 12, left to right boundary closure only)
= 0 extrapolated
= 1 reflected

NUAC12 Top boundary condition
= 0 extrapolated
= 1 reflected

NUAC13 Right boundary condition (always required), set to -1 if NUAC11 is -1
= 0 extrapolated
= 1 reflective
= 2 90 degree rotational symmetry (right to bottom boundary, slabs only)
= 3 inverted reflection (180 degree rotational symmetry, slab only)

NUAC14 Bottom boundary condition (required for 2-D)
= 0 extrapolated
= 1 reflected
= 2 60 degree rotational symmetry, triangulars

NUAC15 Front boundary condition (required for 3-D)
= 0 extrapolated
= 1 reflected

NUAC16 Back boundary condition (required for 3-D)
= 0 extrapolated
= 1 reflected

NUAC17 Number of zone to be an internal black absorber and to have the non-return boundary condition applied at its edges (see XMIS2 CARD 4; this zone will be black to all groups unless additional data are supplied)

NUAC18 Option to allow negative neutron flux if > 0

NUAC19 Override use of Chebychev polynomials in adjusting the parameters if > 0

NUAC20 Line relax only on rows if > 0; if -1, force alternating direction line relaxation on rows and columns, and also fore and aft for 3-D; if -2, use only on rows and columns. When left unspecified, the code selects line relaxation on rows only with one inner iteration for all problems involving upscattering, otherwise three inner iterations for 3-D problems without I/O and five with data I/O during iteration, and alternating direction line relaxation for all 2-D problems.

NUAC21 –

NUAC22 –

NUAC23 Specified number of inner iterations normally not specified (see NUAC20 above)

NUAC24 –

CARD 3: Iteration convergence criteria (6E12.5)

EPSI1 Maximum relative flux change for the last iteration of each initialization eigenvalue problem <0.0001>

EPSI2 Maximum relative change in the eigenvalue for the last iteration of eigenvalue problems. This applies to the multiplication factor calculation, and the direct buckling or 1/v search parameter. <0.00001>

EPSI3 –

EPSI4 Replaces EPSI1 for all eigenvalue problems except those for initialization or station calculations. <0.0001>

EPSI5 –

EPSI6 –

CARD 4: Miscellaneous data (6E12.5)

XMIS1 External extrapolated boundary constant (–D/Phi*dPhi/dx)
> 0 specifies the constant for all extrapolated boundaries (see NUAC11–16) for all groups
< 0 this is the total number of energy groups (negative) and other cards are to follow this card which give the extrapolated boundary constants for problem boundaries beginning with those for all energies for the left boundary (6E12.5); thus data is required for only the left and right boundaries for 1–D problems, four boundaries for 2–D, and for six boundaries for 3–D problems. For the periodic boundary condition, NUAC11 =–1, skip the left and right boundaries.
= 0 the code will use the built–in value for all extrapolated boundaries <0.4692>

XMIS2 Internal black absorber boundary constant ($-D/\varphi*d\varphi/dx$)
> 0 the constant for all groups applying to zone NUAC17
< 0 this is the total number of groups (negative) and another card(s) is to follow, after any required above , which gives the internal black absorber boundary constants for each energy group (6E12.5). Any zero values indicate that the rod condition is not to apply to that group.
= 0 the code will use the built–in value for all groups and the absorber will be black over all energy <0.4692>

XMIS3 Core power level, Mwth; if macroscopic cross sections have

been specified but not watts/fission data (see Section 008), this is to be the total neutron production rate (n/sec) . Normally this number must be specified and attention should be given to achieve the desired useful power
level using the supplied fission cross sections which may be artifitial so as to include the (n,2n) reaction.

XMIS4 Conversion factor, ratio of thermal energy to fission energy (XMIS3 is divided by this, normally < 1.0 <1.0>)

XMIS5 Fraction of the core considered; the mass balances are divided by this number and XMIS3 is multiplied by this number <1.0>

XMIS6 Initial overrelaxation factor, normally calculated by the code and not specified here <0.0>

Section 004: Geometric mesh description
****

For any succeeding case, if the number of regions is changed this section must be followed by data for section 005; further, any required overlay must be included, section 006.

CARD 1: 004

CARD 2: (6(i3,E9.0))

Specify the number of mesh points and the region width for each vertical region going from left to right. For a two-domensional problem also specify the number of mesh points and the region width for each horizontal region going from the top to bottom starting with a new card . For a three dimensional problem also specify the number of mesh points and the region width for each region going from front to back. In referring to the geometric mesh, rows of mesh points go from top to bottom, column of mesh points go from left to right, and planes of mesh points go from front to back. In R and R-Z gepmetry, a row is a radial traverse. in Theta-R and Theta-R-Z geometries, columns radiate downward from a center at the top and a row has constant radius. In hexagonal and triagonal geometries the X and Y are assumed to be at 60 degree (upper left-hand corner), and dimensions are on external boundaries; these treatments are precise only if points on any plane have equal finite-difference volumes. Data must be ended for each traverse by a blank entry; if the last card of data is filled for any traverse, another card is required (blank).

Section 005: Zone placement
****

CARD 1: 005

CARD 2: (24I3)

Specify the zone identification numbers (i.e., location of uniform composition) of each vertical region for the first horizontal row of regions going from left to right. Beginning with a new card, specify the zone numbers of each vertical region in the second horizontal row

of regions. Continue these specifications going from top to bottom. For a one-dimensional problem the zone numbers are specified for only horizontal regions. For a three-dimensional problem, give the two-dimensional grid for each block of mesh-point planes going from front to back. The cross section set later to be associated with the zone numbered 1 will be used as a reference, so zone 1 might be located within a core rather than in a blanket or reflector. Also it will prove convenient to number consecutively zones which will contain the same material.

Section 006: Mesh overlay

New zones may be superimposed within a mesh described with this data.

CARD 1: 006

CARD 2: (I4)

Specify a zone number. Data is read until zero zone number encountered.

CARD 3+: (3(6I4))

Specify blocks of points by left column number and then right column number to give limits along rows, top and bottom row numbers for column limits, and front and back plane numbers for depth limits in that order. If only one row is involved, for example, then that row is repeated. Only 4 entries are needed for each specification in 2-D geometry. Data is read until zero value is encountered on the first column number. The entry '000600060005001500020002' places the new material along column 6 from row 5 through 15 and on plane 2 only. Remember that each mesh point has an associated volume around it - mesh points do not lie on material interfaces; specifying a single mesh point here, say '000600060005000500020002' does involve an associated volume.

Section 008: Macroscopic cross sections

Macroscopic cross sections are supplied from an I/O device (logical 31) where the form of the data is identical with that of this input section in EBCDIC); CARD 1 is required and then CARD 2 containing a negative integer in columns 1-3, which complete the data for this section. We shall complete to describe this section to explain the contents of the I/O device 31.

CARD 1: 008

CARD 2: The number of groups and scattering range (3I3)

KMAX The number of energy groups

IX28 The number of groups for downscatter

IX29 The number of groups for upscatter

CARD 3: Cross sections (2I6,5E12.5)

M	A zone number
K	A group number
SIG1	Diffusion coefficient
SIG3	Absorption cross section
SIG4	Production cross section (Nu*Σf)
SIG5	1/v cross section for primary mode calculation
SIG6	Power per unit flux, watts/fission times Σf for normalization of the flux level and power density maps; if all entries here are zero, then SIG4 is used. Values of Σf is filled by SRAC to complete data for normalization to total fissions.

CARD 4: Scattering cross sections (6E12.5)

Specify the cross section for scatter from group K (above) to each of the other groups 1 to KMAX.

The code continues reading card 3 and card 4 until a zero M is found (blank card).

CARD 5: Fission source distribution function (6E12.5)

Specify the fraction of fission neutrons that are born in groups 1 to KMAX

Section 024: Buckling specification

****

CARD 1: 024

CARD 2: Indicator, IND(I3,E9.0)

If IND = 1, specify a constant buckling in columns 4-12 (E9.0) on this card and no additional data is required.

If IND = 2, specify values of group dependent buckling starting with card 3 (6E12.5).

If IND = 3, specify two zone numbers on card 3(8I3) followed by the group depending buckling on card 4 (6E12.5) which will apply to the set of consecutive zones specified on card 3. Continue with card 3 and card 4 for as many zones as required. A blank card (zero zone number) must be used to end this data.

Section 099: Termination of input

****

Termination of each case is done with a card containing 999 in columns 1-3. In case of separate execution of the original CITATION, cases may be stacked one behind the other with succeeding ones using data provided in previous ones. A normal exit may be affected by

locating a blank card after the 999 card of the last case. In SRAC only a case is allowed.

CARD 1: 999

CARD 2: 000 (blank)

For the use of CITATION in the SRAC, succeeding two BLOCKs are required to complete the input of Sect.II.7

BLOCK for Material number specification /NM/

This item is to select and to number the materials used in CITATION routine from the materials listed in the mixture specification in Sect. II.8.

MAT(M) M=1,NM
Position of material M (zone number in Section 008) in the mixture specification in Sect.II.8

BLOCK for X-region specification /IZN/ required if NXR>0

This item is to edit the averaged few group cross sections using the neutron fluxes of the case where the multi-group problem is solved.

NXREG(I),I=1,IZN
X-region number by region ; IZN is the product of number of horizontal regions, that of vertical regions and that of planes for 3-D case.

****

When the CITATION format macroscopic cross sections are required by IC14 in Block 3 in Sect.II.1, the Block 1 (NM,NXR,ID) and the Block for Material number specification (MAT(M),1,NM) are read in this step.

II.8 Material Specification

BLOCK 1

NMAT Number of mixtures specified. It is not necessary to specify the homogenized mixtures by X-region if they are not used in the later step in the case.

BLOCK 2

MTNAME (8H) Mixture identification expressed by eight characters, composed of five tags as mmmmebxp which appears as MEMBER name in macroscopic cross section files and flux file.

mmmm-tag Effective mixture identification; any alphabetic character for the first and any alphameric character for the remaining three columns is accepted. For a set of X-region averaged cross sections CASENAME (4H) is assigned to this tag.

e-tag the tag internally used to specify the neutron energy range of a set of macroscopic cross sections, as the set for fast and thermal energy range are separately composed and lately concatenated into the set for whole energy range. The code 'F' for fast, 'T' for thermal, 'A' for whole energy range is assigned, respectively.

b-tag the tag to indicate burn-up step as '0','1','2',...,'9' or 'A','B',... corresponding to fresh, step 1, step 2,.., step 9, and step 10, step 11,... respectively, when cell burn-up calculation is called by IC20 in BLOCK 1 of Sect. II.1. In non-depletion case this tag is filled by the character punched in the input.

x-tag the tag to specify X-region number of a set of homogenized cross sections. For a set of mixture cross sections composed in MACROF or in MACROT routine , this tag is filled by the character punched.

p-tag the tag internally used to specify Legendre component and also whether fine or coarse in energy structure. Any character punched in the input is replaced by the tag.
='0' coarse and isotropic
='2' fine and isotropic (after transport correction)
='3' fine and P1 component
='4' fine and P0 component

NCOM Number of comment words in the future use of Data Pool system (fill 0 for the present PDS file)

NISO Number of nuclides to compose the mixture. If punched =0 the program assumes that the mixture exists already in MACROWRK file or will be made in the MIXX step, then the formation of macroscopic cross section is skipped in the MACROF and the MACROT steps. Contrarily if the composition is specified to the already existing mixture the formation of macroscopic cross sections is skipped. When the IRA or the PEACO routine is used, punch the

composition for all costituent mixtures in the cell to calculate resonance absorption even if they are already made up.

TEMP Physical temperature of the mixture (degree K). Although any temperature can be accepted, the program replaces it by the nearest value in the tabulated temperature (See Dictionary VII.7). Actually this temperature is effective in calculating resonance absorption.

XL l ; the mean chord length of the resonance absorber lump used for the heterogeneous effect of the admixture cross section used in the interpolation of resonance shielding factor and in the IRA method , and also used in PEACO for the constant to yield a non-dimensional blackness for the interpolations of the collision probabilities. Enter non-zero value to non resonant mixture.

Generally l is defined as

$$l = 4\frac{V}{S}$$

where V is the volume of the mixture, S is the surface area.
For a one dimensional slab of thickness a
$l = 2a$
For a one dimensional sphere of radius a
$l = 4a/3$
For a one dimensional cylinder of radius a
$l = 2a$
For a hollow cylinder of inner radius a and outerradius b

$$l = 2b(1 - (\frac{a}{b})^2)$$

For more complicated geometry which contains several absorber lumps in a unit cell, punch the mean chord length of a dominant absorber, for example , to BWR lattice that of a pin rod.

DC C ; the Dancoff correction factor used for the heterogeneous effect of the admixture cross section in the interpolation of resonance shielding factor (effective only if IC3=3 in BLOCK 3 of Sect.II.1 is specified; punch 1.0 for the homogeneous approximation)

BLOCK 3 /NCOM/ words

MCOM Comment for the mixture stored in the Data Pool system

BLOCK 4 required NISO times for the mixture if non zero NISO is specified.

IDENT (8H) Nuclide identification expressed by eight characters composed by five tags 'Xzzmciit' as already explained in Sect.II.2. Note that the discrepancy of the mixture

temperature and the temperature indicated by t-tag is accepted with some warning message; the former is used for the resonance shielding calculations , the latter is used to choose the thermal neutron data.

IRES Resonance process indicator (effective if non zero IC5 of BLOCK 3 in Sect.II.1 is specified)
=0 normal non absorbing nuclide
=-1 pseudo non resonant nuclide whose cross sections in resonance energy range have been stored in MICREF file as the output of the IRA approximation in the previous case (effective if IC5=1)
=1 non resonant nuclide whose cross section in resonance energy range are replaced by those of the above pseudo nuclide of the same IDENT (effective if IC5 =1)
Both input records of IRES=-1 and 1 must be entered to the same resonant nuclide in the macroscopic step in the treatment of double heterogeneity.

=2 resonant nuclide (if the BURNUP routine is called this item is set automatically for resonant nuclide)
=3 non resonant but absorbing nuclide whose fine energy dependent absorption cross sections are assumed constant in each user's fine group which are taken from the user fast library (effective if IC5=2)
=4 non resonant 1/v absorbing nuclide whose fine energy dependent absorbing cross sections are treated exactly as 1/v in resonance energy range (effective if IC5=2)

IXMICR Indicator to write the effective cross sections into the effective microscopic cross section file and the the resonance cross section file
=0 no edit
plus 1 write the effective cross section into the effective microscopic cross section file in the same format as in the user fast and thermal libraries
plus 2 write the microscopic cross sections averaged in M-region into the user fast library (if IC5=1) or into the resonance cross section file (if IC5=2), and also rewite the macroscopic cross sections of the mixture using the effective microscopic cross sections in macroscopic file. The IDENT (nuclide identification) of resultant microscopic cross sections are formed as same as the IDENT of this nuclide except ii-tag (two characters) which are taken from the sixth and seventh characters of MTNAME of the mixture. This option is for the output of primary step in the treatment of double-heterogeneity.

DN Nuclide density ($10^{-24} cm^{-3}$)

Repeat BLOCKs from 2 through 4 NMAT times

II.9 Burn-up calculation

The input for cell burn-up calculation is minimized because the information such as chain scheme, yield, decay constant, power per fission, etc. is compiled in the burn-up library. The user has still to choose the chain schemes which are stored in separate members in the library. (See Dictionary VII.5)

Following input data are required if IC20 > 0 in Sect.II.1.

BLOCK 1 Control integers /3/

NEP Number of burn-up steps to modify the effective microscopic cross sections for burnable nuclides (<=15). It means (NEP+1) times cell calculations are executed in a case.

IBUNIT Burn-up unit to control exposure time under the fixed power
=0 MWD (Mega Watt Days)
=1 fraction of U-235 burnt

IBEDIT Edit control
=0 brief edit
=1 detailed edit
=2 more detail for code testing

BLOCK 2 Floating numbers /2/

POWERL Whole core power in MW

CVOL The factor to yield the total amount of fuel volume in core by multiplying it to the cell volume. For example in a one dimensional calculation of a cell of a fuel plate , the thickness of fuel meat stands for the fuel volume, then the factor is the product of width * height of fuel plate * number of fuel plate in the core.

The ratio of the above two items is used to get absolute neutron flux, then any common factor to the above's may be multiplied. For example, the term 'core' appearing in this item can be read as 'fuel element' or 'assembly'.

BLOCK 3 /NEP/

PERIOD Burn-up steps in ascending order (not including fresh step) in unit specified by IUNIT

II.10 MCROSS ; Updating of Resonance Cross Section File

The routine MCROSS is called if IC8 > 0 in Sect.II.1. This routine produces the temperature dependent cross sections of the required resonant nuclides of the specified temperatures in the ultra-fine energy structure which will be used in the ultra fine resonance calculations. The resonance level parameters required are taken from the user fast library. The user who are satisfied with the public resonance cross section file where we have compiled the cross sections for all available nuclides over all range of temperature does not need this routine. Otherwise if the user wants to treat the double heterogeneity

in resonance neutron energy, he has to prepare his own resonance cross section file which contains the information for all concerning nuclides over the necessary temperature range to be able to write the effective cross sections into his file. The above preparation may be substituted by the TSS operations to copy the necessary members from the public into the user library. (see Sect.V.3)

BLOCK 1

IDENT (8H) Nuclide identification of the organization 'Xzzmciit' as described in Sect.II.2 User's microscopic cross section libraries, while the chemical tag 'c' and the temperature tag 't' are not effective in this routine.

BLOCK 2 /3/

NT Number of temperatures

IOUT Print control
=0 brief edit
=1 detailed edit

IPLOT Plotter control
=0 skip plot
=1 plot cross sections vs energy

BLOCK 3 /NT/

TT Temperatures for which cross sections are calculated. Each temperature is replaced by the nearest value in the tabulated values (see Dictionary II.6)

Repeat BLOCKs from 1 through 3 until a blank identification is encountered.

II.11 PEACO ; The Ultra-fine Resonance Absorption Calculation

An integer is required if IC5 = 2 in BLOCK 3 of Sect.II.1 is specified for edit control of the PEACO routine.

BLOCK 1 /1/

IPLOT =0 skip plot
=1 plot neutron spectra by up-to five R-regions in a figure
=2 plot neutron spectra by R-region for a R-region per figure
>0 print modified group cross sections

III. Usage of Auxiliary Programs

III.1 Production of the Fast Group Cross Section Library

The group cross section library for the SRAC code is generated by using the TIMS-PGG code system(Ref.(76)) from the evaluated nuclear data files in the ENDF/B-IV type format. The TIMS-PGG code system consists of several codes and a flowchart of the generation system is shown in Fig.III.1-1. At each intermediate step, a code is used to process intermediate data libraries and produces an output library.

The PROF-GROUCH-GIIR code calculates the group averaged cross sections of the Bondarenko's type by weighting the 1/E and/or fission spectra, depending on the energy range defined by Section I.1.3. The TIMS-1 code generates mainly the group constants for heavy resonant nuclei by solving numerically neutron slowing down equations. The calculated results are stored to the Master Fast Library in the PDS format by XTABPDS. The SPEC code calculates the group fission spectra from using the distribution given in the ENDF/B-IV nuclear data file. Finally, the fast group cross section library for the SRAC code system is produced by using FASTMAKE.

The TIMS-1 code (Ref.(16)) calculates the group constants of heavy nuclides in resonance energy region. The flowdiagram is shown in Fig.III.1.2 and the general features of each subprogram are as follows:

FILEREAD : The resonance parameters and floor correction cross sections are read from the data file in the ENDF/B format, and are suitably arranged for the ARCFIT-2, ARCFIT-3 and MCROSS codes.

ARCFIT-2 : In the unresolved resonance region the average cross sections are calculated by using the average resonance parameters. The formulas of the average cross sections are the same as those used in the ENDF/B-IV processing(Ref.(12)).

ARCFIT-3 : In the unresolved region, the resonance parameters and the level spacings are generated by using Monte Carlo method from the Porter-Thomas and Wigner distributions, respectively. The generated ladder of resonance parameters and the resolved resonance parameters are finally connected and a set of resonance parameters is supplied ranging from the unresolved to the resolved regions.

MCROSS : The Doppler broadened cross sections are calculated on the ultra-fine lethargy meshes of about 5000 groups below 130.7 eV. The calculations are performed by use of either the Breit-Wigner single -level formula or the multilevel formula (Ref.(55)). The symmetric and asymmetric Doppler line shape functions are calculated by using the Buckler method (Ref.(39)).

PEACO : The neutron spectra on the ultra-fine lethargy meshes are calculated by solving numerically the neutron slowing down equation with the use of the recurrence formula developed by Kier(Refs.(40,41)) in which the lethargy meshes are assumed to be extremely narrow compared to the maximum lethargy gain per collision with the heaviest nuclei in the considered system.

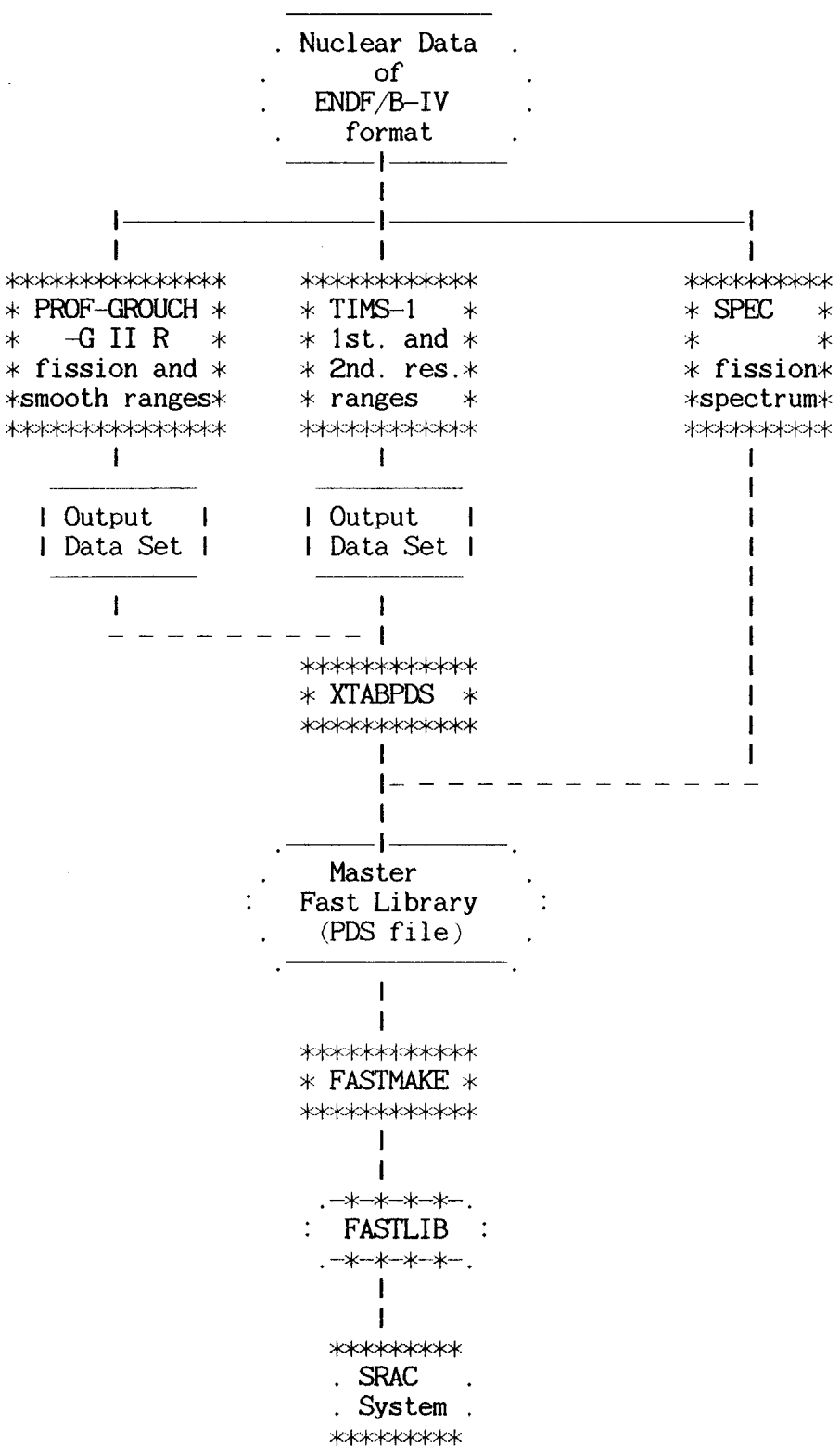


Fig.III.1-1 Flowchart for group constants production code system

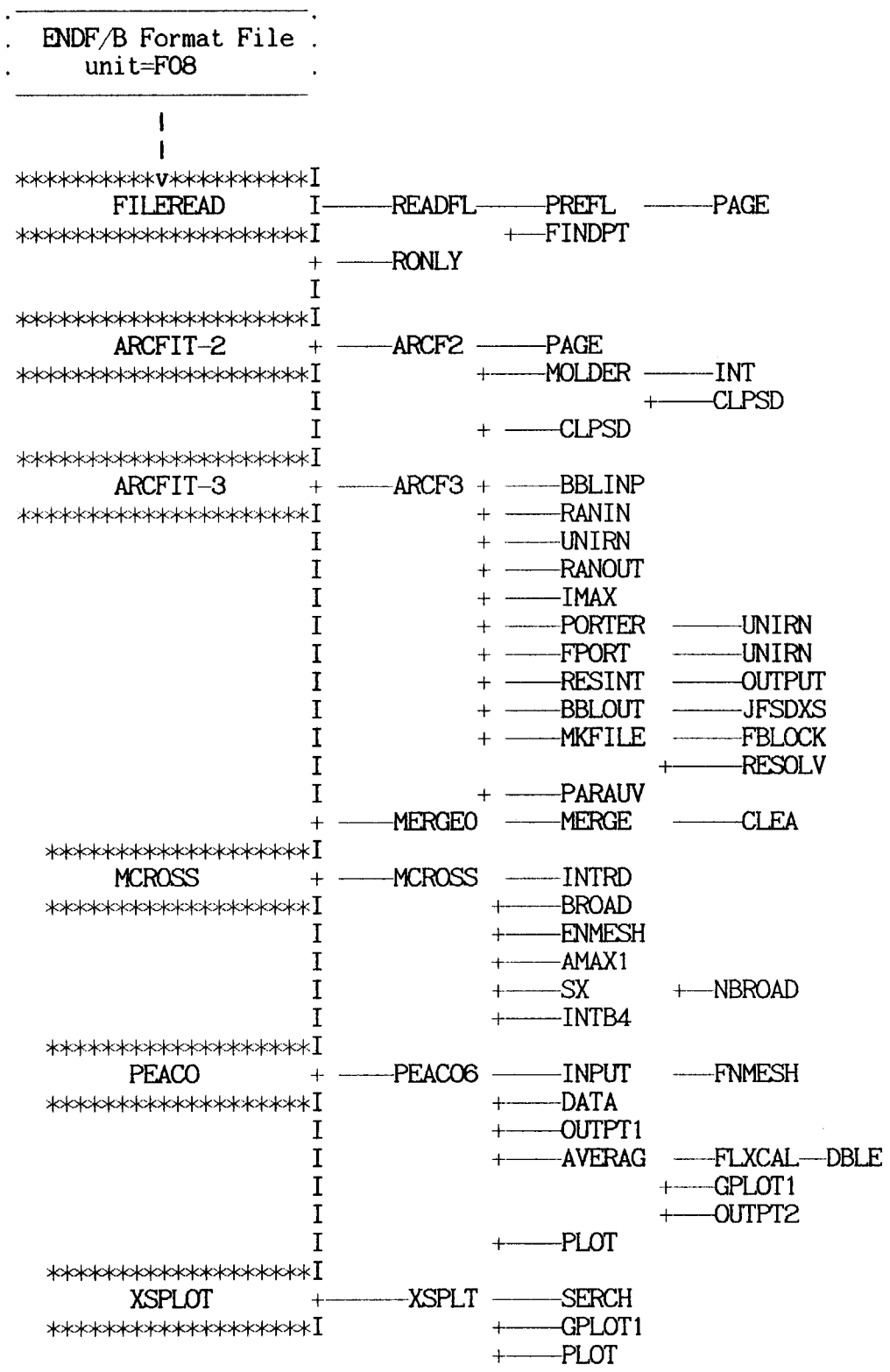


Fig.III.1-2 Flowchart of TIMS-1 for the group constants generation of heavy resonant nuclei

The PROF-GROUCH-G II R code is a modified version of PROF-GROUCH-G II (Ref.(15)), and calculates the group cross sections of light and medium weight nuclei and those for smooth cross section of heavy nuclei. A flow diagram is shown in Fig.III.1-3.

MCFILE : This code produces a file of control parameters for the group cross section library, such as the energy boundaries and admixture background cross sections etc.. Only for the first case, MCFILE is executed.

RESENDD (Ref.(42)): This code is a revision of the RESEND code (Ref.(37)). The pointwise cross sections are generated for a material designated by MAT.No of input data. In the resonance energy region, they are calculated with the given formula using resonance parameters. In the unresolved resonance region, the RESENDD code calculates the temperature independent smooth cross sections. The temperature dependent cross sections for heavy resonant nuclides are calculated by using the TIMS-1 system. The RESENDD code is used for the calculations of pointwise cross sections for light and intermediate nuclides.

SPINPTM : The input data for the SPTGRM code are produced from the control file and the pointwise cross section file. The weighting spectrum for calculation of the averaged group cross sections is determined.

SPTGRM : The group averaged cross sections, and elastic, inelastic and (n, 2n) scattering matrix are calculated. The original version of SPTGRM is the SUPERTOG code(Ref.(43)) which calculates infinitely dilute cross sections.

XTABPDS : The tables of self-shielding factors are generated from using the infinitely dilute and effective cross sections calculated with SPTGRM . The PDS file is accessed by XTABPDS to construct a master library of the group constants.

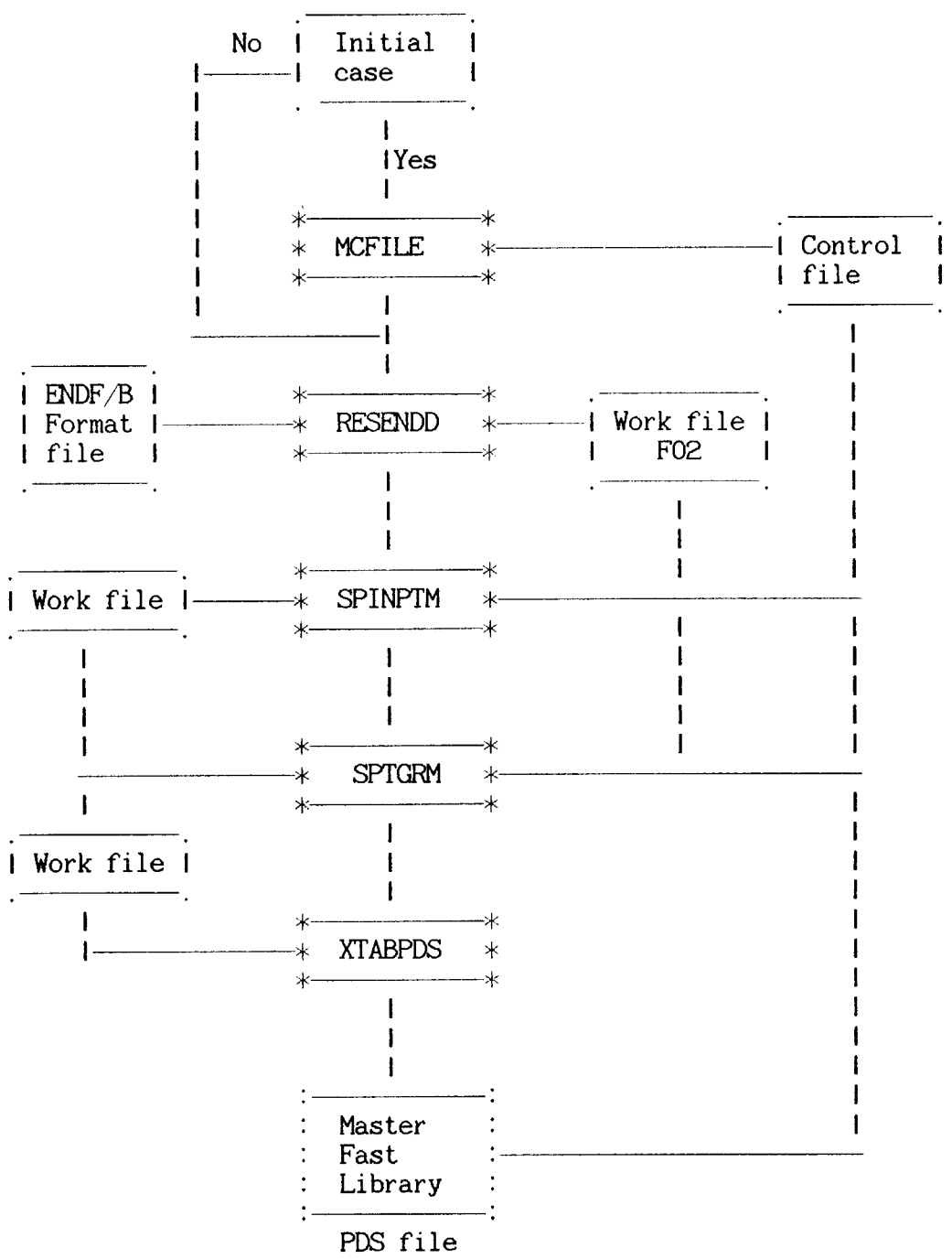


Fig.III.1-3 Flowdiagram of the PROF.GROUCH.GIIR code

III.1.1 Input Data Requirements for TIMS-1

In the TIMS-1 code, the input data are read in the namelist format statement. For example, an input statement

&NAMJ (b) ABC=1.0,X=0.5,KJ=10 (b) &END

means that in the namelist name NAMJ,input variables name ABC,X and KJ are specified by 1.0,0.5 and 10 respectively. &NAMJ is read

from the second column,and one blank is needed between &NAMJ and the first input variable ABC and also between the last variable KJ and &END. A character type data must be enclosed by apostrophes as CASE='TH-232'. A sequential data array can be entered together, each word separated by comma, as XX=1.0,2.0,3.0 for an array variable of XX of length 3.

BLOCK 1

TITLE(72H) Descriptive information

BLOCK 2 / 1 /

MATNO(I5) Material number defind in the ENDF/B or JENDL library

BLOCK 3 / Max(6) /

IOP(6(A8,2X)) Calculating flow is designated by the six words ,FILEREAD, ARCFIT-2, ARCFIT-3, MCROSS , PEACO and XSPLOT. These designation data mean the following performance :

Designation	BLOCK NO.	Performance
FILEREAD	1,2,3,4	Resonance parameters and smooth data are read from the nuclear data file ,and the data files for ARCFIT and MCROSS are produced.
ARCFIT-2	1,2,3, 5	Averaged resonance cross sections are calculated in the unresolved regions.
ARCFIT-3	1,2,3, 5	Production of a ladder of resonance parameters for s-,p- and d-wave neutrons
MCROSS	1,2,3, 6,7	Calculations of temperature dependent cross sections at ultrafine lethargy mesh widths.
PEACO	1,2,3, 8,9,10	Calculations of neutron spectra and effective group cross sections.
XSPLOT	1,2,3, 11,12,13	Plotting for temperature dependent cross sections

BLOCK 4 (NAM1-FILEREAD)

NAM1 Namelist name.

MPRINT Print control of nuclear data file (Default value = 1).
= 1, print the nuclear data of ENDF/B or JENDL
= 0, no effect

IRONLY Control for resolved and unresolved region
(Default value=0)
= 0, unresolved and resolved regions are processed.

= 1, only resolved region is processed.

BLOCK 5 (NAM2-ARCFIT-2)

NAM2 Namelist name.

ICFT Fissile or fertile material option (Default value = 1).
= 1, fertile material
= 2, fissile material

BLOCK 6 (NAM3-MCROSS)

NAM 3 Namelist name.

NT No. of temperatures (Max.= 5)(Default value = 1).

NOMS If NOMS = 0, the group boundary is searched automatically. (Default value = 0) The group number for the upper energy group where calculation is started are shown as follows:

NOMS	Energy boundary(eV)
1	100000.0
2	46500.0
3	21500.0
4	10000.0
5	4650.0
6	1000.0
7	465.0
8	100.0
9	46.5
10	10.0
11	0.2

CASE Name of nuclide to be processed.
By assigning 'Name'of nuclide,the s-,p- and d-wave neutron resonance levels to be summed up in the calculations are defined in the program. The installed nuclide names are TH-232,U-233,U-234,U-235,U-236,U-238, PU-239,PU-240 and PU-241,and for the others,the 'Name' of nuclide to which resonance structure is analogous,may be selected.

TT Temperature(°K) (Default values = 300., 800., 2100., and 4500.).

BLOCK 7 (NAM4-MCROSS)

NAM 4 Namelist name.

IOUT Print control of ultrafine group cross sections (Default value = 0).
= 1, print ultrafine group cross sections.
= 0, no effect.

BLOCK 8 (NAM5-PEACO)

NAM 5 Namelist name.

NTEMP No. of temperatures (Max.= 5) (Default value = 4).

KRAT No. of the atomic density ratio, R - values (Max.= 5) (Default value= 1).

KSIGM No. of admixture cross sections(Max.= 10) (Defalut value = 8).

KRES No. of resonant materials (Max.= 2) (Default value = 1).

NPLOT Plot control of neutron spectra (Default value = 0).
= 1, neutron spectra are plotted.
= 0, no plotting.

NPRINT Print control (Default value = 0). = 0, only the results for resonant material of interest are printed. = 1, also the results for background resonant material are printed.

KMOD Starting energy group number for the group structure of the JFS-2,SRAC or JFS-3 type. The SRAC group structure is shown in Dictionary VII.4. =-1,-2,and -3 search automatically the starting number for JFS-2,SRAC or JFS-3. If you designate specially upper group number,as for SRAC type, input KMOD = 200 plus group number shown in Dictionary VII.4, and if KMOD=0 ,the number of energy groups(KBG) and energy boundaries(EN) are read in BLOCK 9.

BLOCK 9 (NAM6-PEACO)

NAM 6 Namelist name.

KBG No. of energy groups (Default value = 0). When users want to calculate group constants for a different structure from JFS or SRAC type, (KBG+1) energy boundaries should be read for EN (Max.= 60).

EN Group energy boundaries (eV) (Default values are shown in Dictionary VII.4).

SIGM Admixture background cross sections of Bondarenko type (Default values = 0, 1, 10, 100,1000,10000,100000,1000000)

RATIO R - values of the number density ratios between the two resonant nuclei. These values are defined at each admixture cross sections.(Max.=5). (Default values=0.1,1.0,10.0,1000.0,1000000.0)

NUCLID Nuclide identified with 4 letters (Max.= 3). For example, NUCLID='U238','P239','A30'. If two resonant materials are considered, NUCLID='U238', 'U235', 'A30'.

TEMP Temperature (^{o}K) (Default values = 300., 800., 2100., and 4500.).

AMASS Average mass for background moderator nuclide (Default value= 30).

BLOCK 10 / 3 / Neutron spectra plotting

XTITL(40H) Title of x coordinate,

YTITL(40H) Title of y coordinate.

TITLE(40H) Figure caption.

BLOCK 10's cards are repeated by (KRAT * KSIGM) times.
Do not enter unless NPLOT=1.

BLOCK 11 (XSPLOT) / 1 /

NCFN(I5) Reaction type option for the cross sections to be plotted.
=0, capture, fission, scattering and total.
=1, capture.
=2, fission.
=3, scattering.
=4, total.

NTT(I5) No. of temperatures(Max.=3)

NEN(I5) No. of energy groups.

BLOCK 12 (XSPLOT) / 1 /

EL (E12.4) Lower energy boundary to be plotted.

EH (E12.4) Upper energy boundary to be plotted.

WX (E12.4) Length of x coordinate (mm)(Default value = 230).

WY (E12.4) Length of y coordinate (mm)(Default value = 230).

RX (E12.4) Linear or log scale control for x coordinate (Default values = 1.0). If RX is less than EH/EL, the scale of x coordinate is linear, otherwise log scale.

RY (E12.4) Linear or log scale control for y coordinate (Default value = 1.0).

BLOCK 13 (XSPLOT) / 3 /

XTITLE(40H) Title of x coordinate.

YTITLE(40H) Title of y coordinate.

TITLE (40H) Figure caption.

Unless NCFN = 0, BLOCKs 12 and 13 are repeated NEN times, and if NCFN = 0, BLOCKs 12 and 13 are repeated 4 * NEN times.

III.1.1.1 Input and Output Files Used in TIMS-1

Logical unit number	Program	Comments
8	FILEREAD	Evaluated nuclear data file of JENDL or ENDF/B
1	FILEREAD	Work file for resolved parameters
2	FILEREAD	Work file for unresolved parameters
3	FILEREAD	Work file for smooth and floor cross sections
10	FILEREAD	Work file
20,21,22,25, 30,31,32 71.72.73 91,92,93	ARCFIT-3	Work files for generating ladders of s-, p- and d-wave neutrons resonance parameters
40,41,42,23	ARCFIT-3	Files of ladders generated finally of s-, p-, d-wave and (u,v) resonance parameters
50,51,52,53, 54	MCROSS	Ultra-fine group cross sections for five temperatures of nuclide 1
55,56,57,58, 59	MCROSS	The same above for nuclide 2
60	PEACO	Group constants to be used in XTABPDS
80	PEACO	Ultra-fine group fluxes used in PLOT
66,67	XSPLOT	Work files for plotting cross sections

III.1.1.2 Sample Data and JCL-statements in TIMS-1

```
//JCLG JOB
// EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER 64902031,HI.TAKANO,0431.000
  T.8 C.2 W.6 P.0 I.8 NGT
OPTP  MSGLEVEL=(1,1),NOTIFY=J2031,MSGCLASS=R
// EXEC LMGO,LM='J2031.TIMS',PNM=TIMSHM
//FT01F001 DD UNIT=WK10,SPACE=(TRK,(50,10))
//FT02F001 DD UNIT=WK10,SPACE=(TRK,(50,10))
//FT03F001 DD UNIT=WK10,SPACE=(TRK,(50,10))
//FT08F001 DD DSN=J1615.JENDL1. DATA,DISP=SHR,LABEL=(,,,IN)
//FT09F001 DD UNIT=WK10,SPACE=(TRK,(50,10))
//FT10F001 DD UNIT=WK10,SPACE=(TRK,(50,10))
//FT20F001 DD UNIT=WK10,SPACE=(TRK,(250,10))
```

```
//FT21F001 DD UNIT=WK10,SPACE=(TRK,(250,10))
//FT25F001 DD UNIT=WK10,SPACE=(TRK,(250,10))
//FT30F001 DD UNIT=WK10,SPACE=(TRK,(250,10))
//FT31F001 DD UNIT=WK10,SPACE=(TRK,(250,10))
//FT40F001 DD UNIT=WK10,SPACE=(TRK,(250,10))
//FT41F001 DD UNIT=WK10,SPACE=(TRK,(250,10))
//FT42F001 DD UNIT=WK10,SPACE=(TRK,(250,10))
//FT50F001 DD DSN=J2031.J1P40T30. DATA,DISP=(NEW,CATLG),UNIT=F478V,
//         SPACE=(CYL,(3,1)),MSVGP=MSS02A
//FT51F001 DD DSN=J2031.J1P40T80. DATA,DISP=(NEW,CATLG),
//         UNIT=F478V,SPACE=(CYL,(3,1)),MSVGP= MSS02A
//FT52F001 DD DSN=J2031.J1P40T21. DATA,DISP=(NEW,CATLG),
//         UNIT=F478V,SPACE=(CYL,(3,1)),MSVGP= MSS02A
//FT53F001 DD DSN=J2031.J1P40T45. DATA,DISP=(NEW,CATLG),
//         UNIT=F478V,SPACE=(CYL,(3,1)),MSVGP= MSS02A
//FT60F001 DD DSN=J2031.J1 RP240.DATA,DISP=(NEW,CATLG),UNIT= TDS,
//       SPACE=(TRK,(2,1),RLSE)
// EXPAND DISK,DDN=FT66F001
// EXPAND DISK,DDN=FT67F001
// EXPAND DISK,DDN=FT80F001
//FT70F001 DD UNIT=WK10,SPACE=(TRK,(250,50))
//FT71F001 DD UNIT=WK10,SPACE=(TRK,(250,50))
//FT72F001 DD UNIT=WK10,SPACE=(TRK,(250,50))
//FT90F001 DD UNIT=WK10,SPACE=(TRK,(250,50))
//FT91F001 DD UNIT=WK10,SPACE=(TRK,(250,50))
//FT92F001 DD UNIT=WK10,SPACE=(TRK,(250,50))
//SYSIN DD *
JFS-3-J1 **TIMS-1** PU-240  (JENDLE-1)
 1942
FILEREAD  ARCFIT-2  ARCFIT-3  MCROSS    PEACO
 &NAM1 MPRINT=0  &END
 &NAM2 ICFT=1 &END
 &NAM3 NOMS=0,NT=4,CASE='PU-240' &END
 &NAM4 &END
 &NAM5 KMOD=-3 &END
 &NAM6 NUCLID='PU238','A30'  &END
/*
//*  ** XTABPDS   *******
//XTABPDS  EXEC LMGO,LM='J2031.JFSLOAD', PNM=XTABPDS
//FT01F001 DD UNIT=WK10,SPACE=(TRK,(50,10))
//FT02F001 DD UNIT=WK10,SPACE=(TRK,(50,10))
//FT21F001 DD DSN=J2031.J1RP240.DATA, DISP=SHR
//USERPDS  DD DSN=J2031.JFS3PDS.DATA, DISP=MOD
//SYSIN    DD *
0 0 1 0 0 0.0  /KP IPROP ISKIP ITOT ICUT EPS ( 1)
0 1 0 /IGRCH ITIMS IFTCUT                    ( 2)
21    /INN                                   ( 6)
P240J1  /NUCLID                              ( 7)
P240J1R /NAMEP                               ( 8)
0 0 0  /IGRCH ITIMS IFCUT                    ( 2)
/*
++
//
```

III.1.2 Input Data Requirements for PROF-GROUCH-G II R

All input data in the PROF-GROUCH-G II R code are read in the free format described in Section II, except for input in RESENDD.

III.1.2.1 MCFILE (free format)

This routine is performed only for the initial case to produce a file of control data which is used in the SPINPTM and SPTGRM codes.

BLOCK 1 / 3 /

NCASE Number of admixture background cross sections (< 8).

NGRMX Number of energy groups (< 199).

LPMAX Number of Legendre orders for scattering matrices (< 20)

BLOCK 2 / NCASE /

SIGO Admixture background cross sections(barn) in the Bondarenko scheme.

BLOCK 3 / 2 /

IOPT Option for group boundary.
= 0 give the group boundaries in the energy unit (eV).
= 1 give those in the lethargy unit.

IPDS Output option for master fast library file.
= 0 skip output to the master file.
= 1 output.

BLOCK 4 / NGRMX + 1 /

EN Energy boundaries are read in decreasing order of energy. Do not enter unless IOPT=0.

BLOCK 5 / NGRMX + 1 /

EMAX The highest energy boundary (eV).

DU The lethargy widths from the first energy group to NGRMX group. Do not enter unless IOPT=1.

BLOCK 6 Constant parameters of weighting functions (1/E**a) or A exp(-E/b) sinh(SQRT(c*E)).

ALPHA is a-value of (1/E**a).

A1 is b in the second function.

A2 is c in the second function.

E1 is the boundary energy between both weighting functions (the recommended value is 1 MeV). If ALPHA=0, the parameters are defined as follows;
a = 1.0, b = 1.0E+06 and c = 2.0E-06.

III.1.2.2 RESENDD (special format)*

BLOCK 1 / 4 /

MAT Material number defined in the ENDF/B-IV or JENDL nuclear data file.

TEMP Temperature (°K). (Default value = 0.0)

ERR Tolerance for cross section variation in the interpolation scheme. (Default value = 0.01)

LSIG > 0 , when cross section calculated is smaller than LSIG, the value is replaced in LSIG-value(barns).
< 0 , no effect
*Note; Example of input is read as 'MAT = 1262' and 'LSIG = 0.001' in separate card.

BLOCK 2

LABLE (5H) The statement that LABLE must be read from the first column.

BLOCK 3

TITLE (2A4) Identification of nuclide

BLOCK 4

GO (2H) GO statement must be read.

BLOCK 5

EOF (3H) EOF statement must be read.

BLOCK 6

STOP (3H) STOP statement must be read.

III.1.2.3 SPINPTM (free format)

BLOCK 1 / 5 /

IPRNT = 0 print the infinitely dilulte cross sections.
= 1 detailed results are printed.

ICSTP = 1 only the infinitely dilute cross sections are calculated.
= 0 the effective cross sections are calculated for the admixture cross sections ,SIGO read in MCFILE.

EPS1 convergence criterion in the Romberg integrations.
(ESP1 > 1.0E-05)

IFLXW Option of weighting function.
= 0 (1/E + X(fission)) spectrum.
= 1 collision density spectrum which must be stored in the

file of logical unit (IFLXW). The data are read in the decreasing order of energy in the following record format :
record 1 MAXG, number of data points.
record 2 (E(I), I=1,MAXG), energy points (eV).
record 3 (FAI(I),I=1, MAXG),collision density spectrum

III.1.2.4 XTABPDS (free format)

BLOCK 1 / 6 /

KP = 1 print the results calculated in SPTGRM.
= 0 no effect.

IPROP =-1 print the group cross sections of one-dimensional data.
= 0 print one and two-dimensional data.

ISKIP = 1 group cross sections are written to the master file in the PDS format.
= 0 no output.

ITOT = 0

ICUT = 74

ESP = 0.0

BLOCK 2 / 3 /

IGRCH = 1 process the results calculated by the SPTGRM routine.
= 0 no process.

ITIMS = 1 process the results calculate by the TIMS-1 code.
= 0 no process.

IFTCUT = 1

When IGRCH = 0 and ITIMS = 0 are encountered, the job of XTABPDS is finished.

BLOCK 3 / 3 /

INN Logical unit number of output file in SPTGRM of NTEMP>1, the NTEMP files are set from INN to INN + NTEMP-1. Do not enter it, unless IGRCH = 1.

NCASE Number of admixture cross sections plus 1.
If NCASE = 0, NCASE is defined by 9.
Maximum number is 10.

NTEMP Number of temperatures.

BLOCK 4 / NTEMP /

TEMP Temperature (K)
Do not enter them, unless IGRCH = 1 and NTEMP>0.

BLOCK 5

NAMEP(7H) Member name stored in the master file.
Do not enter it unless IGRCH = 1 and ISKIP = 1.

BLOCK 6 / 1 /

LIN Logical unit number of output file in TIMS-1.
Do not enter it unless ITIMS = 1.

BLOCK 7

NUCLID(8H) Nuclide identification. Do not enter it unless
ITIMS = 1.

BLOCK 8

NAMET(7H) Member name stored in the master file.
Do not enter it unless ITIMS = 1 and ISKIP = 1.

III.1.2.1 Sample Data and JCL-statements in PROF. GROUCH.GIIR

```
//JCLG JOB
// EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER 64902031,HI.TAKANO,0431.110
  T.4 C.5 W.3 I.5 P.0 CLS
 OPTP NOTIFY=J2031,MSGCLASS=R
//*      ******************************************
//*      *         R E S E N D D G                *
//*      ******************************************
//RESENDDG EXEC LMGO,LM='J2031.JFSLOAD', PNM=RESENDDG
//FT01F001 DD DSN=J1615.ENDFB407. DATA,LABEL=(,,,IN),DISP=SHR
//FT02F001 DD DSN=&&BTOB,SPACE=( TRK,(100,10)),DISP=(NEW,PASS),
//    DCB=(RECFM=FB,LRECL=80, BLKSIZE=3200,BUFL=3200,DSORG=
PS),UNIT=WK10
//FT96F001 DD UNIT=WK10,SPACE=(TRK,(100,20))
//FT97F001 DD UNIT=WK10,SPACE=(TRK,(100,20))
//FT98F001 DD UNIT=WK10,SPACE=(TRK,(100,20))
//FT99F001 DD UNIT=WK10,SPACE=(TRK,(100,20))
//SYSIN DD *
 MAT=1297,TEMP=300.0,ERR=0.01
LABEL
PA233-B4
GO
EOF
STOP
/*
//*  ************************************************
//*  *              S P I N P T M                   *
//*  ************************************************
//SPINPTM  EXEC LMGO,LM='J2031.JFSLOAD', PNM=SPINPTM
//FT01F001 DD DSN=J2031.CONTTEST. DATA,DISP=SHR,LABEL=(,,,IN)
//FT02F001 DD DSN=&&BTOB,DISP=(OLD,PASS)
//FT08F001 DD DSN=&&DATA,SPACE=( TRK,(100,10)),DISP=(NEW,PASS),
//    DCB=(RECFM=FB,LRECL=80, BLKSIZE=3200,BUFL=3200,DSORG=
PS),UNIT=WK10
```

```
//FT10F001 DD DSN=J2031.MONJSPTG. DATA,DISP=SHR,LABEL=(,,,IN)
//SYSIN  DD  *
  0 1 0.0001  0 0.0   / IPRNT ICSTP EPS1 IFLXW ECUT
/*
//*      ****************************************
//*      *        S P T G 4 Z 2 T               *
//*      ****************************************
//SPTG4Z2T EXEC LMGO,LM='J2031.JFSLOAD', PNM=SPTGREMO
//FT05F001 DD DSN=&&DATA,DISP=(OLD,DELETE)
//FT01F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT02F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT03F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT04F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT08F001 DD DSN=&&BTOB,DISP=(OLD,DELETE)
//FT09F001 DD DSN=J2031.PA233B4. DATA,DISP=(NEW,CATLG),UNIT=TDS,
//        SPACE=(TRK,(2,1),RLSE)
//FT10F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT21F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT22F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT23F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT30F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT31F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT43F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT44F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT51F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT52F001 DD UNIT=WK10,SPACE=(TRK,(200,30))
//FT53F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT97F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT98F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//FT99F001 DD UNIT=WK10,SPACE=(TRK,(100,30))
//*      ****************************************
//*      *             X T A B P D S            *
//*      ****************************************
//XTABPDS  EXEC LMGO,LM='J2031.JFSLOAD', PNM=XTABPDS
//FT01F001 DD UNIT=WK10,SPACE=(TRK,(50,10))
//FT02F001 DD UNIT=WK10,SPACE=(TRK,(50,10))
//FT21F001 DD DSN=J2031.PA233B4.DATA, DISP=SHR
//USERPDS  DD DSN=J2031.SRACFP.DATA,DISP= MOD
//SYSIN    DD *
0 0 0 0  0 0.0 /KP IPROP ISKIP ITOT ICUT EPS ( 1)
1 0  0  /IGRCH ITIMS IFTCUT                   ( 2)
21 1 1  /LIN NCASE NTEMP                      ( 3)
300.    /TEMP                                 ( 4)
0 0  0  /IGRCH ITIMS IFTCUT                   ( 2)
/*
++
//
```

III.1.3 SPEC : Fission Spectrum Calculation Code

The fission spectrum is calculated by using the energy distribution laws given in ENDF/B. The distribution laws available in SPEC are as follows :

(1) General evaporation spectrum (LF = 5)
(2) Simple fission spectrum (LF = 7)
(3) Evaporation spectrum (LF = 9)
(4) Watt spectrum (LF = 10)

All input data are read in the free format.

BLOCK 1 / 2 /

IMAX Number of energy groups

ISWR Energy boundary
= 0 give the energy boundaries (eV) in the decreasing order of energy.
= 1 give the highest energy boundary (eV) and the lethargy widths at each energy group.
= 2 energy boundaries are read from the member ENBND of the master library.

BLOCK 2 / IMAX + 1 /

E Energy boundaries (eV), if ISWR = 0.

BLOCK 3 / IMAX + 1 /

EMAX The highest energy boundary (eV)

DU Lethargy widths of IMAX groups
Do not enter unless ISWR = 1

BLOCK 4 / 5 /

NUCL(8H) Nuclide identification read from the first column.

MATNO Material number defined in ENDF/B and/or JENDL

IOUT Output option of fission spectra
= 1 Output to the master file
= 0 Only print
=-1 Card output to FT07F001

ICUT = 74 in SRAC-FASTLIB

EINT Incident neutron energy (eV)

BLOCK 3 is repeated until MATNO = 0.

III.1.3.1 Sample Data and JCL-statements in SPEC

```
//JCLG    JOB
//        EXEC JCLG
//SYSIN   DD DATA,DLM='++'
// JUSER 64902031,HI.TAKANO,0431.110
 T.1 C.0 W.1 I.2 OPN
 OPTP NOTIFY=J2031,MSGCLASS=R
//SPEC EXEC LMGO,LM='J2031.JFSLOAD',PNM= SPEC
//USERPDS   DD DSN=J2031.JFS3OLD.DATA, DISP=MOD
//FT10F001  DD DSN=J1615.ENDFB404. DATA,DISP=SHR,LABEL=(,,,IN)
//SYSIN    DD *
70 2  / IMAX ISWR
TH232B4   1296   1 0 20000000.   / NUCL MATNO IOUT ICUT EINT
             0   0 0 0.0          / STOP
```

III.2 Production of the Thermal Neutron Library

The thermal neutron cross section library for the SRAC is generated by a series of codes from the evaluated neutron data files compiled in the ENDF/B format. The data flow is shown in **Fig.III.2-1**.

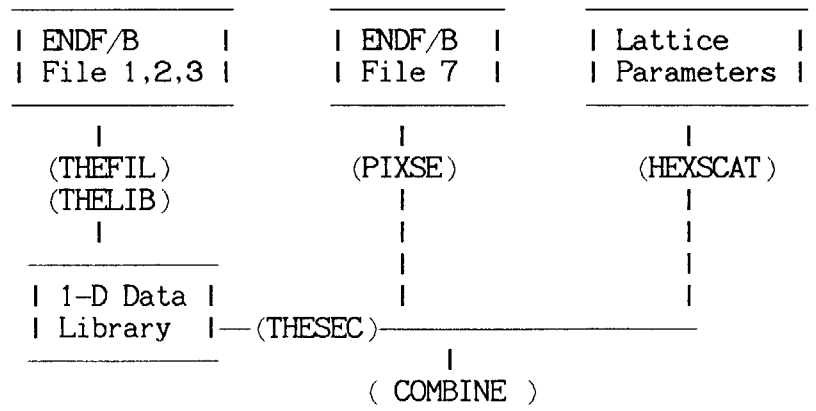


Fig.III.2-1 Flow of Thermal Neutron Data in SRACTLIB

The program THEFIL extracts the data related to thermal neutron fission and absorption into a file which is formed in the ENDF/B format to avoid the repetitive running of the following steps on separate tapes. The files considered are File 1 for ν values of thermal fission, File 2 for resolved resonance parameters, and Mat 1,2,18,101,102,103,.. in File 3 ; tabulated cross sections. The tabulation in File 3 is shortened to include only thermal neutron energy.

The program THELIB converts the data arrangement in EBCDIC mode of a nuclide into a sequential array and stores the array into a PDS file in binary mode with the 'material number' as member name. The structure of PDS file makes easy the search of the nuclide required by the following step for the proper process by nuclide. In the above conversion also the resonance parameters of which resonance energy are located far from thermal neutron energy are replaced into the additional data in File 3 tabulation. On the other hand the resonance parameters in thermal neutron energy range are still kept as resonance parameters to take account of the Doppler effect. In the formulation to treat the Doppler effect, not only Breit-Wigner formula but also Adler-Adler formula can be processed.

The routine COMBINE yields group constants of a nuclide for thermal neutron energy in the format of the SRAC thermal library by combining the data from 1) the routine PIXSE (Ref.75) for thermal scattering expressed by scattering law , $S(\alpha,\beta)$ or by free gas model, 2) the routine THESEC for capture and fission, and 3) the routine HEXSCAT (Ref.19) for coherent scattering of crystalline material if required.

The basic data file must be compiled in the ENDF/B format (Ref.(1)); fission neutron yield in File 1, resonance parameters in File 2, capture and fission cross section in File 3, and scattering law in File 7.

As the thermal cross sections are temperature dependent, the

calculations are generally executed on consecutive temperature points and the results are referred by the temperature index.

For the nuclide which has resonance level in thermal neutron range (up to 3.928 eV), we may provide the table of self-shielding factors to get the effective cross sections.

The computed results are stored in a PDS file where several members are produced per nuclide as follows;

MEMBER NAME	CONTENTS
'CzzmcOOO'	a control member
'KzzmcOOt'	P0 matrix with capture, fission, and total vectors
'MzzmcOOt'	P0 matrix after transport correction with capture, fission, and total vectors (alternate of matrix K)
'PzzmcOOt'	P1 matrix (if required)
'NzzmcOOt'	P0 matrix after second order expanded transport correction (alternative combination with matrix P instead of that of K and P)
'FzzmcOOt'	shielding factor tabulation (if required)

where 'zz' denotes element chemical symbol as described in Sect. II.8, and 'm' for last digit of mass number to discriminate isotopes, 't' for the temperature index as shown in Dictionary VII.6, and 'c' for chemical compound state as shown in Dictionary VII.2. Matrices K, M, P, and F are organized to have the same length. The contents of each member will be described in Sect.V.2.

Input requirements of THEFIL

To the THEFIL, the following five items are required.

Keyword	Key codes	Remarks	Defaulted
TAPE	OLD NEW	The output tape is old or new	NEW
OLD-MAT		The last material on the old output tape if any	-1
NEW-MAT		The last material to be processed in the input tape	-1
EMAX		The thermal cut off energy	5.0
	PRINT	Print or not detailed data on the output tape	No print

Input requirements of SRACTLIB

The input requirement is designed as little as possible, hence the information stored in a PDS file of DD name THCON supplies normal input data which may be modified beforehand by using the code PDSEDT decsribed in Sect.III.6.

BLOCK 1 Title (72H)

BLOCK 2 Number of cases (nuclides) /1/

BLOCKs from 3 through 10 are to be repeated by case

BLOCK 3 Control integers for a nuclide /10/

MATID Material (nuclide) identification number in ENDF/B format input file

MATID2 Material identification number of secondary nuclide in the case of chemical compound for which the scattering law is given to the principal nuclide only. We have to cover whole thermal energy range beyond the range in which the scattering law is given by using free gas model to complete the matrix. Enter 0 if not required.

IADDP Indicator whether or not the scattering of the secondary nuclide is included in the scattering law S(alpha,beta)
=0 no
=1 yes

IFT Indicator to compose the shielding factor table
=0 skip
=1 compose

IFFS Indicator whether or not fission cross sections are included
=0 no
=1 yes

IHEXS Indicator to add coherent elastic cross sections to scattering
=0 skip
=1 add

NTEMP Number of temperature points to calculate; the lower NTEMP points in the tabulated temperatures are taken.

ISAB Specification of scattering law
=0 free gas model and calculate P0 only
=1 read S(alpha,beta)
=-1 free gas model and calculate P0 and P1

IFACT Integer number of the principal nuclide contained in the compound to which the scattering law is given. For example , to hydrogen in benzene C6H6 IFACT = 6.

MATRIX Indicator to calculate scattering matrix
=0 skip to calculate scattering cross section and fill 0 scattering part in matrix
=1 calculate scattering matrix

BLOCK 4 Nuclide identification (8H) to the SRAC library as described in Sect.II.2.

BLOCK 5 Parameters for scattering of principal nuclide; required if MATRIX = 1 /2/

SFREE Free atom scattering cross section

AMASS Atomic mass

BLOCK 6 Nuclide identification (8H) of the secondary nuclide; required if IADD = 1

BLOCK 7 Parameters for scattering of secondary nuclide; required if IADD = 1 /2/

SFREE2 Free atom scattering cross section

AMASS2 Atomic mass

BLOCK 8 Parameters for coherent elastic scattering; requires if IHEXS = 1 /4/ (see Ref(19))

SCOH Coherent cross section (barn)

ALPHA Alpha (cm) : magnitude of base plane lattice vector

C c (cm) : magnitude of lattice vector out of base plane

EXACT Cut off energy to switch the exact treatment (eV)

BLOCK 9 /NTEMP/ required if IHEXS = 1

WAL2 Temperature dependent Debye-Waller integral/AMASS (non-dimensional)

BLOCK 10 Integers for scaterring matrix /4/

NG Order of Gauss approximation for energy integration within group; enter = 3, 5, or 7

NGA Order of Gauss approximation for angular integration; enter = 3, 5, or 7

ISGMN Indicator for Legendre expansion; inefficient if ISAB = 0
= 0 consistent P0 only
= 1 consistent P0 and P1
=-1 P0 with extended transport correction of order 1
=-2 P0 with extended transport correction of order 2 and P1

ISWT not used; enter 0 always

Here we shall describe the contents of THCON file (PDS) which contains built-in values which are necessary in the calculation but not in the card input.

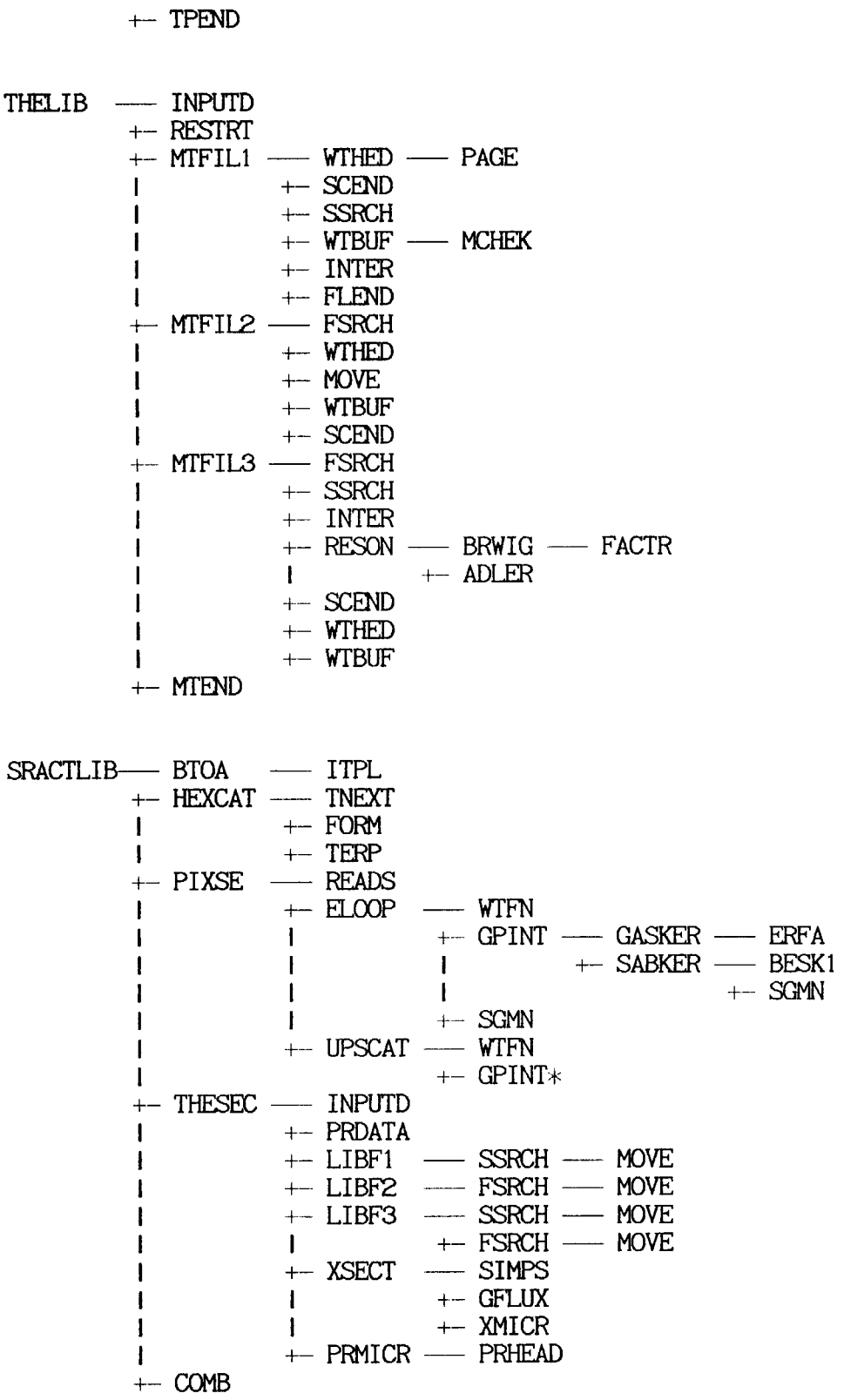
MEMBER		LENGTH		CONTENTS
SIGZ	*	NSIG <8>	*	Sig 0 : admixture cross sections for self-
	*		*	shielding factors; the same values must be used
	*		*	to form the table in the fast neutron library.
	*		*	< 0.1, 1.0, 10., 100., 1000., 1.E+4, 1.E+5,
	*		*	1.E+6 >
	*		*	
TEMP	*	NTEMP <11>	*	Tabulated material temperatures: Tm
	*		*	(see Dictionary VII.6)

```
        *              *
TEMPN   *  NTEMP <11> * Tabulated neutron temperature: Tn <Tn=Tm+50>
        *              *
ENERGY  *  NET+2       * Number of groups: NET and boundary energies
        *    <48+2>    * (see Dictionary VI.4)
        *              *
IPRT    *      5       * Print control
        *              * 1 : S(alpha,beta) interpolation process <0>
        *              * 2 : HEXSCAT process                     <0>
        *              * 3 : fission and capture process         <0>
        *              * 4 : scattering process                  <0>
        *              * 5 : combination process                 <0>
        *              * IPRT(N)=0  message only
        *              *       =1  message and input
        *              *       =2  message and detailed edit
        *              *
DEFAULT*       7       * 1 : dummy Debye-Waller factor for PIXSE <0.0>
        *              * 2 : factor to build up consecutively upper
        *              *     groups for calculating up-scatter cross
        *              *     section <1.13>
        *              * 3 : truncation criterion for differential
        *              *     scattering vector <0.0001>
        *              * 4 : y value as y*kTn is cut off between Maxwell
        *              *     and 1/E spectrum  <5.0>
        *              * 5 : reciprocal width for effective width model;
        *              *     if non-zero value entered, the free gas
        *              *     model is substituted. <0.>
        *              * 6 : NAV value for HEXSCAT  <0>
        *              * 7 : NAVX value for HEXSCAT <2>
*************************************************************************
```

The data values enclosed by < > are standard values.

Now we describe the overlay structure of the SRACTLIB code. As shown below the codes THEFIL, and THELIB are separated from the main structure because the aboves are used only once for an original data file.

```
THEFIL ── INPUTD
       +- RESTRT
       +- MTFIL1 ── MCHEK
       |            +- WTHED ── PAGE
       |            +- SCEND
       |            +- SSRCH
       |            +- WTBUF
       +- MTFIL2 ── WTHED
       |            +- WTBUF
       |            +- SCEND
       |            +- FSRCH
       |            +- MCHEK
       +- MTFIL3 ── FSRCH
       |            +- MCHEK
       |            +- WTBUF
       |            +- SCEND
       +- MTEND
       +- MSRCH
```


+- TPEND
THELIB —— INPUTD
+- RESTRT
+- MTFIL1 —— WTHED —— PAGE
+- SCEND
+- SSRCH
+- WTBUF —— MCHEK
+- INTER
+- FLEND
+- MTFIL2 —— FSRCH
+- WTHED
+- MOVE
+- WTBUF
+- SCEND
+- MTFIL3 —— FSRCH
+- SSRCH
+- INTER
+- RESON —— BRWIG —— FACTR
+- ADLER
+- SCEND
+- WTHED
+- WTBUF
+- MTEND
SRACTLIB—— BTOA —— ITPL
+- HEXCAT —— TNEXT
+- FORM
+- TERP
+- PIXSE —— READS
+- ELOOP —— WTFN
+- GPINT —— GASKER —— ERFA
+- SABKER —— BESK1
+- SGMN
+- SGMN
+- UPSCAT —— WTFN
+- GPINT*
+- THESEC —— INPUTD
+- PRDATA
+- LIBF1 —— SSRCH —— MOVE
+- LIBF2 —— FSRCH —— MOVE
+- LIBF3 —— SSRCH —— MOVE
+- FSRCH —— MOVE
+- XSECT —— SIMPS
+- GFLUX
+- XMICR
+- PRMICR —— PRHEAD
+- COMB

III.3 Plotting and Printing of Neutron Flux and Cross Section

An auxiliary code MFPLOT permits the plotting of 1) Spatial distribution of R-region-(Zone-)wise neutron fluxes of specified groups or/and average of specified groups, 2) Energy spectrum of specified X-region, 3) Energy dependence of specified reactions of a set of macroscopic cross sections. Another code MACRPR permits printing the above quantities.

Input Requirements of MFPLOT

I.1 LABEL (A8) Member name in PDS file to be plotted. A member of such a name caseA002 (the case name in the first four characters and 00 on sixth and seventh characters) in the FLUX file keeps the spatial distribution of neutron fluxes. A member of the form of caseA0n2 (the case name on the first four characters and X-region number on the seventh character) in the FLUX file keeps the neutron spectrum of the X-region.
Any member in the MACRO file keeps the macroscopic cross section which may be plotted.

I.2 MACRO(I) I=1,8

If MACRO(I) =0 skip
If MACRO(I) =0 Plot the cross section of I-th reaction

Reaction is assigned by I as,

I=1 Σact Activation cross section
I=2 Σf Fission cross section
I=3 νΣact Nu*fission cross section
I=4 Σt Total cross section
I=5 X Fission neutron yield
I=6 D1 Diffusion coefficient 1
I=7 D2 Diffusion coefficient 2
I=8 Σa Absorption cross section

I.2' IRFLX =0 Skip
=1 Plot the spatial distribution of neutron flux

I.3 IXFLX =0 Skip
=1 Plot the neutron spectrum

I.3' IGEOM,IGROUP's

IGEOM =0 Equal interval by region
=1 Sphere geometry in the order of region number from left to right
=2 Slab geometry in the order of region number from the center to outer
=3 Cylinder geometry in the order of region number from the center to outer

IGROUP Enter the group numbers of which fluxes are to be plotted. If a negative number entered, the average between the group entered before and the current group is plotted. For example an input record (2 4 10 5 -8) indicates to plot the fluxes of groups

2,4,and 10 and the average among groups 5,6,7 and 8.

A combination of I.1, I.2, and I.3 or another combination of I.1, I.2', and I,3' is used. The former is for the plot of the spatial distribution of neutron fluxes and the latter is for the plot of cross sections and the neutron spectra.

Sample JCL and Input of MFPLOT

```
   T.2 C.3 W.0 P.0 I.2  GRP
 OPTPMSGCLASS=R,PASSWORD=?
//MFPLOT EXEC LMGO,LM='J1480.LAMPLM',SYSOUT=R,PNM=MFPLOT
//MACRO   DD DSN=J1480.IALBMAC.DATA,DISP=SHR,LABEL=(,,,IN)
//FLUX    DD DSN=J1480.IALBFLX.DATA,DISP=SHR,LABEL=(,,,IN)
//FT10F001 DD UNIT=WK10,SPACE=(TRK,(10,10)),
//         DCB=(LRECL=80,BLKSIZE=3120,RECFM=FB),DISP=(NEW,DELETE)
//FT11F001 DD DSN=&&DATA,UNIT=WK10,SPACE=(TRK,(10,10)),
//         DCB=(LRECL=80,BLKSIZE=3120,RECFM=FB),DISP=(NEW,PASS)
//SYSIN   DD *
TRX1T002 / Sample 1
0 /
3     10 15 25 26-29/
TRX1T012 / Sample 2
0 1 0 1 0 1 0 1 /
1/
TRX1A012 / Sample 3
0 1 0 1 0 1 0 1 /
1/
MOD4TM62 / Sample 4
0 1 0 1 0 1 0 1 /
1/
DRIVTM22 / Sample 5
0 1 0 1 0 1 0 1 /
1/
/*
//GPLP EXEC LMGO,LM='J3268.GPLP',PNM=TEMPNAME
// EXPAND GRNLP,SYSOUT=M
//FT04F001 DD UNIT=WK10,SPACE=(TRK,(10,10)),
//         DCB=(LRECL=6208,BLKSIZE=6212,RECFM=VBS),DISP=(NEW,DELETE)
//FT10F001 DD UNIT=WK10,SPACE=(TRK,(10,10)),
//         DCB=(LRECL=80,BLKSIZE=3120,RECFM=FB),DISP=(NEW,DELETE)
//SYSIN    DD DSN=&&DATA,DISP=(OLD,DELETE)
++
//
```

As shown in the above JCL list, the execution is divided into two steps. The latter step named GPLP is a universal graph plotting code (Ref.(44). By the use of GPLP, our programming effort for plotting is concentrated in preparation of the data and selection of options.

Now we describe how to use the MACRPR. We are required two input files ; one for the macroscopic cross section file, the other for the neutron flux file. The sysin input is simply to indicate the member name. If the member is not found in the flux file, only the cross sections are interpreted. A blank name terminate the execution.

Sample JCL and Input of MACRPR

```
  T.0 C.0 W.1 P.0 I.0
```

```
 OPTPMSGCLASS=R,MSGLEVEL=(1,1),NOTIFY=J1480,PASSWORD=?
// EXEC LMGO,LM='J1480.SRACSC',PNM=MACRPR
//MACRO   DD DSN=J1480.IALBMAC.DATA,DISP=SHR,LABEL=(,,,IN)
//FLUX    DD DSN=J1480.IALBFLX.DATA,DISP=SHR,LABEL=(,,,IN)
//SYSIN   DD *
TRX1T002
TRX1T012
TRX1A012

++
//
```

III.4 PDS Edit Programs

Two service programs PDSEDT and PDSEDGRP have been used to edit the contents of PDS files.

The program PDSEDT executes the commands listed below in the input format.

```
*PRINT    NAME1        TYPE < FIRST LAST >
*PUNCH    NAME1        TYPE < FIRST LAST >
*PUT      NAME1        TYPE   LAST  list
*DELETE   NAME1             < FIRST LAST >
*UPDATE   NAME1        TYPE   FIRST LAST  list
*RENAME   NAME1 NAME2
*DIVIDE   NAME1 NAME2        FIRST LAST
*CONCAT   NAME1 NAME2  NAME3
*INFORM   NAME1
*TRANSFER NAME1 NAME2
*FIN
```

where

NAME1,NAME2,NAME3 : member name

TYPE : data type indicated by the codes; I (integer), E (floating number), A (character), B (binary)

FIRST,LAST : data location in the one-dimensional array

list : list records

* before command code must be entered on column 1 of input record . Command code is identified only by the first four characters including *. Succeeding items must be separated by more than one blank. The data corresponding to 'list' are entered on new record in the 'free' format.

PRINT : print the contents of NAME1 < from FIRST to LAST location > in TYPE format.

PUNCH : punch the contents of NAME1 < from FIRST to LAST location > in TYPE format.

PUT : put the new member by reading 'list' record of length LAST. If TYPE is indicated B, the data are read from a sequential file FT08F001.

DELETE: rewrite the member by deleting the contents of the member from

FIRST to LAST. If the location is not assigned, the whole member is deleted.

UPDATE: update the contents of the member by feeding the data from FIRST to LAST locations.

RENAME: rename the member from NAME1 to NAME2.

DIVIDE: put a new member NAME2 by extracting the data from FIRST to LAST in NAME1, and rewrite NAME1 composed of the remaining data

CONCAT: concatenate two members NAME1 and NAME2 and put a new member NAME3. If the member name NAME3 is identical with NAME1 or NAME2, the member is rewritten.

INFORM: print the array length of the member.

TRANSF: transfer the member NAME1 in DDN=USERPDS to DDN=USERPDS2 by the name NAME2.

FIN : terminate the execution.

Sample JCL and Input of PDSEDT

```
      P.0 I.0 T.0 C.0 W.0
 OPTPMSGCLASS=R,NOTIFY=J1480,PASSWORD=?
//    EXEC  LMGO,LM='J1480.LAMPLM',PNM=PDSEDT
//USERPDS DD DSN=J1480.THERMLB2.DATA,DISP=MOD
//SYSIN DD *
*UPD  CH01P000 I 2 2
  6
*FIN
//
```

Sample Macroscopic Command for PDSEDT

```
PROC 1 DSN1 DSN2(OFF) DSN3(OFF)
/* DSN1 PDS FILE TO READ WRITE DELETE RENAME DDN=USERPDS
/* DSN2 PDS FILE TO TRANSFER                DDN=USERPDS2
/* DSN3 PS FILE TO READ WRITE IN BINARY MODE DDN=FT08F001
CONTROL PROMPT LIST MSG
FREE DA(&DSN1)
FREE ATTRLIST(A)
ATTR A  LRECL(133) RECFM(U)
FREE F(FT05F001)
FREE F(FT06F001)
ALLOC DA(*) F(FT05F001)
ALLOC DA(*) F(FT06F001) USING(A)
ALLOC DA(&DSN1) F(USERPDS) OLD
IF &DSN2 NE OFF THEN DO
FREE DA(&DSN2)
ALLOC DA(&DSN2) F(USERPDS2)
END
IF &DSN3 NE OFF THEN DO
FREE F(FT08F001)
FREE DA(&DSN3)
ALLOC DA(&DSN3) F(FT08F001)
END
CALL 'J1480.LAMPLM.LOAD(PDSEDT)'
FREE DA(&DSN1)
FREE ATTRLIST(A)
IF &DSN2 NE OFF THEN FREE DA(&DSN2)
IF &DSN3 NE OFF THEN FREE DA(&DSN3)
END
```

The program PDSEDGRP executes the commands as the PDSEDT does to a

group of members.

A specification of a group is to indicate the first and the last members among the member names ordered alphabetically; as
FIRSTNAME LASTNAME
where FIRSTNAME,LASTNAME are member names to indicate the range. If the member does not exist, the members within the range are processed.

Another specification is to indicate the common string in the member names; as
STRING—— , -STRING— , ——STRING , or specially ————
where STRING is partial string of eight characters of member name, and the character '-' denotes the masking. The string may be on any column in a member name. The members which have the common string on the same columns are processed. If the member name is fully masked, all member are in a group.

Here we show the combination of commands and the group specification

1 PRINT FIRSTNAME LASTNAME TYPE < FIRST LAST >
1' PRINT STRING—— TYPE < FIRST LAST >

Print the contents of the members, where TYPE is a character to indicate the data type as I (integer), E (floating number), and A (character), FIRST and LAST are the first and last location of data in one-dimensional array. If the locations are not specified, the whole data in each member are printed.

2 INFORM FIRSTNAME LASTNAME
2' INFORM STRING——

Print the length of each member.

3 DELETE FIRSTNAME LASTNAME
3' DELETE STRING——

Delete the members.

4 TRANSFER FIRSTNAME LASTNAME
4' TRANSFER STRING——

Transfer the members from DDN=USERPDS to DDN=USERPDS2

5 RENAME STRING1—— STRING2——

Rename the members which have common string STRING1 to the names which have STRING2 instead of STRING1.

6 FIN

Terminate the execution.

MACRO-COMMAND 'PDSEDGRP'

Following is a list of the macroscopic command 'PDSEDGRP' to use the program.

```
PROC 1 NAME1 NAME2(OFF) NAME3(OFF) NAME4(OFF)
/* NAME1 PDS FILE  TO READ WRITE DELETE & RENAME  DDN=USERPDS
/* NAME2 PDS FILE  TO COPY                        DDN=USERPDS2
/* NAME3 PS FILE   TO PRINT IN EBCDIC MODE   DDN=FT10F001
/* NAME4 PS FILE   TO WRITE IN BINARY MODE   DDN=FT11F001
CONTROL PROMPT LIST MSG
FREE F(FT01F001)
FREE F(FT05F001)
FREE F(FT10F001)
.DISK DD(FT01F001)
ALLOC DA(*) F(FT05F001)
ALLOC DA(*) F(FT06F001)
ALLOC DA(*) F(FT10F001)
FREE DA(&NAME1)
ALLOC DA(&NAME1) F(USERPDS)
IF &NAME2 NE OFF THEN DO
FREE DA(&NAME2)
ALLOC DA(&NAME2) F(USERPDS2)
END
IF &NAME3 NE OFF THEN DO
FREE F(FT10F001)
FREE DA(&NAME3)
ALLOC DA(&NAME3) F(FT10F001)
END
IF &NAME4 NE OFF THEN DO
FREE DA(&NAME4)
FREE F(FT11F001)
ALLOC DA(&NAME4) F(FT11F001)
END
CALL 'J1480.SRACSC.LOAD(PDSEDGRP)'
FREE DA(&NAME1)
IF &NAME2 NE OFF THEN FREE DA(&NAME2)
IF &NAME3 NE OFF THEN FREE DA(&NAME3)
IF &NAME4 NE OFF THEN FREE DA(&NAME4)
END
```

Sample I/O list of the macro-command 'PDSEDGRP'

```
.PDSEDGRP SAMPLE.DATA DSN2(IALBMAC.DATA) NAME3(PRINT.DATA)
 ALLOC DA(*) F(FT05F001)
 ALLOC DA(*) F(FT06F001)
 ALLOC DA(SAMPLE.DATA) F(USERPDS)
 ALLOC DA(IALBMAC.DATA) F(USERPDS2)
 ALLOC DA(PRINT.DATA) F(FT10F001)
 CALL 'J1480.SRACSC.LOAD(PDSEDGRP)'
 00015?INF ----------
 00015?PRINT CC-------  I  1  10
 00015?PRINT CB-------  CD030000  I   1   10
 00015?TRANSFER CB000000 CD030000
 00015?TRA CF-------
 00015?RENAME CC------- CX-------
 00015?DELETE CAL80000 CB020000
 00015?DEL CF-------
 00015?INF ----------
```

```
00015?FIN
FREE DA(SAMPLE.DATA)
FREE DA(IALBMAC.DATA)
FREE DA(PRINT.DATA)
END
READY
```

Sample output list of PDSEDGRP on PRINT.DATA

```
 TRANSFER CB000000 CD030000

 MEMBER CB000000 OF LENGTH    41 IS STORED IN USERPDS2 FILE
 MEMBER CB010000 OF LENGTH    41 IS STORED IN USERPDS2 FILE
 MEMBER CCDN0000 OF LENGTH    41 IS STORED IN USERPDS2 FILE
 MEMBER CC090000 OF LENGTH    41 IS STORED IN USERPDS2 FILE
 MEMBER CC020000 OF LENGTH    41 IS STORED IN USERPDS2 FILE
 MEMBER CD020000 OF LENGTH    41 IS STORED IN USERPDS2 FILE

TRA  CF--------

 MEMBER CF5N0000 OF LENGTH    41 IS STORED IN USERPDS2 FILE
 MEMBER CF5R0000 OF LENGTH    41 IS STORED IN USERPDS2 FILE
 MEMBER CF5S0000 OF LENGTH    41 IS STORED IN USERPDS2 FILE
 MEMBER CF9N0000 OF LENGTH    41 IS STORED IN USERPDS2 FILE
 MEMBER CF9R0000 OF LENGTH    41 IS STORED IN USERPDS2 FILE
 MEMBER CF9S0000 OF LENGTH    41 IS STORED IN USERPDS2 FILE


FIN
```

IV User Information

IV.1 Core Storage and Machine Time Requirement

The SRAC is designed to work normally in 768 KB core (C2 rank) . The standard load module with 240 KB (60,000 words) work area requires 420 KB as program size. The remaining area may be used as I/O buffer area. The accumulation of the graph plotter output may cause the shortage of core extent when the NLP plotter is used. It is to be noted that any routine which may require a large amount of core storage shares the labelled common /work/ as work area. At each entry point of such a routine, the storage requirement is announced in message print. If the user wants to extend the core requirement, the following procedure may be used to create a temporary module with enlarged work area.

```
*********1*********2*********3*********4*********5*********6*********7
//JCLG JOB
// EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER statement
> T.1 C.2 W.1 P.0 I.4
//   EXEC FORTHE,A=NOSOURCE
//FORT.SYSIN DD *
C********** SRAC MAIN PROGRAM TO BE MODIFIED *****************
      COMMON /MAINC/ IOPT(95),MEMORY,DUMMY(24),II(880)
C
C    FOR THE EXECUTION OF SRAC WITH ENLARGED CORE STORAGE,
C    CHANGE THE ARRAY LENGTH OF LABELLED COMMON /WORK/
C    FROM 60,000 WORDS TO THE DESIRED VALUE,
C    AND SET THE VARIABLE 'MEMORY' TO THIS VALUE
C
>      COMMON /WORK/ A(60000)
       CALL DTLIST
       CALL ICLEA(IOPT,1000,0)
>      MEMORY = 60000
       CALL SRAC
       STOP
       END
/*
//    EXEC LKED,A=OVLY,B=NOLIST,GPLIB=PNL,CNTL=NO
>//LINK.SYSLMOD DDDSN=J????.SRACTEMP.LOAD,SPACE=(TRK,(150,10,1),RLSE),
//     DCB=(BLKSIZE=19069,RECFM=U),UNIT=TSSWK,DISP=(NEW,CATLG,DELETE)
//LINK.OLDLM DD DSN=J1480.SRACLM.LOAD,DISP=SHR,LABEL=(,,,IN)
//LINK.SYSIN DD DSN=J1480.OVERLAYH.DATA(SRAC),DISP=SHR,LABEL=(,,,IN)
>//SRAC EXECLMGO,LM='J????.SRACTEMP',PNM=SRAC2
//*****************************************************************
//*                                                               *
//*      DD STATEMENTS FOLLOWS                                    *
//*                                                               *
//*****************************************************************
++
//
```

The machine time requirement varies in large scale by the number of

energy groups, the geometry (specially 1D or 2D), the number of spatial division, and the repetitive cell calculations by burn-up step. Also the use of PDS files which is always accompanied by file open/close process at each read/write execution increases the core time (not cpu time).

For example a series of burn-up calculation for a DIDO type lattice cell consumes 7 cpu minutes by FACOM M-200 computer by the following specifications.

Geometry : One dimensional cylinder
18 groups, 19 spatial regions in fast range
4 groups, 19 spatial regions in resonance range
31 groups, 25 spatial regions in thermal range
Method : Collision probability (in fast, thermal)
also in resonance range by 5,000 energy pts. 19 region collision probability method
Number of burn-up steps : 6 (including fresh step)

In the above example the computer time is almost used in the one dimensional numerical integration for collision probabilities. It is to be noted that the another process for resonance integral by IR method needs the same order of computing time as the exact method, because the latter process saves the computing time for collision probability calculation by interpolation of the tabulated values which correspond to 11 group calculation.

Another series of similar calculations for a MTR type fuel plate cell requires only 50 cpu seconds due to the analytical expression of collision probability for one dimensional slab, and fewer spatial division (3 regions).

It is not necessary to mention about the machine time requirement for the well experienced routines ANISN, TWOTRAN, and CITATION, however, we should mention about the one dimensional diffusion routine TUD sometimes requires several ten cpu seconds when it is used in 60-70 group structure (30-40 thermal groups) for a well reflected core due to slow convergence in thermal flux distribution.

IV.2 Job Control Statements

The following list is an example of JCL for FACOM M-200 computer, where the DD statements listed are required in a cell burn-up calculation to use collision probability method. Several PDS files which are used in work area may be replaced by catalogued file.

```
*********1*********2*********3*********4*********5*********6*********7
//JCLG JOB
// EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER statement
 T.4 C.2 W.1 P.0 I.4 GRP
 OPTPNOTIFY=J????,MSGCLASS=?,PASSWORD=?
//SRAC EXEC LMGO,LM='J1480.SRACLM',PNM=SRAC2,OBSIZE=274
?//SYSINDDSN=J????.????.DATA(????),DISP=SHR,LABEL=(,,,IN)
// EXPAND GRNLP
//FT03F001 DD DSN=&&WRK03,SPACE=(TRK,(30,10)),UNIT=WK10,
//          DCB=(RECFM=VS,BLKSIZE=800,BUFNO=2)
//FT04F001 DD DSN=&&WRK04,SPACE=(TRK,(30,10)),UNIT=WK10,
//          DCB=(RECFM=VS,BLKSIZE=800,BUFNO=2)
//FT21F001  DD DSN=&&WRK21,SPACE=(TRK,(30,10)),UNIT=WK10,
//          DCB=(RECFM=VS,BLKSIZE=800,BUFNO=2)
//FT22F001  DD DSN=&&WRK22,SPACE=(TRK,(2,1)),UNIT=WK10,
//          DCB=(RECFM=VS,BLKSIZE=800,BUFNO=2)
//FT32F001  DD DSN=&&WRK32,SPACE=(TRK,(20,5)),UNIT=WK10,
//          DCB=(RECFM=VS,BLKSIZE=800,BUFNO=2)
//FT33F001  DD DSN=&&WRK33,SPACE=(TRK,(20,5)),UNIT=WK10,
//          DCB=(RECFM=VS,BLKSIZE=800,BUFNO=2)
//FT50F001  DD DSN=J1480.BURN.DATA(ENDFB2),DISP=SHR,LABEL=(,,,IN)
//FT51F001  DD DSN=&&WRK51,SPACE=(TRK,(5,1)),UNIT=WK10,
//          DCB=(RECFM=VBS,BLKSIZE=6212,LRECL=6208,BUFNO=2)
//FT52F001  DD DSN=&&WRK52,SPACE=(TRK,(2,1)),UNIT=WK10,
//          DCB=(RECFM=VS,BLKSIZE=800,BUFNO=2)
//FT81F001  DD DSN=&&WRK81,SPACE=(TRK,(30,10)),UNIT=WK10,
//          DCB=(RECFM=VBS,BLKSIZE=4084,LRECL=4080,BUFNO=2)
//FT82F001  DD DSN=&&WRK82,SPACE=(TRK,(30,10)),UNIT=WK10,
//          DCB=(RECFM=VBS,BLKSIZE=4084,LRECL=4080,BUFNO=2)
//FT83F001  DD DSN=&&WRK83,SPACE=(TRK,(30,10)),UNIT=WK10,
//          DCB=(RECFM=VBS,BLKSIZE=4084,LRECL=4080,BUFNO=2)
//FASTP     DD DSN=J1480.FASTLIB.DATA,DISP=SHR,LABEL=(,,,IN)
//THERMALP  DD DSN=J1480.THERMLB2.DATA,DISP=SHR,LABEL=(,,,IN)
//MCROSS    DD DSN=J1480.MCROSS2.DATA,DISP=SHR,LABEL=(,,,IN)
//MACROWRK   DD DSN=&&WRKMCW,SPACE=(TRK,(50,5,5)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(,PASS)
//MACRO     DD DSN=&&WRKMCR,SPACE=(TRK,(50,5,5)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(,PASS)
//FASTU     DD DSN=&&WRKFTU,SPACE=(TRK,(50,5,5)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(,DELETE)
//THERMALU   DD DSN=&&WRKTHU,SPACE=(TRK,(50,5,5)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(,DELETE)
//MICREF    DD DSN=&&WRKMIC,SPACE=(TRK,(5,1,2)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(,PASS)
//FLUX      DD DSN=&&WRKFLX,SPACE=(TRK,(5,1.5)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(,PASS)
//FT99F001  DD SYSOUT=R,DCB=(RECFM=FBA,LRECL=133,BLKSIZE=5320)
++
//
```

Several sequential files are required when the component routines ANISN, TWOTRAN, or/and CITATION are used.
The following table shows the assignment of numerical DD name to the variable name defined by Fortran statements. Their physical contents are explained in Chapt.V.

DD Name	MAIN	ANISN	TWOTRAN	CITATION	RECFM	LRECL
FT01		NT1	NEDIT	I01	V	
FT02		NT2		I02	V	
FT03	(MACROF)	NT3	IVMESH	I03	V	
FT04	(SRAC)	NT4	NEXTRA		V	
FT05		NIN	NINP		V	
FT06	NOUT1	NOUT1	NOUT		FBS	137
FT07						
FT08		NT8	NAFLUX		V	
FT09			LAFLUX	IOFLX	V	
FT10		NT6,NT7	ISOTXS	IX77	V	
FT11			ISNCON	IX78	V	
FT12			NDUMP1		V	
FT13			NDUMP2	IX80	V	
FT14			IZMESH	IX81	V	
FT15				IX82	V	
FT16				IX83	V	
FT17				IX84	V	
FT18				IX85	V	
FT19				IX86	V	
FT20				IX87	V	
FT21	(PIJ)				V	
FT22	(DR,DZ)			IX137	V	
FT31	(CVMACT)			IX138	FB	80
FT32	(MACROF)	NSOUC	IFIXSR	IX139	V	
FT33	ITFLUX				V	
FT50	(BURNUP)				FB	136
FT51	(BURNUP)				V	
FT52	(MICREF)				V	
FT81	(PIJ)				VBS	4090
FT82	(PIJ)				VBS	4090
FT83	(PIJ)				VBS	4090
FT91				IOIN	FB	80
FT92	(CVMACT)				FB	80
FT99	NOUT2				FBS	137

* Note Name enclosed by ( ) is the subroutine name to use the unit.

IV.3 Program Flow Diagram

The program flow diagram is shown in **Fig.IV.3-1**. We shall give the brief description to the process following the flow.

INPUT1 : Read the control data and energy structure information

USER.FASTLIB : Compose the user fast neutron library on FASTU

USER.THERMAL.LIB : Compose the user fast neutron library on THERMALU

PIJ.INPUT : Read the input for the collision probability method and compose the trace table on FT81,82,83

PLOT.GEOMETRY : Figure out the cell geometry and region number map

SN.INPUT : Read the input for ANISN and/or TWOTRAN

DIFFUSION.INPUT : Read the input for TUD and/or CITATION

INPUT2 : Read in the material specification

BURN.IN : Read in the input for burn-up calculation and compose the case-dependent nuclide and chain tables

MACRO.FAST : Compose macroscopic cross section sets with self shielding factor in fast neutron energy range

SHIELD : Calculate the self shielding factor by table look-up

PIJ2(M) : Calculate collision probabilities by M-region for Dancoff correction factor

MACRO.THERMAL : Compose macroscopic cross section sets with self shielding factor in thermal neutron energy range

GAM.P1B1 : Modify transport cross sections and diffusion coefficients in the macroscopic cross section sets prepared by MACRO. FAST and MACRO.THERMAL by P1 or B1 approximation for homogeneous material and for component materials in a cell. In the latter case the neutron flux and current are calculated in a homogenized material.

IR.METHOD : Modify capture cross section in the macroscopic cross section sets by resonance integral for resolved level by IR approximation

MCROSS : Compose the user's resonance neutron cross section file when double heterogeneity is treated by PEACO option

PEACO : Calculate ultra-fine neutron spectrum in multiregion cell in Resonance I energy range by collision probability method, and modify the absorption and fission cross sections of resonant material.

PIJ(R) : Calculate collision probabilities for R-region

PIJ(T) : Calculate collision probabilities for T-region

MIX.XSECTION : Homogenize the macroscopic cross section by X-region

PIJ3 : Solve linear equations by S.O.R for collision probability method

SN : Solve Sn equations

DIFFUSION : Solve diffusion theory equations

HOMOSP : Solve a bare reactor equation by P1 or B1 approximation

CONDENSE : Collapse the energy structure of the macroscopic cross sections to get few group cross sections

BURNUP : Calculate the change of nuclide concentrations in a burn-up step

CVMACT : Convert the format of the macroscopic cross sections for separate execution of the original CITATION code

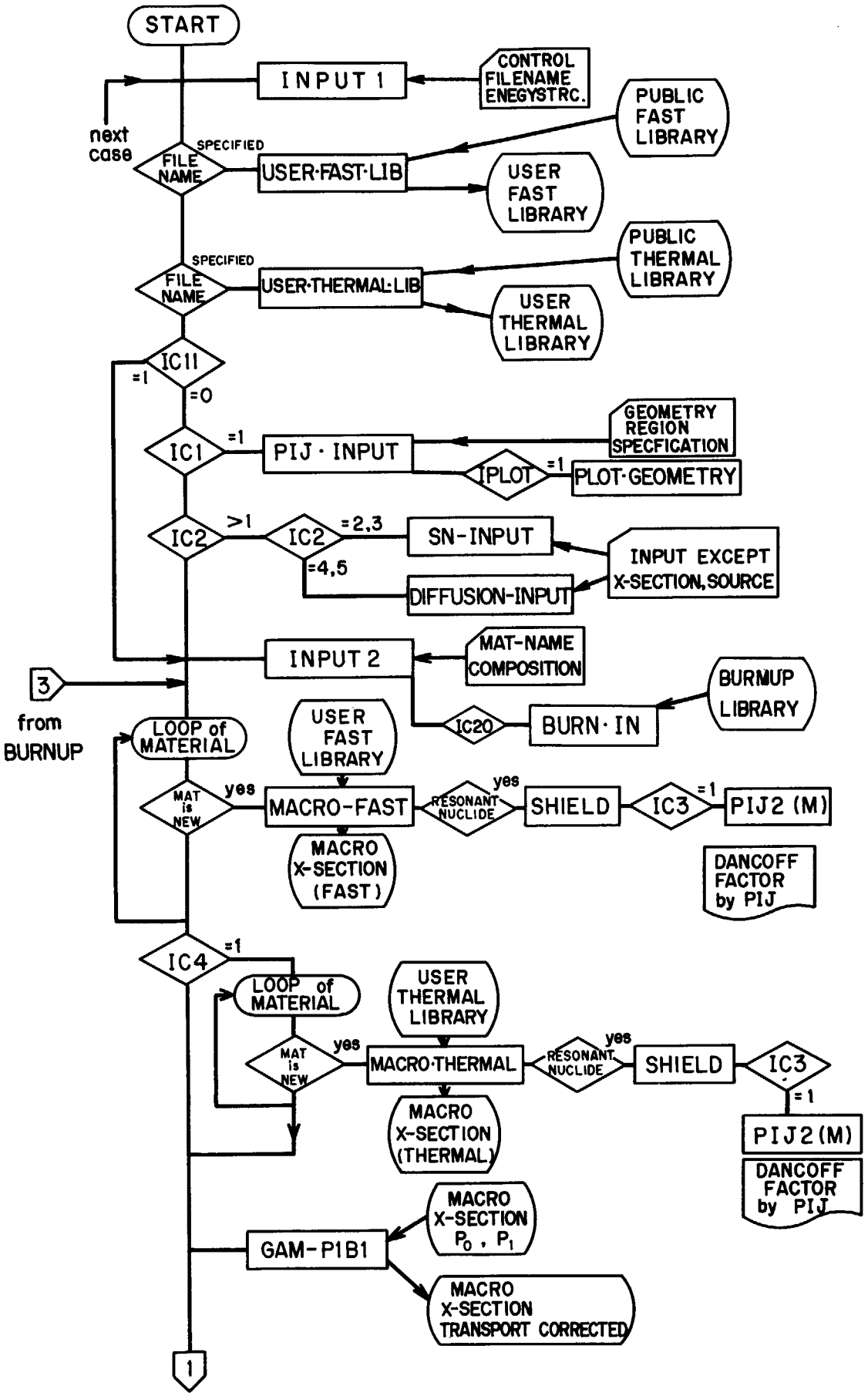


Fig.IV.3-1 Flow diagram of SRAC

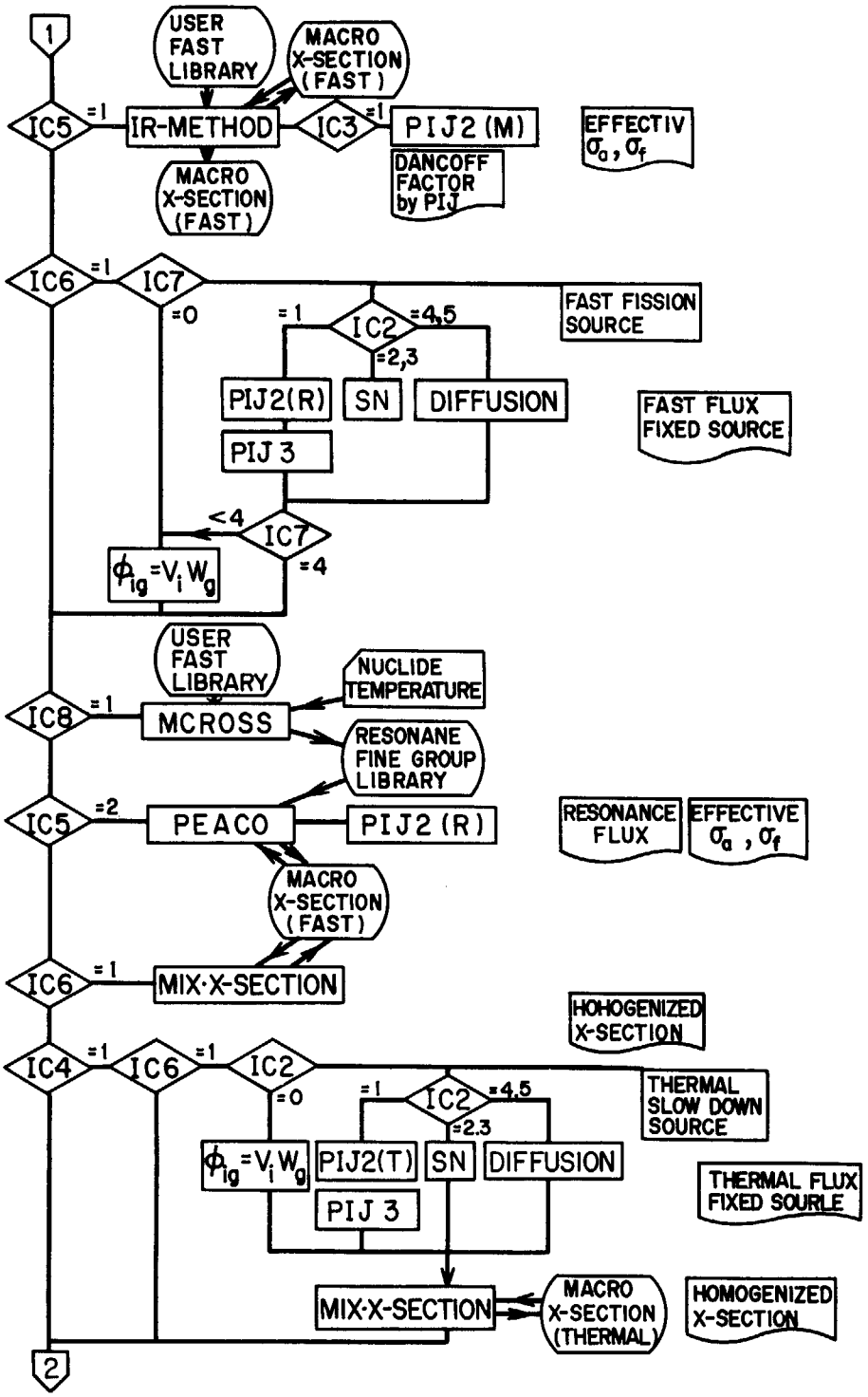


Fig.IV.3-1 (Cont'd)

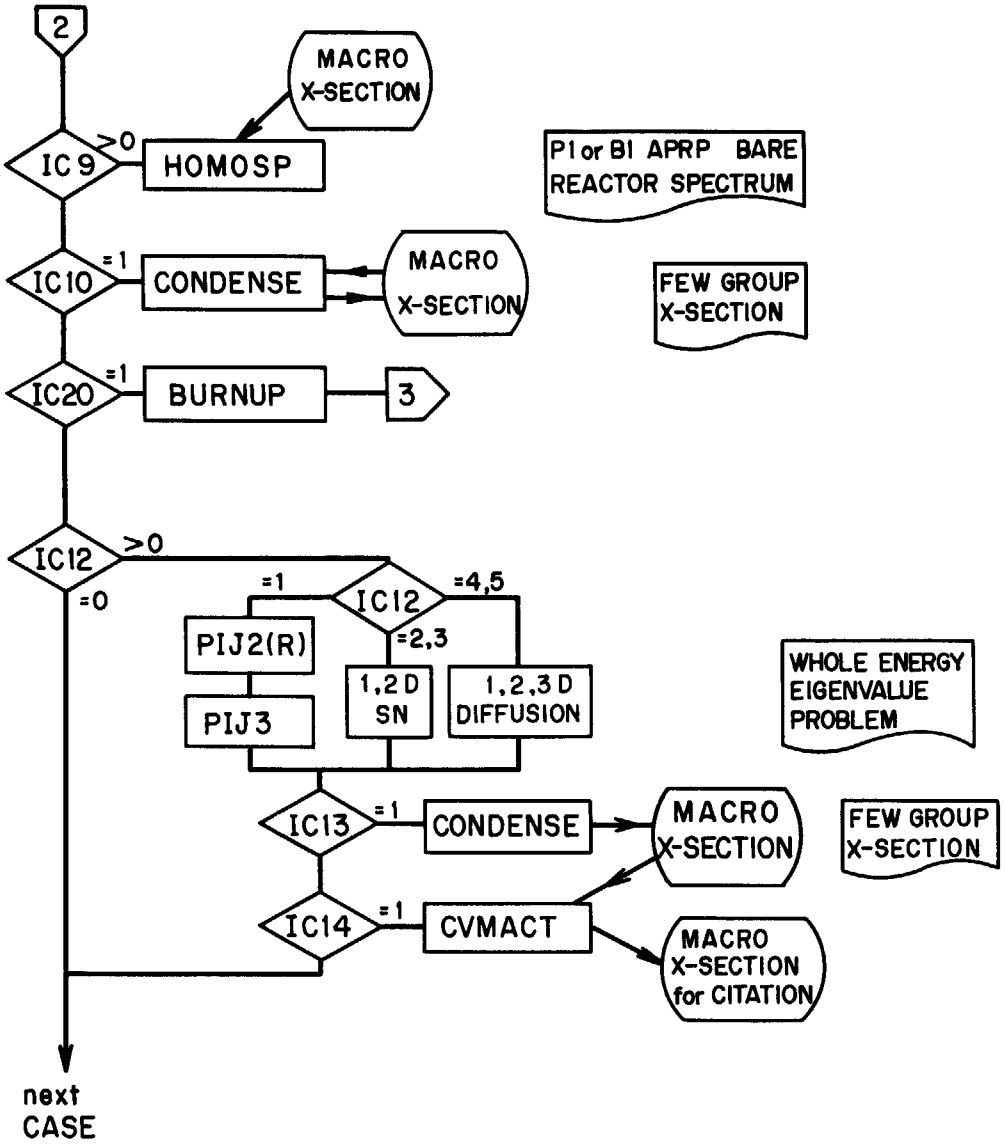


Fig.IV.3-1 (Cont'd)

IV.4 Overlay structure

The overlay structure which is currently applied is shown below. The built-in routines and the labelled common blocks in LEVEL 0 are ommited from listing

```
LEVEL 0
            MAIN,SRAC,CLEA,SPLINE,STEPCOM
            READ,PDSERR,RWPDSF,MSGPR,SEARCH,OPNBUF,OVRWRT (PDS file)
            REAG,PACK,PACKX (Built-in free format routines)
            *SEPTE,*TITLEP
            *PCOWK2,*AA,*TMPSET
 LEVEL1
            INPUT1,INPUT2

 LEVEL1     (User microscopic libraries)
            COLLAP,*USERIX

  LEVEL2    USERFL,UFLCAL,UFLCON,*UFLCNT

  LEVEL2    USERTL

 LEVEL1     (Macroscopic cross section in thermal energy range)
            MACROT,MACRTR

 LEVEL1     (Collision probability preparation)
            PIJIN,PIJ1,CHECK,ELIM,INSERT,*PIJC,*PIJ1C

  LEVEL2    GEOMTY,INUM,ORGSET,GTITLE,CIRCLG,CNTPLY,CIRCLH

    LEVEL3    GEOM01,GEOM02,GEOM08,GEOM09,GEOM13

    LEVEL3    GEOM04,GEOM05,GEOM06,GEOM07,CIRCLS

    LEVEL3    GEOM10,GEOM11,CIRCLP,NUMCYL,NUMPIN,CHKIN,SRTCYL

  LEVEL2    PATH,PAIN,PREPA,MAKEPT,CYL,HEX,HEX2,SQ,SQ2
            COMPAR,DIVIDE,SLAB,HHHH,PATHHH,PREHH,SARCOS

  LEVEL2    CLUP,CLIN,GEOM,MAKETC

  LEVEL2    CLUP77,RDX1,GEOM7,MAKET,INSET7
            LOCF,IPRINT,PREX7

  LEVEL2    CLUPH,CLINH,MAKETH,SECT,INTRP,GEOMH,*HTGR

  LEVEL2    PATHXY,PREXY,GEOMXY,MAKETX,IPRTX

 LEVEL1     (Collision probability calculation)
            PIJ2,SIGRD,DELT,FORM,PAINT,ONE,TWO,ENX,FKIN,PINTH,*ABC

  LEVEL2    (Macroscopic cross section in fast energy range)
            MACROF,*MAFWRK,*MAFCNL,*MAFCNT,*MAFSX1,*MAFSX2

   LEVEL3   MAFDAT,MAFSFX

   LEVEL3   MAFCAL,MAFCON
```

```
    LEVEL4   MAFLTM,MAFOUT,MAFPRT,MAFTTL

    LEVEL4   MAFSIG

  LEVEL2     (IRA resonance integral)
             IRA,IRACON,*IRACNL,*IRADAT,*IRASX1,*IRACNT,*IRAWRK

    LEVEL3   IRACMP,IRASFX

    LEVEL3   IRACAL,*IRAPRM,*IRAPSE

      LEVEL4 IRASIG

      LEVEL4 IRARSP

      LEVEL4 IRAPRT,IRAOUT

      LEVEL4 IRASET,IRAMIX

  LEVEL2     (Fine resonance integral)
             PEACO,PCOCON,*PCOWK1,*PCOWK3,*PCOSF1,*PCOSF2,*PCOSF3

    LEVEL3   PCOSFX

    LEVEL3   PCOPRE,PCOIN1,PCOIN2,PCODAT

    LEVEL3   PCOAVG,PCOFIN

      LEVEL4 PCOPIJ

      LEVEL4 PCOINT

    LEVEL3   PCOOUT

    LEVEL3   PCOMCR

    LEVEL3   PCOPLT

LEVEL1       (Solve linear eqs. by collision probabilities)
             PIJ3

  LEVEL2     INP2F,INP3F,TEDIT

  LEVEL2     ITER,RELAX,MATINV

LEVEL1       (Resonance cross section file)
             MCROSS,MCRCAL,MCRNBR
             *MCRDI1,*MCRDI2,*MCRTKN,*MCRMAT,*MCRBLK
             *MCRSUB,*MCRDBL,*MCRBRE,*MCRCAR,*MCRELH

  LEVEL2     MCRBRD

  LEVEL2     MCRPLT

  LEVEL2     MCRINT

  LEVEL2     MCRNTL
```

```
 LEVEL2     MCRRSP

 LEVEL2     MCRSX

LEVEL1      (TUD preparation)
            TUD1

LEVEL1      (TUD execution)
            TUD2,VINT,FINT

 LEVEL2     INPT2,INPT3,INPT4,OUTPT2

 LEVEL2     ITUD,PROD,RELAXT

LEVEL1      (ANISN)
            WOT,ERRO,ADDR,ITIME,CLEAR,WOT8,ANISN1,SN1AR
            S814B,S804,CVMASN,FSPVEL

 LEVEL2     ANISN2,CONTRL

  LEVEL3    PLSNT,TP,ADJNT,S805,S814,S966

  LEVEL3    GUTS,S807,S810,S821,S824,DT,S833,S851,CELL

  LEVEL3    FINPR,FINPR1,PUNSH,DTFPUN,FLTFX

   LEVEL4   BT,SUMARY,AFACTOR

LEVEL1      (Effective microscopic cross section)
            MICREM,MICREF,MOVEVT,*MICRC

LEVEL1      (One point reactor spectrum)
            HOMOSP

LEVEL1      (Write R-region flux into FLUX file)
            FWRITE

LEVEL1      (Fixed thermal source)
            TSOURC

LEVEL1      (Few group macroscopic cross sections)
            CONDEN,CONDEM

LEVEL1      (Mix cross sections by X-region)
            MIXX

LEVEL1      (Fixed fission source)
            FSOURC

LEVEL1      (Concatenate fast and thermal macroscopic cross sections)
            CONCAT

LEVEL1      (Collision cross sections for PIJ2)
            SIGT

LEVEL1      (Modify transport cross sections)
            GAM,P1B1
```

```
LEVEL1      (Cross section for CITATION)
            CVMACT

LEVEL1      (CITATION preparation)
            CIT1,CIT2,GRIT,RQED,CALR,CTFLUX

LEVEL1      (CITATION execution)
            CIT1,CIT2,GRIT,RQED,CALR,PDSWRT
            *COOPD,*CMARY,*ASRCH,*AKADD,*AVDLM
            *ABURN,*AFLUX,*AMESH,*ALSUB

 LEVEL2     RSET

 LEVEL2     IPTM

  LEVEL3    *SLSUB,*SKADD,OPT1,GETC,GETE
            CSTC,CRDR,UPDT,RONE,RALL,COPY,WALL
            WART,RAEN,SNSN,KRST

  LEVEL3    SETV,CNTR,HIST,GEOMC,LVMX,MESH,COMP,CMOT,KOMP,KMOT
            OVER,MACR,SSET,KXNX,KSIG,TAPE

  LEVEL3    CLAS,DENS,BKLE,FXSO,BEER,SRCH,RODI,DCAY,YELD,CHAN,IPRT
            TAPX,CNIO,BNSB,CPNC,DISK,SIZE,RSTR,TRAN,SHOX,WIO3,DYPD
            IMXS,MYSH,RODO

  LEVEL3    GEDT

 LEVEL2     EIGN,BIGS,XSET,EXTR,CYCR,GINS,ITED,UDTE,SSZU

  LEVEL3    WFCC,WFAC,WNSS,RNSS,HOWE,INFX,KNFX,RODX

  LEVEL3    FLUX,DNSD,ABPR,LOOP,FINS

   LEVEL4   CNST

   LEVEL4   BEGN,RDUE

   LEVEL4   FWRD,FXRD

   LEVEL4   DPER

   LEVEL4   HWRD,HXRD

   LEVEL4   WFLX

   LEVEL4   FTRI

  LEVEL3    KLUX

   LEVEL4   KNST

   LEVEL4   KTRI

  LEVEL3    NMBL,WSTR,DISH,DIRT,FASP,DASH,KASH,DODA,KRAN,CRSH,MASH

 LEVEL2     OUTC,POUT,KOUT
```

```
  LEVEL3    PDWT,KDWT,HEAT,PTAB,KTAB,KUDN,DTOR,NUDN,EDIN,TABL,RERT
            CMXS

  LEVEL3    PERT

 LEVEL2     TSCL,BURN,CYED

 LEVEL2     *FMSUB,*FMADD,*FMIMS,MEDT
            DCAX,I120
            SADD,MBED,INCO,EQTS,I2T4

LEVEL1      (Burn-up)
            AA,BB,BURN11

 LEVEL2     (Burn-up preparation)
 LEVEL2     BURNIN,BURN03,BURN04,BURN05,BURN06,BURN07,BURN08,BURN09
            BURN10

LEVEL1      (Burn-up execution)
            BURNUP,BURN01,BURN02,ERRBRN,BURN12

LEVEL1      (TWOTRAN)
            ERROR,CLEARW,WWRITE,REED,RITE,DATE1,ECRD,ECWR,SECOND
            LOAD,WRFQ

 LEVEL2     FSOURW (Fast neutron source)

 LEVEL2     TSOURW (Thermal neutron source)

 LEVEL2     TWTRN1,AUXTWT (Twotran preparation)

  LEVEL3    SN1TR

  LEVEL3    INPT14,SNCON,IFINSN,PNGEN

  LEVEL3    INPT15,CSMESH,MAPPER

 LEVEL2     (Twotran execution)
            TWTRN2,MONITR,MPLY,ECHECK,DUMPER,PCMBAL

  LEVEL3    INPUTW

   LEVEL4   INPT11,DUMPRD

    LEVEL5  CVMASW,FSPVEW

   LEVEL4   INPT12,CSPREP,S966W

   LEVEL4   INPT13,READQF

  LEVEL3    GRIND2,REBAL

   LEVEL4   GRID21,INITAL,INITQ,FISCAL

   LEVEL4   GRID22,OUTER,INNER,IN,OUT,FIXUP,SETBC,STORAF,SAVEAF
            GSUMS

   LEVEL4   GRID23,TESTS,NEWPAR
```

```
   LEVEL3     OUTPT3

    LEVEL4    OUTT31,FINAL,PDSFLX

    LEVEL4    OUTT32,EDCALL,GENFLO,EDITOR,EDMAP

    LEVEL4    IFOUT,IFRITE

NEWLVL1(REGION)
              ENTAPR
NEWLVL1(Plotter routines)
              CIRCL,SIMBOL,SYMB4,POLY,AXIS,NUMBER,SCALE,YSERCH
              CHAINP,GETCHA,FLPLOT,FSCALE,GPLOTI,GPLOTZ,ISERCH
              LSCALE,PLTCL
              PLOT,PLOTS,NEWPEN,FACTOR,WHERE
              GCHAR2,GCLS,GLINE,GOPN,GPCLS
              GPOPN,GSCHAR,NFACTR,NNEWPN,NPLOT,NWHERE
              NOPAGE,XDTTOX,YDTTOY,IXTODT,IYTODT

  *** Name marked by '*'denotes labelled common block ***
```

IV.5 Common Data

We classify the common data used in the SRAC into following category;

Rank A Common data throughout the system
Rank B Common data between the particular routine and the main
Rank C Common data within the particular routine.

We shall describe the contents of common data in Rank A and B.

Rank A MAINC /2000/
****

1	IOPT /20/	Control integers read in Sect.II.1
21 - 36		Reserved area for file names in Data Pool system (4*16H)
37	IUPSCAT	Control if the upscattering is assumed as selfscatter.
38	IBSPEC	Control whether if the P1 or B1 approx. spectrum is used to collapse the energy structure.
39 - 52		not used
53	NEFL	Number of energy groups in the public fast library (= 0 if the public file is not used)
54	NETL	Number of energy groups in the public thermal library (= 0 if the public file is not used)
55	NEF	Number of energy groups in the user fast library
56	NET	Number of energy groups in the user thermal library
57	NERF	Number of energy groups in the condensed fast group structure (= 0 if CONDENSE routine is not used)
58	NERT	Number of energy groups in the condensed thermal group structure (= 0 if CONDENSE routine is not used)
59	NMAT	Number of mixtures specified in Sect.II.8
60	NTL1	Group number in the public thermal library corresponds to the highest user's group below the thermal cut off
61	BSQ	Buckling in cm-2 read in BLOCK 4 in Sect.II.1
62	NIN1	Card input device unit number (set to 5)
63	NIN2	not used
64	NOUT1	Logical device number for message print (set to 6)
65	NOUT2	Logical device number for edit print (set to 99)
66	ITO	Initial clock in second used to check elapsed time

67 NEFL1 The lowest group number of the fast fission energy range in the public fast library (=10 ; above 930.14 keV)

68 NEFL2 The lowest group number of the smooth energy range in the public fast library (=22 ; above 40.87 keV)

69 NEFL3 The lowest group number of the resonance I energy range in the public fast library (=45 ; above 130.07eV)

70 NEF1 The user's group number corresponding to NEFL1-th group of the public group structure

71 NEF2 The user's group number corresponding to NEFL2-th group of the public group structure

72 NEF3 The user's group number corresponding to NEFL3-th group of the public group structure

73 ISTP Indicator of process step

74 NSOUC Logical device number for reading fixed source

75 NFIN Logical device number for reading flux guess

76 NFOUT Logical device number for writing flux

77 ITYPE Indicator if inhomogeneous or eigenvalue problem

78 IMCEF Indicator whether if any effective cross sections are required

79 IBNSTP Integer to indicate burn-up step used if IOPT(20).NE.0 ; set 0 during fresh composition.

80 MEMFST The first location in the common WORK allowed to use until the last location MEMORY (96-th word in MAINC)

81 LCNEGF The starting address of the sub-array NEGF in the array AA of length 880 (= 1 ; always)

82 LCNEGT The starting address of the sub-array NEGT in the array AA (= LCNEGF+NEF)

83 LCNECF The starting address of the sub-array NECF in the array AA (= LCNEGT+NET)

84 LCNECT The starting address of the sub-array NECT in the array AA (= LCNECF+NERF)

85 LCMTNM The starting address of the sub-array MTNAME in the array AA (= LCNECT+NERT)

86 LCNISO The starting address of the sub-array NISO in the array AA (= LCMTNM+2*NMAT)

87 LCTEMP The starting address of the sub-array TEMP in the array AA (= LCNISO+NMAT)

88 LCXL — The starting address of the sub-array XL in the array AA (= LCTEMP+NMAT)

89 LCXCDC — The starting address of the sub-array DC in the array AA (= LCXL+NMAT)

90 LCLISO — The starting address of the sub-array LISO in the array AA (= LCXCDC+NMAT)

91 LCIDNT — The starting address of the sub-array IDNT in the array AA (= LCLISO+NMAT)

92 LCDN — The starting address of the sub-array DN in the array AA (= LCIDNT+2*NTISO) where NTISO is total number of nuclides appearing in the mixture specification Sect. II.8.

93 LCIRES — The starting address of the sub-array IRES in the array AA (= LCDN+NTISO)

94 LCIXMC — The starting address of the sub-array IMCR in the array AA (= LCDN+NTISO)

95 NTOT — Logical device number for macroscopic collision cross sections used in Pij production (set to 4)

96 MEMORY — Maximum length of the common WORK ; normally set to 60,000, which is commonly used as work area in transport or diffusion codes. The user has to recompile the MAIN routine after chainging the common length so as to use more core storage.

97 IPLOT — Indicater for opening the files relating to plotter to avoid dupulicative opening (0/1:no/yes)

98 IRANG — Indicator in which energy range the present step is;
=0 in fast
=1 in thermal
=2 in whole energy range

99 ICF — Indicator in which energy structure the present step is;
='0000' processing in condensed energy structure
='0002' processing in fine energy structure

100 INITL — Case counter
=0 in first case of job step
>0 in secondary case

101 CASEID — Case identification (8H)

103 TITLE — Case description (72H)

121 AA /1880/ — The array which includes sub-arrays ; NEGF, NEGT, NECF, NECT, MTNM, NISO, TEMP, XL, DC, LISO, IDNT, DN, IRES, and IMCR ; as described below

NEGF /NEF/ — Number of public fast groups in each user's fine group

NEGT /NET/ Number of public thermal groups in each user's fine group

NECF /NERF/ Number of user's fine groups in each condensed fast group

NECT /NERT/ Number of user's fine groups in each condensed thermal group

MTNM /NMAT/ Mixture identification (8H) of each mixture

NISO /NMAT/ Number of nuclides to compose the mixture

TEMP /NMAT/ Phisical temperature of the mixture

XL /NMAT/ Mean chord length of the mixture (input)

DC /NMAT/ Dancoff correction factor of the mixture (input only)

LISO /NMAT/ Relative location of the first nuclide of the mixture on the nuclide vector

IDNT /NTISO/Nuclide identification (8H) of the nuclide (input)

DN /NTISO/Atomic number density of the nuclide (changed by burn-up)

IRES /NTISO/Resonance indicator of the nuclide (input)

IMCR /NTISO/Edit indicator for effective microscopic cross sections

Rank B PIJ2C /1000/

****

The common data between input step and executing step of collision probability method routine.

1 IGT Geometry type

2 NZ Total number of sub-regions

3 NR Total number of T-regions

4 NRR Total number of R-regions

5 NXR Total number of X-regions

6 IBOUND Outer boundary condition of lattice cell

7 IDRECT Indicator to compute directional collision probabilities
=1 isotropic only
=2 anisotropic also

8 LCOUNT Total number of neutron paths stored on path table files

9 IEDPIJ Edit control for calculated collision probabilities
=0 no
=1 print

10 IFORM Indicator of definition of collision probabilities (internal use)

11 NTTAB Maximun length of path table for a unit cell

12 NUTAB Maximum length of for concatenated path table

13 SZ Outer surface area of unit cell

14 ITYPE Problem type (internal use)
=0 eigenvalue
=1 fixed fource

15 NGLAST The last group number of collision probabilities which were calculated in the previous run and are read from the unit of logical device number 20. This item is for restart purpose.

16–18 for local use

19 IEDFLX Edit control for calculated neutron flux and reaction rate

20–31 Iteration parameters read in BLOCK 2 of Sect.II.3 or defaulted values

32 ICOOD Coordinate of lattice cell used in directional diffusion coefficients
=0 plane
=1 cylinder (one or two dimensional)
=2 shere

33 NMP Number of mixtures used in collision probability method

34–39 not used

40 LCMMR The address of the sub-array MMR in the array PAA
(= LCMATD + NMP)

41 LCNREG The address of the sub-array NREG in the array PAA
(= 1 always)

42 LCIRR The address of the sub-array IRR in the array PAA
(=LCNREG + NZ)

43 LCIXR The address of the sub-array IXR in the array PAA
(=LCNREG + NR)

44 LCMAR The address of the sub-array MAR in the array PAA
(=LCIXR + NRR)

45 LCMAT The address of the sub-array MAT in the array PAA
(=LCMAR + NRR)

46 LCVOL The address of the sub-array VOL in the array PAA
(=LCMAT + NR)

47 LCVOLR The address of the sub-array VOLR in the array PAA
(=LCMAT + NR)

48 LCVOLX The address of the sub-array VOLX in the array PAA (=LCVOLR + NRR)

49 LCVOLM The address of the sub-array VOLM in the array PAA (=LCVOLX + NXR)

50 LCMATD The address of the sub-array MATD in the array PAA (=LCVOLM + NMP)

51 PAA /950/ The array which includes the sub-arrays ; NREG, IRR, IXR, MAR, MAT, VOL, VOLR, VOLX, MATD, and MMR ; as described below;

NREG /NZ/ T-region number by sub-region

IRR /NR/ R-region number by T-region

IXR /NRR/ X-region number by R-region

MAR /NRR/ Material code number by R-region

MAT /NR/ M-region number by T-region

VOL /NR/ Volume of T-region

VOLR /NRR/ Volume of R-region

VOLX /NXR/ Volume of X-region

VOLM /NMP/ Volume of M-region

MATD /NMP/ Material code number by M-region

MMR /NRR/ M-region number by R-region

Rank B SN1C /1000/

****

The common data between input and executing step of ANISN

1 D(1) dummy

2 LIM1 available data locations

(Array address)

3 LR R(IM+1) radii

4 LW W(MM) Sn weights

5 LDSN DSN(MM) Sn cosines

6 LMA MA(IM) zone numbers by interval

7 LMZ MZ(IZM) material numbers by zone

8 LMB MB(MS) mixing number in mixing table

9 LMC MC(MS) component number in mixing table

10	LXMD	XMD(MS)	number density in mixing table
11	LFIX	(36)	integer parameters
12	LFLT	(14)	floating parameters
13	LJ5	J5(IZM)	order of scatter by zone
14	LRM	RM(IZM)	radius modifier by zone
15	LDF	DF(IM*IDFM)	densisy factors
16	LJ3	J3(ID3)	material numbers for activities
17	LJ4	J4(ID3)	position for activities
18	LIGT	IGT(IGM)	Sn/diffusion/homogeneous cell indicators, only if IDAT2.NE.0
19	LART	ART(IGM)	albedo-right boundary, if IBR=3 only
20	LALFT	ALFT(IGM)	albedo-left boundary, if IBL=3 only
21	LCNXRA	FGP(IZM)	X-region number by zone
22	LFF		dummy
23	LEND		last address
24	LV	V(IM)	volumes
25	LAA	AA(IM+1)	areas
26	LWD	WD(MM)	weight*cosine
27	LMR	MR(MM)	reflective direction indices
28	LPNC	PNC(MM,IT)	Pl coefficients, if ISCT.NE.0 only

(Integer parameters (see Sect.II.4))

29	ID	Problem ID number.
30	ITH	=0 forward solution =1 adjoint solution
31	ISCT	Maximum order of scatter found in any zone
32	ISN	Order of angular quadrature
33	IGE	Geometry
34	IBL	Left boundary condition
35	IBR	Right boundary condition
36	IZM	Number of zones or regions

37	IM	Number of mesh intervals
38	IEVT	Eigenvalue type
39	IGM	Number of energy group
40	IHT	Position of total cross section in cross section table
41	IHS	Position of self-scatter cross section in cross section table
42	IHM	Length of cross section table
43	MS	Cross section mixing table length
44	MCR	Number of cross section sets to be read from cards
45	MTP	Number of cross section sets to be read from tape
46	MT	Total number of cross section sets
47	IDFM	Control for density factors
48	IPVT	K0 value
49	IQM	Indicator for distributed source
50	IPM	
51	IPP	Interval number which contains shell source
52	IIM	Inner iteration maximum
53	ID1	Print control
54	ID2	
55	ID3	Number of activity computed by zone
56	ID4	Number of activity by interval
57	ICM	Outer iteration maxmum
58	IDAT1	Indicator for data storage
59	IDAT2	Indicator for diffusion solution
60	IFG	Edit control for P1 cross sections
61	IFLU	Indicator for negative flux
62	IFN	Control for initial guess
63	IPRT	Print control for cross sections
64	IXTR	Indicator for P1 scattering constants

(Floating point parameters)

65 EV Initial guess for eigen value

66 EVM Eigenvalue modifier

67 EPS Epsilon - accuracy desired

68 BF Buckling factor, normally 1.420892

69 DY Cylinder or plane height for buckling correction (may include extrapolation length)

70 DZ Plane depth for buckling cirrection

71 DFM1 Transverse dimension for void streaming correction

72 XNF Normalization factor

73 PV =0.0, or = K0 according to IPVT=0, or =1

74 RYF Relaxation factor (suggested value = 0.5)

75 XLAL Point flux convergence criterion

76 XLAH Upper limit for |1.0-lambda 1| used in linear search

77 EQL =0.0 (defaulted)

78 XNPM =0.0 (defaulted)

(Additional parameters)

79 NMPA Number of materials excluding anisotropic blocks

80 LCNXRA Array address of X-regions by zone

81 NXRA Total number of X-regions

82 LCMACA Material number by M-region

83 LCVLMA Array address of volumes of M-regions

(Miscellaneous)

91 NIN Input device number

92 NOU Print device number

93 NT1 Device number for flux and current

94 NT2 Device number for flux and current of the previous iteration

95 NT3 Device number for cross section and source

96 NT4 scratch unit for normalization of source and cross section mixing

97	NT5	Initial time
98	NT6	Device number of library (not used)
99	NT7	Device number for specially group independent cross sections
100	NT8	Device number for weighted cross sections
101-1000		Dummy vector which contains the arrays

Rank B TWC1 /2000/
****

The common data between input step and executing step of TWOTRAN. We shall introduce only the data exchanged between TWOTRAN and external routines.

4	IGM	Number of groups
5	IM	Number of coarse meshes in X-direction
6	JM	Number of coarse meshes in Y-direction
15	MT	Total number of materials
18	IHT	Position in table of total cross section
19	IHS	Position in table of self-scatter cross section
20	IHM	Length of cross section table
21	IQOPT	Source input option
22	IQAN	Distributed source anisotropic order
25	IPVT	Parametric eigenvalue or Keff indicator
34	IGEOM	Geometry type
58	IMJM	Product of IM * JM
66	IT	Total number of radial fine-mesh intervals
67	JT	Total number of axial fine-mesh intervals
68	ITJT	Product of IT * JT
157	NOUT	Print device logical number <99>
170	NMPW	Number of materials not counting anisotropic block
181	LZRNUM	Address of array for M-region number by coarse mesh
204	LFGP	Address of array for X-region number by coarse mesh
205	LZRDUC	Address of array for Material code number by M-region

206 LDC Address of array for Material code number by coarse mesh

207 LVOLMA Address of VOLM : volumes of M-regions

209 IVMESH Device number of work file to keep volumes of coarse mesh IM*JM <3>

210 IZMESH Device number of work file to keep material numbers of coarse mesh IM*JM <2>

211 NXRW Number of X-regions

Rank B TUD1C /550/

****

The common data between input step and executing step of TUD : one dimensional diffusion calculation routine.

1 NR Number of regions

2 NMP Number of materials used in TUD calculation

3 NG Number of energy groups

4 NGS Number of energy groups having fixed source

5 NGK Number of energy groups having fission source

6 NNMAX Total number of mesh-intervals

7 IG Geometry type (0/1/2;slab/cylinder/sphere)

8 IBOUND Boundary condition

9 IGUESS Initial flux indicator

10 ID Select of diffusion coefficients

11 ITMAX Maximum thermal iterations per outer iteration

12 ITMOUT Maximum power iterations

13 ITBG Minimum number of iterations before extrapolation

14 LCMX Number of iterations for testing extrapolations

15 ITDM Minimum delay between extrapolations

16 IPT Monitor print indicator

17 EPSI Convergence criterion for thermal iterations

18 EPSO Convergence criterion for power iterations

19 EPSG Extrapolation criterion

20 RELCA Initial over-relaxation factor

21 OVERX Maximum over-relaxation factor

22 FACTOR Under extrapolation factor

23 XLAMD Extrapolation length in cm

24 BSQ1 Trnasverse buckling in cm**-2

25 IPTXEC Print indicator for cross sections

26 ITFLUX Print indicator for final fluxes

27 IPTS Print control for fixed source

28 IDOPT Selection of diffusion coefficients

29 NXR Number of X-regions

(Array address)

30 LCIK IK(NR) Material code numbers by region

31 LCNK NK(NR) Number of mesh intervals by region

32 LCXR XR(NR) X-region numbers by region

33 LCRX RX(NR) Outer radii by region

34 LCNN1 NN1(NNMAX1) Material code numbers by mesh interval where NNMAX1 is total mesh intervals counting doubly at region boundaries.

35 LCVOLR VOLR(NR) Volumes by region

36 LCMTM MTM(NMP) Material code number by M-region

37 LCMTR MTR(NR) M-region numbers by region

38 LCVLMT VLMT(NMP) Volumes by M-region

39-50 Not used

51-550 Dummy array which includes arrays of IK,NK,.....,VLMT

Rank B DEPLET /87/

****

The common data between the main and BURNUP step; cell burn-up calculation.

1 NEP Number of broad exposure steps (<=15)

2 NDEPZ(20) Flag to indicate depleting or not by R-region

3 PERIOD(15) Exposure steps

4 IBEND Normally =0, =1 if all exposure steps are finished

5	Dummy	Not used
6	TIMESC(16)	Exposure time in second corresponding to input exposure steps PERIOD
7	TIMEMW(16)	Exposure steps in MWD
8	TIMEU5(16)	Exposure steps in relative U-235 burnt
9	POWERL	Power level given in input

Rank B CIT1C /3008/

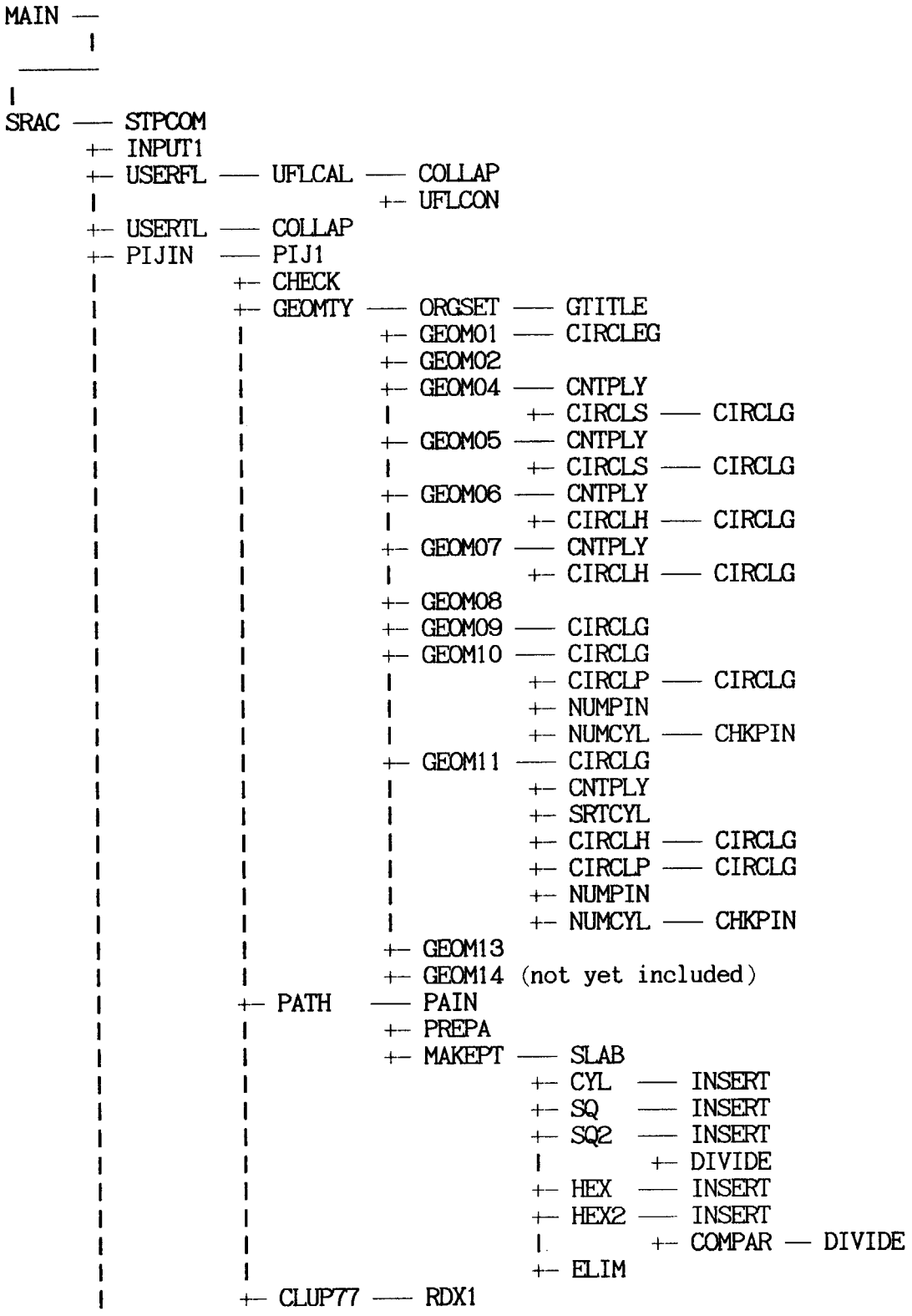
****

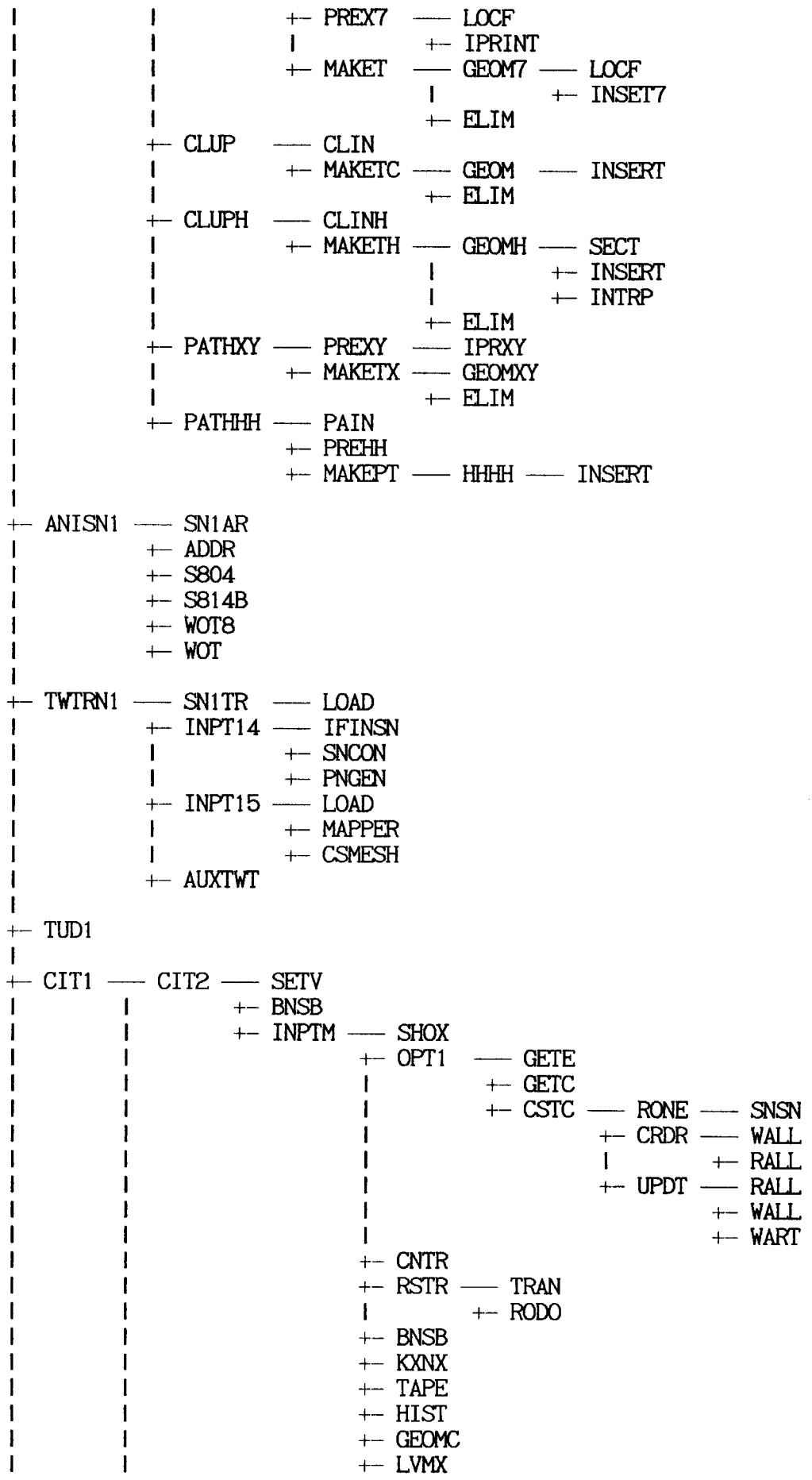
The common data between input step and executing step of CITATION: the multi-dimensional diffusion.

1	NM	Number of materials used in the CITATION
2	NXR	Number of regions for cross section edit
3	ID	Option for diffusion coefficients
4	IRN	Number of regions (NREGI*NREGJ*NREGK)
5	LCNM	Array adress of material position number in the mixture specifications (Sect.II.8) , length NM.
6	LCNXR	Array adress of X-region numbers by region , length IRN
7	LCMAC	Array adress of material position number by region , length IRN, if NXR>0.
8	LCVOL	Array adress of volumes by region , length IRN
9	IC(3000)	Dummy vector which contains the above four arrays.

IV.6 Subroutines

We shall show tree structure how to call subroutines in the SRAC. In the following figure FACOM built-in functions, frequently appearing I/O routines, their associated error print routines, and clock routines are omitted.

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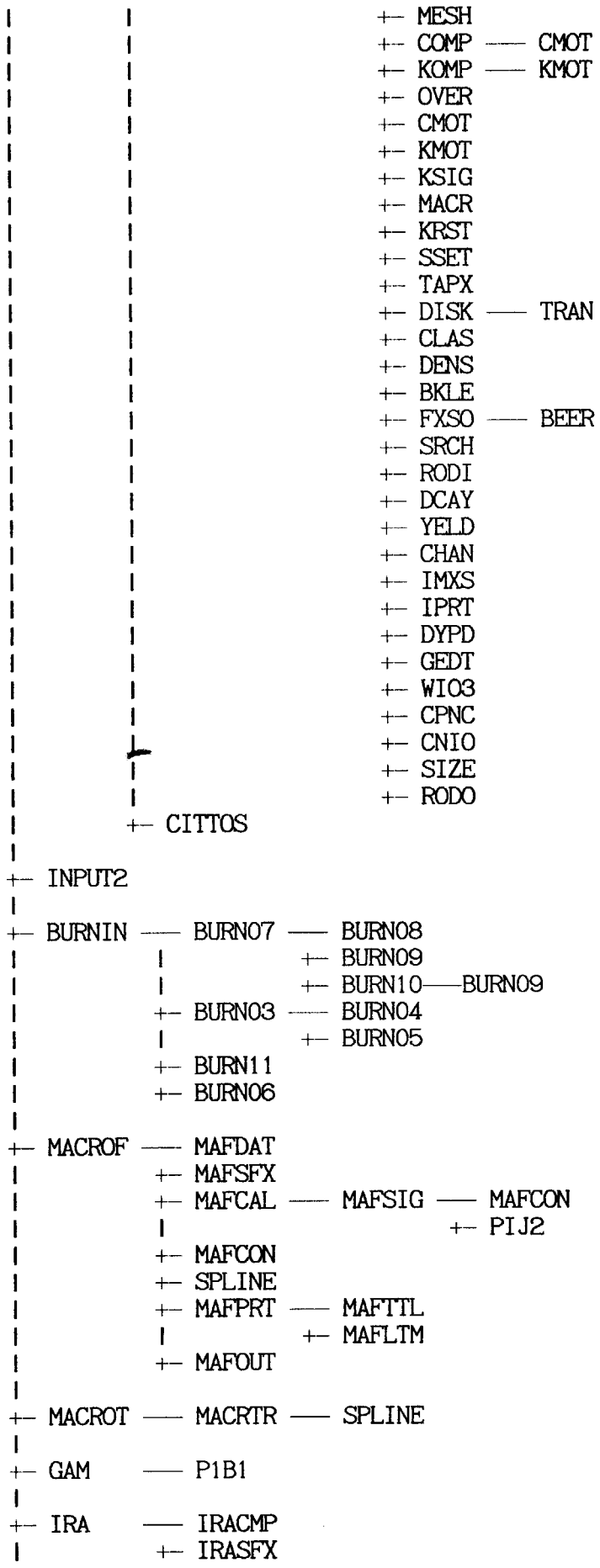
IPRINT
MAKET — GEOM7 — LOCF
INSET7
ELIM
CLUP — CLIN
MAKETC — GEOM — INSERT
ELIM
CLUPH — CLINH
MAKETH — GEOMH — SECT
INSERT
INTRP
ELIM
PATHXY — PREXY — IPRXY
MAKETX — GEOMXY
ELIM
PATHHH — PAIN
PREHH
MAKEPT — HHHH — INSERT
ANISN1 — SN1AR
ADDR
S804
S814B
WOT8
WOT
TWTRN1 — SN1TR — LOAD
INPT14 — IFINSN
SNCON
PNGEN
INPT15 — LOAD
MAPPER
CSMESH
AUXTWT
TUD1
CIT1 — CIT2 — SETV
BNSB
INPTM — SHOX
OPT1 — GETE
GETC
CSTC — RONE — SNSN
CRDR — WALL
RALL
UPDT — RALL
WALL
WART
CNTR
RSTR — TRAN
RODO
BNSB
KXNX
TAPE
HIST
GEOMC
LVMX


MESH
COMP — CMOT
KOMP — KMOT
OVER
CMOT
KMOT
KSIG
MACR
KRST
SSET
TAPX
DISK — TRAN
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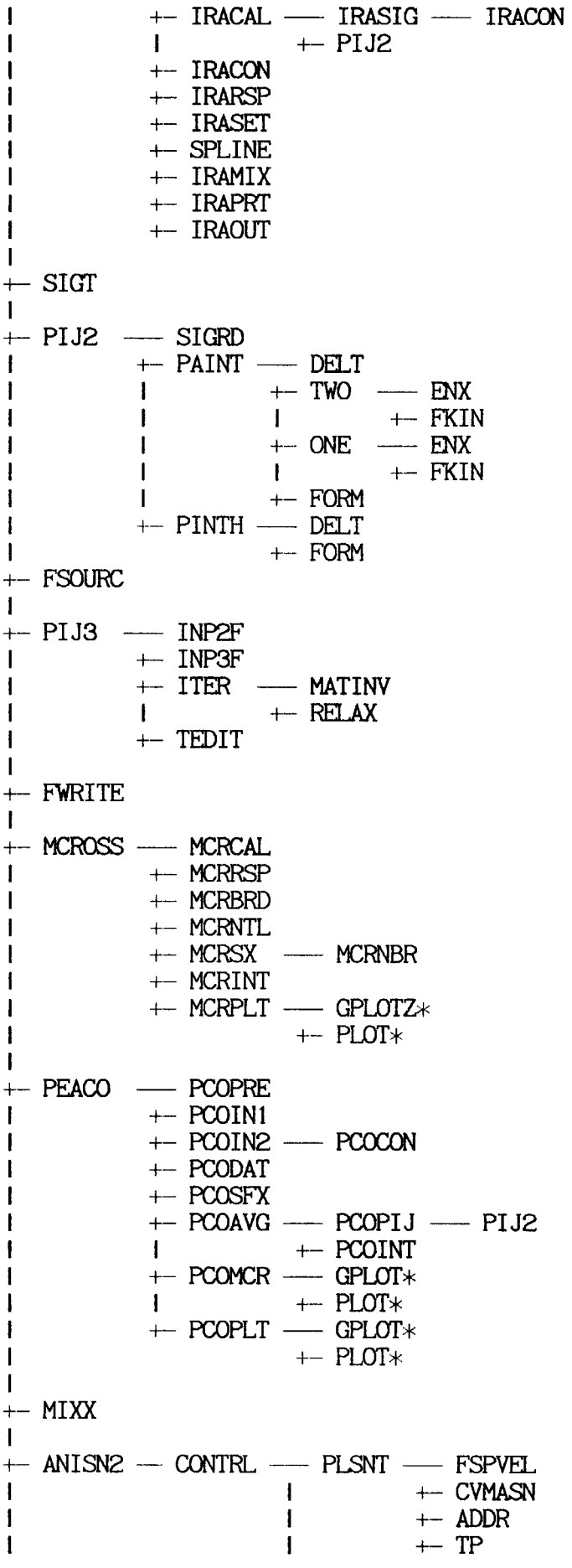
DENS
BKLE
FXSO — BEER
SRCH
RODI
DCAY
YELD
CHAN
IMXS
IPRT
DYPD
GEDT
WIO3
CPNC
CNIO
SIZE
RODO
CITTOS
INPUT2
BURNIN — BURN07 — BURN08
BURN09
BURN10—BURN09
BURN03 — BURN04
BURN05
BURN11
BURN06
MACROF — MAFDAT
MAFSFX
MAFCAL — MAFSIG — MAFCON
PIJ2
MAFCON
SPLINE
MAFPRT — MAFTTL
MAFLTM
MAFOUT
MACROT — MACRTR — SPLINE
GAM — P1B1
IRA — IRACMP
IRASFX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+- PIJ2
+- IRACON
+- IRARSP
+- IRASET
+- SPLINE
+- IRAMIX
+- IRAPRT
+- IRAOUT
+- SIGT
+- PIJ2 -- SIGRD
+- PAINT -- DELT
+- TWO -- ENX
+- FKIN
+- ONE -- ENX
+- FKIN
+- FORM
+- PINTH -- DELT
+- FORM
+- FSOURC
+- PIJ3 -- INP2F
+- INP3F
+- ITER -- MATINV
+- RELAX
+- TEDIT
+- FWRITE
+- MCROSS -- MCRCAL
+- MCRRSP
+- MCRBRD
+- MCRNTL
+- MCRSX -- MCRNBR
+- MCRINT
+- MCRPLT -- GPLOTZ*
+- PLOT*
+- PEACO -- PCOPRE
+- PCOIN1
+- PCOIN2 -- PCOCON
+- PCODAT
+- PCOSFX
+- PCOAVG -- PCOPIJ -- PIJ2
+- PCOINT
+- PCOMCR -- GPLOT*
+- PLOT*
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+- PLOT*
+- MIXX
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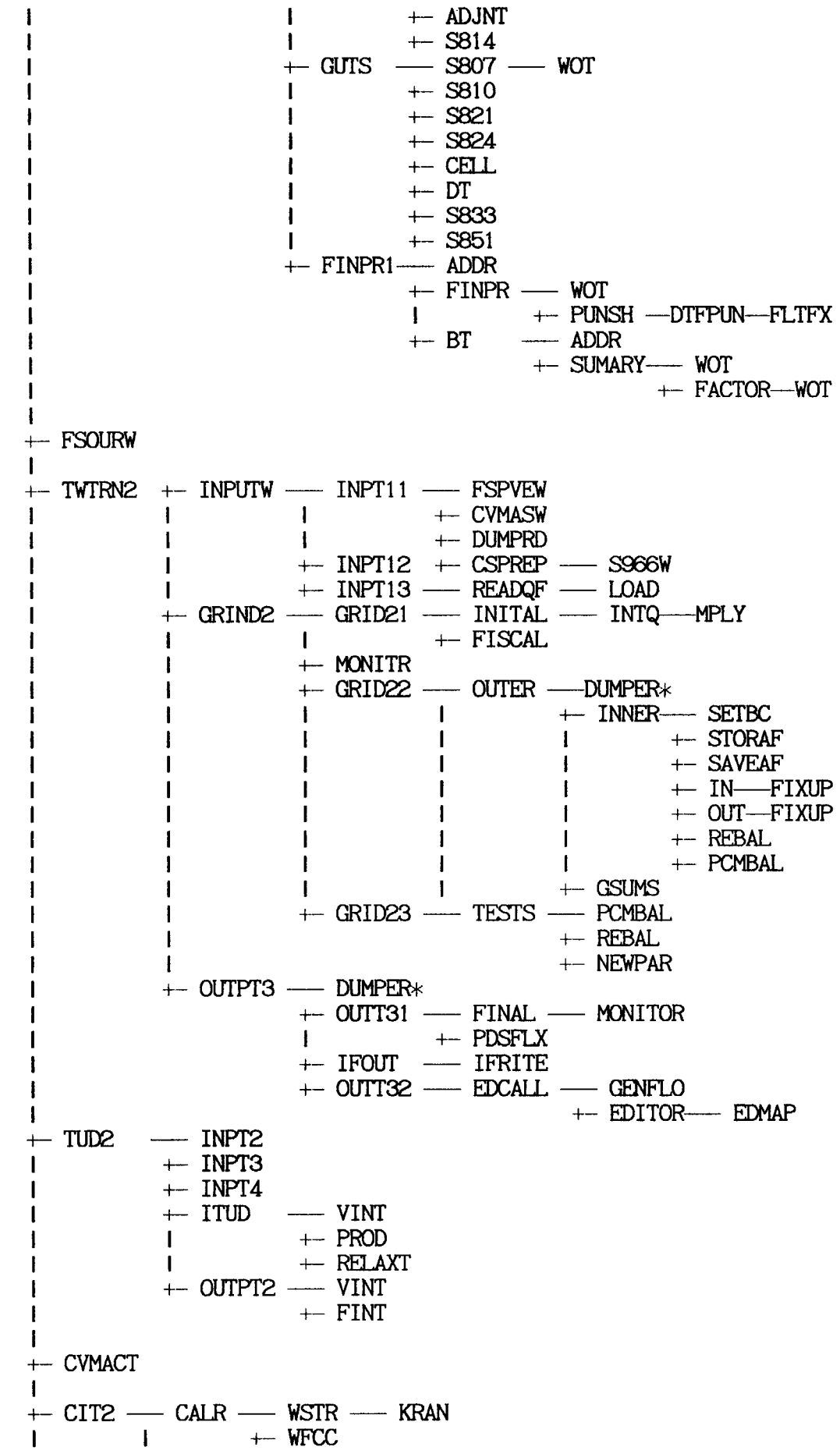
+- CVMASN
+- ADDR
+- TP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WOT
PUNSH
DTFPUN
FLTFX
BT
ADDR
SUMARY
WOT
FACTOR
WOT
FSOURW
TWTRN2
INPUTW
INPT11
FSPVEW
CVMASW
DUMPRD
INPT12
CSPREP
S966W
INPT13
READQF
LOAD
GRIND2
GRID21
INITAL
INTQ
MPLY
FISCAL
MONITR
GRID22
OUTER
DUMPER*
INNER
SETBC
STORAF
SAVEAF
IN
FIXUP
OUT
FIXUP
REBAL
PCMBAL
GSUMS
GRID23
TESTS
PCMBAL
REBAL
NEWPAR
OUTPT3
DUMPER*
OUTT31
FINAL
MONITOR
PDSFLX
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EDITOR
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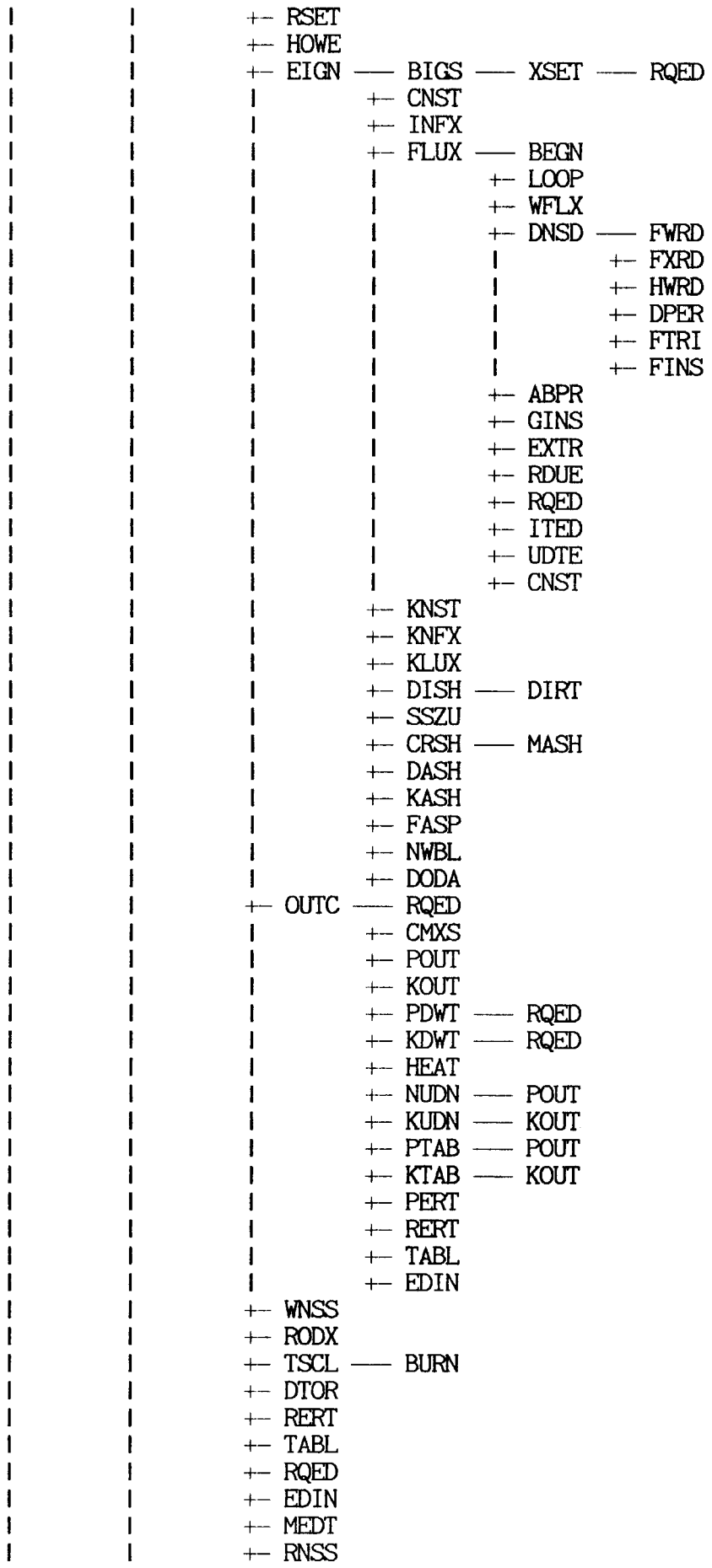
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DPER
FTRI
FINS
ABPR
GINS
EXTR
RDUE
RQED
ITED
UDTE
CNST
KNST
KNFX
KLUX
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KASH
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DODA
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RQED
CMXS
POUT
KOUT
PDWT
RQED
KDWT
RQED
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TABL
EDIN
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TSCL
BURN
DTOR
RERT
TABL
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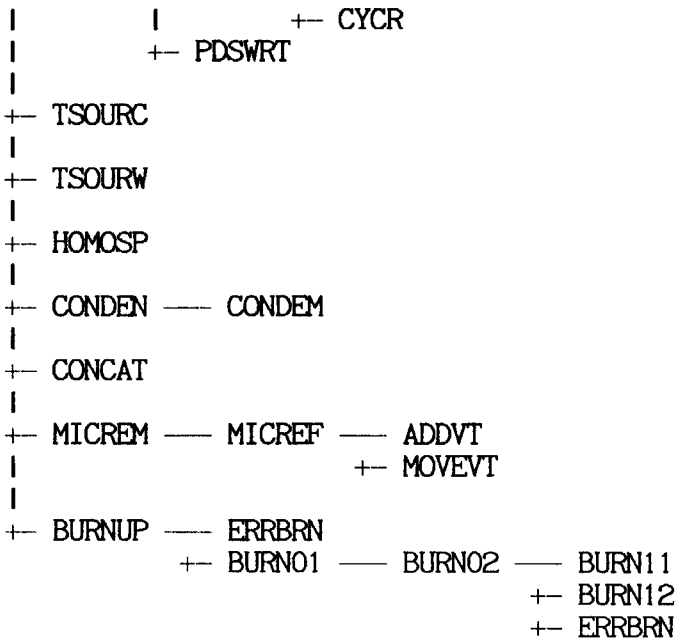
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+- PDSWRT
+- TSOURC
+- TSOURW
+- HOMOSP
+- CONDEN —— CONDEM
+- CONCAT
+- MICREM —— MICREF —— ADDVT
+- MOVEVT
+- BURNUP —— ERRBRN
+- BURN01 —— BURN02 —— BURN11
+- BURN12
+- ERRBRN

V. Structure of I/O Files

We shall describe the structure of I/O files and their physical contents which will be helpful to prepare the input and utilize the output.

The most of data libraries and output files are stored in PDS files . For the user who are not familiar with PDS file we shall mention about it. The PDS ,we concern, is a Partitioned Data Set (DS organization PO with undefined record format, and maximum block-size of the device) charaterized by storing a data array by a member name in binary mode. To manage the file by Fortran program we use a subroutine RWPDSF written by assembler language. The subroutine permits us to switch the data set under process and also indicate the member. The processes available are to inform the array length, read, write, delete, and rename a member. This routine is used not only in the SRAC but also in the auxiliary programs which allows us to manage the file by our macroscopic TSS commands PDSEDT, and PDSEDGRP together with built-in TSS commands such as LISTD, DELETE, RENAME like to manage a PO file of EBCDIC mode.

Another advantage of using the PDS file arises from the control of member name of eight characters. We will see in the following description that the member name is composed of some characters to denote case ID, mixture ID, or nuclide ID, and other characters to denote physical contents such as reaction, temperature, spatial index, burn-up step, ... etc..

In the following description for the member name, for example, 'Member Czzm0000', capital letters and digits denote fixed characters for the member name and lowercase letters denote variable characters.

V.1 Fast Neutron Microscopic Cross Section File (PDS)

The microscopic group cross sections, the tables for self-shielding factors, and resonance level parameters are stored in a PDS file which will be used by DD name of FASTP for the public (basic) or FASTU for the user library. Both of them have the same organization as described below. The public library keeps the information for all available nuclei in the 74 group structure, and the user library does that for the selected nuclei in the user's group structure.

The following eight kinds of members are stored in the library.

MEMBER NAME	CONTENTS
'FASTLIB '	The control information for the library
'FISSYILD'	The pseudo fission neutron yield
'Czzm0000'	The control information for the nuclide zzm
'Mzzm0000'	The principal data for the nuclide zzm
'Fzzm0000'	The self-shielding factor table for the nuclide zzm
'Rzzm0000'	The control information for resonance parameters of the nuclide zzm
'Pzzm000l'	The resonance parameters for the l-state of the nuclide
'Bzzm000r'	The background cross sections for the reaction r of the nuclide zzm. r = F for fission r = C for capture

r = E for elastic

****

Member FASTLIB

****

The member keeps the information about the energy group structure in a vector.

NGF,NGF1,NGF2,NGF3,(Wg,g=1,NGF),(Eg,g=1, NGF+1)

NGF The total number of energy groups.
= 74 (Ec = 0.414 eV) in the public library

NGF1,NGF2,NGF3 The lowest group number in each energy range of the fast (1 MeV), smooth (50 keV), resonance I (130.07 eV), respectively. The NGF-th group must be the lowest in the resonance II range.

Wg The weighted lethargy widths which will be used in collapsing the energy group structure.

Eg The boundary enrgies in eV.

Member FISSYILD

****

The member keeps the fission neutron yield of U-235 to provide the fixed source in calculating the fast neutron spectrum for non fissile material.

Xg,g=1,NGF The fission neutron yields in the g-th group normalized as sum of Xg = 1

Member CzzmCONT

****

zzm (3H) The nuclide identification composed of chemical symbol of the nuclide and the last digit of the mass number. The available combination of zzm is listed in Dictionary VII.3.

The member keeps the control information of the nuclide in a vector of fixed length of 41.

ICAP = 0 no capture
= 1 capture cross sections stored

IFISS = 0 no fission
= 1 fission cross sections stored

IRES = 0 no resonance parameter
= 1 resonance parameters stored

LTOT The vector length of the member Mzzm0000 described below.

LTHi,i=1,4 The partial vector length which contains the i-th scattering matrix.

LAi,i=1,4 The lowest group number of the energy range where the i-th scattering occurs.

LDi,i=1,4 The number of energy groups to which the slowing-down occurs in the i-th scattering.

IFS Index for shielding factor tables
= 0 no shilding factor
= 1 shilding factors for any reaction are tabulated

IFTR,IFC,IFF,IFE, and IFER
Indices for self-shielding factor tables for partial reaction of transport, capture, fission, elastic, and elastic slowing-down, respectively.

NGMIN and NGMAX
The highest and lowest group number for the self-shielding factor tabulation.

NSIG The number of admixture cross sections which are used as one of the arguments for interpolation of self-shielding factor.

NTEMP The number of temperatures which are used as the other argument for interpolation of self-shielding factor.

AMASS,SIGP, and SIGCO
The properties of the nuclide; atomic mass in amu, potential scattering cross section, and the 2200 m value of capture cross section.

TEMPi,i=1,NTEMP
The temperatures for tabulation.

SIGi,i=1,NSIG
The admixture cross sections for tabulation.

Member Mzzm0000
****
The member keeps the principal neutron cross sections.

CAPTi,i=1,NGF
The capture cross sections if ICAP=1.

FISSi,i=1,NGF
The fission cross sections if IFISS=1.

FNUi,i=1,NGF
The ν fission neutron yield per fission if IFISS=1.

FSPCi,i=1,NGF
The fission neutron spectrum if IFISS=1.

TRi,i=1,NGF
The transport cross sections

WEIGHTi,i=1,NGF

The lethargy widths weighted by fission neutron spectrum

ELASi, i=1, NGF
The total elastic cross sections

N-Ni, i=1, LTH(1)
The inelastic scattering matrix of the length LTH(1) = (LD(1)+1)*LA(1), ordered as,

$\sigma_{1\to1}, \sigma_{1\to2}, \ldots, \sigma_{1\to1+LD(1)}$,

$\sigma_{2\to2}, \sigma_{2\to3}, \ldots, \sigma_{2\to2+LD(1)}$,

.....

$\sigma_{g\to g}, \sigma_{g\to g+1}, \ldots, \sigma_{g\to g+LD(1)}$,

.....

N2Ni, i=1, LTH(2)
The N2N scattering matrix of the length LTH(2) = (LD(2)+1)*LA(2).

ELP0i, i=1, LTH(3)
The elastic scattering matrix of the length LTH(3) = (LD(3)+1)*LA(3).

ELP1i, i=1, LTH(1)
The elastic scattering matrix of the length LTH(4) = (LD(4)+1)*LA(4).

Member Fzzm0000 if IFS=1
****
The member keeps the self-shielding factor tables of the length NSIG*NTEMP*(NGMAX-NGMIN+1)*(IFTR+IFC+IFF+IFE+IFER+1).

Member Rzzm0000 if IRES=1
****
The member keeps the control information for resonance level parameters of the length 6.

NLS	Number of neutron orbital angular momenta
SPI	The nuclear spin
AP	The scattering radius in unit of 10 **(-12) cm
AWR	The ratio of the mass of the nuclide to that of a neutron
EL	The lower limit for a energy range
EH	The upper limit for a energy range

Member Pzzm0001 if IRES=1
****

The member keeps the control information for a given l -value, repeatedfor l =0,NLS-1.

L The value of l

NRS The number of resolved resonances for a given l -value.

The 15 resolved resonance parameters per a level so as to express the multi-level formula; repeated NRS times

ERj	resonance energy (eV)
AJj	statistical factor (2J+1)/2/(I+1)
GTj	total width (eV)
GNj	neutron width (eV)
GGj	gamma width (eV)
GFj	fission width (eV)
SIGZj	peak value of total cross section $\Sigma_0=2.6*10^6*AJ_J\Gamma_N/(\Gamma ER_J)$
SIGZPj	peak value of scattering cross section $\Sigma_{0p}=(\sigma_0\sigma_pAJ_J\Gamma_N/\Gamma)^{1/2}$
BETAj	$\beta_\infty=(1.0+(\sigma_0\Gamma_N/\sigma_p\Gamma))^{1/2}$
ETAj	$\eta_\infty=\sigma_0/\sigma_p$
RIIj	$I_\infty=\pi\sigma_0\Gamma_\gamma/2ER_j$
UTj	U-value for total
VTj	V-value for total
UFj	U-value for fission
VFj	V-value for fission

for j=1,NRS, and σp is potential scattering.

Member BzzmOOOr if IRES=1

****

The member keeps the background cross sections for a given reaction

NR The number of energy ranges that have been given. A different interpolation scheme may be given for each range.

NP The total number of energy points used to specify the data.

NBTi,INTi,i=1,NR
The interpolation schemes

Ei,σi,i=1,NP
The background cross sections

V.2 Thermal Neutron Microscopic Cross section File (PDS)

The microscopic group cross sections, the tables for self-shielding factors in the thermal neutron range are stored in a PDS with DD name of THERMALP for those of public (basic) and THERMALU for those of user libraries. Both have the same organization as described below. The public library keeps the information for all available nuclei in the 48 group structure, and the user library does that for selected nuclei in the user's group structure. A file stores in a PDS file seven kinds of membersas listed below.

MEMBER NAME CONTENTS

****

```
'THERMALt'  The control information of the library                    *
'CzzmcOOO'  A control member for the nuclide zzm                      *
'KzzmcOOt'  P0  matrix with capture, fission, and total vectors       *
'MzzmcOOt'  P0  matrix after transport correction with capture,       *
            fission, and total vectors (alternate of matrix K)        *
'PzzmcOOt'  P1  matrix (given for moderating nuclide only)            *
'NzzmcOOt'  P0  matrix after second order expanded transport          *
            correction (alternative combination with matrix P         *
            instead of that of K and P)                               *
'FzzmcOOt'  Shielding factor tabulation (for nuclide which has        *
            any sharp resonance in thermal range)                     *
**********************************************************************
```

where 'zz' denotes element chemical symbol as described in Sect.II.2, and 'm' for last digit of mass number to discriminate isotopes, 't' for the temperature index as shown in Dictionary VII.6, and 'c' for chemical compound state as shown in Dictionary VII.4. Matrices K, M, P, and F are organized to have the same length. The contents of each member will be described in Sect.V.2.

We shall describe the physical contents of a member.

Member THERMALt
****

NGT — The number of energy groups

WTg, — The integrated asymptotic neutron spectrum in the group which will be used as weights for collapsing the energy group structure. The spectrum is prepared to form the Maxwellian of neutron temperature Tn (=Tm+50) and 1/E above the cut off of 5*kTm, given for g=1,NGT.

Eg, — The boundary energies, for g=1,NGT+1

Member CzzmcOOO length 30
****

INT(1)
- = 0 non fissile
- = 1 fissile

INT(2)
- = 0 keep M-matrix in which scattering matrix is filled by zero values
- = 1 keep M-matrix
- = 2 keep K-matrix
- = 3 keep K and P-matrices
- = 4 keep N and P-matrices

INT(3)
- = 0 no self-shielding factor tabulation
- = 1 self-shielding factor tabulation

INT(4) — The highest group which has the self-shielding factor

INT(5) — The lowest group which has the self-shielding factor

INT(6) — The number of admixture for the self-shielding factor tabulation

INT(7) The number of temperatures for which the cross sections given

INT(8),INT(9),INT(10) not used

DMi,i=1,10 not used

SIG0i,i=1,8 The admixture cross sections for the tabulation

XNU ν-value in thermal range

DM20 not used

Member MzzmcOOt of length NGT*(NGT+4)
****

σs,g-g' The scattering cross sections from group g to g', for g'=1,NGT, and g=1,NGT

σup,g The up-scattering cross sections for g=1,NGT

σc,g The capture cross sections for g=1,NGT

σt,g The total cross sections for g=1,NGT

σf,g The fission cross sections for g=1,NGT

Members KzzmcOOt, NzzmcOOt, and PzzmcOOt have the same structure as the member MzzmcOOt, except that the P-matrix keeps zero values of capture and fission cross sections.

Member FzzmcOOt
****

FTCPi,i=NGMIN,NGMAX
The self-shielding factors for capture cross sections

FTFSi,i=NGMIN,NGMAX
The self-shielding factors for fission cross sections

FTTRi,i=NGMIN,NGMAX
The self-shielding factors for total cross sections

V.3 Resonance Neutron Microscopic Cross Section File (PDS)

The microscopic ultra-fine group cross sections are stored in PDS file which is used by the DD name of MCROSSP for the resonance integral calculations.

The following two kinds of members are stored in the library.

MEMBER NAME	CONTENTS
'CzzmOOOt'	The control information for the nuclide zzm of temperature tagged t
'FzzmrOOt'	The fine resonance cross sections for the nuclide zzm of reaction r of temperature tagged t

r = F for fission
r = C for capture
r = E for elastic

****

Member CzzmOOt
****

The member keeps the control information for the nuclide zzm of temperature tagged t.

IA	not used
NOMESH	The number of broad group
NOIG	The number of fine group
NFI	The number of ultra-fine group in a fine group
NFII	Maximum number of ultra-fine group
IFISS	= 0 non-fissile = 1 fissile
MVOGO	S-S interference effect option
NMS	The number of S-wave resonances
NMP	The number of P-wave resonances
TEMP	The nuclide temperature in Kelvin
AM	The ratio of atomic mass of the nuclide to that of a neutron
EEUP	The upper limit of the energy range where cross sections are given
EELW	The lower limit of the energy range where cross sections are given
UIGP	The lethargy width of a fine group

Member FzzmrOOt
****

The member keeps the cross sections for the nuclide zzm of the reaction r of temperature on the ultra-fine energy group.

σri	for i=1,NOIG*NFI

V.4 Macroscopic Cross Section Files (PDS)

The macroscopic cross sections are stored in PDS files with DD name of MACROWRK for those of fine group and MACRO for those of few group energy structure. Both may be assigned to a same Data Set. The separation is effective if the user needs to keep only the few group cross

sections on the catalogued file. The both have the same organization as described below.

Member CONTe00p (8H)
****

e (1H) Index for energy range where the cross sections are defined
=F Fast energy range
=T Thermal energy range
=A Whole energy range

p (1H) Index for Legendre component and energy group structure
=0 Coarse group
=2 Fine group

This member keeps the information about the energy group structure in a vector of length 2*(NG+1), as follows;

NG Number of energy groups

Wg,g=1,NG Weighted lethargy widths which may be utilized in collapsing the group structure further.

Eg,g=1,NG+1 Energy boundaries starting at the highest energy

Member nameebnp (8H)
****

name (4H) Mixture identification or case identification

e (1H) Index for energy range ; =F, =T, or =A.

b (1H) Index for the burn-up step
=0 for initial composition
=N for N-th burn-up step; as 1,2,3....., 9,A,B,...,I, J

n (1H) X-region index coupled with case identification to identify the homogenized cross sections.
=0 with the mixture identification
=1 for the cell averaged cross sections (number of X-region = 1) with the case identification
=N for N-th X-region with the case identification

p (1H) Index for Legendre component and energy group structure
=0 P0 component and coarse group
=2 P0 component(after transport correction) and fine group
=3 P1 component and fine group
=4 P0 component(consistent) and fine group

The member keeps a complete set of group cross sections in a one-dimensional array. We shall show below a partial vector corresponding to a energy group.

1 LSSg Position of the self-scatter on the scattering vector

2 LGTg Length of the scattering vector

3 $\Sigma_{act,g}$ Activation cross section

4 $\Sigma f,g$ Fission cross section

5 $\nu\Sigma f,g$ ν*fission cross section

6 $\Sigma t,g$ Total cross section

7 Xg Fission neutron yield

8 D1g Diffusion coefficient 1

9 D2g Diffusion coefficient 2

10 $\Sigma a,g$ Absorption cross section followed by the scattering vector

1 $\Sigma_{g\to g1}$

2 $\Sigma_{g\to g1+1}$

– where g1 = g – LSSg + 1
– g2 = g + LGVg – LSSg

LSS_g $\Sigma_{g\to g}$

LSS_g+1 $\Sigma_{g\to g+1}$

–
–

LGV_g $\Sigma_{g\to g2}$

The above organization is repeated NG times in a vector. The vector length amounts to 10*NG + sum of LGVg.

V.5 Neutron Flux File (PDS)

The neutron fluxes integrated spatially in 'R-region' (collision probability method), in 'zone' (ANISN), in 'coarse mesh zone' (TWOTRAN) in 'region' (TUD), or in 'zone' (CITATION) and those integrated in X- region by X-region are stored in a PDS file. For plotting and homogenization purpose the volumes of each spatial region are written.

The following three kinds of members are stored in the file.

```
MEMBER NAME   CONTENTS
**************************************************************************
'caseeb0p'    The neutron fluxes by R-regione by group
'caseebnp'    The neutron fluxes of n-th X-region by group
'caseeVOL'    The volumes of R-regions
**************************************************************************

Member  caseeb0p  /Number of R-regions * number of groups/
****************
```

case(4H) The case identification

e (1H) The tag for neutron energy range
=F Fast energy range
=T Thermal energy range
=A Whole energy range

b (1H) The burn-up step indicator
=0 no burn-up problem or initial step
=N burn-up step as 0,1,2,....,9,A,B,C,...,etc.

p (1H) The tag for energy group structure
=0 coarse group
=2 fine group

PHIi,g The neutron fluxes multiplied by volume i and lethargy width g for i=1,NRR,g=1,NG

Member caseebnp /Number of groups/
****

case(4H) The case identification

e (1H) The tag for neutron energy range
=F Fast energy range
=T Thermal energy range
=A Whole energy range

b (1H) The burn-up step indicator
=0 no burn-up problem or initial step
=N burn-up step as 0,1,2,....,9,A,B,C,...,etc.

p (1H) The tag for energy group structure
=0 coarse group
=2 fine group

n (1H) X-region index
=1 for the cell averaged cross sections (number of X-region = 1)
=N for N-th X-region

PHIg The neutron fluxes of n-th X-region multiplied by volume of the X-region and lethargy width g.

Member caseeVOL /Number of R-region/
****

case(4H) The case identification

e (1H) The tag for neutron energy range
=F Fast energy range
=T Thermal energy range
=A Whole energy range

Vi The volumes of R-regions.

V.6 I/O Files for ANISN (PS)

DD Name	Variable Name	Remarks

FT01F001	NT1	Flux and current storage unit
FT02F001	NT2	Flux and current storage unit in the previous iteration
FT03F001	NT3	Scratch unit for cross section and fixed source storage (not used for IDAT1 is set to zero)
FT04F001	NT4	Scratch unit for normalization of source
FT05F001	NT5	Initiatl flux unit
FT10F001	NT6	Interface file for macroscopic cross sections
FT08F001	NT8	Weighted cross section unit
FT32F001	NSOUC	Interface file for fixed source
FT99F001	NOUT2	Print unit

V.7 I/O Files for TWOTRAN (PS)

DD Name	Variable Name	Remarks
FT01F001	NEDIT	Edit input storage
FT02F001	IXMESH	X-region output storage to SRAC
FT14F001	IVMESH	Zone output storage to SRAC
FT04F001	NEXTRA	Scratch unit
FT05F001	NINP	Extra input in TWTRN2 routine
FT06F001	NOUT	System print
FT08F001	NAFLUX	Angular flux by group
FT09F001	LAFLUX	Output form of angular flux
FT10F001	ISOTXS	Input for X-sections fed by SRAC main
FT11F001	ISNCON	Output for SN constants
FT12F001	NDUMP1	First restart dump unit
FT13F001	NDUMP2	Second restrart dump unit
FT32F001	IFIXSR	Input of inhomogeneous source fed by SRAC main
FT33F001	ITFLUX	Output form of total flux

V.8 I/O Files for CITATION (PS)

The current version CITATION compiled in SRAC does not work for cross section mixing, nor density search, nor fuel management. The device list following is for the current version.

D Define Name	Variable Name	Remarks
FT01F001	I01	Scratch unit, always required.
FT02F001	I02	Scratch unit, always required.
FT03F001	I03	Scratch unit, always required.
FT04F001	I04	Storage for microscopic scattering cross sections, not required in SRAC.
FT05F001		System input unit
FT06F001		System print only for message.
FT07F001		
FT08F001	IOSIG	Microscopic cross section library, not used.
FT09F001	IOFLX	Used to store forward neutron flux map by option. See NGC6 section 001. Also used to store forward and adjoint fluxes for use in

	perturbation calculation. Required if NGC6>0, or if the adjoint problem is specified.
FT10F001	Scratch unit, always required.
FT11F001	Scratch unit, always required.
FT12F001	Storage for zone densities, not used in SRAC.
FT13F001	Output unit for restart, required if NGC2 and/or NGC3 >0.
FT14F001	Scratch unit to store macroscopic cross sections, always required.
FT15F001	Scratch unit to store equation constants if I/O during the iterative calculation is necessary.
FT16F001	Scratch unit, always required.
FT17F001	Scratch unit used to store fluxes if a double iteration type search is being done, used to to store space-energy fixed source, and used to save point neutron source.
FT18F001	Scratch unit used in perturbation calculation
FT19F001	Scratch unit, always required.
FT21F001	Scratch unit in fuel managemaent calculation, not used in SRAC.
FT22F001	Scratch unit used in perturbation calculation
FT26F001	Scratch unit used in perturbation calculation
FT28F001	Scratch unit used in perturbation calculation
FT31F001	Macroscopic cross section input unit supplied by CVMACT format conversion routine.
FT32F001	Power density and heat to coolant, if required.
FT91F001	Scratch unit to transfer input data in EBCDIC mode from CIT1 step to CIT2 step
FT92F001	Scratch unit used in preparation of macroscopic cross section in CVMACT step.

****

V.9 Burn-up Library File (PO)

The current burn-up library file stores several optional chain schemes in separate member of a PO file ; their physical contents are explained in Dictionary VII.5.

The user who wants to use his own burn-up chain scheme has to prepare the information in a file which will be read from FT50F001 by the SRAC as the burn-up library.

We shall describe the organization of a burn-up scheme.

File attribute: Blocksize 4200 bytes, Record length 130 bytes, Record format FB, Dataset organization PO, No numbered

Record 1 format(4I4)

NCH	Number of chains
NTNUC	Total number of nuclides contained in a scheme.
NABSB	ID number of the first burnable absorber appearing in the nuclide table.
NFIRST	ID number of the first F.P. nuclide appearing in the nuclide table

Record 2 format(32I4)

NSTP(I) Number of sub-steps in I-th burn-up step; in each sub-step the flux normalization is executed to keep the given power level. The value 3 is filled for all steps. The number of burn-up step is an input item entered in II.11. As we adopt an analytic form for the solution of burn-up chain equations, the results very weakly depend on the the period through which we assume the fixed flux level. By this reason we put this item in the library.

After the above two record the file contains the following three tables;

(1) Nuclide table (N-table)
(2) Chain description table (C-table)
(3) Fission yield table (F-table)

(1) N-table /7/ items in NTNUC lines in format(I4,A4,2I4,2 E13.5)

ITBL(I,1)* Numerical ID of not more three digits for the nuclide
ITBL(I,2) Alphabetic ID commonly used in the SRAC system
ITBL(I,3) Fissile indicator (0/1;no/fissile)
IRES(I) Resonance indicator (0/2;no/resonant)
DCAY(I) The decay constants
TBL(I,4)** The power emitted per a fissin in watt. This item is to supplied for possible fissile nuclei.

Note * The nuclei have to be ordered in NTNUC lines as first the nuclei appearing in heavy nuclide chains, second the burnable nuclei, third F.P. nuclei.
Note ** EQUIVALENCE(ITBL,TBL)

(2) C-table /any/ signed integers in NCH lines in format(32I4)

Each burn-up chain is expressed by a series of signed integers of three digits (numerical nuclide ID defined in N-table). In a line, 32 integers may be entered. The first and second integers are used for control purpose, then the actual nuclide ID appears at the third integer.

Note This table is read by Fortran statemens as

```
      DO 10 I=1,NCH
   10 READ(50,20) (ICHAIN(J,I),J=1,32)
   20 FORMAT(32I4)
```

, therefore first 128 columns in each record does not allow any string such as comment.

ICHAIN(1,I): the position of the nuclide in the vector ICHAIN(*,I) to which the calculation of concentration started, normally this item is filled by '3' i.e. start at the first nuclide.
ICHAIN(2,I): the physical length of vector -2 (= the number of nuclides in the vector when none of interpreter accompanied)
ICHAIN(J,I) for J>2: a signed three digit integer to denote a nuclide

or an interpreter to the previous nuclide.

Meaning of sign
+ : capture reaction yield the next nuclide
- : decay yield the next nuclide

Meaning of absolute value
1 through NTNUC : normal chain
201 through 200+NTNUC : interlocked chain
501 through 500+NTNUC : partial capture, partial decay, or N-2N
for N-2N case the ID followed by '001', for partial reactions the ID followed by four digit integer which is a ratio to 10000 of the partial reaction.

Ex. 1
002+001+042 : Accumulation of Kr-83 in fission fragment
Ex. 2
003+003-061-064+068: I-135 decay yields Xe-135, Xe-135 decay yields Cs-135
Ex. 3
003+003-063+066+067: Xe-133 decay yields Cs-133, Cs-133 capture yields Cs-134.

(3) F-table /1/ alphabetic string and /13/floating numbers in format(A3,13E9.3); repeated for possible F.P. nuclei in the order as appear in N-table.

We restrict that the possible fissile nuclei to 13 nuclei of Th-230, Th-232, Pa233, U-233, U-234, U-235, U-236, U-238, Pu-238, Pu-238, Pu-240, Pu-241, and Pu-242.

AID (3H) The alphabetic ID of the F.P. given in N-table

FYIELDi The fission yield of the nuclei from the i-th fissile (order as appears in the above paragraph) for i=1,13.

V.10 Auxiliary files (PS)

Here we shall describe the usage of sequential files which have not yet been mentioned but appears in the table in Sect.IV.2.

****
FT03 : Scratch file, required always
FT04 : Scratch file, required always
FT05 : System input file to feed the input data
FT06 : System print file, mainly used for running message and
: input data listing
FT21 : Scratch file to store collision probabilities
FT22 : Scratch file to store the diffusion coefficients defined
: by Benoist Behrens term
FT31 : Output file to keep the macroscopic cross sections in
: CITATION format
FT32 : Scratch file to feed the fixed source distribution
: formed in FSOURC or TSOURC routine to any of transport
: routine
FT33 : Scratch file used to transfer the neutron fluxes solved

```
            : by any of the transport routine to MIXX routine
   FT50     : Input file to keep the chain scheme for burn-up calcula-
            : tion
   FT51     : Scratch file to transfer the information from the pre-
            : paration step of burn-up calculation to the execution
            : step
   FT52     : Scratch file to transfer the effective microscopic cross
            : sections from MICREF step to BURNUP step
   FT81     : Scratch file to keep the neutron traces across T-region
            : boundaries in collision probability calculation ; written
            : by the fixed logical record length of 4096 bites in
            : binary mode
   FT82     : Scratch file to keep the neutron traces across R-region
            : boundaries
   FT83     : Scratch file to keep the neutron traces across M-region
            : boundaries
   FT92     : Scratch file to prepare the file FT31
   FT99     : System print file
*****************************************************************************
```

VI Mathematical Formulations

Presented here are the equations programmed into the SRAC system.

VI.1 Group Cross Sections in the Fast Neutron Library

The multigroup transport or diffusion calculations are often based on the concept of group constants such as the ABBN(Ref.(2)) or JAERI-Fast set(Refs.(45 and 46)). The principal advantage of the multigroup constant method is that reactor calculation can be made by using the same group constant set for the various reactors with different compositions and sizes. Consequently, some processing code such as ETOX(Ref.(47)), MINX(Ref.(48)) and PROF-GROUCH-G(Ref.(15)) were developed to calculate economically and conveniently the group constants using a nuclear data file. However, the calculational method of group constants used in these codes are different. Especially, the effective cross sections in resonance energy region are calculated by using various methods.

Though the ETOX and MINX codes can take into account for both the composition and temperature dependence of group cross sections, the effective group cross sections are calculated by assuming the constancy of collision density. Moreover, the isolated narrow-resonance approximation is used for unresolved resonance region, and the mutual interference between resonances of different nuclides and the self-overlapping effects are ignored. Furthermore, the self-shielding of elastic removal cross section is assumed to be neglected.

In the TIMS-PGG code system (Ref.(76)) for group constants production, the TIMS-1 code calculates the effective cross sections by solving numerically the neutron slowing down equation using the recurrence formula (Refs.(40 , 41)) for slowing down source, in order to avoid the errors (Refs.(49, 50)) caused from the approximations used in the ETOX and MINX codes. In the unresolved resonance region for this purpose, TIMS-1 generates required resonance levels and parameters in the unresolved resonance region by using Monte Carlo method(Refs.(51, 52, 53)). TIMS-1 is a code system which is composed of some codes, ARCFIT-2(Ref.(54)), ARCFIT -3(Ref.(54)), MCROSS-2(Ref.(17)) and PEACO-II(Ref.(5)) developed for the production of the JAERI Fast set(JFS).

In this Chapter, the calculational methods used in the TIMS-PGG code system are described. Section VI.1.1 describes the expression of Doppler broadened cross sections based on the multilevel formula. The ultrafine group cross sections are calculated by using this formula. The expressions of average cross sections in the unresolved resonance region are described in Section VI.1.2. In Section VI.1.3, a ladder of resonance parameters and levels which reproduces the average cross sections are generated by a random sampling technique. Also for the unresolved region, the ultrafine group cross sections are calculated by using the ladder of resonance parameters on the basis of the formula described in Section VI.1.1. The ultrafine group neutron spectra are calculated by solving numerically the neutron slowing down equation in an infinite homogeneous system. This calculational method is described in Section VI.1.4. The group constants are calculated by weighting the ultrafine group neutron spectra as described in Section VI.1.5. In Section VI.1.6,the interpolation method for the self-shielding factors are described.

VI.1.1 Doppler broadened cross sections

The Doppler broadened pointwise cross sections for total and capture or fission are written by the following expressions (Ref. (18)) :

$$\sigma_t^J(E)=\sigma_p(E)+\frac{4\pi}{k^2}\sqrt{\alpha/2m\pi}\ g_J \sum_\lambda \{(\Gamma_{\lambda n}{}^0+\frac{1}{2}u_\lambda^t)\ \Phi_\lambda^r\}$$
$$+\frac{8\pi R}{k}\sqrt{\alpha/2m\pi}\ g_J \sum_\lambda \{(\Gamma_{\lambda n}{}^0+\frac{1}{4kR}v_\lambda{}^t)\ \Phi_\lambda^i\} \qquad (VI.1.1\text{-}1)$$

and

$$\sigma_y^J(E)=\frac{4\pi}{k^2}\sqrt{\alpha/2m\pi}g_J \sum_\lambda \{(\frac{\Gamma_{\lambda n}^0\Gamma_{\lambda n}}{\Gamma_\lambda}+\frac{1}{2}u_\lambda{}^y)\ \Phi_\lambda{}^r+\frac{1}{2}v_\lambda{}^y\Phi_\lambda{}^i\}, \qquad (VI.1.1\text{-}2)$$

where

$$\Phi_\lambda{}^r = R_e(F(\omega_1)-F(\omega_2)), \qquad (VI.1.1\text{-}3)$$

$$\Phi_\lambda{}^i = I_m(F(\omega_1)-F(\omega_2)), \qquad (VI.1.1\text{-}4)$$

$$F(\omega)=\int_0^\infty \exp(-x^2)\ \frac{\omega}{x^2+\omega^2}\ dx\ , \qquad (VI.1.1\text{-}5)$$

$$\omega_1=\sqrt{\alpha}(b+i(a-v_0)), \qquad (VI.1.1\text{-}6)$$

$$\omega_2=\sqrt{\alpha}(b+i(a+v_0)), \qquad (VI.1.1\text{-}7)$$

$$a-ib=(2E_\lambda-i\Gamma_\lambda)^{1/2}, \qquad (VI.1.1\text{-}8)$$

$$\alpha=A_m/2KT, \qquad (VI.1.1\text{-}9)$$

$$R=(1.23\times A_m^{1/3}+0.8)\times 10^{-1} \text{ in unit of } 10^{-12} \text{ cm}, \qquad (VI.1.1\text{-}10)$$

$$u_\lambda^t=\sum_{\lambda'\neq\lambda}\{\sqrt{\Gamma_{\lambda n}^0\Gamma_{\lambda' n}^0}\ G_{\lambda\lambda'}\{2(E_\lambda-E_{\lambda'})\sin 2\varphi_n+(\Gamma_\lambda-\Gamma_{\lambda'})\cos 2\varphi_n\}$$
$$/|Z_\lambda-Z_{\lambda'}|^2\}, \qquad (VI.1.1\text{-}11)$$

$$v_\lambda^t=\sum_{\lambda'\neq\lambda}\{\sqrt{\Gamma_{\lambda n}^0\Gamma_{\lambda' n}^0}\ G_{\lambda\lambda'}\{-(\Gamma_\lambda-\Gamma_{\lambda'})\sin 2\varphi_n+(E_\lambda-E_{\lambda'})\cos 2\varphi_n\}$$

$$/ | Z_\lambda - Z_{\lambda'} |^2 \}, \qquad (VI.1.1\text{-}12)$$

$$u_\lambda^y = \sum_{\lambda' \neq \lambda} \{ \sqrt{\Gamma_{\lambda n}^0 \Gamma_{\lambda' n}^0}\ G_{\lambda\lambda'} (\Gamma_\lambda - \Gamma_{\lambda'}) \ / \ | Z_\lambda - Z_{\lambda'} |^2 \}, \qquad (VI.1.1\text{-}13)$$

$$v_\lambda^y = \sum_{\lambda' \neq \lambda} \{ -2\sqrt{\Gamma_{\lambda n}^0 \Gamma_{\lambda' n}^0}\ G_{\lambda\lambda'} (E_\lambda - E_{\lambda'}) \ / \ | Z_\lambda - Z_{\lambda'} |^2 \}, \qquad (VI.1.1\text{-}14)$$

$$G_{\lambda\lambda'} = \sum_c (\Gamma_{\lambda c}^{1/2}\ \Gamma_{\lambda' c}^{1/2}), \qquad (VI.1.1\text{-}15)$$

$$Z_\lambda = (E_\lambda - E) - i\Gamma_\lambda / 2,$$

$$\Gamma_\lambda = \sum_c \Gamma_{\lambda c} = \Gamma_\gamma + \Gamma_f + \Gamma_n,$$

K = Boltzmann constant,
T = temperature (°K),
v_0 = neutron velocity,
σ_p = potential scattering cross section (barn),
A_m = atomic mass,
$\Gamma_{\lambda n}^0$ = reduced neutron width,
φ_n = hard sphere scattering phase shift,
E_λ = resonance energy,
k = neutron wave number,
and y stands for capture and fission reactions.

The resonance scattering cross section is given by

$$\sigma_n^J(E) = \sigma_t^J(E) - \sigma_p(E) \quad - \sum_y \sigma_y^J(E)$$
$$= \frac{4\pi}{k^2} \sqrt{\alpha/2m\pi}\ g_J \sum_\lambda \{ \left(\frac{\Gamma_{\lambda n}^0 \Gamma_{\lambda n}}{\Gamma_\lambda} + \frac{1}{2} u_\lambda^n \right) \Phi_\lambda^r \}$$
$$+ \frac{8\pi R}{k} \sqrt{\alpha/2m\pi}\ g_J \sum_\lambda \{ \left(\Gamma_{\lambda n}^0 + \frac{1}{4kR} v_\lambda^n \right) \Phi_\lambda^i \}, \qquad (VI.1.1\text{-}16)$$

where

$$u_\lambda^n = u_\lambda^t - \sum_y u_\lambda^y \qquad (VI.1.1\text{-}17)$$

and

$$v_\lambda^n = v_\lambda^t - \sum_y v_\lambda^y. \qquad (VI.1.1\text{-}18)$$

When $\sqrt{E_\lambda / E}$ is assumed to be nearly to unity and $(1/2)\Gamma_\lambda / E_\lambda \ll 1$, the functions Φ_λ^r and Φ_λ^i can be related to the usual Doppler line shape functions $\Psi_\lambda(\theta, x)$ and $X_\lambda(\theta, x)$, respectively. The assumption will

be sufficiently satisfied near resonance energy in relatively higher energy region.

In Eqs. (VI.1.1 -1), (VI.1.1 -2) and (VI.1.1-16), each reaction cross section is represented by the summation of a symmetric and an asymmetric functions. These expressions are the same as the single-level ones except for containing the coefficients u_λ and v_λ ,which show the interference effect between levels. If the cross sections will be well fitted by the single-level expression, the coefficients u_λ and v_λ will be nearly equal to zero. Hence, they may be considered as a kind of correction parameters for the single-level fits obtained on the base of the R-matrix theory. They will be easily obtained from the least squares method (Ref. (18)) coupled with the single-level fit.

The reaction cross sections for p- and d- wave neutrons are assumed to be respectively expressed by the single-level formula, due to their smaller contribution, as follows :

$$\sigma_y^J(E)=\frac{4\pi}{k^2}\sqrt{\alpha/2m\pi}\ \frac{(\mu_l)}{E}\ g_J\sum_\lambda \Gamma_{\lambda n l}^0\ \{E_\lambda\Phi_\lambda{}^r(\omega)+\frac{1}{2}\Gamma_\lambda\Phi_\lambda{}^i(\omega)\}, \qquad (VI.1.1\text{-}19)$$

$$\sigma_y^J(E)=\frac{4\pi}{k^2}\sqrt{\alpha/2m\pi}\ \frac{(\mu_l)}{E}\ g_J\sum_\lambda \frac{\Gamma_{\lambda n l}{}^0\Gamma_{\lambda y}}{\Gamma_\lambda}\{E_\lambda\Phi_\lambda{}^r(\omega)+\frac{1}{2}\Gamma_\lambda\Phi_\lambda{}^i(\omega)\}, \qquad (VI.1.1\text{-}20)$$

where

$$\mu_l=\begin{cases}(kR)^2/(1+(kR)^2) & for\ l=1,\\ (kR)^4/(9+3(kR)^2+(kR)^4) & for\ l=2,\end{cases} \qquad (VI.1.1\text{-}21)$$

$$\Gamma_{\lambda n l}^0=\Gamma_{\lambda n}/\mu_l\sqrt{E}. \qquad (VI.1.1\text{-}22)$$

VI.1.2 Average Cross Sections in the Unresolved Resonance Region

The average cross sections are defined by

$$\langle\sigma_x\rangle=\frac{1}{\Delta E}\int_{\Delta E}\sigma_x(E)dE, \qquad (VI.1.2\text{-}1)$$

where x stands for the capture, fission, elastic and inelastic scattering reactions. The energy interval ΔE is calculated from the energy points designated in the nuclear data file. For the average resonance parameters provided in the file, Eq. (VI.1.2 -1) is expressed by using a single-level Breit-Wigner formula as follows (Ref. (57)) :

$$\langle\sigma_c\rangle_{\Delta E}=\frac{2\pi}{k^2}\sum_l\sum_J g_J\left\langle\frac{\Gamma_n\Gamma_\gamma}{\Gamma}\right\rangle_{l,J}/D_{l,J}, \qquad (VI.1.2\text{-}2)$$

$$\langle\sigma_f\rangle_{\Delta E}=\frac{2\pi}{k^2}\sum_l\sum_J g_J\left\langle\frac{\Gamma_n\Gamma_f}{\Gamma}\right\rangle_{l,J}/D_{l,J}, \qquad (VI.1.2\text{-}3)$$

$$\langle\sigma_{in}\rangle_{\Delta E} = \frac{2\pi}{k^2}\sum_l \sum_J g_J \left\langle\frac{\Gamma_n\Gamma_{in}}{\Gamma}\right\rangle_{l,J}/D_{l,J}, \qquad (VI.1.2\text{-}4)$$

$$\langle\sigma_n\rangle_{\Delta E} = \frac{4\pi}{k^2}\sum_l \sum_J g_J \ (2l+1)\sin^2\varphi_l + \frac{2\pi^2}{k^2}\sum_l \sum_J g_J \frac{g_J}{D_l}\left\{\left\langle\frac{\Gamma_n\Gamma_n}{\Gamma}\right\rangle_{l,J} - 2\Gamma_{n,l,J}\sin^2\varphi_l\right\}, \qquad (VI.1.2\text{-}5)$$

$$\langle\sigma_t\rangle_{\Delta E} = \langle\sigma_c\rangle_{\Delta E} + \langle\sigma_{in}\rangle_{\Delta E} + \langle\sigma_f\rangle_{\Delta E} + \langle\sigma_n\rangle_{\Delta E}, \qquad (VI.1.2\text{-}6)$$

where the brackets mean the statistical average over the χ -square distribution with a designated degree of freedom and D the mean level spacing.

The average neutron widths for l -wave neutrons and spin J -states are defined as

$$\Gamma_{nlJ} = \Gamma_{nlJ}{}^0 \sqrt{E}\ \mu_l \nu_{nlJ}. \qquad (VI.1.2\text{-}7)$$

where ν the number of degree of freedom for the neutron width distribution and E the neutron energy.

In Eq. (VI.1.2 -5), φ_l is the phase shift given by

$$\varphi_{l=0} = kR, \qquad (VI.1.2\text{-}8)$$

$$\varphi_{l=1} = kR - \tan^{-1}kR, \qquad (VI.1.2\text{-}9)$$

$$\varphi_{l=2} = kR - \tan^{-1}\left\{\frac{3kR}{3-(kR)^2}\right\}, \qquad (VI.1.2\text{-}10)$$

where R is the effective scattering radius (in units of 10^{-12} cm).

VI.1.3 Generation of Resonance Parameters by Monte Carlo Method

Resonance widths and level spacing are distributed around their mean values according to the χ^2 - and Wigner distributions, respectively. The χ^2 - distribution is known as the distribution that a statistical variable

$$x = X_1{}^2 + X_2{}^2 + \ldots\ldots + X_n{}^2 \qquad (VI.1.3\text{-}1)$$

follows, where the n samples X_1 , X_2 ....., X_n distribute normally

$$P(x)dx = \exp(-x^2/2)dx/\sqrt{2\pi}. \qquad (VI.1.3\text{-}2)$$

Hence, the statistical variables distributed according to the χ^2 -

distribution are produced by generating the samples which distribute normally.

Now, let ξ_1 and ξ_2 are independent samples distributed uniformly in the range from 0 to 1, then, the variables x_1 and x_2 defined by

$$x_1=(-2ln\xi_1)^{1/2} \sin 2\pi\xi_2, \qquad (VI.1.3\text{-}3)$$

$$x_2=(-2ln\xi_1)^{1/2} \cos 2\pi\xi_2, \qquad (VI.1.3\text{-}4)$$

are distributed normally. That is, the Jacobian is

$$\frac{\partial(\xi_1,\xi_2)}{\partial(x_1,x_2)}=\frac{1}{2\pi}\exp\left(-\frac{x_1^2+x_2^2}{2}\right), \qquad (VI.1.3\text{-}5)$$

or

$$d\xi_1 d\xi_2 = P(x_1)dx_1 . P(x_2)dx_2. \qquad (VI.1.3\text{-}6)$$

The Wigner distribution needed for level spacing is obtained from the χ^2 - distribution with 2 degrees of freedom,

$$P_2(x)dx=e^{-x}dx, \qquad (VI.1.3\text{-}7)$$

by the variable inversion

$$y=2\sqrt{x/\pi}, \qquad (VI.1.3\text{-}8)$$

that is,

$$P_2(x)dx=\frac{\pi}{2}y\exp(-\pi y^2/4)dy. \qquad (VI.1.3\text{-}9)$$

The ladders of resonance parameters and levels are repeatedly generated until a ladder satisfies the following conditions:

$$\left|\langle\sigma_x\rangle-\sigma_x\right|/\langle\sigma_x\rangle\Big|_{\Delta E_n} \leqq \varepsilon_1, \qquad (VI.1.3\text{-}10)$$

$$\left|\langle\Gamma_x\rangle-\Gamma_x\right|/\langle\Gamma_x\rangle\Big|_{\Delta E_n} \leqq \varepsilon_2, \qquad (VI.1.3\text{-}11)$$

$$\Delta E_n=\frac{1}{2}(E_{n-1}+E_n)-\frac{1}{2}(E_n+E_{n+1}), \qquad (VI.1.3\text{-}12)$$

where x stands for the fission, capture, elastic and/or inelastic reaction, $\langle\sigma_x\rangle$ is the evaluated average cross section calculated with the evaluated average partial width $\langle\Gamma_x\rangle$, $\bar{\sigma}_x$ and $\bar{\Gamma}_x$ mean the values of the cross section and partial width averaged over a generated ladder of resonance parameter, respectively, and ε_1 and ε_2 are the assumed errors for the average cross section and partial width in the energy range ΔE_n, respectively.

VI.1.4 Calculation of Neutron Spectra

Neutron balance equation in an infinite homogeneous system can be expressed as

$$\sigma_t(u)\varphi(u)=\sum_i \frac{1}{1-\alpha_i}\int_{u-\varepsilon_i}^{u} e^{u'-u}\Sigma_{si}(u')\varphi(u')du', \qquad (VI.1.4\text{-}1)$$

$$\varepsilon_i=-ln\alpha_i, \qquad (VI.1.4\text{-}2)$$

where Σ_{si} is the scattering cross section of the i-th scatter and $\varphi(u)$ the neutron spectrum at the lethargy u.

Letting $\psi(u)=\varphi(u)e^u$,Eq.(VI.1.4 -1) is written as

$$\Sigma_t(u)\psi(u)=\sum_i \frac{1}{1-\alpha_i}\int_{u-\varepsilon_i}^{u} \Sigma_{si}(u')\psi(u')du'. \qquad (VI.1.4\text{-}3)$$

This equation is more simpler than Eq.(VI.1.4-1).In TIMS-1, Eq.(VI.1.4-3) is used in place of Eq.(VI.1.4 -1), because the factor exp(u) multiplied to ψ may serve to reduce round errors appeared in the recurrence formula introduced later for the numerical calculation of neutron slowing-down source.

The energy range of interest is divided into so extremely narrow lethargy width (ultrafine groups) that the resonance cross sections can be described enough accurately, and the ultrafine group is assumed to be less than maximum lethargy gain per collision with the heaviest nuclide. On the ultrafine group representation, the flux and collision density are defined by

$$\psi^m=\int_{u_0}^{u_+}\psi(u)du=\int_{u_0}^{u_+}\varphi(u)e^u du, \qquad (VI.1.4\text{-}4)$$

$$F_i^m=\int_{u_0}^{u_+}\Sigma_{si}(u)\psi(u)du=\Sigma_{si}^m\ \psi^m, \qquad (VI.1.4\text{-}5)$$

where u_+ and u_0 are the upper and lower lethargy boundaries of the ultrafine group m, respectively. Then, the slowing down source is shown to be written as

$$\begin{aligned} S_i^m&=\int_{u_0}^{u_+}S_i(u)du=\frac{1}{1-\alpha_i}\int_{u_0}^{u_+}du\int_{u-\varepsilon_i}^{u}\Sigma_{si}(u')\varphi(u')du' \\ &=\frac{\Delta u_m}{(1-\alpha_i)}\int_{u_0-\varepsilon_i}^{u_0}F_i(u)du \\ &=S_i^{m-1}+\frac{\Delta u_m}{(1-\alpha_i)}\{F^m-F^{m-L_i^m}\}, \end{aligned} \qquad (VI.1.4\text{-}6)$$

where Δu_m is the lethargy width of ultra-fine group m and L_i^m is the maximum number of groups which corresponds to the maximum lethargy gain by elastic collision. In the derivation of Eq.(VI.1.4-6),the self-scatter term was neglected because the effect of self-scatter was shown to be quite insignificant (Ref.(58)). In the TIMS-1 code, the scattering rate $F_i^{m-L_i^m-1}$ is approximately calculated by using the

intermediate-group method of Kier (Ref.(41)). The accuracy of this approximation was also studied and shown to be quite satisfactory. Assuming the asymptotic flux distribution below the lethargy range under consideration, the neutron spectra can be calculated recurrently by using Eq.(VI.1.4 -6) for the slowing down source. Using the calculated neutron spectra, the group constants are calculated as described in Section VI.1.5.

In Eq.(VI.1.4-3), two resonant materials and a fictitious moderator with the admixture potential scattering cross section σ_0, are considered for calculating the group constants. In the FASTLIB production, the average moderator mass A =1.0 is assumed.

VI.1.5 Group Constants

The effective group cross section for reaction x is calculated by using the neutron spectra as follows :

$$\bar{\sigma}_x(\sigma_0,T,R) = \sum_{m\in\Delta E} \sigma_x^m(T)\varphi^m(\sigma_0,T,R) / \sum_{m\in\Delta E} \varphi^m(\sigma_0,T,R), \qquad (VI.1.5\text{-}1)$$

where φ^m is the neutron spectrum at the ultra-fine lethargy group (m), σ_x^m the energy-dependent cross section at temperature T(°K), σ_0 the admixture background potential scattering cross section of the Bondarenko type and R the atomic density ratio of the resonant element (2) to the resonant element (1) of interest. Furthermore, a special 'total' cross section is calculated as(Ref.(16))

$$\bar{\sigma}_t(\sigma_0,T,R) = \sum_{m\in\Delta E} \varphi^m(\sigma_0,T,E) / \sum_{m\in\Delta E} \{(\varphi^m(\sigma_0,T,R) / (\sigma_{t,1}^m(T) + R\sigma_{t,2}^m(T+\sigma_0))\}$$

$$- \sum_{m\in\Delta E} \varphi^m(\sigma_0,T,R) / \sum_{m\in\Delta E} \{\varphi^m(\sigma_0,T,R) / (R\sigma_{t,2}(T)+\sigma_0)\}, \qquad (VI.1.5\text{-}2)$$

where $\sigma_{t,1}$ is the total cross section of the resonant element (1) of interest and $\sigma_{t,2}$ that of the other resonant element (2). This special total cross section thus defined is used to calculate the effective diffusion coefficient.The special total cross section of Eq.(VI.1.5-2) is shown to approach to the commonly defined total cross section of Eq.(VI.1.5 -1), when σ_0 value become infinite.

The elastic removal cross section is given by

$$\bar{\sigma}_{er}(\sigma_0,T,R) = \sum_{m\in\Delta E'} \sigma_n^m(T)\varphi^m(\sigma_0, T,R) \frac{(E_L-\alpha E_m)}{(1-\alpha)E_m}$$

$$/ \sum_{m\in\Delta E} \varphi^m(\sigma_0,T,R), \qquad (VI.1.5\text{-}3)$$

$$\Delta E' = E_L / \alpha - E_L, \qquad (VI.1.5\text{-}4)$$

$$\alpha = (A_1-1)^2 / (A_1+1)^2, \qquad (VI.1.5\text{-}5)$$

where E_L is the lower energy boundary of integration interval ΔE, $\sigma_n^m(T)$ the elastic scattering cross section and A_1 the atomic mass for the resonant element (1) of interest.

The resonance self-shielding factors are defined by

$$f_x(\sigma_0,T,R) = \sigma_x(\sigma_0,T,R) / \sigma_x(\infty,300,0). \qquad (VI.1.5\text{-}6)$$

In the PROF.GROUCH G II R code, under the assumption of the constant collision density, the neutron spectrum is defined by

$$\varphi^m(\sigma_0,T) = \varphi_0{}^m / \{\sigma_t{}^m(T)+\sigma_0\}, \qquad (VI.1.5\text{-}7)$$

where $\varphi_0{}^m$ represents the broad energy behavior of the 1/E or fission spectrum. The total cross section for Eq. (VI.1.5 -2) is calculated without considering the mutual interference effect as follows :

$$\bar{\sigma}_t(\sigma_0,T) = \sum_{m\in\Delta E} \{\sigma_t{}^m(T)\varphi_0{}^m / (\sigma_t{}^m(T)+\sigma_0)^2\} / \sum_{m\in\Delta E} \{\varphi_0{}^m/(\sigma_t{}^m(T)+\sigma_0)^2\}.$$

The average number of neutorns per fission is

$$\bar{\nu} = \sum_{m\in\Delta E} \nu^m\sigma_f{}^m\varphi_0{}^m / \sum_{m\in\Delta E} \sigma_f{}^m\varphi_0{}^m, \qquad (VI.1.5\text{-}8)$$

and the average scattering cosine is

$$\bar{\mu} = \sum_{m\in\Delta E} \mu^m\sigma_e{}^m\varphi_0{}^m / \sum \sigma_e{}^m\varphi_0{}^m. \qquad (VI.1.5\text{-}9)$$

The fission neutron spectrum is a slowly varying function over the most important energy range, thus the group average fission spectrum is calculated by

$$\chi = \int_{\Delta E} \chi(E',E)dE. \qquad (VI.1.5\text{-}10)$$

Assuming a 1/E flux spectrum within each group, the elastic scattering matrices is calculated by (Ref. (43))

$$\sigma_s^l(g,g') =$$

$$\frac{(2l+1)}{(1-\alpha)\Delta u_g} \int_{g'} dE' \int_g dE \frac{1}{E^2} \sigma_s(E)P_l(\mu) \sum_{l'} (2l'+1) f_{l'}(E)P_{l'}(\eta), \qquad (VI.1.5\text{-}11)$$

where l is the order of Legendre expansion coefficient, P_l the l th order Legendre polynomial, g the source group index, g' the sink group index, μ the cosine of scattering angle in the laboratory system, η the cosine of scattering angle in the center of mass system and $f_{l'}$ the Legendre expansion coefficient. The hydrogen elastic scattering matrix is a simplification of Eq. (VI.1.5-11) and assuming isotropic in the center of mass system is given by

$$\sigma_s^l(g,g') = \frac{(2l+1)}{\Delta u_g} \int_{g'} dE' \int_g dE \frac{1}{E^2} \sigma_s(E)P_l(\mu). \qquad (VI.1.5\text{-}12)$$

The inelastic scattering transfer cross section is

$$\sigma_{in}(g,g') = \sum_i \int_g dE \int_{g'} dE' \ \sigma_{in}^i(E,E')\varphi_0(E) \ / \int_g \varphi_0(E)dE \qquad (VI.1.5\text{-}13)$$

where i is the discrete level. When the data for the evapolation model is given in the form of probabilities for an evapolation scattering versus energy, the transfer matrix is obtained by combining both the scattering matrices of the level and evapolation models.

VI.1.6 Spline Interpolation for Resonance Self-Shielding Factors

The multigroup constant set prepared for SRAC-FASTLIB has the tables, σ_0 , of resonance self-shielding factors which are given as function of temperature T and effective background cross section σ_0 of Bondarenko scheme. In reactor calculation, the effective micro- scopic cross sections for each material zone are calculated from the infinitely dilute cross sections and the self-shielding factors inter- polated for the effective temperature and background cross sections using σ_0 -tables.

As for the interpolation schemes of σ_0 -tables, many methods have been studied(Ref.(20)). For example,in the 1-D diffusion code EXPANDA-4(Ref.(22)) a rational function is used for both the σ_0 - and T-interpolations, and the Kidman's schemes(Ref.(23)) used in the 1-D diffusion code IDX are a Lagrange three-point interpolation on the ln tabular points and the four-parameter interpolation formula for the σ_0 -table. Furthermore, there are several methods described in Ref.(21). Among these schemes, however, the superiority of the accuracy and computing cost is not compared. At the present time, the f-tables can numerically be calculated with high accuracy by the use of a processing code such as TIMS-1. Hence the error caused by interpolating the f-tables should be as small as possible.

In Ref.(21), various interpolation methods has been compared in connection with computing cost and accuracy. Moreover, the effects of the difference among various interpolations on integral quantities in fast reactor have been studied.As the results, the cubic spline method offers a very practical means with good accuracy and short computing time, as to the interpolation for the 'table look-up' method of resonance self-shielding factors. It was shown that the errors caused by the differences among the various interpolation methods produce non- negligible effects on effective multiplication factor, control rod reactivity worth and Doppler effect, because the effects are comparable to the goal accuracy requested from design study for a large fast breeder reactor. Furthermore, the difference produces a slightly anxious effect on reaction rate distribution in blanket region (Ref.(21)).

The cubic spline interpolation is adopted in the SRAC code system.

VI.2 Group Cross Sections in the Thermal Neutron Library

The thermal neutron library of the SRAC keeps the group cross sections and also Bondarenko type self-shielding factor tables for the nuclides which have any strong resonance level in the thermal neutron energy range. The data for the library is prepared by the SRACTLIB code system which is composed of several codes, the THERMO- series for one-dimensional cross sections, the PIXSE code (Ref.(75)) for group scattering cross sections, and the HEXSCAT code (Ref.(19)) for elastic scattering cross sections of crystalline material.

The input data for the SRACTLIB are assumed to be formed in ENDF/B format. The information needed for thermal calculations are listed below.

ν neutron yield per fission

Cross sections of the form $\sigma_x(E)$, where E is the neutron energy, and x denotes elastic, absorption, fission, total, or(n, γ)

The thermal neutron scattering law, $S(\alpha,\beta,T)$, which is represented as a tabulated part plus analytic part of the free gas or diffusive motion form.

Resolved resonance parameters in the single level Breit-Wigner, the multi-level Breit-Wigner, or Adler-Adler representation.

Lattice constant, Debye Waller factor, etc. for coherent elastic scattering of polycrystaline material.

The input requirements for the SRACTLIB described in Sect.III.2 are minimized by preparing the built-in values, however, the user can reset the standard values as he likes.

The resultant group constants are always averaged by the asymptotic neutron spectrum, as follows,

$$\sigma_{xgTT'} = \int_{\Delta E_g} \sigma_x(E,T)\varphi(E,T')dE \;/\; \int_{\Delta E_g} \varphi(E,T')dE \qquad (VI.2\text{-}1)$$

$$\sigma_{sgg'lTT'} = \int_{\Delta E_g}\int_{\Delta E'} \sigma_l(E\rightarrow E',T)\varphi(E,T')dE'dE \;/\; \int_{\Delta E_g} \varphi(E,T')dE \qquad (VI.2\text{-}2)$$

where l denotes the order of the Legendre expansion; g and g' are the group number of the energy interval ΔE_g , T is the material temperature, T' is the neutron temperature (exactly the average temperature of Maxwellian neutron density distribution). It is to be noted that even if the cross sections are not dependent on the material temperature, the group constants are temperature dependent due to the average by the temperature dependent neutron spectrum which is formed by the combination of the Maxwellian of the neutron temperature T' and 1/E part above the given value of E/kT'.

However it is very difficult to predict the neutron temperature in general. We can say only the neutron temperature is always higher than that of material. In active reactor core, the neutron temperature

is much higher than that of material due to the absorption by fuel, contrary, in reflector the neutron temperature is almost same as that of material. Fortunately the effect of neutron temperature on the group constants is very weak in our fine group structure.

The data have been prepared on the certain number of temperature points under the assumption that the neutron temperatures are 50° C higher than the material temperatures for the simplicity of the data structure. Hereafter we denote the dual temperature dependence simply by T. Such an organization of the library forces the user of the SRAC to calculate on any of the tabulated temperatures. The interpolation code is available on the reactor constants which are the output of the SRAC.

VI.2.1 Process for one-dimensional cross section

The $\sigma_x(E)$ values from the tabulated data in the original file as ν values in Mat 452 in File 1, or total, fission, parasitic absorption, capture cross sections in Mat 1, 18, 101, 102, respectively in File 3 are obtained according to the interpolation formula indicated in the file.

Here we describe the formulation of cross sections expressed by resonance parameters.

Single level Breit-Wigner representation

Capture cross section

$$\sigma_c(E) = \sum_{NRS} \sigma_0 \ (\Gamma_\gamma/\Gamma) \sqrt{E_0/E} \ \varphi(\xi, X) \qquad (VI.2.1\text{-}1)$$

Fission cross section

$$\sigma_f(E) = \sum_{NRS} \sigma_0 \ (\Gamma_f/\Gamma) \sqrt{E_0/E} \ \varphi(\xi, X) \qquad (VI.2.1\text{-}2)$$

Elastic scattering cross section

$$\sigma_n(E) = \sum_{NRS} \sigma_0 \ \Gamma_n/\Gamma \ \varphi(\xi, X) + \sum_{NRS} (\sigma_0 \sigma_p g \frac{\Gamma_n}{\Gamma})^{1/2} \chi(\xi, X) + \sigma_p \qquad (VI.2.1\text{-}3)$$

where

$$\sigma_0 = 4\pi g \lambda_0 \ \Gamma_n/\Gamma \qquad (VI.2.1\text{-}4)$$

$$\xi = (\frac{AWR \ \Gamma^2}{4\pi E_0 kT_m})^{1/2} \qquad (VI.2.1\text{-}5)$$

$$X = \frac{2(E - E_0)}{\Gamma} \qquad (VI.2.1\text{-}6)$$

$$g = \frac{(2J+1)}{2(2I+1)} \qquad (VI.2.1\text{-}7)$$

$$1/\lambda_0 = K = 2.196771*10^{-3} \left(\frac{AWR}{AWR+1}\right)\sqrt{E_0} \qquad (VI.2.1\text{-}8)$$

J : Spin of nucleus
I : Spin of target nucleus
K : Neutron wave number
AWR : Ratio of neutron mass
E_0 : Resonance energy
Γ : Resonance total width
Γ_n : Neutron width
Γ_f : Fission width
Γ_γ : Radiation width
σ_p : Potential scattering cross section
k : Boltzmann constant
T_m : Mixture temperature

The line shape functions φ and χ relating to Doppler broadning are calculated using the complex integral W as follows;

$$\varphi(\xi,X) = \xi\sqrt{\pi}\, R_eW\left(\frac{\xi X}{2},\frac{\xi}{2}\right)/2 \qquad (VI.2.1\text{-}9)$$

$$\chi(\xi,X) = \xi\sqrt{\pi}\, I_mW\left(\frac{\xi X}{2},\frac{\xi}{2}\right)/2 \qquad (VI.2.1\text{-}10)$$

where R_eW and I_mW are real and imaginary part of W. The values of W are obtained by the interpolation from the table which are prepared before the execution.

If the multi-level parameters are given, the code works as if the sigle level parameters are given.

Adler-Adler representation

The Adler-Adler representation has not yet utilized, however we would like to introduce it's Doppler broadened formulation for reaction x.

$$\sigma_x(E) = \frac{2C}{E}(1-\cos\omega) + C\sqrt{\pi}\sum_{NRS} U(x)\{G(x)\cos\omega + H(x)\sin\omega\}/(\Delta\sqrt{E})$$
$$+ C\sqrt{\pi}\sum_{NRS} V(x)\{H(x)\cos\omega + G(x)\sin\omega\}/(\Delta\sqrt{E}) \qquad (VI.2.1\text{-}11)$$

where the term due to the potential scattering is summed-up in calculation of total cross sections.

$$\Delta = 0.3177\left(\frac{T_m}{293.}\right)^{1/2}\left(\frac{E}{AWR}\right)^{1/2}$$

$$\frac{C}{E} = \pi\lambda^2,$$

$$1/\lambda = K = 2.196771*10^{-3} \left(\frac{AWR}{AWR + 1} \right)\sqrt{E}$$

$$\omega = 2\,K\,a$$

Tm : Mixture temperature
AWR : Ratio of neutron mass
K : Neutron wave number
a : Effective scattering radius
G(x): Adler-Adler parameters
H(x): Adler-Adler parameters

As in the single Breit-Wigner representation, the Doppler broadened functions U(x), V(x) are calculated using the complex integral

$$U = I_m\{\frac{1}{\pi} \int_{-\infty}^{\infty} dt exp(-t^2) \,/(\, Z-t \,)\} \qquad (VI.2.1\text{-}12)$$

$$V = R_e\{\frac{1}{\pi} \int_{-\infty}^{\infty} dt exp(-t^2) \,/(\, Z-t \,)\} \qquad (VI.2.1\text{-}12)$$

where
$Z = \xi + i\eta,\ \xi = (\mu - E\ ) / \Delta\ ,\ \eta = \nu / \Delta$

E : Energy
Δ : Doppler width
μ : Adler- Adler parameter
ν : Adler- Adler parameter

Note: The Adler-Adler parameters are optionally given in ENDF/B format file by the flag LI, so the calculation has to follow the flag to decide which quantity is primary or secondary.

LI	Given parameters	Primary quantity	Secondary quantity
5	Total,capture width	Total,capture no fission	Elastic
6	Fission ,capture width	Fission, capture	Total, no elastic
7	Total,fission, capture width	Total, fission, capture	Elastic

VI.2.2 Group transfer cross sections

For the general nuclide, the group transfer cross sections are calculated by the free gas model. For the principal moderators such as H2O, D2O, Be metal, and graphite, they are calculated from the tabulation of the scattering law $S(\alpha,\beta)$.

The differential scattering cross section $\sigma(E\rightarrow E',\theta)$ has the relation :

$$\sigma(E\rightarrow E',\theta) = \frac{\sigma_p}{2kT}\left(\frac{E'}{E}\right)^{1/2}\exp(-\beta/2)\,S(\alpha,\beta) \qquad (VI.2.2\text{-}1)$$

where E and E' are incident and secondary neutron energies, respectively, and θ is the angle of scattering. Above the energy to which the scattering law is not given, the free gas model is used.

The Legendre expansion coefficient of the scattering cross sections is calculated by Gaussian quadrature

$$\sigma_l(E\rightarrow E') = \int_{-1}^{1} P_l(\mu)\sigma(E\rightarrow E',\mu)d\mu \qquad (VI.2.2\text{-}2)$$

with $\mu = \cos\theta$

The present library keeps only l =0 and l =1 components for the principal moderators (see Dictionary VII.3)

To the thermal source (down-scatter from epi-thermal to thermal energy range), the fast library keeps the group transfer cross sections which are calculated by

$$\sigma_0(E\rightarrow E') = \sigma_s(E) \,/\, (1 - \alpha)E \qquad (VI.2.2\text{-}3)$$

$$\sigma_1(E\rightarrow E') = 3\sigma_s(E)(A + 1)^2 \,/\, 8A(E*E')^{1/2} \qquad (VI.2.2\text{-}4)$$

where

$$\alpha = \left(\frac{A - 1}{A + 1}\right)^2$$

It is to be noted that the there is no temperature dependency in the thermal source.

VI.2.3 Coherent elastic scattering cross sections

The HEXSCAT module calculates the P0 and P1 components of the polycrystaline coherent elastic neutron cross section per nucleus for a hexagonal lattice.

The same input data (Ref.(35)) as used for computing tabulated ENDF/B-III data are used in calculating the sharp structure due to Bragg peaks for Be, BeO, and graphite.

The incoherent approximation for the polycrystaline elastic neutron cross section is available using the function installed in the PIXSE routine.

VI.3 Optional Processes for Calculational Methods of Transport Cross Section and Diffusion Coefficient

This section is concerned with the options for the specification of the input data IC15,IC16 and IC17 in Sect.II.1. Discussions will be made on the calculational methods of the effective total cross sections used for the S_N and/or collision probability method, and of the diffusion coefficient for the diffusion code. In connection with the diffusion coefficient, the calculational methods adopted for the transport cross section will be also presented. Furthermore, descriptions will be given to the cell averaged diffusion coefficients.

(A) Total and Transport Cross Sections in Homogeneous Media

Firstly, we start with the P_1 equations in multigroup representation. The P_1 multigroup equations can be written as (Ref.(62))

$$\nabla \cdot J_g(r) + \Sigma_{t,g}\, \Phi_g(r) = \sum_{g'} \Sigma_{so,g\to g'}\, \Phi_{g'}(r) + Q_g(r) \qquad (VI.3\text{-}1)$$

$$\nabla \Phi_g(r) + 3\, \Sigma_{t,g}\, J_g(r) = 3 \sum_{g'} \Sigma_{s1,g'\to g}\, J_{g'}(r) \qquad (VI.3\text{-}2)$$

where notation is conventional.

If a group-dependent form of Fick's law is postulated, i.e.,

$$J_g(r) = -D_g(r)\nabla\Phi_g(r), \qquad (VI.3\text{-}3)$$

then from Eq.(VI.3-2) the diffusion coefficient can be formally expressed by

$$D_g(r) = \frac{1}{3\Sigma_{tr,g}(r)} \qquad (VI.3\text{-}4)$$

with

$$\Sigma_{tr,g}(r) \equiv \Sigma_{t,g} - \sum_{g'} \Sigma_{s1,g'\to g}\, J_{g'}(r)/J_g(r). \qquad (VI.3\text{-}5)$$

If the group width is wider compared with the maximum energy degradation of neutron by elastic collision and the quantity $EJ(r,E)$ can be assumed to be energy-independent in the width, we can show (Ref. (62))

$$\sum_{g'} \Sigma_{s1,g'\to g} J_{g'}(r)/J_g(r) \sim \sum_{g'} \Sigma_{s1,g\to g'} = (\bar{\mu})_g\, \Sigma_{s,g} \qquad (VI.3\text{-}6)$$

In this case, we have

$$\Sigma_{tr,g}(r) = \Sigma_{t,g} - \sum_{g'} \Sigma_{s1,g\to g'} \qquad (VI.3\text{-}7)$$

i) Effective total cross section

There are two ways to define the effective microscopic total cross section in the SRAC system. In either way, the total cross sections are

given at the first stage for all the compositions concerned.

** IC15 = 1 **

One way is based on the concept of the Bondarenko type cross section set (Ref. (2)), i.e.,

$$\sigma_{t,g} = \sigma_{t\infty,g} \cdot f_{t,g}(\sigma_0), \qquad (VI.3\text{-}8)$$

where the suffix g stands for the group number, and $f_t(\sigma_0)$ is the self-shielding factor for the total cross section in Sect. VI.1. The total cross section thus defined is valid only for the calculation of the diffusion coefficient.

** IC15 = 2 **

On another way, the total cross section is defined as the summation of all partial reaction cross sections, i.e.,

$$\sigma_{t,g} = \sum_{x} \sigma_{x\infty,g} \cdot f_{x,g}(\sigma_0), \qquad (VI.3\text{-}9)$$

where $f_{x,g}(\sigma_0)$ is the self-shielding factor for reaction type x. This cross section can be used to obtain the transport cross section for the diffusion coefficient calculation or for the extended transport approximation (Ref. (63)).

ii) Effective transport cross section

Four methods are prepared to calculate the transport cross section.

** IC16 = 0 **

The first one is used to treat the anisotropic scattering in the multigroup transport S_N and/or collision probability calculation by the extended transport approximation. Here, the macroscopic transport cross section is defined by

$$\Sigma_{tr,g} = \Sigma_{t,g} - \sum_{g'} \Sigma_{s1,g' \to g}, \qquad (VI.3\text{-}10)$$

where the macroscopic total cross section $\Sigma_{t,g}$ is calculated by Eq. (VI. 3-9).

It should be noted that the shielding factor is not prepared yet for the P_1 component $\Sigma_{s1,g' \to g}$ of scattering matrix. Moreover, the total cross section, instead of the transport cross section, is used to calculate the collision probability in the second resonance range, where the ultra-fine spectrum is calculated by assuming isotropic neutron scattering.

** IC16 = 1 or 2 **

The second and third method is based on the P_1 and B_1 approximation, respectively, for fast neutron spectrum calculation in homogeneous media (Ref. (64)). By choice of the option IC16 = 1 or 2, the

system under study is at first homogenized by simply smearing the atomic number densities if heterogeneous, and the P_l or B_l equations are solved by assuming an appropriate buckling. Then, using the homogeneous spectrum obtained and assuming the flat flux, the effective cross section of each region is calculated by

$$\Sigma_{tr,g} = (\gamma_g \Sigma_{t,g} - \sum_{g'} \Sigma_{s1,g'\to g} \Phi_{1,g'} / \Phi_{1,g}) \qquad (VI.3\text{-}11)$$

where

$$\gamma_g \equiv \begin{cases} 1 & for\ P_l \text{ approximation} \\ \alpha_g tan^{-1}\alpha_g / [\ 3\{1-(tan^{-1}\alpha_g)/\alpha_g\}\ ] & \\ \text{for } B_l \text{ approximation with } \alpha_g = B/\Sigma_{t,g}. & \end{cases} \qquad (VI.3\text{-}12)$$

Here, $\Phi_{0,g}$ and $\Phi_{1,g}$ is, respectively, the P_0 and P_1 component of the angular flux, and B is the square root of the geometrical buckling. This approach is essential for the calculation of the lattice cell or homogenous medium including hydrogeneous materials.

** IC16 = 3 **

The last method is based on the S_N transport calculation used for cell or core calculation. Here, the transport cross section is define , following to Eq. (VI.3-5), by

$$\Sigma_{tr,g}(r) = \Sigma_{t,g} - \sum_{g'} \Sigma_{s1,g'\to g} \Phi_{1,g'}(r) / \Phi_{1,g}(r) \qquad (VI.3\text{-}13)$$

This quantity is averaged over each volume or region in the system under consideration to give the representative value

(B) Diffusion Coefficient in Lattice Cell

Benoist presented a theory of the diffusion coefficient in reactor lattice, leading to expressions valid in full generality (Ref. (36)). For the diffusion coefficient of the direction k , omitting the absorption correction and angular terms, this theory gives

$$D_k = \sum_i \sum_j V_i \Phi_i \lambda_j P_{ij,k} / (3 \sum_i V_i \Phi_i). \qquad (VI.3\text{-}14)$$

where i and j stand for the region number, λ_j the transport mean free path for region j and $P_{ij,k}$ is a directional first collision probability given by Eq. (VI.5-3).

The SRAC system is provided with the three methods for calculating the diffusion coefficient D_k based on Eq. (VI.3-14).

** IC17 = 1 **

The first method is a rather sophisticated one which has been proven by experience to be fairly accurate, and the diffusion coeffic-

ient is given by the inverse of the the cell averaged transport cross section, i.e.,

$$D_k = \sum_i V_i\Phi_i / (3 \sum_i V_i\Phi_i \ \Sigma_{tr,i}). \qquad (VI.3\text{-}15)$$

When the flux Φ_i are all equal in the mixture and we can assume

$$P_{ij,k} = V_i\Sigma_{tr,i} / (3 \sum_j V_j\Sigma_{tr,j}) \qquad \text{(homogenenous limit)} \qquad (VI.3\text{-}16)$$

independent of k, we have

$$D_k' = \sum_i V_i / (3 \sum_i V_i \ \Sigma_{tr,i}). \qquad (VI.3\text{-}17)$$

On the other hand, if a neutron born in any region i is certain to suffer its first collision in this region (the assumption that each region is large compared to the mean free path), then

$$P_{ij,k} = \delta_{ij} \qquad (VI.3\text{-}18)$$

and

$$D_k'' = \sum_i V_i\Phi_i\lambda_i / (3 \sum_i V_i\Phi_i \ ) \ . \qquad (VI.3\text{-}19)$$

It follows at once from the fundamental theorem of algebra

$$D_k'' \geqq D_k \ . \qquad (VI.3\text{-}20)$$

The use of Eq. (VI.3 - 19) to the cell with voided regions falls into the drawback that the diffusion coefficient is unreasonably too large to be used in practical calculation.

** IC17 = 2 **

The mean diffusion coefficient is used and defined by

$$D_0 = \frac{1}{3} \sum_k D_k \ . \qquad (VI.3\text{-}21)$$

Then, from the definition of the directional collision probability by Eq. (VI.5 - 2), this diffusion coefficient can be calculated using the isotropic collision probability defined by Eq. (VI.5-1), i.e.,

$$D_k = \sum_i \sum_j V_i\Phi_i\lambda_j \ P_{ij} / (3 \sum_i V_i\Phi_i) \ . \qquad (VI.3\text{-}22)$$

** IC17 = 3 **

The anisotropic diffusion coefficients defined by Eq. (VI.3-14) are calculated. These coefficients are used for the 2-D or 3-D diffusion calculation using the CITATION code.

VI.4 Optional Processes For Resonance Absorption

Concerning thermal reactor analysis, a more accurate treatment, compared with fast reactor, is needed for the calculations of the effective resonance cross sections in the lower energy regions. The upper energy boundary of the energy range requiring the special attention is assigned to be 130.07 eV in the SRAC system, as previously mentioned. In this resonance energy region, the effective fission and capture cross sections formerly obtained from the table-look-up of the f-tables based on the NR approximation are replaced by those from the more accurate methods.

In the followings, descriptions will be given to the methods which are incorporated in the SRAC system for the accurate calculation of the effective resonance cross sections.

VI.4.1 Table-Look-Up Method of f-Tables Based on IR Approximation (The Method Adopted in Subroutine 'IRA')

At first, we start with the intermediate resonance approximation (IRA) of resonance absorption in homogeneous systems, to give an insight into the relationship with the table-look-up method.

Consider an infinite homogeneous mixture consisting of one kind of resonance nuclide and of nonabsorbing moderator. When the narrow resonance approximation (NRA) is applied for the slowing down due to the moderator nuclide, the slowing down equation can be written as

$$(\sigma_t(u) + \sigma_b)\varphi(u) = K(\sigma_s\varphi) + \sigma_b \qquad (VI.4.1\text{-}1)$$

with

$$\sigma_t(u) = \sigma_a(u) + \sigma_s(u) \qquad (VI.4.1\text{-}2)$$

where $\varphi(u)$ is flux per unit lethargy, K, the slowing down kernel, σ_a , σ_s , microscopic absorption and scattering cross sections of the resonance absorber, and σ_b is the scattering cross section of the moderator. Equation (VI.4.1-1) is the basic equation that has been used in the construction of the SRAC library.

From the two extreme cases representing the limits of narrow resonance (NR) and wide resonance (WR),for the slowing down kernel, the first-order solution for $\varphi(u)$ can be written as (Refs.(65, 66))

$$\varphi(u) \sim \frac{\lambda\sigma_p + \sigma_b}{\sigma_a + \lambda\sigma_s + \sigma_b} \qquad (VI.4.1\text{-}3)$$

where λ is the intermediate resonance (IR) parameter for the absorber and σ_p is the potential scattering cross section. The value of λ can be determined by solving a transcendental equation for λ (Refs.(65, 66)).

For a homogeneous system including many moderator nuclides, the slowing down equation can be written as

$$\{\sigma_t(u) + \sigma_m\}\varphi(u) = K(\sigma_s\varphi) + \sum_j \sigma_j K_j(\varphi) \qquad (VI.4.1\text{-}4)$$

where

$$\sigma_m = \sum_j \sigma_j \qquad (VI.4.1\text{-}5)$$

and K_j is the slowing down kernel for moderator j. The corresponding

first-order solution can be given by

$$\varphi(u) \sim \frac{\lambda\sigma_p + \sigma_b'}{\sigma_a + \lambda\sigma_s + \sigma_b'} \qquad (VI.4.1\text{-}6)$$

with $$\sigma_b' = \sum_j \lambda_j \sigma_j, \qquad (VI.4.1\text{-}7)$$

where λ_j is the IR parameter for moderator j and can be again determined by solving a coupled set of transcendental equations (Ref.(66)).

Here, it should be noted that both the fluxes obtained from a numerical integration of Eq.(VI.4.1-1) and given by Eq.(VI.4.1-2) or (VI.4.1-6) are the weighting functions for cross section averaging. Hence, they can be assumed to give the same value for the effective cross section in the extent of the accuracy of the IRA. Consequently, a homogeneous system with σ_b has the same effective cross section as the homogeneous system with the same σ_b of Eq.(VI.4.1-1). That is, the effective cross sections can be calculated by determining the IR parameters and σ_b' as to be used for the table-look-up of the resonance shielding tables.

The IRA method described above can be applied only to a zero temperature system. For nonzero temperature, the IR parameter λ for absorber depends on temperature when the interference between potential and resonance scattering is taken into consideration (Ref.(67)). A simple way to take account of this dependence is to multiply the interference scattering term by a factor with temperature dependence (Ref.(67)).

Next consider the IR treatment of resonance absorption in heterogeneous systems. Assuming a flat flux, the slowing down equation in the two-region system, consisting of an absorbing lump (f) and a nonabsorbing moderator (m), can be written as

$$\sigma_f\varphi_f = p_{ff}\{\sigma_{am}K_{am}(\varphi_f) + K_f(\sigma_{sf}\varphi_f)\} + (1-p_{ff})\sigma_f\sum_k\{R_kK_k(\varphi_m)\} \qquad (VI.4.1\text{-}8)$$

$$\sigma_f\varphi_f + \sigma_m\varphi_m = \sigma_{am}K_{am}(\varphi_f) + K_f(\sigma_{sf}\varphi_f) + \sigma_m\sum_k\{R_kK_k(\varphi_m)\} \qquad (VI.4.1\text{-}9)$$

where

$\varphi_f(u)$, φ_m = flux per unit lethargy in the lump and moderator region, respectively

$\sigma_f(u) = \sigma_a(u) + \sigma_s + \sigma_{am}$ = microscopic total cross section of the lump

σ_{am} = scattering cross section of admixed moderator per absorber atom

$\sigma_m = \Sigma_m v_m/(n_f v_f)$, $R_k = \Sigma_k/\Sigma_m$, $\Sigma_m = \sum_k \Sigma_k$

v_f, v_m = volumes of the lump and the moderator regions, respectively

n_f = number density of the resonance absorber in the lump

K = slowing down operator

p_{ff} = effective collision probability in the fuel lump.

Other notations are conventional.

We make use of the simple interpolation formula for the collision probability as proposed by Wigner:

$$p_{ff} = \frac{x}{x + g(c)(1 - c)} = \frac{\sigma_f}{\sigma_f + s} \qquad (VI.4.1\text{-}10)$$

with $$x = l_f n_f \sigma_f = l_f \Sigma_f \qquad (VI.4.1\text{-}11)$$

$$s = g(c)(1 - c)/(l_f n_f) \quad \text{and} \quad g(c) = \frac{a}{1 + (a-1)c} \qquad (VI.4.1\text{-}12)$$

where l_f the lump mean chord length, c, the Dancoff factor, and a is a purely geometrical quantity.

Generalized collision probability theory shows

$$p_{ff} \sim 1 - \frac{1 - c}{x} \quad \text{for} \quad x \to \infty, \qquad (VI.4.1\text{-}13)$$

where the Dancoff factor, c, is zero for isolated lumps. A rational interpolation of p_{ff} leads to g = 1 in Eq. (VI.4.1-10) when use is made only of the behavior of p_{ff} at $\Sigma = \infty$, given by Eq. (VI.4.1-13). On the other hand, since the bulk of resonance absorption occurs at finite values of Σ_f, we need some corrections for the rational approximation. It is this quantity a that has been introduced for the corrections (Refs. (68,69,70)).

Being based on Eq. (VI.4.1-13), the Dancoff factor, c, is calculated by using the value of p_{ff} for a sufficiently large value of Σ_f, as $\Sigma_f = 300\ cm^{-1}$,

$$1 - c = \{1 - p_{ff}(\Sigma_f)\} x \,|_{\Sigma_f \to \infty} . \qquad (VI.4.1\text{-}13')$$

As the quantity, a, usually referred to as the Bell or Levine factor, somewhat fluctuate among resonances, there might be some minor problems with the choice. The exact choice of this quantity is not believed to be important, considering from the results of many studies done in this field. The values recommended in the SRAC system for the geometric quantities are shown in the following table:

Table VI.4.1-1

Geometry	l_f	a	Remarks
Sphere of Radius r	4r/3	1.69	Ref. (4,71)
Slab of Thickness r	2r	1.2	Ref. (71,73)
Infinite Cylinder of Radius r	2r	1.2	Ref. (68,69)
Infinite Long Cylinder of Radius r	$2r\cos^2\theta$	1.2	Ref. (72,73)

Substituting Eq. (VI.4.1-10) into Eq. (VI.4.1-8) and subtracting Eq. (VI.4.1-9) from the resulting equation, we obtain the following set of equations for neutron balance :

$$(\sigma_f + s)\varphi_f = \sigma_{cm} K_{cm}(\varphi_f) + K_f(\sigma_{sf}\varphi_f) + s\sum_k \{R_k K_k(\varphi_m)\} \qquad (VI.4.1\text{-}14)$$

$$s\varphi_f = \sigma_m\varphi_m + (s-\sigma_m)\sum_k \{R_k K_k(\varphi_m)\} \qquad (VI.4.1\text{-}15)$$

Then, from the two extreme cases representing the limits of narrow resonance (NR) and wide resonance (WR), respectively, for the slowing down kernels, the first-order solution for φ_f and φ_m can be written as (Ref.(3))

$$\varphi_f^{(1)}(u) = \frac{\lambda\sigma_b + \kappa\sigma_{am} + \mu^* s}{\sigma_a + \lambda\sigma_{sf} + \kappa\sigma_{am} + \mu^* s} \qquad (VI.4.1\text{-}16)$$

$$\varphi_m^{(1)}(u) = 1 - s\{1-\varphi_f^{(1)}(u)\}/\{\mu\sigma_m + (1-\mu)s\} \qquad (VI.4.1\text{-}17)$$

with

$$\mu^* = \mu\sigma_m/\{\mu\sigma_m + (1-\mu)s\} \quad \text{and} \quad \mu = \sum_k R_k\mu_k \ , \qquad (VI.4.1\text{-}18)$$

where μ_k is the IR parameter for the outside moderator,k. Here, a set of the IR parameters can be determined by the same procedure as those in a homogeneous system (Ref.(3)).

The simple result of Eq.(VI.4.1-16) again reveals the following equivalence relation with practical usefulness : A homogeneous system with $\sigma_b = \kappa\sigma_{am} + \mu^* s$ has the same effective resonance cross section as a homogeneous system of Eq.(VI.4.1-1) with the same σ_b . In other words, the effective cross sections in a heterogeneous system can be estimated by using a cross section set of the Bondarenko type (Ref.(4)) which is calculated based on Eq.(VI.4.1-1).

VI.4.2 Numerical Method for Calculating Neutron Flux Distribution (The Method Adopted in Subroutine 'PEACO')

We assume that heterogeneous systems are built up of an infinite number of 'unit cells' and the neutron balance in a heterogeneous system can be described by using the first-flight collision probabilities. To reduce the numerical errors caused by the flat-flux assumption, each region of the system may be divided into subregions as many as necessary or possible. Then, assuming the isotropic elastic scattering, the neutron balance in a cell may be written by the neutron slowing down equation

$$V_i\Sigma_i(u)\Psi_i(u) = \sum_{j=1}^{J} P_{ji}(u)V_j \sum_{k=1}^{K} S_{jk}(u) \qquad (VI.4.2\text{-}1)$$

$$S_{jk}(u) = \frac{1}{1-\alpha_k}\int_{u-\varepsilon_k}^{u} \exp\{-(u-u')\}\Sigma_{sjk}(u')\Psi_j(u')du' \qquad (VI.4.2\text{-}2)$$

with

$$\alpha = \left(\frac{A-1}{A+1}\right)^2 \quad \text{and} \quad \varepsilon = -\ln\alpha. \qquad (VI.4.2\text{-}3)$$

Here, the subscript i or j stands for the subregion number and the k corresponds to the nuclear species. The quantity P_{ji} is the effective probability in a unit cell that a neutron scattered isotropically in region j ,into lethargy u,will have its first collision in region i , and other notation has the customary meanings.

By letting $V_i\Psi_i(u)\exp(u) = \psi_i(u)$,we have

$$\Sigma_i(u)\psi_i(u) = \sum_{j,k} P_{ji}(u)\, S^0{}_{jk}(u) \qquad (VI.4.2\text{-}4)$$

with

$$S^0{}_{ik}(u) = \frac{1}{1-\alpha_k}\int_{u-\varepsilon_k}^{u} F_{jk}(u')du' \qquad (VI.4.2\text{-}5)$$

$$F_{jk}(u) = \Sigma_{sjk}(u)\psi_j(u). \qquad (VI.4.2\text{-}6)$$

Here,note that the equations (VI.4.2-4) and (VI.4.2-5) for $\psi_i(u)$ is more simple than Eqs.(VI.4.2-1) and (VI.4.2-2).

For the computation of the neutron spectra $\psi_i(u)$ on discrete lethargy meshes, we use the RABBLE Method developed by Kier (Refs.(40, 41)). Hence, the lethargy meshes used are assumed to be extremely narrow compared to the maximum lethargy gain per collision with the heaviest nuclides in the system under consideration. Furthermore, we assume that the resonance cross sections are given at the mid-point of this fine group and the collision probabilities $P_{ji}(u)$ is constant over a fine group. Now define

$$\psi^m{}_i = \int_{u_0}^{u_+}\psi_i(u)du \qquad (VI.4.2\text{-}7)$$

$$F^m{}_{jk} = \int_{u_0}^{u_+}F_{jk}(u)du = \Sigma^m{}_{sjk}\psi^m{}_j \qquad (VI.4.2\text{-}8)$$

where u_+ and u_0 are upper and lower lethargy bounds, respectively, corresponding to the fine group m. Then, the integration of Eq.(VI.4.2-4) over the lethargy range from u_0 to u_+ gives

$$\Sigma^m{}_i\psi^m{}_i = \sum_{j,k} P^m{}_{ji}Q^m{}_{jk} \qquad (VI.4.2\text{-}9)$$

where

$$Q^m{}_{jk} = \frac{1}{1-\alpha_k}\int_{u_0}^{u_+}du\int_{u-\varepsilon_k}^{u}F_{jk}(u')du' \qquad (VI.4.2\text{-}10)$$

$$\sim \frac{\Delta u_m}{1-\alpha_k}\int_{u_0-\varepsilon_k}^{u_0}F_{jk}(u)du$$

$$= Q^{m-1}{}_{jk} + \frac{\Delta u_m}{1-\alpha_k}\{F^m{}_{jk} - (F_{jk})^{m-L^m_k}\} \qquad (VI.4.2\text{-}11)$$

with

$$(F_{jk})^{m-L^m_k} = \int_{u_--\varepsilon_k}^{u_0-\varepsilon_k}F_{jk}(u)du. \qquad (VI.4.2\text{-}12)$$

Here, u_- is the lower bound of the (m-1)th fine group and $L^m{}_k$ is an integral number of groups which corresponds of the maximum lethargy gain by elastic collision. Note that the self-scatter was neglected in deriving the above equations because the effect of the self-scatter was shown to be quite insignificant (Ref.(58)).

It is moreover assumed that the scattering rate given by Eq. (VI.4.2-12) can be approximated by using the intermediate group scattering rate (Ref. (39)). The accuracy of this approximation was also investigated and shown to be quite satisfactory when an adequate group structure is adopted for representing the fine and intermediate groups (Ref. (39)). Assuming the asymptotic flux distribution below the lethargy range under consideration, the neutron flux distribution can be recurrsively calculated, until the entire energy range of interest is covered.

VI.4.2-1 Interpolation Method for Collision-Probability Calculation

The lattice cell under study may consist of several materials and each number referring to the materials corresponds generally to few different region-numbers. The resonance-absorbing isotopes are contained in some materials (fuel-materials) and other materials are assumed to have constant cross sections. Now, for such a system we select a resonance absorber, say ^{238}U , and define the microscopic resonance cross section per the absorber under consideration in the fuel materials by

$$\bar{\sigma}_I(E) = \frac{1}{N_I} \sum_K N_{IK}\sigma_K(E) \qquad (N_I \neq 0) \qquad (VI.4.2\text{-}13)$$

where I corresponds to the fuel material, the summation on k is extended over all the resonance absorbers, $\sigma_K(E)$ is the microscopic total cross section of the k'th absorber, and N_I and N_{IK} are the atomic number densities of the absorber under consideration and of the k'th resonance absorber in the I'th material, respectively.

Some fuel materials may have the same relative densities N_{IK}/N_I , hence the same value for $\bar{\sigma}_I(E)$. In such materials, the value of $\bar{\sigma}_I(E)$ can be considered as a common variable to express the macroscopic total cross sections. There may be another possibility to find such an independent variable in other fuel materials. In the present treatment, it is assumed that the macroscopic total cross sections can be expressed or be approximated by using at most two such independent variables.

Now, let us denote by Σ_{0J} the smallest macroscopic total cross section made by the nonresonance isotopes in the fuel materials belonging to the independent variable $\bar{\sigma}_J(E)$. Let us introduce new variables defined by

$$X_J = (N_J\bar{\sigma}_J + \Sigma_{0J})\bar{l}_J \qquad J=1, IMAX, \qquad (VI.4.2\text{-}14)$$

where IMAX is the number of the independent variables (IMAX $\leqq$2) and $\bar{l}_J$ is the averaged value of double widths of the regions which belong to the variable $\bar{\sigma}_J(E)$. Then, the collision probability, $P_{ji}(u)$, in Eqs. (VI.4.2-1) or (VI.4.2-5) can be considered to be a function of X (J=1, IMAX).

In the subroutine 'PEACO', two kinds of computational methods are adopted for the calculation of the $X_J(J=1, IMAX)$: In the first method which is used for $P_{ji}(X_1, X_2)$, the values of $X_1, X_2 \leqq 9$. needed for the

interpolation are calculated by calling the subroutine 'PIJ-2'. The second one uses the asymptotic expansion of $P_{ji}(X_1,X_2)$ when X_1 or $X_2 \geqq 9$. .

(I) One Resonance-Absorbing Composition Problem (IMAX=1)

It is easy to show (Refs. (69,71,73))

$$P_{ij}(X) \rightarrow \eta_{ij} + \gamma_{ij}/X \quad for\ X \rightarrow \infty \tag{VI.4.2-15}$$

with

$$\eta_{ij} \equiv P_{ij}(X=\infty) \tag{VI.4.2-16}$$

and

$$\gamma_{ij} \equiv \begin{cases} \{P_{ij}(X)-\eta_{ij}\}X\,|_{X\rightarrow\infty} & if\ i \in RAC \\ 0 & \text{otherwise,} \end{cases} \tag{VI.4.2-17}$$

where RAC stands for the resonance-absorbing material.

The above equation (VI.4.2-15) is used for $X > 9$ and the generalized Dancoff factor γ_{ij}, given by Eq. (VI.4.2-17) is calculated at X = 9, while η_{ij} is obtained as the value of $P_{ij}(X)$ at $X = 10^4$.

On the other hand, for the range of $X < 9$ we introduce a new variable

$$Z \equiv \frac{X}{X+1} \quad or \quad X = \frac{Z}{1-Z} . \tag{VI.4.2-18}$$

The interpolation of the collision probability is made by using the values of $P_{ij}(X)$ calculated on ten points of the variable Z with Z=0.1 and the Lagrangian three points interpolation formula (Ref. (74))

$$f(Z_0 + \delta Z) = \frac{1}{2}u(u-1)f_{-1} + (1-u^2)f_0 + \frac{1}{2}u(u+1)f_1 \tag{VI.4.2-19}$$

with

$$u = \frac{\delta Z}{\Delta Z} \quad and \quad |u| \leqq 1, \tag{VI.4.2-20}$$

where the quantity ΔZ is the mesh width of the variable Z at $Z = Z_0$ corresponding to the direction of the increment δZ .

The values of Z and X used for the interpolation are shown in the following table:

Table VI.4.2-1

No.	Z	X
1	0	0
2	0.1	0.111111
3	0.2	0.25
4	0.3	0.423571
5	0.4	0.666667
6	0.5	1.

7	0.6	1.5
8	0.7	2.333333
9	0.8	4.
10	0.9	9.
11	0.9999	10E+4

(II) Two Resonance-Absorbing Composition Problem (IMAX=2)

We can prove also for the two resonance-absorbing composition problem

$$P_{ij}(\infty, X_2) - \gamma^1{}_{ij}(X_2)/X_1 \qquad X_1 > 9,\ X_2 \leqq 9 \qquad (VI.4.2\text{-}21)$$

$$(\gamma^1{}_{ij}(X_2) \equiv 0 \quad if\ i \notin X_1)$$

$$P_{ij}(X_1, X_2) \sim$$

$$P_{ij}(X_1, \infty) - \gamma^2{}_{ij}(X_1)/X_2 \qquad X_1 \leqq 9,\ X_2 > 9 \qquad (VI.4.2\text{-}22)$$

$$(\gamma^2{}_{ij}(X_1) \equiv 0 \quad if\ i \notin X_2)$$

$$P_{ij}(\infty, \infty) - \gamma^{\infty}_{ij}/X_1 \quad (i \in X_1)$$

$$P_{ij}(\infty, \infty) - \gamma^{\infty}{}_{ij}/X_2$$

$$P_{ij}(X_1, X_2) \sim \quad (i \in X_2) \qquad X_1, X_2 > 9 \qquad (VI.4.2\text{-}23)$$

$$P_{ij}(\infty, \infty)$$

(otherwise).

Here, P_{ij} and γ_{ij} in Eqs. (VI.4.2-21) and (VI.4.2-22) are calculated by the exactly same way as the case of the interpolation of $P_{ij}(X)$ for $X < 9$ in the one resonance-absorbing composition problem.

For the range of $X_1,\ X_2 \leqq 9$, we again introduce two variables

$$Z_I = \frac{X_I}{1+X_I} \quad or \quad X_I = \frac{Z_I}{1-Z_I} \qquad (I=1,2) \qquad (VI.4.2\text{-}24)$$

and $P_{ij}(X_1, X_2)$ is interpolated on the these variables by using one of the following formulae (Ref. (74))

$$f(Z_{10} + \delta Z_1,\ Z_{20} + \delta Z_2) = \{1 - \frac{1}{2}(3-u-v)(u+v)\} f_{00}$$
$$+ (2-u-v)(uf_{10} + vf_{01}) + uvf_{11}$$
$$+ \frac{1}{2}u(u-1)f_{20} + \frac{1}{2}v(v-1)f_{02}\ , \qquad (VI.4.2\text{-}25)$$

with $u=\frac{\delta Z_1}{\delta Z}$, $v=\frac{\delta Z_2}{\Delta Z}$, $|u| \leqq 1, |v| \leqq 1,$ (VI.4.2-26)

$$f(Z_{10}+\delta Z_1, Z_{20}+\delta Z_2)=\frac{1}{2}u(u-1)f_{-10}+\frac{1}{2}v(v-1)f_{0-1}$$
$$+(1-uv-u^2-v^2)f_{00}+\frac{1}{2}u(u-2v-1)f_{10}$$
$$+\frac{1}{2}v(v-2u+1)f_{01}+uvf_{11}, \quad (VI.4.2\text{-}27)$$

$$f(Z_{10}+\delta Z_1, Z_{20}+\delta Z_2)$$
$$=(1-u)(1-v)f_{00}+u(1-v)f_{10}+v(1-u)f_{01}+uvf_{11}. \quad (VI.4.2\text{-}28)$$

The choice of Eqs.(VI.4.2-25), (VI.4.2-27) or (VI.4.2-28) depends on the values of Z_1, Z_2 on which the $P_{ij}(X_1, X_2)$ is to be interpolated.

(III) Conservation and Reciprocity of Collision Probability

The first-flight collision probability must satisfy the two important relations, that is, the conservation law

$$\sum_j P_{ij} = 1 \qquad for\ all\ i \quad (VI.4.2\text{-}29)$$

and the reciprocity relation (Ref.(62))

$$V_i\Sigma_i P_{ij} = V_j\Sigma_j P_{ji} \quad for\ all\ i,j. \quad (VI.4.2\text{-}30)$$

In the subroutine 'PEACO', at first, the values of P_{ij} are calculated only for $i \leqq j$ by Eqs. from (VI.4.2-15) and (VI.4.2-28). Then, the collision probabilities satisfying Eqs.(VI.4.2-29) and (VI.4.2-30) are successively obtained by the following equations starting from i=1:

$$P_{ij}^* = \frac{1-\beta_1}{\beta_0} P_{ij} \ , \ P_{ji}^* = \frac{1-\beta_1}{\beta_0} P_{ji} \quad (j=i, J) \quad (VI.4.2\text{-}31)$$

where

$$\beta_0 = \sum_{j=i}^{J} P_{ji} \quad and \quad \beta_1 = \sum_{i=1}^{j-1} P_{ij}^* \quad (VI.4.2\text{-}32)$$

with

$$P_{ij} = V_j\Sigma_j P_{ji} / (V_i\Sigma_i) \qquad for\ j > i. \quad (VI.4.2\text{-}33)$$

The collision probabilities P_{ij}^* given by Eq.(VI.4.2-31) will be readily known to satisfy Eqs.(VI.4.2-29) and (VI.4.2-30).

Using the interpolation and the asymptotic expansion, combined with the method mentioned above, we can guarantee the accuracy of 0.1%

for the calculation of the collision probability, including the one resonance-absorbing composition problem. Furthermore it should be emphasized that most of practical problems can be executed in a computing time of the same order as the convenient method based on the IRA.

VI. 4.3 Treatment of Double Heterogeneity

We frequently encounter a situation in calculating the effective cross sections that the geometry under study can be assumed to have a double heterogeneity, which can not be treated by a one-through calculation due to its complexity : For example, a high-temperature gas-cooled reactor (HTGR) uses fuel in the form of small spherical coated particles. A coated particle consists of a fuel kernel of UO_2 and several layers of pyrolitic carbon and S_iC . Such coated particles, together with a graphite binder, are formed into annular fuel rods that are then inserted into a graphite block. In this case, a coated particle is assumed to constitute a unit cell (the primary cell), and an annular fuel rod does another unit cell (the secondary cell). Then the resonance cross sections are shielded through both the microscopic grain structure and macroscopic rod configuration. Another example of the double heterogeneity is that between fuel pin and subassembly lattice in light water reactors and/or fast reactors.

In this section, the two homogenization procedures adopted in the SRAC system will be presented; the first one is based on the direct numerical-method described in the previous section, while the second one uses the table-look-up method shown in Sect. VI. 4.1. Both the procedures are composed of the two step calculation : At the first step, the primary cell is spatially smeared as to give a representative homogenized mixture. At the next step, the homogenization procedure is executed for the secondary cell with gross heterogeneity.

(I) Direct Numerical Method

The direct method incorporated in the subroutine 'PEACO' is used in each calculation step for homogenizing the double heterogeneity. At first the spatial fine structure of ultra-fine group is calculated in the primary cell, and is spatially smeared without collapsing the groups to give a representative resonance cross section for the cell under study. The resonance cross-section variation with neutron energy thus obtained is much smooth, compared with the original library cross-section variation ,due to the spatial shielding.

The ultrafine cross sections are prepared for all the reactions of the resonant nuclides in each R-region and stored in the other data files assigning the new Member Names. At the second step, the spatial flux distribution is again calculated by using these cross sections and calling the collision-probability routine for the secondary lattice cell . The cell is smeared and the ultra-fine groups are collapsed into few groups to give the homogenized cross sections.

(II) Table-Look-Up Method with IR Approximation

The first-order solution for the primary cell is assumed to be given by Eq.(VI.4.1-16). This solution shows another important feature of resonance absorption in a heterogeneous system, that is, it shows that a heterogeneous system described by that equation has the same effective cross section as a homogeneous system that consists of the absorber lump composition admixed with an additional moderator with the

scattering cross section, s, and the IR parameter, μ^* . Here, it should be noted that the quantities s and μ^* appear in the product form $\mu^* s$ both in Eq. (VI.4.1-16) and in the coefficients determining the IR parameters (Ref. (4)). Hence it is possible to define the additional moderator in the following two ways: One is to determine the mass M^* for the given values of κ, λ and $\mu^* s$ by solving the transcendental equation (Ref. (4))

$$\mu^* = 1 + X^* + \frac{\delta^* \eta}{2C_1} Y^* \quad . \tag{VI.4.3-1}$$

Another method is to determine the value s by calculating μ^* from Eq. (VI.4.3-1) for the beforehand given value of $\mu^* s$ and the fictitiously given value of M^* , for which the use of the main moderator mass is much simple and realistic.

Consequently, a heterogeneous system expressed by Eq. (VI.4.1-16) is equivalent, in the sense of the IRA, to a homogeneous system described by

$$\sigma_f \varphi = K(\sigma_{sf} \varphi) + \sigma_{cm} K_{cm}(\varphi) + sK^*(\varphi) \tag{VI.4.3-2}$$

where the slowing down kernel K^* corresponds to the fictitious mass M^* determined by Eq. (VI.4.3-1) or assumed beforehand. In other words, the primary cell can be effectively homogenized by introducing the fictitious moderator. Hereafter, the secondary cell with gross heterogeneity is treated. Each gross region which includes the primary cells can be considered to be homogeneous, because both the heterogeneity characteristics and the slowing down property in the primary cells are the same as in the infinite medium with the fictitious moderator.

The neutron balance in a multiregion secondary cell can be written by the following slowing down equation using collision probabilities :

$$V_i \Sigma_i \varphi_i = \sum_j P_{ji}(u) \{V_j S_j(u)\} \tag{VI.4.3-3}$$

Here, it should be noted that Eq. (VI.4.3-3) can be also solved by the subroutine 'PEACO' to give the exact solution instead of the following two-region formulation : We assume that each fuel and nonfuel region ,respectively, is homogeneous and has a flat flux. The latter assumption for the moderator region is completely reasonable for the NR moderators in which the slowing down can be accurately treated by the NR approximation.

From the above discussion, the averaged fluxes Φ_f and Φ_m can be assumed to satisfy the same equations as Eqs. (VI.4.1-8) and (VI.4.1-9), respectively :

$$\bar{\sigma}_f \Phi_f = p_{ff} \{\sigma_{cm} K_{cm}(\Phi_f) + K_f(\sigma_{sf} \Phi_f) + sK^*(\Phi_f)\} + (1 - p_{ff}) \bar{\sigma}_f \sum_k \{R_k K_k(\Phi_m)\} \tag{VI.4.3-4}$$

$$\bar{\sigma}_f \Phi_f + \bar{\sigma}_m \Phi_m = \sigma_{cm} K_{cm}(\Phi_f) + K_f(\sigma_{sf} \Phi_f) + sK^*(\Phi_f)$$

$$+ \bar{\sigma}_m \sum_k \{R_k K_k(\Phi_m)\} \qquad (VI.4.3\text{-}5)$$

where

$$\bar{\sigma}_f(u) = \sigma_f(u) + s = \sigma_a(u) + \sigma_{sf}(u) + \sigma_{cm} + s \qquad (VI.4.3\text{-}6)$$

$$\bar{\sigma}_m = \Sigma_m V_m / (N_f V_f) \qquad (VI.4.3\text{-}7)$$

$N_f = n_f v_f / (v_f + v_m)$ = averaged number density of resonance absorber

V_f, V_m = volumes of the lump and moderator regions of the secondary cells, respectively.

Similarly as in the last section, we use the rational approximation for the collision probability :

$$p_{ff} = \frac{\bar{\sigma}_f}{\bar{\sigma}_f + S} \qquad (VI.4.3\text{-}8)$$

with

$$S = \frac{G(1 - \bar{C})}{N_f L_f} = \frac{G(1 - \bar{C})(v_f + v_m)}{v_f n_f L_f} \qquad (VI.4.3\text{-}9)$$

where $G(\bar{C})$ and L_f are the Bell-Levine factor and the mean chord length of fuel rod, respectively, and $\bar{C}$ is the Dancoff factor for the secondary cell. These quantities can be estimated by applying the same procedure as used to obtain those for the heterogeneity of the primary cell. That is, the quantity $G(\bar{C})$ is assumed to be given by

$$G(\bar{C}) = \frac{\bar{a}}{1 + (\bar{a} - 1)\bar{C}}, \qquad (VI.4.3\text{-}10)$$

where the Dancoff factor, $\bar{C}$, is calculated by using an equation corresponding to Eq. (VI. 4.1-13') and the subroutine 'PIJ2', and the Bell-Levine factor, $\bar{a}$, and the mean chord length L_f are given in Table VI.4.1-1.

By following a procedure analogous to that used in obtaining Eqs. (VI.4.1-7) through (VI.4.1-17), we find that

$$\Phi_f^{(1)}(u) = \frac{\lambda\sigma_p + \kappa\sigma_{cm} + \mu^* s + \mu^{**} S}{\sigma_a + \lambda\sigma_{sf} + \kappa\sigma_{cm} + \mu^* s + \mu^{**} S} \qquad (VI.4.3\text{-}11)$$

with

$$\mu^{**} = \mu\bar{\sigma}_m / \{\mu\bar{\sigma}_m + (1 - \mu)\bar{\sigma}_m\} \quad \text{and} \quad \mu = \sum_k R_k \mu_k . \qquad (VI.4.3\text{-}12)$$

Here, the IR parameters (κ, λ, μ^*, μ^{**}) are determined by solving a coupled set of the transcendental equations.

Our scenario for homogenization of the double heterogeneity is as

follows: First,in a practical situation, the heterogeneity characteristics, s and S for the primary and secondary cells, are independently obtained by calculating the respective Dancoff factor. In the next stage, a fictitious moderator is determined for the infinite medium composed of the primary cell through the IR treatment to eliminate the heterogeneity. Finally, the secondary heterogeneity is homogenized using the fictitious moderator, also through the IR procedure, where each fuel lump is treated as homogeneous.

The simple results of Eq.(VI.4.3-11) reveal the following equivalence relation with practical usefulness : For the determined values of a set of IR parameters, we define a scattering cross section by

$$\sigma_b = \kappa\sigma_{cm} + \mu^* s + \mu^{**} S \qquad (VI.4.3\text{-}13)$$

Then, the IRA solution of Eq.(VI.4.3-11) becomes equivalent to that of the simple equation (VI.4.1-1).

VI.5 Collision Probability and Applications

VI.5.1 General theory

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

$$P_{ij} = \frac{1}{4\pi V_i}\int_{V_i} d\bar{r}\int_{V_j} d\bar{r}'\frac{\Sigma(\bar{r}')}{R^2}\exp\{-\int_0^R \Sigma(s)ds\} \qquad (VI.5\text{-}1)$$

where R denotes the distance from the emitting point to the collision point. We have also another expression of the Pij equivalent to Eq. (VI.5-1):

$$P_{ij} = \frac{1}{4\pi V_i}\int_{V_i} d\bar{r}\int_{4\pi} d\bar{\Omega}\int_{R_{j-}}^{R_{j+}} dR\Sigma_j\exp\{-\int_0^R \Sigma(s)ds\} \qquad (VI.5\text{-}2)$$

where Rj- and Rj+ denote respectively the distances from point r to the inner and outer boundary of the j-th region along the line through the points r and r'.

Similarly the directional probability Pijk which is called as the 'Behrens term' (Ref.(36)) is given by

$$P_{ijk} = \frac{1}{4\pi V_i}\int_{V_i} d\bar{r}\int_{4\pi} d\bar{\Omega}3\Omega_k^2\int_{R_{j-}}^{R_{j+}} dR\Sigma_j\exp\{-\int_0^R \Sigma(s)ds\} \qquad (VI.5\text{-}3)$$

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

We shall define a sub-region as a minimum spatial sub-division surrounded by a boundary surface . Several sub-regions which may be apart from each other, can be treated as in a same region by assigning a common region number to the sub-regions. The region is the spatial variable in the collision probability method. In the SRAC we can change the assignment by neutron energy range, so that we find T-region, R-region, and M-region. The correspondence between regions and sub-region is given by assigning the region number to each sub-region by the input and the geometrical disposition of the sub-regions is set implicitly in the program. Such hierachic configuration is introduced to save the computer time and storage. Therefore, when we consider the lattice cell system, the collision region j appears repeatedly in the neighbouring cells and even in a cell also. In the case where a neutron path enters into region j more than once, a sum of such term as shown by Eq. (VI.5-2) or Eq.(VI.5-3) is required for calculating the collision probabilities.

When we apply the isotropic (white) condition on the outer surface S of the cell, the probability that a neutron emitted from the region i

crosses S, Pis, is required.

$$P_{is} = \frac{1}{4\pi V_i}\int_{V_i} d\bar{r}\int d\bar{S}\,\frac{1}{R_s^2}\exp\{-\int_0^{R_s}\Sigma(s)ds\} \qquad (VI.5\text{-}4)$$

or as an alternative expression

$$P_{ij} = \frac{1}{4\pi V_i}\int_{V_i} d\bar{r}\int_{4\pi} d\bar{\Omega}\exp\{-\int_0^{R_s}\Sigma(s)ds\} \qquad (VI.5\text{-}5)$$

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

The isotropic boundary condition is frequently used for the lattice cell calculation not only in the collision probability method but also in the Sn calculation. We shall describe its physical meaning and the application in the collision probability method.

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

$$\Sigma_i\varphi V_i = \frac{\varphi}{4}SG_i + \sum_{j=1}^{N} P_{ji}\Sigma_j\varphi V_j \qquad (VI.5\text{-}6)$$

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

Σ_i ;The total macroscopic cross section,
φ ; the uniform scalar flux,
V_i ;the volume,
S ; the area of the surface,
G_i ;the probability that a neutron impinging on the surface has its first collision in the i-th region,
P_{ji} ;the probability that a neutron emitted in the j-th region has its first collision in the i-th region.

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

$$P_{ij}\Sigma_j\varphi V_j = P_{ij}\Sigma_i\varphi V_i \qquad (VI.5\text{-}7)$$

and the conservation theorem :

$$\sum_{j=1}^{N} P_{ji} + P_{is} = 1 \qquad (VI.5\text{-}8)$$

where Pis is the probability that a neutron emitted in the i-th region escapes from the outer surface S without suffering any collision, we

have

$$G_i = \frac{4V_i}{S}\Sigma_i P_{is} \qquad (VI.5\text{-}9)$$

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

$$G_s = 1-\sum_{i=1}^{N} G_i \qquad (VI.5\text{-}10)$$

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

$$P_{ij}(lattice) = P_{ij}(isolated)+P_{is}G_j+P_{is}G_sG_j+P_{is}G_s^2G_j+\ldots.$$

which can be rewritten as

$$P_{ij}(lattice) = P_{ij}(isolated)+P_{is}\frac{G_j}{1-G_s} \qquad (VI.5\text{-}11)$$

VI.5.2 Collision probabilities for slab lattice

In one-dimensional slab geometry shown in Fig.VI.5-1, we have

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

$$dr=dx,$$

$$d\Omega=2\pi\sin\theta d\theta.$$

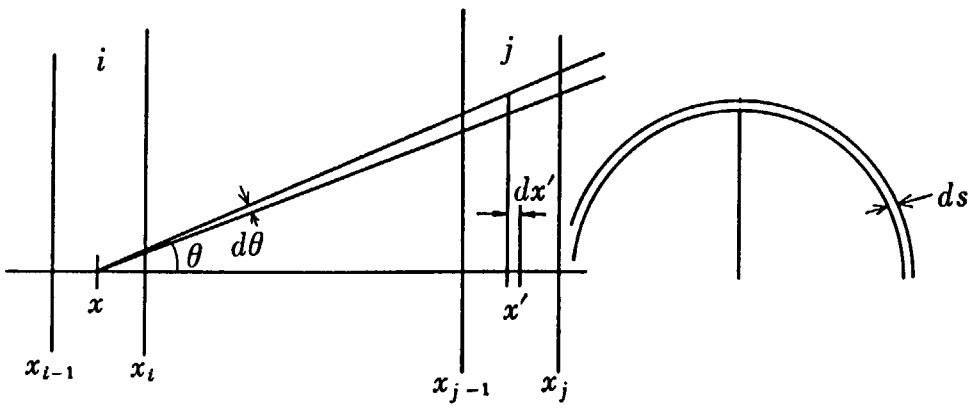


Fig.VI.5-1 Coordinate in slab geometry

We assume that the system is divided into an array of slabs. The i-th slab has its left edge at xi and its total cross section denoted by Σi. Then we have the Pij's as

$$P_{ij}=\frac{1}{2(x_i-x_{i-1})}\int_{x_{i-1}}^{x_i}dx\int_{x_{j-1}}^{x_j}dx'\int_0^{\pi/2}\Sigma(x')\frac{\sin\theta}{\cos\theta}* \\ *\exp\{\int_x^{x'}\Sigma(t)dt/\cos\theta\mid\}d\theta, \qquad (VI.5\text{-}12)$$

for the case xi < xj-1, and the optical distance which appears in the exponential term in Eq. (VI.5-12) is reduced to

$$\int_x^{x'}\Sigma(t)dt = \Sigma_i(x_i-x)+\Sigma_j(x'-x_{j-1})+\sum_{k=i+1}^{j-1}\lambda_k,$$

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

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

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

Now we introduce the Ein function defined by

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

We have the final form of Pij for the case xi < xj as follows

$$P_{ij} = \frac{1}{2\lambda_i}\{E_{i3}(\lambda_{ij})-E_{i3}(\lambda_{ij}+\lambda_i)-E_{i3}(\lambda_{ij}+\lambda_j)+E_{i3}(\lambda_{ij}+\lambda_i+\lambda_j)\}, \qquad (VI.5\text{-}13)$$

where

$$\lambda_{ij} = \sum_{k=i+1}^{j-1}\lambda_k, \quad \text{for} \quad x_i<x_j. \qquad (VI.5\text{-}14)$$

Next we shall consider the case where xi>xj, the optical distance is reduced to

$$\int_x^{x'}\Sigma(t)dt = \Sigma_i(x-x_{i-1})+\Sigma_j(x_j-x')+\sum_{k=j+1}^{i-1}\lambda_k,$$

by using the same procedure as xi < xj we get the same expression as Eq. (VI.5-13) except for the definition of λij. In this case we have

$$\lambda_{ij}=\sum_{k=j+1}^{i-1}\lambda_k, \quad \text{for} \quad x_i>x_j. \tag{VI.5-15}$$

In the last case where xi = xj, the optical distance in Eq. (VI.5-12) is reduced to

$$\left|\int_x^{x'}\Sigma(t)dt\right| = \begin{cases}\Sigma_i(x'-x), & \text{for} \quad x'>x.\\ \Sigma_i(x-x'), & \text{for} \quad x'<x.\end{cases}$$

Integrating over x and x' gives the final form of Pii as

$$P_{ii} = 1-\frac{1}{\lambda_i}\{E_{i3}(0)-E_{i3}(\lambda_i)\} \tag{VI.5-16}$$

If the λi's are so small that the differences in Eq. (VI.5-13) and in Eq. (VI.5-16) can not be obtained accurately in the numerical calculation, we should use the following differential forms instead of Eqs. (VI.5-13) and (VI.5-16),

$$P_{ij} = \frac{\lambda_j}{2}E_{i1}(\lambda_{ij}+\lambda_i/2+\lambda_j/2), \tag{VI.5-17}$$

$$P_{ii} = \lambda_i E_{i1}(\lambda_i/2). \tag{VI.5-18}$$

We, however, should take care of the property of the Ei1 function which has the logarithmic singularity.

We consider now the lattice cell system where a unit cell is divided into N regions and the collision region j lies periodically. A sum of such terms as Eq. (VI.5-13) gives

$$\begin{aligned}P_{ij} = \frac{1}{2\lambda_i}\sum_{l=0}\{&E_{i3}(\lambda_{ij}^{l1})-E_{i3}(\lambda_{ij}^{l1}+\lambda_i)-E_{i3}(\lambda_{ij}^{l1}+\lambda_j)+E_{i3}(\lambda_{ij}^{l1}+\lambda_i+\lambda_j)\\ &+E_{i3}(\lambda_{ij}^{l2})-E_{i3}(\lambda_{ij}^{l2}+\lambda_i)-E_{i3}(\lambda_{ij}^{l2}+\lambda_j)+E_{i3}(\lambda_{ij}^{l2}+\lambda_i+\lambda_j)\},\end{aligned} \tag{VI.5-19}$$

where

$$\begin{aligned}\lambda_{ij}^{l1} &= \sum_{k=i+1}^{j-1}\lambda_k + l*\sum_{k=1}^{N}\lambda_k,\\ \lambda_{ij}^{l2} &= \sum_{k=1}^{i-1}\lambda_k + \sum_{k=j+1}^{N}\lambda_k + \sum_{k=1}^{i-1}\lambda_k + l*\sum_{k=1}^{N}\lambda_k,\end{aligned} \tag{VI.5-20}$$

The summation over l is achieved separately by λ_{ij}^{l1} or λ_{ij}^{l2} series until λ_{ij}^{l1} or λ_{ij}^{l2} exceeds the fixed optical length of 6, respectively. For the

case where j<i, i and j in Eq.(VI.5-20) must be replaced by j and i, respectively.

For the case i=j we have

$$P_{ii} = 1-\frac{1}{\lambda_i}\{E_{i3}(0)-E_{i3}(\lambda_i)\}$$
$$+\frac{1}{\lambda_i}\sum_{l=0}\{E_{i3}(\lambda_{ij}^{l})-E_{i3}(\lambda_{ij}^{l}+\lambda_i)-E_{i3}(\lambda_{ij}^{l}+\lambda_j)+E_{i3}(\lambda_{ij}^{l}+\lambda_i+\lambda_j) \qquad (VI.5\text{-}21)$$

where

$$\text{where} \qquad \lambda_{ij}^{l} = (l+1)*\sum_{k=1}^{N}\lambda_k-\lambda_i. \qquad (VI.5\text{-}22)$$

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

Now we consider the explicit form of the directional probability. For the perpendicular direction to the boundary plane, we have $3\cos^2\theta$ as $3\Omega_\perp^2$, c.f. Fig.VI.5-1, by which the integrand in Eq.(VI.5-12) has to be multiplied. The similar procedure gives us the expression of $P_{ij\perp}$ corresponding to Eq. (VI.5-13) as follows:

$$P_{ij\perp}=\frac{3}{2\lambda_i}\{E_{i5}(\lambda_{ij})-E_{i5}(\lambda_{ij}+\lambda_i)-E_{i5}(\lambda_{ij}+\lambda_j)+E_{i5}(\lambda_{ij}+\lambda_i+\lambda_j)\}, \qquad (VI.5\text{-}23)$$

and corresponding to Eq. (VI.5-16)

$$P_{ii\perp} = 1-\frac{3}{2\lambda_i}\{E_{i5}(0)-E_{i5}(\lambda_i)\} \qquad (VI.5\text{-}24)$$

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

$$P_{ij} = \frac{1}{3}P_{ij\perp} + \frac{2}{3}P_{ij\parallel} \qquad (VI.5\text{-}25)$$

The relation comes from

$$1 = \sum_k \Omega_k^2 = \Omega_\perp^2 + 2\Omega_\parallel^2$$

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

At the end of this section we show the expression for the escape probability Pis:

$$P_{is}=\frac{1}{2\lambda_i}\{E_{i3}(\lambda^1_{is})-E_{i3}(\lambda^1_{is}+\lambda_i)+E_{i3}(\lambda^2_{is})-E_{i3}(\lambda^2_{is}+\lambda_i)\}, \qquad (VI.5\text{-}26)$$

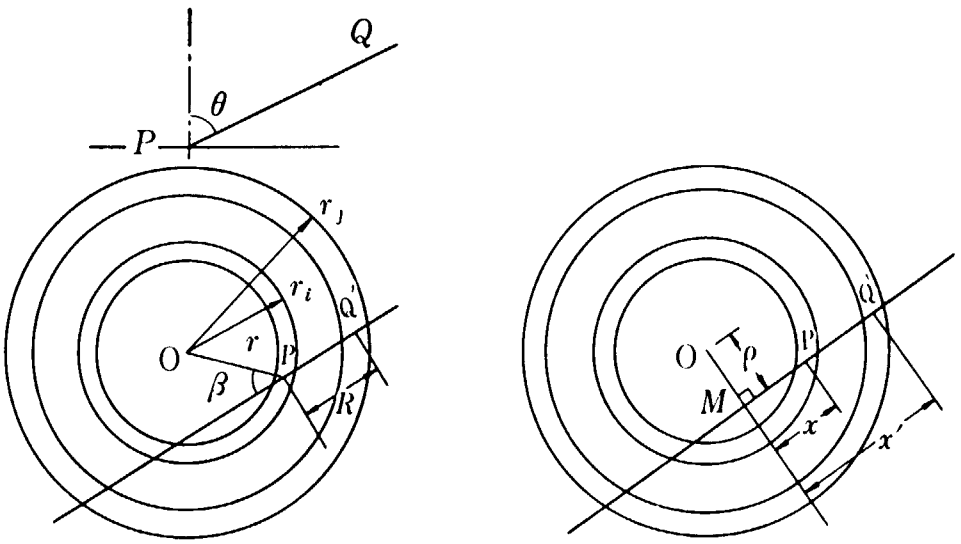
where

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

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

VI.5.3 Collision probabilities for annularly cylindrical lattice

We consider the infinitely long cylinder which is divided into several annular shells. The outer radius of the i-th shell is ri. We suppose that a neutron emitted at the point P in the i-th shell has it first collision at the point Q in the j-th shell. The position of P is defined by only the distance from the cylindrical axis;r. The line PQ makes an angle θ with the vertical line. We define the point Q' as the projection of the point Q on the horizontal cross section so that the line PQ' makes an angle β with the line PO. The distance between P and Q' is R. In the cylindrical coordinate system as shown in Fig.VI.5-2 **a**, we have the collision probability Pij as,



a **b**

Fig.VI.5-2 Cylindrical coordinate

$$P_{ij}=\frac{2}{V_i}\int_{r_{i-1}}^{r_i} r dr\int_0^{\pi} d\beta\int_0^{\pi/2}\sin\theta d\theta\int_{R_{j-}}^{R_{j+}} dR\frac{\Sigma_j}{\sin\theta}\exp\{-\int_0^R\frac{\Sigma(s)ds}{\sin\theta}\}, \qquad (VI.5\text{-}27)$$

where

$$V_i=\pi(r_i^2-r_{i-1}^2). \qquad (VI.5\text{-}28)$$

Then we transform the variables r, β ,R into new ones ρ, x and x' as illustrated in Fig.VI.5-2b. We define the perpendicular distance OM from O to the line PQ' as ρ, the distance between P and M as x, and the distance between Q' and M as x'. There are three relations among variables:

$$r^2 = \rho^2 + x^2,$$

$$r \sin\beta = \rho,$$

$$R = x'-x. \qquad (VI.5\text{-}29)$$

Using the relations we have the Jacobian

$$\frac{\partial (r,\beta,R)}{\partial (\rho,x,x')} = \frac{1}{r}, \qquad (VI.5\text{-}30)$$

Then we can rewrite Eq. (VI.5-27) by new variables:

$$P_{ij} = \frac{2}{V_i}\int_0^{r_i} d\rho \int_0^{\pi/2} d\theta \int_{x_{i-1}}^{x_i} dx \int_{x_{j-1}}^{x_j} dx' \Sigma_j \{\exp\{-|\int_x^{x'} \Sigma(t)dt|/\sin\theta\}$$
$$+\exp\{-|\int_{-x}^{x'} \Sigma(t)dt|/\sin\theta\}\}, \qquad (VI.5\text{-}31)$$

where

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

$$x_i = 0, \quad \text{for} \quad r_i<\rho.$$

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

For the case rj>ri, the optical distances which appears in the exponential terms in Eq. (VI.5-31) are reduced to

$$\int_x^{x'} \Sigma(t)dt = \Sigma_i (x_{i-1}-x)+\Sigma_j (x'-x_j')+\sum_{k=j+1}^{i-1} \lambda_k,$$

$$\int_{-x}^{x'} \Sigma(t)dt = \Sigma_i (x-x_{i-1})+\Sigma_j (x_j-x')+\sum_{k=1}^{i-1}\lambda_k+\sum_{k=1}^{i-1}\lambda_k,$$

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

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

$$P_{ij} = \frac{2}{\Sigma_i V_i}\int_0^{r_i} d\rho \int_0^{\pi/2} \sin\theta d\theta \{1-\exp(-\frac{\lambda_j}{\sin\theta})\}*$$

$$* \{exp(-\sum_{k=i+1}^{j-1} \lambda_k/sin\theta) + \exp(-2\sum_{k=1}^{i-1}\lambda_k+\lambda_i+\sum_{k=i+1}^{j-1} \lambda_k/\sin\theta)\} \}.$$

Now we introduce Kin function defined by

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

We have the final form of Pij for the case ri<rj, as follows:

$$P_{ij} = \frac{2}{\Sigma_i V_i}\int_0^{r_i} d\rho \{K_{i3}(\lambda^1_{ij})-K_{i3}(\lambda^1_{ij}+\lambda_i)-K_{i3}(\lambda^1_{ij}+\lambda_j)+K_{i3}(\lambda^1_{ij}+\lambda_i+\lambda_j)$$
$$+K_{i3}(\lambda^2_{ij})-K_{i3}(\lambda^2_{ij}+\lambda_i)-K_{i3}(\lambda^2_{ij}+\lambda_j)+K_{i3}(\lambda^2_{ij}+\lambda_i+\lambda_j)\},$$

(VI.5-33)

where

$$\lambda^1_{ij} = \sum_{k=i+1}^{j-1} \lambda_k,$$
$$\lambda^2_{ij} = \sum_{k=1}^{i-1}\lambda_k + \sum_{k=1}^{j-1}\lambda_k \quad \text{for} \quad r_i<r_j. \qquad (VI.5\text{-}34)$$

Next we consider Pij for the case ri>rj where the optical distances differ from the case ri<rj because Q is located inside of P so that x'< x. In this sense the symbol of the absolute value is required. We rewrite locally the optical distance as follows:

$$\left|\int_x^{x'} \Sigma(t)dt\right| = \int_{x'}^{x} \Sigma_i(t)dt = \Sigma_j(x_j-x') + (\Sigma_i(x-x_{i-1}) + \sum_{k=j+1}^{i-1} \lambda_k.$$

Integration over x and x' gives the final form of Pij for the case rj<ri, which is exactly same expression as in Eq. (VI.5-33) but λ^1_{ij} in Eq. (VI.5-34) must be replaced by

$$\lambda^1_{ij} = \sum_{k=j+1}^{i-1} \lambda_k \quad \text{for} \quad r_i>r_j. \qquad (VI.5\text{-}35)$$

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

$$\left|\int_x^{x'} \Sigma(t)dt\right| = \begin{cases} \Sigma_i(x'-x), & \text{for} \quad x'>x, \\ \Sigma_i(x-x'), & \text{for} \quad x'<x, \end{cases}$$

$$\int_{-x}^{x'} \Sigma(t)dt = \Sigma_j(x'-x_i) + (\Sigma_i(x-x_{i-1}) + 2\sum_{k=1}^{i-1}\lambda_k$$

In the integration over x' for the first term on R.H.S. of Eq.(VI.5 31), we must divide the range into (xi-1,x) and (x,xi) and then we have

$$P_{ii}=\frac{2}{\Sigma_i V_i}\int_0^{r_{i-1}} d\rho\int_0^{\pi/2} d\theta\ \{2\lambda_i\sin\theta-2\sin^2\theta\ \{1-\exp(-\frac{\lambda_i}{\sin\theta})\}$$
$$+\ \{1-\exp(-\frac{\lambda_i}{\sin\theta})\}^2\sin^2\theta\exp(-2\sum_{k=1}^{i-1}\frac{\lambda_k}{\sin\theta})\ \}$$
$$+\frac{2}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i} d\rho\int_0^{\pi/2} d\theta\{2\lambda_i\sin\theta-\sin^2\theta\ \{1-\exp(-2\frac{\lambda_i}{\sin\theta})\}\ \}$$

By using Kin function we get the final form of Pii as follows:

$$P_{ii}=\frac{2}{\Sigma_i V_i}\int_0^{r_{i-1}} d\rho\ \{2\lambda_i-2K_{i3}(0)+2K_{i3}(\lambda_i)$$
$$+K_{i3}(\lambda_{ii})-2K_{i3}(\lambda_{ii}+\lambda_i)+K_{i3}(\lambda_{ii}+2\lambda_i)\}$$
$$+\frac{2}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i} d\rho\ \{2\lambda_i-K_{i3}(0)+K_{i3}(2\lambda_i)\}\ , \qquad (VI.5\text{-}36)$$

where

$$\lambda_{ii}=2\sum_{k=1}^{i-1}\lambda_k . \qquad (VI.5\text{-}37)$$

If the λi's are so small that the differences in the brackets of Eq.(VI.5-33) and (VI.5-36) can not be obtained accurately in numerical calculation, we should use instead of Eqs.(VI.5-33) and (VI.5-36) the following differential forms:

$$P_{ij}=\frac{2}{\Sigma_i V_i}\int_0^{r_i} d\rho\lambda_i\lambda_j\ \{K_{i1}(\lambda_{ij}^1)+K_{i1}(\lambda_{ij}^2)\} \qquad (VI.5\text{-}38)$$

$$P_{ii}=\frac{1}{\Sigma_i V_i}\int_0^{r_{i-1}} d\rho\ \{\lambda_i^2K_{i1}(\frac{\lambda_i}{2})+\lambda_i^2K_{i1}(\lambda_{ii})\}+\frac{2}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i} d\rho\lambda_i^2K_{i1}(\lambda_i) \qquad (VI.5\text{-}39)$$

If we assume the cylindricalized cell with the perfect reflecting outer boundary, more terms like those in Eq.(VI.5-35) are required as follows:

$$K_{i3}(\lambda_{ij}^3)-K_{i3}(\lambda_{ij}^3+\lambda_i)-K_{i3}(\lambda_{ij}^3+\lambda_j)+K_{i3}(\lambda_{ij}^3+\lambda_i+\lambda_j)$$
$$+K_{i3}(\lambda_{ij}^4)-K_{i3}(\lambda_{ij}^4+\lambda_i)-K_{i3}(\lambda_{ij}^4+\lambda_j)+K_{i3}(\lambda_{ij}^4+\lambda_i+\lambda_j)$$
$$+\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots$$
$$+K_{i3}(\lambda_{ij}^n)-K_{i3}(\lambda_{ij}^n+\lambda_i)-K_{i3}(\lambda_{ij}^n+\lambda_j)+K_{i3}(\lambda_{ij}^n+\lambda_i+\lambda_j)$$

$$+\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots$$

where

$$\lambda^3_{ij} = \lambda^1_{ij} + \lambda_j + 2\sum_{k=i+1}^{N} \lambda_k,$$

$$\lambda^4_{ij} = \lambda^2_{ij} + \lambda_j + 2\sum_{k=i+1}^{N} \lambda_k,$$

$$\lambda^n_{ij} = \lambda^{n-2}_{ij} + \lambda_j + 2\sum_{k=i+1}^{N} \lambda_k.$$

As regards the directional probability in the cylindrical coordinate, we know for the axial direction $3\Omega_z^2=3\cos^2\theta$ and for the radial direction $3\Omega_r^2=(3/2)\sin^2\theta$. For the latter Pijr is obtained by multiplying the integrand in Eq. (VI.5-27) by $(3/2)\sin^2\theta$. It is tedious and meaningless, however, to write here the whole expressions for each condition. It is enough for us to know only the fact that all the terms expressed by Kin function must be replaced by $(3/2)K_{i(n+2)}(x)$. Similarly to the slab system the following relation holds:

$$P_{ij} = \frac{1}{3} P_{ijz} + \frac{2}{3} P_{ijr} \qquad (VI.5\text{-}40)$$

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

$$P_{is} = \frac{2}{\Sigma_i V_i}\int_0^{r_n} d\rho \,\{K_{i3}(\lambda^1_{is}) - K_{i3}(\lambda^1_{is}+\lambda_i) + K_{i3}(\lambda^2_{is}) - K_{i3}(\lambda^2_{is}+\lambda_i)\}, \qquad (VI.5\text{-}41)$$

where

$$\lambda^1_{is} = \sum_{k=i+1}^{N} \lambda_k,$$

$$\lambda^2_{is} = \sum_{k=1}^{i-1} \lambda_k + \sum_{k=1}^{N} \lambda_k. \qquad (VI.5\text{-}42)$$

VI.5.4 Collision probabilities for spherical system

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

$$P_{ij} = \frac{1}{4\pi V_i} \int_{V_i} dV \int_{4\pi} d\bar{\Omega} \int_{R\subset V_j} dR\Sigma_j \exp\{-\int_0^R \Sigma(s)ds\} \qquad (VI.5\text{-}43)$$

The integrand on R.H.S. of Eq. (VI.5-43) is interpreted as follows by seeing Fig.VI.5-3a. A neutron emitted at a point P in the region i moves toward the point Q which is in distance R from point P, has the exponential decay by the optical length $\int_0^R \Sigma(s)ds$ and suffers its collision at the layer of thickness dR in region j of the cross section Σ_j. In the spherically symmetric system the position of the point P is defined only by the distance r from the center C of the system. The position of point Q is defined at the distance R from the point P, and the lines PQ and PC make an angle θ (see Fig.VI.5-3a). In this coordinate system,

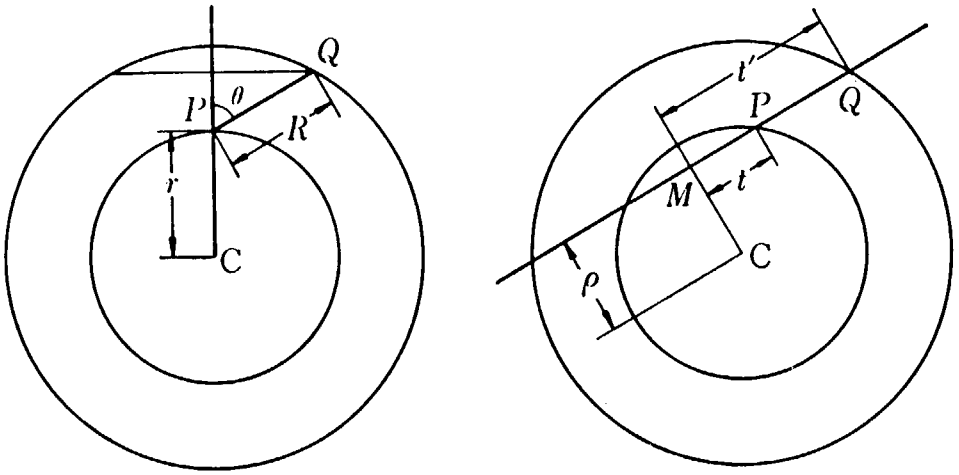


Fig.VI.5-3 Spherical coordinate

$dV = 4\pi r^2 dr, \quad 0 < r < R_N,$

$d\bar{\Omega} = 2\pi \sin\theta d\theta, \quad 0 \leq \theta \leq \pi.$

and Eq. (VI.5-43) is rewritten by the triple integral:

$$P_{ij} = \frac{4\pi\Sigma_j}{V_i} \int_0^{R_n} r^2 dr \int_0^{2\pi} \sin\theta d\theta \int_{R\subset V_j} dR \exp\{-\int_0^R \Sigma(s)ds\}. \qquad (VI.5\text{-}44)$$

To perform analytically the integration as far as possible, the coordinate shown in Fig.VI.5-3a is transformed into the new coordinate shown in Fig.VI.5-3b where the perpendicular length CM is ρ. The positions of points P and Q are defined by the distances t and t', respectively, from the point M. The following relations among variables are found:

$$r^2 = t^2 + \rho^2,$$

$$r \sin\theta = \rho,$$

$$R = t' - t. \qquad (VI.5\text{-}45)$$

The Jacobian is then obtained as follows:

$$\frac{\partial(r,\theta,R)}{\partial(\rho,t,t')} = -\frac{1}{r}. \qquad (VI.5\text{-}46)$$

The probability is rewritten by the new variables as

$$P_{ij} = \frac{2\pi\Sigma_j}{V_i}\int_0^{R_n}\rho d\rho\int_{t\subset V_i}dt\int_{t'\subset V_j}dt'\exp\{-\int_0^{t-t'}\Sigma(s)ds\}\ . \qquad (VI.5\text{-}47)$$

As the nuclear cross section in each shell is uniform, we can integrate Eq.(VI.5-47) over t and t'. Finally the shell-to-shell collision probabilities are given in the form of single integral (see Fig.VI.5-4);

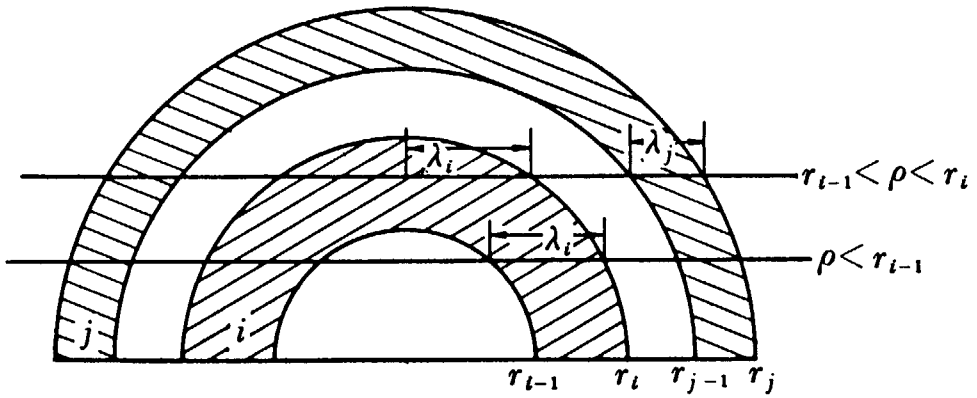


Fig.VI.5-4 Neutron paths in case i<j

$$P_{ij} = \frac{2\pi}{\Sigma_i V_i}\int_0^{r_{i-1}}\rho d\rho\{1-\exp(-\lambda_i)\}\{1-\exp(-\lambda_j)\}$$
$$*\{\exp\{-\sum_{k=i+1}^{j-1}\lambda_k\}+\exp\{-2\sum_{k=i}^{i-1}\lambda_k-\lambda_i-\sum_{k=i+1}^{j-1}\lambda_k\}\}$$
$$+\frac{2\pi}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i}\rho d\rho\{\{1-\exp(-2\lambda_i)\}\{1-\exp(-\lambda_j)\}\exp\{-\sum_{k=i+1}^{j-1}\lambda_k\}\},$$
$$\text{for}\quad i<j, \qquad (VI.5\text{-}48)$$

$$P_{ii} = \frac{2\pi}{\Sigma_i V_i}\int_0^{r_{i-1}}\rho d\rho\{\{\lambda_i-1+\exp(-\lambda_i)\}\{1-\exp(-\lambda_j)\}^2\{\exp\{-2\sum_{k=1}^{i-1}\lambda_k\}\}$$
$$+\frac{2\pi}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i}\rho d\rho\{2\lambda_i-1+\exp(-2\lambda_i)\},$$
$$\text{for}\quad i=j, \qquad (VI.5\text{-}49)$$

where

$$t_i = \sqrt{r_i^2-\rho^2} \quad \text{for} \quad r_i \geqq \rho,$$

$$t_i = 0 \quad \text{for} \quad r_i < \rho, \tag{VI.5-50}$$

$$\lambda_i = \Sigma_i (t_i - t_{i-1}). \tag{VI.5-51}$$

For i>j, the similar expression as Eq. (VI.5-48) can easily be obtained but the reciprocity theorem gives Pij directly from Pij.

Now we have the escape probability Pis as

$$P_{is} = \frac{2\pi}{\Sigma_i V_i}\int_0^{r_{i-1}} \rho d\rho \{1-\exp(-\lambda_i)\} * \{\exp\{-\sum_{k=i+1}^{N}\lambda_k\} \{1+\exp(-2\sum_{k=i}^{i-1}\lambda_k - \lambda_i)\}\} + \frac{2\pi}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i} \rho d\rho \{1-\exp(-2\lambda_i)\}\exp(-\sum_{k=i+1}^{N}\lambda_k). \tag{VI.5-52}$$

In our computer code, the integrands in Eqs. (VI.5-48) and (VI.5-49) with possible pairs of (i,j) are firstly calculated for a fixed ρ. Then, the integration over ρ is accomplished by changing the value of ρ .

VI.5.5 Collision probabilities for two-dimensional cylindrical lattice

In the cylindrical system with general shape of its cross section and of infinite height, the collision probability from a region i to another region j, Pij, is expressed by the following Eq.(VI.5-53) in the coordinate system of Fig.VI.5-5, assuming the flat flux in each region and an isotropic emission in the laboratory system.

$$P_{ij} = \{\int_{-\infty}^{\infty}\rho d\rho\int_0^{2\pi}d\varphi\int_0^{\pi/2}\sin\theta d\theta\int_{AB}^{AC}dt\exp\{-\frac{\Sigma_i|AC-t|}{\sin\theta}\}\int_{AD}^{AE}dt'\frac{\Sigma_j}{\sin\theta} * \exp\{-\frac{\Sigma_j|t'-AD|}{\sin\theta}\}\exp\{-|\int_{AC}^{AD}\Sigma(s)ds|/\sin\theta\}\} / \int_{-\infty}^{\infty}\rho d\rho\int_0^{2\pi}d\varphi\int_0^{\pi/2}\sin\theta d\theta\int_{AB}^{AC}dt \tag{VI.5-53}$$

In Fig.VI.5-5 the line PQ' defined by ρ and φ is the projection of the neutron path PQ on the horizontal plane. The points P and Q are, respectively, the source and collision positions. The point A is the origin of measures of t, t' and s. The points B, C, D and E are the points of intersection of the line PQ' with the region boundaries. A restriction on the moving direction of a neutron is imposed so that a neutron moves only to the positive direction of t along the line PQ'. If the line PQ' enters region j more than once, a sum of Eq.(VI.5-53) is required.

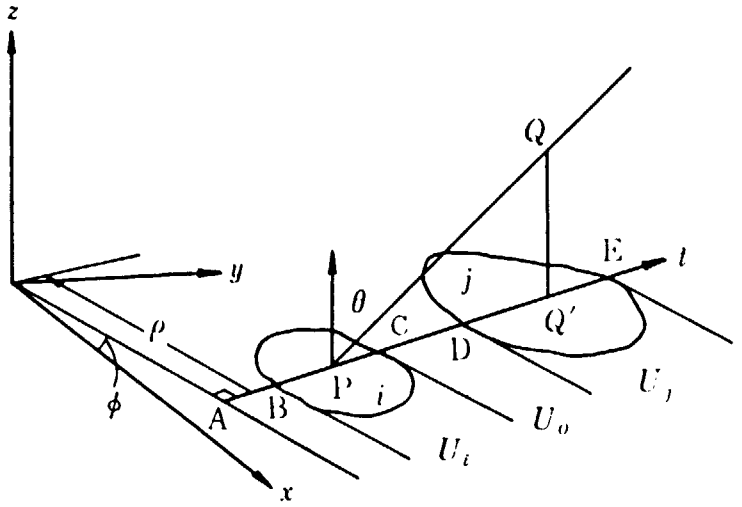


Fig.VI.5-5 Cylindrical coordinate system

The self collision probability, Pii is expressed by the following Eq. (VI.5-54), where the point Q is in region i.

$$P_{ii} = \int_{-\infty}^{\infty}\rho d\rho\int_0^{2\pi}d\varphi\int_0^{\pi/2}\sin\theta d\theta\int_{AB}^{AC}dt\int_t^{AC}dt'\frac{\Sigma_i}{\sin\theta}\exp\{-\frac{\Sigma_i\mid t-t'\mid}{\sin\theta}\}$$
$$/\int_{-\infty}^{\infty}\rho d\rho\int_0^{2\pi}d\varphi\int_0^{\pi/2}\sin\theta d\theta\int_{AB}^{AC}dt \qquad (VI.5\text{-}54)$$

If the line PQ' reenters region i, a sum of such a term as Eq. (VI.5-53) is required for obtaining Pii.

The six-fold integrals of Eqs. (VI.5-53) and (VI.5-54) are reduced to the double integrals as follows:

$$P_{ij} = \frac{1}{2\pi\Sigma_i V_i}\int_{-\infty}^{\infty}d\rho\int_0^{2\pi}d\varphi\{K_{i3}(U_0)-K_{i3}(U_0+U_i)$$
$$-K_{i3}(U_0+U_j)+K_{i3}(U_0+U_i+U_j)\}, \qquad (VI.5\text{-}55)$$

$$P_{ii} = \frac{1}{2\pi\Sigma_i V_i}\int_{-\infty}^{\infty}d\rho\int_0^{2\pi}d\varphi\{U_i-K_{i3}(0)+K_{i3}(U_i)\}, \qquad (VI.5\text{-}56)$$

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

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

$$P_{is} = \frac{1}{2\pi\Sigma_i V_i}\int_{-\infty}^{\infty}d\rho\int_0^{2\pi}d\varphi\{K_{i3}(U_{is})-K_{i3}(U_{is}+U_i)\}, \qquad (VI.5\text{-}57)$$

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

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

$$P_{ijr} = \frac{3}{4\pi\Sigma_i V_i}\int_{-\infty}^{\infty} d\rho \int_0^{2\pi} d\varphi \{K_{i5}(U_0) - K_{i5}(U_0+U_i) - K_{i5}(U_0+U_j) + K_{i5}(U_0+U_i+U_j)\}, \quad (VI.5\text{-}58)$$

$$P_{iir} = \frac{1}{4\pi\Sigma_i V_i}\int_{-\infty}^{\infty} d\rho \int_0^{2\pi} d\varphi \{2U_i - 3K_{i3}(0) + 3K_{i3}(U_i)\}. \quad (VI.5\text{-}59)$$

Thus we have the double integration of the linear combination of Kin function as a final form of the collision probability for the two-dimensional cylindrical system.

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

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

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

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

where LO = the number of regions in the base cell, LLL = the total number of regions on the line, W = the weight of the line, T(L) = the path length and II(L) = the region identification number. The source region is picked up sequentially from the first LO regions in the table. If the L-th region is set to the source region (the region identification number is II(L)), the collision region is picked up sequentially from the L-th region to the LLL-th region until the optical path length between the source and the collision exceeds the limited value (which is provisionally set to 7.0). Then the source region is set to the (L+1)-th region. If the vacuum boundary condition or the isotropic boundary condition is adopted, the table is cut off at

LLL=L0, and Pis is calculated.

If a neutron path traverses a region which is empty or nearly empty, the integrands of Eqs. (VI.5-55) and (VI.5-56) tend to zero but Pij/Σi which is used for the diffusion coefficient defined by Ref. ((36)) has a finite non zero value, and the forms of the integrands bring the growth of the truncated error. To avoid this growth of error, the subroutine DELT integrates the symmetric elements Δij defined by

$$\Delta_{ij} = \frac{1}{2\pi}\int_{-\infty}^{\infty} d\rho \int_0^{2\pi} d\varphi \{K_{i3}(U_0) - K_{i3}(U_0+U_i) - K_{i3}(U_0+U_j) + K_{i3}(U_0+U_i+U_j)\}/\Sigma_i\Sigma_j, \quad (VI.5\text{-}60)$$

$$\Delta_{ii} = \frac{1}{2\pi}\int_{-\infty}^{\infty} d\rho \int_0^{2\pi} d\varphi \{U_i - K_{i3}(0) + K_{i3}(U_i)\}/\Sigma_i\Sigma_i, \quad (VI.5\text{-}61)$$

$$\Delta_{is} = \frac{1}{2\pi}\int_{-\infty}^{\infty} d\rho \int_0^{2\pi} d\varphi \{K_{i3}(U_{is}) - K_{i3}(U_{is}+U_i)\}/\Sigma_i, \quad (VI.5\text{-}62)$$

The Δij relates with Pij as

$$P_{ij} = \frac{\Delta_{ij}\Sigma_j}{V_i}, \quad (VI.5\text{-}63)$$

$$P_{is} = \frac{\Delta_{is}}{V_i}, \quad (VI.5\text{-}64)$$

The reciprocity theorem

$$\Delta_{is} = \Delta_{ji} \quad (VI.5\text{-}65)$$

follows immediately from Eq. (VI.5-60).

If a neutron path traverses an empty or a nearly empty region, the integrand of Δij is reduced to the differential form of Bickley function as Eq. (VI.5-66), Eq. (VI.5-67), Eq. (VI.5-68), and Δii and Δis are reduced to the forms as Eqs. (VI.5-69) and (VI.5-70), respectively:

$$\Delta_{ij} = \frac{1}{2\pi}\int_{-\infty}^{\infty} d\rho \int_0^{2\pi} d\varphi T_j \frac{\{K_{i2}(U_0) - K_{i3}(U_0+U_i)\}}{\Sigma_i} \quad \text{for small } \Sigma_j \quad (VI.5\text{-}66)$$

$$\Delta_{ij} = \frac{1}{2\pi}\int_{-\infty}^{\infty} d\rho \int_0^{2\pi} d\varphi T_i \frac{\{K_{i2}(U_0) - K_{i3}(U_0+U_j)\}}{\Sigma_j} \quad \text{for small } \Sigma_i \quad (VI.5\text{-}67)$$

$$\Delta_{ij} = \frac{1}{2\pi}\int_{-\infty}^{\infty} d\rho \int_0^{2\pi} d\varphi T_i T_j \; K_{i1}(U_0) \quad \text{for small } \Sigma_i, \Sigma_j \quad (VI.5\text{-}68)$$

$$\Delta_{ii} = \frac{1}{4\pi}\int_{-\infty}^{\infty} d\rho \int_0^{2\pi} d\varphi T_i T_i \; K_{i1}(0) \quad \text{for small } \Sigma_i \quad (VI.5\text{-}69)$$

$$\Delta_{is} = \frac{1}{2\pi}\int_{-\infty}^{\infty} d\rho \int_{0}^{2\pi} d\varphi T_i \; K_{i2}(U_{is}) \qquad \text{for small } \Sigma_i \qquad (VI.5\text{-}70)$$

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

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

$$P_{ij} = \frac{\Delta_{ij}}{V_i}, \qquad (VI.5\text{-}71)$$

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

$$X_i \left\{ \sum_{j=1}^{N} P_{ij}^{\#}\Sigma_j + P_{is} \right\} = 1 \qquad (VI.5\text{-}72)$$

where X_i is a normalization factor to satisfy Eq.(VI.5-72). After finding X_i , $P_{ij}^{\#}$ for j=i NR are normalized. Such a procedure is repeated for i=1 N, and then, if required, $P_{ij}^{\#}$ is replaced by Pij. It should be noted that if X_i deviates from unity the reciprocity relation does not hold but it affects to the flux only slightly.

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

$$a_m = \frac{y_{m-1} - 2y_{m-1/2} + y_m}{2\Delta x^2} \qquad (VI.5\text{-}73)$$

$$b_m = \frac{y_m - y_{m-1}}{\Delta x} - a_m(x_{m-1} + x_m) \qquad (VI.5\text{-}74)$$

$$c_m = y_{m-1} - b_m x_{m-1} - a_m x_{m-1}^2 \qquad (VI.5\text{-}75)$$

$$y_m = K_{in}(x_m) \qquad (VI.5\text{-}76)$$

$$\Delta x = (x_m - x_{m-1}) \qquad (VI.5\text{-}77)$$

The pitch and range of the tabulation of **a**, **b** and **c** is as follows:

Δx/x :0.01/(0.0, 0.48), 48 points,
0.04/(0.48, 2.4), 48 points,
0.1/(2.4, 9.6), 72 points,
∞ /(9.6 ∞),1 points,

where x>9.6, Kin(x) is assumed to be zero.

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

$$K_{in}(x) = (a_{nm}x + b_{nm})x + c_{nm} \qquad (VI.5\text{-}78)$$

where $x_{m-1} \leqq x_i \leqq x_m$

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

VI. 6 Collapsing of Energy Structure for Few Group Calculation

Description will be given on the collapsing methods of the multi-group energy structure into the few group. Shown in the following table is the nomenclature used for the description.

Table VI.6-1 Nomenclature

Symbols	Meaning
g , g'	Multi-group number
G , G'	Few group number
i , j	A) For cell calculation: T- or R-region number, which is used for describing the neutron balance equation in cell. The cell calculation is made by the T- and R-region representation, respectively, in the thermal and resonance energy region. B) For Sn and/or diffusion calculation: Zone number.
I	X-region number for representing effective cross section (usually I=1, except for the case where a supercell model is used.)
M (*Note*: $i,j \in M \in I$)	M-region number for representing materials. Different material numbers must be assigned to the same material if the effective cross sections are expected to be different.
z	Nuclear reaction for process z (fission , capture, scattering etc.) This symbol is also used to express the total cross section in the meanings of Sect.VI.2.
n	Nuclide
l	Order of Legendre expansion
$\chi^n{}_g$	Fission spectrum (n = U-235)
$\sigma^n{}_{z,M,g}$, $\nu\sigma^n{}_{f,M,g}$	Effective multi-group cross section of M'th M-region (z = fission and capture) , which is obtained by the interpolation of the self-shielding factors in the U-library, except for the 2nd. resonance energy region where the ultrafine group calculation or IR method can be used. This cross section is mainly used for burn-up calculation by being collapsed as the following few-group cross section and can be also used for activation calculation.

Table VI.6-1 (Cont'd)

Symbols	Meanings
$\sigma^n_{z,M,G}$, $\nu\sigma^n_{f,M,G}$	Effective few-group cross section of M'th M-region (z = fission and capture).
$\Sigma_{z,M,g}$, $\nu\Sigma_{f,M,g}$	Macroscopic multi-group cross section of M'th M-region. This cross section is composed of the microscopic cross sections given by the same procedure as used to obtain $\sigma^n_{z,M,g}$.
$\Sigma_{tr,M,g}$	Transport cross section of M-region. (See Sect.VI.3-(A)-(ii))
$\Sigma_{sl,M,g\to g'}$ $(l = 0,1)$	Total energy transfer cross section of M-region.
$\Sigma_{z,i,g}$, $\Sigma_{tr,i,g}$, $\nu\Sigma_{f,i,g}$	Multi-group cross section of region i, which is asigned from the above M-region cross section by the correspondence of M-region to T- or R-region.
$\Sigma_{sl,i,g\to g'}$ $(l = 0,1)$	Total energy transfer cross section of region i.
$\overline{\Sigma}_{z,I,g}$, $\nu\overline{\Sigma}_{f,I,g}$	Effective multi-group cross section of I'th X-region. Usually one X-region is asigned to each lattice cell unless a supercell model is used.
$\overline{\Sigma}_{sl,I,g\to g'}$ $(l = 0,1)$	Total energy transfer cross section of I'th X-region.
$\overline{\Sigma}_{tr,I,g}$	Effective transport cross section of I'th X-region (See Eq.(VI.3-15)).
$\overline{D}_{I,g}$	Effective diffusion coefficient. (See Sect. VI.3-(B)).
$\overline{\chi}_G$	Few group fission spectrum.
$\overline{\Sigma}_{z,I,G}$, $\nu\overline{\Sigma}_{f,I,G}$	Few group cross section of I'th X-region.
$\overline{\Sigma}_{sl,I,G\to G'}$ $(l = 0)$	Few group energy transfer cross section of I'th X-region.
$\overline{\Sigma}_{tr,I,G}$	Few group transport cross section.
$\overline{D}_{I,G}$	Few group diffusion coefficient.

Table VI.6-1 (Cont'd)

Symbols	Meanings
$\Phi_g(r) = \int_{\Delta u_g} \Phi(r,u)du$	Space dependent flux of group g.
$\bar{\Phi}_{i,g} = \int_{V_i} \Phi_g(r)dr/V_i$	Average flux of region i and of group g.

$$\Phi_{l,g} = \sum_{i\in l} V_i\bar{\Phi}_{i,g} = \int_{V_l} \Phi_g(r)dr, \qquad \Phi_{l,G} = \sum_{g\in G} \Phi_{l,g} \tag{VI.6-1}$$

$$\Phi_{M,g} = \sum_{i\in M} V_i\bar{\Phi}_{i,g} = \int_{V_m} \Phi_g(r)dr, \qquad \Phi_{M,G} = \sum_{g\in G} \Phi_{M,g} \tag{VI.6-2}$$

(A) Region Smearing and Group Collapsing of Cross Section

Smeared- and collapsed- group cross sections for the few group calculations are obtained from the following prescriptions. Another description will be given for the transport cross section and diffusion coefficient in the next subsection.

$$\sigma^n_{z,M,G} = \sum_{g\in G} \sigma^n_{z,M,g}\ \Phi_{M,g} \ /\ \Phi_{M,G} \qquad (z = f \text{ or } c) \tag{VI.6-3}$$

$$\nu\sigma^n_{f,M,G} = \sum_{g\in G} \nu\sigma^n_{f,M,g}\ \Phi_{M,g}\ /\Phi_{M,G} \tag{VI.6-4}$$

$$\bar{\Sigma}_{z,l,g} = \sum_{i\in l} \bar{\Sigma}_{z,i,g}\ V_i\bar{\Phi}_{i,g}\ /\Phi_{l,g} \tag{VI.6-5}$$

$$\nu\bar{\Sigma}_{f,l,g} = \sum_{i\in l} \nu\bar{\Sigma}_{f,i,g}V_i\bar{\Phi}_{i,g}\ /\Phi_{l,g} \tag{VI.6-6}$$

$$\bar{\Sigma}_{sl,l,g\to g'} = \sum_{i\in l} \bar{\Sigma}_{sl,i,g\to g'}V_i\bar{\Phi}_{i,g}\ /\Phi_{l,g} \tag{VI.6-7}$$

$$\bar{\chi}^n_G = \sum_{g\in G} \bar{\chi}^n_g \tag{VI.6-8}$$

$$\bar{\Sigma}_{z,l,G} = \sum_{g\in G} \bar{\Sigma}_{z,l,g}\ \Phi_{l,g}\ /\Phi_{l,G} \tag{VI.6-9}$$

$$\overline{\nu\Sigma}_{f,I,G} = \sum_{g\in G} \nu\Sigma_{f,I,g} V_i \overline{\Phi}_{I,g} \,/\Phi_{I,g} \qquad (VI.6\text{-}10)$$

$$\overline{\Sigma}_{sl,I,G\to G'} = \sum_{g\in G} \sum_{g'\in G'} \overline{\Sigma}_{sl,I,g\to g'} \Phi_{I,g} \,/\, \Phi_{I,G} \qquad (VI.6\text{-}11)$$

(B) Group collapsing of diffusion coefficient

The multi-group diffusion coefficients in homogeneous medium and lattice cell are given by the prescriptions, respectively, in Sect.VI. 3-(A) and (B). As mentioned there, the diffusion coefficient in homogeneous system is directly obtained from the transport cross section, while the one substituted for a heterogeneous lattice cell is formulated using the collision probability. Hence, the multi-group diffusion coefficients thus obtained are assumed to be given for each homogenized zone (X-region).

The group-dependent diffusion equation in the I'th zone can be written as

$$-\overline{D}_{I,g}\nabla^2\Phi_g(\boldsymbol{r}) + \overline{\Sigma}_{t,I,g}\,\Phi_g(\boldsymbol{r}) = S_g(\boldsymbol{r}). \qquad (VI.6\text{-}12)$$

Let us sum up the above equation over g to collapse the multigroup equations into the few group and denote each few group by G, then we have for the diffusion term

$$-\sum_{g\in G} \overline{D}_{I,g}\nabla^2\Phi_g \sim B^2 \sum_{g\in G} \overline{D}_{I,g}\Phi_g(\boldsymbol{r}) = B^2\overline{D}_{I,G}\Phi_G \sim -\overline{D}_{I,G}\nabla^2\Phi_G \qquad (VI.6\text{-}13)$$

where

$$\Phi_G \equiv \sum_{g\in G} \Phi_g \qquad (VI.6\text{-}14)$$

and

$$D_{I,G} \equiv \sum_{g\in G} \overline{D}_{I,g}\Phi_g / \Phi_G. \qquad (VI.6\text{-}15)$$

Hence, the few group transport cross section can be defined by

$$\Sigma_{tr,I,G} = 3\Phi_G \Big/ \sum_{g\in G} \overline{D}_{I,g}\,\Phi_g = \Phi_G \Big/ \sum_{g\in G} \frac{\Phi_g}{\overline{\Sigma}_{tr,I,g}}. \qquad (VI.6\text{-}16)$$

As the quantities $D_{I,G}$ and $\Sigma_{tr,I,G}$ still have the space dependence in the I'th zone, the volume-integrated flux of Eq.(VI.6-1) are used in Eqs.(VI.6-15 and -16) to give their effective values $\overline{D}_{I,G}$ and $\overline{\Sigma}_{tr,I,G}$.

VI.7 Burn-up calculations

The SRAC installs a burn-up routine to be called after each cell-calculations to return the change of nuclide densities due to burn-up during a exposure step under a given power level, and fixed material temperatures. This process is , so called, cell burn-up, then the process for core burn-up under development utilizes the tabulated sets of macroscopic cross sections prepared for possible range of burn-up, material temperatures,and various cell compositions.

As seen in Sect.II.11, the input for cell burn-up is minimized to save the work to prepare the data for many daughter nuclei and their chains. All necessary information is stored in a burn-up library so that the user can choose one of F.P. chain models prepared in separate members.

We shall describe here the formulation for birth and decay of individual nuclide along the chain under a fixed power level.

We use explicit form for the solution of the equations as used in the CITATION code (Ref.(10)) to avoid accumulation of numerical error which happens by improper selection of time mesh used in the numerical method such as the Runge-Kutter-Gill method. On the other hand, the given condition 'fixed power' which means the neutron fluxes are time-dependent makes the following equations not to hold exactly, then we have to divide an exposure step into several sub-steps so as to be able to assume the neutron fluxes are time-independent.

The differential equation expressing the chain relationship between nuclide due to nuclear reactions is written as

$$\frac{dN_n(t)}{dt} = -A_n(t)N_n(t) + Y_n(t) + \sum_j G_{j,n-1\to n}(t)N_{n-1}(t) \qquad (VI.7\text{-}1)$$

where
$N_n(t)$: a nuclide concentration at some location at the time t,
$A_n(t)$: the specific loss rate of nuclide n at time t,
$Y_n(t)$: the direct yield rate from neutron fission to a nuclide n,
$G_{j,n-1\to n}(t)$: the specific generation rate from a precursor n-1 to n along the chain j,

The explicit form of solutions of Eq. (VI.7-1) chosen for an exposure time θ is

$$N_n(t) = N_n(t-\theta)\exp(-A_n\theta) + \frac{Y_n(\theta)\{1-exp(-A_n\theta)\}}{A_n} + \sum_j \sum_{i=1}^{n-1} \{N_i(t-\theta)Q_{j,n,i}(\theta) + Y_{j,i}(\theta)U_{j,n,i}(\theta)\}. \qquad (VI.7\text{-}2)$$

That is, the nuclide concentration at the end of an exposure step is dependent on the start-of-the-step nuclide concentrations back along the chains, the yield rate of nuclides along the chains and certain factors Q and U. These factors Q and U are independent of nuclide concentrations and yield rates and therefore may be obtained once and applied directly to obtain spatial behavior (within the approximation

of exposure to the constant neutron fluxes). These factors are given by

$$Q_{j,n,i}(\theta) = \sum_{m=i}^{n-1} \frac{\exp(-A_m\theta) - \exp(-A_n\theta)}{A_n - A_m} * G_{j,m\to m+1}(\theta) * \prod_{\substack{k=i \\ k\neq m}}^{n-1} \frac{G_{j,k\to k+1}(\theta)}{A_k - A_m} \qquad (VI.7\text{-}3)$$

$$U_{j,n,i}(\theta) = \frac{1-\exp(-A_n\theta)}{A_n} * \prod_{m=i}^{n-1} \frac{G_{j,m\to m+1}(\theta)}{A_m} - \sum_{m=i}^{n-1} \frac{\exp(-A_m\theta) - \exp(-A_n\theta)}{A_m(A_n - A_m)} * G_{j,m\to m+1}(\theta) * \prod_{\substack{k=i \\ k\neq m}}^{n-1} \frac{G_{j,i\to k+1}(\theta)}{A_k - A_m}. \qquad (VI.7\text{-}4)$$

In the above equations the term An represents the total specific loss rate of a nuclide, neglecting time change of neutron flux,

$$A_n = \lambda_n + 10^{-24} \sum_g \Phi_{g,z}\sigma_{a,n,g} \qquad (VI.7\text{-}5)$$

where $\Phi_{g,z}$ denotes neutron flux integrated in spatial region z and energy width g, and the generation rate from a precursor is given by

$$G_{j,n\to n+1}(\theta) = \begin{cases} 10^{-24} \sum_g \Phi_{g,z}\sigma_{c,n,g} & \textit{by capture} \\ \lambda_{j,n} & \textit{by decay} \\ H_n\lambda_{j,n} & \textit{by partial decay} \\ 10^{-24} E_n \sum_g \Phi_{g,z}\sigma_{c,n,g} & \textit{by partial capture} \\ 10^{-24} \sum_g \Phi_{g,z}\sigma_{n2n,n,g} & \textit{by n-2n reaction} \end{cases} \qquad (VI.7\text{-}6)$$

depending on the chain route specifications, where Hn and En denote the fraction of the reaction to the nuclide n.

It is assumed that F.P. nuclides follow in order all fissile nuclides, and strong absorbers. The end-of-step concentrations are calculated by processing the chain relationship specified. When the first F.P. nuclide is encountered, all the yield rates in Eq. (VI.7-2) are calculated as

$$Y_n(\theta) = 10^{-24} \sum_m Y_{m\to n} \frac{N_m(t) + N_m(t+\theta)}{2} \sum_g \Phi_{g,z}\sigma_{f,n,g} \qquad (VI.7\text{-}7)$$

where $Y_{m\to n}$ is the specified fission yield (fraction) of product nuclide n from fissile nuclide m, and the arithmetic average of the end-point values of the fissile nuclide concentrations are used as shown in Eq. (VI.7-7).

For a non-depleting, one-nuclide chain, the equation solved is

$$N_n(t) = N_n(t-\theta) + \theta Y_n(\theta). \qquad (VI.7\text{-}8)$$

The code has provision for a strong absorbing, one-nuclide chain (e.g. burnable poison), to be solved as,

$$N_n(t) = N_n(t-\theta)\exp(-A_n\theta) \qquad (VI.7\text{-}9)$$

For all exposure calculations, reaction rates are based on the neutron flux available at the start of an exposure time step. Each time step may be sub-divided into the set number (stored in the Burn-up Library by time step) of intervals to obtain renormalization of the neutron flux after each interval to the desired power level.

When more than one spatial regions contain any burnable nuclide, the above process is repeated by region in a sub-step and the flux renormalization is done to cover all regions. The effective microscopic cross sections are supplied for each region.

VII Dictionaries

VII.1 Element Tag

Z	Element Name	Tag	Z	Element Name	Tag	Z	Element Name	Tag
1	Hydrogen	HO	35	Bromine	BR	71	Lutetium	LU
1	Deuterium	DO	36	Krypton	KR	72	Hafnium	HF
1	Tritium	TO	37	Rubidium	RB	73	Tantal	TA
2	Helium	HE	38	Strontium	SR	74	Tungsten	WO
3	Lithium	LI	39	Yttrium	YO	75	Rhenium	RE
4	Beryllium	BE	40	Zirconium	ZR	76	Osmium	OS
5	Boron	BO	41	Niobium	NB	77	Iridium	IR
6	Carbon	CO	42	Molybdenum	MO	78	Platinium	PT
7	Nitrogen	NO	43	Technetium	TC	79	Gold	AU
8	Oxygen	OO	44	Ruthenium	RU	80	Mercury	HG
9	Fluorine	FO	45	Rhodium	RH	81	Thallium	TL
10	Neon	NE	46	Palladium	PD	82	Lead	PB
11	Sodium	NA	47	Silver	AG	83	Bismuth	BI
12	Magnesium	MG	48	Cadmium	CD	84	Polonium	PO
13	Aluminium	AL	49	Indium	IN	85	Astatine	AT
14	Silicon	SI	50	Tin	SN	86	Radon	RN
15	Phosphorus	PO	51	Antimony	SB	87	Francium	FR
16	Sulfur	SO	52	Tellurium	TE	88	Radium	RA
17	Chlorine	CL	53	Iodine	IO	89	Actinum	AC
18	Argon	AR	54	Xenon	XE	90	Thorium	TH
19	Potassium	KO	55	Cesium	CS	91	Protactinium	PA
20	Calcium	CA	56	Barium	BA	92	Uranium	UO
21	Scandium	SC	57	Lanthane	LA	93	Neptunium	NP
22	Titan	TI	58	Cerium	CE	94	Plutonium	PU
23	Vanadium	VO	59	Praseodymium	PR	95	Americium	AM
24	Chromium	CR	60	Neodymium	ND	96	Curium	CM
25	Manganese	MN	61	Promethium	PM	97	Berkelium	BK
26	Iron	FE	62	Samarium	SM	98	Californium	CF
27	Cobalt	CO	63	Europium	EU	99	Einsteinium	ES
28	Nickel	NI	64	Gadolinium	GD	100	Fermium	FM
29	Copper	CU	65	Terbium	TB	101	Mendelevium	MD
30	Zinc	ZN	66	Dysprosium	DY	102	Nobelium	NO
31	Garium	GA	67	Holmium	HO	103	Lawrencium	LR
32	Germanium	GE	68	Erbium	ER	104	Kurchatovium	KU
33	Arsenic	AS	69	Thulium	TM	**		
34	Selenium	SE	70	Ytterbium	YB	**		

VII.2 Compound symbol

Compound	Key code	Chemical symbol	Remarks
Beryllium metal	B	Be	
Beryllium oxide	E	BeO	O scattering included
Benzene	Q	C6H6	C scattering included
Graphite	C	C	
Polyethylene	P	(CH2)n	Add C comp. by free
Uranium metal	U	U	
Uranium carbite	V	UC	Not yet compiled
Uranium oxide	W	UO2	U scattering included
Water light	H	H2O	Add O comp. by free
Water heavy	D	D2O	Add O comp. by free
Zirconium hydrate	Z	ZrH	Both Zr H comps. given
Simple	0	Free atom	

VII.3 Nuclides in SRAC public libraries

Nuclide	Key code	Thermal scatter	Thermal F-TAB	Fast F-TAB	Reso lib	Orig File Mat No.	vol
H-001	XH01000A	free P1				ENDFB 1269	4
H-001H	XH01H008	S(αβ)P1				ENDFB 1002	3
H-001Q	XH01Q00A	S(αβ)P1				ENDFB 1095	3
H-001Z	XH01A00A	S(αβ)P1				ENDFB 1097	3
H-001P	XH01P004	S(αβ)P1				ENDFB 1114	3
D-002	XD02000A	free P1				ENDFB 1120	4
D-002D	XD02D008	S(αβ)P1				ENDFB 1004	3
T-003	XT03000A	free				ENDFB 1169	4
HE003	XHE3000A	free				ENDFB 1146	4
LI006	XLI6000A	free		13-17		ENDFB 1271	4
LI007	XLI7000A	free		14-17		ENDFB 1272	4
BE009	XBE9000A	free P1		5-12		ENDFB 1289	4
BE009E	XBE9E00A	S(αβ)P1				ENDFB 1099	3
BE009B	XBE9B00A	S(αβ)P1				ENDFB 1064	3
B-010	XB00000A	free				ENDFB 1273	4
B-011	XB01000A	free				ENDFB 1160	4
C-012	XC02000A	free P1		1- 7		ENDFB 1274	4
C-012C	XC02C00A	S(αβ)P1				ENDFB 1065	3
C-012P	XC02P00A	S(αβ)P1				ENDFB 1114	3
C-012Q	XC02Q001	cap				ENDFB 1095	3
N-014	XN04000A	free P1		3-13		ENDFB 1275	4
O-016	XO06000A	free 10		1-13		ENDFB 1276	4
O-016E	XO06E001	cap				ENDFB 1099	3
O-016W	XO06W00A	S(αβ)				ENDFB 1167	3
F-019	XF090001	free				ENDFB 1277	4
NA023	XNA3000A	free		7-35		ENDFB 1156	4
MG000	XMGN000A	free		2-26		ENDFB 1280	4
AL027	XAL7000A	free		6-30		ENDFB 1193	4
SI000	XSIN0001	cap				ENDFB 1194	4
S-000						not yet	
K-000	XKON0001	free				ENDFB 1150	4
CA000	XCAN0001	cap				ENDFB 1195	4
TI000	XTIN0001	free				ENDFB 1286	4

V-000	XVON0001	free				ENDFB 1196	4
CR000	XCRN000A	free		6-35		ENDFB 1191	4
MN055	XMN50001	cap				ENDFB 1197	4
FE000	XFEN000A	free		6-37		ENDFB 1192	4
CO059	XCO90001	free				ENDFB 1199	4
NI000	XNIN000A	free		9-34		ENDFB 1190	4
CU000	XCUN0001	cap				ENDFB 1295	4
CU063	XCU30001	cap		not yet fast		ENDFB 6411	4
KR083	XKR30001	cap				ENDFB 1184	4
ZR000	XZRN000A	free		20-44		ENDFB 1284	4
ZR000Z	XZRNZ00A	S($\alpha\beta$)P1				ENDFB 1096	3
NB093	XNB30001	free				ENDFB 0236	3
MO000	XMON000A	free		35-55		ENDFB 1287	4
MO099	XMO90001	cap				ENDFB 0269	4
TC-99	XTC90001	cap				ENDFB 1137	4
RU101	XRU10001	cap				ENDFB 0310	4
RU103	XRU30001	cap				ENDFB 0312	4
RU105	XRU50001	cap				ENDFB 0314	4
RH103	XRH30001	cap				ENDFB 0330	4
RH105	XRH50001	cap				ENDFB 0334	4
PD105	XPD50001	cap				ENDFB 0359	4
PD107	XPD70001	cap				ENDFB 0361	4
PD108	XPD80001	cap				ENDFB 0363	4
AG107	XAG70001	cap			1	ENDFB 1138	4
AG109	XAG9000A	free			1	ENDFB 1139	4
CD000	XCDN000A	free				ENDFB 1281	4
CD113	XCD30001	cap				ENDFB 1282	4
IN113	XIN3000A	free		49-65	1	ENDFB 0445	4
IN115	XIN5000A	free		39-69	1	ENDFB 0449	4
I-131	XI010001	cap				ENDFB 0570	4
I-135	XI050001	cap				ENDFB 0576	4
SB121	XSB10001	free				ENDFB 0511	4
SB123	XSB30001	free				ENDFB 0514	4
XE131	XXE10001	cap				ENDFB 0592	4
XE133	XXE30001	cap				ENDFB 0433	S
XE135	XXE50001	cap				ENDFB 1294	4
CS133	XCS30001	cap				ENDFB 0613	4
CS134	XCS40001	cap				TAKAN00434	S
CS135	XCS50001	cap				ENDFB 0616	4
PR143	XPR30001	cap				ENDFB 0695	4
ND143	XND30001	cap				ENDFB 0714	4
ND145	XND50001	cap				ENDFB 0716	4
ND147	XND70001	cap				ENDFB 0718	4
PM147	XPM70001	cap				ENDFB 0733	4
PM148g	XPMG0001	cap				ENDFB 0734	4
PM148m	XPMM0001	cap				ENDFB 0735	4
PM149	XPM90001	cap				ENDFB 0736	4
PM151	XPM10001	cap				ENDFB 0738	4
SM147	XSM70001	cap				ENDFB 0753	4
SM148	XSM80001	cap				ENDFB 0754	4
SM149	XSM90001	cap				ENDFB 1027	4
SM150	XSM00001	cap				ENDFB 0756	4
SM151	XSM10001	cap				ENDFB 0757	4
SM152	XSM20001	cap				ENDFB 0758	4
EU153	XEU30001	cap				ENDFB 0776	4
EU154	XEU40001	cap				ENDFB 0777	4
EU155	XEU50001	cap				ENDFB 0778	4
EU156	XEU60001	cap				ENDFB 0779	4

GD000	XGDN0009	free		39-63		:ENDFB 1030		4
GD154	XGD40001	cap				:ENDFB 0791		4
GD155	XGD50001	cap				:ENDFB 0792		4
GD156	XGD60001	cap				:ENDFB 0793		4
GD157	XGD70001	cap				:ENDFB 0794		4
GD158	XGD80001	cap				:ENDFB 0795		4
DY164	XDY40001	cap				:ENDFB 1031		4
LU175	XLU50001	cap	1 -48			:ENDFB 1032		4
LU176	XLU60001	cap	1 -48			:ENDFB 1033		4
HF000	XHFN0001	cap				:TAKAN00872		S
HF174	XHF40001	cap				:TAKAN00874		S
HF176	XHF60001	cap				:TAKAN00876		S
HF177	XHF70001	cap				:TAKAN00877		S
HF178	XHF80001	cap				:TAKAN00878		S
HF179	XHF90001	cap				:TAKAN00879		S
HF180	XHF00001	cap				:TAKAN00880		S
W-182	XW020001	cap				:ENDFB 1128		4
W-183	XW030001	cap				:ENDFB 1129		4
W-184	XW040001	cap				:ENDFB 1130		4
W-186	XW060001	cap				:ENDFB 1131		4
AU197	XAU7000A	free		35-61		:ENDFB 1283		4
PB000	XPBN000A	free		10-35		:ENDFB 1288		4
TH232	XTH2000A	free		23-74	11	:ENDFB 1296		4
PA233	XPA3000A	free		50-70		:ENDFB 1297		4
U-233	XU03000A	free	1 -48	51-74	11	:ENDFB 1260		4
U-234	XU04000A	free		38-74	11	:ENDFB 1043		4
U-235	XU05000A	free	1 -48	26-74	11	:ENDFB 1261		4
U-236	XU06000A	free		38-74	11	:ENDFB 1163		4
U-238	XU08000A	free		23-74	11	:ENDFB 1262		4
U-238W	XU08W00A	S(αβ)				:ENDFB 1167		3
PU239	XPU9000A	free	1 -48	26-74	11	:ENDFB 1264		4
PU240	XPU0000A	free	1 -48	26-74	11	:ENDFB 1265		4
PU241	XPU1000A	free	1 -48	23-74	11	:ENDFB 1266		4
PU242	XPU2000A	free		29-74	11	:ENDFB 1161		4
U-233RF	XF3R0001	cap				:ENDFB 1042		2
U-235RF	XF5R0001	cap				:ENDFB 1045		2
PU239RF	XF9R0001	cap				:ENDFB 1052		2
U-233SF	XF3S0001	cap				:ENDFB 1066		2
U-235SF	XF5S0001	cap				:ENDFB 1068		2
PU239SF	XF9S0001	cap				:ENDFB 1070		2
U-233NF	XF3N0001	cap				:ENDFB 1067		2
U-235NF	XF5N0001	cap				:ENDFB 1069		2
PU239NF	XF9N0001	cap				:ENDFB 1071		2
pseudo	XPSD0001	cap				:TAKAN00400		S
pseu25	XPS20001	cap				:VSOP 163		
pseu40	XPS40001	cap				:VSOP 160		

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Note 1. in key code column
n on the eighth character : the temperature dependent matrices are prepared on the first n temperatures.

Note 2. in Thermal scatter column
free : scattering matrices calculated by free gas model
S(αβ) : scattering matrices calculated from tabulated S(αβ) scattering law
cap : only capture cross sections with no scattering assumed P1
: scattering matrices for P0 and P1 are prepared

Note 3. in Thermal FTAB column
n n' : shielding factor tabulation starting from n to n'

Note 4. in Fast FTAB column
n n' : shielding factor tabulation starting from n to n'

Note 5. in Res lib column
n : ultra-fine cross sections for n temperature points are stored in MCROSS library:

Note 5. in Orig File and Mat No. column
ENDFB nnnn m: material no. in ENDF/B version m
TAKANOnnnn : material no. in TAKANO evaluated file
VSOP nnn : material no. in VSOP code

VII.4 Energy Group Structure of SRAC Public Libraries

Fast Energy Group Structure in Public Library
(m=1.67482E-24 gram, eV=1.60210E-12 erg)

Group	Energy Range (eV) Upper	Lower	Upper Velocity (m/sec)	Lethargy Upper	Width	
1	0.10000E+8	0.77880E+7	0.43740E+8	0.0	0.25	
2	0.77880E+7	0.60653E+7	0.38600E+8	0.25	0.25	
3	0.60653E+7	0.47237E+7	0.34065E+8	0.50	0.25	
4	0.47237E+7	0.36788E+7	0.30062E+8	0.75	0.25	
5	0.36788E+7	0.28651E+7	0.26529E+8	1.00	0.25	
6	0.28651E+7	0.22313E+7	0.23412E+8	1.25	0.25	
7	0.22313E+7	0.17377E+7	0.20661E+8	1.50	0.25	
8	0.17377E+7	0.13534E+7	0.18233E+8	1.75	0.25	
9	0.13534E+7	0.10540E+7	0.16091E+8	2.00	0.25	
10	0.10540E+7	0.82085E+6	0.14200E+8	2.25	0.25	*
11	0.82085E+6	0.63928E+6	0.12532E+8	2.50	0.25	
12	0.63928E+6	0.49787E+6	0.11059E+8	2.75	0.25	
13	0.49787E+6	0.38774E+6	0.97596E+7	3.00	0.25	
14	0.38774E+6	0.30197E+6	0.86129E+7	3.25	0.25	
15	0.30197E+6	0.23518E+6	0.76008E+7	3.50	0.25	
16	0.23518E+6	0.18316E+6	0.67077E+7	3.75	0.25	
17	0.18316E+6	0.14264E+6	0.59195E+7	4.00	0.25	
18	0.14264E+6	0.11109E+6	0.52240E+7	4.25	0.25	
19	0.11109E+6	0.86517E+5	0.46101E+7	4.50	0.25	
20	0.86517E+5	0.67380E+5	0.40684E+7	4.75	0.25	**
21	0.67380E+5	0.52475E+5	0.35904E+7	5.00	0.25	
22	0.52475E+5	0.40868E+5	0.31685E+7	5.25	0.25	
23	0.40868E+5	0.31828E+5	0.27962E+7	5.50	0.25	
24	0.31828E+5	0.24788E+5	0.24676E+7	5.75	0.25	
25	0.24788E+5	0.19305E+5	0.21777E+7	6.00	0.25	
26	0.19305E+5	0.15034E+5	0.19218E+7	6.25	0.25	
27	0.15034E+5	0.11709E+5	0.16960E+7	6.50	0.25	
28	0.11709E+5	0.91188E+4	0.14967E+7	6.75	0.25	
29	0.91188E+4	0.71017E+4	0.13208E+7	7.00	0.25	
30	0.71017E+4	0.55308E+4	0.11656E+7	7.25	0.25	
31	0.55308E+4	0.43074E+4	0.10287E+7	7.50	0.25	
32	0.43074E+4	0.33546E+4	0.90779E+6	7.75	0.25	
33	0.33546E+4	0.26126E+4	0.80112E+6	8.00	0.25	
34	0.26126E+4	0.20347E+4	0.70699E+6	8.25	0.25	
35	0.20347E+4	0.15846E+4	0.62391E+6	8.50	0.25	
36	0.15846E+4	0.12341E+4	0.55060E+6	8.75	0.25	
37	0.12341E+4	0.96112E+3	0.48590E+6	9.00	0.25	
38	0.96112E+3	0.74852E+3	0.42881E+6	9.25	0.25	
39	0.74852E+3	0.58295E+3	0.37842E+6	9.50	0.25	
40	0.58295E+3	0.45400E+3	0.33396E+6	9.75	0.25	
41	0.45400E+3	0.35357E+3	0.29472E+6	10.00	0.25	
42	0.35357E+3	0.27536E+3	0.26009E+6	10.25	0.25	
43	0.27536E+3	0.21445E+3	0.22952E+6	10.50	0.25	
44	0.21445E+3	0.16702E+3	0.20255E+6	10.75	0.25	
45	0.16702E+3	0.13007E+3	0.17875E+6	11.00	0.25	***
46	0.13007E+3	0.10130E+3	0.15775E+6	11.25	0.25	
47	0.10130E+3	0.78893E+2	0.13921E+6	11.50	0.25	
48	0.78893E+2	0.61442E+2	0.12286E+6	11.75	0.25	
49	0.61442E+2	0.47851E+2	0.10842E+6	12.00	0.25	

50	0.47851E+2	0.37266E+2	0.95680E+5	12.25	0.25
51	0.37266E+2	0.29023E+2	0.84437E+5	12.50	0.25
52	0.29023E+2	0.22603E+2	0.74516E+5	12.75	0.25
53	0.22603E+2	0.17604E+2	0.65760E+5	13.00	0.25
54	0.17604E+2	0.13710E+2	0.58033E+5	13.25	0.25
55	0.13710E+2	0.10677E+2	0.51214E+5	13.50	0.25
56	0.10677E+2	0.83153E+1	0.45196E+5	13.75	0.25
57	0.83153E+1	0.64760E+1	0.39885E+5	14.00	0.25
58	0.64760E+1	0.50435E+1	0.35199E+5	14.25	0.25
59	0.50435E+1	0.39279E+1	0.31063E+5	14.50	0.25
60	0.39279E+1	0.30590E+1	0.27413E+5	14.75	0.25
61	0.30590E+1	0.23824E+1	0.24192E+5	15.00	0.25
62	0.23824E+1	0.18554E+1	0.21349E+5	15.25	0.25
63	0.18554E+1	0.16374E+1	0.18841E+5	15.50	0.125
64	0.16374E+1	0.14450E+1	0.17699E+5	15.625	0.125
65	0.14450E+1	0.12752E+1	0.16627E+5	15.750	0.125
66	0.12752E+1	0.11253E+1	0.15619E+5	15.875	0.125
67	0.11253E+1	0.99312E+0	0.14673E+5	16.000	0.125
68	0.99312E+0	0.87643E+0	0.13784E+5	16.125	0.125
69	0.87643E+0	0.77344E+0	0.12949E+5	16.250	0.125
70	0.77344E+0	0.68256E+0	0.12164E+5	16.375	0.125
71	0.68256E+0	0.60236E+0	0.11427E+5	16.500	0.125
72	0.60236E+0	0.53158E+0	0.10735E+5	16.625	0.125
73	0.53158E+0	0.46912E+0	0.10085E+5	16.750	0.125
74	0.46912E+0	0.41399E+0	0.94736E+5	16.875	0.125

* Lower boundary of Fast Fission Range
** Lower boundary of Smooth Range
*** Lower boundary of Resonance I Range

Thermal Energy Group Structure in Public library
(m=1.67482E-24 gram, eV=1.60210E-12 erg)

Group	Energy Range (eV)		Upper Velocity	Lethargy	
	Upper	Lower	(m/sec)y	Upper	Width
1	0.39279E+1	0.30590E+1	0.27413E+5	14.75	0.25
2	0.30590E+1	0.23824E+1	0.24192E+5	15.00	0.25
3	0.23824E+1	0.18554E+1	0.21349E+5	15.25	0.25
4	0.18554E+1	0.16374E+1	0.18841E+5	15.50	0.125
5	0.16374E+1	0.14450E+1	0.17699E+5	15.625	0.125
6	0.14450E+1	0.12752E+1	0.16627E+5	15.750	0.125
7	0.12752E+1	0.11254E+1	0.15619E+5	15.875	0.125
8	0.11254E+1	0.99312E+0	0.14673E+5	16.000	0.125
9	0.99312E+0	0.87642E+0	0.13784E+5	16.125	0.125
10	0.87642E+0	0.77344E+0	0.12949E+5	16.250	0.125
11	0.77344E+0	0.68256E+0	0.12164E+5	16.375	0.125
12	0.68256E+0	0.60236E+0	0.11427E+5	16.500	0.125
13	0.60236E+0	0.53158E+0	0.10735E+5	16.625	0.125
14	0.53158E+0	0.46912E+0	0.10085E+5	16.750	0.125
15	0.46912E+0	0.41399E+0	0.94737E+4	16.875	0.125
16	0.41399E+0	0.38926E+0	0.88996E+4	17.000	0.0616
17	0.38926E+0	0.36528E+0	0.86297E+4	17.0616	0.0636
18	0.36528E+0	0.34206E+0	0.83597E+4	17.1252	0.0657
19	0.34206E+0	0.31961E+0	0.80896E+4	17.1909	0.0678
20	0.31961E+0	0.29792E+0	0.78196E+4	17.2587	0.0703
21	0.29792E+0	0.27699E+0	0.75496E+4	17.3290	0.0729

22	0.27699E+0	0.25683E+0	0.72796E+4	17.4019	0.0755
23	0.25683E+0	0.23742E+0	0.70097E+4	17.4774	0.0786
24	0.23742E+0	0.21878E+0	0.67396E+4	17.5560	0.0818
25	0.21878E+0	0.20090E+0	0.64696E+4	17.6378	0.0230
26	0.20090E+0	0.18378E+0	0.61996E+4	17.7230	0.0891
27	0.18378E+0	0.16743E+0	0.59296E+4	17.8121	0.0932
28	0.16743E+0	0.15183E+0	0.56597E+4	17.9053	0.0987
29	0.15183E+0	0.13700E+0	0.53896E+4	18.0031	0.1028
30	0.13700E+0	0.12293E+0	0.51196E+4	18.1059	0.1083
31	0.12293E+0	0.10963E+0	0.48496E+4	18.2142	0.1145
32	0.10963E+0	0.97080E-1	0.45797E+4	18.3287	0.1216
33	0.97080E-1	0.85397E-1	0.43096E+4	18.4503	0.1282
34	0.85397E-1	0.74276E-1	0.40420E+4	18.5785	0.1396
35	0.74276E-1	0.64017E-1	0.37696E+4	18.7181	0.1486
36	0.64017E-1	0.54520E-1	0.34996E+4	18.8667	0.1606
37	0.54520E-1	0.45785E-1	0.32296E+4	19.0273	0.1746
38	0.45785E-1	0.37813E-1	0.29596E+4	19.2019	0.1913
39	0.37813E-1	0.30602E-1	0.26897E+4	19.3932	0.2116
40	0.30602E-1	0.24154E-1	0.24196E+4	19.6048	0.2366
41	0.24154E-1	0.18467E-1	0.21497E+4	19.8414	0.2685
42	0.18467E-1	0.13543E-1	0.18796E+4	20.1099	0.3101
43	0.13543E-1	0.93805E-2	0.16097E+4	20.4200	0.3672
44	0.93805E-2	0.59804E-2	0.13396E+4	20.7872	0.4502
45	0.59804E-2	0.33423E-2	0.10696E+4	21.2374	0.5818
46	0.33423E-2	0.14663E-2	0.79965E+3	21.8192	0.8239
47	0.14663E-2	0.35238E-3	0.52965E+3	22.6431	1.4258
48	0.35238E-3	0.10000E-4	0.25965E+3	24.0689	3.5621
48L	0.10000E-4	**********	0.43738E+2	27.6310	******

****

Energy Group Structure for Ultra Fine Resonance Calculation

The energy group structure in the ultra fine group calculation performed by PEACO routine for the Resonance II range is prepared in the MCROSS library file as follows;

****

Upper boundary energy 130.07 eV
Lower boundary energy 0.4139 eV
Equal lethargy width 0.00125
Number of ultra fine groups on cross section table 4600
Number of fine energy groups on flux calculation 460

****

VII.5 Burn-up Chains

We have installed four burn-up libraries for the optional use of burn-up chain schemes. They are stored in the following files.

J1480.BURN.DATA(ENDFB2) for the Garrison model J1480.BURN.DATA(IIJIMA) for the Iijima model
J1480.BURN.DATA(VSOP25) for the VSOP 25 model
J1480.BURN.DATA(VSOP40) for the VSOP 40 model

The differences among them are in the F.P. chain models.

First we shall describe common information among libraries.

1) Nuclide Table

Maximum number of heavy nuclide : 22
Maximum number of absorber : 20
Maximum number of F.P. nuclide : 67

NO	NUC	FIS	RES	DECAY CONST.	WATT/FISS.
1	TH8	0	0	0.0	3.10800E-11
2	TH0	1	0	0.0	3.10800E-11
3	TH2	1	2	0.0	3.10800E-11
4	PA1	0	0	0.0	3.10800E-11
5	PA2	0	0	0.0	3.10800E-11
6	PA3	1	2	0.0	3.10800E-11
7	U02	0	0	0.0	3.10800E-11
8	U03	1	2	0.0	3.10800E-11
9	U04	1	2	0.0	3.10800E-11
10	U05	1	2	0.0	3.10800E-11
11	U06	1	2	0.0	3.10800E-11
12	U08	1	2	0.0	3.10800E-11
13	PU8	1	2	0.0	3.22000E-11
14	PU9	1	2	0.0	3.22000E-11
15	PU0	1	2	0.0	3.22000E-11
16	PU1	1	2	1.68000E-09	3.22000E-11
17	PU2	1	2	1.68000E-09	3.22000E-11
18	NP7	0	0	0.0	0.0
19	NP8	0	0	0.0	0.0
20	NP9	0	0	0.0	0.0
21	AM3	0	0	0.0	0.0
22	CM4	0	0	0.0	0.0
23	B00	0	2	0.0	0.0
24	AG7	0	0	0.0	0.0
25	IN3	0	2	0.0	0.0
26	IN5	0	2	0.0	0.0
27	HF6	0	0	0.0	0.0
28	HF7	0	0	0.0	0.0
29	HF8	0	0	0.0	0.0
30	HF9	0	0	0.0	0.0
31	HF0	0	0	0.0	0.0
32	DMY	0	0	0.0	0.0
33	DMY	0	0	0.0	0.0
34	DMY	0	0	0.0	0.0
35	DMY	0	0	0.0	0.0

36	DMY	0	0	0.0	0.0
37	DMY	0	0	0.0	0.0
38	DMY	0	0	0.0	0.0
39	DMY	0	0	0.0	0.0
40	DMY	0	0	0.0	0.0
41	DMY	0	0	0.0	0.0
42	KR3	0	0	0.0E-00	0.0
43	ZR5	0	0	0.0	0.0
44	MO5	0	0	0.0	0.0
45	MO7	0	0	0.0	0.0
46	MO9	0	0	2.9164E-06	0.0
47	TC9	0	0	0.0E-00	0.0
48	RU1	0	0	0.0E-00	0.0
49	RU3	0	0	2.0259E-07	0.0
50	RU5	0	0	4.3365E-05	0.0
51	RH3	0	0	0.0E-00	0.0
52	RH5	0	0	5.4237E-06	0.0
53	PD5	0	0	0.0E-00	0.0
54	PD7	0	0	3.3615E-15	0.0
55	PD8	0	0	0.0E-00	0.0
56	PD9	0	0	1.4305E-05	0.0
57	AG9	0	0	0.0E-00	0.0
58	CD3	0	0	0.0E-00	0.0
59	I01	0	0	9.9770E-07	0.0
60	I03	0	0	0.0	0.0
61	I05	0	0	2.9239E-05	0.0
62	XE1	0	0	0.0E-00	0.0
63	XE3	0	0	1.5165E-06	0.0
64	XE5	0	0	2.0997E-05	0.0
65	XE6	0	0	0.0	0.0
66	CS3	0	0	0.0E-00	0.0
67	CS4	0	0	1.0652E-08	0.0
68	CS5	0	0	9.5563E-15	0.0
69	PR1	0	0	0.0	0.0
70	PR3	0	0	5.9076E-07	0.0
71	ND3	0	0	0.0E-00	0.0
72	ND4	0	0	0.0	0.0
73	ND5	0	0	0.0E-00	0.0
74	ND6	0	0	0.0	0.0
75	ND7	0	0	7.2999E-07	0.0
76	PM7	0	0	8.3738E-09	0.0
77	PMM	0	0	1.9425E-07	0.0
78	PMG	0	0	1.4939E-06	0.0
79	PM9	0	0	3.6260E-06	0.0
80	PM0	0	0	7.1844E-05	0.0
81	PM1	0	0	6.7796E-06	0.0
82	SM7	0	0	2.0541E-19	0.0
83	SM8	0	0	2.7474E-24	0.0
84	SM9	0	0	2.1980E-24	0.0
85	SM0	0	0	0.0E-00	0.0
86	SM1	0	0	2.3633E-10	0.0
87	SM2	0	0	0.0E-00	0.0
88	EU3	0	0	0.0E-00	0.0
89	EU4	0	0	2.5558E-09	0.0
90	EU5	0	0	4.5792E-09	0.0
91	EU6	0	0	5.2580E-07	0.0
92	GD4	0	0	0.0E-00	0.0
93	GD5	0	0	0.0E-00	0.0

94	GD6	0	0	0.0E-00	0.0
95	GD7	0	0	0.0E-00	0.0
96	GD8	0	0	0.0E-00	0.0
97	PSD	0	0	0.0	0.0 pseudo FP in Iijima medel
98	F3N	0	0	0.0	0.0
99	F3S	0	0	0.0	0.0
100	F3R	0	0	0.0	0.0
101	F5N	0	0	0.0	0.0
102	F5S	0	0	6.00000E-10	0.0
103	F5R	0	0	2.10000E-10	0.0
104	F9N	0	0	0.0	0.0
105	F9S	0	0	4.40000E-10	0.0
106	F9R	0	0	1.66000E-10	0.0
107	PS2	0	0	0.0	0.0 pseudo FP in VSOP 25 model
108	PS4	0	0	0.0	0.0 pseudo FP in VSOP 40 model

2) Heavy nuclide chains

The routine is programed to process (n,2n) reaction. However it is not so important in thermal reactors, then we neglect it in the present preparation.

Th232 -> U233 -> U234 -> U235 -> U236
U238 -> Pu239 -> Pu240 -> Pu241 -> Pu242

3) Absorber chains

We can consider the decrease of concentration of absorbing nuclides contained in burnable poison or control rod.

B10
Gd155 -> Gd156 -> Gd157
Ag107
Ag109
In113
In115
Cd113
Hf176 -> Hf177 -> Hf178 -> Hf179 -> Hf180

4) F.P. chains

We have four possible options for F.P. chains

4.1) Garrison model

The model of Garrison and Roos (Ref.(20)) consists of rapidly saturating, slowly saturating, and non-saturating groups plus two explicit nuclides, Xe-135 and Sm-149. Their neutron cross sections are taken from ENDF/B-II. Each is separately prepared for U-233, U235, and Pu-239 thermal fission.

F3N,F3S,F3R from fission of U03

F5N,F5S,F5R from fission of U05

F9N,F9S,F9R from fission of PU9

Fission yields
****

	TH2	PA3	U03	U04	U05	U06
XE5	0.052	0.062	0.062	0.062	0.066	0.065
SM9	0.008	0.008	0.008	0.008	0.011	0.018
F3N	0.0	0.0	1.5884	0.0	0.0	0.0
F3S	0.0	0.0	0.338	0.0	0.0	0.0
F3R	0.0	0.0	0.0036	0.0	0.0	0.0
F5N	0.0	0.0	0.0	0.0	1.5403	0.0
F5S	0.0	0.0	0.0	0.0	0.378	0.0
F5R	0.0	0.0	0.0	0.0	0.0047	0.0

	U08	PU9	PU0	PU1	PU2
XE5	0.065	0.075	0.073	0.072	0.073
SM9	0.018	0.013	0.024	0.015	0.024
F5N	0.0	0.0	0.0	0.0	0.0
F5S	0.0	0.0	0.0	0.0	0.0
F5R	0.0	0.0	0.0	0.0	0.0
F9N	0.0	1.507	0.0	0.0	0.0
F9S	0.0	0.394	0.0	0.0	0.0
F9R	0.0	0.011	0.0	0.0	0.0

4.2) Iijima model

Iijima et al.(Ref.(22)) have proposed a model containing 45 explicit nuclides and one pseudo nuclide for BWR lattice calculation. We take their chain model and fission yields, but neutron cross sections from ENDF/B-IV except for the pseudo nuclide. We have another option to use fission yields taken from ENDF/B-IV which might be considered more systematic.

Following descriptions for F.P. chains are shown in the format of C-table desribed in Sect.V.9.

F.P. chains in Iijima model
****

```
003+001+042
003+002-046+047
003+001+048
003+002-049+051
003+003-050-052+053
003+003+054+055+057
003+001+058
003+002-059+062
003+003-063+066+067
003+003-061-064+068
003+002-070+071
003+001+073
003+001+097
003+015-075-076+082+083+084+085+086+087+088-089+092+093+094+095+096
013+015-075-076+082+083+084+085+086+087+088+089-090+293+294+295+296
014+015-075-076+082+083+084+085+086+087+088+089+090-091+294+295+296
006+016-075+5765330+077-079+284+285+286+287+288-289+292+293+294+295+296
014+016-075+5765330+077-079+284+285+286+287+288+289-290+293+294+295+296
```

015+016-075+5765330+077-079+284+285+286+287+288+289+290-291+294+295+296
006+016-075+5764670+078-279+284+285+286+287+288-289+292+293+294+295+296
014+016-075+5764670+078-279+284+285+286+287+288+289-290+293+294+295+296
015+016-075+5764670+078-279+284+285+286+287+288+289+290-291+294+295+296
008+015-075+5765330+077+079+285+286+287+288-289+292+293+294+295+296
013+015-075+5765330+077+079+285+286+287+288+289-290+293+294+295+296
014+015-075+5765330+077+079+285+286+287+288+289+290-291+294+295+296
008+015-075+5764670+078+279+285+286+287+288-289+292+293+294+295+296
013+015-075+5764670+078+279+285+286+287+288+289-290+293+294+295+296
014+015-075+5764670+078+279+285+286+287+288+289+290-291+294+295+296
007+016-075+5765330-077+283+284+285+286+287+288-289+292+293+294+295+296
014+016-075+5765330-077+283+284+285+286+287+288+289-290+293+294+295+296
015+016-075+5765330-077+283+284+285+286+287+288+289+290-291+294+295+296
007+016-075+5764670-078+283+284+285+286+287+288-289+292+293+294+295+296
014+016-075+5764670-078+283+284+285+286+287+288+289-290+293+294+295+296
015+016-075+5764670-078+283+284+285+286+287+288+289+290-291+294+295+296
003+010-081+286+287+288-289+292+293+294+295+296
008+010-081+286+287+288+289-290+293+294+295+296
009+010-081+286+287+288+289+290-291+294+295+296

Fission yields in Iijima model

****

	TH2	PA3	U03	U04	U05	U06	U08
KR3	2.1662-2	2.1662-2	1.0189-2	3.9210-3	5.3817-3	3.9210-3	3.9210-3
ZR5	0.0	0.0	0.0	0.0	0.0	0.0	0.0
MO5	0.0	0.0	0.0	0.0	0.0	0.0	0.0
MO7	0.0	0.0	0.0	0.0	0.0	0.0	0.0
MO9	2.9249-2	2.9249-2	4.8905-2	6.2358-2	6.1046-2	6.2358-2	6.2358-2
TC9	0.0	0.0	0.0	0.0	0.0	0.0	0.0
RU1	7.4330-3	7.4330-3	3.2346-2	6.0791-2	5.0774-2	6.0791-2	6.5794-2
RU3	1.6066-3	1.6066-3	1.6749-2	6.2363-2	3.0404-2	6.2363-2	6.2363-2
RU5	3.4437-4	3.4437-4	4.8211-3	3.9864-2	9.6605-3	3.9864-2	3.9864-2
RH3	0.0	0.0	0.0	0.0	0.0	0.0	0.0
RH5	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PD5	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PD7	5.2437-4	5.2437-4	1.1709-3	1.3108-2	1.4274-3	1.3108-2	1.3108-2
PD8	6.2454-4	6.2454-4	6.3049-4	6.0817-3	6.8771-4	6.0817-3	6.0817-3
PD9	0.0	0.0	0.0	0.0	0.0	0.0	0.0
AG9	5.9371-4	5.9371-4	4.4515-4	3.0623-3	3.3076-4	3.0623-3	3.0623-3
CD3	6.9367-4	6.9367-4	1.3515-4	5.1225-4	1.1899-4	5.1225-4	5.1225-4
I01	1.6148-2	1.6148-2	3.6801-2	3.2073-2	2.8866-2	3.2073-2	3.2073-2
I03	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I05	5.2749-2	5.2749-2	4.9201-2	6.7981-2	6.3047-2	6.7981-2	6.7981-2
XE1	0.0	0.0	0.0	0.0	0.0	0.0	0.0
XE3	3.7992-2	3.7992-2	6.0218-2	6.6853-2	6.7031-2	6.6853-2	6.6853-2
XE5	2.9140-4	2.9140-4	1.2836-2	2.6640-4	2.4280-3	2.6640-4	2.6640-4
XE6	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CS3	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CS4	3.6400-9	3.6400-9	1.3010-5	3.2800-9	1.3500-7	3.2800-9	3.2800-9
CS5	0.0	0.0	1.8400-4	1.0000-7	2.0000-5	1.0000-7	1.0000-7
PR1	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PR3	6.6282-2	6.6282-2	5.8667-2	4.5270-2	5.9471-2	4.5270-2	4.5270-2
ND3	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ND4	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ND5	5.3631-2	5.3631-2	3.3831-2	3.7311-2	3.9173-2	3.7311-2	3.7311-2
ND6	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ND7	3.0985-2	3.0985-2	1.7100-2	2.5141-2	2.2296-2	2.5141-2	2.5141-2

PM7	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PMM	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PMG	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PM9	8.9415-3	8.9415-3	7.5678-3	1.6012-2	1.0748-2	1.6012-2	1.6012-2
PM0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PM1	1.7411-3	1.7410-3	3.1830-3	7.9770-3	4.1650-3	7.9770-3	7.9770-3
SM7	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SM8	1.348-08	1.348-08	1.187-06	8.220-09	6.848-08	8.220-09	8.220-09
SM9	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SM0	3.340-06	3.340-06	2.743-03	8.121-06	3.039-05	8.121-06	8.121-06
SM1	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SM2	7.5860-4	7.5860-4	2.0100-3	5.1850-3	2.6810-3	5.1850-3	5.1850-3
EU3	3.2880-4	3.2880-4	1.0450-3	3.9610-3	1.6130-3	3.9610-3	3.9610-3
EU4	2.01 -09	2.01 -09	2.16 -06	3.51 -08	2.55 -06	3.51 -08	3.51 -08
EU5	4.0320-5	4.0320-5	2.1130-4	1.3340-3	3.2080-4	1.3340-3	1.3340-3
EU6	2.5880-5	2.5880-5	1.1300-4	6.8570-4	1.3200-4	6.8570-4	6.8570-4
GD4	0.0	0.0	0.0	0.0	0.0	0.0	0.0
GD5	0.0	0.0	0.0	0.0	0.0	0.0	0.0
GD6	0.0	0.0	0.0	0.0	0.0	0.0	0.0
GD7	9.8190-6	9.8190-4	6.3610-5	3.8940-4	6.1560-5	3.8940-4	3.8940-4
GD8	5.3680-6	5.3680-6	2.2120-5	1.6970-4	2.9310-5	1.6970-4	1.6970-4
PSD	1.6673-0	1.6673-0	1.6014-0	1.3233-0	1.3890-0	1.3233-0	1.3233-0

PU8	PU9	PU0	PU1	PU2
3.9210-3	2.9512-3	3.9210-3	2.9512-3	3.9210-3
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
6.2358-2	6.1691-2	6.2358-2	6.1691-2	6.2358-2
0.0	0.0	0.0	0.0	0.0
6.0791-2	5.9532-2	6.0791-2	5.9532-2	6.0791-2
6.2363-2	6.9508-2	6.2363-2	6.9508-2	6.2363-2
3.9864-2	5.3828-2	3.9864-2	5.3828-2	3.9864-2
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
1.3108-2	3.3968-2	1.3108-2	3.3968-2	1.3108-2
6.0817-3	2.2041-2	6.0817-3	2.2041-2	6.0817-3
0.0	0.0	0.0	0.0	0.0
3.0623-3	1.6539-2	3.0623-3	1.6539-2	3.0623-3
5.1225-4	7.6830-4	5.1225-4	7.6830-4	5.1225-4
3.2073-2	3.8707-2	3.2073-2	3.8707-2	3.2073-2
0.0	0.0	0.0	0.0	0.0
6.7981-2	6.2905-2	6.7981-2	6.2905-2	6.7981-2
0.0	0.0	0.0	0.0	0.0
6.6853-2	7.0278-2	6.6853-2	7.0278-2	6.6853-2
2.6640-4	1.1126-2	2.6640-4	1.1126-2	2.6640-4
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
3.2800-9	9.8800-6	3.2800-9	9.8800-6	3.2800-9
1.0000-7	1.5300-4	1.0000-7	1.5300-4	1.0000-7
0.0	0.0	0.0	0.0	0.0
4.5270-2	4.4281-2	4.5270-2	4.4281-2	4.5270-2
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
3.7311-2	2.9880-2	3.7311-2	2.9880-2	3.7311-2
0.0	0.0	0.0	0.0	0.0

```
2.5141-2 2.0503-2 2.5141-2 2.0503-2 2.5141-2
     0.0      0.0      0.0      0.0      0.0
     0.0      0.0      0.0      0.0      0.0
     0.0      0.0      0.0      0.0      0.0
1.6012-2 1.2436-2 1.6012-2 1.2436-2 1.6012-2
     0.0      0.0      0.0      0.0      0.0
7.9770-3 7.7440-3 7.9770-3 7.7440-3 7.9770-3
     0.0      0.0      0.0      0.0      0.0
8.220-09 2.741-05 8.220-09 2.741-05 8.220-09
     0.0      0.0      0.0      0.0      0.0
8.121-06 1.159-03 8.121-06 1.159-03 8.121-06
     0.0      0.0      0.0      0.0      0.0
5.1850-3 5.9010-3 5.1850-3 5.9010-3 5.1850-3
3.9610-3 3.6660-3 3.9610-3 3.6660-3 3.9610-3
3.51 -08 9.19 -05 3.51 -08 9.19 -05 3.51 -08
1.3340-3 1.6790-3 1.3340-3 1.6790-3 1.3340-3
6.8570-4 1.1820-3 6.8570-4 1.1820-3 6.8570-4
     0.0      0.0      0.0      0.0      0.0
     0.0      0.0      0.0      0.0      0.0
     0.0      0.0      0.0      0.0      0.0
3.8940-4 7.4550-4 3.8940-4 7.4550-4 3.8940-4
1.6970-4 4.1950-4 1.6970-4 4.1950-4 1.6970-4
1.3233-0 1.3675-0 1.3233-0 1.3675-0 1.3233-0
```

4.3) V.S.O.P models

The code system V.S.O.P. (Ref.(24)) has been employed in extensive calculations of fuel cycle and life history of pebble bed HTR in KFA Jeulich. It keeps four chain schemes. We have taken two of them; Chain 25 most simple, and Chain 40 most precise one.

```
 F.P. chains in VSOP 25
 **********************
+003+001+047
+003+002-049+051
+003+001+052
+003+001+062
+003+003-063+066+067
+003+003-061-064+068
+003+001+107
+003+015-070+071+072+073+074-076+082+083+084+085+086+087+088+089+090
+004+014+070+272+273+274-276+282+283+284+285+286+287+288+289+290
+010+016-070+071+072+073+074+5764670-078+283+284+285+286+287+288+289
+290
+009+015+070+272+273+274+8764670-278+283+284+285+286+287+288+289+290
+011+015-070+071+072+073+074+5764670+078+284+285+286+287+288+289+290
+010+014+070+272+273+274+8764670+278+284+285+286+287+288+289+290
+010+018-070+071+072+073+074+5765330-5770460-278+283+284+285+286+287
+288+289+290
+009+017+070+272+273+274+8765330-8770460-278+283+284+285+286+287+288
+289+290
+012+017-070+071+072+073+074+5765330-5779540+283+284+285+286+287+288
+288+289+290
+011+016+070+272+273+274+8765330-8779540+283+284+285+286+287+288+289
+290
+013+017-070+071+072+073+074+5765330-5770460+078+284+285+286+287+288
+289+290
```

```
+012+016+070+272+273+274+8765330-8770460+278+284+285+286+287+288+289
+290
+011+015-070+071+072+073+074+5765330+077+284+285+286+287+288+289+290
+010+014+070+272+273+274+8765330+277+284+285+286+287+288+289+290

 F.P. chains in VSOP 40
 **********************
+003+001+042
+003+002-043+044
+003+001+045
+003+001+048
+003+001+047
+003+002-049+051
+003+002-052+053
+003+002+055+057
+003+001+058
+003+002-059+062
+003+003-063+066+067
+003+003-061-064+068
+003+001+069
+003+001+108
+003+018-070+071+072+073+074-076+082+083+084+085+086+087+088+089-090
+093+094+095
+018+017-070+071+072+073+074-076+082+083+084+085+086+087+088+089+090
+294+295
+004+017+070+272+273+274-276+282+283+284+285+286+287+288+289-290+293
+294+295
+017+016+070+272+273+274-276+282+283+284+285+286+287+288+289+290+294
+295
+010+019-070+071+072+073+074+5764670-078+283+284+285+286+287+288+289
-290+293+294+295
+019+018-070+071+072+073+074+5764670-078+283+284+285+286+287+288+289
+290+293+294+295
+009+018+070+272+273+274+8764670-278+283+284+285+286+287+288+289-290
+293+294+295
+018+017+070+272+273+274+8764670-278+283+284+285+286+287+288+289+290
+294+295
+011+018-070+071+072+073+074+5764670+078+284+285+286+287+288+289-290
+293+294+295
+018+017-070+071+072+073+074+5764670+078+284+285+286+287+288+289+290
+294+295
+010+017+070+272+273+274+8764670+278+284+285+286+287+288+289-290+293
+294+295
+017+016+070+272+273+274+8764670+278+284+285+286+287+288+289+290+294
+295
+010+021-070+071+072+073+074+5765330-5770460-278+283+284+285+286+287
+288+289-290+293+294+295
+021+020-070+071+072+073+074+5765330-5770460-278+283+284+285+286+287
+288+289+290+294+295
+009+020+070+272+273+274+8765330-8770460-278+283+284+285+286+287+288
+289-290+293+294+295
+020+019+070+272+273+274+8765330-8770460-278+283+284+285+286+287+288
+289+290+294+295
+012+020-070+071+072+073+074+5765330-5779540+283+284+285+286+287+288
+289-290+293+294+295
+020+019-070+071+072+073+074+5765330-5779540+283+284+285+286+287+288
+289+290+294+295
+011+019+070+272+273+274+8765330-8779540+283+284+285+286+287+288+289
```

```
-290+293+294+295
+019+018+070+272+273+274+8765330-8779540+283+284+285+286+287+288+289
+290+294+295
+013+020-070+071+072+073+074+5765330-5770460+078+284+285+286+287+288
+289-290+293+294+295
+020+019-070+071+072+073+074+5765330-5770460+078+284+285+286+287+288
+289+290+294+295
+012+019+070+272+273+274+8765330-8770460+278+284+285+286+287+288+289
-290+293+294+295
+019+018+070+272+273+274+8765330-8770460+278+284+285+286+287+288+289
+290+294+295
+011+018-070+071+072+073+074+5765330+077+284+285+286+287+288+289-290
+293+294+295
+018+017-070+071+072+073+074+5765330+077+284+285+286+287+288+289+290
+294+295
+010+017+070+272+273+274+8765330+277+284+285+286+287+288+289-290+293
+294+295
+017+016+070+272+273+274+8765330+277+284+285+286+287+288+289+290+294
+295
```

Fission yields in VSOP models

****

	TH2	PA3	U03	U04	U05	U06	U08
	*1	*2	*3	*4	*5	*6	*
KR3	1.03-2	1.03-2	1.03-2	5.26-3	5.26-3	5.26-3	3.01-3
ZR5	6.17-2	6.17-2	6.17-2	6.10-2	6.10-2	6.10-2	4.85-2
MO5	2.30-3	2.30-3	2.30-3	3.50-3	3.50-3	3.50-3	1.00-4
MO7	5.52-2	5.52-2	5.52-2	5.86-2	5.86-2	5.86-2	5.64-2
MO9	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TC9	5.16-2	5.16-2	5.16-2	6.14-2	6.14-2	6.14-2	5.61-2
RU1	3.27-2	3.27-2	3.27-2	5.03-2	5.03-2	5.03-2	6.50-2
RU3	1.60-2	1.60-2	1.60-2	2.90-2	2.90-2	2.90-2	5.79-2
RU5	0.0	0.0	0.0	0.0	0.0	0.0	0.0
RH3	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
RH5	5.00-3	5.00-3	5.00-3	9.00-3	9.00-3	9.00-3	5.47-2
PD5	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
PD7	1.20-3	1.20-3	1.20-3	1.70-3	1.70-3	1.70-3	3.60-2
PD8	7.00-4	7.00-4	7.00-4	7.00-4	7.00-4	7.00-4	2.60-2
PD9	0.0	0.0	0.0	0.0	0.0	0.0	0.0
AG9	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CD3	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I01	3.51-2	3.51-2	3.51-2	2.79-2	2.79-2	2.79-2	3.60-2
I03	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I05	4.85-2	4.85-2	4.85-2	6.33-2	6.33-2	6.33-2	6.30-2
XE1	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
XE3	6.06-2	6.06-2	6.06-2	6.73-2	6.73-2	6.73-2	7.18-2
XE5	1.34-2	1.34-2	1.34-2	0.255-2	0.255-2	0.255-2	1.152-2
XE6	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CS3	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
CS4	0.0000	0.0000	0.0000	8.90-6	8.90-6	8.90-6	8.72-6
CS5	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PR1	6.24-2	6.24-2	6.24-2	5.73-2	5.73-2	5.73-2	6.11-2
PR3	5.85-2	5.85-2	5.85-2	5.90-2	5.90-2	5.90-2	4.48-2
ND3	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
ND4	4.67-2	4.67-2	4.67-2	5.42-2	5.42-2	5.42-2	3.78-2
ND5	3.37-2	3.37-2	3.37-2	3.86-2	3.86-2	3.86-2	3.03-2
ND6	2.53-2	2.53-2	2.53-2	2.95-2	2.95-2	2.95-2	2.49-2
ND7	0.0	0.0	0.0	0.0	0.0	0.0	0.0

PM7	1.78-2	1.78-2	1.78-2	2.12-2	2.12-2	2.12-2	2.15-2
PMM	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
PMG	4.46E-6	4.46E-6	4.46E-6	6.00-6	6.00-6	6.00-6	6.20-6
PM9	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PM0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PM1	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SM7	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
SM8	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
SM9	7.73-3	7.73-3	7.73-3	1.00-2	1.00-2	1.00-2	1.24-2
SM0	1.00-5	1.00-5	1.00-5	1.30-5	1.30-5	1.30-5	2.10-5
SM1	3.65-3	3.65-3	3.65-3	4.08-3	4.08-3	4.08-3	8.11-3
SM2	1.86-3	1.86-3	1.86-3	2.12-3	2.12-3	2.12-3	5.81-3
EU3	1.12-3	1.12-3	1.12-3	1.69-3	1.69-3	1.69-3	3.40-3
EU4	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
EU5	3.00-4	3.00-4	3.00-4	2.91-4	2.91-4	2.91-4	1.50-3
EU6	0.0	0.0	0.0	0.0	0.0	0.0	0.0
GD4	0.0	0.0	0.0	0.0	0.0	0.0	0.0
GD5	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
GD6	1.25-4	1.25-4	1.25-4	1.25-4	1.25-4	1.25-4	6.20-4
GD7	6.56E-5	6.56E-5	6.56E-5	6.00-5	6.00-5	6.00-5	7.00-4
GD8	2.40-5	2.40-5	2.40-5	3.12-5	3.12-5	3.12-5	3.50-4
PS2	1.281-0	1.281-0	1.281-0	1.185-0	1.185-0	1.185-0	1.204-0
PS4	1.281-0	1.281-0	1.281-0	1.185-0	1.185-0	1.185-0	1.204-0

PU8	PU9	PU0	PU1	PU2
—7—*—	—8—*—	—9—*—	—0—*—	—1—
3.01-3	3.01-3	2.00-3	2.00-3	2.00-3
4.85-2	4.85-2	3.92-2	3.92-2	3.92-2
1.00-4	1.00-4	0.0	0.0	0.0
5.64-2	5.64-2	4.76-2	4.76-2	4.76-2
0.0	0.0	0.0	0.0	0.0
5.61-2	5.61-2	6.17-2	6.17-2	6.17-2
6.50-2	6.50-2	5.94-2	5.94-2	5.94-2
5.79-2	5.79-2	6.20-2	6.20-2	6.20-2
0.0	0.0	0.0	0.0	0.0
0.0000	0.0000	0.0000	0.0000	0.0000
5.47-2	5.47-2	5.47-2	5.47-2	5.47-2
0.0000	0.0000	0.0000	0.0000	0.0000
3.60-2	3.60-2	3.04-2	3.04-2	3.04-2
2.60-2	2.60-2	2.60-2	2.60-2	2.60-2
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
3.60-2	3.60-2	3.15-2	3.15-2	3.15-2
0.0	0.0	0.0	0.0	0.0
6.30-2	6.30-2	6.30-2	6.30-2	6.30-2
0.0000	0.0000	0.0000	0.0000	0.0000
7.18-2	7.18-2	6.71-2	6.71-2	6.71-2
1.152-2	1.152-2	1.152-2	1.152-2	1.152-2
0.0	0.0	0.0	0.0	0.0
0.0000	0.0000	0.0000	0.0000	0.0000
8.72-6	8.72-6	8.72E-6	8.72E-6	8.72E-6
0.0	0.0	0.0	0.0	0.0
6.11-2	6.11-2	6.11-2	6.11-2	6.11-2
4.48-2	4.48-2	4.48-2	4.48-2	4.48-2
0.0000	0.0000	0.0000	0.0000	0.0000
3.78-2	3.78-2	4.13-2	4.13-2	4.13-2
3.03-2	3.03-2	3.19-2	3.19-2	3.19-2

2.49-2	2.49-2	2.68-2	2.68-2	2.68-2
0.0	0.0	0.0	0.0	0.0
2.15-2	2.15-2	2.22-2	2.22-2	2.22-2
0.0000	0.0000	0.0000	0.0000	0.0000
6.20-6	6.20-6	6.20-6	6.20-6	6.20-6
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
0.0000	0.0000	0.0000	0.0000	0.0000
0.0000	0.0000	0.0000	0.0000	0.0000
1.24-2	1.24-2	1.43-2	1.43-2	1.43-2
2.10-5	2.10-5	2.10-5	2.10-5	2.10-5
8.11-3	8.11-3	5.00-3	5.00-3	5.00-3
5.81-3	5.81-3	7.25-3	7.25-3	7.25-3
3.40-3	3.40-3	2.00-3	2.00-3	2.00-3
0.0000	0.0000	0.0000	0.0000	0.0000
1.50-3	1.50-3	1.00-3	1.00-3	1.00-3
0.0	0.0	0.0	0.0	0.0
0.0	0.0	0.0	0.0	0.0
0.0000	0.0000	0.0000	0.0000	0.0000
6.20-4	6.20-4	6.20-4	6.20-4	6.20-4
7.00-4	7.00-4	7.00-4	7.00-4	7.00-4
3.50-4	3.50-4	3.50-4	3.50-4	3.50-4
1.204-0	1.204-0	1.204-0	1.204-0	1.204-0
1.204-0	1.204-0	1.204-0	1.204-0	1.204-0

VII.6 Temperature

	Temperature (degree K)	tag
1	300.	1
2	325.	2
3	350.	3
4	400.	4
5	450.	5
6	500.	6
7	550.	7
8	600.	8
9	900.	9
10	1200.	A
11	1600.	B

VIII Applications ; Benchmark Calculations

A series of benchmark tests has been done for K_{eff} values of the various types of thermal reactor critical assemblies. The assemblies adopted for these benchmark test are summalized in Table VIII-1. Here, the benchmark problems for fast reactors were settled to assess the accuracy of the library adopted in the fast energy region.

Table VIII-1 Critical assemblies used for benchmark tests

No.	Assembly	Case	Remarks	Refs.
1	TCA	9	light-water moderated UO_2 lattices	26)
2	JMTRC	4	critical facility for Japan MTR	27)
3	DCA	1	D_2O moderated assembly for ATR (initial loading core)	28)
4	SHE	7	Graphite moderated critical assembly with 20% enriched ^{235}U fuel.	29)
5	Benchmark Problem for Fast Reactors	21	CSEWG Benchmark problems + four ZPR-3 assemblies + FCA assembly (Ref. (31))	30)

For thermal reactor problems, the effective macroscopic cross sections in the resonance energy region were calculated by using both the IRA method and the ultrafine group code PEACO to check the applicability of the former simple and time saving method. In the thermal energy region, the cell calculations were made with the collision probability method, and the effective cross sections were calculated by a cell- homogenization procedure without collapsing the U-library energy group. The resulting macroscopic cross sections were collapsed into few groups constants by using the 1D diffusion code TUD to take account of their space dependence. The overall K_{eff} calculations were carried out through the diffusion routine with the 2D or 3D option depending on the geometry under study.

A detailed description of these prescriptions, together with the additional calculated results, was published in Ref. (25).

VIII.1 Benchmark Calculation of TCA (Light-Water Moderated Assembly)

The benchmark calculation was made for the UO_2 -critical systems selected from a sequence of the critical experiments at TCA (Ref. (26)). The fuel rods are composed of UO_2 with 2.6W/O ^{235}U enrichment. The 1.25cm diameter pellets are clad into an aluminium tube (Fig.VIII-1). The effective length of the fuel rod is 144.2cm. The geometry of the horizontal cross-sectional view of the lattices is rectangular and the lattice cell is square (Fig.VIII-2).

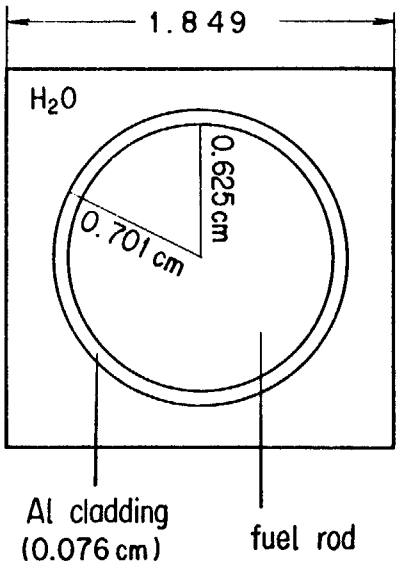


Fig.VIII-1 Unit cell of TCA core

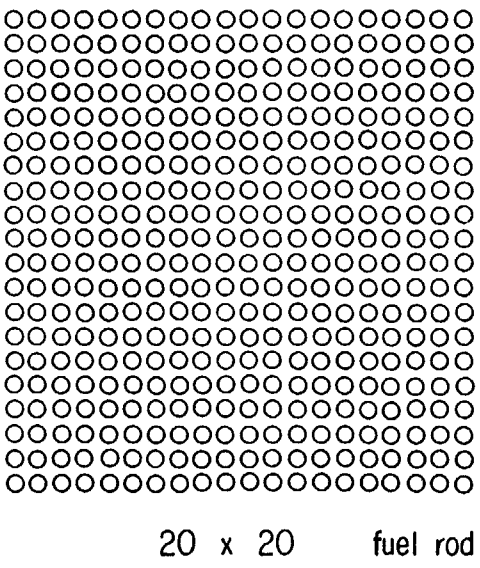


Fig.VIII-2 Pattern of lattice configuration

Investigations have been made for the effects of the cell model (cylindrical or square lattice), the number of spatial meshes in the unit cell, and the calculational model of the resonance absorption in the second resonance range on the K_{eff} calculation. The results showed that these effects, except for the number of the T-regions, had no essential importance for the K_{eff} calculation on the TCA analysis. Here, the K_{eff} calculations were made through the 3D (X-Y-Z) diffusion routine with 3 group approximation. The collapsing of 60 U-library groups (27 for thermal, 33 for fast) into 3 groups was carried out by the 1D TUD code with appropriate bucklings. For the structural materials at the top and lower parts of the core, the effective cross sections were obtained by setting another regions in reflector for their reprensentation.

Table VIII-2 Calculated K_{eff} values for TCA

Core pattern (V_{H_2O}/V_f)	Critical water level (cm)	K_{eff}*
20 (20 x 20) (1.50)	73.49	0.9960 (0.9926)**
24 (20 x 20) (1.50)	53.12	0.9976 (0.9915)
28	43.89	0.9985

(24 x 24) (1.50)		(0.9883)
1 (12 x 29) (1.83)	131.94	0.9975
2 (14 x 27) (1.83)	69.01	0.9998
15 (18 x 18) (1.83)	75.32	0.9983
20 (20 x 20) (1.83)	51.65	0.9992
5 (15 x 15) (3.00)	90.75	1.0044
18 (19 x 19) (3.00)	41.54	1.0078

*) The exposed part of fuel pin above the water level was corrected by a simple model based on the diffusion calculation.

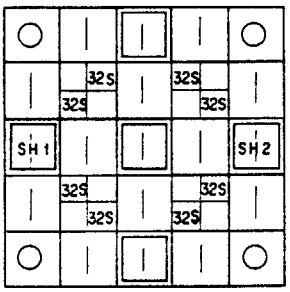
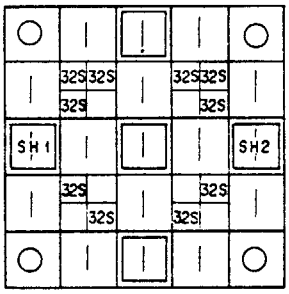
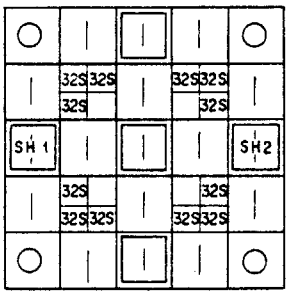
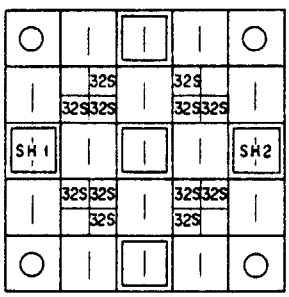
**) The parentheses show the uncorrected value.

Table VIII-2 shows the final results of the TCA benchmark calculation, which were obtained by assuming 5 regions both for the T-and R-regions. This table also shows the corrected K_{eff} values for the exposed part of the fuel rods, which were obtained from simply homogenizing the exposed parts. This correction is more important for the assemblies with lower water level. The results are in an excellent agreement with the experimental results, though a qualitative trend seems to exist in the case of $V_{H_2O}/V_f = 3$, compared with $V_{H_2O}/V_f = 1.5$ and 1.83. It should be noted that high accuracy can not be anticipated without a 3D calculation for a rectangular geometry with reflector such as treated here.

VIII.2 Benchmark Calculation of JMTRC (Japan Material Testing Reactor Critical Facility)

The criticality calculations were made for JMTRC to assess the applicability of the SRAC system to the plate type fuel elements. Considering the correspondence of the calculated results to the experimental values and the minimization of the estimate errors of excess reactivity, the four cores, which were critical at the almost full withdrawal of control rods and geometrically symmetric at their core centers, were selected for the benchmark ones. These were composed of 12 fuel elements with ^{235}U of 135g, 5 fuel followers, and Be and Al reflectors in a light-water pool. The critical conditions were

accomplished by changing the location of the pseudo irradiation materials (32S) to obtain a control rod configuration with nearly full withdrawal. **Figure VIII-3** shows each configuration of the four cores.

case	core configuration	control rod position of SH1 and SH2 (mm)	reactivity
1		638.1	638.1 → 0.015 (mm) (%ΔK/K/mm) $\rho_{ex}=\frac{(855-638.1)}{2}\times 0.015$ = 1.627 % ΔK/K K_{eff} = 1.01654
2		681.1	681.1 → 0.01205 (mm) (%ΔK/K/mm) ρ_{ex} = 1.048 % ΔK/K K_{eff} = 1.01059
3		755.5	755.5 → 0.00687 (mm) (%ΔK/K/mm) ρ_{ex} = 0.342 % ΔK/K K_{eff} = 1.00343
4		794.5	794.5 → 0.0042 (mm) (%ΔK/K/mm) ρ_{ex} = 0.127 % ΔK/K K_{eff} = 1.00127

Al reflector fuel element fuel follower

Fig.VIII-3 Measured results for various core configurations

All the cell calculations were made through collision probability routine using 61 U-library groups (24 for fast, 37 for thermal with the cut off energy of 0.68256 eV). The cell calculations for the fuel elements were carried out by a two step calculation for considering the double heterogeneity ; the calculations for meat parts were followed by those for side plates. Moreover for the fuel followers, one more step was taken between the above two steps to consider the guide tubes. Here, all the calculations were made on plane geometry, and the second resonance region was treated by the ultrafine groups. Concerned for the reflectors in core which were affected by the surrounding fuel elements and followers, two super-cell models were used, of which one is shown in Fig.VIII-3. The other model is the conventional one that has an additional cylindrical region with a homogenized composition for the surrounding regions. On the other hand, the collapse of the cross sections in reflector were performed by the 1D diffusion model using the TUD code in slab geometry after the cell calculations.

Table VIII-3 Calculated K_{eff} values for JMTRC

Case	Experiment	SRAC				THERMOS-JMTR,GGC4	
		1	2	3	4	5	6
1	1.01654	1.01974	1.01921	1.01656	1.01598	1.00896	1.00839
2	1.01059	1.01400	1.01347	1.01047	1.00986	1.00286	1.00223
3	1.00343	1.00772	1.00717	1.00383	1.00315	0.99620	0.99551
4	1.00127	1.00484	1.00428	1.00086	1.00011	0.99309	0.99229
Cell and Collapsing Model	Fuel Follower 32S 4F,ALFR in Core	No Extra Region	Extra Fuel Region (Cylinder)		Extra Fuel Region (Plane) (2cm)	No Extra Region	Extra Fuel Region (2cm) Thermal
	Reflector	No Extra Region	TUD	TUD	TUD	No Extra Region	

The overall core calculations were made by the CITATION code using a 2D (X-Y) models. Table VIII-3 shows the calculated results, together with those obtained from the combined use of the GGC-4 (Ref.(32) and THERMOS (Ref.(33)) codes. The results from the SRAC system are in an excellent agreement with the experimental results. Especially those of the model 3 agree well within 0.04%(ΔK_{eff}), while there are some discrepancies of about 1%(ΔK_{eff}) between the experimental results and those obtained from the existed data and method of the models 5 and 6.

VIII.3 Benchmark Calculation of DCA (Heavy-Water Moderated Assembly) Initial Core.

The DCA standard cell is the heavy-water moderated one of the pressurized tube-in-type light water cooling in which 28 fuel rods are concentrically located (see Fig.VIII-4). These cells constitute a square lattice cell of 22.5cm pitch. This cell was treated by the collision probability routine using the subroutine 'CLUP', where the cylindricalized approximation with symmetrically reflected boundary conditions was assumed for the outer boundary, as shown in Fig.VIII-4.

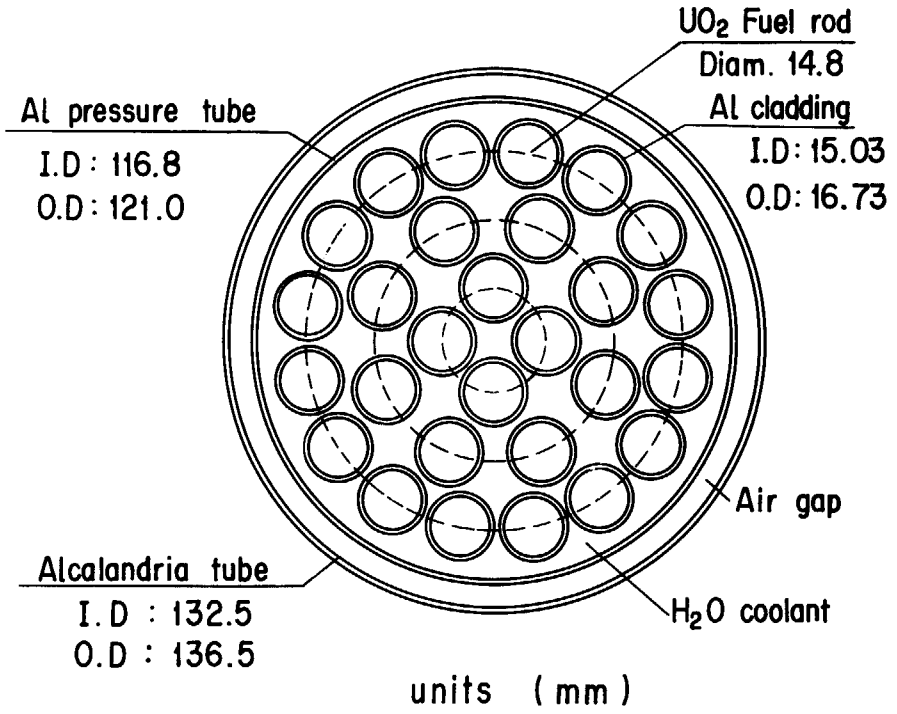


Fig.VIII-4 Calculation model of DCA lattice cell

Concerning to the energy group structure of the U-library , 33 groups were taken for the fast energy region with 0.5 lethargy width, and 35 for thermal with the cut off energy of 0.64825eV. The calculation of the effective resonance cross sections was made both by the exact method using the subroutine 'PEACO' and the simple method based on the IRA. The SRAC system has three options on the calculational methods of isotropic diffusion coefficients, as described in the Sect. II.1. The diffusion coefficients were calculated by all these three methods to investigate the effect of the differences on the K_{eff} calculations.

The DCA core are assembled in the cylindrical tank with the diameter of 3m, filled with heavy water where the calandria tubes of 121 channels are arranged in a square lattice. The fuel rods, pressurized tubes, calandria tubes and so on are exposed out of the water level, though the bottom is provided with the absorber plate which can be assumed to be black for thermal neutron. Hence, excepting this complexity, it is possible to treat this geometry by a 1D model. In practice, the core calculations were made by the 1D diffusion code 'TUD' without collapsing the 70 U-library groups.

Table VIII-4 Calculated K_{eff} values for DCA Initial Core

Conditions	K_{eff}	ΔK_{eff} (%)
SRAC Standard:	1.0007	———
σ_t by Summation	1.0008	0.01
Σ_{tr} by Consistent Transport Correction	0.9997	-0.10
D by Isotropic Behrens Benoist	0.9896	-1.11
Resonance Integral by I.R.A	1.0020	0.13
Resonance Integral by Table-Look-Up	0.9980	-0.27
LAMP-A (Ref.6)	1.0113	1.04

*) Standard Specifications

Lattice Pitch	22.5 cm square
Critical Radius	139.2 cm
Critical Height	96.2 cm (heavy water)
Theoretical Critical Height	105.0 cm(B_z^2 =.000895 cm^{-2})
Core Radius	150.0 cm

σ_{tr} by Harmonic Average
Σ_{tr} by B_1 Approximation
D by $1/(3\Sigma_{tr})$

Table VIII-4 shows the results obtained under the various calculational conditions for the DCA initial loading cores. Here is a problem on how to estimate an extrapolated distance for the core heights, though a total extrapolation distance of 9cm assumed for the top and bottom sides was added to the physical core height for calculating the bucklings. It should be noted that the difference of 1cm in the extrapolation distance gives $\Delta K_{eff} \sim 0.2$. The calculational models for the various group constants do not affect so much the K_{eff} values, because of their cancellation effects among energy groups. From the discussions made, it will be concluded that the SRAC system can predict, with higher accuracy, the criticality of the DCA.

VIII.4 Benchmark Calculations of SHE (Graphite Moderated Assembly)

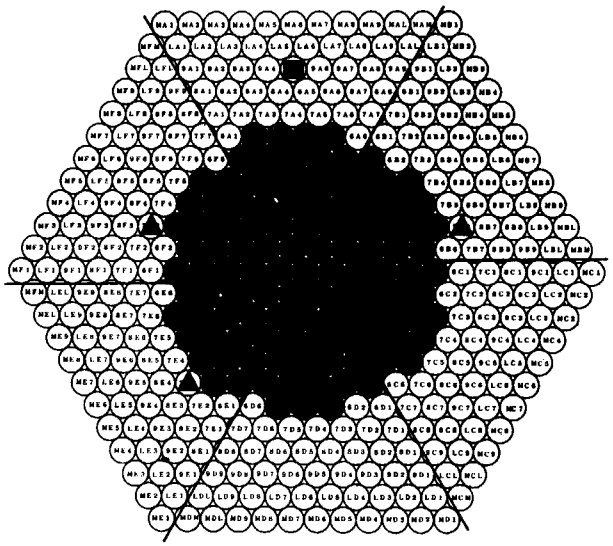
The semi-homogeneous experiment (SHE) assembly is a critical system with 20% enriched ^{235}U fuel and moderated and reflected by carbon (Ref. (29)). The benchmark calculations were made for the K_{eff} values of the following cores :

1) The homogeneous core with the different ratio of atomic number densities $C/^{235}U$ from 5383 to 2316 (SHE-5 to 8 Core).

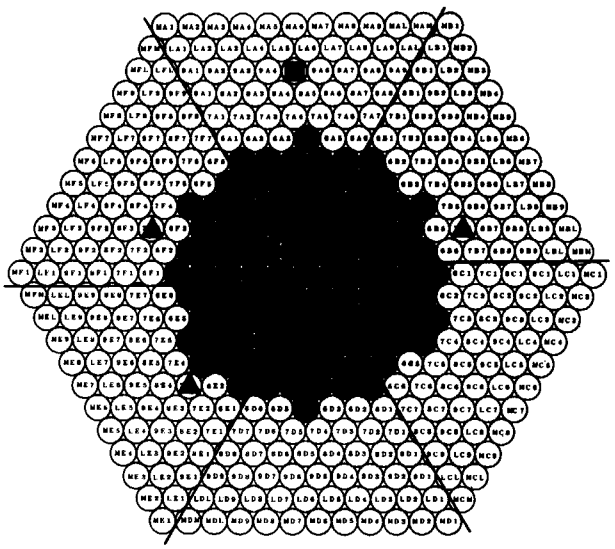
2) The heterogeneously loaded cores (SHE-12 and 13 cores).

3) The mock-up core for the multi-purpose high temperature gas cooled reactor (SHE-14).

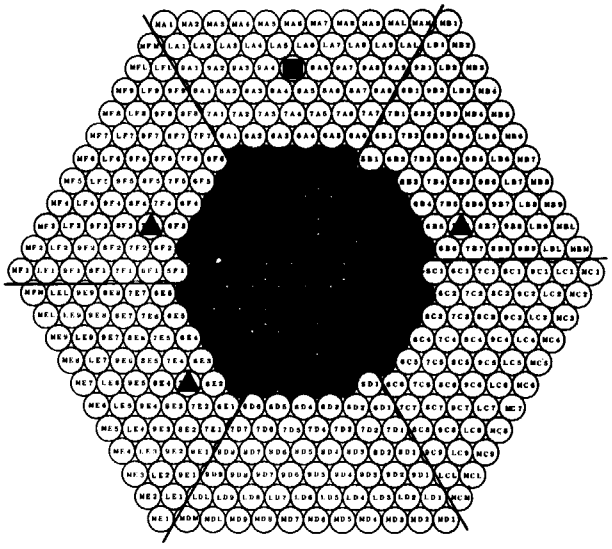
The **figure VIII-5** shows the loading pattern of each core.



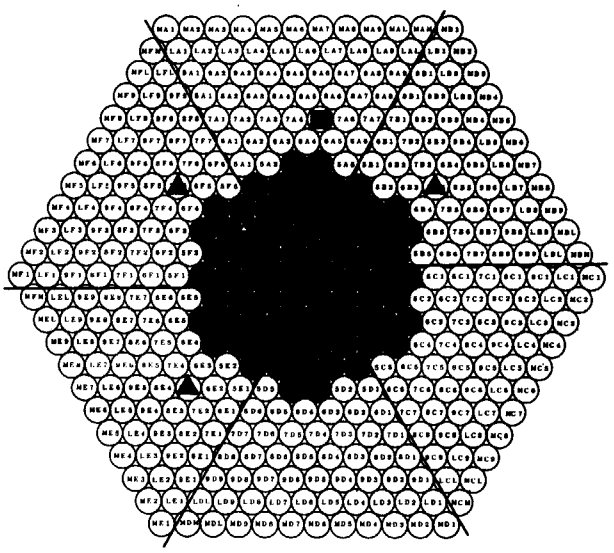
SHE－5



SHE－6

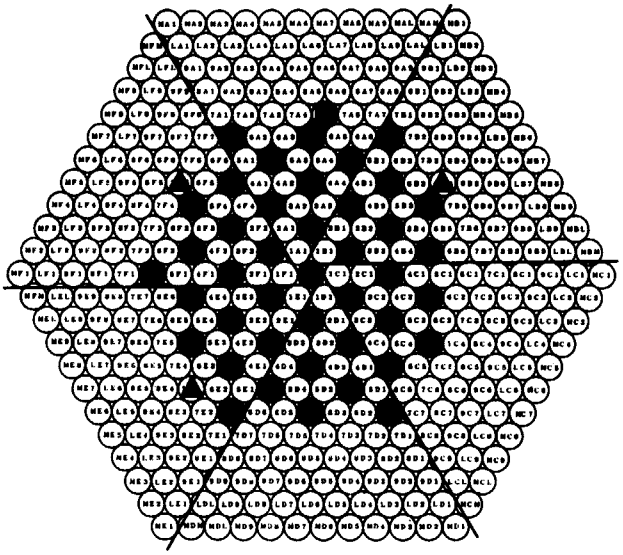


SHE－7



SHE－8

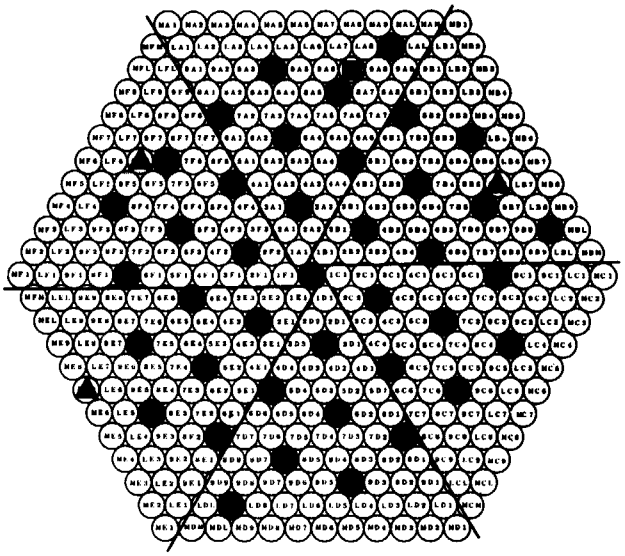
Fig.VIII-5 Fuel rods loading pattern of SHE cores



S H E - 12

Key

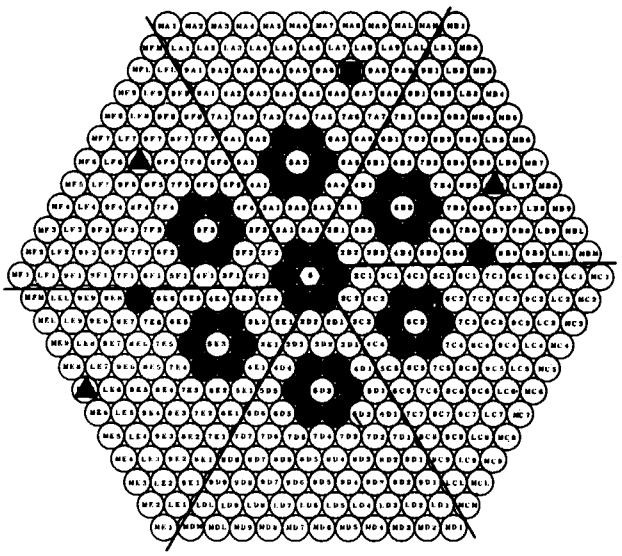
- Fuel rod
- Graphite rod
- Safety rod
- Control rod



S H E - 13

Key

- Fuel rod
- Graphite rod
- Safety rod
- Control rod



S H E - 14

Key

- Fuel rod
- Graphite rod
- Safety rod
- Control rod

Fig.VIII-5 (continued)

The standard cell calculations were performed by the collision probability routine making use of 22 groups for fast energy region and 39 for thermal with the cut off energy of 1.1254 eV. The second resonance range was treated again by both the ultra-fine spectrum code 'PEACO' and the IRA method. The core calculations were made mainly by the 1D diffusion routine on the cylindrical geometry with a radial reflector, since the core were nearly cylindrically loaded with fuel rods but not installed with any axial reflector. Here, the axial neutron leakage was treated by considering the axial buckling B_z^2 .

Table VIII-5 Calculated effective multiplication factor K_{eff} for SHE

Core Configuration	$C/^{235}U$	Core Radius(cm)	Calculated Values of K_{eff}
SHE- 5	5483	35.10	1.0015
SHE- 6	4395	33.17	1.0050
SHE- 7	3359	31.04	1.0071
SHE- 8	2316	28.71	1.0075
SHE-12	6785	37.28	1.0140 (1.0053)*
SHE-13	15724	55.76	1.0184 (1.0097)*
SHE-14	7158	39.54	1.0127 (1.0040)*

*) Corrected values for neutron streaming and end cap effect.

Table VIII-5 shows the results obtained for the K_{eff} values by the 1D diffusion and the IRA. It will be seen that the calculated values agree with the experimental ones within 0.7%(ΔK_{eff}) for the homogeneous cases of SHE-5 ~ 8. Brief discussions are made on the calculated results in the followings.

i) Effects of the thermal cut off energy on the K_{eff} values

A change of about 1% was observed in the calculated K_{eff} values for the change of the cut off energy from 0.68256 to 1.1254 eV, while there is no appreciable difference for the rise of the cut off energy from 1.1254 to 1.8554 eV. This change is attributed to the neglect of neutron up-scattering in the first energy region.

ii) Transport effect

For the SHE-8 core, there is a difference of only about 0.1% (ΔK_{eff}) between the calculated results obtained by the 1D diffusion code 'TUD' and the S_N code 'ANISN' based on the S_8P_1 approximation.

iii) Effect of the treatment of resonance absorption

The ultra-fine spectrum calculation and the IRA gives, respectively , 60.63 and 59.42 barns for the resonance integral of ^{238}U in the SHE- 8 core in the first resonance energy range. This difference results in the change of about 0.02% in reactivity.

iv) Mesh and geometry effects in diffusion calculation

The criticality calculation of the SHE-14 core, which has a nearly hexagonal geometry, was made also by a 2D triangular geometry using the CITATION code (11 groups for fast and 13 for thermal with 1.1254

eV). The difference between the 1D and 2D criticality calculations was shown to be so small (about 0.02%) as to be neglected. This can be attributed to the larger width of reflector (91cm), compared with the core radius of 38 cm.

v) Streaming effect through air gaps in core

The SHE cores are composed of the carbon matrix tubes and the carbon rods, respectively having the diameter of 6.5 cm. Hence there are triangular air gaps, which occupy an area of about 10% of one carbon rod. The axial neutron leakage effect through the air gaps was investigated for the SHE-8 core. The radial and axial diffusion coefficients (Dr and Dz) were calculated by the Benoist model in the unit cell of the SHE-8 core. The resulting ratio Dz/Dr was 1.04. The criticality calculation using B^2Dz as axial leakage gave a lower criticality factor by 0.57%, compared with the case where the isotropic diffusion coefficient was used.

Table VIII-5 shows that the calculated K_{eff} values for the heterogeneous cores also agree with the measured ones within 1%(ΔK_{eff}), when the various corrections above mentioned are made.

VIII.5 Benchmark Calculations of Fast Critical Assemblies for Assessment of Fast Energy Group Constants

Benchmarking was made for 21 fast critical assemblies which consist of 6 cases of uranium core and 15 cases of plutonium core. These assemblies have been used to assess the nuclear data libraries for fast reactor analysis (Refs.(30,34)).

Table VIII-6 Calculated K_{eff} values for benchmark test problems of fast reactors

No.	Assembly	Fuel	JFS data: EXPANDA code for fast reactor	JENDL-IIB: for fast reactor	ENDF/B-4 data: for fast reactor	ENDF/B-4 data: SRAC
1	VERA-11A	Pu	0.99240	0.99236	0.9855	0.9861
2	VERA-1B	U	1.00360	0.99537	0.9954	0.9953
3	ZPR-3-6F	U	1.01660	1.01225	1.0095	1.0095
4	ZEBRA-3	Pu	0.99797	0.99825	0.9917	0.9850
5	ZPR-3-12	U	1.00697	1.00114	0.9987	0.9985
6	SNEAK-7A	Pu	1.00508	1.00135	0.9915	0.9892
7	ZPR-3-11	U	1.00800	1.00211	1.0050	1.0041
8	ZPR-3-54	Pu	0.95435	0.95443	0.9338	0.9347
9	ZPR-3-53	Pu	0.99650	0.99116	0.9777	0.9747
10	SNEAK-7B	Pu	1.00436	0.99852	0.9896	0.9842
11	ZPR-3-50	Pu	0.99848	0.99709	0.9817	0.9747
12	ZPR-3-48	Pu	1.00306	1.00049	0.9891	0.9852
13	ZEBRA-2	U	0.98523	0.98314	0.9781	0.9751
14	ZPR-3-49	Pu	1.00416	1.00546	0.9924	0.9882
15	ZPR-3-56B	Pu	0.99668	0.98885	0.9775	0.9750
16	ZPR-6-7	Pu	1.00332	0.99368	0.9816	0.9736
17	ZPR-6-6A	U	1.00191	0.99884	0.9895	0.9857
18	ZPPR-2	Pu	1.00874	1.00037	0.9887	0.9789
19	MZA	Pu	1.00119	0.99446	0.9842	0.9904

: 20	MZB	: Pu	: 1.00129	: 0.99114	: 0.9809	: 0.9784 :
: 21	FCA-V-2	: Pu	: 1.00910	: 1.00567	: 0.9787	: 0.9797 :

The criticality calculations were made by using the 1D diffusion code TUD without collapsing the P-library. The fission spectrum of ^{235}U was assumed for all the cases. The calculated results are shown in Table VIII-4-6 in comparison with those obtained from other group constant libraries. Here, the symbol JFS-II stands for JAERI-Fast Set, Version II (Ref.(34)), and JFS-IIB is a library with the same group structure as JFS-II but produced using the nuclear data file JENDL-IIB (Ref.(14)). On the other hand, the symbol ENDF/B-4 shows a library based on the nuclear data file ENDF/B-IV, which has the same group structure as the P-library of the SRAC system and contrasts with it in the following points :

i) The resonance shielding factors of JFS-II were calculated by assuming M = 30 for moderator mass, while the NRA was assumed for the moderator slowing-down on the P- library production.

ii) The self-shielding factors of medium weight nuclides were taken to be unity when they were close to unity, since the SRAC library is to be used in thermal reactor analysis.

iii) The temperature dependence of the shielding factors were neglected for the capture reaction of structural materials.

iv) The shielding factor of the scattering cross sections were substituted for those of the removal, while the inherent shielding factors were prepared in the library of JAERI Fast Set.

The item ii) above described is already modified to be consistent with the JAERI Fast Set library.

Table VIII-6 shows that the qualitative trend of ENDF/B-IV that the calculated values of K_{eff} decrease with increase of the fertile to fission ratio of the core. This tendency is also seen in the P-library. There are, however, some quantitative differences; for example, the amount of this difference is nearly 1% on the No.18 core, ZPPR-2. These causes are not so clear, but may be attributed to an overall result of the above four contrasts. Particularly for the ZPR-3-54 which has an iron reflector, the discrepancy between measured and calculated results is remarkable, while the S_N calculation gives almost $K_{eff} = 1$. Hence, there seem to be some problems on the calculating method of the transport cross section to obtain diffusion coefficient, especially for a simple composition($\sigma_0 = 0$).

IX Sample Input and Output

Here we present two sample input data to help the user understand the input specifications and compose his own calculation scheme.

IX.1 Pin-Rod Cell and Burn-up Calculation

The following is an input data which was used for a burn-up calculation of single pin rod cell as a reference of NEA benchmark calculations of BWR elements with adjacent poisoned fuel pins which were performed in 1981.

Specification
Fuel: Material = UO2, density = 10 g/cc, temperature = 600 °C,
U-235 enrichment = 3 w/O, diameter = 1 cm.
Clad: Material = Zircalloy-2, density =6.55 g/cc, temperature = 300 °C
inside diameter = 1 cm, outside diameter = 1.2 cm.
Moderator: Material = H2O, void content = 0 %, temperature = 286 °C,
lattice pitch = 1.6 cm.
Power density = 20 w/U-metal gram in the fresh fuel.

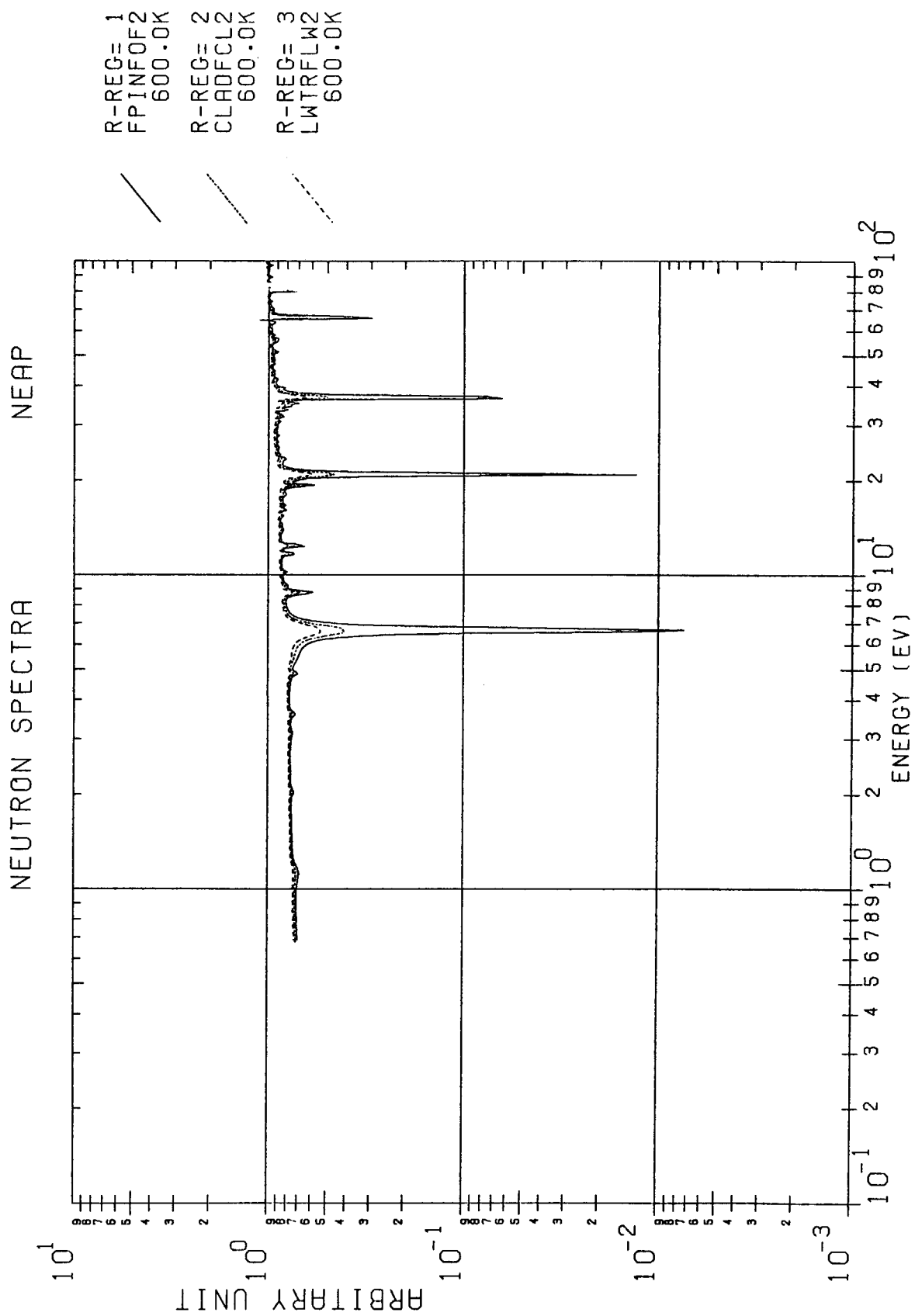
Calculation model
Geometry: Cylinder of five regions
Resonance absorption: By PEACO
Energy group structure: 33 fast groups 37 thermal groups, thermal cut off = 0.6825 eV.
Burn-up scheme: Garrison model

```
*********1*********2*********3*********4*********5*********6*********7*
NEAP
BWR BENCHMARK SINGLE PIN ROD CELL ; BURN-UP
 1 1 1 1 5 1 4 0 2 0   0 0 0 0 1  2 1 0 0 1 / SRAC CONTROL
 .00020 / AXIAL BUCKLING ONLY
J1480            0 0 / UPDATE USER FAST LIB
J1480            0 0 / UPDATE USER THERMAL LIB
J1480            0 0
J1480            0 0
33 37 0 0 / NEF NET NERF NERT
  31(2) 2(4)/ FINE FAST GROUP MESH
  37(1) /      FINE THERMAL GROUP MESH
XH01H008
XO060008
XU050009
XU060009
XU08W009
XO06W009
XGD50009
XGD70009
XZRN0008
XFEN0008
XNIN0008
XCRN0008
XPU90009
XPU00009
XPU10009
XPU20009
XF5N0001
```

```
XF5R0001
XF5S0001
XF9N0001
XF9R0001
XF9S0001
XXE50009
XSM90009

  3 5 5 3 1 0 5 0 0 0   0 0 6 1 0  0 0 0   / PATH  1-D CYL
0 6(0) 6(0.) /
 1 1   2  3  3 / T-R
 1 1   1 / X-R
 2 4   1  /  M-R
 0. 0.4 0.5   0.6   0.8  0.90270 / RX
4 / NMAT
LWTRXLWX  0 2 600. 2.   1. /  MAT 1 : LIGHT WATER 600 D K
XH01H008  0 0 .049737
XO060008  0 0 .024868
FPINXOFX  0 4 900. 1.   1. /  MAT 2 : FUEL ROD NORMAL
XU08W009  2 1 .021637    / 1  PU ISOTOPES AND PSEUDO FP'S
XU050009  2 1 .0006777   / 2  ARE AUTOMATICALLY ADDED AFTER
XO06W009  0 1 .044630    / 3  BURN-UP
GPINXOGX  0 5 900. 1.   1. /  MAT 3 : FUEL PIN INCLUDING GD
XU08W009  2 0 .020988    /    NOT USED IN THIS CASE
XU050009  2 0 .00065738 /
XO06W009  0 0 .044786    /
XGD50009  2 1 .00014850 / 4
XGD70009  2 1 .00015647 / 5
CLADXCLX  0 4 600. 0.18 1. /    MAT 4 : CLADDING
XZRN0008  0 1 .04294     / 19
XFEN0008  0 1 .0000845   /20
XNIN0008  0 1 .0000341   /22
XCRN0008  0 1 .0000759   /23
10 0 0          / BURNUP STEP , BURN UNIT , EDIT
0.00013846  1.    / M WATT, HEIGHT OF PIN
0.0069194 9*0.0069194/   BURN UP STEP (1/10) PER CM HEIGHT
 0 / PEACO
           / NO NEXT CASE FOLLOWS

++
//
```

```
      **** CASE ID === NEAP     *** TITLE *** NEA BENCHMARK SINGLE PIN ROD CELL ; BURN-UP                    ***

     USE OF COLLISION PROB. ROUTINE (NO,CALL:0,1)                    1
     SELECT OF ROUTINE FOR FIX S CALC(NO,PIJ,1D SN ,2D SN ,
      1D DIFF,2D DIFF:0,1,2,3,4,5)                                   1
     DANCOFF FACTOR BY(INPUT,PIJ,FORMULA:0,1,2)                      1
     CALC OF THERMAL RANGE (EXCLUDED,INCLUDED:0,1)                   1
     PROCESS OF RESONANCE 2 RANGE(TABLELOOK,IR,PEACO:0,1,2)          2
     FLUX CALCULATION BY RANGES(SKIP,CALC:0,1)                       1
     FAST RANGES FOR TRANSPORT CALC(0,1,2,3,4)                       3
     UPDATE OF MCROSS FILE (NO,CALL:0,1)                             0
     CALC OF BARE REACTOR SPECTRUM(SKIP,P1,B1:0,1,2)                 2
     COLLAPSE MACRO X-SECTION BY RANGE(SKIP,CALL:0,1)                1
     INPUT OF GEOMETRY (NEW,SAME AS PREVIOUS CASE:0,1)               0
     SELECT ROUTINE FOR EIGENVALUE CALC(NO,PIJ,1D SN,2D SN,
      1D DIFF,2D DIFF:0,1,2,3,4,5)                                   0
     COLLAPSE MACRO AFTER EIGENVALUE CALC(SKIP,CALL:0,1)             0
     PREPARE MACRO FOR CITATION (SKIP,CALL:0,1)                      0
     TOTAL MICRO X-SECTION BY (HARMONIC,ARITHMETIC AV:1,2)           1
     TRANSPORT MACRO X-SECTION BY (P0,P1,B1,SN:0,1,2,3)              2
     DIFF COEF (INVERSE TR,BENDIST ISO,BENDIST ANISO:1,2,3)          1
     PLOT OF FLUX MACRO X-SECTION (1+2)                              0
     PRINT OF MACRO-X (0,1,2)                                        0
     BURN-UP CALCULATION (0,1:SKIP,EXECUTE)                          1
     NUMBER OF USER'S FAST GROUPS                                   33
     NUMBER OF USER'S THERMAL GROUPS                                37
     NUMBER OF USER'S CONDENSED FAST GROUPS                          2
     NUMBER OF USER'S CONDENSED THERMAL GROUPS                       1

     NUMBER OF LIBRARY GROUPS IN THE USER'S FAST GROUP
                2          2          2          2          2          2          2          2          2          2

                2          2          2          2          2          2          2          2          2          2

                2          2          2          2          2          2          2          2          2          2

                2          4          4
     NUMBER OF LIBRARY GROUPS IN THE USER'S THERMAL GROUPS
                1          1          1          1          1          1          1          1          1          1

                1          1          1          1          1          1          1          1          1          1

                1          1          1          1          1          1          1          1          1          1

                1          1          1          1          1          1          1
     NUMBER OF USER'S GROUPS IN THE CONDENSED FAST GROUP
               17         16
     NUMBER OF USER'S GROUPS IN THE CONDENSED THERMAL GROUP
               37


NEAP     NEA BENCHMARK SINGLE PIN ROD CELL ; BURN-UP                          *** STEP *** INPUT FOR PIJ ***
                    GEOMETRY TYPE                                  3          ( CIRCULAR CYLINDER )
                    NUMBER OF SUB - REGIONS                        5
                    NUMBER OF  T  - REGIONS                        5
                    NUMBER OF  R  - REGIONS                        3
                    NUMBER OF  X  - REGIONS                        1
                    OUTER BOUNDARY CONDITION (-1,0,1,2)            0
                    DIRECTIONAL PIJ (1,2)                          1  INDICATED BY IC17
                    NUMBER OF R OR X MESH                          5
                    NUMBER OF THETA OR Y MESH                      0
                    TOTAL NUMBER OF PIN RODS                       0
                    NUMBER OF RINGS OF PIN ROD ARRAY               0
                    NUMBER OF LATTICE CELLS TRACED                 1
                    PRINT CONTROL OF PIJ (0,1)                     0
                    ORDER OF GAUSS RADIAL INTEGRATION              6
                    NO. OF DIVISION FOR ANGULAR INTEGRATION        1
                    NUMBER OF ANNULAR DIVISION IN A PIN ROD        0
                    DIVISION BY RPP  (0,1,2)                       0
                    ANGLE RANGE BY DEGREE                          0
                    PLOTER OPTION  (0,1)                           0

          ***R-REGION NO./T  ***
                    1          1          2          3          3

          ***X-REGION NO./R  ***
                    1          1          1

          ***MATERIAL NO./R  ***
                    2          4          1

          ***R-X DIVISION    ***
           0.0          0.40000E+00 0.50000E+00 0.60000E+00 0.80000E+00 0.90270E+00

 STRAGE USED   2625 FROM  60000

          VOLUME OF (S)-REGION
            1)5.02654E-01  2)2.82743E-01  3)3.45575E-01  4)8.79646E-01  5)5.49360E-01
          VOLUME OF (T)-REGION
            1)5.02654E-01  2)2.82743E-01  3)3.45575E-01  4)8.79646E-01  5)5.49360E-01
          VOLUME OF (R)-REGION
            1)7.85398E-01  2)3.45575E-01  3)1.42901E+00
          VOLUME OF (X)-REGION
            1)2.55998E+00

          RATIO OF VOLUMES OF T-REGIONS NUMERICALLY INTEGRATED TO ANALYTIC
            1)1.000269     2)0.999087     3)0.999676     4)0.999628     5)0.999761


             30 LINES DRAWN *** ELAPSED TIME     11 SEC

     NO. OF MATERIALS USED IN THE CASE =  4
```

```
   ***  BARE REACTOR SPECTRUM CALCULATION *** OF STEP 15 ***          BURN STEP = 0
***NEAP    ***NEA BENCHMARK SINGLE PIN ROD CELL ; BURN-UP                          ***
*** BY B1 APPROXIMATION ***
RATIO OF THERMAL SOURCE TO REMOVAL  0.10011E+01
NEUTRON SPECTRUM PER LETHARGY
 0.65522E+00 0.25982E+01 0.57266E+01 0.62978E+01 0.62082E+01 0.62750E+01 0.45802E+01 0.37613E+01 0.29652E+01 0.24462E+01
 0.21092E+01 0.19106E+01 0.17711E+01 0.16787E+01 0.16246E+01 0.15844E+01 0.15505E+01 0.15219E+01 0.14982E+01 0.14716E+01
 0.14435E+01 0.14101E+01 0.13526E+01 0.13300E+01 0.13017E+01 0.12293E+01 0.11464E+01 0.11763E+01 0.97420E+00 0.11078E+01
 0.11143E+01 0.10976E+01 0.10637E+01 0.11474E+01 0.12065E+01 0.12304E+01 0.13353E+01 0.14015E+01 0.14741E+01 0.15647E+01
 0.16996E+01 0.18006E+01 0.19613E+01 0.21619E+01 0.24169E+01 0.26477E+01 0.29117E+01 0.31910E+01 0.34189E+01 0.36444E+01
 0.37863E+01 0.38766E+01 0.38965E+01 0.38014E+01 0.36357E+01 0.33573E+01 0.30183E+01 0.26142E+01 0.21713E+01 0.17350E+01
 0.13149E+01 0.93961E+00 0.62576E+00 0.38300E+00 0.20828E+00 0.97948E-01 0.36013E-01 0.94006E-02 0.12004E-02 0.28307E-04
NEUTRON CURRENT PER LETHARGY
 0.47004E-03 0.15459E-02 0.30813E-02 0.28722E-02 0.20527E-02 0.18733E-02 0.11062E-02 0.85757E-03 0.64060E-03 0.49365E-03
 0.39853E-03 0.35227E-03 0.30908E-03 0.28889E-03 0.26879E-03 0.26080E-03 0.25248E-03 0.25026E-03 0.24684E-03 0.24290E-03
 0.23556E-03 0.23047E-03 0.23324E-03 0.21086E-03 0.20864E-03 0.20158E-03 0.19337E-03 0.19181E-03 0.17008E-03 0.18204E-03
 0.18309E-03 0.18026E-03 0.17541E-03 0.89817E-04 0.93291E-04 0.93649E-04 0.99233E-04 0.10232E-03 0.10627E-03 0.11124E-03
 0.11842E-03 0.12336E-03 0.13235E-03 0.14449E-03 0.16000E-03 0.17304E-03 0.18819E-03 0.20280E-03 0.21345E-03 0.22312E-03
 0.22733E-03 0.22726E-03 0.22187E-03 0.20946E-03 0.19341E-03 0.17181E-03 0.14830E-03 0.12253E-03 0.94862E-04 0.71135E-04
 0.50186E-04 0.32702E-04 0.19756E-04 0.11052E-04 0.52422E-05 0.22484E-05 0.64267E-06 0.12968E-06 0.12021E-07 0.14017E-09
FAST FISSION              0.24107E+00
FAST ABSORPTION           0.34167E+00
FAST LEAKAGE              0.10429E-01
TOTAL FISSION             0.13164E+01
TOTAL ABSORPTION          0.98910E+00
TOTAL LEAKAGE             0.10895E-01
K-EFF=1.3164 KINF=1.3309 UNDER GEOMTRICAL BUCKLING= 0.20000E-03

          COAR    WHOL ENERGY GROUP STRUCTURE (M=1.67482E-24 GRAM,  EV=1.60210E-12 ERG)

 GROUP          ENERGY RANGE (EV)                VELOCITY RANGE (CM/SEC)          LETHARGY RANGE
   1       0.10000E+08    0.20347E+04       0.43740E+10    0.62391E+08        0.0        8.5000
   2       0.20347E+04    0.68256E+00       0.62391E+08    0.11427E+07        8.5000    16.5000
   3       0.68256E+00    0.10010E-04       0.11427E+07    0.43761E+04       16.5000    27.6300


   ***  BARE REACTOR SPECTRUM CALCULATION *** OF STEP 15 ***          BURN STEP = 1
***NEAP    ***NEA BENCHMARK SINGLE PIN ROD CELL ; BURN-UP                          ***
*** BY B1 APPROXIMATION ***
RATIO OF THERMAL SOURCE TO REMOVAL  0.10011E+01
NEUTRON SPECTRUM PER LETHARGY
 0.65870E+00 0.26054E+01 0.57342E+01 0.63018E+01 0.62109E+01 0.62747E+01 0.45810E+01 0.37616E+01 0.29654E+01 0.24465E+01
 0.21095E+01 0.19109E+01 0.17714E+01 0.16790E+01 0.16250E+01 0.15848E+01 0.15509E+01 0.15224E+01 0.14987E+01 0.14723E+01
 0.14442E+01 0.14110E+01 0.13534E+01 0.13316E+01 0.13044E+01 0.12325E+01 0.11495E+01 0.11810E+01 0.97723E+00 0.11115E+01
 0.11181E+01 0.11018E+01 0.10688E+01 0.11518E+01 0.12098E+01 0.12309E+01 0.13300E+01 0.13880E+01 0.14519E+01 0.15300E+01
 0.16476E+01 0.17302E+01 0.18771E+01 0.20697E+01 0.23186E+01 0.25417E+01 0.27949E+01 0.30618E+01 0.32763E+01 0.34883E+01
 0.36171E+01 0.36960E+01 0.37073E+01 0.36087E+01 0.34474E+01 0.31818E+01 0.28616E+01 0.24799E+01 0.20610E+01 0.16483E+01
 0.12501E+01 0.89391E+00 0.59568E+00 0.36484E+00 0.19846E+00 0.93377E-01 0.34305E-01 0.89565E-02 0.11439E-02 0.26986E-04
NEUTRON CURRENT PER LETHARGY
 0.46663E-03 0.15390E-02 0.30809E-02 0.28479E-02 0.20135E-02 0.19419E-02 0.10760E-02 0.87177E-03 0.64585E-03 0.49636E-03
 0.40009E-03 0.35312E-03 0.30970E-03 0.28926E-03 0.26919E-03 0.26108E-03 0.25264E-03 0.24957E-03 0.24797E-03 0.24312E-03
 0.23578E-03 0.23082E-03 0.21125E-03 0.23674E-03 0.20809E-03 0.20359E-03 0.19441E-03 0.19292E-03 0.17114E-03 0.18264E-03
 0.18381E-03 0.18112E-03 0.17664E-03 0.89449E-04 0.92675E-04 0.92602E-04 0.97710E-04 0.10010E-03 0.10303E-03 0.10641E-03
 0.11242E-03 0.11511E-03 0.12318E-03 0.13448E-03 0.14978E-03 0.16367E-03 0.17772E-03 0.19140E-03 0.20168E-03 0.21090E-03
 0.21371E-03 0.21241E-03 0.20767E-03 0.19634E-03 0.18146E-03 0.16163E-03 0.13838E-03 0.11473E-03 0.89186E-04 0.67465E-04
 0.47357E-04 0.31381E-04 0.18965E-04 0.10565E-04 0.50097E-05 0.21489E-05 0.61349E-06 0.12377E-06 0.10429E-07 0.13376E-09
FAST FISSION              0.23605E+00
FAST ABSORPTION           0.33922E+00
FAST LEAKAGE              0.10431E-01
TOTAL FISSION             0.12632E+01
TOTAL ABSORPTION          0.98913E+00
TOTAL LEAKAGE             0.10872E-01
K-EFF=1.2632 KINF=1.2771 UNDER GEOMTRICAL BUCKLING= 0.20000E-03


                                  --------------------------------
                                  RESULT OF DEPLETION CALCULATION
                                  --------------------------------

  DAYS  4.99890E+01 9.99946E+01 1.50022E+02 2.00066E+02 2.50126E+02 3.00198E+02 3.50281E+02 4.00374E+02 4.50475E+02 5.00587E+02

   MWD  6.91940E-03 1.38388E-02 2.07582E-02 2.76776E-02 3.45970E-02 4.15164E-02 4.84358E-02 5.53552E-02 6.22746E-02 6.91940E-02

U235-%  4.09682E-02 8.03340E-02 1.18280E-01 1.54951E-01 1.90451E-01 2.24862E-01 2.58249E-01 2.90666E-01 3.22154E-01 3.52743E-01


 ( MATERIAL NO.= 2  VOLUME= 7.85398E-01 CM-3  FACTOR= 1.00000E+00 )

 ID  NAME     MWD    6.919E-03 1.384E-02 2.076E-02 2.768E-02 3.460E-02 4.152E-02 4.844E-02 5.536E-02 6.227E-02 6.919E-02

---- ----  --------- --------- --------- --------- --------- --------- --------- --------- --------- --------- ---------
   1  U05  6.777E-04 6.499E-04 6.233E-04 5.975E-04 5.727E-04 5.486E-04 5.253E-04 5.027E-04 4.807E-04 4.594E-04 4.386E-04
   2  U06  0.0       4.764E-06 9.350E-06 1.375E-05 1.798E-05 2.206E-05 2.599E-05 2.978E-05 3.344E-05 3.697E-05 4.037E-05
   3  U08  2.164E-02 2.163E-02 2.161E-02 2.160E-02 2.159E-02 2.158E-02 2.156E-02 2.155E-02 2.154E-02 2.153E-02 2.151E-02
   4  PU9  0.0       9.815E-06 1.867E-05 2.645E-05 3.332E-05 3.938E-05 4.473E-05 4.945E-05 5.361E-05 5.727E-05 6.046E-05
   5  PU0  0.0       2.481E-07 9.213E-07 1.920E-06 3.165E-06 4.597E-06 6.170E-06 7.850E-06 9.609E-06 1.142E-05 1.341E-05
   6  PU1  0.0       8.997E-09 6.649E-08 2.069E-07 4.475E-07 7.924E-07 1.240E-06 1.785E-06 2.417E-06 3.126E-06 3.774E-06
   7  PU2  0.0       7.219E-11 1.076E-09 5.083E-09 1.495E-08 3.387E-08 6.512E-08 1.119E-07 1.773E-07 2.641E-07 3.730E-07
   8  XE5  0.0       6.101E-09 6.154E-09 6.144E-09 6.120E-09 6.085E-09 6.042E-09 5.991E-09 5.935E-09 5.873E-09 5.799E-09
   9  SM9  0.0       5.229E-08 5.331E-08 5.325E-08 5.300E-08 5.262E-08 5.213E-08 5.156E-08 5.091E-08 5.021E-08 4.940E-08
  10  F5N  0.0       3.699E-05 7.262E-05 1.070E-04 1.403E-04 1.727E-04 2.040E-04 2.345E-04 2.642E-04 2.931E-04 3.212E-04
  11  F5S  0.0       9.019E-06 1.758E-05 2.574E-05 3.351E-05 4.093E-05 4.801E-05 5.477E-05 6.123E-05 6.740E-05 7.328E-05
  12  F5R  0.0       8.031E-08 1.168E-07 1.315E-07 1.356E-07 1.345E-07 1.310E-07 1.264E-07 1.213E-07 1.162E-07 1.110E-07
  13  F9N  0.0       6.811E-07 2.631E-06 5.718E-06 9.824E-06 1.486E-05 2.076E-05 2.747E-05 3.494E-05 4.314E-05 5.205E-05
  14  F9S  0.0       1.772E-07 6.813E-07 1.473E-06 2.518E-06 3.790E-06 5.266E-06 6.931E-06 8.768E-06 1.076E-05 1.291E-05
  15  F9R  0.0       3.410E-10 9.901E-10 1.670E-09 2.300E-09 2.865E-09 3.367E-09 3.815E-09 4.213E-09 4.569E-09 4.883E-09
```


R-REG= 1
FPINFOF2
600.0K
R-REG= 2
CLADFCL2
600.0K
R-REG= 3
LWTRFLW2
600.0K
NEUTRON SPECTRA
NEAP
ARBITRARY UNIT
ENERGY (EV)

IX.2 Plane Cell and Core Calculation

The following is a part of input data of the IAEA benchmark calculations which were carried out to validate the computer codes of the participants of the Specialist Meetings on the enrichment reduction proposals for their individual research reactors. The calculations are based on an idealized model of a MTR type 10 MW research reactor. The calculations requested are aimed at providing a through test of core physics computing code and data in several areas of relevance to enrichment reduction.

The first case is for a cell of the standard fuel element composed of a fuel meat, cladding, coolant water, and a pseudo mixture corresponding to side plate and water gap. The second case is for a smeared cross sections of a part of control element. A super cell of a fork type absorber plate and adjacent three fuel plates is solved. The third case is for a core calculation by two dimensional diffusion calculation.

```
*********1*********2*********3*********4*********5*********6*********7*
HS11
IAEA BENCHMARK HEU  STANDARD ELEMENT WITH CELL BURN-UP
 1 1 1 1 1  1 4 0 2 1  0 0 0 0 1  2 1 0 0 1 / SRAC CONTROL
 .00120 /  TEMPORARY BUCKLING
J1480            0 0/
J1480            0 0/
J1480            0 0/ NOT EFFECTIVE
J1480            0 0/ NOT EFFECTIVE
22 31 5  5  /
7(2) 6(4) 8(3) 4 / FINE ENERGY GROUP STRUCTUREFAST
2(4) 2 2 1 1 2 2 23(1)/ FINE ENERGY GROUP STRUCTURE THERMAL
 4 4 4 5 5 /
 6 6 6 6 7 /
XH01H001
  *
  *
(Insert possible combination of nuclides )
  *
  *
XC02C001

 2  7  4  4  1 1 7 1 0 0 5 0 1 2 0 0 1 0 /PATH 1-D SLAB 0 10 20 5 5 5 0
 0.0001 0.0001 0.01 1.2 100. 0.8/
 4  3  2  1  2  3  4 /T BY SUB
 1 1 1 1 / X BY R
 1 2 3 4/ M BY R
 0. 0.04 1*0.1115 1*0.038 1*.051 1*.038 1*.1115 1*0.04/ RX
  4 / NMAT
FH1DXD1X  0  3  300.  0.102   0.    / MAT1 FUEL MEAT
XAL70001  0  0  0.057011   /
XU050001  2  0  1.61790E-3 /
XU080001  2  0  1.20050E-4 /
CLADXCLC  0  1  300.    0.076  0.72/ MAT2 CLADDING
XAL70001  0  0  0.06023
LWTRXLWX  0  2  300.    0.223  0.99/ MAT3 H2O
XH01H001  0  0  0.06692
XO060001  0  0  0.03346
```

```
EXTRAEXR  0  3  300.    0.08   1.  / MAT4 EXTRA H2O+AL
XAL70001  0  0  0.05091
XH01H001  0  0  0.01673
XO060001  0  0  0.00837
  6 1 0  / FOR BURN UP
 10.0 2.10361E+05 / REACTOR POWER, TOTAL FUEL VOLUME BY CELL WIDTH
       0.05 0.1 0.25 0.3 0.45 0.5 / BURN-UP STEP BY U-235 %
HC25
 BOC PARTIAL CONTROL ELEMENT 25% BURNUP
 1 1 0 1 2  1 3 0 2 1  0 0 0 0 1  2 1 0 0 0 / SRAC CONTROL
 .00120 /  TEMPORARY BUCKLING
 2 20 20 20 1 0 20 1 0 0 1 0 1 2 0 0 1 0 /PATH 1-D SLAB 0 10 20 5 5 5 0
 0.0001  0.0001 0.01 0.9 100. 0.7/ FOR STRONG ABSORBER 13(1) 7(0)/ X BY
 R
 3 3(1 3) 2  3 4 3 2 3 3(3 1) 3 / M BY R
 0. 0.1115 1*0.127 1*.223 $
           1*0.127 1*.223 $
           1*0.127 1*.223  1*0.127 1*0.1276 1*0.318 $
  1*0.1276 1*0.127 2*0.1115 1*0.127 $
                    1*0.223 1*0.127 $
                    1*0.223 1*0.127 $
                    1*0.1115 / RX
  4 / NMAT
FH25XCCX 0 15 300. 0.102 0.889 / HOMOGENIZED FUEL PLATE 25 % BURNT
XAL70001 0 0 0.05893 /
XU050001  2  0  0.487  E-3 /
XU080001  2  0  0.4731 E-4 /
XPU90001  2  0  0.6450 E-6 /
XPU00001  2  0  0.6478 E-7 /
XPU10001  0  0  1.3023 E-8 /
XPU20001  2  0  0.6221 E-9 /
XXE50001  0  0  0.5566 E-8 /
XSM90001  0  0  0.4361 E-7 /
XF5N0001  0  0  2.088  E-4 /
XF5S0001  0  0  0.4924 E-4 /
XF5R0001  0  0  0.8811 E-7 /
XF9N0001  0  0  2.464  E-7 /
XF9S0001  0  0  0.6257 E-7 /
XF9R0001  0  0  2.806 E-10 /
CLADXCLX  0  1  300.  0.  1.  / CLADING; ALREADY DEFINED INTHE PREVIOUS
XAL70001 0 0 0.06023 / CASE, REQUIRED FOR PEACO ROUTINE
LWTRXLWX  0  2  300. 0. 1. / WATER; THE SAME AS ABOVE
XH01H001  0  0  0.06692
XO060001  0  0  0.03346
ABSBXAAX 0 2 300 0. 1.  /  TO  BE  REPLACED  BY  CONTROL  ROD  MATERIAL
XH01H001 0 0 0.06692
XO060001  0  0  0.03346
 0 / PEACO
HBOL
HEU BOC CORE CALCULATION BY 2-D DIFFUSION
 0 0 0 1 0  0 0 0 0 1  0 5 0 0 1  2 0 0 0 0 / SRAC CONTROL
 .00120 /  TEMPORARY BUCKLING
 7 7 1/ CIT1
IAEA BENCHMARK PROBLEM  *HIGH ENRICHMENT FUEL BEGINNING OF CRE
10 GROUP,A QUADRANT SOLVED, CITATION INPUT FOLLOWS
001

  1     1  1                          0  0  1
```

```
100
 1.5
003
                6                   1  1  0  0  0  0
 0
 0.
004
  2  2.5       1  1.35     4  7.7       4  7.7       4  7.7      10 23.1
  0
  2  2.5       1  1.55     2  4.05      1  2.1       2  3.9       1  2.1
  4  8.1      10  24.3     0
005
  4  7  1  2  5  4
  7  7  1  2  5  4
  1  1  1  2  5  4
  1  1  6  3  5  4
  1  1  2  3  5  4
  1  1  6  3  5  4
  2  2  3  4  4  4
  4  4  4  4  4  4
008
-24 24 13
024
  1  0.00171
999

 1 6*1  / MATERIAL NO.
  4  7  1  2  5  4    $ R-X REGION
  7  7  1  2  5  4    $
  1  1  1  2  5  4    $
  1  1  6  3  5  4    $
  1  1  2  3  5  4    $
  1  1  6  3  5  4    $
  2  2  3  4  4  4    $
  4  4  4  4  4  4    /
  7  /NMAT
HS11X51X  0  0  300.    0.      1. / FUEL CELL 45 % BURNT
HS11X31X  0  0  300.    0.      1. / FUEL CELL 25 % BURNT
HS11X11X  0  0  300.    0.      1. / FUEL CELL  5 % BURNT
RELHXHHX  0  2  300.    0.      1. / LIGHT WATER FOR REFLECTOR
X0060001  0  0  0.03346
XH01H001  0  0  0.06692
GRAPX00X  0  1  300.    0.      1. / GRAPHITE REFLECTOR
XC02C001  0  0  0.08532            /
TRAPXALX  0  1  300.   0.   1.     /    FLUX TRAP
XAL70001  0  0  0.06023
HC25X01X  0  0  300.    0.      1. /


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     **** CASE ID === HS11     *** TITLE === IAEA BENCHMARK HEU  STANDARD CELL BURN-UP                              ***

    USE OF COLLISION PROB. ROUTINE (NO,CALL:0,1)                1
    SELECT OF ROUTINE FOR FIX S CALC(NO,PIJ,1D SN ,2D SN ,
     1D DIFF,2D DIFF:0,1,2,3,4,5)                               1
    DANCOFF FACTOR BY(INPUT,PIJ,FORMULA:0,1,2)                  1
    CALC OF THERMAL RANGE (EXCLUDED,INCLUDED:0,1)               1
    PROCESS OF RESONANCE 2 RANGE(TABLELOOK,IR,PEACO:0,1,2)      1
    FLUX CALCULATION BY RANGES(SKIP,CALC:0,1)                   1
    FAST RANGES FOR TRANSPORT CALC(0,1,2,3,4)                   4
    UPDATE OF MCROSS FILE (NO,CALL:0,1)                         0
    CALC OF BARE REACTOR SPECTRUM(SKIP,P1,B1:0,1,2)             2
    COLLAPSE MACRO X-SECTION BY RANGE(SKIP,CALL:0,1)            1
    INPUT OF GEOMETRY (NEW,SAME AS PREVIOUS CASE:0,1)           0
    SELECT ROUTINE FOR EIGENVALUE CALC(NO,PIJ,1D SN,2D SN,
     1D DIFF,2D DIFF:0,1,2,3,4,5)                               0
    COLLAPSE MACRO AFTER EIGENVALUE CALC(SKIP,CALL:0,1)         0
    PREPARE MACRO FOR CITATION (SKIP,CALL:0,1)                  0
    TOTAL MICRO X-SECTION BY (HARMONIC,ARITHMETIC AV:1,2)       1
    TRANSPORT MACRO X-SECTION BY (PO,P1,B1,SN:0,1,2,3)          2
    DIFF COEF (INVERSE TR,BENOIST ISO,BENOIST ANISO:1,2,3)      1
    PLOT OF FLUX MACRO X-SECTION (1+2)                          0
    PRINT OF MACRO-X (0,1,2)                                    0
    BURN-UP CALCULATION (0,1:SKIP,EXECUTE)                      1
    NUMBER OF USER'S FAST GROUPS                               22
    NUMBER OF USER'S THERMAL GROUPS                            31
    NUMBER OF USER'S CONDENSED FAST GROUPS                      5
    NUMBER OF USER'S CONDENSED THERMAL GROUPS                   5

    NUMBER OF LIBRARY GROUPS IN THE USER'S FAST GROUP
              2          2          2          2          2          2          2          4          4          4

              4          4          4          3          3          3          3          3          3          3

              3          4
    NUMBER OF LIBRARY GROUPS IN THE USER'S THERMAL GROUPS
              4          4          2          2          1          1          2          2          1          1

              1          1          1          1          1          1          1          1          1          1

              1          1          1          1          1          1          1          1          1          1

              1
    NUMBER OF USER'S GROUPS IN THE CONDENSED FAST GROUP
              4          4          4          5          5
    NUMBER OF USER'S GROUPS IN THE CONDENSED THERMAL GROUP
              6          6          6          6          7


HS11    IAEA BENCHMARK HEU  STANDARD CELL BURN-UP                              *** STEP *** INPUT FOR PIJ ***
                  GEOMETRY TYPE                                 2         ( INFINITE SLAB )
                  NUMBER OF SUB - REGIONS                       7
                  NUMBER OF  T  - REGIONS                       4
                  NUMBER OF  R  - REGIONS                       4
                  NUMBER OF  X  - REGIONS                       1
                  OUTER BOUNDARY CONDITION (-1,0,1,2)           1
                  DIRECTIONAL PIJ (1,2)                         1  INDICATED BY IC17
                  NUMBER OF R OR X MESH                         7
                  NUMBER OF THETA OR Y MESH                     1
                  TOTAL NUMBER OF PIN RODS                      0
                  NUMBER OF RINGS OF PIN ROD ARRAY              0
                  NUMBER OF LATTICE CELLS TRACED                5
                  PRINT CONTROL OF PIJ (0,1)                    0
                  ORDER OF GAUSS RADIAL INTEGRATION             1
                  NO. OF DIVISION FOR ANGULAR INTEGRATION       2
                  NUMBER OF ANNULAR DIVISION IN A PIN ROD       0
                  DIVISION BY RPP  (0,1,2)                      0
                  ANGLE RANGE BY DEGREE                         1
                  PLOTER OPTION  (0,1)                          0

         ===T-REGION NO./SUB===
                  4          3          2          1          2          3          4

         ===X-REGION NO./R  ===
                  1          1          1          1

         ===MATERIAL NO./R  ===
                  1          2          3          4

         ===R-X DIVISION    ===
          0.0         0.40000E-01 0.15150E+00 0.18950E+00 0.24050E+00 0.27850E+00 0.39000E+00 0.43000E+00

 STRAGE USED   7431 FROM  60000

         VOLUME OF (S)-REGION
           1)4.00000E-02  2)1.11500E-01  3)3.80000E-02  4)5.09999E-02  5)3.80000E-02  6)1.11500E-01  7)4.00000E-02
         VOLUME OF (T)-REGION
           1)5.09999E-02  2)7.60000E-02  3)2.23000E-01  4)7.99999E-02
         VOLUME OF (X)-REGION
           1)4.30000E-01

         RATIO OF VOLUMES OF T-REGIONS NUMERICALLY INTEGRATED TO ANALYTIC
           1)1.000000     2)1.000000     3)0.999999     4)1.000000


             2 LINES DRAWN *** ELAPSED TIME     13 SEC

    NO. OF MATERIALS USED IN THE CASE =  4
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   ***  BARE REACTOR SPECTRUM CALCULATION *** OF STEP 15 ***          BURN STEP = 0
***HS11     ***IAEA BENCHMARK HEU  STANDARD CELL BURN-UP                          ***
*** BY B1 APPROXIMATION ***
RATIO OF THERMAL SOURCE TO REMOVAL  0.10004E+01
NEUTRON SPECTRUM PER LETHARGY
 0.55506E+00 0.23063E+01 0.48932E+01 0.53833E+01 0.48249E+01 0.41493E+01 0.31564E+01 0.22496E+01 0.15807E+01 0.13288E+01
 0.11422E+01 0.11075E+01 0.10831E+01 0.10701E+01 0.10571E+01 0.10432E+01 0.10251E+01 0.10185E+01 0.99683E+00 0.97103E+00
 0.96591E+00 0.95501E+00 0.97620E+00 0.10444E+01 0.10523E+01 0.10846E+01 0.10835E+01 0.11020E+01 0.11636E+01 0.12652E+01
 0.14189E+01 0.15322E+01 0.17106E+01 0.19001E+01 0.21541E+01 0.24474E+01 0.27265E+01 0.30084E+01 0.32144E+01 0.33424E+01
 0.33274E+01 0.31540E+01 0.28461E+01 0.24136E+01 0.19099E+01 0.13939E+01 0.92378E+00 0.53948E+00 0.26850E+00 0.10644E+00
 0.29255E-01 0.39293E-02 0.97376E-04
NEUTRON CURRENT PER LETHARGY
 0.22931E-02 0.78960E-02 0.14710E-01 0.13697E-01 0.98290E-02 0.75696E-02 0.49741E-02 0.30772E-02 0.19301E-02 0.14982E-02
 0.12948E-02 0.12214E-02 0.11836E-02 0.11601E-02 0.11383E-02 0.11178E-02 0.10911E-02 0.10874E-02 0.10596E-02 0.10190E-02
 0.10141E-02 0.10008E-02 0.39629E-03 0.41001E-03 0.38755E-03 0.38666E-03 0.37577E-03 0.37602E-03 0.38931E-03 0.41209E-03
 0.45038E-03 0.47648E-03 0.51983E-03 0.56319E-03 0.62170E-03 0.68626E-03 0.74047E-03 0.78918E-03 0.81232E-03 0.80812E-03
 0.76417E-03 0.68335E-03 0.57781E-03 0.45497E-03 0.33434E-03 0.22540E-03 0.13735E-03 0.74065E-04 0.34010E-04 0.12058E-04
 0.28310E-05 0.27684E-06 0.35855E-08
FAST FISSION              0.12030E+00
FAST ABSORPTION           0.90909E-01
FAST LEAKAGE              0.52736E-01
TOTAL FISSION             0.16465E+01
TOTAL ABSORPTION          0.94491E+00
TOTAL LEAKAGE             0.55094E-01
K-EFF=1.6465 KINF=1.7425 UNDER GEOMTRICAL BUCKLING= 0.12000E-02

          COAR    WHOL ENERGY GROUP STRUCTURE (M=1.67482E-24 GRAM,  EV=1.60210E-12 ERG)

 GROUP          ENERGY RANGE (EV)                VELOCITY RANGE (CM/SEC)          LETHARGY RANGE
    1       0.10000E+08    0.13534E+07       0.43740E+10    0.16091E+10        0.0        2.0000
    2       0.13534E+07    0.11109E+06       0.16091E+10    0.46101E+09        2.0000     4.5000
    3       0.11109E+06    0.20347E+04       0.46101E+09    0.62391E+08        4.5000     8.5000
    4       0.20347E+04    0.37266E+02       0.62391E+08    0.84437E+07        8.5000    12.5000
    5       0.37266E+02    0.11254E+01       0.84437E+07    0.14673E+07       12.5000    16.0000
    6       0.11254E+01    0.27699E+00       0.14673E+07    0.72796E+06       16.0000    17.4019
    7       0.27699E+00    0.13700E+00       0.72796E+06    0.51196E+06       17.4019    18.1059
    8       0.13700E+00    0.64017E-01       0.51196E+06    0.34996E+06       18.1059    18.8667
    9       0.64017E-01    0.18467E-01       0.34996E+06    0.18796E+06       18.8667    20.1099
   10       0.18467E-01    0.10010E-04       0.18796E+06    0.43761E+04       20.1099    27.6300

   ***  BARE REACTOR SPECTRUM CALCULATION *** OF STEP 15 ***          BURN STEP = 1
***HS11     ***IAEA BENCHMARK HEU  STANDARD CELL BURN-UP                          ***
*** BY B1 APPROXIMATION ***
RATIO OF THERMAL SOURCE TO REMOVAL  0.10004E+01
NEUTRON SPECTRUM PER LETHARGY
 0.55528E+00 0.23069E+01 0.48943E+01 0.53838E+01 0.48258E+01 0.41491E+01 0.31577E+01 0.22500E+01 0.15818E+01 0.13300E+01
 0.11423E+01 0.11082E+01 0.10837E+01 0.10708E+01 0.10581E+01 0.10443E+01 0.10265E+01 0.10210E+01 0.99965E+00 0.97088E+00
 0.96761E+00 0.95632E+00 0.97861E+00 0.10479E+01 0.10566E+01 0.10896E+01 0.10890E+01 0.11076E+01 0.11694E+01 0.12703E+01
 0.14226E+01 0.15338E+01 0.17092E+01 0.18939E+01 0.21413E+01 0.24259E+01 0.26952E+01 0.29684E+01 0.31688E+01 0.32946E+01
 0.32809E+01 0.31116E+01 0.28100E+01 0.23848E+01 0.18886E+01 0.13794E+01 0.91481E+00 0.53456E+00 0.26620E+00 0.10557E+00
 0.29028E-01 0.38993E-02 0.96598E-04
NEUTRON CURRENT PER LETHARGY
 0.22915E-02 0.79048E-02 0.14715E-01 0.13686E-01 0.98330E-02 0.75579E-02 0.49839E-02 0.30788E-02 0.19358E-02 0.15037E-02
 0.12934E-02 0.12245E-02 0.11848E-02 0.11617E-02 0.11402E-02 0.11199E-02 0.10935E-02 0.10924E-02 0.10644E-02 0.10157E-02
 0.10170E-02 0.10023E-02 0.39764E-03 0.41193E-03 0.38970E-03 0.38916E-03 0.37837E-03 0.37864E-03 0.39207E-03 0.41447E-03
 0.45203E-03 0.47733E-03 0.51963E-03 0.56135E-03 0.61757E-03 0.67849E-03 0.72981E-03 0.77641E-03 0.79794E-03 0.79384E-03
 0.75236E-03 0.67282E-03 0.56907E-03 0.45010E-03 0.33068E-03 0.22302E-03 0.13596E-03 0.73340E-04 0.33636E-04 0.12080E-04
 0.27979E-05 0.28758E-06 0.35530E-08
FAST FISSION              0.11482E+00
FAST ABSORPTION           0.89623E-01
FAST LEAKAGE              0.52774E-01
TOTAL FISSION             0.15505E+01
TOTAL ABSORPTION          0.94488E+00
TOTAL LEAKAGE             0.55117E-01
K-EFF=1.5505 KINF=1.6410 UNDER GEOMTRICAL BUCKLING= 0.12000E-02

                                  ---------------------------------
                                  RESULT OF DEPLETION CALCULATION
                                  ---------------------------------

  DAYS  2.62457E+01 5.25105E+01 1.33064E+02 1.59548E+02 2.41315E+02 2.68124E+02

   MWD  2.60274E+02 5.20622E+02 1.30333E+03 1.56522E+03 2.35282E+03 2.61682E+03

U235-%  5.00000E-02 1.00000E-01 2.50000E-01 3.00000E-01 4.50000E-01 5.00000E-01

 ( MATERIAL NO.= 1  VOLUME= 5.09999E-02 CM-3  FACTOR= 2.10361E+05 )

ID  NAME    U235-%  5.000E-02 1.000E-01 2.500E-01 3.000E-01 4.500E-01 5.000E-01
--- ----  --------- --------- --------- --------- --------- --------- --------- --------- --------- ---------
  1  U05  1.618E-03 1.537E-03 1.456E-03 1.213E-03 1.133E-03 8.900E-04 8.092E-04
  2  U06  0.0       1.344E-05 2.683E-05 6.633E-05 7.917E-05 1.169E-04 1.290E-04
  3  U08  1.200E-04 1.196E-04 1.192E-04 1.177E-04 1.172E-04 1.155E-04 1.149E-04
  4  PU9  0.0       4.059E-07 7.845E-07 1.703E-06 1.926E-06 2.409E-06 2.434E-06
  5  PU0  0.0       7.894E-09 3.055E-08 1.644E-07 2.244E-07 4.230E-07 5.022E-07
  6  PU1  0.0       3.731E-10 2.967E-09 4.099E-08 6.549E-08 1.830E-07 2.195E-07
  7  PU2  0.0       2.955E-12 4.910E-11 1.962E-09 4.103E-09 2.072E-08 3.159E-08
  8  XE5  0.0       1.675E-08 1.608E-08 1.399E-08 1.287E-08 1.066E-08 9.441E-09
  9  SM9  0.0       1.344E-07 1.285E-07 1.093E-07 1.002E-07 8.115E-08 7.163E-08
 10  F5N  0.0       1.038E-04 2.076E-04 5.189E-04 6.228E-04 9.344E-04 1.038E-03
 11  F5S  0.0       2.530E-05 5.019E-05 1.223E-04 1.453E-04 2.108E-04 2.314E-04
 12  F5R  0.0       1.877E-07 2.389E-07 2.218E-07 2.025E-07 1.587E-07 1.392E-07
 13  F9N  0.0       2.494E-08 1.017E-07 6.527E-07 9.495E-07 2.218E-06 2.779E-06
 14  F9S  0.0       6.484E-09 2.629E-08 1.656E-07 2.393E-07 5.454E-07 6.770E-07
 15  F9R  0.0       1.050E-11 2.868E-11 7.550E-11 8.715E-11 1.137E-10 1.140E-10
```

```
**** CASE ID === HC25     *** TITLE === IAEA BENCHMARK PROBLEM HEU BOC CONTROL  25% BURNUP                    ***

   USE OF COLLISION PROB. ROUTINE (NO,CALL:0,1)            1
   SELECT OF ROUTINE FOR FIX S CALC(NO,PIJ,1D SN ,2D SN ,
    1D DIFF,2D DIFF:0,1,2,3,4,5)                           1
   DANCOFF FACTOR BY(INPUT,PIJ,FORMULA:0,1,2)              0
   CALC OF THERMAL RANGE (EXCLUDED,INCLUDED:0,1)           1
   PROCESS OF RESONANCE 2 RANGE(TABLELOOK,IR,PEACO:0,1,2)  2
   FLUX CALCULATION BY RANGES(SKIP,CALC:0,1)               1
   FAST RANGES FOR TRANSPORT CALC(0,1,2,3,4)               3
   UPDATE OF MCROSS FILE (NO,CALL:0,1)                     0
   CALC OF BARE REACTOR SPECTRUM(SKIP,P1,B1:0,1,2)         2
   COLLAPSE MACRO X-SECTION BY RANGE(SKIP,CALL:0,1)        1
   INPUT OF GEOMETRY (NEW,SAME AS PREVIOUS CASE:0,1)       0
   SELECT ROUTINE FOR EIGENVALUE CALC(NO,PIJ,1D SN,2D SN,
    1D DIFF,2D DIFF:0,1,2,3,4,5)                           0
   COLLAPSE MACRO AFTER EIGENVALUE CALC(SKIP,CALL:0,1)     0
   PREPARE MACRO FOR CITATION (SKIP,CALL:0,1)              0
   TOTAL MICRO X-SECTION BY (HARMONIC,ARITHMETIC AV:1,2)   1
   TRANSPORT MACRO X-SECTION BY (PO,P1,B1,SN:0,1,2,3)      2
   DIFF COEF (INVERSE TR,BENOIST ISO,BENOIST ANISO:1,2,3)  1
   PLOT OF FLUX MACRO X-SECTION (1+2)                      0
   PRINT OF MACRO-X (0,1,2)                                0
   BURN-UP CALCULATION (0,1:SKIP,EXECUTE)                  0

HC25    IAEA BENCHMARK PROBLEM HEU BOC CONTROL  25% BURNUP                    *** STEP *** INPUT FOR PIJ ***
                GEOMETRY TYPE                              2        ( INFINITE SLAB )
                NUMBER OF SUB - REGIONS                   20
                NUMBER OF  T  - REGIONS                   20
                NUMBER OF  R  - REGIONS                   20
                NUMBER OF  X  - REGIONS                    1
                OUTER BOUNDARY CONDITION (-1,0,1,2)        0
                DIRECTIONAL PIJ (1,2)                      1  INDICATED BY IC17
                NUMBER OF R OR X MESH                     20
                NUMBER OF THETA OR Y MESH                  1
                TOTAL NUMBER OF PIN RODS                   0
                NUMBER OF RINGS OF PIN ROD ARRAY           0
                NUMBER OF LATTICE CELLS TRACED             1
                PRINT CONTROL OF PIJ (0,1)                 0
                ORDER OF GAUSS RADIAL INTEGRATION          1
                NO. OF DIVISION FOR ANGULAR INTEGRATION    2
                NUMBER OF ANNULAR DIVISION IN A PIN ROD    0
                DIVISION BY RPP  (0,1,2)                   0
                ANGLE RANGE BY DEGREE                      1
                PLOTER OPTION  (0,1)                       0

        ===X-REGION NO./R  ===
                1          1          1          1          1          1          1          1          1
                1          1          1          0          0          0          0          0          0

        ===MATERIAL NO./R  ===
                3          1          3          1          3          1          3          2          3          4
                3          2          3          3          1          3          1          3          1          3

        ===R-X DIVISION    ===
         0.0         0.11150E+00 0.23850E+00 0.46150E+00 0.58850E+00 0.81150E+00 0.93850E+00 0.11615E+01 0.12885E+01 0.14161E+01
         0.17341E+01 0.18617E+01 0.19887E+01 0.21002E+01 0.22117E+01 0.23387E+01 0.25617E+01 0.26887E+01 0.29117E+01 0.30387E+01
         0.31502E+01

 STRAGE USED   2686 FROM  60000

        VOLUME OF (S)-REGION
          1)1.11500E-01  2)1.27000E-01  3)2.23000E-01  4)1.27000E-01  5)2.23000E-01  6)1.27000E-01  7)2.22999E-01  8)1.27000E-01
          9)1.27600E-01 10)3.18000E-01 11)1.27600E-01 12)1.27000E-01 13)1.11500E-01 14)1.11500E-01 15)1.27000E-01 16)2.23000E-01
         17)1.27000E-01 18)2.23000E-01 19)1.27000E-01 20)1.11500E-01
        VOLUME OF (T)-REGION
          1)1.11500E-01  2)1.27000E-01  3)2.23000E-01  4)1.27000E-01  5)2.23000E-01  6)1.27000E-01  7)2.22999E-01  8)1.27000E-01
          9)1.27600E-01 10)3.18000E-01 11)1.27600E-01 12)1.27000E-01 13)1.11500E-01 14)1.11500E-01 15)1.27000E-01 16)2.23000E-01
         17)1.27000E-01 18)2.23000E-01 19)1.27000E-01 20)1.11500E-01
        VOLUME OF (X)-REGION
          1)2.10020E+00
        VOLUME OF (M)-REGION
          1)1.81620E+00  2)7.61999E-01  3)2.54000E-01  4)3.18000E-01

        RATIO OF VOLUMES OF T-REGIONS NUMERICALLY INTEGRATED TO ANALYTIC
          1)1.000000     2)1.000000     3)1.000000     4)1.000000     5)1.000000     6)1.000000     7)0.999999     8)1.000000
          9)1.000000    10)1.000000    11)1.000000    12)1.000000    13)1.000000    14)1.000000    15)1.000000    16)1.000000
         17)1.000000    18)1.000000    19)1.000000    20)1.000000


            2 LINES DRAWN *** ELAPSED TIME     91 SEC

   NO. OF MATERIALS USED IN THE CASE =  4


           ***  BARE REACTOR SPECTRUM CALCULATION *** OF STEP 15 ***
        ***HC25    ***IAEA BENCHMARK PROBLEM HEU BOC CONTROL  25% BURNUP                    ***
        *** BY B1 APPROXIMATION ***
        RATIO OF THERMAL SOURCE TO REMOVAL  0.10002E+01
        NEUTRON SPECTRUM PER LETHARGY
         0.54759E+00 0.22072E+01 0.45618E+01 0.48850E+01 0.43168E+01 0.37137E+01 0.27058E+01 0.18686E+01 0.13005E+01 0.10809E+01
         0.97657E+00 0.94779E+00 0.92997E+00 0.92244E+00 0.91595E+00 0.90872E+00 0.89944E+00 0.88741E+00 0.87784E+00 0.87325E+00
         0.86894E+00 0.86308E+00 0.86729E+00 0.94071E+00 0.96450E+00 0.10083E+01 0.10234E+01 0.10538E+01 0.11409E+01 0.13207E+01
         0.15895E+01 0.18392E+01 0.22033E+01 0.26478E+01 0.32297E+01 0.39173E+01 0.46458E+01 0.53963E+01 0.60538E+01 0.65552E+01
         0.67761E+01 0.66546E+01 0.61891E+01 0.54007E+01 0.43870E+01 0.32790E+01 0.22196E+01 0.13247E+01 0.67200E+00 0.27052E+00
         0.75241E-01 0.10179E-01 0.25171E-03
        NEUTRON CURRENT PER LETHARGY
         0.22565E-02 0.74522E-02 0.13536E-01 0.12258E-01 0.85938E-02 0.67270E-02 0.42302E-02 0.25954E-02 0.15785E-02 0.11746E-02
         0.98084E-03 0.92433E-03 0.89414E-03 0.88042E-03 0.87026E-03 0.86034E-03 0.84755E-03 0.83648E-03 0.82246E-03 0.80981E-03
         0.80099E-03 0.79670E-03 0.29009E-03 0.30534E-03 0.29620E-03 0.30139E-03 0.29881E-03 0.30303E-03 0.32120E-03 0.36082E-03
         0.42238E-03 0.47870E-03 0.55976E-03 0.65652E-03 0.77963E-03 0.91852E-03 0.10548E-02 0.11851E-02 0.12812E-02 0.13308E-02
         0.13072E-02 0.12116E-02 0.10565E-02 0.86008E-03 0.64788E-03 0.44787E-03 0.28087E-03 0.15508E-03 0.73409E-04 0.26879E-04
         0.63562E-05 0.64352E-06 0.82264E-08
        FAST FISSION              0.48713E-01
        FAST ABSORPTION           0.48256E-01
        FAST LEAKAGE              0.44803E-01
        TOTAL FISSION             0.13135E+01
        TOTAL ABSORPTION          0.95208E+00
        TOTAL LEAKAGE             0.47916E-01
        K-EFF=1.3135 KINF=1.3796 UNDER GEOMTRICAL BUCKLING= 0.12000E-02
```

```
**** CASE ID *** HBOL     *** TITLE *** IAEA BENCHMARK PROBLEM HEU BOC CORE                                    ***

 USE OF COLLISION PROB. ROUTINE (NO,CALL:0,1)                0
 SELECT OF ROUTINE FOR FIX S CALC(NO,PIJ,1D SN ,2D SN ,
  1D DIFF,2D DIFF:0,1,2,3,4,5)                               0
 DANCOFF FACTOR BY(INPUT,PIJ,FORMULA:0,1,2)                  0
 CALC OF THERMAL RANGE (EXCLUDED,INCLUDED:0,1)               1
 PROCESS OF RESONANCE 2 RANGE(TABLELOOK,IR,PEACO:0,1,2)      0
 FLUX CALCULATION BY RANGES(SKIP,CALC:0,1)                   0
 FAST RANGES FOR TRANSPORT CALC(0,1,2,3,4)                   0
 UPDATE OF MCROSS FILE (NO,CALL:0,1)                         0
 CALC OF BARE REACTOR SPECTRUM(SKIP,P1,B1:0,1,2)             0
 COLLAPSE MACRO X-SECTION BY RANGE(SKIP,CALL:0,1)            1
 INPUT OF GEOMETRY (NEW,SAME AS PREVIOUS CASE:0,1)           0
 SELECT ROUTINE FOR EIGENVALUE CALC(NO,PIJ,1D SN,2D SN,
  1D DIFF,2D DIFF:0,1,2,3,4,5)                               5
 COLLAPSE MACRO AFTER EIGENVALUE CALC(SKIP,CALL:0,1)         0
 PREPARE MACRO FOR CITATION (SKIP,CALL:0,1)                  0
 TOTAL MICRO X-SECTION BY (HARMONIC,ARITHMETIC AV:1,2)       1
 TRANSPORT MACRO X-SECTION BY (P0,P1,B1,SN:0,1,2,3)          2
 DIFF COEF (INVERSE TR,BENOIST ISO,BENOIST ANISO:1,2,3)      0
 PLOT OF FLUX MACRO X-SECTION (1+2)                          0
 PRINT OF MACRO-X (0,1,2)                                    0
 BURN-UP CALCULATION (0,1:SKIP,EXECUTE)                      0


      STEP - CIT1 -
      NUMBER OF MATERIAL                    7
      NUMBER OF X-REGION                    7
      SELECT DIFF.COEF (1,2:D1,D2)          1

            **********CITATION - REVISION 2 (JULY 1971) - SUPPLEMENT 3 (JULY 1972)**********

                  *********THIS JOB WAS RUN ON  82-12-10  ON THE FACOM M-200  **********



IAEA BENCHMARK PROBLEM NO3 *HIGH ENRICHMENT FUEL
BOL


GENERAL CONTROL INPUT - SECTION 001

   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0
   1   0   1   1   0   0   0   0   0   0   0   0   0   1   0   0   0   0   0   0   0   0   0   0
 100 100  10   2   3   0   0   0   0   0   0   0   0   0   0   0   0   0  60  30  60  30  60 120
  1.500000E+00  5.000000E-01  9.999999E+09  1.000000E+24  0.0           1.000000E+00


NEUTRON FLUX PROBLEM DESCRIPTION - SECTION 003

   0   0   0   0   6   0   0   0   0   0   0   1   1   0   0   0   0   0   0   0   0   0   0   0   0   0
  1.000000E-04  1.000000E-05  1.000000E-04  1.000000E-04  1.000000E-04  0.0
  0.0           0.0           1.000000E+00  1.000000E+00  1.000000E+00  0.0


LEFT,TOP,RIGHT,BOTTOM,FRONT,BACK BOUNDARY CONDITIONS ARE
  0.0           0.0           4.692000E-01  4.692000E-01  4.692000E-01  4.692000E-01

 TWO DIMENSIONAL SLAB GEOMETRY (X,Y)  WIDTH  5.005000E+01  HEIGHT  4.859999E+01


 REGION SPECIFICATIONS
   PTS  REGION WIDTH
     2  2.500000E+00     1  1.350000E+00     4  7.700000E+00     4  7.700000E+00     4  7.700000E+00    10  2.310001E+01

   PTS  REGION HEIGHT
     2  2.500000E+00     1  1.550000E+00     2  4.050000E+00     1  2.100000E+00     2  3.900000E+00     1  2.100000E+00
     4  8.100000E+00    10  2.430000E+01

 X-DIR. POINTS  25         Y-DIR. POINTS  23

 DISTANCES TO MESH INTERVAL INTERFACES

    J   DIST.
    2   1.250    3   2.500    4   3.850    5   5.775    6   7.700    7   9.625    8  11.550    9  13.475   10  15.400
   11  17.325   12  19.250   13  21.175   14  23.100   15  25.025   16  26.950   17  29.260   18  31.570   19  33.880
   20  36.190   21  38.500   22  40.810   23  43.120   24  45.430   25  47.740   26  50.050

    I   DIST.
    2   1.250    3   2.500    4   4.050    5   6.075    6   8.100    7  10.200    8  12.150    9  14.100   10  16.200
   11  18.225   12  20.250   13  22.275   14  24.300   15  26.730   16  29.160   17  31.590   18  34.020   19  36.450
   20  38.880   21  41.310   22  43.740   23  46.170   24  48.600

 DISTANCES TO FLUX POINTS

    J   DIST.
    1   0.625    2   1.875    3   3.175    4   4.813    5   6.737    6   8.662    7  10.587    8  12.512    9  14.437
   10  16.362   11  18.287   12  20.212   13  22.137   14  24.062   15  25.987   16  28.105   17  30.415   18  32.725
   19  35.035   20  37.345   21  39.655   22  41.965   23  44.275   24  46.585   25  48.895

    I   DIST.
    1   0.625    2   1.875    3   3.275    4   5.063    5   7.087    6   9.150    7  11.175    8  13.125    9  15.150
   10  17.212   11  19.237   12  21.262   13  23.287   14  25.515   15  27.945   16  30.375   17  32.805   18  35.235
   19  37.665   20  40.095   21  42.525   22  44.955   23  47.385
```

```
ZONE INPUT BY REGION
   4   7   1   2   5   4
   7   7   1   2   5   4
   1   1   1   2   5   4
   1   1   6   3   5   4
   1   1   2   3   5   4
   1   1   6   3   5   4
   2   2   3   4   4   4
   4   4   4   4   4   4

ZONE NUMBER AT EACH MESH INTERVAL

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

  1    4  4  7  1  1  1  1  2  2  2  2  5  5  5  5  4  4  4  4  4  4  4  4  4  4
  2    4  4  7  1  1  1  1  2  2  2  2  5  5  5  5  4  4  4  4  4  4  4  4  4  4
  3    7  7  7  1  1  1  1  2  2  2  2  5  5  5  5  4  4  4  4  4  4  4  4  4  4
  4    1  1  1  1  1  1  1  2  2  2  2  5  5  5  5  4  4  4  4  4  4  4  4  4  4
  5    1  1  1  1  1  1  1  2  2  2  2  5  5  5  5  4  4  4  4  4  4  4  4  4  4
  6    1  1  1  6  6  6  6  3  3  3  3  5  5  5  5  4  4  4  4  4  4  4  4  4  4
  7    1  1  1  2  2  2  2  3  3  3  3  5  5  5  5  4  4  4  4  4  4  4  4  4  4
  8    1  1  1  2  2  2  2  3  3  3  3  5  5  5  5  4  4  4  4  4  4  4  4  4  4
  9    1  1  1  6  6  6  6  3  3  3  3  5  5  5  5  4  4  4  4  4  4  4  4  4  4
 10    2  2  2  3  3  3  3  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 11    2  2  2  3  3  3  3  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 12    2  2  2  3  3  3  3  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 13    2  2  2  3  3  3  3  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 14    4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 15    4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 16    4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 17    4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 18    4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 19    4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 20    4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 21    4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 22    4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4
 23    4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4  4

MEMORY LOCATIONS RESERVED FOR DATA STORAGE---  60000
MEMORY LOCATIONS USED FOR THIS PROBLEM-------   8589
MEMORY LOCATIONS NOT USED--------------------  51411
        STRAGE USED       152 FROM      3000 IN CIT1 STEP
        MATERIAL NO.
                  1        2         3         4         5         6         7
        X-REGION NO.
                  4        7         1         2         5         4         7         7         1         2

                  5        4         1         1         1         2         5         4         1         1

                  6        3         5         4         1         1         2         3         5         4

                  1        1         6         3         5         4         2         2         3         4

                  4        4         4         4         4         4         4         4

    NO. OF MATERIALS USED IN THE CASE =  7


IAEA BENCHMARK PROBLEM NO3 *HIGH ENRICHMENT FUEL
BOL
LINE RELAXATION WILL BE DONE ON ROWS -   1 INNER ITERATION(S)
ITERATION  FLUX CHANGE     BETA        MU-1        MU-2        MU-3              K
    1      1.78913E+00 1.000E+00 3.578E+00-1.762E+00 0.0          0.544637
    2      3.57574E+00 1.898E+00 5.574E+00 1.324E-01 5.733D-01    0.672889
    3      5.61506E+00 1.814E+00 7.185E+00 1.936E-02 1.054D+00    0.735441
    4      4.58133E+01 1.751E+00 5.397E+01 3.884E-02 1.305D+00    0.854300
    5      1.72789E+02 1.707E+00 1.766E+02 4.945E-02 4.818D-01    0.876838
    6      2.18395E+01 1.677E+00 2.197E+01 5.041E-02-1.401D-01    0.869971
    7      2.90354E+01 1.657E+00 3.036E+01 2.158E-02-7.525D+00    0.932220
    8      2.71007E+01 1.645E+00 2.803E+01 3.245E-02 5.513D-01    0.929622
    9      1.19594E+01 1.637E+00 1.240E+01 3.113E-02 2.364D-01    0.949276
   10      1.77538E+01 1.631E+00 1.924E+01 1.898E-02 4.662D+00    0.977166
   11      1.22010E+01 1.628E+00 1.289E+01 6.021E-02 2.958D-01    0.976793
   12      5.51422E+00 1.626E+00 5.966E+00 1.055E-01-4.195D-01    0.981779
   13      2.24759E+01 1.625E+00 2.655E+01 1.882E-02-8.611D+00    1.006561
   14      1.39336E+01 1.624E+00 1.455E+01 5.032E-02 1.811D-02    1.001621
   15      5.88634E+00 1.624E+00 6.309E+00 1.358E-01 7.950D+00    1.010450
   16      2.60260E+01 1.623E+00 3.045E+01 1.703E-02 5.019D+00    1.020894
   17      1.00505E+01 1.623E+00 1.044E+01 9.542E-02-1.969D-01    1.010977
   18      6.09537E+00 1.623E+00 6.702E+00 1.280E-01 8.856D-01    1.015512
   19      5.09498E+01 1.623E+00 5.931E+01 1.603E-02-6.275D+00    1.026013
   20      1.25169E+01 1.623E+00 1.276E+01 9.940E-02-1.282D+00    1.005324
   21      1.50389E+01 1.498E+00 1.624E+01 5.279E-02-2.154D-01    1.015141
   22      3.37939E+01 1.498E+00 3.604E+01 3.969E-02 7.458D-01    1.012926
   23      7.95942E+00 1.498E+00 8.195E+00 1.449E-01-7.340D-02    1.016189
   24      1.46720E+01 1.498E+00 1.652E+01 3.821E-02-3.677D+00    1.016568
   25      6.79331E+00 1.498E+00 7.256E+00 1.212E-01 5.777D-02    1.016261
   26      6.48816E+00 1.498E+00 7.443E+00 1.170E-01-1.677D+01    1.016391
   27      5.77063E+00 1.498E+00 6.660E+00 1.027E-01-8.253D-02    1.016215
   28      1.30877E+00 1.498E+00 1.536E+00 4.479E-01 1.218D+01    1.016933
   29      2.85214E+00 1.498E+00 5.031E+00 1.113E-01-4.513D-01    1.016994
   30      8.45385E-01 1.498E+00 1.142E+00 4.486E-01 3.055D-01    1.016860
   31     -4.39502E-01 1.427E+00-9.594E-01-1.788E-01-9.403D-01    1.016935
   32      5.13164E-01 1.427E+00-6.544E-01-8.631E-01-1.450D+00    1.016939
   33     -1.79080E-01 1.427E+00-5.281E-01-1.172E+00-8.613D-01    1.017053
   34     -1.25026E-01 1.427E+00 5.731E-01 9.145E-01 3.390D-01    1.017127
   35      1.20955E-01 1.427E+00-8.465E-01-2.604E-01-2.620D+00    1.017087
   36     -5.96132E-02 1.427E+00-5.525E-01-5.218E-01-4.006D-01    1.017139
   37      1.76077E-02 1.427E+00-2.778E-01-1.324E+00-2.419D-01    1.017153
   38     -9.37325E-03 1.427E+00-5.417E-01-6.065E-01 1.542D+00    1.017168
   39     -7.80308E-03 1.427E+00 8.247E-01 8.748E-01-6.824D-01    1.017193
   40      6.73962E-03 1.427E+00-8.570E-01-8.546E-01-1.049D+00    1.017200
   41      6.21796E-03 1.427E+00 9.288E-01 8.796E-01 7.905D-01    1.017207
   42      5.73158E-03 1.427E+00 9.275E-01 8.942E-01 2.251D-01    1.017217
   43      5.28240E-03 1.427E+00 9.269E-01 8.934E-01 3.033D+00    1.017222
   44      4.86565E-03 1.427E+00 9.260E-01 8.935E-01 8.960D-01    1.017227
   45      4.48036E-03 1.427E+00 9.253E-01 8.932E-01 7.634D-01    1.017232
   46      4.12941E-03 1.427E+00 9.258E-01 8.930E-01 1.240D+00    1.017234
   47      3.80898E-03 1.427E+00 9.262E-01 8.929E-01 8.810D-01    1.017236
   48      3.51048E-03 1.427E+00 9.251E-01 8.927E-01 8.964D-01
           3.49199E-02   EXTRAPOLATION WITH       9.9823         1.017253
   49      6.79016E-04 1.000E+00 1.941E-01 1.547E-01 9.931D+00    1.017251
   50      5.64575E-04 1.427E+00 8.320E-01 5.349E-01 6.387D-02    1.017246
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 51      4.73976E-04 1.427E+00 8.400E-01 7.600E-01 2.997D-01      1.017243
 52      4.44412E-04 1.427E+00 9.381E-01 1.045E+00 2.353D+00      1.017240
 53      3.68118E-04 1.427E+00 8.287E-01 7.654E-01 4.632D-01      1.017237
 54      3.37601E-04 1.427E+00 9.174E-01 9.127E-01 7.465D-01      1.017234
 55      2.97546E-04 1.427E+00 8.817E-01 8.594E-01 1.110D+00      1.017232
 56      2.66075E-04 1.427E+00 8.945E-01 8.306E-01 7.075D-01      1.017230
 57      2.37465E-04 1.427E+00 8.927E-01 8.499E-01 7.366D-01      1.017228
 58      2.11716E-04 1.427E+00 8.918E-01 8.775E-01 8.426D-01      1.017226
 59      1.88828E-04 1.427E+00 8.921E-01 8.406E-01 5.978D-01      1.017224
 60      1.68800E-04 1.427E+00 8.941E-01 8.803E-01 5.771D-01      1.017223
 61      1.49727E-04 1.427E+00 8.872E-01 8.448E-01 4.296D-01      1.017222
 62      1.33514E-04 1.427E+00 8.919E-01 8.634E-01-3.687D-01      1.017221
 63      1.19209E-04 1.427E+00 8.930E-01 8.646E-01 4.133D+00      1.017220
 64      1.05858E-04 1.427E+00 8.881E-01 8.638E-01 1.670D+00
         7.47493E-04   EXTRAPOLATION WITH       7.0613           1.017215
 65      1.14441E-05 1.000E+00 1.081E-01 2.423E-01 1.020D+01      1.017215
 66      1.90735E-05 1.427E+00 1.667E+00 4.369E-01 4.385D-01      1.017215
 67     -2.77758E-05 1.427E+00-1.456E+00-4.267E+00 9.342D-01      1.017215

END OF EIGENVALUE CALCULATION - ITERATION TIME  0.216 MINUTES

CONVERGENCE INDICATION BY MINIMIZING THE SUM OF THE SQUARES OF THE RESIDUES - RELATIVE ABSORPTION  1.0000000   K  1.0172167

    LEAKAGE  4.58803E+03  TOTAL LOSSES  2.38004E+06  TOTAL PRODUCTIONS  2.42101E+06  REACTOR POWER(WATTS)  1.00000E+06
```

Acknowlegments

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