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Revised SRAC Code System

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Revised SRAC Code System

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Abstract

Since the publication of JAERI-1285 in 1983 for the preliminary version of the SRAC code system, a number of additions and modifications to the functions have been made to establish an overall neutronics code system. Major points are (1) addition of JENDL-2 version of data library, (2) a direct treatment of doubly heterogeneous effect on resonance absorption, (3) a generalized Dancoff factor, (4) a cell calculation based on the fixed boundary source problem, (5) the corresponding edit required for experimental analysis and reactor design, (6) a perturbation theory calculation for reactivity change, (7) an auxiliary code for core burnup and fuel management, *etc.*

This report is a revision of the users manual which consists of the general description, input data requirements and their explanation, detailed information on usage, mathematics, contents of libraries and sample I/O.

Keywords: Neutronics, Cell Calculation, Cell Burnup, Core Calculation, Core Burnup, Fuel Management, Resonance Absorption, NR, IR, Table-Look-Up, Double Heterogeneity, Collision Probability Method, Sn, Diffusion, Core Calculation, Smearing, Collapsing, Group Constants, Analysis, Benchmark, Thermal Reactor

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SRACコードシステム
改訂版

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（1986年4月17日受理）

要　　旨

1983年にSRACコードシステムの第1版のレポートがJAERI-1285として出版されたのち，総合的な核計算コードシステムを目指して数多くの機能の改良と追加が行われてきた。主なものは，(1)JENDL－2版のデータライブラリー，(2)共鳴吸収の二重非均質効果に対する直接的な方法，(3)一般化されたダンコフ係数，(4)固定面中性子源に基く格子計算，(5)実験解析や設計の要求に対応する出力データ，(6)反応度変化のための摂動計算，(7)炉心燃焼と燃料管理のための補助コード等である。

この報告書は利用手引書の改訂版であり，一般的記述に始まり入力データとその説明，使用上の情報，数式，ライブラリーの内容及び入力例から成っている。

＊　日本情報サービス（株）
＊＊　I.S.L.（株）

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-380 or FACOM VP-100
3. Nature of physical problem solved : Overall neutronics calculation including cell calculation with burn-up, core calculation for any type of thermal reactor. Core burn-up calculation and fuel management by an auxiliary code.
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 : 20 regions for a continuous energy resonance absorption calculation and 16 steps for cell burn-up.
6. Typical running time : It depends on the number of energy group, geometry option, and with or without burn-up calculation.
7. Unusual features of the program : Flexible energy group structure in cell and core calculation. 13 types of cell geometries for collision probability method. Optional treatments for resonance absorption by table-look-up based on NR or IR, or the continuous energy cell calculation in dominant resonance energy range. Successive cell calculation to treat double heterogeneity.
8. Related and auxiliary programs : PROF-GROUCH G-II 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 PDS files. COREBN for core burn-up and fuel management.
9. Status : under testing
10. References : JAERI-1285, this report
11. Machine requirement : 1400 KB core memories for normal work area of 60,000 words. Plotter facility
12. Programming language used : FACOM Fortran 77
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 PDSFUTY enables to read, write, rename, and delete a member by Fortran statements
15. Name and establishment of author : K. Tsuchihashi et al., JAERI Tokai Research Establishment, Tokai-mura, Ibaraki-ken 319-11, Japan
16. Material available :

Foreword

This report is a revision of the users manual of the SRAC code system published on the report JAERI 1285, January 1983.

The JAERI standard thermal reactor nuclear design code system SRAC has been developed to be used with high reliability in design and analysis of a variety of thermal reactors as a demand has been increased for accurate estimate of reactor characteristics, safety aspects, fuel cycle strategy, *etc.*

The primary version of the SRAC code system as been released in FY 1982 after the verification by extensive benchmark calculations on various types of critical assemblies such as TCA (Tank-type Critical Assembly for light water reactor), DCA (Deuterium Critical Assembly for Advanced Thermal Reactor of H_2O cooled D_2O moderated pressure tube type), SHE (Semi-Homogeneous Experimental facility of 20% enriched uranium graphite moderated type), JMTRC (Critical facility for JMTR), TRX benchmark cores and a series of FBR benchmark cores.

After the release, a number of additions and changes have been made to the functions including the auxiliary codes. The major points follow;

(1) The whole thermal scattering law data which were taken from ENDF/B-3 were recalculated in order to include the data on an additional temperature 325 K, because it was found that the interpolation on the scattering law $S(\alpha,\beta)$ was not suitable. The higher Legendre components of H_2O up to P_5 are prepared to permit anisotropic transport calculations.

(2) The JENDL-2 version of data libraries are provided for an alternative of the ENDF/B-4 version, whereas the lack of thermal scattering law data in JENDL allows us to utilize the new version in a mixed mode.

(3) The treatment of resonance absorption in the doubly heterogeneous system as occurs in VHTR with coated fuel particles is replaced by a direct method based on the continuous energy cell calculation.

(4) A generalized Dancoff factor is introduced for infinite arrays of multi-region cells including several absorber lumps with different nuclide concentration.

(5) A fixed boundary source problem is available in the cell calculation by the collision probability method. It can give a proper spectrum to an isolated cell which can not have its own spectrum. In a doubly heterogeneous cell, the microscopic cell is calculated first, and the resultant averaged macroscopic cross sections are used in the succeeding macroscopic cell calculation. Although the cell burn-up calculation is executed only on the microscopic cell, this function gives the burn-up routine the proper reaction rate which could be obtained after passing the macroscopic cell calculation.

(6) An edit function is added to calculate the reaction rate of the neutron detectors, the spectrum parameters, and the conversion rate. It is called after the cell calculation or after the core calculation.

(7) The self-shielded cross sections calculated for a particular composition and stored in a separate file can be utilized as if they are infinitely dilute ones. These cross sections are used in the above reaction rate calculation.

(8) The reactivity caused by the change of cross section is calculated by using the first order perturbation theory.

(9) The assembler routine RWPDS to read/write the data from/to PDS files has been replaced

by a new routine PDSFUTY. An interface Fortran routine READ is also modified to accept the change of usage. The major modification is to mount dynamically the directory of each PDS file on the core memory during the execution, and to keep the member length in the directory so that the frequent call of the routine to ask whether or not a member exists, if affirmative, then to get the member length may not need to access to I/O unit. This modification results in the decrease of I/O times into half and the elapsed time considerably.

(10) An auxiliary code COREBN for a core burn-up calculation has been developed. Through the data storage file, a fuel management is also available. An interpolation of tabulated macroscopic cross section by degree of burn-up and temperature gives space-dependent cross sections to 2D or 3D diffusion core calculation at each time step. The table is prepared in the cell burn-up step in the main SRAC code.

Parallel to the modification, validation and application works have been continued as shown below.

An international RERTR program has offered occasions to show the validity of the prediction for the research reactor conversions from HEU to LEU upon such as an international intercomparison of benchmark calculations for DIDO type heavy water moderated reactor, an analysis of the initial LEU core of the Ford Nuclear Reactor, analyses for temperature and void coefficients of KUCA (Kyoto University Critical Assembly) MEU (medium enriched uranium) cores, and a series of analyses for the critical experiments in JMTRC MEU core.

SRAC has been successfully applied to the design works for the upgrading plan of the JRR-3 and for the core conversion of JAERI research and test reactors. Its application has been also made for the reconstruction of the SHE for the neutronics study of the VHTR and for the benchmark study of the criticality safety facility.

Recent experimental analyses show a good prediction of the SRAC on the critical approach of the VHTRC, the thorium plates inserted in the test zone of the KINKI-UTR (KINKI University Training Reactor), the beryllium plates in the graphite zone of KUCA.

Fairly good agreement shown in a preliminary analysis for highly compact LWR experiments performed in the Proteus reactor encourages us to extend the SRAC code system to predict more accurately the reaction rates occurring in the intermediate energy range.

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I. General Descriptions

The SRAC system is designed to permit overall neutronic calculation for any type of thermal reactors; which covers microscopic library compilation, macroscopic constant generation, cell and core calculations including burn-up and fuel management. The key parameters required in reactor design or experimental analysis are also provided.

The unique functions implemented in the SRAC code system are in the following ;

1) The wide applicability to all type of thermal reactors owing to the complete set of collision probability routines for 13 types of geometries described in Section II.3.

2) Three options for the treatment of resonance absorption in the dominant resonance energy range. The effective cross sections by the conventional table-look-up method based on the NR approximation can be replaced by those based on the IR approximation. Moreover a rigorous method to solve a multi-region cell problem is also provided.

3) The capability of multi-group cell calculation performed separately in each energy range of thermal, resonance and fast neutrons asa fixed source problem, or continuously through the whole neutron energy range as an eigenvalue problem or as a fixed boundary source problem.

4) A doubly heterogeneous system can be solved by successive cell calculations since smearing and/or collapsing of macroscopic cross sections is carried out separately. Especially, the resonance absorption of which double heterogeneity effect should be solved simultaneously, can be treated as far as the microscopic cell can be approximated by any of 1D cells.

5) The Dancoff correction factor required in the interpolation of the self-shielding factors of resonance nuclides is automatically calculated by the installed collision probability routines. It is to be noted that the factor is given not for an absorber lump but for each constituent nuclide for the lattice cell which contains a resonant nuclide in two material with different composition.

6) The calculation methods and the energy group structure should be selected by the user depending on his purpose. For example, a variety of the transport codes are available for cell calculations (collision probability method, 1D and 2D Sn codes). The one space-point solution by the P_1 or B_1 approximation is also available after smearing the cross sections. For the core calculation, 1D, 2D and 3D diffusion codes in addition to the above transport codes are also available.

These functions give SRAC a wide applicability and a high accuracy. As described in the users manual of the preliminary version[1] , and in the related references[2,3] , for validation purpose, extensive benchmark calculations have been made for various types of critical assemblies such as TCA (Tank-type Critical Assembly for light water reactor), DCA (Deuterium Critical Assembly for Advanced Thermal Reactor of H2O cooled D2O moderated pressure tube type), SHE (Semi-Homogeneous Experimental facility of 20 % enriched uranium graphite moderated type), JMTRC (Critical facility for JMTR), TRX benchmark cores and a series of FBR

benchmark cores.

An international RERTR (Reduction of Enrichment of Research and Test Reactor fuels) program has offered occasions to show the validity of the SRAC code system; such as an intercomparison of benchmark calculations for DIDO type heavy water moderated reactor[4], an analysis of the initial LEU core of the Ford Nuclear Reactor[5], analyses for temperature and void coefficient of KUCA (Kyoto University Critical Assembly) MEU (medium enriched uranium) cores[6,7], and a series of analyses for the critical experiments at JMTRC MEU core[8,9].

The SRAC code system has been successfully applied to the designs for the upgrading plan of the JRR-3[10] and for the reduction of uranium enrichment of JAERI research and test reactors[11]. Its application has been also made for the reconstruction of the SHE[12] for the neutronics study of the VHTR and for the benchmark study of a criticality safety facility[13].

A good prediction accuracy of SRAC has been shown through the analyses of the initial critical approach of the VHTRC,[14], the reactivity worth of thorium plate and flux distribution in the test zone of the KINKI UTR (University Training Reactor)[15], and the reactivity worth of beryllium plate in the graphite zone of KUCA[16].

A fairly good agreement shown in the preliminary analysis[17] performed for high conversion LWR experiment in the Proteus reactor[18] encourages us to improve the SRAC code system to predict more accurately the reaction rates occurring in the intermediate energy range.

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 also for its auxiliary codes , user information needed in use of the SRAC system, structure of I/O file, mathematical formulations used in the solution methods adopted, dictionaries and sample I/O.

I.1 Calculation Scheme

In Fig.I-1, a flow diagram of the SRAC is shown. Except the initial read step, all the steps are optionally used. Here we shall follow the 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 IR approximation

Core calculation for simplified core of a homogenized active core surrounded by reflector in eigenvalue problem in multigroup energy group structure by 1D diffusion theory calculation

Condense of macroscopic cross sections for an external use of the CITATION code

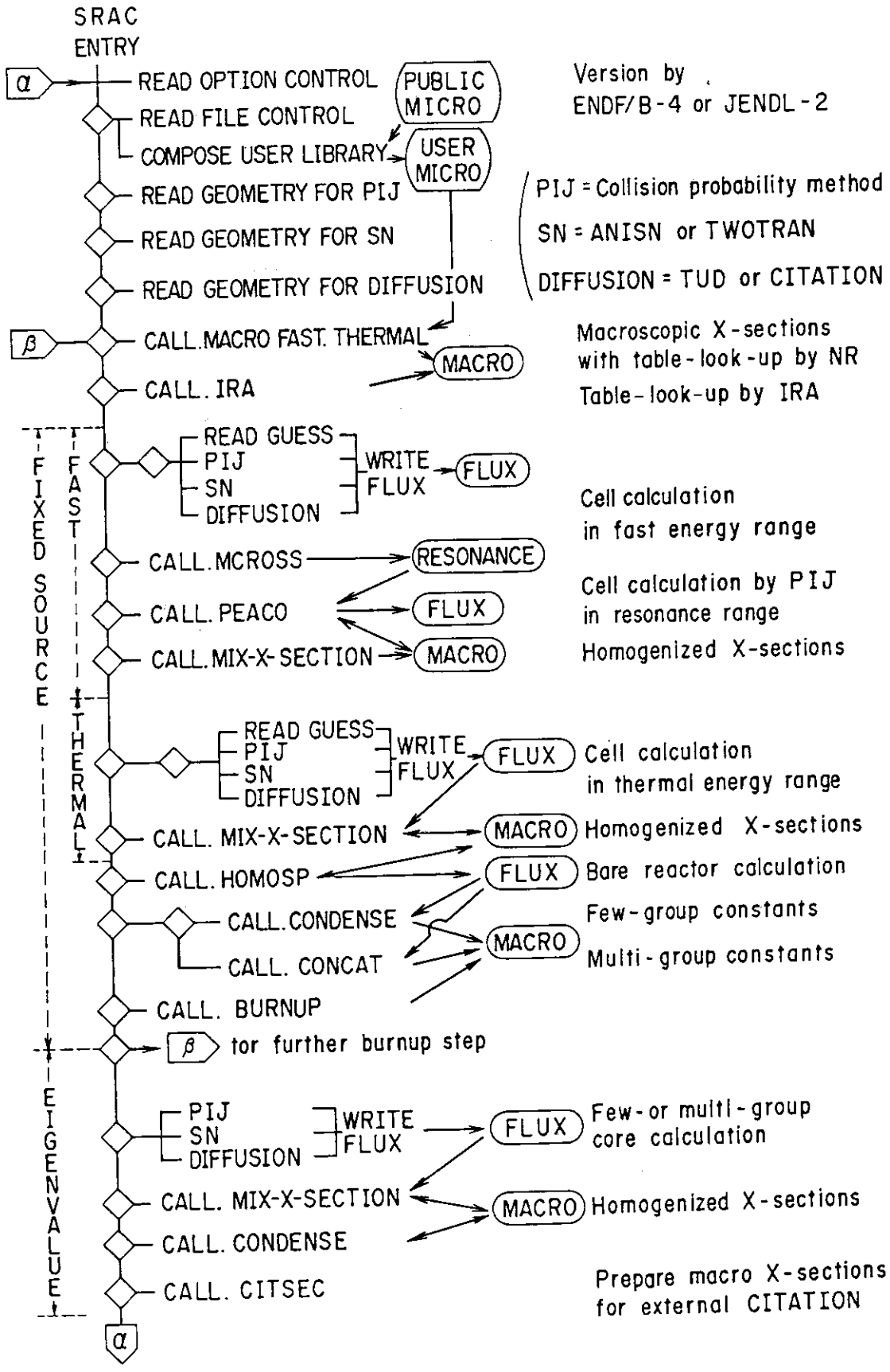


Fig.I.1-1 Flow diagram of SRAC

READ OPTION CONTROL
READ FILE CONTROL
COMPOSE USER LIBRARY
Prepare the User fast and thermal libraries by specifying the energy group structure, nuclides, temperatures
READ GEOMETRY FOR PIJ
Read the geometry of the pin rod cell for collision probability method
READ GEOMETRY FOR DIFFUSION
Read the geometry of the core for 1D diffusion calculation. Read the edit control for the macroscopic cross sections to be written in the format of the CITATION
MACRO FAST.THERMAL
Read the material specifications. Compose macroscopic cross sections in the fast and thermal neutron energy range
IRA Modify cross sections in resolved resonance energy range by IR approximation
PIJ Compute collision probability in the fast energy range, and solve linear equations in the fast energy range
MIX-X-SECTION
Homogenize cross sections using the spatial distribution of neutron flux obtained by the above cell calculation
PIJ Compute collision probability in the thermal energy range, and solve linear equations in the thermal energy range
MIX-X-SECTION
Homogenize cross sections using the spatial distribution of neutron flux obtained by the above cell calculation
HOMOSP Calculate k_∞, k_{eff} by a point model
CONCAT Concatenate fast and thermal cross sections into a set for the succeeding core calculation
DIFFUSION
Solve an eigenvalue problem for a core calculation by 1D diffusion routine
CONDENSE
Condense cross sections into those of few group structure using neutron fluxes obtained in the above core calculation
CITSEC Store macroscopic cross sections in a catalogued file for an external use of the CITATION code

I.2 Data Libraries

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

The present Public library was produced by processing the nuclear data files ENDF/B-III[19] and IV[20]. The Public library based on the JENDL-2 nuclear data file[21] is also available as a substitute of that based on the ENDF/B.

i) Energy group structure

The energy group structure of the current Public libraries consists of 107 groups (48 groups for thermal and 74 for fast energy ranges, respectively, with 12 overlapping groups). In the fast energy range, scattering of non-resonant nuclide is treated as independent of temperature so that the slowing-down equation for resonance absorption can be strictly expressed by a form dependent of temperature only for resonant nuclide. Contrary, in the thermal energy range where temperature dependent scattering takes important role, resonance absorption is evaluated only on group constants.

The Public library is classified into the library corresponding to the fast neutron energy range (En > 0.41399 eV) and that to the thermal (En < 3.9279 eV). The user can choose the thermal cut off energy from the boundary energies of the fine groups within the overlapping range. The higher cut off energy may be used to take account of the up-scattering of thermal neutron of the material under the high temperature. By specifying the cut off energy below 1 eV, the first resonance level of ^{240}Pu is treated in the resonance range those of fertile nuclides. Otherwise, the lethargy mesh width of the as overlapping range is sufficiently narrow to treat this level in the multigroup scheme in the thermal range to take account of the up-scattering effect.

ii) 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, that is, the self-shielding factors are given for scattering, removal, capture, fission and transport cross sections, which are tabulated by temperature T and background cross section σ_0, which is widely used in fast reactor analysis. This library carries also the standard spectrum for collapsing the Public library into the User library. The angular dependence of elastic scattering is taken into consideration up to P_5 component. The present library was produced by use of PROF GROUCH-GII code[22] and TIMS-1 code[23].

iii) Resonance parameters and MCROSS library

The effective group cross sections in the second resonance range defined in Sect.I.3 can be, on simple treatment, modified by the combined use of the IR approximation and the table-look-up method of the resonance shielding factors, while those are always obtained by the method based on the NR approximation, as in the first resonance range. The resonance parameters needed for the IR are accommodated in the fast group constants library.

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 an almost continuous energy group structure of 4600 points. The resonance cross section library needed for the ultra-fine group calculation is given, for every temperature used, by the MCROSS-2 code[24] and stored in the PDS files (See Sect.I.5). Here, in general, a multilevel formalism is used for the cross section representation[25], and the necessary multilevel parameters are also prepared for the energy range (<130.07 eV) in the fast group constant library.

vi) Group constants in thermal energy region

The thermal neutron library consists of the matrices with fixed dimension of 48 energy groups. For moderator materials, the scattering law $S(\alpha,\beta)$ data were calculated by the GASKET code[26], the group-group transfer matrices are tabulated on the fixed eleven temperatures. Especially for the moderators with crystalline structure, the coherent elastic scattering cross sections calculated by the HEXSCAT code[27] have been 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 scattering 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 Public library into the User library was assumed to be (the Maxwellian distribution corresponding to the above temperature & $1/E$ spectrum), commonly for all the nuclides concerned. Moreover, only capture cross sections are prepared to most of FP (fission product) nuclides.

v) Nuclear data for depletion calculation

Several fission product models[28-30] for depletion calculations 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.

First, the model of Garrison and Roos[28] consists of two explicit nuclides (^{135}Xe and ^{149}Sm) 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 for two explicit nuclides, respectively. The nuclear data of ENDF/B-IV are used for half-lives and fission yields for these nuclides.

Second, a detailed model by Iijima *et al.*[29] for BWR 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-lives and fission yield data. The group constants of the explicit nuclides were mostly produced from ENDF/B-IV, except for ^{113}Cd, ^{133}Xe, ^{134}Cs and pseudo fission product.

Third, the VSOP-chain models[30] developed in KFA Juelich are available. Among these models, two branches can be selected, which consist of 27 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 energy group structure not only in few group calculation but

also in multi-group calculation is flexible. Spatial sub-division for the flux distribution is, in general, changeable by the energy range. Here the energy range is defined as follows ;

i) Fast fission energy range

This range corresponds to the fast energy region higher than the fission threshold energy of fertile nuclides where the weighting spectrum used for producing the Public library is assumed to be fission spectrum. The energy averaged spectrum in each group is used as the standard one to collapse the Public library into the User library. For low 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 reactors. The group constants in the Public library are processed assuming the neutron spectrum to be $1/(E\Sigma(E))$, as well as in the following two energy ranges. The $1/E$ spectrum is used to collapse the cross sections of the Public library into those of the User library.

iii) First resonance range

Below about 50 KeV, fine structure appears in the neutron spectrum due to isolated and/or statistical resonance levels of heavy nuclides, and the Doppler effect must be taken into account. For each heavy resonant nuclide, an exact calculation is made for the resonance shielding factor production using TIMS-1 code[23]. There is, however, no special difference in programming between the smooth and resonance energy ranges in the SRAC system.

vi) Second resonance range

This energy range corresponds to the lower resonance energy region where are many sharp and strong resonance levels of fissile and fertile nuclides. A special attention must be paid for this range in thermal reactor analyses, because most of resonance absorption occurs in these strong resonances. The resonance shielding factors for a heavy resonant nuclide are evaluated for the homogeneous mixtures with an imaginary nuclide of the constant cross section. 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. Here, for a simple calculation, the IR approximation is used for calculating the effective resonance cross sections, and the ultra-fine 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 on the given temperatures ; 300 K, 325 K, 350 K, 400 K,... etc.,

some interpolation means are available for the group constants on the real temperature. The effect of up-scattering from the thermal range into the range above the cut off energy can be reflected only on the whole energy calculation to hold the neutron conservation (usually in the eigenvalue problem).

I.4 Optional Transport Codes and Their Usage

Several kinds of optional paths are available for neutron transport and/or diffusion calculations. The path based on the collision probability method can treat 13 types of geometries shown in Fig. (II.3-1,a ~ 1,m). The SN path adopts the ANISN code[31] for 1D calculation and the TWOTRAN code[32] for 2D, respectively. On the other hand, the diffusion code CITATION[33] is generally used for the diffusion path, though a 1D code TUD[34] is also available which is preferred by its simple and small input requirement. Any selection from these paths is possible for each energy range except for the second resonance energy range in which the ultra-fine group calculation to a multi-region cell for resonance absorption is uniquely based on a use of the collision probability method.

The neutron spectrum calculation for smearing and/or collapsing of macroscopic cross sections can be made by a proper choice of sequence of paths. Moreover, the P_1 or B_1 approximation[35] based on the fundamental mode assumption is available for collapsing the multi-group cross sections into few group ones 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 purely geometrical sub-division that is bounded by the lines or circles used to define the geometry under consideration for the collision probability routines.

ii) T-region (Thermal-region)

A T-region consists of a few sub-regions where the neutron flux distributions in the sub-regions are same due to geometrical symmetry or where the fluxes in the adjacent sub-regions can be assumed to be same because of the thin optical thickness. This T-region is used for the calculation of the spatial fine structure of the neutron flux in the thermal energy range.

In the SN or diffusion routines, the finest spatial mesh is treated as a T-region.

iii) R-region (Resonance region)

Since the neutron distribution in the fission or resonance energy range is rather flat than in the thermal range, it is not always necessary to sub-divide the geometry into so many meshes as in the thermal energy range. In such a case, several T-regions are collected to form an R-region.

To the SN or diffusion routines, while any allocation to the R-region is not required, the coarse regions are assumed to form R-regions.

iv) X-region (X-section edit region)

An X-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 whole unit cell. On the other hand, for a special case such as 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 be excluded from any of the X-regions when they are added as extra regions to an isolated cell to simulate the surrounding boundary condition.

v) M-region

An M-region is formed by several R-regions which have the same composition. On the calculation of the background cross section σ_0 based on the NR or IR approximation, the collision probabilities are calculated to the M-region. Effective microscopic cross sections are transferred to the burn-up routine by M-region.

I.5 Data Storage in PDS File

The variety of data are stored in the special PDS (Partitioned Data Set) files[36] . A series of data is stored in a file in binary format. It is specified by the member name and length of the member. A PDS file can contain any number of members. A member name of 8 Bites is composed of the characters assigned to a proper material and characters assigned to a physical quantity. An assembler routine PDSFUTY[37] permits member control by Fortran statements. A few auxiliary programs are also prepared to read/write/plot the content of the specified member(s). The TSS terminal commands such as LIST DIRECTORY, DELETE, CONDENSE, COPY, RENAME can be also used for file/member control.

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) Three types of data array (character string of four Bites, integer, and floating point number of single precision) can be read.

2) Columns 1 to 72 of a card image record is used as data field.

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

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

5) A word must be completed in a card image record.

6) One or more blank columns may be inserted between sign code and digit code.

7) For E type, one or more blank columns may be inserted between the last digit of mantissa and the succeeding exponent code character 'E'.

8) For character type, there is no free format. Any character including blank or control character described below on the specified columns is taken as data.

   Column position of character type variables is always organized to start at the first column of the record.

9) Repetition function is available. An integer before the code '(' is taken as the repetition number of a data or a series of data which are enclosed by ( ).

10) Accumulation function is also available. An integer before '*' is taken as the number of times of accumulation, and the data value after '*' is taken as increment to be added to the previous data.

11) The character '/' is taken as the termination code. If the termination code character is encountered, a check whether or not the array length meets with one required by the program. However the character '/' 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 the code '/' is read at the beginning of the next call.

12) 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.

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

14) A series of strings for 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 (character, integer, or floating) of variable or array in the following description is not always mentioned, the user can recognize character type data by finding Hollerith count after the variable name as (A8), and concerning numerical data the user can discriminate integer type or floating type by the first character of the variable name whether if it is one of characters from I to 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 one 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 number of data required in a BLOCK is shown as /20/ or /NRR/. If mixed types of data are required in a BLOCK, they are read in the order of character, integer, then floating type, and the data requirement is expressed by /A8,3,2/ for character, integer and floating variables, respectively.

II.1 General Control and Energy Structure Specifications

BLOCK 1 /A4/

CASENAME — 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 and one more blank card after the last case to terminate the SRAC execution.

BLOCK 2 /A72/

TITLE — 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 separate energy ranges (fast & thermal). If any negative value is specified, a spatial distribution will be read from FT05 in the execution time of the routine to give the realistic fixed source distribution, otherwise a uniform thermal flux distribution is assumed.

=0 None of routines is used
=± 1 Collision probability routine
=± 2 ANISN (one dimensional SN)
=± 3 TWOTRAN (two dimensional SN)
=± 4 TUD (one dimensional diffusion)
=± 5 CITATION (multi-dimensional diffusion)

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
=1 the collision probability method
=2 the empirical formula (not yet installed in SRAC)

Note : When the double heterogeneity is solved by the PEACO routine, feed the Dancoff correction of the microscopic heterogeneity in the material specification.

IC4 Indicator for the thermal energy range.

=0 Thermal range is excluded (for FBR calculation)
=1 Thermal range is included

IC5 Selection of the process for resonance absorption in the resonance II range (above thermal cut off below 130.0eV).

=0 Table look-up of Bondarenko type table[1] by NRA
=1 Table look-up by IRA (Intermediate Resonance Approx.) The IRA routine works for unique resonant R-region in a cell.
=2 The PEACO routine (ultra fine group calculation by the collision probability method
=-2 The PEACO routine to treat more than two resonant materials by an approximation to assum two pseudo resonant materials. Additional input is required to assign the materials to which resonant material.

Note. The PEACO routine generally does not work for more than two resonant mixtures in a cell because the two-dimensional interpolation of collision probabilities is done for completely different resonant materials. When a depletion problem is solved for a multi-region cell, several compositions which have been unique at the clean stage have to be considered in a cell. The similarity of

cross sections can permit the above mentioned approximation.

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 foreseen by the user.

Energy Range	*	Fast Fission	Smooth	Resonance I	Resonance II	Thermal
Upper	*	10 MeV	.82 MeV	67.4KeV	130. eV	cut off
Lower	*	.82 MeV	67.4KeV	130. eV	cut off	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 ; the user has to prepare his own file of DD= MCROSS to write the data.

IC9 Indicator to call HOMOSP routine to calculate the one point (bare) reactor neutron spectrum and K_∞,K_{eff} (see Sect. VI.4).

=0 Skip
=± 1 Call and P_l approximation
=± 2 Call and B_l approximation

If negative value is entered, P0 components of the solution of P_l or B_l equations are used as the weight to

collapse the homogenized cross sections, otherwise the spectrum obtained by the previous cell calculation is used.

Note: The collision probability method is used upon infinite cell approximation.

: The geometrical buckling given in BLOCK 4 is used in the leakage term.

: The one-group microscopic cross sections required in the cell burn-up calculation will be collapsed using the spectrum specified by this item.

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 whole energy calculation.

=0 Skip
=1 Call

Note: This indicator is also used to specify the energy group structure for the whole energy calculation activated by IC12 whether in the fine group or in the few group.

IC11 Indicator to read the geometric information required in II.3 through II.7 for this case.

=0 Read the new geometry
=1 Skip reading and use the same as the previous case

IC12 Selection of the routine for the whole energy range calculation.

=0 No calculation ; go to the CITATION cross section formation.
=± 1 Collision probability method If the negative value is entered, the incident current at the outer boundary is read from FLUX file by the member caseAbSp so that a fixed source problem will be solved.
= 2 ANISN (one dimensional SN)
= 3 TWOTRAN (two dimensional SN)
= 4 TUD (one dimensional diffusion)
= 5 CITATION (multi-dimensional diffusion)

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 whole energy calculation.

=0 skip
=1 call

Note: If either IC10=1 or IC13=1 is entered, the specification of the few group structure by BLOCKs 9,10 is required only in the first case.

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 few group cross

sections will be written if either IC10 or IC13 is 1. Input requirement is described in Sect,II.

=0 skip
=1 write

IC15 Selection of the process to compose (or define) the microscopic total cross sections in the resonance energy range. (See Sect.VI.2)

=± 1 Interpolation of the self-shielding factor tabulation of the total cross sections which have been constructed by the inverse average of the total cross sections in sufficiently fine energy intervals.
The in-group scattering cross sections are adjusted to hold the neutron balance after the table look-up.

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

=± 2 Summation of all partial reactions

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

The negative value activates an approximation on the treatment of up-scattering from the thermal range into the epi-thermal range to replace the up-scatter by the same amount of self-scatter in order to save the cpu in a few group calculation.

IC16 Indicator how to form the macroscopic transport (collision) cross sections of each mixture which are required in the isotropic transport routine. (See Sect.VI.2)

=0 the extended transport approximation

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

however, in the resonance shielding calculation

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

is assumed

=1 P_1 approximation[2] of the multi-group one point problem is solved for a media made by homogenizing the mixtures in the system for the fixed source problem (specified by IC2) assuming the flat flux distribution. As for the other isolated mixtures which will be used in the whole energy calculation, an equation is solved for one pure mixture. If any of constituent mixtures in a cell has already been formed in the previous case, other new mixtures are treated as if isolated ones.

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

=2 B_1 approximation[2] of the multi-group calculation of the homogeneous media.

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

In the above equations, the geometrical buckling given in BLOCK4 is used in the leakage term, and the source is assumed to be same as the fission neutron spectrum of ^{235}U .

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

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

Note : For a material strongly absorbing or almost empty, do not use the options other than IC16=0, otherwise, the spectrum calculation will fail to converge.
: If the cell burn-up calculation is specified by IC20, this item is fixed to 0 in order to pass a common process through every burn-up steps.

IC17 Indicator how to compose the cell averaged diffusion coefficients on D1 position of macroscopic cross section format. The absolute value of this item defines the way of spatial average of the coefficients in the fine energy group stage, and the sign of IC17 defines how to collapse them into the few group constants. Normally enter positive value, then the few group transport cross sections are made from the inverse of the few-group diffusion coefficients.

$$D_G = \sum_{g \in G} D_g \varphi_g / \sum_{g \in G} \varphi_g$$

If a negative value entered, the few group transport cross sections are calculated by the flux-volume weighted average of multi-group transport cross sections. This option should be used for highly absorbing material. Even if the negative value is entered, the few group diffusion coefficient are not changed by the sign of this entry.

$$\Sigma_{tr,G} = \sum_{g \in G} \Sigma_{tr,g} \varphi_g / \sum_{g \in G} \varphi_g \qquad \text{for} \quad IC17<0$$

=± 1 Inverse of the spatially averaged transport cross section. The values required by this option are usually stored in

D2 position unless IC17=3. For the homogeneous mixture this option is unconditionally taken as IC17=1 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[3] which are written into D1, whereas the values $\bar{D}_t$ defined by Eq.VI.5-11 in Sect.VI.5 remain in D2 positions in the macroscopic format (see Sect.V.5).

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

=± 3 The anisotropic components of the Behrens' term of the Benoist model[3]. 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.

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

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 the reaction rate calculation.

=0 skip
=1 call reaction routine. The input specified in II.9 is required.

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 the cell calculation where all mixtures used in the problem are to be specified with nuclide composition)
The input is specified in II.10.

Note. All relevant daughter nuclides depending on the chain scheme have to be prepared in User libraries.

BLOCK 4 /1/

BSQ Geometrical buckling (cm^{-2}) commonly used in the P_1 or B_1 approximation in one point spectrum calculation specified by IC9 and/or IC16. The negative value is accepted.

BLOCK 5* Data set specification for PDS library files. 8*/A8/
Each in one line.

DATASET(I),I=1,8
Any alphameric characters indicate the need of the corresponding file in the job step. If a blank line is entered, it indicates that the file is not used. For example, When the user fast and thermal files are prepared by the previous job, 1st and 2nd lines may be blank because no public library is necessary.

1st line for the public fast library
2nd line for the public thermal library
3rd line for the user fast library
4th line for the user thermal library
5th to 8th lines are always necessary, but their function is obsolete.

BLOCK 6* Specification for energy group structures /4/

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

NET Number of the thermal neutron groups of 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* Required if DATASET(1) is specified in BLOCK 5 /NEF/

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

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

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

BLOCK 9* Required if non-zero NERF is specified /NERF/

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

BLOCK 10* Required if non-zero NERT is specified /NERT/

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

Note 1 BLOCK's marked by * *i.e.* BLOCK 6 through BLOCK 10 are required only in the first case. The energy group structure is kept during the job step. If the CONDENSE routine will be called in any of succeeding cases, the condense information NERF, NERT, NECF's, NECT's have to be specified in the first case. To feed these items, either of condense indicator IC10 or IC13 must be

non-zero while the CONDENSE routine is not called in the initial case.

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 user library files are updated. The use of the public file(s) must be indicated on BLOCK 5 of II.1. The information for the specified nuclide and temperature will be transferred from the public file(s) into the user('s).

BLOCK 1 /A8/

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

Note: The same organization of IDENT is used to specify a nuclide in the material specification on II.8:

X-tag the tag used internally to specify physical quantities, then any character is accepted in the input. The user will be aware that several members with 'C', 'F', 'M', or 'K' for this tag are transferred at this step.

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. 'N' is used to specify the element with natural abundance. 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)

b-tag punch '0' always in this BLOCK. This tag is used in the mixture specification (II.8) to identify the effective microscopic cross sections of the b-th burn-up step associated to the mixture because the common BLOCK

organization is applied in the mixture specification.

i-tag punch '0' always in this BLOCK. This tag is used to identify the effective microscopic cross sections associated to the mixture when this organization is applied in the mixture specification (II.8).

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

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

A special function is added to treat the library in which the data are prepared on the different temperature points. If IDENT='TEMPSET ' is entered before the blank line, the vector (TEMPi,i=1,11) punched in the free format are read immediately after this special nuclide identification in order to replace the tabulation of temperature.
For this function, the corresponding thermal library must be prepared.

II.3 Collision Probability Method

The input of this section is required if the combination of IC1=1 and IC11=0 is specified in BLOCK 3 of Sect.II.1.

BLOCK 1 Control integers /18/

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

=1 One dimensional sphere of multi-shells.

Fig.II.3-1-a shows the cross section of a spherical cell. This cell model has been utilized in the evaluation of the resonance integral of coated particles of an HTGR assuming the spherical cell of the unit fuel grain and associated graphite binder with the isotropically reflective condition at the outer boundary.

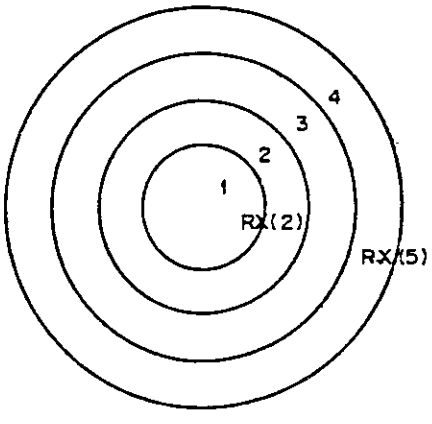


Fig.II.3-1-a Spherical and cylindrical cell

=2 One dimensional slab of multi-layers.

In Fig.II.3-1-b shown is a sample plane cell composed of several layers of infinite planes. Care should be taken of the boundary condition. Since SRAC does not provide the perfect reflective (mirror) boundary condition but the periodic condition for this geometry, an asymmetric cell can be treated. On the other hand, if a symmetric lattice is considered, the full geometry must be given. The symmetric condition is reflected on T-region assignment *i.e.* by assigning a common T-region number to two S-regions in symmetric position. It is to be noted that since the evaluation of the collision probability does not require any numerical integration, a shorter computer time is expected in the application of this geometry.

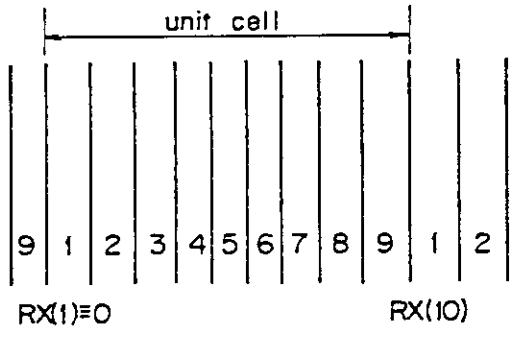


Fig.II.3-1-b Infinite plane cell

=3 One dimensional circular cylindrical of multi-shell.

In Fig.II.3-1-a shown is also the cross section of an infinitely long cylindrical cell which is a frequently used geometry because the most power reactors install fuel in the form of pin rod. The cylindrical model for the regular array of pin rod cell together with the isotropically reflective boundary condition provides a sufficient accuracy in the numerical results compared with those by the more exact model such as the square or hexagonal cell with the perfect reflective boundary condition. The fact that the former requires only one-dimensional numerical integration, recommends the use of the cylindrical model as far as the accuracy is assured.

=4 Square cylinder divided by concentric annuli.

Figure II.3-1-c shows a sample square cell divided by the concentric circles into several regions. A pin rod cell of BWR- and PWR-type lattice cell can be accurately expressed by this module. It is to be noticed that the coolant region can be sub-divided by the circle of which radius exceeds the distance from the center to the side line.

=5 Square cylinder of two-dimensional division

Figure II.3-1-d shows a sample square cell sub-divided by

the concentric circles and further by four lines crossing the central axis. Each line makes an angle of 67.5° with a side line of the square. While an annular ring is divided into eight pieces, because of the octant symmetry assumed, two adjacent pieces per annular division are left as independent regions.

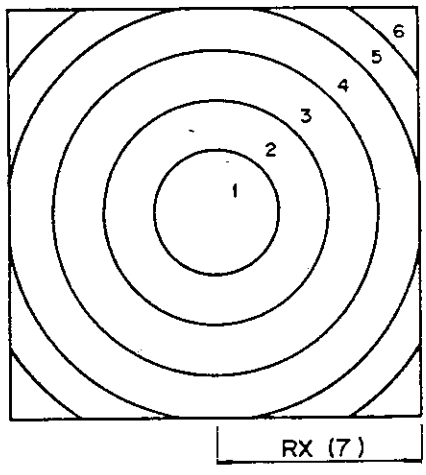


Fig.II.3-1-c Square cell

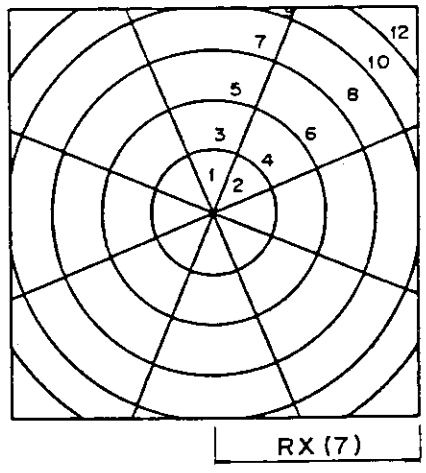


Fig.II.3-1-d Two-dimensional square cell

=6 Hexagonal cylinder divided by concentric annuli

Figure II.3-1-e shows a sample hexagonal cell divided by the concentric circles into several regions. A pin rod cell of the LMFBR, and the SHE can be simulated by this module.

=7 Hexagonal cylinder of two-dimensional division.

Figure II.3-1-f shows a sample hexagonal cell sub-divided by the concentric circles and also by six lines crossing the central axis. Each line makes an angle of 75° with a side line of the hexagon. While an annular ring is divided into twelve pieces, because of the 60° rotational symmetry assumed, two adjacent pieces per annular division remain as independent regions.

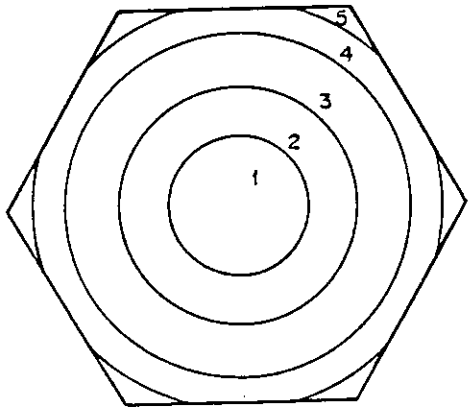


Fig.II.3-1-e Hexagonal cell

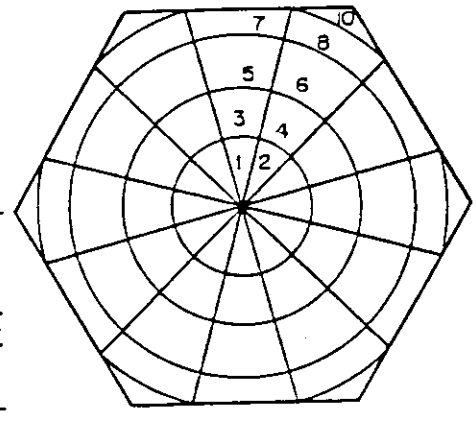


Fig.II.3-1-f Two dimensional hexagonal cell

=8 Octant symmetric square pillar divided by X-Y coordinates

Figure II.3-1-g shows a square cell divided by the Cartesian coordinates.

=9 Octant symmetric square pillar divided by X-Y coordinates with square array of pin rods.

Figure II.3-1-h shows a square cell containing the square array of pin rods. The CLUP77 module for these geometries is developed to compute collision probability in a BWR-type fuel cluster with diagonal symmetry with or without the explicit representation of fuel pins.

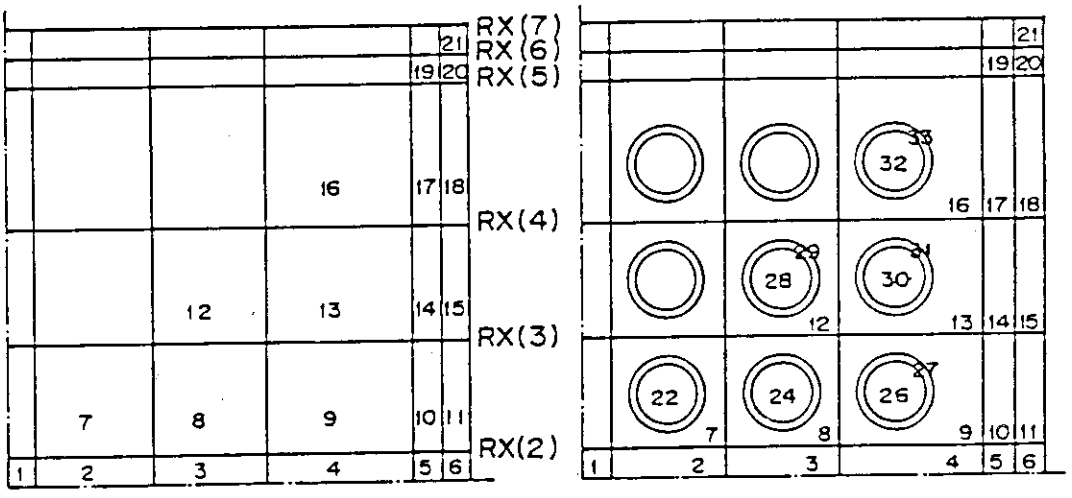


Fig.II.3-1-g
A quadrant of an octant symmetric square assembly

Fig.II.3-1-h
Octant symmetric square assembly with pin rods

Note that a standard 7 * 7 BWR bundle in a 12 * 12 x-y mesh will have, allowing for the diagonal symmetry, 78 x-y mesh regions and a minimum of 56 fuel pin regions (one fuel and one can region per pin) making a total of 134 regions. Subdividing the coolant zones into quadrature increases this to 246.

An application focussing the analysis of the cruciform control rod worth or the alternative water gap peaking effect requires the more detailed x-y subdivision to simulate the sharp flux distribution near the central axes. The running of such a case might exceed the limit of the computer time and memory. This restriction might be mitigated by some compromised approach such as to take account of heterogeneous geometry only of a part of the assembly under consideration and to substitute some fuel pin regions by the homogenized fuel regions in order to reduce the total number of regions.

=10 Annular assembly with annular array of pin rods

As shown in Fig.II.3-1-i, the geometry allowed consists of a number of annului and a number of circular fuel rods, equally spaced in circular rings. The rods consists of several concentric layers, and they may, together with the coolant, be divided further by the circles through the centers by an indicator IDIVP. Several pin rods on a circle are assumed equivalent. The sub-division of annular coolant regions by the concentric circles : the function of IDIVP which indicate the radial positions of pin rods is optional as also shown in Fig.II.3-1-i. Note that the cylindrical approximation is made for the outer shape of a unit assembly. S-region is numbered first to the inner-most pin rod starting from inner to outer of a pin rod center if IDIVP<2, and from inner to outer measured from the cell center if IDIVP=2, then to the pin rod on the outer ring. After the outer-most pin rod, the moderator region is from the inner to the outer.

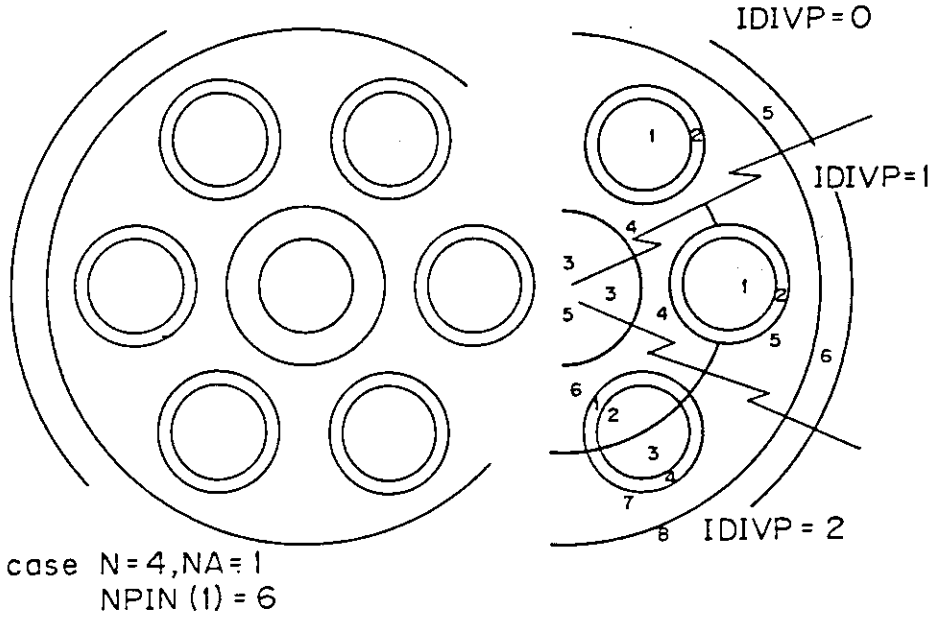


Fig.II.3-1-i Annular assembly with annular arrays of pin rods

=11 Annular assembly with asymmetric array of pin rods

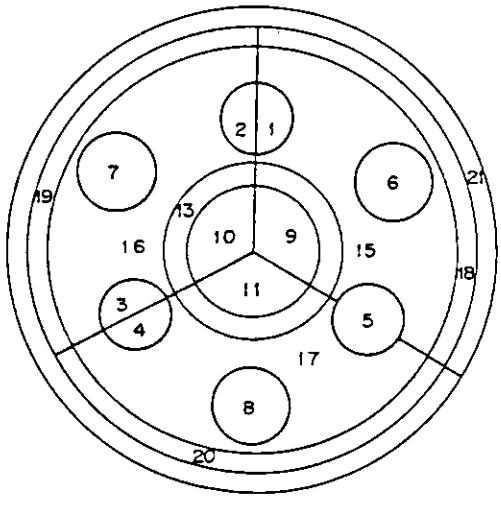
A module to permit an asymmetric disposition of pin rods is provided. Any size of pin rod can be mounted at an

arbitrary position as far as pin rods do not intersect each other. The moderator regions can be subdivided two-dimensionally by the concentric circles and by the radiating lines as illustrated in Fig.II.3-1-j.

Care should be taken in applying the isotropically reflective boundary condition at the outer surface where the neutron flux is assumed uniform and isotropic even if the fluxes in the segment regions adjacent to the surface are not uniform in the rotational direction. It is suggested to use this module in the so-called super cell structure in which an actual asymmetric cell is surrounded by enough thick symmetric material and the isotropic boundary condition is applied at the outer boundary of this external material.

=12 Hexagonal assembly with asymmetric array of pin rods

A module is provided to permit the calculation of the collision probability in a hexagonal block with asymmetrical array of pin rods. The moderator regions can be subdivided two-dimensionally by the concentric circles and by the radiating lines as illustrated in Fig.II.3-1-k. Optional boundary conditions; periodic or 60° rotational, are provided to reflect the realistic disposition of burnable poison rods in the neighbouring blocks.



case N = 5, M = 3
NP = 6

Fig.II.3-1-j Annular assembly with asymmetric pin rods

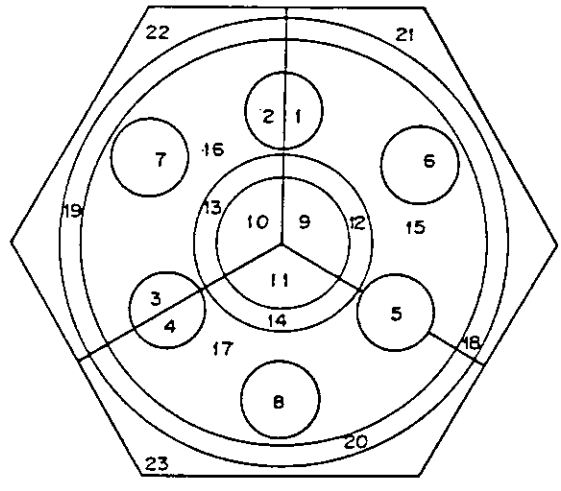


Fig.II.3-1-k Hexagonal assembly with asymmetric pin rods

=13 Rectangular pillar divided by X-Y coordinates with pin rods on grid points

This type has permitted an x-y division of a rectangular lattice cell. By the recent modification, annularly sub-divided pin rods can be mounted on any grid point (x_i, y_j). Arbitrary radii for the annular sub-division are given by each pin rod. A special example shown in Fig.II.3-1-*l* is a multi-rod expression of a hexagonal array

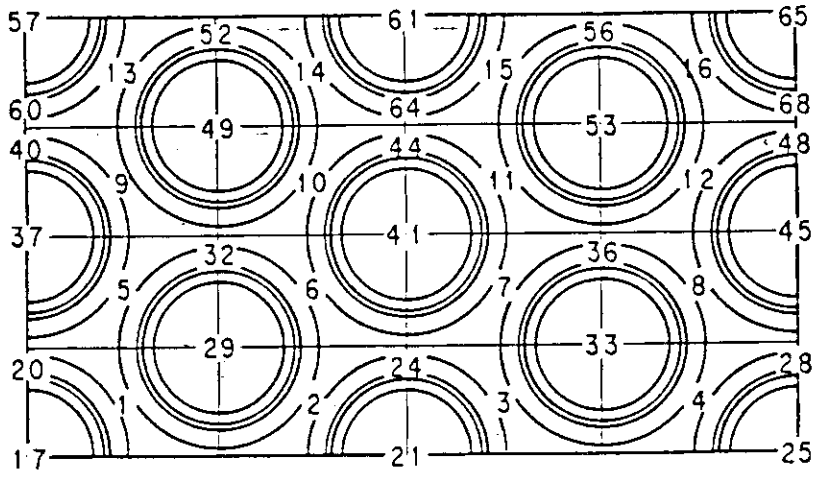


Fig.II.3-1-*l*
X-y two-dimensional cell with pin rods on arbitrary grid points (Illustrated is a multi-rod expression of hexagonal array of pin rods)

of unit pin rods. S-region numbers appearing in the figure is purely geometrical, and the user is requested to allocate T-region number for which the neutron fluxes are calculated. As the periodic condition in x- and y-direction is supposed, the region allocation must satisfy this condition. For example, the region number allocated to a fuel pin located on the left edge must be coincide to that on the right edge.

It has been used in an analysis of the two-rod heterogeneity in the experiments achieved at the PROTEUS reactor for the feasibility study of LWHCR, a closely packed LWR. In the above experiment PuO_2 rods and depleted UO_2 rods are alternatively arranged in a hexagonal array. It has been found that this module is quite useful through the analyses for a three-rod heterogeneity composed of ThO_2, enriched UO_2, and graphite rods in a hexagonal array in the SHE core, and also for a checker board pattern of MO_X and UO_2 rods in the TCA core.

=14 Concentric layer of hexagons with pin rod array of hexagonal symmetry

The original module for this geometry has permitted the concentric hexagonal division of a hexagonal cell. By the modification, on the side of the arbitrary concentric hexagon, pin rods can be mounted. Number of pin rods on a hexagon must be a multiple of six, since the 60° rotational symmetry is assumed. They are placed with equal interval starting at a corner of a hexagon. Pin rods on a hexagon are treated to have the same fluxes. A special case is shown in Fig.II.3-1-m where a hexagonal array is formed by 19 rods. The calculations of direct collision probability of a pin type LMFBR assembly enclosed by a wrapper tube become available to solve the double heterogeneity effect which is one of topics in the FBR physics.

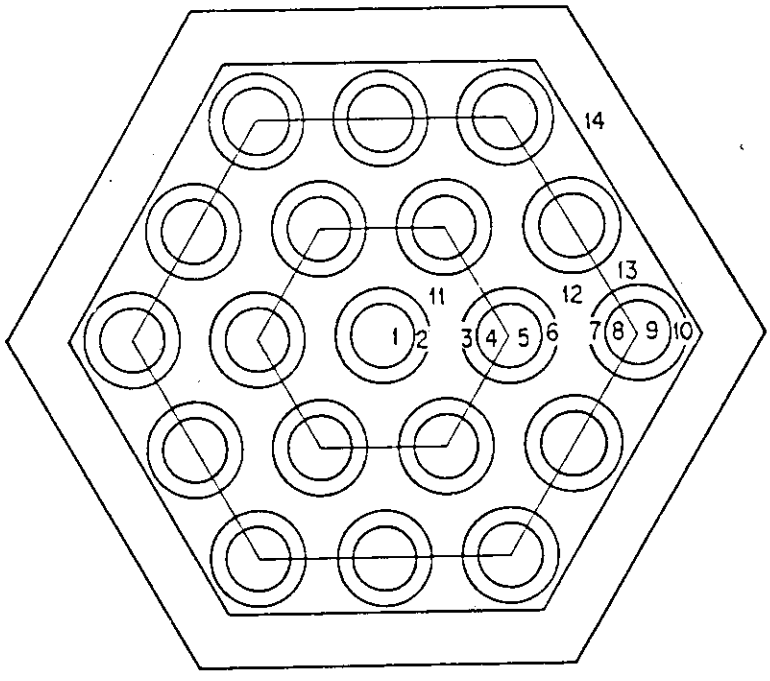


Fig.II.3-1-m Concentric layers of hexagons with equi-distant pin rod arrays

2 NZ Total number of sub-regions. A sub-region is the smallest unit enclosed by line or by circle. It is merely geometrical sub division used to assign the following T- , R , and X-region by integer. The accuracy of the calculation is decided not by sub-region but by the latters.

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

4 NRR Total number of R-regions (in resonance neutron calculations) An R region is composed of one or more T-regions.

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

6 IBOUND Outer boundary condition of the cell calculation

=0 Isotropic (white) reflection
=1 Perfect (mirror) reflection, but for IGT=2 (1D slab) or IGT=13 (2D X-Y pillar) periodic condition is taken. In the latter cases, symmetric condition if any has to be specified by region numbering and dimension.
=2 Isolated (black)
=-1 60 degree rotational (applicable only for IGT=12)

Note: If the fixed boundary source problem is solved by the specification of IC12, IBOUND is automatically set to black. For the Dancoff factor calculation even in this case, IBOUND =0 or =1 should be entered.

7 NX Number of mesh intervals for X or R division of a cell

8 NY Number of mesh intervals for Y or angular division

9 NTPIN Total number of pin rods (effective for IGT=10,11,12,13 or 14, calculated internally for IGT=9)

10 NAPIN Number of pin rods in an array (for IGT=9), or number of circles on which the pin rods are located (for IGT=10,14)

11 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. It is desirable to trace a neutron beyond an optical length of 6.0 if the computer time allows. Recommended value to this item is NCELL = 2 for a cell enough large in the sense of optical path, or NCELL = 5 for a transparent or small cell. 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 and redistribution of collision probabilities.

12 IEDPIJ Edit control for collision probabilities

=0 Skip print
=1 Print collision probabilities

13 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.

14 NDA Number of division of the range IBETM (described below) degree for the numerical angular integration of Pij, required for IGT =4,5,6,7,8,9,10,11,12,13, and 14. Sufficient accuracy will be obtained if approximately half of IBETM is entered as NDA.

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.

15 NDPIN Number of annular division of a pin rod, effective for IGT= 9,10,11,12,13 or 14

16 IDIVP Control of sub-division by RPP's, used for IGT=9,10,11,12,13 or 14.

=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 further divide the pin rod regions into inner and outer regions, (inefficient for IGT=9, or 13)

17 IBETM Range of angular integration in degree. Punch =45 in octant symmetric geometry, =30 in hexagonal symmetry, inefficient for one dimensional geometry. Set double if IBOUND=1 is specified. Enter =360 if symmetric only on a plane.

18 IPLOT Indicator to call plotter routine for geometric mapping

=0 Skip plotting
=1 Call plotter routine

BLOCK 2 /0,7,6/

Parameters for the iterative solution of linear equations for neutron fluxes; the values in < > shows defaulted values used when ITMINN=0 is specified. (See Sect.VI.4)

1 IEDIT Edit control

=0 No edit
add 1 print reaction balance and flux distribution
add 2 print macroscopic cross sections
add 4 print collision probabilities
add 8 print fixed source distribution

2 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

3 ITMOUT Maximum number of outer iterations for the eigenvalue problem < 50 >

4 ITBG Minimum number of iterations before extrapolation < 5 >

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

6 ITDM Minimum delay between extrapolation < 5 >

7 IPT Control of monitor print at each iteration < 0 >

=0 suppress print
=1 print record

1 EPSI Convergence criterion for inner iterations < .0001 >

2 EPSO Convergence criterion for outer iterations < .001 >

3 EPSG Extrapolation criterion < .001 >

4 RELC Initial over-relaxation factor < 1.2 >

5 OVERX Maximum extrapolation < 100. >

6 FACTOR Under extrapolation factor < 0.8 >

BLOCK 3 Required if NR < NZ /NZ/

NREG T-region number by sub-region

BLOCK 4 Required if NRR < NR /NR/

IRR R-region number by T-region

BLOCK 5 Required if NXR < NRR /NRR/

IXR X-region number by R-region. If entered =0, this R-region is excluded from the average.

BLOCK 6 /NRR/

MAR ± Material number by R-region ; sequential order of a material appearing in the mixture specification is used as material number.

Negative value of the material number indicates that this material is heterogeneous in the sense of the double heterogeneity. It is supposed that two composite materials of this material are specified in the preceding positions. That is, if the number -3 appears in MAR's, the first and second materials specified in II.8 are the composites of the microscopic cell. The relevant input data are required in BLOCK 14.

BLOCK 7 Required only if IGT=10 or 14 /NAPIN/

NPIN Number of pin rods on each ring. If IGT=10, the pin rod on the central axis is not counted in NPIN, but the summation of NPIN values is less than NTPIN by one, a pin rod is automatically positioned at the center. If IGT=14, the center pin rod is counted by NPIN. Defaulted values are prepared if NPIN(1)=0 for IGT=14 as 1,6,12,18,....

BLOCK 8 /NX+1/

RX X-abscissae, radii, or the distance from the center to each side of hexagon or square in *cm*; RX(1)=0 always.

BLOCK 9 Required if IGT=11 or 12 and if NY > 0 /NY/

TY Angular division by θ's in degree.

BLOCK 9' Required if IGT=13 and if NY>1 /NY+1/

TY Y-abscissae in *cm* ; TY(1)=0 always.

BLOCK 10 Required if IGT=9,10 or 14 /NAPIN/

RPP X-positions of pin rods for IGT=9. Radii of the circles on which pin rods are located for IGT=10. Distances from the center to the sides of hexagons for IGT=14.

BLOCK 10' Required if IGT=11, or 12 /NTPIN/

RPP Radial position of each pin rod for IGT=11 or 12

BLOCK 10' Required if IGT=13 /NTPIN/

IXP X-position of each pin rod on RX's. Enter integers ranging 0 and NX.

BLOCK 11 Required if IGT=10,11, or 12 /NTPIN/

THETA Angular position of each pin rod by θ in degree

BLOCK 11' Required if IGT=13 /NTPIN/

IYP Y-position of each pin rod on TY's. Enter integers ranging 0 and NY.

BLOCK 12 Required if IGT=9,10 or 14 /NDPIN+1/

RDP Radii for annular sub-division in a pin rod ; where RDP(1)=0. The radii are common through all pin rods.

BLOCK 12' Required if IGT=11,12 or 13 /(NDPIN+1)*NTPIN/

RDP Radii for annular sub-division of individual pin rod ; where RDP(1,J)=0 always.

BLOCK 13 /3/

Plotter control integers required if non zero value of IPLOT in BLOCK 1 is specified

1 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
add 1 Sub-region
add 2 T-region
add 4 R-region
add 8 Material number
add 16 X-region

Positive value indicates printing of assignment of region numbers in the figure, and negative value requires only figure.

2 ISCAL Indicator of the scale of figures

=1 One 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

3 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 separate 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.

BLOCK 14 Required if any negative MAR is entered /A0,3,2/
Control integers for the treatment of the double heterogeneity[4].

1 IDB Energy range indicator

=1 Resonant II range by the PEACO routine
=2 Thermal range (not yet available)
=3 Resonance II and thermal range (not yet available)

2 IGEOM Geometry indicator of the microscopic heterogeneity

=1 Slab
=2 Cylinder
=3 Sphere

3 MODEL Model indicator for the definition of the collision rate ratio in the two-region microscopic cell

=1 Transmission
=2 Neutron from moderator
=3 Neutron escaping from absorber lump
=4 Simplified transmission

4 RF Outer radius of absorber lump

5 RM Outer radius of microscopic cell. Because the escape probability is evaluated by the analytical expression by Case *et al.*[5], the Dancoff correction must be fed in the material specification II.8.

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[6] 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/

1 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.

2 ITH
=0 forward solution
=1 adjoint solution

3 ISCT — Maximum order of scatter found in any zone
=0 P_0 component
=1 P_i components

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

5 IGE — Geometry
=1 slab
=2 cylinder
=3 sphere

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

7 IBR — Right boundary condition, same as IBL

8 IZM — Number of zones or regions

9 IM — Number of mesh intervals

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

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

12 IHT Position of total cross section in cross section table (internally set)

13 IHS Position of self-scatter cross section in cross section table (internally set)

14 IHM Length of cross section table (internally set)

15 MS Cross section mixing table length (mixing function is suppressed) punch =0

16 MCR Number of cross section sets to be read from cards in the original format. (suppressed) punch =0

17 MTP Number of cross section sets to be read from tape (internally set)

18 MT Total number of cross section sets (internally set)

19 IDFM Control for density factors (suppressed)
=0 density factors (21*) not used
=1 density factors used

20 IPVT Indicator of parameter EV, effective if IEVT=2
=0 no effect
=1 EV as k
=2 EV as alpha

21 IQM =0 no distributed source in eigenvalue problem
=1 distributed source formed in SRAC
=2 distributed source read in execution time of ANISN

22 IPM =0 no shell source
=1 shell source IGT*MM
=N shell source N*IGT*MM

23 IPP =0
=N interval number which contains the shell source if IPM=1.

24 IIM Inner iteration maximum (suggested value, IIM = 35)

25 ID1 =0 no effect
=1 print angular flux

26 ID2 =0 (suppressed)

27 ID3 =0 no effect
=N compute N activity by zone

28 ID4 =0 no effect
=1 compute activity by interval

29 ICM Outer iteration maximum (suggested value for K calculation ICM=50) if fixed source ICM=1

30 IDAT1 =0 all data in core (other options suppressed)

31 IDAT2	=0 no effect =1 execute diffusion solution for specified group (24$)
32 IFG	=0 no effect (other options suppressed)
33 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
34 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)
35 IPRT	=0 print cross sections =1 do not print cross sections
36 IXTR	=0 calculate P_l scattering constants (Legendre coefficients,suggested option) =1 read P_l constants from BLOCK 34*
BLOCK 16*	Floating point parameters /14/
1 EV	Initial guess for eigenvalue
2 EVM	Eigenvalue modifier for first eigenvalue change.
3 EPS	Epsilon - accuracy desired (suggested value, EPS=0.0001)
4 BF	Buckling factor, normally 1.420892
5 DY	Cylinder or plane height for buckling correction (may include extrapolation length)
6 DZ	Plane depth for buckling correction
7 DFM1	Transverse dimension for void streaming correction
8 XNF	Normalization factor (If XNF=0.0, no normalization is done), Suggested value, XNF=0.
9 PV	=0.0, or = α_0 if IPVT=2.
10 RYF	Relaxation factor (suggested value = 0.5)
11 XLAL	Point flux convergence criterion if punched greater than zero (suggested value = 2.0*EPS)
12 XLAH	Upper limit for |1.0-lambda| used in linear search, normally =0.05
13 EQL	Eigenvalue change option, 0.001< EQL < 3* EPS
14 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, if IDFM=1 /IM/

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

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

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

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

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

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

BLOCK 34* P_l scatter constants, if IXTR = 1 /JT*MM/
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

The following blocks are required in the execution step of the ANISN routine

BLOCK 17* Distributed source, if IQM=1 /IGM*IM/

BLOCK 18* Shell source, if IPM=1 /IGM*IPM*MM/

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[7] is replaced by the 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 (NTITLE)/A72/

TITLE Job title or job description.

BLOCK 3 Control integers /42/

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

2 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 acceptable for ordinary moderators, N=5 for H2O)

3 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.

4 IGM Number of energy groups (internally set)

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

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

7 IBL Left boundary condition
=0 vacuum
=1 reflective

8 IBR Right boundary condition
=0 vacuum
=1 reflective

=2 white

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

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

11 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)

12 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 FT12F001

Other options in this item are suppressed.

13 MT Total number of cross section blocks including anisotropic cross sections

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

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

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

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

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

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

20 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

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

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

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

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

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

26 IITL Maximum number of inner iterations allowed per group

27 IXM I-direction zone thickness modifier (suppressed)

28 IYM J-direction zone thickness modifier (suppressed)

29 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.

30 IGEOM Geometry
=1 X-Y
=2 R-Z
=3 R-θ

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

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

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

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

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

36 I4 Final fission print indicator
=0 yes
=1 no

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

38 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

39 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

40 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 in the SRAC)

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

42 IF0 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/

1 EV Eigenvalue guess. It is satisfactory to punch =0.0

2 EVM Eigenvalue modifier used only if IEVT > 1.

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

4 XLAL Lower limit for eigenvalue searchs.

5 XLAH Search lambda upper limit.

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

7 EPS Convergence precision.

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

9 POD Parameter oscillation dumper used in eigenvalue search.

10 BHGT Sighed value of buckling height in cm used to simulate the axial leakage by adding an absorption given by

$$\Sigma_{a,BHGT}=\frac{\Sigma_t}{3}\quad(\pi/(BHGT*\Sigma_t\ +\ 1.4209))^2$$

Here 1.4209 is twice the Milne problem extrapolation distance, and Σ_t is the total cross section. Used in (r,θ) and (x,y) geometry only. If the negative value is entered, the Milne extrapolation distance is not added to the buckling height.

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 14 /IM+1/

XRAD Coarse i-mesh boundaries. Must form increasing sequence.

BLOCK 15 /JM+1/

YRAD Coarse j-mesh boundaries. Must form increasing sequence.

BLOCK 16 /IM*JM/

IDCS Cross section zone identification numbers. 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 . IT = sum of IHX vector.

BLOCK 18 /JT/

YDF Axial fine-mesh density factors. Do not enter if ISDF.EQ.0 JT = sum of IHY vector.

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 /1/

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

BLOCK 20 Required if NEDS > 0 /2/

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 /IM*JM/

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

BLOCK 22 /IM*JM/

IXZ X-region numbers by zone.

Note: The following 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 PEACO input data.

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	Required if IQR=1 where JT is total number of fine mesh in j-direction, equal to the sum of IHY vector , and MM is total number of angular mesh; MM = ISN*(ISN+2)/8	/JT*MM/
QR1	Right boundary source (flux) in the in-down directions.	
BLOCK 9	Required if IQR=1	/JT*MM/
QR2	Right boundary source (flux) in the in-up directions.	
BLOCK 10	Required if IQB=1 where IT is total number of fine mesh in i-direction, equal to the sum of IHX vector.	/IT*MM/
QB1	Bottom boundary source (flux) in the in-up directions.	
BLOCK 11	Required if IQB=1	/IT*MM/
QB2	Bottom boundary source (flux) in the out-up directions.	
BLOCK 12	Required if IQT=1	/IT*MM/
QT1	Bottom boundary source (flux) in the in-down directions.	
BLOCK 13	Required if IQT=1	/IT*MM/
QT2	Bottom boundary source (flux) in the out-down directions.	

II.6 TUD : One Dimensional Diffusion

The one-dimensional diffusion routine TUD[9] is preferred to the CITATION by the simpler input requirement. 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/

1 NRMAX Number of regions

2 IG ρ ; geometry

ρ = 0 slab
ρ = 1 cylinder
ρ = 2 sphereer

3 IBOUND Outer boundary condition

=-1 zero flux at the outer boundary, r_B
=0 reflective
=1 zero flux at the group dependent extrapolated distance calculated as

$$D_g \frac{d\varphi_g}{dr} + \omega_g \varphi_g = 0$$

$$\text{with} \quad \omega_g = (1 + 0.7105 \frac{3 \rho D_g}{2 r_b}) / (3+0.7105)$$

=2 zero flux at the constant extrapolated distance

$$D_g \frac{d\varphi_g}{dr} + \omega \varphi_g = 0$$

with ω specified by the input item XLAMD.

Note. The boundary condition at r=0 is always set as reflective.

4 IGUESS Initial flux guess control

=-1 read from FT33F001 prepared by the user
=0 uniform; no input required
=N read from FTONF001 in binary mode as wrriten by ITFLUX

5 IPTXEC Print control for cross sections

=0 skip print
=1 print

6 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=,NNMAX+1),g=1,NGMAX)
```

where F(i,g) = $\varphi((r_i)\Delta E_g$ for slab, cylinder
= $r_i\varphi(r_i)\Delta E_g$ for sphere
= $\varphi(0)$ for sphere & r=0

=5 action 1 and 4

7 IPS Print control for the fixed source distribution.

=0 skip print
=1 print

8 IDOPT Selection of the diffusion coefficients

=1 use D_1 in the macroscopic cross section organization
=2 use D_2 in the macroscopic cross section organization

9 NXR Number of X-regions

BLOCK 2 Integer parameters for the iterative process /6/

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

< 100 > for the fixed source problem
< 10 > for the eigenvalue problem

2 ITMOUT Maximum number of power iterations < 25 >

3 ITBG Minimum number of inner iterations before extrapolation < 5 >

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

5 ITDM Minimum delay between extrapolation < 5 >

6 IPT Control for monitor print < 0 >

=0 skip print
=1 monitor print at each inner iteration

BLOCK 3 Floating point parameters /6/

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

2 EPSO Convergence criterion for outer iteration < .0001 >

3 EPSG Extrapolation criterion < 0.001>

4 RELC Initial over-relaxation factor < 1.2 >

5 OVERX Maximum extrapolation factor < 100. >

6 FACTOR	Under extrapolation factor < 0.8 >	
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	Required even if NXR=0	/NRMAX/
IXR	X-region number by region. Enter all 0, if not necessary	
BLOCK 7		/NRMAX/
RK	Outer radii by region	
BLOCK 8		/1/
BSQ	Transverse buckling in cm^{-2}	
BLOCK 9	Required if IBOUND=2	/1/
XLAMD	Extrapolation distance in cm	

II.7 CITATION : Multi-Dimensional Diffusion

The input for the CITATION routine keeps the original input format[9] so that the same input data except the leading BLOCKs and final two BLOCKs can be used in separate execution of the original CITATION code.

Several additional functions are available as seen in the leading input specification such as the perturbation calculation, the application of anisotropic diffusion coefficients, the material dependent fission neutron spectrum, and the calculations of the generation time and the effective delayed neutron fraction.

BLOCK 1 Control integers, always required /3/

NM ± Number of materials used in this routine (i.e., count the materials read in Section 008).
Negative value activates the entry of the perturbation calculation (see VI.6).

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 through the CITATION routine
=± 2 select D2
=± 3 select D1 or D2 for the particular direction specified by the following BLOCK 3 by mixture.

Negative value activates the entry for the calculations of the material dependent fission neutron spectrum and/or the effective delayed neutron fraction.

BLOCK 2 Additional control integers required if ID < 0 /2/

IXKI Option to prepare the material dependent fission neutron spectrum.

=0 Unique spectrum is used (the spectrum of the fissile mixture which appears first in the material specification will be used).
=1 Spectrum by mixture will be used.

IDELAY Option to write delayed neutron data into MACRO file for the succeeding effective generation time calculation.

=0 Skip
=1 Write

Note: Do not forget to specify non-zero value to NGC12 in Section 001 to activate the adjoint flux calculation.

BLOCK 3 /NM/
Selection of directional diffusion coefficients by mixture, required if ABS(ID)=3

IXYZm, m=1, NM

IXYZ	Dx	Dy	Dz
=1	D1	D1	D1
=2	D2	D1	D1
=3	D1	D2	D1
=4	D2	D2	D1
=5	D1	D1	D2
=6	D2	D1	D2
=7	D1	D2	D2
=8	D2	D2	D2

Note: The correspondence of Dx,Dy,Dz to the real direction will be mentioned in the item for the geometry selection NUAC5 in the CARD 2 in Section 003.
The diffusion coefficients used for the transverse leakage term together with the buckling values entered in Section 024 are taken from Dz.

BLOCK 4 Required if NM < 0 /1/
Control for perturbation calculation

Note: Do not forget to specify non-zero value to NGC12 in Section 001 to activate the adjoint flux calculation.

ICASE Number of cases for perturbation calculation after a CITATION calculation. Repeat BLOCK 5 through BLOCK 8 ICASE times.

BLOCK 5 /A8,2,1/

SAMPLE (A8) Member name of the macroscopic cross section to be used in the perturbed region. Deviation of cross sections from those unperturbed is automatically calculated.

IOPT Option to specify the perturbed region
=-N Whole region of N-th zone
= 1 Region is specified by the mesh intervals given in BLOCK 6.
= 2 Region is specified by the X-Y-Z abscisa given in BLOCK 7.

IDOPT Selection of the diffusion coefficients of the member SAMPLE
=1 use D1 (Dx=Dy=Dz=D1)
=2 use D2 (Dx=Dy=Dz=D2)
=3 Dx=D2, Dy=D1, Dz=D1
=4 Dx=D1, Dy=D2, Dz=D1
=5 Dx=D2, Dy=D2, Dz=D1
=6 Dx=D1, Dy=D1, Dz=D2
=7 Dx=D2, Dy=D1, Dz=D2
=8 Dx=D1, Dy=D2, Dz=D2

Note: The correspondence between each of Dx,Dy,Dz and the direction of geometry will be mentioned in the item for the geometry selection NUAC5 in the CARD 2 in Section 003. The coefficient for the pseudo leakage term uses always Dz.

BKLE Buckling value for the perturbed region
>=0. Constant buckling
< 0. floating number of groups. Group dependent buckling values will be read by BLOCK 8.

BLOCK 6 Required if IOPT=1 /6/

Region specification by mesh intervals
IXYZ(1,1) X-mesh interval number of the right boundary
IXYZ(2,1) X-mesh interval number of the lefy boundary
IXYZ(1,2) Y-mesh interval number of the upper boundary
IXYZ(2,2) Y-mesh interval number of the lower boundary
IXYZ(1,3) Z-mesh interval number of the top boundary
IXYZ(2,1) Z-mesh interval number of the bottom boundary

Note: Even one- or two-dimensional calculation requires 6 entries, feed zero values if not significant.
: The direction X Y Z to the different geometries are mentioned in the description for NUAC5 in CARD 2 in Section 003.

BLOCK 7 Required if IOPT=2 /6/

Region specification by X-Y-Z abscisa.
XYZ(1,1) X-abscisa of the right boundary
XYZ(2,1) X-abscisa of the lefy boundary
XYZ(1,2) Y-abscisa of the upper boundary
XYZ(2,2) Y-abscisa of the lower boundary
XYZ(1,3) Z-abscisa of the top boundary
XYZ(2,1) Z-abscisa of the bottom boundary

Note: A mixed node of spacification is available. If any abscisa coincides with the abscisa of the original mesh point, enter negative floating number of the mesh interval number. It is to avoid the truncation error to compare two almost equal floating numbers.

BLOCK 8 Required if BKLE < 0. /BKLE/

BKLE Group dependent buckling values.

Following data starting at Title cards, Section 001, 003,,,,through Section 999, and a blank card have the same organization as the original CITATION.

Title Card /A72,A72/
Each individual case has two title cards at the beginning.

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 5 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 flux calculation is required. If a negative value is entered, the adjoint flux will be printed out.

NGC13 Option to input the adjoint flux. Enter =1 if the adjoint fluxes are read from FT28. Enter =-1 if both of the forward and adjoint are read from FT28. In this case, all input data is required as if the fluxes are newly calculated.

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 résidues 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 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 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

	Dx	Dy	Dz
=1 One-dimensional slab (X)	X		T**
=2 One-dimensional cylinder (R)	R		T
=3 One-dimensional sphere (S)	R		T

=4	-			
=5	-			
=6	Two-dimensional slab (X,Y)	X	Y	T
=7	Two-dimensional cylinder (R,Z)	R	Z	T
=8	Two-dimensional circle (θ,R)	θ	R	T
=9	Two dimensional hexagonal (H)	X	Y	T
=10	Two-dimensional triangular (T)	X	Y	T
=11	Three-dimensional slab (X,Y,Z)	X	Y	Z
=12	Three-dimensional cylinder (θ,R,Z)	θ	R	Z
=13	Three-dimensional hexagonal (H,Z)*	X	Y	Z
=14	Three-dimensional triangular (T,Z)	X	Y	Z

Note * Hexagonal cylinder has another boundary surface than X,Y, the leakage from this surface is calculated by Dx, and counted in the right and left leakage.

** Symbol T denotes that the transverse leakage term DB^2 are calculated by Dz multiplied by the buckling value specified in Section 024.

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 (required for 2-D)

= 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 slabs)
60 degree rotational symmetry (triangular)
= 3 inverted reflection (180 degree rotational symmetry, slab only)

NUAC14 Bottom boundary condition (required for 2-D)

= 0 extrapolated
= 1 reflected

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 Maximum extrapolation

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 ($-D/\varphi*d\varphi/dx$)
> 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 geometry, a row is a radial traverse. in θ-R and θ-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: Number of groups and scattering range (3I3)

KMAX Number of energy groups

IX28 Number of groups for downscatter

IX29 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 ($\nu*\Sigma_f$)

SIG5 1/v ($1/\sqrt{E}$) cross section for primary mode calculation

SIG6 Power per unit flux, watts/fission times Σ_f for normalization of the flux level and power density maps; if all entries here are zero, then SIG4 is used. Values of Σ_f is filled by the 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 and buckling value (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 4. In this case the KMAX value in Section 008 must be proper

If IND = 3, specify two zone numbers on CARD 3 followed by the group depending buckling on CARD 4 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.

CARD 3: Zone indicators; required if IND=3 (2I3)

CARD 4: Group dependent bucklings, required if IND=2,3 (6E12.5)

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 one blank card after the 999 card. The CITATION routine compiled in SRAC allows only one case.

CARD 1: 999

CARD 2: (blank)

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

BLOCK 9 Material number specification /NM/

This item is to select and to number the materials used in the 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 10 X-region specification, required if NXR>0 /IZN/

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 /1/

NMAT Signed number of materials to be used. Here "material" includes several kinds of mixtures;

1) A mixture with composition to which macroscopic cross sections will be formed in the case;
2) A mixture without composition to which macroscopic cross sections were formed in the previous case and are kept in MACROWRK file;
3) A mixture with composition to which macroscopic cross sections were formed in the previous case and are kept in MACROWRK file; The macroscopic cross sections will not be further formed but the mixture forms a cell to which resonance integral by IRA or PEACO will be calculated together with a new resonant mixture. The composition can not include any resonant nuclide.
4) A homogenized mixture with a CASENAME label to which macroscopic cross sections were formed in the previous case;
5) A homogenized mixture with the CASENAME label of this case to which macroscopic cross sections are formed in this case, and will be used in the later step in this case. Note that it is not necessary to specify the homogenized mixtures which are made in this case but will not be used in the later step in the case.
6) A fictitious mixture with composition to be used only for reaction rate calculation.
7) A fictitious mixture without composition to be used only for reaction rate calculation.

Note: If a negative number is specified, the atomic number densities at the final burn-up step will be stored on FT07 so as to be used in the material specification in a restart run for the succeeding burn-up step. Moreover an automatic restart is prepared by checking whether the next cell burn-up steps will terminate within the specified cpu and I/O limit. This function is effective only if IC20 (cell burn-up indicator) =1.

BLOCK 2 /A8,2,3/

MTNAME Material identification expressed by eight characters, composed of five tags as mmmmebxp which appears as MEMBER name in macroscopic cross section files.

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 Tag internally specified to choose the neutron energy range of a set of macroscopic cross sections, because the sets 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. The user can realize in the member list of MACROWRK file.

b-tag Tag to indicate burn up step as '0' , '1', '2', ... ,'9' and 'A', 'B',... corresponding to fresh, step 1, step 2,.., step 9, and step 10, step 11,... respectively, when the cell burn-up calculation is done to make up this mixture. Otherwise this tag is filled by the character given in the input.

x-tag Tag to specify X-region number which is composed at the MIXX step for 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 seven-th character of the mixture name. In the latter case, this tag is used to discriminate the effective microscopic cross sections stored in the MICREF file when a resonant nuclide is commonly used in more than one mixture.

p-tag Tag internally used to specify Legendre component and also whether fine or coarse in energy structure. Any character given 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: storage in MACROWRK file is not yet prepared : fill 0 for the present version.

NISO Number of nuclides to compose the mixture. If specified =0 the program assumes that the mixture exists already in MACROWRK file or will be created in the MIXX step. Contrary 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, give the composition for all constituent mixtures in the cell to evaluate the resonance absorption even if they are already made up.

TEMP Physical temperature of the mixture (degree K). Although any temperature can be accepted for the interpolation of f-table, the data on THERMAL and MCROSS are tabulated on the fixed temperature points. The program replaces it by the nearest value in the tabulated temperature (See Dictionary VII.6). Actually this temperature is effective in calculating resonance absorption. Confining the calculation in the energy range higher than thermal cut-off, this temperature tabulation can be replaced by that given in II.2. Since the temperature in thermal energy range is specified by the temperature tag of each nuclide IDENT given in BLOCK 4 or BLOCK 4'. The thermal library must be prepared to correspond to the new temperature tabulation.

XL l ; the mean chord length of the resonance absorber lump used in the interpolation of resonance shielding factor in MACROF, MACROT and IRA routines, and also used in PEACO for the constant to yield a non-dimensional blackness for the interpolations of the collision probabilities. Give a suitable value to the mixture including resonant nuclides.

Generally l is defined as

$$l = 4\frac{V}{S} \quad (\text{in } cm)$$

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

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

For more complicated geometry which contains several absorber lumps in a unit cell, enter 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 background cross section in the interpolation of resonance shielding factor (effective only if IC3=0 in BLOCK 3 of Sect.II.1 is specified; enter 1.0 for the homogeneous approximation) If C<0 is entered, BLOCK 4' will be read instead of BLOCK 4, which includes nuclide-dependent Dancoff factor.

Remind that l and C is used in a conventional table-look-up for the heterogeneous effect to the background cross section σ_b in the following form; (See Eq.(VI.3.3-20))

$$\sigma_b = \sum_{\mu' \neq \mu} N_{\mu'} \sigma_{t\mu'} / N_\mu + g(C)(1-C)/l/N_\mu$$

where $g(C)$ is a geometric correction factor defined by Eq. (VI.3.2-8) to the simple NR approximation, μ denotes the resonant nuclide under cosideration and μ' any nuclide admixed in the material.

BLOCK 3 Required if NCOM > 0 /(NCOM)A4/

MCOM Comment for the mixture (storage is obsolete)

BLOCK 4 Required if NISO > 0 NISO times /A8,2,1/

IDENT Nuclide identification expressed by eight characters composed by five tags 'Xzzmcbit' as already appeared in Sect.II.2.

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

zz-tag 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 '9' for PU-239. 'N' is used to specify the element with natural abundance. 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 Chemical compound status tag to select the proper thermal scattering law (see Dictionary VII.2). If the next item IRES is specified to 1, this tag is internally set to 'F' or 'T' to select 'fast' or 'thermal' data because the whole sets of effective cross sections are stored in MICREF file. If no effective cross section is found in MICREF file, infinite dilution cross sections will be taken from FASTU or THERMALU file.

b-tag Punch '0' to search the cross sections stored in 'Users' libraries. To specify the data in MICREF file with IRES=1, punch the sixth character of the mixture name with which these effective cross sections are associated. Remind that the sixth character is used for burn-up indicator.

i-tag Punch '0' to search the cross sections stored in 'Users' libraries. Otherwise, punch the seventh character of the mixture name to which the effective microscopic cross sections are produced. The latter search in MICREF file will be activated by setting 1 for the next item IRES.

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

Note: The discrepancy of the mixture temperature specified by TEMP in the previous BLOCK 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. This item is closely related with the resonance indicators stored in the library; IFS of F-table, IRP of resonance level parameters for IRA and PEACO.

Note: When the cell burn-up process is specified, this indicator to the depleting nuclide is replaced by the value stored in the burn-up library, and that to the non-depleting nuclide is set to IRES=0 at the BURNIN routine(see VII.6).

=0 No process for resonance treatment even if IRP=1, however, to the resonant nuclide which is specified by IFS=1, the table-look-up will be done at the MACROF routine.

=1 Effective cross sections which have been calculated in the previous case will be read from MICREF file as if this nuclide is non-resonant.

=2 Treatment as a resonant nuclide. No process for non-resonant nuclide is taken.

=3 Effective only in the PEACO routine and if this nuclide is a constituent of non-resonant mixture. The Pij tabulation will be made by each fine group considering the change of cross section of this nuclide by group.

=4 Effective only in the PEACO routine and if this nuclide is a constituent of non-resonant mixture, two-dimensional Pij tabulation will be made assuming the behavior of cross section of this nuclide as 1/v.

Note: Unless IRES=3 or IRES=4 is given to the nuclides of non-resonant mixture, the Pij tabulation will be made considering the variation of cross sections of resonant mixture but assuming the value of the highest energy group is constant through the resonance II range.

IXMICR Indicator to write the effective cross sections into the effective microscopic cross section file MICREF and/or the resonance cross section file MCROSS.

=0 No edit. IF IC5-2 is entered, and IFS=1 is specified on FASTU, and this nuclide is a constituent of the mixture in the cell, the effective cross sections in stead of dilute cross sections will be written to be used in PEACO routine, as if IXMICR=1 is entered.

=1 Write the effective group cross sections into the MICREF file in the same format of the user fast and thermal libraries if the nuclide has any resonant property in the corresponding energy range. If the cell burn-up routine is activated, this item is automatically set to 1 for all possible depleting nuclides so that the effective fission

and absorption cross sections are written on FT52 for the burn-up calculation. The effective cross sections are written in MICREF file even if IXMICR=0 is entered for the nuclide who has resonant property but ultra-fine group cross sections are not prepared in MCROSS file, if IC5=2 is chosen.

=2 Write the microscopic cross sections averaged in the mixture into MCROSS file (if IC5=2). In this case a private MCROSS file must be prepared to allow the write option. This option may be utilized for the output of primary step in the treatment of double heterogeneity.

=3 both of functions 1 & 2.

Note: When the cell burn-up process is specified, this indicator, if this nuclide is depleting, is replaced by the value stored in the burn-up library, for non-depleting nuclide IXMICR=0 is set at the BURNIN routine (see VII.6).

: The IDENT (nuclide identification) of effective microscopic cross sections is formed as same as the IDENT of this nuclide except cbi-tag (three characters). 'c' tag is replaced by 'F' or 'T' to stand for the energy range, and the 'bi' tag is taken from the mixture name which contains the nuclide. If the cell burn-up calculation is executed, the b-tag in sixth character is overridden by the burn-up step indicator. When a common nuclide is a constituent of two mixtures used in a case or in succeeding cases, the member name must not be identical to be distimguished.

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

BLOCK 4' Required if DC on BLOCK 2 < 0., instead of BLOCK 4 NISO*/A8,2,2/

IDENT same as in BLOCK 4

IRES same as in BLOCK4

IXMICR same as in BLOCK 4.

DN same as in BLOCK 4.

DCN Dancoff correction factor of this nuclide. Effective if IC3=0 in Block 3 of Sect.II.1.

Repeat BLOCK 2 through BLOCK 4 or BLOCK 4' NMAT times

II.9 Reaction Rate Calculation

A common edit can be executed after any component routine for flux distribution. The edit routine REACT permits the calculation of reaction (capture, fission) rate distribution of the detector with or without the filter. The reaction rates are calculated as follows; For the detector without filter

$$R(r) = \sum_g \Sigma_{xg} \cdot \Phi_g(r) \qquad (II.9\text{-}1)$$

For the detector with filter

$$R(r) = \sum_g f_g \quad \Sigma_{xg} \cdot \Phi_g(r) \qquad (II.9\text{-}2)$$

This edit also permit the calculation of the spectrum indice such as ρ_{28} δ_{25} δ_{28} and C^*. They are defined as follows;

The resonance capture ratio:

$$\rho^{28} = \int_{E_{cd}}^{\infty} \sigma_c^{28}(E) N^{28} \varphi(E) dE \Big/ \int_0^{E_{cd}} \sigma_c^{28}(E) N^{28} \varphi(E) dE \qquad (II.9\text{-}3)$$

The epithermal fission ratio:

$$\delta^{25} = \int_{E_{cd}}^{\infty} \sigma_f^{25}(E) N^{25} \varphi(E) dE \Big/ \int_0^{E_{cd}} \sigma_f^{25}(E) N^{25} \varphi(E) dE \qquad (II.9\text{-}4)$$

The fast fission ratio:

$$\delta^{28} = \int_0^{\infty} \sigma_f^{28}(E) N^{28} \varphi(E) dE \Big/ \int_0^{\infty} \sigma_f^{25}(E) N^{25} \varphi(E) dE \qquad (II.9\text{-}5)$$

The capture ratio:

$$C^* = \int_0^{\infty} \sigma_c^{28}(E) N^{28} \varphi(E) dE \Big/ \int_0^{\infty} \sigma_f^{25}(E) N^{25} \varphi(E) dE \qquad (II.9\text{-}6)$$

where E_{cd} is the Cadmium cut off energy, and the superscripts ; 25 and 28 denotes the nuclides ^{235}U and ^{238}U, respectively. According to the input specification, any other combination of the nuclides can be accepted.

The epi-cadmium reactions are calculated as;

$$R_{epi} = \sum_g f_g \ \sigma_{\mu mg} \ \frac{\varphi_{ig}}{\varphi_{xg}} \quad \Phi_g(r) \cdot N_{\mu m} \qquad (II.9\text{-}7)$$

The sub-cadmium reactions are calculated as;

$$R_{th} = \sum_g (1 - f_g) \ \sigma_{\mu mg} \ \frac{\varphi_{mg}}{\varphi_{xg}} \ \Phi_g(r) \cdot N_{\mu m} \qquad (II.9\text{-}8)$$

In the above equations, $\sigma_{\mu mg}$ denotes the effective microscopic cross section of the nuclide μ included in the mixture m of the energy group g, φ_{ig} the flux of ith spatial region composed of the mixture m, φ_{xg} the average flux of x-th X region in the cell calculation, and $\Phi_g(r)$ is the flux at r in the core calculation.

This routine is applicable for both of the routines called by IC2 and IC12 specified in BLOCK 3 of II.1. If both problems are executed in a case, the edit is called after the routine called by IC12.

The cross sections required in this step have to be prepared before entering this routine (macroscopic in MACRO or MACROWRK for reaction

rate calculation, and microscopic in MICREF for spectrum index). For the latter purpose, IXMICR=1 in II.8 must be specified to store the effective microscopic cross sections in the MICREF file.

Following input data are required if IC18 > 0 in Sect.II.1.

BLOCK 1 Control integers for reaction rate calculation /4/

IOPT(1) Number of detectors used without filter.

IOPT(2) Number of detecters used with filter.

IOPT(3) Number of cases to calculate the spectrum index.

MREC Signed number of mixtures which contain the nuclides with IXMICR =1. Care should be taken in depletion problem where IXMICR is automatically assigned and this edit works after the final burn-up step.

Negative number activates a spatial integration of the reaction of fictitious materials specified by BLOCK 5 to obtain the convertion ratio of the whole core.

BLOCK 2.1 Control for the detectors without filter, required if IOPT(1)>0. /A8,2,0/

MTNAME Member name of the detector cross sections stored in MACROWRK or MACRO file.

IREAC Specification of reaction
=0 fission
=1 capture
=2 fission and capture

NMESH Number of mesh intervals to calculate the reaction rates.

BLOCK 2.2 Spatial position by mesh number /3*NMESH/

MESH(i,n) i=1,3, n=1,NMESH
i=1 corresponds to X-direction, i=2 to Y-direction, i=3 to Z-direction. In one- or two-dimensional case, enter zero values into irrelevant data.

Note: If the output of the collision probability method is processed, specify R-region number as if one-dimensional calculation.

Repeat BLOCK 2.1 through BLOCK 2.2 IOPT(1) times.

BLOCK 3.1 Control for detectors with filter, required if IOPT(2)> /A8,2,0/

MTNAME Member name of the detector cross sections stored in MACROWRK or MACRO file.

IREAC Specification of reaction
=0 fission
=1 capture
=2 fission and capture

NMESH Number of mesh intervals to calculate the reaction rates.

BLOCK 3.2 Specification of mesh intervals. /3*NMESH/

MESH(i,n) i=1,3, n=1,NMESH
i=1 corresponds to X-direction, i=2 to Y-direction, i=3 to Z-direction. In one- or two-dimensional case, enter zero values into irrelevant data.

BLOCK 3.3 Filter transmission by energy group /IGMAX/

FGg, g=1,IGMAX
Transmission has to range between 0.0 and 1.0. If the exact values are required around the cut off energy, obtain the collision probabilities for an isolated cell composed of a detector covered by a filter. Calculate Gi defined in Eq. VI.1-22 where Gi = Gd (i is detector region) the probability that a neutron impinging into filtered detector from the outer surface with isotropic angular distribution using printed Pis. IGMAX = NEF+NET for the fine group calculation and IGMAX= NERF+NERT for the coarse group calculation.

Repeat BLOCK 3.1 through BLOCK 3.3 IOPT(2) times.

BLOCK 4.1 Input for spectrum index, required if IOPT(3)>0. /MREC/

NRECi i=1,MREC
Number of nuclides which have the IXMICR =1 in i-th mixture counted in MREC. Count i in the order of appearance in the material specification in II.8.

BLOCK 4.2 Specification of nuclides for spectrum index /A0,6,IGMAX/

MPOSI Position of the mixture in MREC mixtures.
LU235 Position of the nuclide U235 in NREC(MPOS) nuclides.
LU238 Position of the nuclide U238 in NREC(MPOS) nuclides.
The nuclide U235 U238 may be read as for another combination such as U233 and Th232.
IX X-position to take the neutron spectrum
IY Y-position to take the neutron spectrum
IZ Z-position to take the neutron spectrum
FGSg,g=1,IGMAX
Filter transmission by energy group to obtain the epithermal index. Transmission ranges between 0.0 and 1.0.

Repeat BLOCK 4.2 IOPT(3) times.

BLOCK 5.1 Required if MREC<0 /A8,1,0/

MTNAME Member name of a fictitious mixture composed only by fissile nuclides like U-233, U-235, Pu-239 and Pu-241.

NZONE Number of zones to be integrated

BLOCK 5.2 /A8,NZONE,0/

MTNAME Member name of a fictitious mixture composed only by fertile nuclides like Th-232, U-238, Pu-240, and Pu-242.

IZONEi i=1,NZONE
Zone numbers

Repeat BLOCK 5.1 and 5.2 IMREC1 times.

II.10 Cell 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). The cell calculations of the final step is skipped.

Note: If NMAT' in Sect. II.8 is specified by a negative number, the atomic number densities at the final step will be stored on FT07 so as to be used in the material specification in a restart run for the suceeding burn-up step. Moreover the final step is automatically judged by checking whether or noy the next cell burn-up steps will terminate within the specified cpu and I/O limit.

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 /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 count the initial step) in unit specified by IUNIT. A modification is made for the action at the burn-up step to yield only the composition, but to skip the cell calculation.

II.11 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 /A8/

IDENT 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 Control integers /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 VII.6)

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

II.12 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 Plot control /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 an R-region per figure
- =3 plot neutron spectra by up-to five R-regions in a figure in the energy ranges which are specified in BLOCK3 and BLOCK4.
- >0 print modified group cross sections

BLOCK 2 Required if IC5=-2. /NM/

When more than two mixtures containing any resonant nuclide are used in a cell, due to the restriction on the PEACO routine, a resonant material is assumed to be one of two resonant mixtures in the interpolation process of the collision probabilities. Such an approximation may be satisfied in case of the burn-up calculations of multiple fuel regions which contain similar composition each other through the burn-up steps.

IRX
- =0 non-resonant region
- =1 assigned to the first resonant material
- =2 assigned to the second resonant material

BLOCK 3 Required if IPLOT=3 /4/

EL The lower energy boundaries for plotting the spectra. Four ranges are assumed.

BLOCK 4 Required if IPLOT=3 /4/

EH The upper energy boundaries for plotting the spectra. Four ranges are assumed.

The first graph will be drawn between EL(1) and EH(1). Then the second..... until the fourth.

III Core Burn-up and Fuel Management

III.1 COREBN code

The COREBN code permits 2-D and 3-D burn-up calculation of a reactor core and fuel management.

The operation of a reactor is generally described by the spatial distribution of physical quantities such as neutron flux, nuclide concentration and temperature. The depletion calculation has to take into account of time-behavior of these parameters. The usual neutronic calculation treats these changes as those of few group cross sections in which consequent change of neutron spectrum is included.

In the SRAC code system, the depletion process is divided into two steps. First, a cell burn-up process implemented in the main program yields few group macroscopic cross sections at each burn-up step. These are prepared on the discrete values of fuel temperature, coolant temperature and degree of burn-up for a cell.

An auxiliary program COREBN to execute core burn-up utilizes this tabulation of few group macroscopic cross sections. Owing to the scheme to interpolate the data for a three dimensional, realistic continuous distribution of the physical quantities in a reactor core can be simulated as far as computer time and space is available. The diffusion routine calculates the power distribution to give increased burn-up of each spatial node.

Information before initial and after final step of burn-up is read/written from/to a history file which keeps the history of individual fuel element. It serves as fuel management.

It is a feature of the code that macroscopic cross sections are essential mediums between cell burn-up and core burn-up calculation. Therefore nuclide concentrations are not indispensable variables to this code, but calculated and written in the history file for fuel management.

III.1.1 Process

We shall follow briefly the process of the code illustrated in Fig.III.1-1.

(1) Get input data : operation condition, loading pattern of fuel elements, temperatures.
(2) Read degree of burn-up for each spatial node from the history file.
(3) Interpolate macroscopic cross sections for each spatial node corresponding to the degree of burn-up, fuel temperature and moderator temperature.
(4) 2-D or 3-D diffusion theory core calculation.
(5) Increase degree of burn-up by multiple of operation time and power level at each node.
(6) Repeat steps (3), (4) and (5) for the given number of steps.
(7) Interpolate nuclide concentrations from the table by the degree of burn-up.
(8) Update the history file.

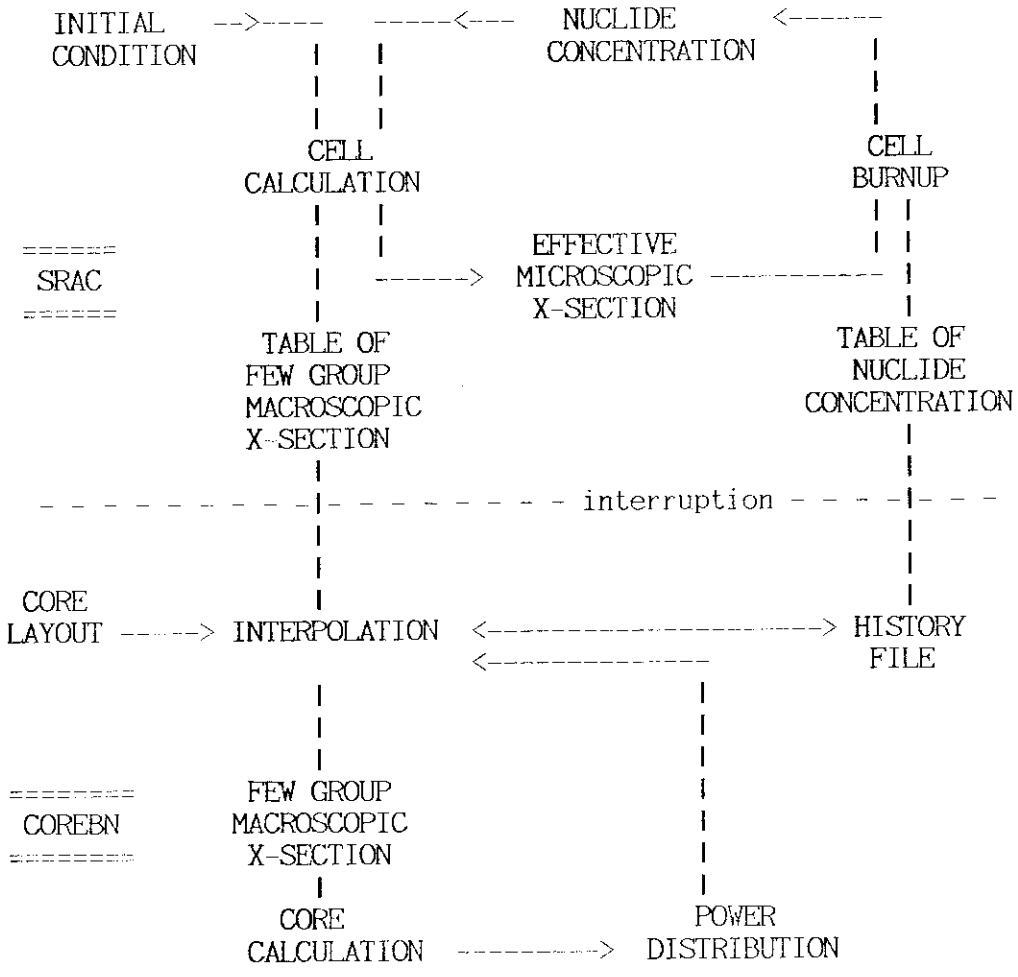


Fig. III.1-1 Flow diagram for burn-up calculation

III.1.2 Formulations

1 Interpolation of macroscopic cross sections and nuclide concentrations

Macroscopic cross sections and nuclide concentrations are obtained by an interpolation scheme described below as a function of three variables, degree of burn-up, fuel temperature and moderator temperature.

First, two burn up steps to include the required burn-up (MWD/cm^3) are found. At each step, a function $F(x,y)$ is obtained from four values $F_1 = F(x_1, y_1)$, $F_2 = F(x_2, y_1)$, $F_3 = F(x_1, y_2)$ and $F_4 = F(x_2, y_2)$ which are the tabulated values corresponding to the possible combinations of either of variables x_1, x_2 and either of variables y_1, y_2 where $x_1 < x < x_2$ and $y_1 < y < y_2$.
Let

$$R_x = \frac{x - x_1}{x_2 - x_1}, \qquad R_y = \frac{y - y_1}{x_2 - x_1}$$

and

$$Fx_1 = F_1 + (F_2 - F_1) * R_x, \qquad Fx_2 = F_3 + (F_4 - F_3) * R_x$$

Finally the interpolated value is obtained by

$$F(x,y) = Fx_1 + (Fx_2 - Fx_1) * R_y$$

The required value is further interpolated between two values of F(x,y) for two steps.

Special case is taken care of. If both of fuel and moderator temperatures are coincident to the tabulated temperatures, no interpolation is done. If either of fuel and moderator temperatures is coincident to any of tabulated ones, interpolation on one variable is executed. Note that as no extrapolation is allowed, the tabulation has to cover all possible range of variables.

2 Assignment of fuel and moderator temperatures

The assignment of fuel and moderator temperature is controlled by input which specifies one of the following options.

(1) A fixed value is used over all nodes.

(2) Values are given by node.

(3) Values are calculated by node (not yet available)

The following formulation is used to calculate the temperature of a node.

$$T(h) = T_{in} + A \int_0^h p(x,y,z)\, dz$$

where T(h) is temperature at h of a channel expressed by x and y, T_{in} the inlet temperature, p(x,y,z) power density at x,y,z and A the constant. For the down-flow, the upper and lower limit of the integration has to be exchanged.

At the initial burn-up step, the temperature distribution has to be given because no power distribution has yet been calculated.

3 Correction for Xe concentration by neutron flux level

The xenon concentration changes mainly by the neutron flux level. The values interpolated by degree of burn-up and temperatures from the tabulated values which were calculated under the fixed power density should be replaced by the value determined by the local neutron flux for each node.

The xenon concentration in the steady state is given by

$$N_{Xe} = \frac{\gamma \Sigma_f \varphi}{\lambda_{Xe} + \sigma_{Xe} \varphi}$$

where γ : fission yield of Xe
Σ_f : macroscopic fission cross section
φ : neutron flux
λ_{Xe} : decay constant of Xe
σ_{Xe} : microscopic absorption cross section of Xe

If N_{Xe0} is obtained by table-look-up under the flux φ_0, the correction to Xe concentration is given by

$$\Delta N_{Xe} = \frac{N_{Xe0}}{\varphi_0} \frac{(\varphi - \varphi_0)\lambda_{Xe}}{\lambda_{Xe} + \sigma_{Xe}\varphi}$$

The cross sections Σ and σ are taken from those of the lowest group assuming few-group calculation. The flux φ is taken from that calculated at the previous step. At the initial step, this correction is not carried out. The correction of Xe concentration is reflected on the macroscopic absorption cross sections for depleting fuel.

III.1.3 Member name on PDS file for tabulation

The macroscopic cross sections provided by the cell burn-up routine are read by the COREBN code as the tabulated values from a PDS file in which the data are read/written by member name of eight characters(columns). Each column is assigned to denote proper physical meaning under the following rules.

Structure of member name IIFMEBIC
Column 1,2 and 7 : Arbitrary alphabetic characters used for material identification.
Column 3 : Numeric character as fuel temperature indicator (see Dict. VII.6)
Column 4 : Numeric character as moderator temperature indicator (see Dict. VII.6)
Column 5 : Alphabetic character as energy range (F for fast, T for thermal and A for whole energy range)
Column 6 : Numeric character as cell burn-up step indicator (0,1,2,3,4,5,6,7,8,9,A,B,.....)
Column 8 : Numeric character as energy mesh indicator (0 for coarse and 2 for fine mesh)

Columns 1,2,5,6,7,8 may be carried over as specified in the cell burn-up step. Care should be taken to columns 3,4 to which any character might be accepted in the cell burn-up step. The material identification of three characters must be common in a kind of fuel elements whether new or old.

The tabulation of nuclide concentration is also referred by a member name of which structure is symbolized as IIFM'NDEN'.

Column 1,2 : Same alphabetic characters as used in a material identification to a material
Column 3 : Numeric character as fuel temperature indicator

Column 4 : Numeric character as moderator temperature indicator

Columns 5,6,7,8 : Fixed character string 'NDEN'

The member names for non-fuel materials may be let as made up by the SRAC code.

III.1.4 Z division on 3-D calculation

In a 3-D calculations, a spatial division is made on Z direction. A unit called "node" is assumed to have uniform degree of burn-up and nuclide density. The boundary of a node must be identical with that of

zone to which the cross section is assigned in the diffusion calculation.

When a follower fuel element which follows the movement of control element is used, division of uniform interval is required to keep the volumes of nodes during successive batches. If only non-fuel material moves along Z-direction, arbitrary division is allowed.

III.1.5 Content of history file

A history file is used to keep the information for fuel management and operation. Before executing the initial depleting calculation, the reactor system and fuel elements have to be registered in the file. An editing program is available for this purpose. This program serves also for the correction and printing out of the contents. The usage of this program will be described in III.2.

The information in the history file is used to restart a next batch after some interruption, and a new file to keep the whole history is written at the end of depletion calculation. It is recommended to keep an old file at certain steps as the back-up of the file for recalculation.

Followings are the definition of the common variables and I/O format.

(1) Definition of the common variables

Corresponding to each record written in the history file, the common variables in the history editing program are arranged as follows;

```
COMMON /REC1/ HEADER(18,2),UDATE(2)
COMMON /REC2/ IGEOM,NREGI,NREGJ,NREGKB,NMAT,NTNUC,NHVNUC,
              NMESHX(MAXX),XX(MAXX),NMESHY(MAXY),YY(MAXY),
              NMESHZ(MAXZ),ZZ(MAXZ),MTNAME(2,MAXMTN),IFORS(MAXMTN),
              VOLFS(MAXMTN),VOLFR(MAXMTN),L235,LXE5,NISO(MXISO),
              IHVNUC(MXHVIS),AMASS(MXHVIS),P1E(MAXX,MAXY)
COMMON /REC3/ NBATCH
COMMON /REC4/ OCOM(18,2),NOB,NOC,NSBSTP,IDATE(2),PERIO(NSBSMX),
              AVRPO(NSBSMX),ACPOW,AVFTPH,AVMTPH,NOFCOR,ID(2,NOFCMX),
              LXYZ(2,3,MXYZF,NOFCMX),IFSAME(NOFCMX),NOCCOR,
              IDC(2,NOCCMX),LCXY(2,2,MXYZC,NOCCMX),
              CLOCZ(NSBSMX,NOCCMX),LCZ(NSBSMX,NOCCMX)
COMMON /REC5/ NFE,NFT,NOT,NRKFMX,NRK1MX,NRKOMX
COMMON /REC6/ FTCOM(2,MAXFT),NREGKF(MAXFT),NREGK1(MAXFT),
              VOLF(MAXK1,MAXFT),TZINV(MAXK1,MXHVIS,MAXFT),
              MATSPC(MAXKF,MAXFT)
COMMON /REC7/ OTCOM(2,MAXOT),NREGKO(MAXOT),MATSPO(MAXKO,MAXOT)
COMMON /REC8/ IDENT(2),NFTYPE,BURNUP(2),BURNUZ(2,MAXK1),
              ZINV(MXISO,MAXK1),STATUS,IBCORR,IDATEF(2),NLOAD,
              LXYZF(2,3,MXYZF),NHIS
COMMON /REC9/ NOB1,NOC1,NSBST1,IDATEH(2),PERIO1(NSBSMX),
              POWZ(NSBSMX,MAXK1),ZFTEMP(NSBSMX,MAXKZ),
              ZMTEMP(NSBSMX,MAXKZ),ACBURN,ZABURN(MAXK1),NHLOAD,
              LXYZ1(2,3,MXYZF)
```

Note. Array size set by following parameter statement in the history

file edit code.

```
      PARAMETER (MAXX=100  ,MAXY=100  ,MAXZ=50   ,
                 MAXMTN=100,MXISO=200 ,MXHVIS=11 ,
                 NSBSMX=10 ,
                 NOFCMX=100,MXYZF=4   ,
                 NOCCMX=50 ,MXYZC=4   ,
                 MAXFT =50 ,MAXKF=50  ,MAXK1=50  ,
                 MAXOT =50 ,MAXKO=50  ,
                 MAXK1I=MAXK1*MXISO   ,
                 MAXK1H=MAXK1*MXHVIS            )
```

(2) Format in History file

Record in history file is written as follows;

```
      REWIND       IFILE
      WRITE(IFILE) HEADER,UDATE
      WRITE(IFILE) IGEOM,NREGI,NREGJ,NREGKB,(NMESHX(I),XX(I),I=1,NREGI),
                   (NMESHY(I),YY(I),I=1,NREGJ),(NMESHZ(I),ZZ(I),I=1,
                   NREGKB),NTNUC,L235,LXE5,(NISO(I),I=1,NTNUC),NHVNUC,
                   (IHVNUC(I),AMASS(I),I=1,NHVNUC),NMAT,
                   ((MTNAME(J,I),J=1,2),IFORS(I),VOLFS(I),VOLFR(I),
                   I=1,NMAT)
      DO 100 KB=1,NREGKB
      WRITE(IFILE) ((P1E(I,J,KB),I=1,NREGI),J=1,NREGJ)
  100 CONTINUE
      WRITE(IFILE) NBATCH
      IF(NBATCH.EQ.0) GO TO 120
      DO 110 NH=1,NBATCH
      WRITE(IFILE) ((OCOM(J,I),J=1,18),I=1,2),NOB,NOC,NSBSTP,
                   (IDATE(I),I=1,2),(PERIO(I),I=1,NSBSTP),
                   (AVRPO(I),I=1,NSBSTP),ACPOW,AVFTPH,AVMTPH,NOFCOR,
                   ((ID(J,I),J=1,2),(((LXYZ(K,J,N,I),K=1,2),J=1,3),
                   ,N=1,MXYZF),IFSAME(I),I=1,NOFCOR),NOCCOR,
                   ((IDC(J,I),J=1,2),(((LCXY(K,J,N,I),K=1,2),J=1,2),
                   N=1,MXYZC),(CLOCZ(J,I),J=1,NSBSTP),
                   (LCZ(J,I),J=1,NSBSTP),I=1,NOCCOR)
  110 CONTINUE
  120 CONTINUE
      WRITE(IFILE) NFE,NFT,NOT,NRKFMX,NRK1MX,NRKOMX
      IF(NEF.GT.0) THEN
      WRITE(IFILE) ((FTCOM(J,I),J=1,2),NREGKF(I),NREGK1(I),I=1,NFT),
                   ((VOLF(K,I),(TZINV(K,J,I),J=1,NHVNUC),K=1,NREGK1(I)),
                   I=1,NFT),((MATSPC(J,I),J=1,NREGKF(I)),I=1,NFT)
      IF(NOT.GT.0) THEN
      WRITE(IFILE) ((OTCOM(J,I),J=1,2),NREGKO(I),I=1,NOT),
                   ((MATSPO(K,I),K=1,NREGKO(I)),I=1,NOT)
      IF(NEF.LE.0) GO TO 999
      DO 150 NF=1,NFE
      WRITE(IFILE) (IDENT(I),I=1,2),NFTYPE,(BURNUP(I),I=1,2),((BURNUZ(J
                   ,K),J=1,2),K=1,NREGK1(NFTYPE)),
                   ((ZINV(J,K),J=1,NTNUC),K=1,NREGK1(NFTYPE)),
                   STATUS,IBCORR,(IDATEF(J),J=1,2),NLOAD,
                   (((LXYZF(J,I,K),J=1,2),I=1,3),K=1,NLOAD),NHIS
      IF(NHIS.LE.0) GO TO 150
      DO 140 NH=1,NHIS
      WRITE(IFILE) NOB1,NOC1,NSBST1,(IDATEH(J),J=1,2),(PERIO1(J),J=1,
```

```
     NSBST1),((POWZ(J,K),J=1,NSBST1),K=1,NREGK1(NFTYPE)),
     ((ZFTEMP(J,K),J=1,NSBST1),K=1,NREGKB),
     ((ZMTEMP(J,K),J=1,NSBST1),K=1,NREGKB),
     ACBURN,(ZABURN(J),J=1,NREGK1(NFTYPE)),NLOADH,
     (((LXYZ1(J,I,K),J=1,2),I=1,3),K=1,NLOADH)
140 CONTINUE
150 CONTINUE
999 CONTINUE
```

(3) Physical meaning of the variables

Followings are physical meaning of the variables listed above. Character type variables are marked by (A).

Record 1. Header

HEADER(A)	File identification
UDATE(A)	Date of the latest update of the file

Record 2. Geometry and miscellaneous for diffusion calculation

IGEOM	Geometry option = 1 to 5 Not available = 6 2-D slab (X,Y) = 7 2-D cylinder (R,Z) = 8 2-D circle (θ,R) = 9 2-D hexagonal(H) =10 2-D triangular(T) =11 3-D slab (X,Y,Z) =12 3-D cylinder(θ,R,Z) =13 3-D hexagonal(H,Z) =14 3-D triangular(T,Z)
NREGI	Number of vertical regions
NREGJ	Number of horizontal regions
NREGKB	Number of planes
NMESHX	Number of mesh points for each vertical region
XX	Region width for each vertical region
NMESHY	Number of mesh points for each horizontal region
YY	Region width for each horizontal region
NMESHZ	Number of mesh points for each plane
ZZ	Region width for each plane
NTNUC	Number of nuclides
L235	Position of ^{235}U in the nuclide table
LXE5	Position of ^{135}Xe in the nuclide table
NISO(A)	Nuclide name
NHVNUC	Number of heavy nuclide for inventory calculation
IHVNUC	Position number of heavy nuclide in the heavy nuclide table
AMASS	Mass of each heavy nuclides
NMAT	Number of materials
MTNAME(A)	Material name on MACRO file
IFORS	Option for fuel or non-fuel
VOLFS	Volume of cell (cm^3) which is used to estimate power density of the cell from the total power specified to the cell
VOLFR	Actual fuel volume fraction in the fuel element

Record 2.1 Thermal neutron flux for Xe-correction

P1E Thermal neutron flux at each zone

Record 3. Control of operation

NBATCH Number of operation batches

Record 4. Operation record in a batch

OCOM(A) Comments for an operation batch. An execution of the COREBN code pursues depletion during a batch in which no refuelling nor withdrawal of control fuel element is permitted.
NOBM Sequential batch number
NOC Cycle number, one or more than one batches are included in a cycle
NSBSTP Number of burn-up steps in the batch
IDATE(1) Date of start-up (YYMMDD)
IDATE(2) Date of shut-down (YYMMDD)
PERIO Operation period for each burn-up step (hour)
AVRPO Average power for each burn-up step (MW)
ACPOW Integrated power (MWD)
AVFTPH Average fuel temperature (K)
AVMTPH Average moderator temperature
NOFCOR Number of fuel elements in the core
ID(A) Identification of each fuel element

Following six items specify the location of a fuel element in the core in term of "region". Remind that a combination of a vertical region, a horizontal region and a plane defines a zone in the 3-D diffusion calculation.

LXYZ(1,1,*,I) Starting region number for x-direction
LXYZ(2,1,*,I) Ending region number for x-direction
LXYZ(1,2,*,I) Starting region number for y-direction
LXYZ(2,2,*,I) Ending region number for y-direction
LXYZ(1,3,*,I) Node number of the element on the first plane
LXYZ(2,3,*,I) conversion factor for fuel volume in the core
IFSAME(I) Indicator for symmetric condition. If =1, the element is located at the symmetric position of the previous element and excluded from the actual calculation. The result of the previous one is duplicated. If 90 rotational symmetry is considered, the three succeeding elements will have the non-zero value for this item.
NOCCOR Number of control elements in the core
IDC(A) Identification of a control element

Following four items specify the location of a control element on a plane in term of "region" as used to specify the position of control element. The degree of insertion is specified by LCZ; the succeeding item.

LCXY(1,1,*,I) Starting region number for x-direction
LCXY(2,1,*,I) Ending region number for x-direction
LCXY(1,2,*,I) Starting region number for y-direction
LCXY(2,2,*,I) Ending region number for y-direction
CLOCZ Control rod insertion depth. This item is not used in the calculation, but as comment.
LCZ Node number on the first plane. If the control element

is fully inserted, enter LCZ=1.

Record 5. Control for individual element

NFE	Total number of fuel elements
NFT	Number of types of fuel element
NOT	Number of types of non-fuel element
NRKFMX	Maximum node number through whole fuel elements
NRK1MX	Maximum node number within active fuel part through whole fuel elements
NRKOMX	Maximum node number through whole non-fuel elements

Record 6. Specification of fuel element type

FTCOM(A)	Name of a type of fuel element
NREGKF	Number of nodes
NREGK1	Number of nodes within active fuel part
VOLF	Volume of each node
ZINV	Initial inventory of heavy nuclides for each node in active fuel part
MATSPC	Material number for each node. Material is numbered in the order appearing in the array MTNAME in Record 2

Record 7 Specification of non-fuel element type

OTCOM(A)	Name of a type of non-fuel element
NREGKF	Number of nodes
MATSPO	Material number for each node. Material is numbered in the order appearing in the array MTNAME in Record 2

Record 8 Information of a fuel element

IDENT(A)	Name of a fuel element
NFTYPE	Fuel element type number which is the order appearing in Record 6.
BURNUP(1)	Average burn-up in MWD/element
BURNUP(2)	Average burn-up in ^{235}U fraction
BURNUZ(1,K)	Axial burn-up distribution in MWD/node
BURNUZ(2,K)	Axial burn-up distribution in ^{235} U fraction/node
ZINV(J,K)	Estimated current nuclide concentration of nuclide J in the node K (* 10^{24} atom/cm^3)
STATUS(A)	Status of fuel element = NEW fresh element = CORE loaded in the core = COOL in cooling = WAST waste (not used)
IBCORR	Status of burn-up correction =0 no correction made =1 corrected except nuclide concentration =2 fully corrected
IDATEF(1)	Date of initialization of record (YYMMDD)
IDATEF(2)	Date of obsolete of record (YYMMDD)

Following items specify the loading location of the fuel element in the core at the latest irradiation;

NLOAD	Number of load data in this fuel element
LXYZF(1,1,*)	Starting region number for x-direction
LXYFZ(2,1,*)	Ending region number for x-direction

LXYFZ(1,2,*)	Starting region number for y-direction
LXYFZ(2,2,*)	Ending region number for y-direction
LXYFZ(1,3,*)	Nnode number of the element on the first plane
LXYFZ(2,3,*)	Conversion factor for fuel volume in the core
NHIS	Number of blocks for the succeeding history records

Record 9. Burn-up history

NOB1	Sequential batch number
NOC1	Cycle number
NSBST1	Number of burn-up steps in the batch
IDATEH(1)	Start-up date (YYMMDD)
IDATEH(2)	Shut-down date (YYMMDD)
PERIO1	Operation period for each burn-up
POWZ	Power for each burn-up step by node
ZFTEMP	Average fuel temperature for each burn-up step by node
ZMTEMP	Average moderator temperature for each burn-up step by node
ACBURN	Integrated burn-up (MWD/element)
ZABURN	Axial distribution of integrated burn-up (MWD/node)
NHLOAD	number of loading data in each operation
LXYZ1(1,1,*)	Starting region number for x-direction
LXYZ1(2,1,*)	Ending region number for x-direction
LXYZ1(1,2,*)	Starting region number for y-direction
LXYZ1(2,2,*)	Ending region number for y-direction
LXYZ1(1,3,*)	Node number of the element on the first plane
LXYZ1(2,3,*)	Conversion factor for fuel volume in the core

III.1.6 Input specification

Followings are the specifications of the input read into FT05. Number of entries required is enclosed by slashes for each block.

BLOCK 1 /72H,72H/

OCOM	Comments for operation

BLOCK 2 Control for reactor operation /0,14,2/

NOB	Batch number in the cycle
NOC	Cycle number
NOFCOR	Number of fuel elements loaded in the core
NOCCOR	Number of control elements loaded in the core
NSBSTP	Number of burn-up steps
IDATE(1)	Start-up date (YYMMDD)
IDATE(2)	Shut-down date (YYMMDD)
IRESTO	Restart option =0 No restart >0 Restart problem. Information stored in FT13 and FT98 will be read in.
IRES1	Preparation for the next restart =0 No information will be written >0 Information for restart will be written into FT13 and FT98
ITCAL	Option for fuel and moderator temperature =0 Constant in time =N Number of pairs of coefficients A,B used in the following polynomials for the analytical expression of space dependent temperature (not yet available)

$$\text{Temp}=A+B\int_0^h P dz$$

ILCAL	Option for the eigenvalue calculation after the interpolation process of the last burn-up step of the batch =0 No execution =1 Execution. Dummy entry for burn-up step will be required.
ICFLOW	Direction of coolant flow which is used to define the upper and lower limit of the above integration =1 left to right =2 right to left =3 top to bottom =4 bottom to top =5 front to back =6 back to front
ID1	Option to select diffusion coefficients in the SRAC macroscopic formulation and delayed neutron data written on macroscopic file =|1| Use D1 for all material =|2| Use D2 for all material =|3| Select D1 or D2 by material < 0 Delayed neutron data is written on macro file for kinetic parameter calculation
IXKI	Option to select fission spectrum

=0 Use the unique spectrum which is taken from the first material positioned in the sequential file of DDNAME FT31
>0 Use material dependent spectra

AVFTMP Average fuel temperature (K)
AVMTMP Average moderator temperature (K)

BLOCK 2-1 Required if ITCAL>0, /2*ITCAL/
Coefficient of polynomials for temperature

COEF(1,1) first coefficient A
COEF(2,1) first coefficient B
:
:
COEF(1,ITCAL) last coefficient A
COEF(2,ITCAL) last coefficient B

BLOCK 2-2 Required if |ID1|=3 /NMAT/

IXYZ(I),I=1,NMAT
Selection of diffusion coefficient for three directions in 3-D calculation by material is specified by IXYZ as following:

IXYZ	vertical Dx	horizontal Dy	axial Dz
=1	D1	D1	D1
=2	D2	D1	D1
=3	D1	D2	D1
=4	D2	D2	D1
=5	D1	D1	D2
=6	D2	D1	D2
=7	D1	D2	D2
=8	D2	D2	D2

Note: NMAT : Number of materials which is kept in the history file

BLOCK 3 Operation period /NSBSTP+ILCAL/

PERIO Period of each burn-up step (hour)
If ILCAL=1, an additional dummy entry is required.

BLOCK 4 Average reactor power /NSBSTP+ILCAL/

AVRPOW Average reactor power of each burn-up step (MW)

Note: If any symmetric core geometry is considered, enter the power of the volume under calculation. For example, a half of core is solved, enter half of reactor power: and in 2-D calculation, enter the power divided by the effective core height (cm).

BLOCK 5 Fuel loading. Repeat NOFCOR times.

BLOCK 5-1 A fuel element /A8,8,0/

ID Identification of the fuel element named in the history file (see III.2.2 , Block 4-3)

Following five items specify the locations of fuel elements in the core in terms of "region". Remind that a combination of a vertical region, a horizontal region and a plane defines a zone in the 3-D diffusion calculation. The Z position of the element is expressed in a different way. Normally the first node of an element is located on the first plane which denotes the first mesh in the Z direction of the 3-D core (LZ(1)=1). If a control element is partially withdrawn from the core, a certain node number greater than one has to be given to LZ(1). Of course, LZ(1)=0 or =-1 is accepted so as to indicate the reverse movement. Remind that any movement can be expressed by an integer as the node is numbered to an interval of uniform mesh. The axial length of any movable element is not necessarily longer than the core height. The vacant space after the movement may be filled by some non-depleting material like moderator as occurs in the actual reactor. In 2-D calculation, LZ(1)=1 should be entered.

LX(1) Starting region number for x-direction
If more loading position data is required , enter negative value and feed BLOCK 5-1-1 after this record

LX(2) Ending region number for x-direction

LY(1) Starting region number for y-direction

LY(2) Ending region number for y-direction

LZ(1) Node number of the element on the first plane

ITEMP Temperature option
=0 Specify by BLOCK 5-2 and 5-3.
=-1 Use the average value entered in BLOCK 2.
=N Calculate by using N-th coefficients within ITCAL pairs.
=100+N
Use the input data for the initial step, and calculate by N-th coefficients

IFACT Conversion factor for the fuel volume in the core
=0 or =1 same
=N N times of the volume under calculation

Note. When the center of an element coincides with the center of symmetry, the actual volume of the element must be N times of the volume under consideration.

IFSAME Indicator for symmetry
=0 Independent fuel element
=1 Dependent fuel element which is located in the symmetric position to the previously specified fuel element. This element is excluded from the diffusion calculation, but the results for the corresponding element will be duplicated. If 90 rotational symmetry is considered, several set of three succeeding elements may have IFSAME=1.

BLOCK 5-1-1 Additional loading position data /4/

LX(1) Starting region number for x-direction
If more loading position data is required , enter negative value and feed BLOCK 5-1-1 after this record

LX(2) Ending region number for x-direction

LY(1) Starting region number for y-direction

LY(2) Ending region number for y-direction

BLOCK 5-2	Required if ITEMP=0 or >100	/NSBSTP1*NREGKB/
ZFTEMP	Fuel temperature (K) for each burn-up step averaged at each core height	

Note: NREGKB=1 in 2-D calculation.
NSBSTP1=NSBSTP+ILCAL if ITEPM=0. NSBSTP1=1 if ITEMP > 100.

BLOCK 5-3.	Required if ITEMP=0 or >100	/NSBSTP1*NREGKB/
ZMTEMP	Moderator temperature (K) for each burn-up step averaged at each core height.	

BLOCK 6 — Loading of control element. Repeat NOCCOR times. If NOCCOR=0, no entry is required for this block. A control element does not contain depleting material. If a control element with follower fuel is treated as a fuel element, it has to be specified in BLOCK 5. Such an element can be treated as two elements; the fuel part (follower fuel) is specified in BLOCK 5, and the absorber part is specified in this BLOCK.

BLOCK 6-1 /A8,4X,A8,4,NSBSTP+ILCAL/

IDC	Identification of a control element
MATNOC	Type name of the control element. Type name is one of the array OTCOM in Record 7 of the history file.
LCX(1)	Starting region number for x-direction. If more loading position data is required, enter negative value and feed BLOCK 6-1-1 after this record
LCX(2)	Ending region number for x-direction
LCY(1)	Starting region number for y-direction
LCY(2)	Ending region number for y-direction
CLOCZ(I)	Position(cm) for each burn-up I=1,NSBSTP+ILCAL. This item is not considered in the calculation, but stored in the history file like comment.

BLOCK 6-1-1	additional loading position data	/4/
LCX(1)	Starting region number for x-direction. If more loading position data is required, enter negative value and feed BLOCK 6-1-1 after this record.	
LCX(2)	Ending region number for x-direction	
LCY(1)	Starting region number for y-direction	
LCY(2)	Ending region number for y-direction	
BLOCK 6-2.	Required only in 3-D calculation	/NSBSTP+ILCAL/
LCZ	Node number of the element on the first plane C.f. LZ(1) in BLOCK 5-1.	

Note: If the control rod position should be recorded in the history file even in 2-D calculation, enter blank value of MATNOC, then the control element will be excluded from the diffusion calculation.

BLOCK 7 — Loading of non-fuel element and material

BLOCK 7-1 Material specification /A8/

MATNO Type name of the non-fuel element. Type name is one of the array OTCOM in Record 7 of the history file.
Automatic allocation of the first material to all zones in the reactor is taken. BLOCK 7-2 data for the first material is not needed. This allocation may be overlayed by the succeeding material specifications.

BLOCK 7-2 Loading position /5/

I1 Starting number of position by x
I2 Ending number of position by x
I3 Starting number of position by y
I4 Ending number of position by y
I5 Node number on the first plane
C.f. LCZ in BLOCK 6-2.

Repeat BLOCK 7-2 until I1=0 is encountered.
Repeat BLOCK 7 until MATNO=blank is encountered.

Note: This loading data are first processed and secondary processed fuel element loading data and finally processed control rod loading data.

BLOCK 8 General control of CITATION routine

CARD 1: 001

CARD 2: Control Options (24I3)

NGC1 =0
NGC2 Restart option. Set by the code.
NGC3 Option to write data on logical device 13 to permit restart. Set by the code.
NGC4 =0
NGC5 =0
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, and also the heat-to-coolant map if >1 provided each is edited.
NGC8 Option to write point neutron source (space-energy) on logical device 17 (see GLIM5 on card 5 of 001)
NGC9 =0
NGC10 =0
NGC11 =0
NGC12 Adjoint indicator , enter =1 if an adjoint calculation is required. In this case enter negative value on ID1 in BLOCK 2.
NGC13 =0
NGC14 =0
NGC15 =0
NGC16 =0
NGC17 =0
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)

NGC20 =0

NGC21 =0

NGC22 =0

NGC23 =0

NGC24 =0

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 =0

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 =0

IEDG8 =0

IEDG9 Print zone average flux values by group (IEDG6=0)

IEDG10 Print point flux values by group

IEDG11 =0

IEDG12 Print zone average power densities. Set by the code.

IEDG13 Print relative power density traverses through peak

IEDG14 Print point power densities. If NGC7>0, enter 1.

IEDG15 Print point cumulative heat deposited in coolant

IEDG16 Print point neutron densities summed over energy

IEDG17 =0

IEDG18 =0

IEDG19 =0

IEDG20 =0

IEDG21 =0

IEDG22 =0

IEDG23 =0

IEDG24 Print zone number by each mesh points (print if=0, no print if=1)

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 =0
ITMX7 =0
ITMX8 =0
ITMX9 =0
ITMX10 =0
ITMX11 =0
ITMX12 =0
ITMX13 =0
ITMX14 =0
ITMX15 =0
ITMX16 =0
ITMX17 =0
ITMX18 =0
ITMX19 Machine time limit (60 min). If restart file is needed, set this item less than that implied in JCL card.

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)

Note: It is recommended to enter GLIM1=2.0 and GLIM2=0.001 for the case in which the Keff value swings over the range defined by the defaulted values during the iteration.

BLOCK 9 Description of the neutron flux problem

CARD 1: 003

CARD 2: General description (24I3)

NUAC1 =0
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. Set by the information in the history file.

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 -

EPSI5 -

EPSI6 -

CARD 4: Miscellaneous data (6E12.5)

XMIS1 External extrapolated boundary constant ($-D/\varphi * d\varphi/dx$)

> 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 set by the code.

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>

BLOCK 10 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 4. In this case the KMAX value in Section 008 must be proper

If IND = 3, specify two zone numbers on CARD 3 followed by the group depending buckling on CARD 4 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.

BLOCK 11 Termination of input

CARD 1: 999

III.1.7 File requirement

The COREBN code requires the following files;

DD name	Remarks	Record format
FT01F001	Scratch unit, always required	VBS
FT02F001	Scratch unit, always required	VBS
FT03F001	Scratch unit, always required	VBS
FT05F001	System input	
FT06F001	System print message only	FBA
FT09F001	Used to store forward neutron flux map by option NGC6 in BLOCK 8.	VBS
FT10F001	Scratch unit, always required	VBS
FT11F001	Scratch unit, always required	VBS
FT13F001	Input/output unit for restart required if IRESTO> or IREST>1	VBS
FT14F001	Scratch unit to store macroscopic cross sections, always required	VBS VBS
FT15F001	Scratch unit to store equation constants, always required. This is the unit to which use of the high speed I/O unit is effective to reduce I/O count	VBS
FT16F001	Scratch unit, always required	VBS
FT19F001	Scratch unit, always required	VBS
FT26F001	Scratch unit, if IRESO>0	VBS
FT31F001	Scratch unit to store macroscopic cross section in BCD format, always required	FB 80 3200
FT32F001	Power density & heat-to-coolant, if required	VBS
FT89F001	Scratch unit, always required	VBS
FT90F001	Scratch unit for interpolation of macroscopic cross sections	VBS
FT91F001	Scratch unit, always required	VBS
FT92F001	History file, old	VBS
FT93F001	History file, new	VBS
FT94F001	Scratch unit to store the input to CITATION	FB 80 3200
FT95F001	Scratch unit, always required	FB 80 3200

FT96F001	Scratch unit, always required	VBS
FT97F001	Scratch unit, always required	VBS
FT98F001	I/O unit for restart, required if IRES0>0 or IRES1>0	VBS

III.1.8 Usages and restrictions

(1) The burn-up chain of Garrison model from ENDF/B-II is generaly model applicable to this code. But other model may be applicable if history file is generated so.
(2) Even if the fuel temperature and the moderator temperature is same, the tabulation of macroscopic cross sections on four pairs of temperatures made by possible combination of T_1 and T_2 is required.
(3) During successive burn-up steps, movement of material is not allowed. It must be specified by separate execution.
(4) The option to calculate temperature is not yet available.
(5) The factor to estimate the power from the fission rate is fixed to the ^{235}U value of $3.108*10^{-11}$ watt/fission.
(6) The number of burn-up steps in the cell burn-up calculation has to be common over whole types of fuel element because their degree of burn-up is referred by the burn-up index of integer.
(7) A cell which has more than one depleting materials is not accepted by the code.
(8) The restart files on FT13 and FT98 are commonly used in I/O. Back-up of the input file may be done by IEBGENER.

III.1.9 Error messages

The error messages, their contents and treatments are described below.

(1) E level "LACK OF WORKING AREA SIZE ..... 9999 XXXXX"
Content : Area size declared in MAIN routine causes lack of 9999 words in XXXX step.
Treat : Enlarge the work area in MAIN routine with sufficient margin.
Detected in SIZEX

(2) E level "EXCEED NSBSMX (CRBNIN) 999 RESET NSBSMX IN SUB-ROUTINE CRBNO and HISTORY FILE EDIT PROGRAM"
Content : Maximum value of burn-up steps 999 exceeds expected value in CRBNO routine
Treat : Enlarge the dimension and the limit in CRBNO routine and the related PARAMETER in the history editing code.
Detected in CRBNIN

(3) E level "EXCEED NOFCMX (CRBNIN) 999 RESET NOFCMX IN SUB-ROUTINE CRBNO and HISTORY FILE EDIT PROGRAM"
Content : Number of fuel element loaded in the core exceeds the maximum value in CRBNO routine
Treat : Enlarge the dimension and the limit in CRBNO routine and the related PARAMETER in the history editing code.
Detected in CRBNIN

(4) E level "EXCEED NOCCMX (CRBNIN) 999 RESET NOCCMX IN SUB-ROUTINE CRBNO and HISTORY FILE EDIT PROGRAM"
Content : Number of control element loaded in the core exceeds the

maximum value in CRBNO routine
Treat : Enlarge the dimension and the limit in CRBNO routine and the related PARAMETER in the history editing code.
Detected in CRBNIN

(5) E level "EXCEED MAXNO (CRBNI2) 999 RESET MAXNO IN SUB-ROUTINE CRBNO"
Content : Number of non-fuel element loaded in the core exceeds the maximum value in CRBNO routine
Treat : Enlarge the dimension and the limit in CRBNO routine
Detected in CRBNI2

(6) E level "EXCEED MAXII (CRBNI2) 999 RESET MAXII IN SUB-ROUTINE CRBNO"
Content : Number of input cards for non-fuel element loaded in the core exceeds the maximum value in CRBNO routine
Treat : Enlarge the dimension and the limit in CRBNO routine and the related PARAMETER in the history editing code.
Detected in CRBNI2

(7) E level "X-Y LOADING POSITION (FUEL) is out of range (9,9,9,9)
Content : Loading position of a fuel element is out of range or up-side-down of top and end position.
Treat : Verify LX(1),LX(2),LY(1),LY(2) in BLOCK 5-1
Detected in CRBNI2

(8) E level "X-Y LOADING POSITION (CONTROL) is out of range (9,9,9,9)
Content : Loading position of a control element is out of range or up side-down of top and end position.
Treat : Verify LCX(1),LCX(2),.....,LCY(2) in BLOCK 6-1
Detected in CRBNI2

(9) E level "X-Y LOADING POSITION (NON-FUEL) is out of range (9,9,9,9)
Content : Loading position of a non-fuel element is out of range or up-side-down of top and end position.
Treat : Verify I1,I2,I3,I4 in BLOCK 7-2
Detected in CRBNI2

(10) E level "NUMBER OF FUEL IS ZERO IN HISTORY FILE"
Content : No fuel element is registered in the history file.
Treat : Check the history file using the history file edit code, and if not yet, register the fuel elements.
Detected in CRBN1

(11) E level "FUEL TYPE NUMBER IS NOT FOUND IN HISTORY FILE IDENT= XXXX
FUEL TYPE NO.=YY IF FUEL TYPE NO.=-1 FUEL ELEMENT IS NOT REGISTERED IN HISTORY FILE"
Content : Fuel element XXXX of type 99 is not found in the history file. If type no.=-1, this element is not yet registered
Treat : Classify the fuel element in the right type, or register the fuel elements into the history file.
Detected in CRBN1

(12) E level "SYMMETRIC FUEL POSITION USED, BUT FUEL TYPE IS NOT EQUAL TO THE ORIGINAL FUEL FUEL ID=XXXX FUEL TYPE = 99 N.E. TYPE(0)= 88"
Content : Fuel type 99 of fuel element XXXX is not identical with

88 of the original fuel element.
Treat : Check the classification of fuel type on the history file. If no wrong classification, symmetric condition can not utilized.
Detected in CRBN1

(13) W level "SYMMETRIC FUEL POSITION USED, BUT FUEL (XXXX) HAS NOT SAME HISTORY OF ORIGINAL FUEL"
Content : Fuel element XXXX on a symmetric position has not the same history of that on the original position.
Treat : Based on the degree of burn-up on the history file, using the power level of the original file, burn-up and nuclide concentration will be calculated.
Detected in CRBN1

(14) E level "BURN-UP INFORMATION RECORD IS NOT FOUND IN MACRO FILE (FUEL TYPE IDENT = (XXXX))"
Content : No member which has XXXX as cell identification is found.
Treat : Check member list of MACRO file whether if the member XXXXNDEN exists or not.
Detected in CRBN12

(15) E level "BURN-UP STEP IN MACRO FILE NOT EQUAL TO ALL FUEL TYPE BURN-UP INFORMATION RECORD XXXXXX YYY
Content : Burn-up step in cell call calation is not common through whole types of fuel elements
Treat : Check number of burn-up steps, unit and steps in each cell calculation. Rerun after unification.
Detected in CRBN12

(16) E level "BURN-UP STEP EXCEED MAXIMUM BURN UP STEPS FUEL TYPE (XXXX) SET = 888 NEEDED 999"
Content : Number of burn-up steps in cell calculation exceeds the expected value in CRBN0.
Treat : Enlarge the dimension for burn-up steps and NBSTP value in CRBN0 routine.
Detected in CRBN12

(17) E level "BURN-UP STEP IN MACRO FILE NOT EQUAL TO ALL FUEL TYPE BURN-UP INFORMATION RECORD XXXXXXXX 8888 9999
Content : Number of burn-up steps in cell calculation is unmatch for all fuel type
Treat : Recalculation cell burn-up with same burn-up steps for each fuel type
Detected in CRBN12

(18) E level "NO. OF NUCLIDED IN BURN-UP INFORMATION RECORD (MACRO FILE) IS NOT EQUAL TO HISTORY FILE XXXXXXXX 9999 8888"
Content : Number of depleting nuclides in MACRO file is not identical with that in the history file.
Treat : Check whether if the Garrison model is used or not in the cell burn-up calculation.
Detected in CRBN12

(19) E level "NO. OF DEPLETION ZONE 8888 IS ILEAGAL"
Content : Multi depletion zone in cell burn-up calculation.
Treat : re-arrange burn-up information record by user program.

Detected in CRBN12

(20) E level "ZONE NO. IS NULL, X-REGION=XX, Y REGION=YY, Z-REGION=ZZ CHECK INPUT (CRBN22)"
ELEMENT TYPE NO."
Content : A zone defined by (XX,YY,ZZ) is vacant.
Treat : Correct zone mapping
Detected in CRBN22

(21) E level "XXXXXXXX IS NOT FOUNDED IN HISTORY FILE. REGISTERED TABLE IS FOLLOWINGS
Content : Material type name is not registered in record 2 or 6 of the history file.
Treat : Correct Material type name
Detected in CRBN24

(22) E level "MEMBER (XXXXXXXX) IS NOT FOUND IN MACRO FILE"
Content : The macroscopic cross section identified by XXXXXXXX is not found in the MACRO file.
Treat : Rename in MACRO file or create new member by a series of cell burn-up calculation.
Detected in CRBN3 CRBN4 CRBN61

(23) E level "RESET LENGB = 99999 IN SUBROUTINE CRBN"
Content : Lack of memory to keep macroscopic cross section is detected. Required as 99999.
Treat : Enlarge MEMORY in COREBN
Detected in CRBN3

(24) E level "MEMBER (XXXXXXXX) READ BUFFER SIZE OVER REQUIRED = 88888 SET = 99999 CHANGE SUBROUTINE CRBN (LBUFFS = 9999)"
Content : Lack of memory to keep macroscopic cross section is detected. Required as 88888.
Treat : Change SUBROUTINE CRBN (LBUFFS = 9999)
Detected in CRBN4

(25) E level "BURN-UP IS OUTSIDE BURN-UP TABLE. MEMBER (XXXX) BURN-UP (MWD/CC) =99999.99 FUEL=(YYYYYYYY)
Content : Degree of burn-up for fuel element YYYYY is out-of-range of tabulation.
Treat : Rerun cell burn-up calculation to include 99999.99 MWD/CC
Detected in CRBN46 CRBN61

(27) E level "FUEL TEMPERATURE IS OUTSIDE FUEL TEMPERATURE TABLE. MEMBER (XXXXXXXX) FUEL TEMP(K)=99999.99 FUEL (XXXXYYYY)
Content : Temperature of the fuel element XXXXYYYY identified by XXXXXXXX is out-of-range of tabulation.
Treat : Correct the temperature for the fuel XXXXYYYY. Temperature is allowed between 300 K and 1600 K.
Detected in CRBN46

(28) E level "MODERATOR TEMPERATURE IS OUTSIDE FUEL TEMPERATURE TABLE. MEMBER (XXXXXXXX) FUEL TEMP(K)=99999.99 FUEL (XXXXYYYY)
Content : Moderator temperature of the fuel element XXXXYYYY identified by XXXXXXXX is out-of-range of tabulation.
Treat : Correct the temperature for the fuel XXXXYYYY. Temperature is allowed between 300 K and 1600 K.
Detected in CRBN46

```
(29) E level 'BURN-UP  TABLE  LENGTH  OVER  MEMBER  (XXXXXXXX) REQUIRED
             =9999 set 8888 CHANGE SUBROUTINE CRBN0 @@@@NBZONE@@@@
     Content : Lack of memory to  keep  the  tabulation  of  macroscopic
               cross sections encountered in interpolation for
               the fuel element XXXX
     Treat   : Expand the limiting value in CRBN0 routine.
     Detected  in CRBN61
```

III.1.10 JCL and modification of work area

A sample JCL for FACOM-380 is shown below.

```
//JCLG  JOB
// EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER  -------  YOUR CURRENT USER CARD
   T.3 C.1 W.1 P.0 I.3
   OPTP MSGCLASS=X,NOTIFY=JXXXX,PASSWORD=XX
// EXEC LMGO,PNM=HIST2,LM=J1480.COREBN2
//FT11F001 DD DSN=&&PSLIB,UNIT=WK10,DISP=(,PASS),SPACE=(TRK,(10,10))
//USERPDS  DD DSN=JXXXX.MACRO.DATA,DISP=SHR ==== MACRO PDS FILE ====
//SYSIN DD *
  99 0 11 / PDS TO PS
/*
// EXEC LMGO,PNM=CRBN2,LM=J1480.COREBN2
//FT06F001 DD SYSOUT=*
//SYSIN DD DSN=JXXXX.INPUT.DATA,DISP=SHR ==== YOUR INPUT DATA ====
//FT01F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT02F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT03F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT09F001 DD SPACE=(TRK,(30,10)),UNIT=WK10 ==== FLUX FILE ====
//FT10F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT11F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//*FT13F001 DD SPACE=(TRK,(30,10)),UNIT=WK10 ==== RESTART FILE 1 ====
//FT14F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT15F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT16F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT19F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT20F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT26F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT31F001 DD SPACE=(TRK,(10,10)),UNIT=WK10,
//            DCB=(RECFM=FB,LRECL=80,BLKSIZE=3200)
//*FT32F001 DD SPACE=(TRK,(30,10)),UNIT=WK10 ==== POWER FILE ====
//FT89F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT90F001 DD DSN=&&PSLIB,UNIT=WK10,DISP=(OLD,DELETE) ==== MACRO ====
//FT91F001 DD SPACE=(TRK,(2,1)),UNIT=WK10,
//            DCB=(RECFM=FB,LRECL=80,BLKSIZE=3200)
//FT92F001 DD DSN=JXXXX.HISTI.DATA,DISP=SHR ==== HISTORY OLD ====
//FT93F001 DD DSN=JXXXX.HISTE.DATA,DISP=SHR ==== HISTORY NEW ====
//FT94F001 DD SPACE=(TRK,(2,1)),UNIT=WK10,
//            DCB=(RECFM=FB,LRECL=80,BLKSIZE=3200)
//FT95F001 DD SPACE=(TRK,(2,1)),UNIT=WK10,
//            DCB=(RECFM=FB,LRECL=80,BLKSIZE=3200)
//FT96F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//FT97F001 DD SPACE=(TRK,(30,10)),UNIT=WK10
//*FT98F001 DD SPACE=(TRK,(30,10)),UNIT=WK10 ==== RESTART FILE-2 ====
//FT99F001 DD SYSOUT=*,DCB=(RECFM=FBA,LRECL=137,BLKSIZE=13700)
```

```
++
//
```

The code is principally organized to run on variable dimensioning. The current version uses 60,000 words as the work area. The following JCL permits the use of larger area, if required. As shown in the list of error messages, certain variable arrays have fixed sizes which are defined in the CRBN0 routine. If any restriction is changed, the corresponding restriction in the history file edit code have to be changed.

```
//JCLG  JOB
// EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER  ------- YUR CURRENT USER CARD
   T.5 C.3 W.1 P.0 I.5
   OPTP MSGCLASS=X,NOTIFY=JXXXX,PASSWORD=XX
//  EXEC FORT77
C
C    CORE BURN-UP MAIN ROUTINE
C
C     FOR THE EXECUTION OF CORE-BURN WITH ENLARGED CORE STORAGE,
C     CHANGE THE ARRAY LENGTH OF BLANK COMMON /      /
C     FROM 60,000 WORDS TO DESIRED VALUE,
C     AND SET THE VARIABLE 'MEMORY' TO THIS VALUE
C
>      COMMON /      / ARAY(600000)
C
>      MEMORY = 600000
       CALL CRBN      (ARAY,MEMORY,1)
       STOP
       END
// EXEC LKEDIT,CNTL=NO,MODS='200,10,1',WORKS='200,10',
//             A='OVLY',LM=J1480.COREBN2
//SYSIN DD DSN=J1480.COREBN2.OVERLAY,DISP=SHR
// EXEC LMGO,PNM=HIST2,LM=J1480.COREBN2
//FT11F001 DD DSN=&&PSLIB,UNIT=WK10,DISP=(,PASS),SPACE=(TRK,(10,10))
//USERPDS  DD DSN=JXXXX.MACRO.DATA,DISP=SHR ==== MACRO PDS FILE ====
//SYSIN DD *
  99 0 11 / PDS TO PS
/*
// EXEC GO,PNM=CRBN2
//*
//* DD STATEMENTS FOLLOWS
//*
++
//
```

III.1.11 Sample input data

A sample input data is shown below.

```
JMTR CORE BURN-UP CALCULATION  12 CYCLE
     CONTROL ROD FULL UP 2-D CALCULATION ( 4 STEPS RUN )      / BLOCK 1
1  12 27  0  4 830701 830818  0  0  0  1  3  1  0 325. 325.  / BLOCK 2
12.0 84.0 2(96.0) 0.0                                        / BLOCK 3
5(.3333333)                                                  / BLOCK 4
FUELA-01   4  4  5  6  1  -1  1  0                           / BLOCK 5
```

```
FUELA-05   4  4  5  6  1  -1  1  1 / SYMMETRIC FUEL          / BLOCK 5
-----------------------------------
--- FUEL ELEMENT LODING DATA ---
-----------------------------------
AREFA000                        / ALL ZONE FILLED AL REFLECTOR / BLOCK 7-1
BEREFA00                         / BE REFLECTOR               / BLOCK 7-1
3  3  3  11  1                                               / BLOCK 7-2
3  8  3   3  1                                               / BLOCK 7-2
3  8 11  11  1                                               / BLOCK 7-2
0  0  0   0  0                  / TERMINATED                 / BLOCK 7-2
-----------------------------------------
--- IRADIATION ELEMENT LOADING DATA ---
-----------------------------------------
                                / BLOCK 7-1 TERMINATED       / BLOCK 7-1
001
  0  0  0  0  0  1  1  0  0  0  0  0  0  0  0  0  0  0  1  0  0  0
  1                                  1  1  1  0  1                    1
200                                                      4
 1.5         0.1
003
              11                          3
 5.0      E-4  1.0     E-4
 0.0           0.0
024
  11.162 E-3
999
```

III.2 History File Edit Code for COREBN

An edit code is available for the history file of the COREBN code described in the previous section. It permits update and edit of the file.

III.2.1 Functions

File update

(a) Initialization and update of core geometory and materials

Register the header, core geometry, materials and their component nuclides and clean-up the required information.
Update core geometory or materials.

(b) Register or update of fuel element type specification

(c) Register or update of non-fuel element type

Register or update of non-fuel element (control element) and materials (reflector)

(d) Register or update of individual fuel element

Register, update or delete the particular fuel element

(e) Obsolete the operation record

Obsolete the old operation record or delete the record for the latest operations for recalculation. Corresponding information in the relevant elements is also deleted.

File edit

(a) Print of core geometry, materials and their component nuclides

(b) Print of operation record.

(c) Print of fuel type specification

(d) Print of non-fuel type specification

(e) Print of individual fuel information

(f) List of operation records

(g) List of fuel elements

(h) Print nuclide concentrations on tabulation
The results of cell burn-up calculation

(i) List of degree of burn-up for fuel elements loaded in the core before and after the operation together with the increase of burn-up.

(j) List of Z-axis structure for fuel and non-fuel element type

(k) Convert from PDS file of MACRO to PS-file for COREBN code

III.2.2 Input specifications

All input read from FT05 may be fed in "free format" described in II.1. The input is organized to feed, feed the selection of the function, then the specification of the function. More than one functions are available in a job.

Block G-1 General form of function selection /3/

IMOD Selection of function
- =0 Terminate of the job
- =1 Initialization or update of core geometory or materials
- =2 Register and update of fuel element type
- =3 Register and update of non-fuel element
- =4 Register and update of individual fuel element
- =5 Delete of operation record
- =6 Print-out of reactor geometry and nuclides
- =7 Print-out of operation records
- =8 Print-out of fuel element type
- =9 Print-out of non-fuel element
- =10 Print-out of individual fuel element
- =11 List of operation records
- =12 List of fuel elements
- =13 Print-out of change of nuclide concentrations during burn-up
- =14 List of degree of burn-up of fuel elements loaded in the reactor
- =15 List of Z-axis structure for fuel and non-fuel elements type
- =99 PDS file of MACRO to PS-file for COREBN code

NHIS1 Logical device number for the old history file :
Enter any two-digit number except 01,05,06

NHIS2 Logical device number for the new history file :
Enter any two-digit number except 01,05,06 or NHIS1.

(1) Initialization or update of core geometory or materials

Block 1-1 Function /3/

IMOD Enter 1

NHIS1 If history file is updated , logical device number for the old history file :
Enter any two-digit number except 01,05,06

NHIS2 Logical device number for the initialized file or new one :
Enter any two-digit number except 01,05,06 or NHIS1.

Block 1-2 Header /A72,A72/

HEADER Any characters in two cards

Block 1-3 Control data /7/

IGEOM Geometry
- =1 to 5 Not available

=6 2D slab (X-Y)
=7 2D cylinder (R-Z)
=8 2D circle (θ-R)
=9 2D hexagonal (H)
=10 2D triangular (T)
=11 3D slab (X-Y-Z)
=12 3D cylinder (θ-R-Z)
=13 3D hexagonal (H-Z)
=14 3D triangular (T-Z)

If IGEOM is negative value , core geometory is updated and Block's 1-4 through 1-9 data are needed for update of core geometory.

NREGI Number of region in X-direction

NREGJ Number of region in Y-direction

NREGKB Number of region in Z-direction; enter 0 except 3D geometry.

NMAT Number of materials used in the core burn-up calculations. Count one for the fuel elements of the same type even if they are in different burn-up steps.
If material information is update , NMAT must be negative value and |NMAT| is number of materials for update. In this case only Block 1-13 is needed.

NTNUC Number of depleting nuclides that is treated in burn-up calculations by SRAC code.
If NTNUC is zero , it is set by 15 and the nuclide names are ' U05',' U06',' U08',' PU9',' PU0',' PU1',' PU2',' XE5', ' SM9',' F5N',' F5S',' F5R',' F9N',' F9S' and ' F9R'. If NTNUC is not zero, Block 1-11 is needed.

NHVNUC Number of heavy nuclides for the inventory calculation in the depleting nuclides.
If NHVNUC is zero, heavy nuclides are automaticaly selected from the depleting nuclides by comparison of heavy nuclide name table in the code. Contents of this table are ' TH2', ' PA3',' U03',' U04',' U05',' U06',' U08',' PU9',' PU0', ' PU1' and 'PU2'.

Block 1-4 Number of mesh division in X-direction /NREGI/

NMESHX Number of intervals by region

Block 1-5 Size in X-direction /NREGI/

XX Widths by region in cm

Block 1-6 Number of mesh division in Y-direction /NREGJ/

NMESHY Number of intervals in a region

Block 1-7 Size in Y-direction /NREGJ/

YY Widths by region in cm

Block 1-8 Number of mesh division in Z-direction , 3D only /NREGKB/

NMESHZ Number of intervals in a region

Block 1-9 Size in Z-direction , 3D only /NREGKB/

ZZ Widths by region in cm

Block 1-10 Material identification and volume NMAT times /A8,1,2/

MATNM Member name on MACRO file
IFORS Option of fuel or other materials
>0 fuel =0 not fuel
VOLFS Volume in cm^3 which is used to convert the degree of burn-up given in MW into MW/cm^3. (The average power density of homogenized fuel region)

Note. Among NMAT materials, non-fuel materials must be fed first. For non-fuel materials, as VOLFS is insignificant, set 1.0. For the internal black absorber used in the CITATION routine, give arbitrary name and enter -1.0 as VOLFS.

VOLFR Volume fraction of fuel meat in a homogenized fuel region which is used to convert Xe concentration in fuel meat into that in an element.

Block 1-11 Depleting nuclide names , if NTNUC > 0 /NTNUC*A4/

NISO Depleting nuclides name treated on SRAC code

Note. This name may be fed in "character free format". "Character free format" permits only repeat of character strings. Blank and comma are deliminator , n(cccc) is that character string 'cccc' is repeated by n times.

Block 1-12 Heavy nuclide names , if NHVNUC > 0 /NHVNUC*A4/

IHVNUC Heavy nuclide name for inventory calculation

Note. This name may be fed in "character free format" and selected from the heavy nuclide table on this code described above.

Block 1-13 Material identification update data , |NMAT| times /2A8,1,2/

MATNMO Old member name on MACRO file
If MATNMO and MATNMN is not BLANK , MATNMO is changed.
If MATNMO is BLANK , MATNMN is added.
IF MATNMN is BLANK , MATNMO is deleted.
MATNMN New member name on MACRO file
IFORS See Block 1-10
VOLFS See Block 1-10
VOLFR See Block 1-10

(2) Register and update of fuel elment type

Block 2-1 Function /3/

IMOD Enter 2

NHIS1 Logical device number for the old history file :
Enter any two-digit number except 01,05,06
NHIS2 Logical device number for the new history file :
Enter any two-digit number except 01,05,06 or NHIS1.

Block 2-2 Option

IOPT Selection of register or update /1/

=1 Register of a new fuel element type
=2 Correction of an old fuel element type
=0 Terminate of Block 2

Following Block 2-3 group is required if IOPT=1 is entered.

Block 2-3-1 Specification of fuel type /A8,2,0/

FTCOM1 Identification of fuel element type

NRKF Number of nodes, set 1 if in 2D.

NRK1 Number of nodes of depleting material, set 1 if in 2D.

Block 2-3-2 Volumes of nodes of depleting material /NRK1/

VOLF1 Volumes of nodes of depleting materials in cm^3 which are used to calculate the inventory of depleting nuclides

Block 2-3-3 Weights by nuclide in a node /NHVNUC*NRK1/

TZINV1(I,K) Weight of nuclide I in gram in the node K
Nuclides are setted in the order by the array given in Block 1-3.

When the nuclide concentration in a fuel element will be registered by Block 4.3.3, all of this entry may be filled by null values.

Block 2-3-4 Material specifications by node /NRKF*A8/

MATSP The material name in the array given in Block 1-10. Enter by "character free format".

Block 2-2 will be read after Block 2-3-4.

Following Block 2-4 group is required if IOPT=2 is entered.

Block 2-4-1 Specification of fuel type /2A8,2,0/

FTCOMN New identification of fuel element type which you want to change. Set blank if no change on ID is required.

FTCOMO Identification of fuel element type in the old file which you want to change.

NRKF Number of nodes, set 0 if no change.

NRK1 Number of nodes of depleting material, set 0 if no change.

Block 2-4-2 Volumes of nodes of depleting material /NRK1/

VOLF1 Volumes of nodes of depleting materials in cm^3. Give non-zero value only for the node on which the correction is required.

Block 2-4-3 Weights by nuclide in a node /NHVNUC*NRK1/

TZINV1(I,K) Weight of nuclide I in gram in the node K. Give non-zero value only for the node on which the correction is required.

Block 2-4-4 Material specifications by node /NRKF*A8/

MATSP The material name in the array given in Block 1-10.
Enter by "character free format".
Give blank if no change is required in the node.

Block 2-2 will be read after Block 2-4-4.

Note. If volume of nodes of depleting materials are changed, related value (ex. Weight of nuclide in node , Degree of burn-up and nuclides density by node of the fuel element of this type.) are automatically corrected by volume average calculation.

(3) Register and update of non-fuel element type

Block 3-1 Function /3/

IMOD Enter 3
NHIS1 Logical device number for the old history file :
Enter any two-digit number except 01,05,06
NHIS2 Logical device number for the new history file :
Enter any two-digit number except 01,05,06 or NHIS1.

Block 3-2 Option /1/

IOPT Selection of register or update

=1 Register of a new non-fuel element
=2 Correction of an old non-fuel element
=0 Terminate of Block 3

Following Block 3-3 group is required if IOPT=1 is entered.

Block 3-3-1 Specification of non-fuel element /A8,1,0/

OTCOM1 Identification of non-fuel element type

NRKO Number of nodes, set 1 if in 2D.

Block 3-3-2 Material specifications by node /NRKO*A8/

MATSP The material name in the array given in Block 1-10.
Enter by "character free format".

Block 3-2 will be read after Block 3-3-2.

Following Block 3-4 group is required if IOPT=2 is entered.

Block 3-4-1 Specification of non-fuel element /2A8,1,0/

OTCOMN New identification of element type which you want to change. Set blank if no change is required

OTCOMO identification of element type in the old file which you want to change.

NRKO Number of nodes, set 0 if no change is required

Block 3-4-2 Material specifications by node /NRKO/

MATSP The material name in the array given in Block 1-10. Enter by "character free format". Set blank if no change is required

Block 3-2 will be read after Block 3-4-2.

(4) Register and update of individual fuel element

Block 4-1 Function /3/

IMOD Enter 4

NHIS1 Logical device number for the old history file : Enter any two-digit number except 01,05,06

NHIS2 Logical device number for the new history file : Enter any two-digit number except 01,05,06 or NHIS1.

Block 4-2 Option /1/

IOPT Selection of register or update

=1 Register of a new fuel element
=2 Correction of an old fuel element
=3 Deletion of a fuel element
=0 Terminate of Block 4

Following Block 4-3 group is required if IOPT=1 is entered.

Block 4-3-1 Specification of fuel element /A8,4X,A8,0,2/

IDENT Identification of fuel element

NFTYPE Fuel element type identification specified in Block 2-3

BURNUP(1) Degree of burn-up of the element (MWD/element)

BURNUP(2) Degree of burn-up of the element (fraction of U-235 burnt)

If a blank IDENT is encountered, the process is terminated and the next function card is read.

Block 4-3-2 Degree of burn-up by node (required only for 3-D

calculation) /2*NRK1/

BURNUZ(1,1) Degree of burn-up of the first node in MWD/node
BURNUZ(2,1) Degree of burn-up of the first node in U-235 fraction
BURNUZ(1,2) Degree of burn-up of the second node in MWD/node
BURNUZ(2,2) Degree of burn-up of the second node in U-235 fraction
....
BURNUZ(1,NRK1) Degree of burn-up of the last node in MWD/node
BURNUZ(2,NRK1) Degree of burn-up of the last node in U-235 fraction

Block 4-3-3 Nuclide density by node ($10^{24} cm^{-3}$) /NTNUC*NRK1/

ZINV(1,1) Nuclide density of the 1st nuclide in the 1st node
ZINV(2,1) Nuclide density of the 2nd nuclide in the 1st node
....
ZINV(NTNUC,1) Nuclide density of the last nuclide in the 1st node
....
ZINV(1,NRK1) Nuclide density of the 1st nuclide in the last node
ZINV(2,NRK1) Nuclide density of the 2nd nuclide in the last node
....
ZINV(NTNUC,NRK1) Nuclide density of the last nuclide in the last node

If in 2D geometry, assume NRK1=1. When the specification of the nuclide density has been set by Block 2-3-3, all entries in this item must be filled by null values.

Repeat Block's 4-3-1 throgh 4-3-3 until a blank IDENT is encountered in Block 4-3-1.

Following Block 4-4 group is required if IOPT=2 is entered.

Block 4-4-1 Specification of fuel element to be corrected /A8,4,0/

IDENT Identification of fuel element which you want to change.

ICOR(1) Indicator to correct the element ID
=0 no change
>0 change

ICOR(2) Indicator to correct the element type
=0 no change
>0 change

ICOR(3) Indicator to correct the degree of burn-up
=0 no change
>0 change

ICOR(4) Indicator to correct the nuclide densities
=0 no change
>0 change

Note: The non-zero value for items ICOR(1), ICOR(2), and ICOR(4) is accepted only for fresh fuel element.
Ident must be in registered order.

Block 4-4-2 Element ID correction, required if ICOR(1)>0 /A8/

IDENT1 Fuel element identification

Block 4-4-3 Element type correction, required if ICOR(2)>0 /A8/

NFTYP1 Fuel element type identification

Block 4-4-4 Degree of burn-up, required if ICOR(3)>0 /2/

BURNU1(1) Degree of burn-up in MWD/element
BURNU1(2) Degree of burn-up in U-235 fraction
If null value is entered, no correction will be made.

Block 4-4-5 Degree of burn-up by node, required if ICOR(3)>0 /2*NRK1/

BURNUZ(1,1) Degree of burn-up of the first node in MWD/node
BURNUZ(2,1) Degree of burn-up of the first node in U-235 fraction
BURNUZ(1,2) Degree of burn-up of the second node in MWD/node
BURNUZ(2,2) Degree of burn-up of the second node in U-235 fraction
....
BURNUZ(1,NRK1) Degree of burn-up of the last node in MWD/node
BURNUZ(2,NRK1) Degree of burn-up of the last node in U-235 fraction

Block 4-4-6 Nuclide density by node ($10^{24} cm^{-3}$) /NTNUC*NRK1/

ZINV(1,1) Nuclide density of the 1st nuclide in the 1st node
ZINV(2,1) Nuclide density of the 2nd nuclide in the 1st node
....
ZINV(NTNUC,1) Nuclide density of the last nuclide in the 1st node
....
ZINV(1,NRK1) Nuclide density of the 1st nuclide in the last node
ZINV(2,NRK1) Nuclide density of the 2nd nuclide in the last node
....
ZINV(NTNUC,NRK1) Nuclide density of the last nuclide in the last node

In 2D geometry, assume NRK1=1. If any null value is entered, no correction will be made.

Repeat Block's 4-4-1 through 4-4-6 until a blank IDENT is encountered in Block 4-4-1.

Block 4-5 Deletion of elements /A8,1,0/
(required if IOPT=3 in Block 4-2)

IDENT1 Identification of fuel element

NSEQ Position of this element in the history file. Fuel element position is defined by registered order.

Repeat Block 4-5 until blank IDENT1 and NSEQ=0 are encountered.

(5) Deletion of operation record

Block 5-1 Function /3/

IMOD Enter 5

NHIS1 Logical device number for the old history file :
Enter any two-digit number except 01,05,06

NHIS2 Logical device number for the new history file :
Enter any two-digit number except 01,05,06 or NHIS1.

Block 5-2 Batch numbers /2/

NBACH1 First batch number to be deleted

NBACH2 Last batch number to be deleted

The operation records for the batches starting from NBACH1 to NBACH2 will be deleted from the history file. If the latest batch record is deleted, the degree of burn-up of individual elements is not replaced to the latest of the remaining ones, then the correction is required.

(6) Print-out of reactor geometry and nuclide table

Block 6-1 Function /3/

IMOD Enter 6

NHIS1 Logical device number for the history file :
Enter any two-digit number except 01,05,06

NHIS2 Dummy number

(7) Print-out of operation records

Block 7-1 Function /3/

IMOD Enter 7

NHIS1 Logical device number for the history file :
Enter any two-digit number except 01,05,06

NHIS2 Dummy number

Block 7-2 Selection of items /2/

IOPT Selection of items
=0 termination of Block 7
=1 List of all operation records
=2 List of the record of a cycle
=3 List of the record of a batch

ICBNO Set 0 if IOPT=1
Cycle number if IOPT=2
Batch number if IOPT=3

Repeat Block 7-2 as required.

(8) Print-out of records for fuel element type

Block 8-1 Function /3/

IMOD Enter 8

NHIS1 Logical device number for the history file :
Enter any two-digit number except 01,05,06

NHIS2 Dummy number

(9) Print-out of records for non-fuel element

Block 9-1 Function /3/

IMOD Enter 9

NHIS1 Logical device number for the history file :
Enter any two-digit number except 01,05,06

NHIS2 Dummy number

(10) Print-out of records for individual element

Block 10-1 Function /3/

IMOD Enter 10

NHIS1 Logical device number for the history file :
Enter any two-digit number except 01,05,06

NHIS2 Dummy number

Block 10-2 Selection of element /A8,1,0/

IDENTL Identification of an element to be printed

IOPT Selection of items
=0 Termination of Block 10
=+-1 List of all elements
=+-2 List of the specified element
If a negative value is entered, the content of the fuel history will not be printed.

Repeat Block 10-2 as required.

(11) List of operation records

Block 11-1 Function /3/

IMOD Enter 11

NHIS1 Logical device number for the history file :
Enter any two-digit number except 01,05,06

NHIS2 Dummy number

(12) List of fuel elements

Block 12-1 Function /3/

IMOD Enter 12

NHIS1 Logical device number for the history file :
Enter any two-digit number except 01,05,06

NHIS2 Dummy number

(13) Print and plot file creation of nuclide densities

Block 13-1 Function /3/

IMOD Enter 13

NHIS1 Logical device number for the history file :
Enter any two-digit number except 01,05,06

NHIS2 Logical device number for plotting file.
Enter 0 if plot is not required.

Note. Plotting file is created for last member only and file format is dynamic file of the GPLP code.

Block 13-2 Member name /A4/

MEMBER Heading four characters of the member name in MACRO file of PDS format. since the member storing the nuclide densities has the fixed string 'NDEN' at the tail position. If this function is used , DD name of 'MACRO' is needed in the JCL statement.

Block 13 will be terminated by entering a blank MEMBER.

(14) List of degree of burn-up in the reactor core

Block 14-1 Function /3/

IMOD Enter 14

NHIS1 Logical device number for the history file storing the information before a batch.

NHIS2 Logical device number for the history file storing the information after a batch.

(15) List of Z-axis structure for fuel and non-fuel element type

Block 15-1 Function /3/

IMOD Enter 15

NHIS1 Logical device number for the history file :
Enter any two-digit number except 01,05,06

NHIS2 Dummy number

(99) Convert from PDS file of MACRO to PS-file for COREBN code

Block 99-1 Function /3/

IMOD Enter 99

NHIS1 Dummy number

NHIS2 Logical device number for the PS-file : Enter any two-digit number except 01,05,06 When this function is used , DD name of 'USERPDS' is needed in the JCL statement.

III.2.3 Sample JCL and input of history file edit code

Sample JCL and input data is shown below.

```
//JCLG  JOB
// EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER  ----- YOUR CURRENT USER CARD
   T.1 C.1 W.2 P.0 I.0
 OPTP PASSWORD=XXXX,NOTIFY=J9999
//HIST EXEC LMGO,PNM=HIST2,LM=J1480.COREBN2
//FT01F001 DD SPACE=(TRK,(10,1),RLSE),UNIT=WK10
//FT10F001 DD SPACE=(TRK,(10,1),RLSE),UNIT=WK10
//FT11F001 DD SPACE=(TRK,(10,1),RLSE),UNIT=WK10
//FT12F001 DD SPACE=(TRK,(10,1),RLSE),UNIT=WK10
//FT13F001 DD DSN=J9999.HIST2D.DATA,DISP=OLD
//SYSIN  DD *
1  0 10                                                    / BLOCK 1- 1
JMTR CORE BURN-UP CALCULATION                              / BLOCK 1- 2
2-DIMENSIONAL CALCULATION
6  8 13  0  17  0  0                                      / BLOCK 1- 3
2  4  8 2(4) 3(2)                                         / BLOCK 1- 4
3(15.44) 2(7.72) 3(3.86)                                   / BLOCK 1- 5
2  4  8  4 2(2)  4  2(2)  4  8  4  2                       / BLOCK 1- 6
3(15.44) 7.72 2(3.86) 7.72 2(3.86) 7.72 3(15.44)           / BLOCK 1- 7
IRA1A000  0 1.0 1.0                                        / BLOCK 1-10
IRA2A000  0 1.0 1.0
--------------------------
--- material name data ---
--------------------------
S9XXAX00  1 110476.6131 0.09484315
2 10 11                                                    / BLOCK 2-1
                                                           / BLOCK 2-2
1                                                          / BLOCK 2-3-1
STFU279G  1  1                                             / BLOCK 2-3-2
425.355                                                    / BLOCK 2-3-3
7(0.0)                                                     / BLOCK 2-3-4
S9XXAX00
----------------------------
--- OTHER FUEL TYPE DATA ---
----------------------------
0            / TERMINATED                                  / BLOCK 2-2
3 11 12                                                    / BLOCK 3-1
                                                           / BLOCK 3-2
1                                                          / BLOCK 3-3-1
B-REFEA0  1                                                / BLOCK 3-3-2
BREFA000
-----------------------------------------------------
--- IRADIATION AND REFLECTOR CONTROL ELEMENT DATA ---
-----------------------------------------------------
0            / TERMINATED                                  / BLOCK 3-2
4 12 13                                                    / BLOCK 4-1
                                                           / BLOCK 4-2
1                                                          / BLOCK 4-3-1
FUELA-01    STFU279G 2(0.0)                                / BLOCK 4-3-3
1.6811E-3 .0 1.2495E-4 12(0.0)
```

```
-----------------------------------
--- INDIVIDUAL FUEL ELEMENT DATA ---
-----------------------------------
                          2(0.0) / TERMINATED          / BLOCK 4-3-1
 6 13  0    / GEOMETRY LISTING                        / BLOCK 6-1
 8 13  0    / FUEL STRUCTURE LISTING                  / BLOCK 8-1
 9 13  0    / COMPONENT STRUCTURE LISTING             / BLOCK 9-1
10 13  0    / FUEL ELEMENT LISTING                    / BLOCK 10-1
          1 / ALL FUEL LIST                           / BLOCK 11-1
          0 / TERMINATED
 0  0  0    / JOB TERMINATED                          / BLOCK G-1
/*
//
```

IV User Information

IV.1 Core Storage and Machine Time Requirement

The SRAC is designed to work normally within 2000 KB core (C2 rank). The standard load module with 240 KB (60,000 words) work area requires 500 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 00931480,KE.TUTIHASI,0434.100
     T.3 C.2 W.0 I.5
 OPTPMSGCLASS=R,PASSWORD=??
//    EXEC FORT77,A=NOSOURCE
//SYSIN DD *
C*****EXECITION OF LARGE CASE NEEDS MODIFICATION OF DIMENSIONS *****
C*****SPECIFIED IN THE FOLLOWING MAIN PROGRAM **********************
      COMMON /MAINC/ IOPT(95),MEMORY
      COMMON /WORK/ A(60000)
      COMMON /TW1C/ CC(1),LIM1,IAA(4000)
      COMMON /SN1C/ BB(1),LIM2,IBB(1000)
      COMMON /CIT1C/ DD(8),LIM3,IDD(4000)
      LIM1=4000
      LIM2=1000
      LIM3=4000
      CALL DTLIST
      CALL ERRSET(202,256,10,2,1)
      CALL ICLEA(IOPT,2000,0)
      MEMORY =60000
      CALL SRAC
      STOP
      END
/*
//     EXEC LKED77,A=OVLY,B=NOLIST,GPLIB=PNL,CNTL=NO
//LINK.SYSLMOD DD DSN=&&LM,SPACE=(TRK,(90,10,1),RLSE),
//      DCB=(BLKSIZE=19069,RECFM=U),UNIT=WK10,DISP=(NEW,PASS,DELETE)
//LINK.OLDLM DD DSN=J1480.SRACLM.LOAD,DISP=SHR,LABEL=(,,,IN)
//LINK.SYSIN DD DSN=J1480.OVERLAYH.DATA(SRAC5),DISP=SHR,LABEL=(,,,IN)
//SRAC EXEC GO,PNM=SRAC5
//*****************************************************************
//*                                                               *
//*       DD STATEMENTS for GO STEP 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 3 cpu minutes by FACOM M-380 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 4,000 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 20 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-380 computer, where the DD statements listed are available for all possible combinations of installed routines. Several 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 00931480,KE.TUTIHASI,0434.100
    T.3 C.2 W.0 I.3 GRP
 OPTP MSGCLASS=R,PASSWORD=TTTTT
//SRAC  EXEC LMGO,LM='J1480.SRAC5',PNM=SRAC5,A='HIO=(15)'
// EXPAND GRNLP,SYSOUT=M
//FT01F001  DD DSN=&&WRK01,SPACE=(TRK,(30,10)),UNIT=WK10,
//          DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT02F001  DD DSN=&&WRK02,SPACE=(TRK,(30,10)),UNIT=WK10,
//          DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
```

```
//FT03F001  DD DSN=&&WRK03,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT04F001  DD DSN=&&WRK04,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//*FT07F001 DD DSN=J1480.BURNRSRT.DATA,SPACE=(TRK,(5,2),RLSE),
//*          UNIT=TSSWK,DISP=(NEW,CATLG,DELETE),
//*          DCB=(RECFM=FB,BLKSIZE=3200,LRECL=80)
//FT08F001  DD DSN=&&WRK08,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT09F001  DD DSN=&&WRK09,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT10F001  DD DSN=&&WRK10,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT11F001  DD DSN=&&WRK11,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT12F001  DD DSN=&&WRK12,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT14F001  DD DSN=&&WRK14,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT15F001  DD SPACE=(CYL,(15,3)),UNIT=WK10,DISP=(NEW,DELETE)
//FT16F001  DD DSN=&&WRK16,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT18F001  DD DSN=&&WRK18,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT19F001  DD DSN=&&WRK19,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT21F001  DD DSN=&&WRK21,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT22F001  DD DSN=&&WRK22,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT26F001  DD DSN=&&WRK26,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT28F001  DD DSN=&&WRK28,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT31F001  DD DSN=&&WRK31,SPACE=(TRK,(5,2)),UNIT=WK10,
//           DISP=(NEW,PASS),DCB=(RECFM=FB,BLKSIZE=6400,LRECL=80)
//FT32F001  DD DSN=&&WRK32,SPACE=(TRK,(20,5)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT33F001  DD DSN=&&WRK33,SPACE=(TRK,(20,5)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT50F001  DD DSN=J1480.BURN2.DATA(ENDFB2),DISP=SHR,LABEL=(,,,IN)
//FT51F001  DD DSN=&&WRK51,SPACE=(TRK,(5,5)),UNIT=WK10,
//           DCB=(RECFM=VS,BLKSIZE=3200,BUFNO=2)
//FT52F001  DD DSN=&&WRK52,SPACE=(TRK,(1,1)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=8164,LRECL=8160,BUFNO=2)
//FT81F001  DD DSN=&&WRK81,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=16324,LRECL=16320,BUFNO=2)
//FT82F001  DD DSN=&&WRK82,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=16324,LRECL=16320,BUFNO=2)
//FT83F001  DD DSN=&&WRK83,SPACE=(TRK,(30,10)),UNIT=WK10,
//           DCB=(RECFM=VBS,BLKSIZE=16324,LRECL=16320,BUFNO=2)
//FT91F001  DD DSN=&&WRK91,SPACE=(TRK,(5,2)),UNIT=WK10,
//           DCB=(RECFM=FB,BLKSIZE=6400,LRECL=80)
//FT92F001  DD DSN=&&WRK92,SPACE=(TRK,(5,2)),UNIT=WK10,
//           DCB=(RECFM=FB,BLKSIZE=6400,LRECL=80)
//FT99F001  DD SYSOUT=R,DCB=(RECFM=FBA,LRECL=137,BLKSIZE=19043)
//FASTP     DD DSN=J1480.FASTLBB4.DATA,DISP=SHR,LABEL=(,,,IN)
//*         DD DSN=J1480.FASTLBB5.DATA,DISP=SHR,LABEL=(,,,IN)
```

```
//*          DD DSN=J1480.FASTLBJ2.DATA,DISP=SHR,LABEL=(,,,IN)
//THERMALP   DD DSN=J1480.THERMLB4.DATA,DISP=SHR,LABEL=(,,,IN)
//*          DD DSN=J1480.THERMLB5.DATA,DISP=SHR,LABEL=(,,,IN)
//*          DD DSN=J1480.THERMLJ2.DATA,DISP=SHR,LABEL=(,,,IN)
//MCROSS     DD DSN=J1480.MCROSS4.DATA,DISP=SHR,LABEL=(,,,IN)
//*          DD DSN=J1480.MCROSB5.DATA,DISP=SHR,LABEL=(,,,IN)
//*          DD DSN=J1480.MCROSSJ2.DATA,DISP=SHR,LABEL=(,,,IN)
//FASTU      DD DSN=&&WRKFTU,SPACE=(TRK,(50,5,50)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(,PASS)
//THERMALU   DD DSN=&&WRKTHU,SPACE=(TRK,(50,5,50)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(,PASS)
//MACROWRK   DD DSN=&&WRKMCR,SPACE=(TRK,(100,5,20)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(,PASS)
//MACRO      DD DSN=&&WRKMCO,SPACE=(TRK,(50,5,10)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(NEW,PASS)
//MICREF     DD DSN=&&WRKMIC,SPACE=(TRK,(50,5,20)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(,PASS)
//FLUX       DD DSN=&&WRKFLX,SPACE=(TRK,(50,5,50)),UNIT=WK10,
//          DCB=(RECFM=U,BLKSIZE=19069),DISP=(,PASS)
//SYSIN      DD DSN=J1480.SRACINPT.DATA(POWER),DISP=SHR,LABEL=(,,,IN)
++
//
```

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

REACT.IN : Read in the input for reaction rate calculation

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 : Calc. self shielding factor by table look-up on NRA

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 of fertile material by table-look-up on IRA

MCROSS : Compose the user's resonance neutron cross section file

PEACO : Calculate ultra-fine neutron spectrum in multiregion cell in Resonance II 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

ANISN2 : Solve one-dimensional Sn equations

TWTRN2 : Solve two-dimensional Sn equations

TUD2 : Solve one dimensional diffusion theory equations

CIT2 : Solve multi-dimensional 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 in whole energy range

CONCAT : Concatenate multi-group constants separately stored for fast and thermal ranges into whole energy range ca;c.

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

CVMACT : Convert the format of the macroscopic cross sections into the original CITATION format.

REAC : Calculate the reaction rate for the specified detector with or without the filter, the spectrum parameters $\delta_{25}, \rho_{28}, \delta_{28}, C^*$, and the convertion rate.

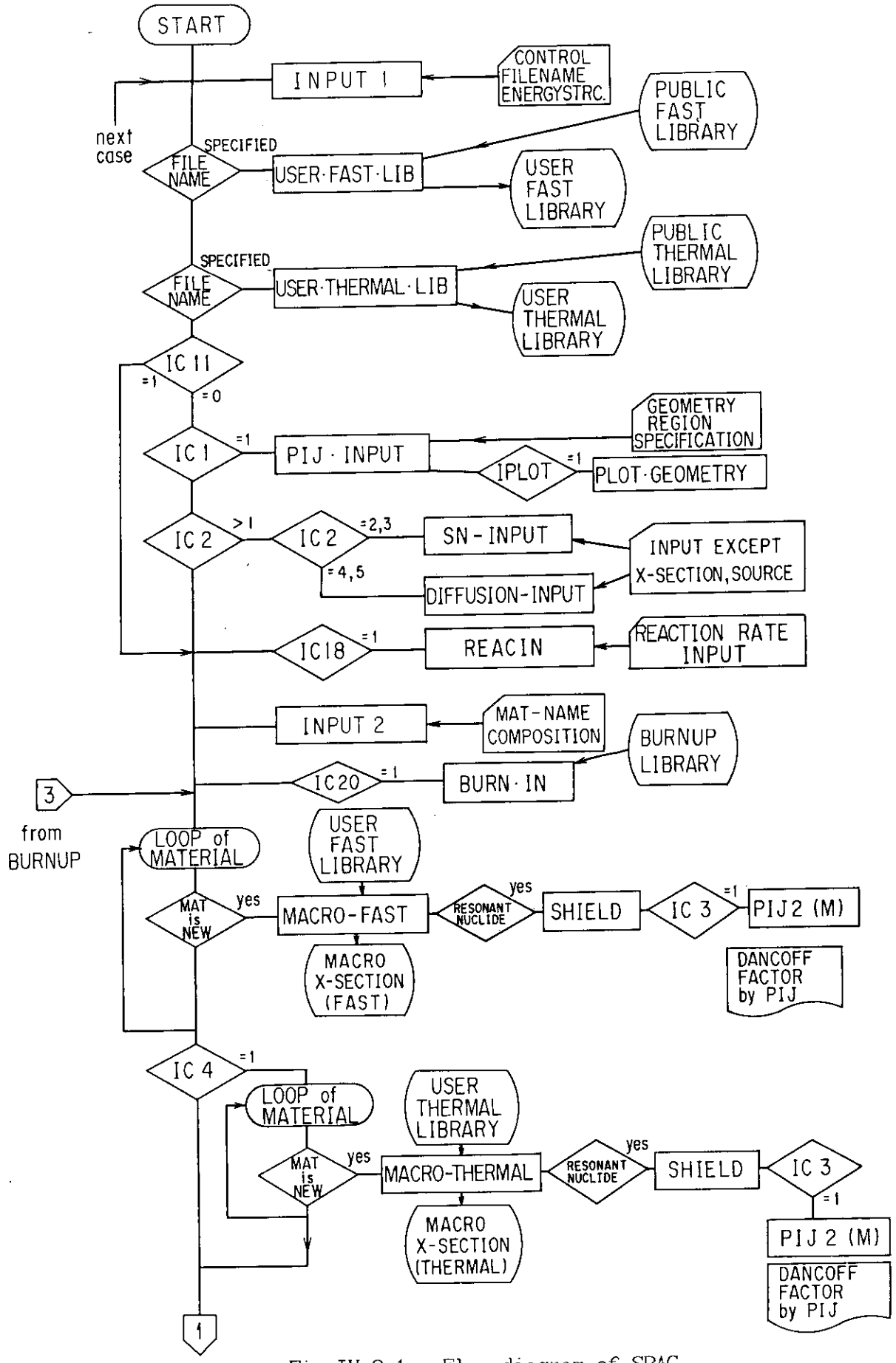


Fig.IV.3-1 Flow diagram of SRAC

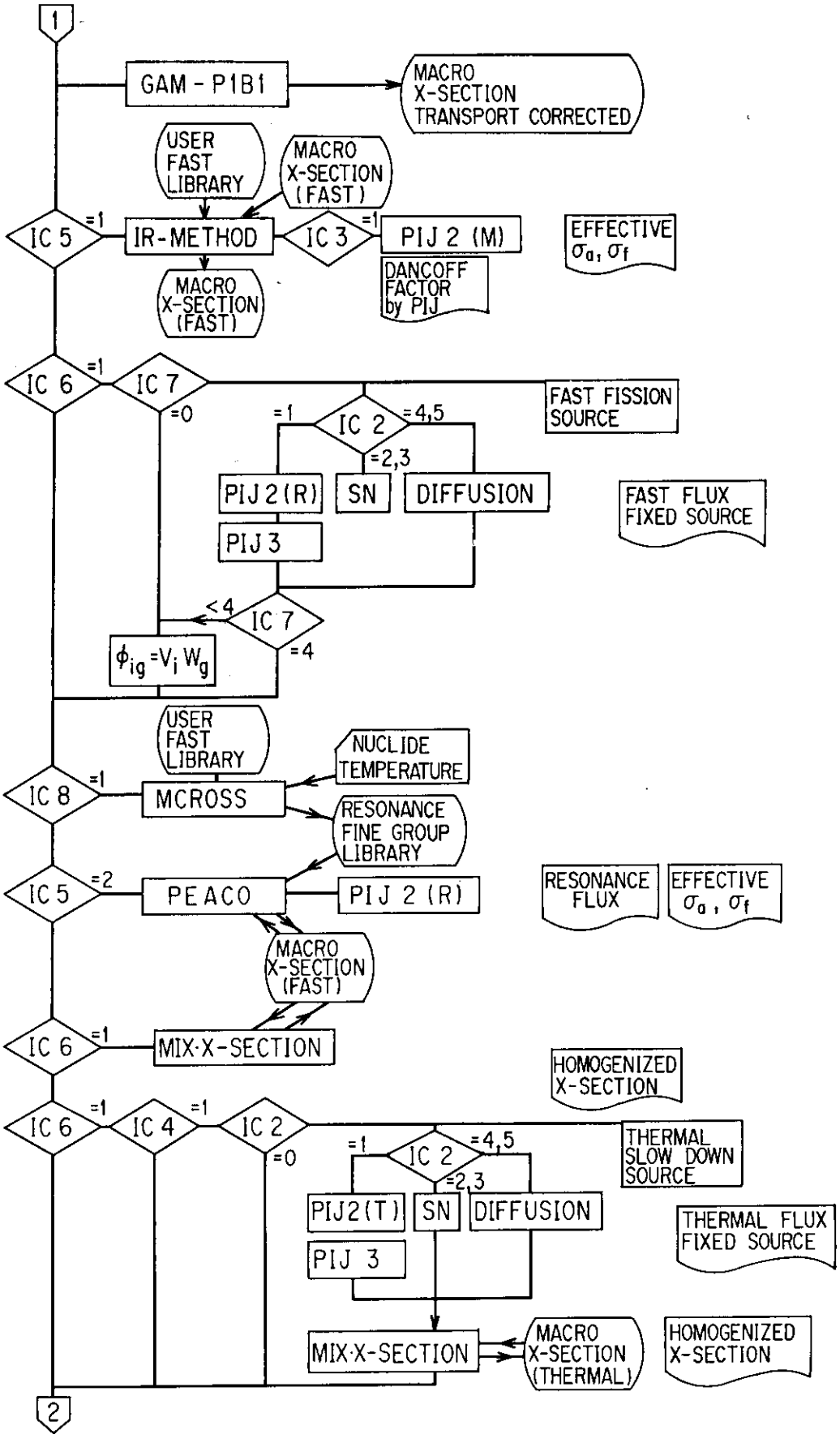


Fig. IV.3-1 (continued)

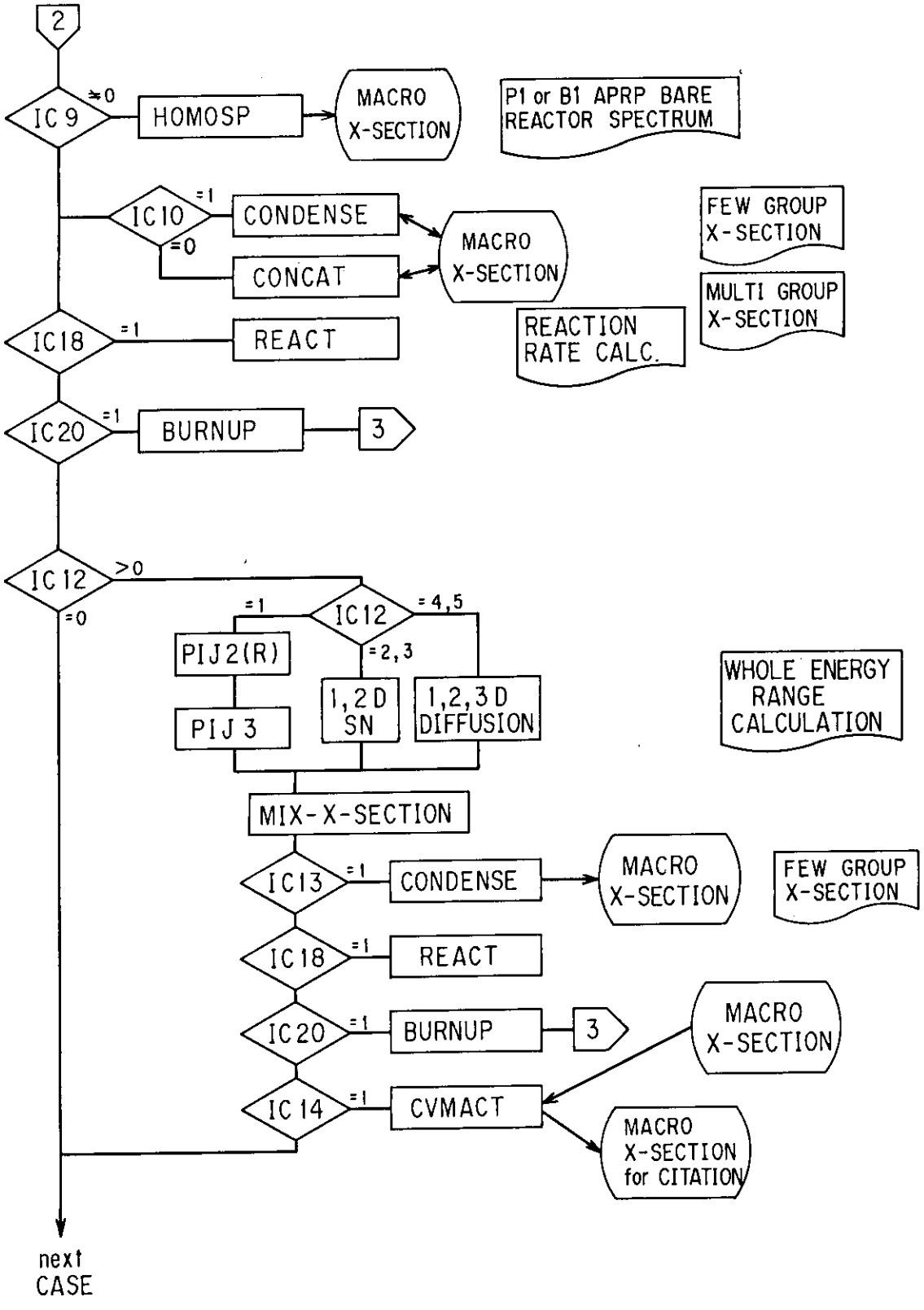


Fig. IV.3-1 (continued)

IV.4 Overlay Structure

The overlay structure is shown below. The labelled common name is marked by *.

```
 LEVEL0 MAIN SRAC CLEA READ PDSERR REAG *SEPTE *TITLEP *PCOWK2
        *TMPSET ENTAPR *ALSUB *SRACIT
   LEVEL1 INPUT1 INPUT2
   LEVEL1 D2OCHK D2OCK1
**** PIJIN ROUTINE ************************************************
   LEVEL1 PIJIN PIJ1 CHECK INSERT ELIM *PIJC *PIJ1C *PIJ1D ASCEND
          VOLPIJ
        LEVEL2 PATH PREPA MAKEPT CYL HEX HEX2 SQ SQ2 COMPAR DIVIDE SLAB
        LEVEL2 MAKETX INSET7
          LEVEL3 PATHHH PREHH GEOMHH
          LEVEL3 PATHXY PREXY GEOMXY IPRTX IPRTXP
          LEVEL3 CLUP77 GEOM7 LOCF IPRINT PREX7
        LEVEL2 MAKETC
          LEVEL3 CLUPH CLINH SECT INTRP GEOMH
          LEVEL3 CLUP CLIN GEOM
**** PLOT GEOMETRY ************************************************
        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 CHKPIN SRTCYL
                    GEOM14 HEXAGP HEXP NUMHEX
**** PIJ INTEGRATION **********************************************
   LEVEL1 PIJ2 SIGRD DELT FORM PAINT ONE TWO ENX FKIN *ABC
          SIGT FFFF PE SPLINE
**** MACROT ROUTINE ***********************************************
        LEVEL2 MACROT MACRTR
**** MACROF ROUTINE ***********************************************
        LEVEL2 MACROF *MAFWRK *MAFCNL *MAFCNT *MAFSX1 *MAFSX2 *MAFSN
             LEVEL3 MAFDAT MAFSFX
             LEVEL3 MAFCAL MAFCON
                  LEVEL4 MAFLTM MAFOUT MAFPRT MAFTTL
                  LEVEL4 MAFSIG
**** IRA ROUTINE **************************************************
        LEVEL2 IRA IRACON *IRACNL IRADAT *IRASX1 *IRACNT *IRAWRK
             LEVEL3 IRACMP IRASFX
             LEVEL3 IRACAL *IRAPRM *IRAPSE
                  LEVEL4 IRASIG
                  LEVEL4 IRARSP
                  LEVEL4 IRAPRT IRAOUT
                  LEVEL4 IRASET IRAMIX IRASRC
**** PEACO ROUTINE ************************************************
        LEVEL2 PEACO *PCOWK1 *PCOWK3 *PCOSF1 *PCOSF2 *PCOSF3
               PCOCON PCODBL
             LEVEL3 PCOSFX
             LEVEL3 PCOPRE PCOIN1 PCOIN2 PCODAT
             LEVEL3 PCOAVG PCOFIN PCOINT PCOQIN
                  LEVEL4 PCOPIJ
                  LEVEL4 PCOQIJ PCOEXP PCOCHK
             LEVEL3 PCOOUT
             LEVEL3 PCOMCR
             LEVEL3 PCOPLT
**** MCROSS ROUTINE ***********************************************
   LEVEL1 MCROSS MCRCAL MCRNBR *MCRDI1 *MCRDI2 *MCRTKN *MCRMAT *MCRBLK
```

```
          *MCRSUB *MCRDBL *MCRBRE *MCRCAR *MCRELH MCRBRD MCRPLT MCRINT
          MCRNTL MCRRSP MCRSX
***** USERFL & USERTL ROUTINE ****************************************
   LEVEL1 COLLAP *USERIX
        LEVEL2 USERFL UFLCAL UFLCON *UFLCNT UFLMPL
        LEVEL2 USERTL
**** PIJ3 ROUTINE ***************************************************
   LEVEL1 PIJ3
        LEVEL2 INP2F INP3F TEDIT
        LEVEL2 ITER RELAX MATINV
**** TUD2 ROUTINE ***************************************************
   LEVEL1 TUD2 VINT FINT
        LEVEL2 INPT2 INPT3 INPT4 OUTPT2
        LEVEL2 ITUD PROD RELAXT
**** ANISN2 ROUTINE *************************************************
   LEVEL1 WOT ERRO ADDR ITIME CLEAR WOT8 ANISN1 SN1AR S814B S804 CVMASN
          FSPVEL SNCONA
        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 AFACTR
**** ROUTINES CALLED BY SRAC
   LEVEL1 TUD1
   LEVEL1 MICREM MICREF *MICRC MOVEVT
   LEVEL1 HOMOSP GAM P1B1
   LEVEL1 CONDEN CONDEM CONCAT
   LEVEL1 FWRITE MIXX MIXX1
   LEVEL1 FSOURC TSOURC SSOURC
**** CITATION ROUTINE ***********************************************
   LEVEL1 CIT1 CIT2 RQED CALR CTFLUX *COOPD *CMARY *ASRCH *AKADD *AVDLM
          *ABURN *AFLUX *AMESH
        LEVEL2 CVMACT CITTOS
        LEVEL2 PERTIN PERTI1
        LEVEL2 IPTM
             LEVEL3 KRST
             LEVEL3 SETV CNTR GEOMC LVMX MESH COMP CMOT KOMP KMOT OVER
                    MACR SSET KSIG
             LEVEL3 BKLE FXSO BEER SRCH CNIO BNSB DISK SIZE RSTR TRAN
                    SHOX WIO3 IMXS DYPD
             LEVEL3 PERTCK
        LEVEL2 EIGN BIGS XSET EXTR CYCR GINS ITED
             LEVEL3 WFCC HOWE INFX KNFX
             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 KNSD KBPR KOOP KINS
                  LEVEL4 KNST
                  LEVEL4 KEGN KDUE
                  LEVEL4 KWRD KXRD KZRD
                  LEVEL4 KPER
                  LEVEL4 MWRD
                  LEVEL4 KTRI
```

```
            LEVEL3 NMBL WSTR DISH DIRT FASP DASH KASH DODA KRAN CRSH
                   MASH
         LEVEL2 OUTC POUT KOUT
            LEVEL3 PDWT KDWT PTAB KTAB KUDN NUDN
            LEVEL3 ELBT ESET
            LEVEL3 PERTC PERTM PERTL PERTL1 PERTC1 PERTC2 PERC11
         LEVEL2 TSCL
**** BURN-UP ROUTINE ************************************************
   LEVEL1 *AA *BB BURN11
         LEVEL2 BURNIN BURN03 BURN04 BURN05 BURN06 BURN06 BURN07 BURN08
                BURN09 BURN10
         LEVEL2 BURNUP BURN01 BURN02 ERRBRN BURN12 BURNXX
**** TWOTRAN ROUTINE ************************************************
   LEVEL1 ERROR CLEARW WWRITE REED RITE DATE1 ECRD ECWR SECOND LOAD WRF
         LEVEL2 FSOURW
         LEVEL2 TSOURW
         LEVEL2 TWTRN1 AUXTWT
            LEVEL3 SN1TR
            LEVEL3 INPT14 SNCON IFINSN PNGEN
            LEVEL3 INPT15 CSMESH MAPPER
         LEVEL2 TWTRN2 MONITR MPLY ECHECK DUMPER PCMBAL
            LEVEL3 INPUTW S966W
               LEVEL4 INPT11 DUMPRD
                  LEVEL5 CVMASW FSPVEW
               LEVEL4 INPT12 CSPREP IFINXS
               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
**** RACTION RATE ROUTINE ******************************************
   LEVEL1 REACIN REACI1
   LEVEL1 REACT FLUXRD RAFX1D RTFX1D RFLX2D RDFX2D RFLX3D REACT1 LSIGMA
          DREACT REAPRT REACT3 REAC31 PHXRD SSIGMA FREACT INTEGP CNVCAL
          CNVPRE CNVSIG CNVCA1
**** PLOTTER ROUTINE ************************************************
          NEWLVL1(REGION)
          CIRCL SYMBOL 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 NPLOTS NWHERE NOPAGE XDTTOX YDTTOY
          IXTODT IYTODT
```

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 (4*16H)
37	IUPSCAT	Control if the upscattering is assumed as selfscatter; activated if any negative IC15 is entered.
38	IBSPEC	Control whether if the P1 or B1 approx. spectrum is used to collapse the energy structure; activated if any negative IC9 in Section II.1 is entered.
39	ISCT	Order of scattering matrix required in the Sn calculation which is transferred after the input of the Sn routine.
40	IFIXS	Control if the fixed source distributions for the collision probability calculations is read in the real time; activated if IC2 = -1 is entered.
41	ICTOT	Control if the few group transport cross sections are formed from the spectrum averaged transport cross sections, activated if IC17 = -1 is entered.
42 - 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(=1) or eigenvalue problem(=0)
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 1880 (= 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	Indicator for opening the files relating to plotter to avoid duplicative 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/ Physical temperature of the mixture
XL /NMAT/ Mean chord length of the mixture (input)
DC /NMAT/ Dancoff correction factor of the mixture (input)
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 source
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	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 sphere
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)	density 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	K_0 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 Pl 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 Pl 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 = K_0 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 | 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	Transverse 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

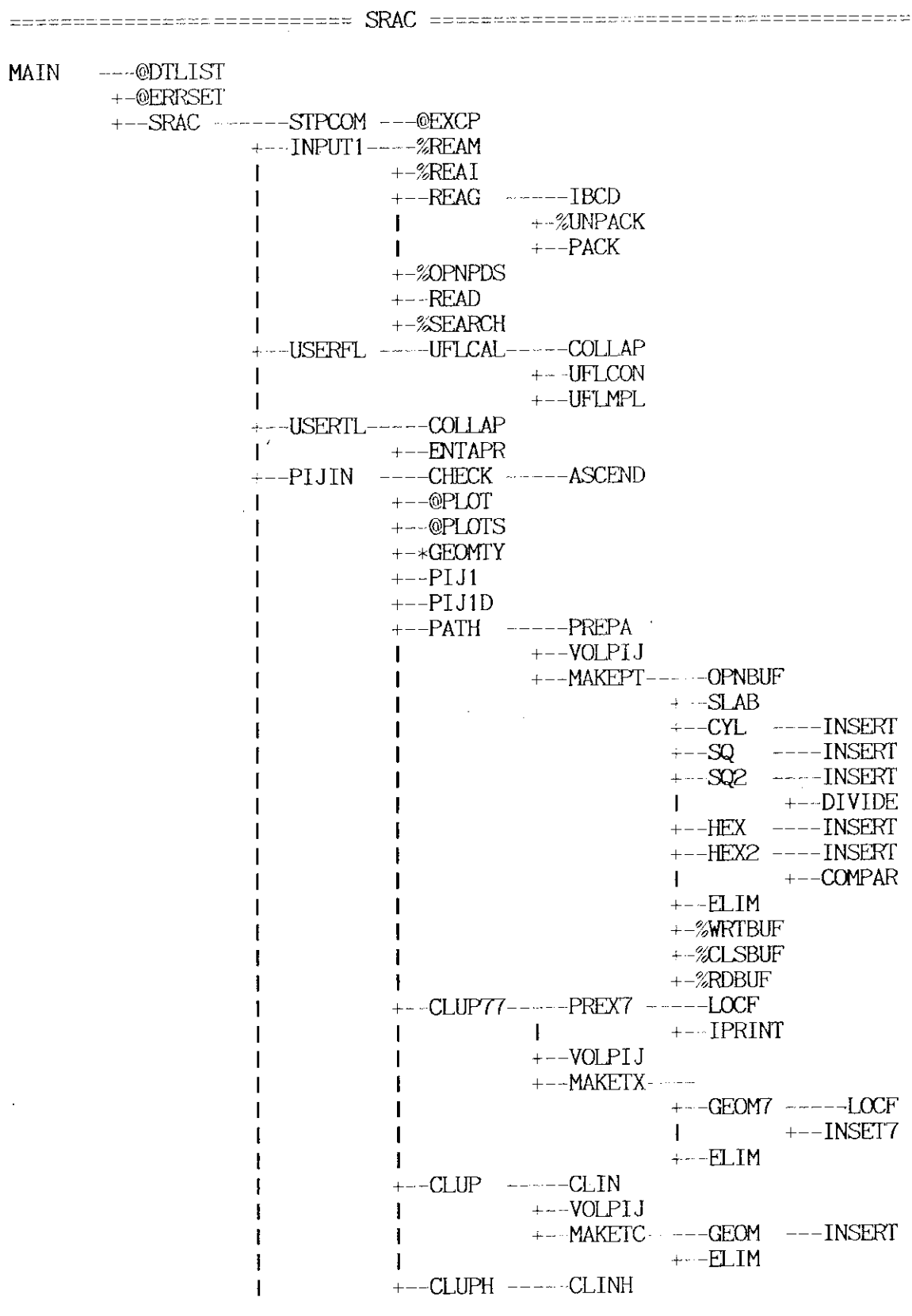
Rank B CIT1C /4000/

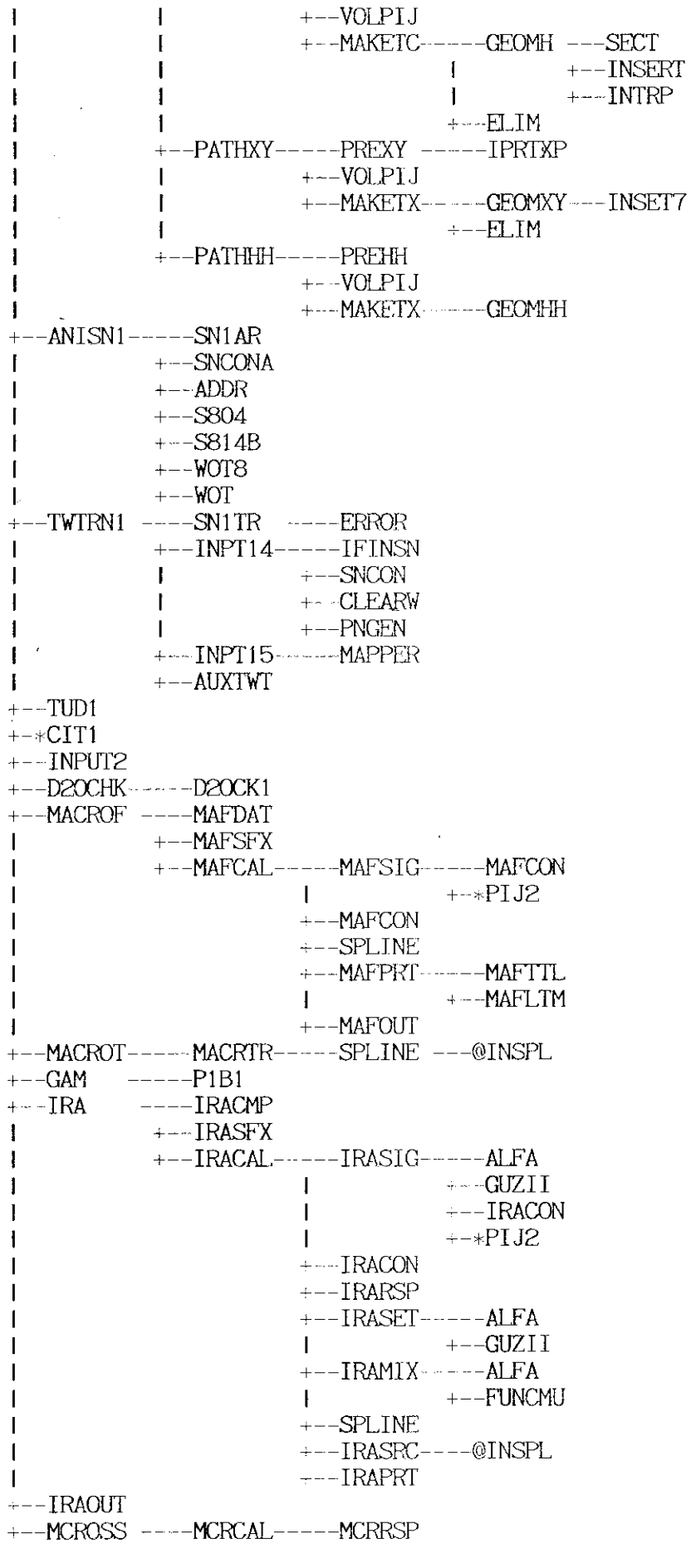
The common data between input step and executing step of CITATION: the multi-dimensional diffusion.

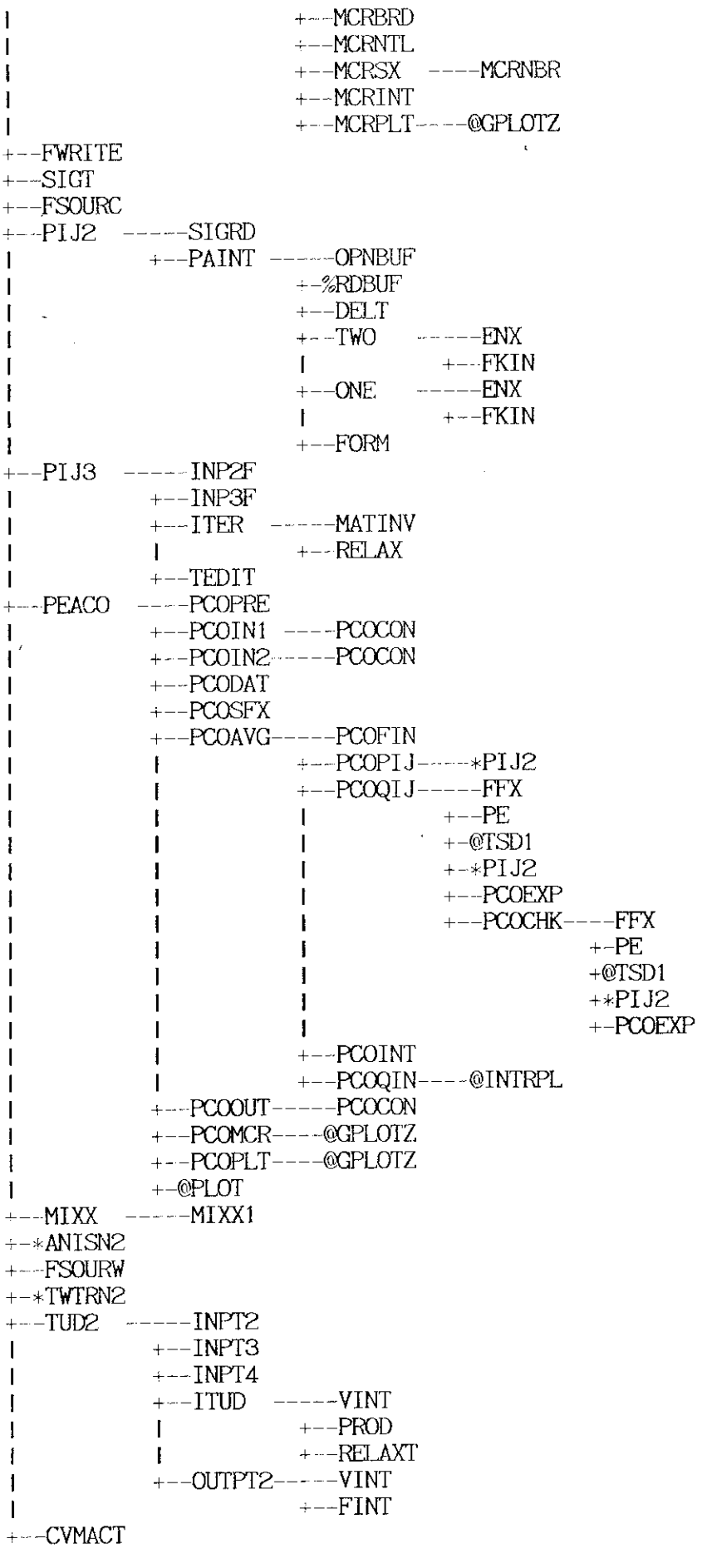
1	NM	Number of materials used in the CITATION
2	NXR	Number of regions for cross section edit
3	ID	Option for diffusion coefficients
4	IRN	Number of regions (NREGI*NREGJ*NREGK)
5	LCNM	Array address of material position number in the mixture specifications (Sect.II.8), length NM.
6	LCNXR	Array address of X-region numbers by region, length IRN
7	LCMAC	Array address of material position number by region, length IRN, if NXR>0.
8	LCVOL	Array address of volumes by region, length IRN
9	IC(3991)	Dummy vector which contains the above four arrays.

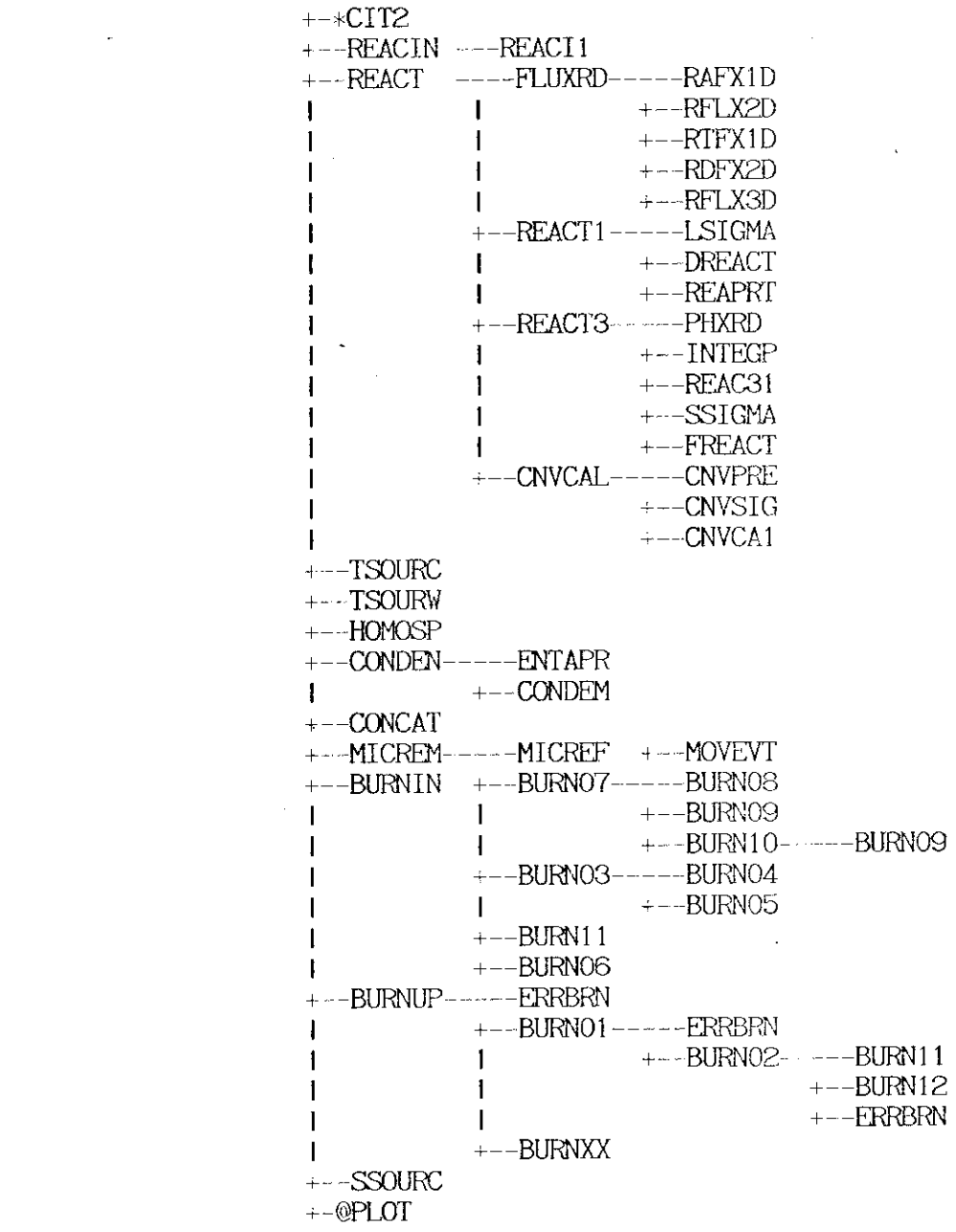
IV.6 Subroutines

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.

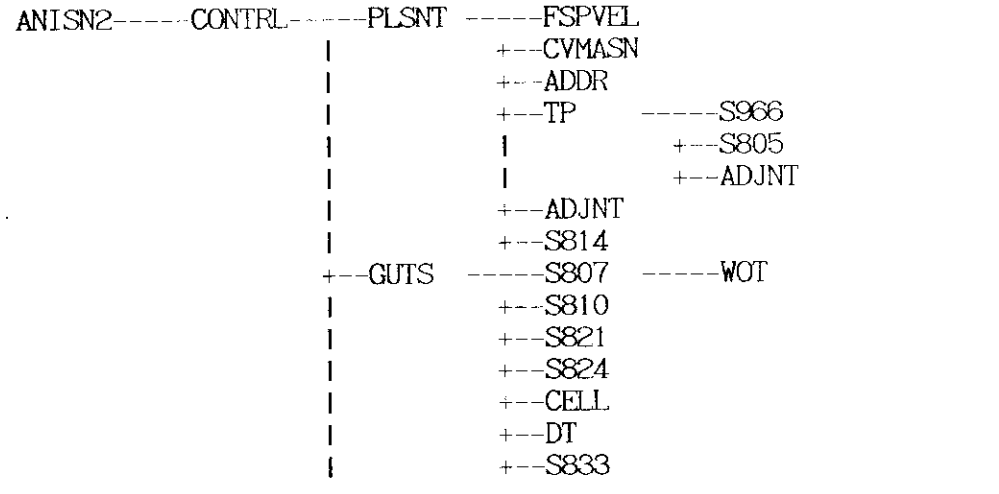


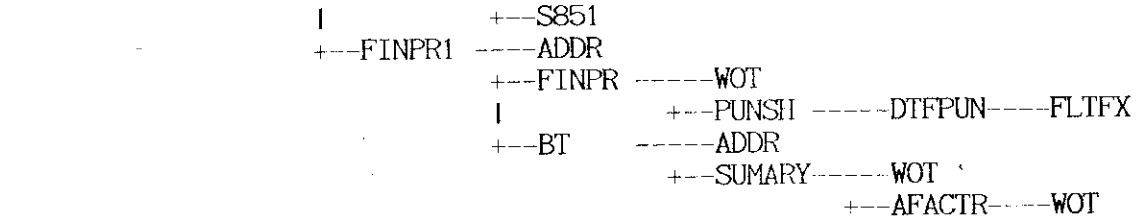

VOLPIJ
MAKETC
GEOMH
SECT
INSERT
INTRP
ELIM
PATHXY
PREXY
IPRTXP
VOLPIJ
MAKETX
GEOMXY
INSET7
ELIM
PATHHH
PREHH
VOLPIJ
MAKETX
GEOMHH
ANISN1
SN1AR
SNCONA
ADDR
S804
S814B
WOT8
WOT
TWTRN1
SN1TR
ERROR
INPT14
IFINSN
SNCON
CLEARW
PNGEN
INPT15
MAPPER
AUXTWT
TUD1
*CIT1
INPUT2
D2OCHK
D2OCK1
MACROF
MAFDAT
MAFSFX
MAFCAL
MAFSIG
MAFCON
*PIJ2
MAFCON
SPLINE
MAFPRT
MAFTTL
MAFLTM
MAFOUT
MACROT
MACRTR
SPLINE
@INSPL
GAM
P1B1
IRA
IRACMP
IRASFX
IRACAL
IRASIG
ALFA
GUZII
IRACON
*PIJ2
IRACON
IRARSP
IRASET
ALFA
GUZII
IRAMIX
ALFA
FUNCMU
SPLINE
IRASRC
@INSPL
IRAPRT
IRAOUT
MCROSS
MCRCAL
MCRRSP


+--MCRBRD
+--MCRNTL
+--MCRSX ----MCRNBR
+--MCRINT
+--MCRPLT----@GPLOTZ
+--FWRITE
+--SIGT
+--FSOURC
+--PIJ2 -----SIGRD
+--PAINT -----OPNBUF
+-%RDBUF
+--DELT
+--TWO -----ENX
+--FKIN
+--ONE -----ENX
+--FKIN
+--FORM
+--PIJ3 -----INP2F
+--INP3F
+--ITER -----MATINV
+--RELAX
+--TEDIT
+--PEACO ----PCOPRE
+--PCOIN1 ----PCOCON
+--PCOIN2-----PCOCON
+--PCODAT
+--PCOSFX
+--PCOAVG-----PCOFIN
+--PCOPIJ-----*PIJ2
+--PCOQIJ-----FFX
+--PE
+-@TSD1
+-*PIJ2
+--PCOEXP
+--PCOCHK----FFX
+-PE
+@TSD1
+*PIJ2
+-PCOEXP
+--PCOINT
+--PCOQIN----@INTRPL
+--PCOOUT-----PCOCON
+--PCOMCR----@GPLOTZ
+--PCOPLT----@GPLOTZ
+-@PLOT
+--MIXX -----MIXX1
+-*ANISN2
+--FSOURW
+-*TWTRN2
+--TUD2 -----INPT2
+--INPT3
+--INPT4
+--ITUD -----VINT
+--PROD
+--RELAXT
+--OUTPT2-----VINT
+--FINT
+--CVMACT

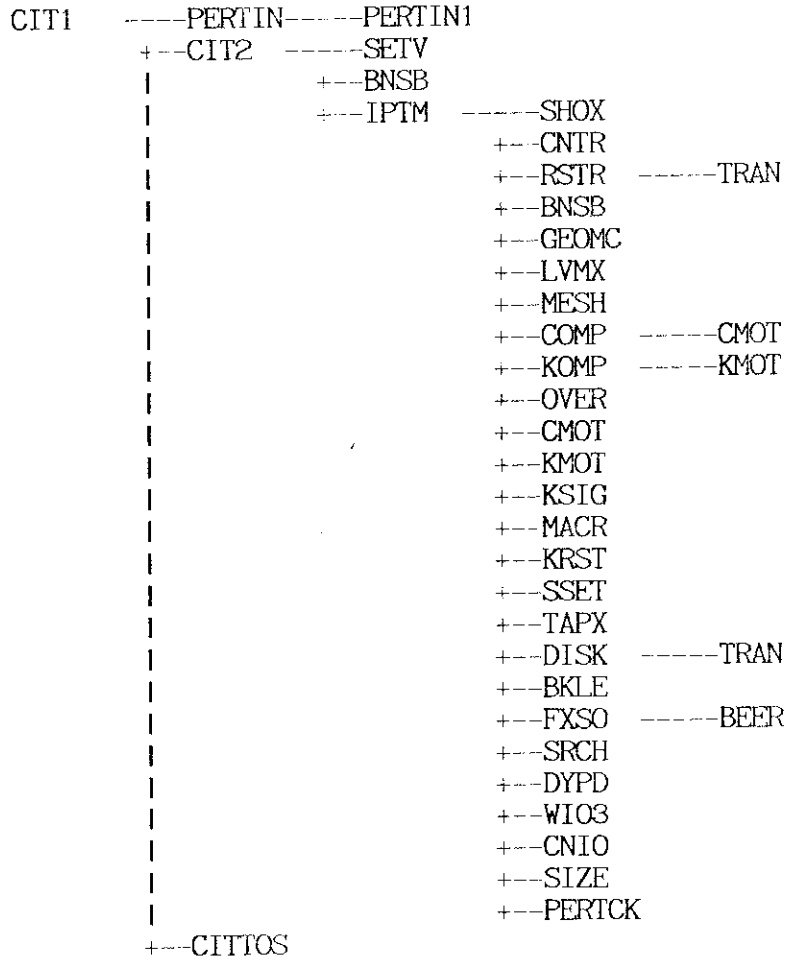


========== ANISN2 : Main part of ANISN routine ==========

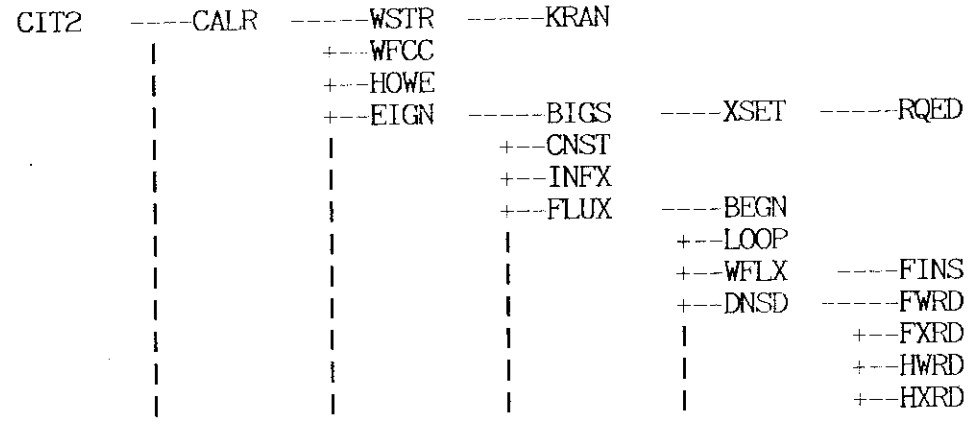


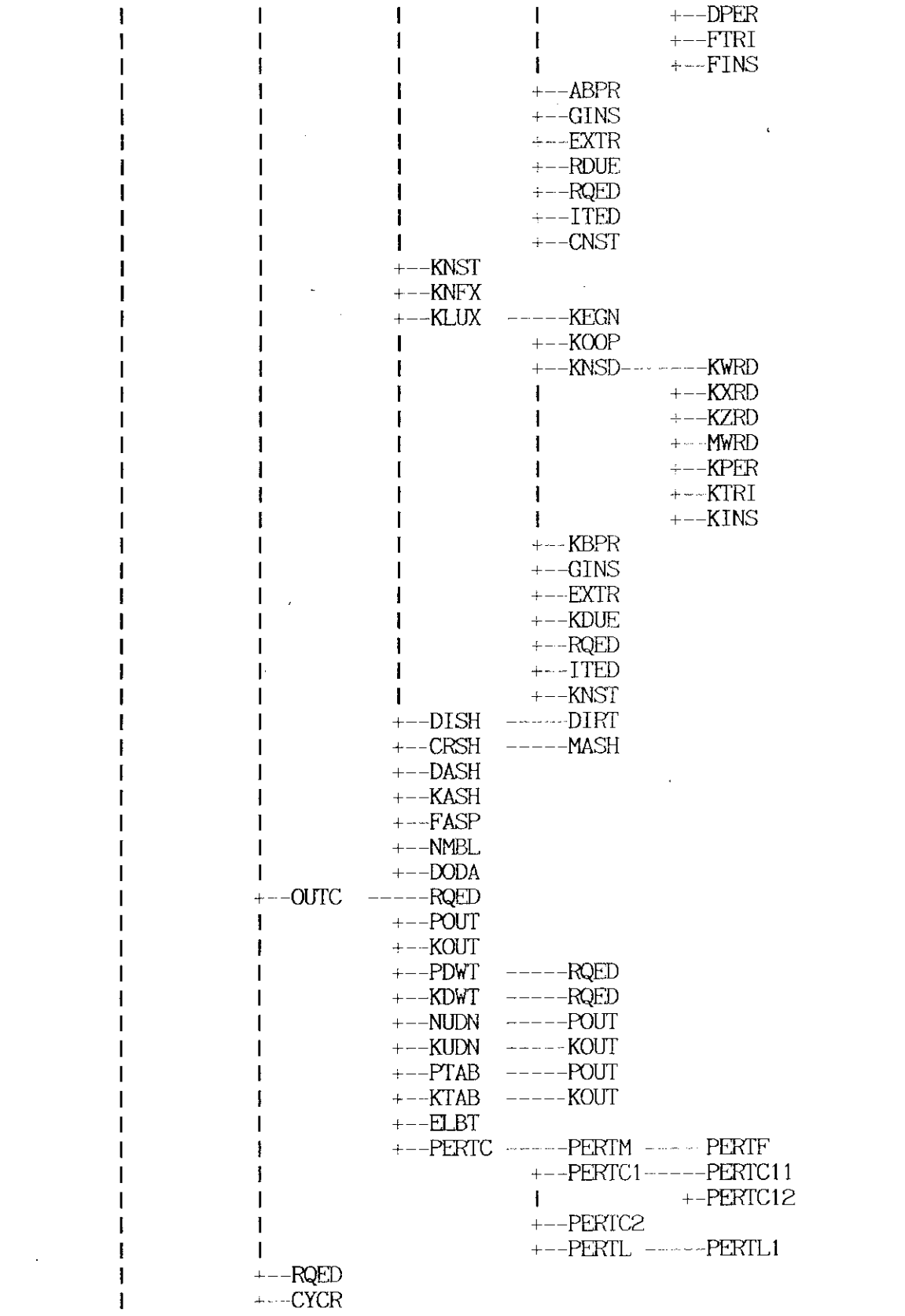


========== CIT1 : Input and check of CITATION ==========



========== CIT2 : Main part of CITATION routine ==========



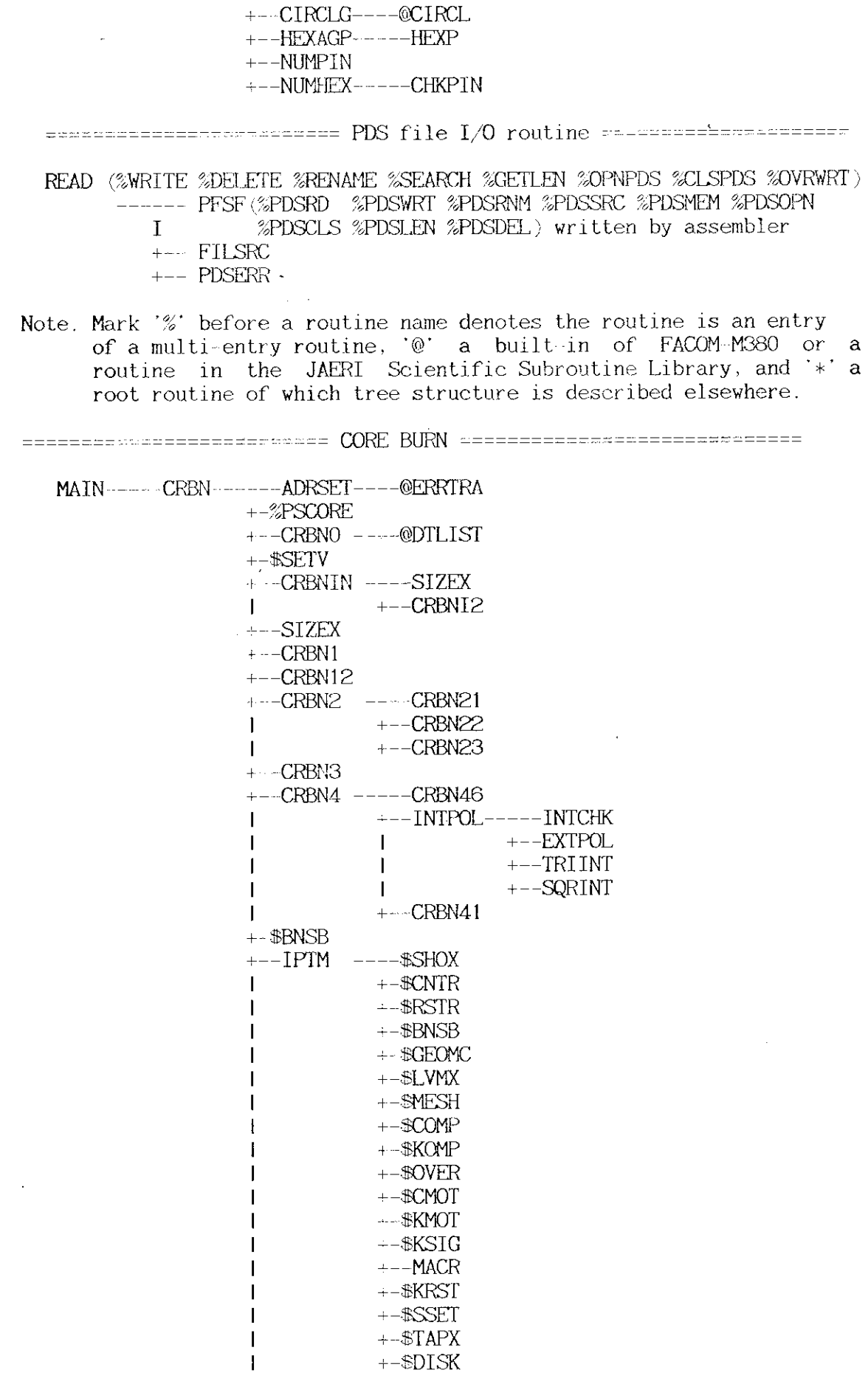


========== TWTRN2 : Main part of TWOTRAN routine ==========

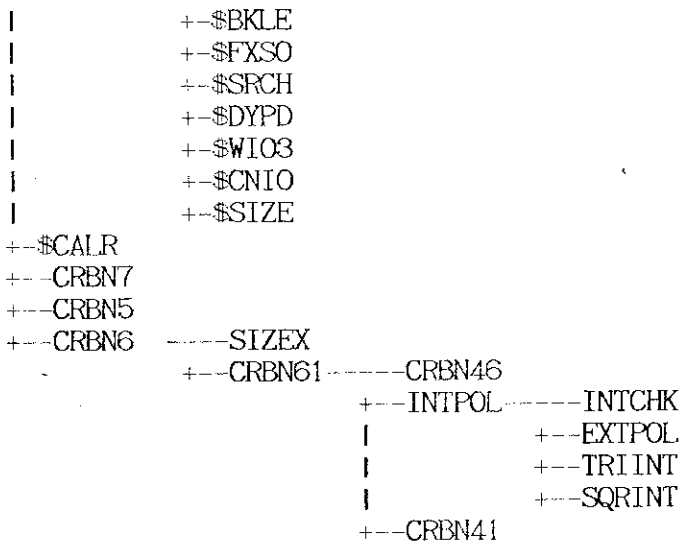
```
|           |           +--CVMASW
|           |           +--DUMPRD
|           +--INPT12-----CSPREP-----IFINXS
|           +--INPT13-----READQF
+--GRIND2-----GRID21-----INITAL-----INITQ -----MPLY
|           |           +--FISCAL
|           +--MONITR-----SECOND
|           +--GRID22-----OUTER -----DUMPER
|           |                       +--INNER -----SETBC
|           |                       |           +--STORAF-----ECWR
|           |                       |           +--SAVEAF
|           |                       |           +--IN     -----FIXUP
|           |                       |           +--OUT    -----FIXUP
|           |                       |           +--REBAL
|           |                       |           +--PCMBAL
|           |                       +--GSUMS
|           +--GRID23-----TESTS -----PCMBAL
|                                   +--REBAL
|                                   +--NEWPAR
+--OUTPT3-----OUTT31-----FINAL -----MONITR------SECONDM
            |           +--PDSFLX
            +--IFOUT -----IFRITE
            +--OUTT32-----EDCALL------GENFLO
                                    +--EDITOR------EDMAP
```

========== GEOMTY : Plot of cell geometry for PIJ ==========

```
GEOMTY ---@NEWPEN
       +--ORGSET------@PLOT
       |           +-@FACTOR
       |           +--GTITLE----@GSCHAR
       +--GEOM01-----CIRCLG ---@CIRCL
       |           +--INUM  ----@GSCHAR
       +--GEOM02
       +--GEOM04------CNTPLY----@POLY
       |           +--CIRCLS ----CIRCLG----@CIRCL
       +--GEOM05------CNTPLY----@POLY
       |           +--CIRCLS------CIRCLG----@CIRCL
       +--GEOM06------CNTPLY----@POLY
       |           +--CIRCLH-----CIRCLG----@CIRCL
       +--GEOM07------CNTPLY----@POLY
       |           +--CIRCLH------CIRCLG----@CIRCL
       +--GEOM08
       +--GEOM09------CIRCLG----@CIRCL
       +--GEOM10------CIRCLG----@CIRCL
       |           +--CIRCLP-----CIRCLG----@CIRCL
       |           +--NUMPIN
       |           +--NUMCYL------CHKPIN
       +--GEOM11------CIRCLG----@CIRCL
       |           +--CNTPLY----@POLY
       |           +--SRTCYL
       |           +--CIRCLH-----CIRCLG----@CIRCL
       |           +--CIRCLP------CIRCLG----@CIRCL
       |           +--NUMPIN
       |           +--NUMCYL-----CHKPIN
       +--GEOM13------CIRCLG----@CIRCL
       |           +-@DASHP
       +--GEOM14------HEXP
```



Note. Mark '%' before a routine name denotes the routine is an entry of a multi-entry routine, '@' a built-in of FACOM-M380 or a routine in the JAERI Scientific Subroutine Library, and '*' a root routine of which tree structure is described elsewhere.



Note. Routines marked by $ are common with those in CIT1 and CIT2.

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, 'CzzmOOOO', 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 nuclides in the 74 group structure, and the user library does that for the selected nuclides in the user's group structure.

The following eight kinds of members are stored in the library.

MEMBER NAME	Contents
'FASTLIB '	Control information for the library
'FISSYILD'	Pseudo fission neutron yield
'Czzm0000'	Control information for the nuclide zzm
'Mzzm0000'	Principal data for the nuclide zzm
'Fzzm0000'	Self-shielding factor table for the nuclide zzm
'Rzzm0000'	Control information for resonance parameters of the nuclide zzm
'Pzzm000*l*'	Resonance parameters for *l*-state of the nuclide
'Bzzm000r'	Background cross sections for the reaction r of the nuclide zzm. r = F for fission r = C for capture

r = E for elastic
'Yzzm0000' Delayed neutron parameters
'Qzzm0000' P_2 component of the elastic scattering
'Szzm0000' P_3 component of the elastic scattering
'Tzzm0000' P_4 component of the elastic scattering
'Uzzm0000' P_5 component of the elastic scattering

****

Member FASTLIB /2*NGF+5/

****

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 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 Weighted lethargy widths which will be used in collapsing the energy group structure.

Eg Boundary enrgies in eV.

Member FISSYILD /NGF/

****

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 Fission neutron yields in the g-th group normalized as sum of Xg = 1

Member Czzm0000 /42/

****

zzm (3H) 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.

ICAP = 0 no capture
= 1 capture cross sections stored

IFISS = 0 no fission
= 1 fission cross sections stored

IRP = 0 no resonance parameter
= 1 resonance parameters stored

LTOT Vector length of the member Mzzm0000 described

below.

LTHi,i=1,4 Partial vector length which contains the i-th scattering matrix, where i=1 for inelastic, i=2 for N2N, i=3 for elastic P_0, and i=4 for elastic P_1.

LAi,i=1,4 The lowest group number of the energy range where the i-th scattering occurs.

LDi,i=1,4 Number of energy groups to which the slowing-down occurs in the i-th scattering.

IFS Index for shielding factor tables
= 0 no shielding factor
= 1 shielding 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 self-shielding factor tabulation.

NSIG Number of admixture cross sections which are used as one of the arguments for interpolation of self shielding factor.

NTEMP Number of temperatures which are used as the other argument for interpolation of self-shielding factor.

AMASS,SIGP, and SIGCO
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
Temperatures for tabulation.

SIGi,i=1,NSIG
Admixture cross sections for tabulation.

IPL Order of elastic scattering. Normally =1, =5 for hydrogen.

Member Mzzm0000 /LTOT/
****
The member keeps the principal neutron cross sections.

CAPTi,i=1,NGF
Capture cross sections if ICAP=1.

FISSi,i=1,NGF
Fission cross sections if IFISS=1.

FNUi,i=1,NGF
ν fission neutron yield per fission if IFISS=1.

FSPCi,i=1,NGF
Fission neutron spectrum if IFISS=1.

TRi,i=1,NGF
Transport cross sections

WEIGHTi,i=1,NGF
Lethargy widths weighted by fission neutron spectrum

ELASi,i=1,NGF
Total elastic cross sections

N-Ni,i=1,LTH(1)
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)
N2N scattering matrix of the length
LTH(2) = (LD(2)+1)*LA(2).

ELP0i,i=1,LTH(3)
Elastic P_0 scattering matrix of the length
LTH(3) = (LD(3)+1)*LA(3).

ELP1i,i=1,LTH(1)
Elastic P_1 scattering matrix of the length
LTH(4) = (LD(4)+1)*LA(4).

Member Fzzm0000 if IFS=1
********** /NSIG*NTEMP*(NGMAX-NGMIN+1)*(IFTR-IFC+IFF+IFE+IFER+1)/
The member keeps the self-shielding factor table.

Member Rzzm0000 if IRES=1 /6/
****
The member keeps the control information for resonance level parameters.

NLS Number of neutron orbital angular momenta

SPI Nuclear spin

AP Scattering radius in unit of $10^{-12}cm$

AWR Ratio of the mass of the nuclide to that of a neutron

EL Lower limit for a energy range

EH Upper limit for a energy range

Member Pzzm000l if IRES=1 /15*NRS+1/
****

The member keeps the control information for a given l -value, repeated for l =0,NLS-1.

L Value of l

NRS Number of resolved resonances for a given l -value.

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
$\Sigma_{0p}=(\sigma_0\sigma_p AJ_J\Gamma_N/\Gamma)^{1/2}$
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 Bzzm000r if IRES=1
****

The member keeps the background cross sections for a given reaction

NR Number of energy ranges that have been given. A different interpolation scheme may be given for each range.

NP Total number of energy points used to specify the data.

NBTi,INTi,i=1,NR
Interpolation schemes

Ei,σ_i,i=1,NP
Background cross sections

Member Yzzm0000 if IFISS=1 /3+2*6+NGF*7/
****
The member keeps the information related delayed fission.

EE Incident neutron energy for the following delayed fission data in eV. (always = 0.0253)

```
   NUT        Total ν value for thermal fission.

   LAMDAi,i=1,6
              Decay constants for delayed neutron family i.

   BETAi,i=1,6
              Fraction of delayed neutron family i. Sum of BETAi=1.

   NUD k,k=1,NGF
              ν value of delayed neutron for group k in fast energy

   NUDT
              ν value of delayed neutron for thermal fission

   KAIDk,i k=1,NGF i=1,6
              Delayed fission neutron spectrum by family i

Member 'QzzmOOOO'  P2 component of the elastic scattering          /LTH/
***************
   ELP2i,i=1,LTH
              Elastic scattering matrix of the length
              LTH = (LD(4)+1)*LA(4).

Member 'SzzmOOOO'  P3 component of the elastic scattering          /LTH/
***************
   ELP3i,i=1,LTH
              Elastic scattering matrix of the length
              LTH = (LD(4)+1)*LA(4).

Member 'TzzmOOOO'  P4 component of the elastic scattering          /LTH/
***************
   ELP4i,i=1,LTH
              Elastic scattering matrix of the length
              LTH = (LD(4)+1)*LA(4).

Member 'UzzmOOOO'  P5 component of the elastic scattering          /LTH/
***************
   ELP5i,i=1,LTH
              Elastic scattering matrix of the length
              LTH = (LD(4)+1)*LA(4).
```

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 nuclides in the 48 group structure, and the user library does that for selected nuclides in the user's group structure. A file stores in a PDS file nine kinds of members as listed below.

```
MEMBER NAME   CONTENTS
*************************************************************
'THERMALt'  Control information of the library
'CzzmcOOO'  A control member for the nuclide zzm
'KzzmcOOt'  Upscattering, capture, total and fission vectors
```

```
            with or without P0 matrix
  'PzzmcOOt'  P1 matrix (given for moderating nuclides only)
  'QzzmcOOt'  P2 matrix (given only for H in H2O)
  'SzzmcOOt'  P3 matrix (given only for H in H2O)
  'TzzmcOOt'  P4 matrix (given only for H in H2O)
  'UzzmcOOt'  P5 matrix (given only for H in H2O)
  '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, through U are organized to have the same length.

We shall describe the physical contents of a member.

Member THERMALt /2*(NGT+1)/
****

NGT Number of energy groups

WTg, 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, Boundary energies, for g=1,NGT+1

Member CzzmcOOO /30/
****

INT(1) = 0 non fissile
= 1 fissile

INT(2) = sum of IM(I)
IM(1)= 0 K-matrix which keeps four vectors of up-scattering, capture, total and fission.
IM(2)= 2 K-matrix keeps P_0 matrix plus the above vectors
IM(3)= 4 plus P-matrix for P_1
IM(4)= 8 plus Q-matrix for P_2
IM(4)=16 plus S-matrix for P_3
IM(4)=32 plus T-matrix for P_4
IM(4)=64 plus U-matrix for P_5

INT(3) = 0 no self-shielding factor tabulation
= 1 self-shielding factor tabulation

INT(4) The highest group which has self-shielding factor

INT(5) The lowest group which has self-shielding factor

INT(6) Number of admixture for self-shielding factor tabulation

INT(7) Number of temperatures for which the cross sections given

INT(8),INT(9),INT(10) not used

DMi,i=1,10 not used

SIGOi,i=1,8 Admixture cross sections for the tabulation

XNU ν value in thermal range

DM20 not used

Member KzzmcOOt /NGT*(NGT+4)/or /4*NGT/
****

$\sigma_{s,g-g'}$ Scattering cross sections from group g to g', for g'=1,NGT, and g=1,NGT

$\sigma_{up,g}$ Up-scattering cross sections for g=1,NGT

$\sigma_{c,g}$ Capture cross sections for g=1,NGT

$\sigma_{t,g}$ Total cross sections for g=1,NGT

$\sigma_{f,g}$ Fission cross sections for g=1,NGT

Members PzzmcOOt, SźzmcOOt, TzzmcOOt, and UzzmcOOt have the same structure as the member KzzmcOOt, except that they keep zero values of capture and fission cross sections.

Member FzzmcOOt /3*(NGMAX-NGMIN+1)/
****

FTCPi,i=NGMIN,NGMAX
Self-shielding factors for capture cross sections

FTFSi,i=NGMIN,NGMAX
Self-shielding factors for fission cross sections

FTTRi,i=NGMIN,NGMAX
Self-shielding factors for total cross sections

V.3 Effective Microscopic Cross Section File (PDS)

The effective microscopic group cross sections in fast and thermal energy range are stored in a PDS file which will be used by DD name MICREF. The content is self-shielded group cross sections as a result of the Table-Look-Up, the IRA or the PEACO calculation. They can be used as if they are infinitely diluted cross sections.

The following four kinds of members are written into the file under the condition that the IXMICR in BLOCK 4 of Sect II.8 is specified to 1 to a nuclide and F-table is prepared in the corresponding energy range.

MEMBER NAME	Contents
'CzzmFbft'	Control information for the nuclide zzm for fast energy range
'MzzmFbft'	Principal data for the nuclide zzm for fast energy range

'CzzmTbft' Control member for the nuclide zzm for thermal energy range
'KzzmTbft' Upscattering, capture, total and fission vectors with or without P_0 matrix for thermal energy range

****

Member CzzmFbft Control for fast range /42/
****

zzm (3H) Nuclide identification composed of chemical symbol of the nuclide and the last digit of the mass number.
b (1H) Burn-up indicator if IC20=1 is specified. Otherwise the sixth character of the mixture name MTNAME in BLOCK 2 of Sect.II.8 is transferred to this tag.
f (1H) Mixture indicator; the seventh character of the mixture name is transferred.
t (1H) Temperature tag taken from that of the nuclide identification IDENT in BLOCK 4 of Sect.II.8.

The member keeps the control information of the nuclide in a vector

ICAP = 0 no capture
= 1 capture cross sections stored

IFISS = 0 no fission
= 1 fission cross sections stored

IRES = 0 no resonance parameter (fixed)

LTOT Vector length of the member MzzmOOOO described below.

LTHi,i=1,4 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 Number of energy groups to which the slowing-down occurs in the i-th scattering.

IFS Index for shielding factor tables
= 0 no shielding factor (fixed)

IFTR,IFC,IFF,IFE, and IFER
Not used

NGMIN and NGMAX
Not used

NSIG Not used

NTEMP Not used

AMASS,SIGP, and SIGCO
Not used.

TEMPi,i=1,NTEMP

Not used

SIGi,i=1,NSIG
Not used

IPL Order of elastic scattering. Normally =1

Member MzzmFbft Fast neutron scattering matrices /LTOT/
****

CAPTi,i=1,NGF
Capture cross sections if ICAP=1.

FISSi,i=1,NGF
Fission cross sections if IFISS=1.

FNUi,i=1,NGF
ν fission neutron yield per fission if IFISS=1.

FSPCi,i=1,NGF
Fission neutron spectrum if IFISS=1.

TRi,i=1,NGF
Transport cross sections

WEIGHTi,i=1,NGF
Lethargy widths weighted by fission neutron spectrum

ELASi,i=1,NGF
Total elastic cross sections

N-Ni,i=1,LTH(1)
Inelastic scattering matrix of the length
LTH(1) = (LD(1)+1)*LA(1).

N2Ni,i=1,LTH(2)
N2N scattering matrix of the length
LTH(2) = (LD(2)+1)*LA(2).

ELP0i,i=1,LTH(3)
Elastic P_0 scattering matrix of the length
LTH(3) = (LD(3)+1)*LA(3).

ELP1i,i=1,LTH(1)
Elastic scattering P_1 matrix of the length
LTH(4) = (LD(4)+1)*LA(4).

Member CzzmTbft Control for thermal range /30/
****

INT(1) = 0 non fissile
= 1 fissile

INT(2) = 0 K-matrix which keeps four vector of up-scattering, capture, total, fission (fixed)

INT(3) = 0 no self-shielding factor tabulation (fixed)

INT(4) through INT(10) Not used

DMi,i=1,10 Not used

SIGOi,i=1,8 Not used

XNU ν-value in thermal range

DM2O not used

Member KzzmTbft /4*NGT/
****

$\sigma_{up,g}$ Up-scattering cross sections for g=1,NGT

$\sigma_{c,g}$ Capture cross sections for g=1,NGT

$\sigma_{t,g}$ Total cross sections for g=1,NGT

$\sigma_{f,g}$ Fission cross sections for g=1,NGT

V.4 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 MCROSS for the resonance integral calculations.

The following two kinds of members are prepared in the library.

MEMBER NAME	CONTENTS
'CzzmOOOt'	Control information for the nuclide zzm of temperature tagged t
'FzzmrOOt'	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 /14/
****

The member keeps the control information for the nuclide zzm of temperature tagged by t.

IA Not used

NOMESH Number of broad group

NOIG Number of fine group

NFI 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	Number of S-wave resonances
NMP	Number of P-wave resonances
TEMP	Nuclide temperature in Kelvin
AM	Ratio of atomic mass of the nuclide to that of a neutron
EEUP	Upper limit of the energy range where cross sections are given
EELW	Lower limit of the energy range where cross sections are given
UIGP	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.

σ_{ri} for i=1,NOIG*NFI, r=F,C, and E.

V.5 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 with DD name 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 NAME	CONTENTS
'CONTeOOp'	Energy group structure
'nameebnp'	Cross section set
'nameebnY'	Delayed neutron data in fine group structure
'nameebnZ'	Delayed neutron data in coarse group structure
'nameebnM'	N2N cross section set in fine group structure
'nameebnN'	N2N cross section set in coarse group structure
'caseNDEN'	Summary information of a cell burnup calculation

Member CONTeOOp /2*(Number of groups+1)/
****

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 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
****

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

Note: The tag 'b' for a mixture in non depletion case keeps the sixth characters entered in the input II.8.

n (1H) X-region index coupled with case identification to identify the homogenized cross sections.
=1 for the cell averaged cross sections (total number of X-region = 1) with the case identification 'name'
=N for N-th X-region with the case identification 'name'

Note: The tag 'n' for a mixture specified by the input keeps the sixth characters entered in the input II.8.

p (1H) Index for Legendre component and energy group structure
=0 P_0 component and coarse group
=2 P_0 component(after transport correction) and fine group
=3 P_1 component and fine group
=4 P_0 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 Σact,g Activation cross section

4 Σf,g Fission cross section

5 $\nu\Sigma$f,g ν*fission cross section

6 Σt,g Total cross section

7 Xg Fission neutron yield

8 D1g Diffusion coefficient 1

9 D2g Diffusion coefficient 2

10 Σa,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.

Member nameebnY
****
The delayed neutron data for the mixture or the homogenized cell tagged by 'nameebn2' of the energy range 'e' in fine group structure.

Member nameebnZ
****
The delayed neutron data for the mixture or the homogenized cell tagged by 'nameebn0' of the energy range 'e' in coase group structure.

Member nameebnM
****
The N2N matrix data for the mixture or the homogenized cell tagged by 'nameebn2' of the energy range 'e' in fine group structure. The data organization is the same as for the standard member 'nameebn2'.

Member nameebnN
****
The N2N matrix data for the mixture or the homogenized cell tagged by 'nameebn0' of the energy range 'e' in coarse group structure. The data organization is the same as for the standard member 'nameebn0'.

Member caseNDEN
****
The member keeps 15 items of information packed in in an array which will be required in the core burn-up calculation.

case (4H) Case identification

1 NSTEP Number of burn-up steps including the initial step.
2 NTNUC Total number of depleting nuclides
3 NZONE Number of material in a cell
4 IDEPT/NZONE/

Flag if depleting (=1)or not (=0) by material

5 CASE (A8) Case identification
6 TITLE (A72) Title of the case
7 DAYS /NSTEP/ Accumulated operation days by step
8 FMWD /NSTEP/ Accumulated MW days by step
9 U235F /NSTEP/ Remaining U-235 fraction by step
10 NUCLID (A8) /NTNUC*NDEPL/
Nuclide ID(A8) for depleting nuclide, repeated by depleting materials
11 DENSTY /NSTEP*NTNUC*NDEPL/
Change of nuclide concentration by step for depleting nuclides, repeated by depleting materials
12 TFLX /NSTEP*NDEPL/
Change of absolute value of thermal flux in $cm^{-2}s^{-1}$, repeated by depleting materials
13 SXE5 /NSTEP*NDEPL/
Change of Xe-135 microscopic cross section of the lowest group as the thermal group, repeated by depleting materials
14 AFISS /NGR*NSTEP*NDEPL/
Change of few group macroscopic absorption cross sections accumulated on fissile nuclides, repeated by depleting materials
15 CFERT /NGR*NSTEP*NDEPL/
Change of few group macroscopic capture cross sections accumulated on fertile nuclides, repeated by depleting materials

V.6 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 of DDNAME 'FLUX'. For plotting and homogenization purpose, the volumes of each spatial region are written.

The following six kinds of members are used in this file.

MEMBER NAME	CONTENTS
'CONTA00p'	Energy group structure
'caseeb0p'	Neutron fluxes by R-regione by group
'caseebnp'	Neutron fluxes of n-th X-region by group
'caseeVOL'	Volumes of R-regions
'caseAbS2'	Fixed boundary source (fed by user)
'mmmmAbx2'	Neutron spectrum given for collapsing (fed by user)

Member CONTA00p /2*(Number of groups + 1)/

Number of energy groups, weights, and energy boundaries for the whole energy range.
This member is not read in SRAC. The information may be be used in some auxiliary program.

p (1H) Tag for energy group structure
=0 coarse group
=2 fine group

Member caseeb0p /Number of R-regions * number of groups/
****

case(4H) Case identification

e (1H) Tag for neutron energy range
=F Fast energy range
=T Thermal energy range
=A Whole energy range

b (1H) 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) Tag for energy group structure
=0 coarse group
=2 fine group

PHIi,g Neutron fluxes multiplied by volume i and lethargy width g for i=1,NRR,g=1,NG

Member caseebnp /Number of groups/
****

case(4H) Case identification

e (1H) Tag for neutron energy range
=F Fast energy range
=T Thermal energy range
=A Whole energy range

b (1H) 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.*

n (1H) X-region index
=1 for the cell averaged cross sections (total number of X-region = 1)
=N for N-th X-region

p (1H) Tag for energy group structure
=0 coarse group
=2 fine group

PHIg Neutron fluxes of n-th X-region multiplied by volume of the X-region and lethargy width g.

Member caseeVOL /Number of R-region/ or /Number of T-regions/
****

case(4H) Case identification

e (1H) Tag for neutron energy range
=F Fast energy range (R-region)
=T Thermal energy range (R-region)

=S Thermal energy range (T-region)
=A Whole energy range (R-region)

Vi Volumes of region i

Member caseAbS2 /Number of fine groups/
****

case(4H) Case identification

This member has to be prepared beforehand the execution. The spectrum will be used as the incurrent flux at the outer boundary of the cell. It is used in the fixed source problem specified by -2 entered to IC12 in BLOCK 3 of II.1.

Member mmmmAbx2 /Number of fine groups/
****

mmmm(4H) Mixture tag of MTNAME appearing in BLOCK 2 of II.8

bx (2H) b- and x-tag of MTNAME appearing in BLOCK 2 of II.8.

This member has to be prepared beforehand the execution. The spectrum will be used as the weight to collapse the fine group cross sections of isolated mixture like reflector material identified by 'mmmm' and 'bx' into few group constants.

V.7 I/O Files in Sequential Organization

Here we shall describe the usage of sequential files. Some of them are used in the special routine and some are commonly used. It is organized that if the DD name is identical between in a routine and in common use, any conflict use will not occur.

The following table shows the assignment of numerical DD name to the variable name defined by Fortran statements. Their physical contents are explained by routine.

DD Name	MAIN	ANISN	TWOTRAN	CITATION	RECFM	LRECL
FT01		NT1	NEDIT	I01	VBS	
FT02		NT2		I02	VBS	
FT03	(MACROF)	NT3	IVMESH	I03	VBS	
FT04	(SRAC)	NT4	NEXTRA		VBS	
FT05		NIN	NINP		VBS	
FT06	NOUT1	NOUT1	NOUT		FBA	137
FT07	(BURNUP)				FB	80
FT08		NT8	NAFLUX		VBS	
FT09			LAFLUX	IOFLX	VBS	
FT10		NT6,NT7	ISOTXS	IX77	VBS	
FT11			ISNCON	IX78	VBS	
FT12			NDUMP1		VBS	
FT13			NDUMP2	IX80	VBS	
FT14			IZMESH	IX81	VBS	
FT15				IX82	VBS	

FT16				IX83	VBS	
FT17				IX84	VBS	
FT18				IX85	VBS	
FT19				IX86	VBS	
FT20				IX87	VBS	
FT21	(PIJ)				VBS	
FT22	(DR,DZ)			IX137	VBS	
FT26				IX89	VBS	
FT28				IX90	VBS	
FT31	(CVMACT)			IX138	FB	80
FT32	(MACROF)	NSOUC	IFIXSR	IX139	VBS	
FT33	ITFLUX				VBS	
FT50	(BURNUP)				FB	136
FT51	(BURNUP)				VBS	
FT52	(MICREF)				VBS	
FT81	(PIJ)				VBS	4080
FT82	(PIJ)				VBS	4080
FT83	(PIJ)				VBS	4080
FT91				IOIN	FB	80
FT92				(CIT1,2)	FB	80
FT99	NOUT2				FBA	137

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

V.7.1 PS files for Collision Probability method

DD Name	Variable Name	Remarks
FT04F001	NFTOT	Total cross section storage unit
FT21F001		Collision probabilities storage
FT22F001		Directional collision probabilities storage unit (required if IC17=2)
FT32F001	NSOUC	Interface file for fixed source
FT33F001	NFIN	Interface file for FLUXES
FT81F001		Scratch unit for path table
FT82F001		Scratch unit for path table
FT83F001		Scratch unit for path table

V.7.2 PS files for ANISN

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
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.7.3 PS files for TWOTRAN

DD Name	Variable Name	Remarks
FT01F001	NEDIT	Edit input storage
FT02F001	IXMESH	X-region output storage to SRAC
FT04F001	NEXTRA	Scratch unit
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
FT14F001	IVMESH	Zone output storage to SRAC
FT32F001	IFIXSR	Input of inhomogeneous source fed by SRAC main
FT33F001	ITFLUX	Output form of total flux

V.7.4 PS files for TUD (1D diffusion)

DD Name	Variable Name	Remarks
FT32F001	ISOURC	Input of inhomogeneous source fed by SRAC main
FT33F001	IFOUT	Output unit of fine flux

V.7.5 PS files for CITATION

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.
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. This is the unit to which the high speed I/O is effective.
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.
FT22F001	Scratch unit used in perturbation calculation
FT26F001	Scratch unit used in perturbation calculation
FT28F001	Scratch unit used in perturbation calculation
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 to transfer reaction rate information in EBCDIC mode from CIT1 step to CIT2 step

****

V.7.6 Common PS files

DD Name	Variable Name	Remarks
FT03F001		Scratch unit for macroscopic X-section formation
FT05F001	NIN	System input unit
FT06F001	NOUT1	System print only for message.
FT32F001		Scratch file to feed the fixed source distribution : formed in FSOURC or TSOURC routine to any of transport routine
FT33F001		Scratch file used to transfer the neutron fluxes solved by any of the transport routine to MIXX routine
FT50F001		Input file to keep the chain scheme for burn-up calculation
FT51F001		Scratch file to transfer the information from the preparation step of burn-up calculation to the execution step
FT52F001		Scratch file to transfer the effective microscopic cross sections from MICREF step to BURNUP step and to the reaction rate calculation.
FT99F001	NOUT2	Print unit

****

V.8 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 3900 Bytes, Record length 130 Bytes, Record format FB, Dataset organization PO

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 records 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,A4,2I4,2E13.5)

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 fission in watt. This item is to supplied for possible fissile nuclides.

Note * The nuclides have to be ordered in NTNUC lines, first the nuclides appearing in heavy nuclide chains, second the burnable nuclides, third F.P. nuclides.

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 statements 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. nuclides in the order as appear in N-table.

We restrict that the possible fissile nuclides to 13 nuclides 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) Alphabetic ID of the F.P. given in N-table

FYIELDi Fission yield of the nuclides from the i-th fissile (order as appears in the above paragraph) for i=1,13.

VI Mathematical Formulations

VI.1 Formulations of Collision Probabilities

We present in this section the formulations to evaluate the collision probabilities for a multi-region cell expressed by either of three one-dimensional coordinate systems (plane, sphere and cylinder) or two-dimensional cylindrical coordinates. They are expressed in a form suitable to apply a numerical scheme "Ray-Trace" method developed by Tsuchihashi[1].

Description of this chapter will be started by the derivation of the linear equation to solve the Boltzmann equation by the method of collision probability, then followed by the formulations of collision probabilities in various coordinate systems, in which care is paid to evaluate the surface problem and also the directional probabilities to yield the anisotropic diffusion coefficients.

For the slab geometry, the formulation of the ordinary collision probabilities expressed by the E_{i3} function has been given by Honeck[2] starting from the plane transport kernel by E_{i1} function. Here, we shall start from the basic point kernel, operate first, the double integration along a neutron line, finally, achieve the angular integration to yield the general form of the directional collision probabilities expressed by the E_{in} function.

For the one-dimensional cylinder, the formulation derived by Takahashi[3] needs that the angular integration to scan the collision regions be repeated for each source region. The similar formulation by Honeck used in the THERMOS code[2] approximates the attenuation of neutron emitted from a source region by that from the mid-point of the source region. This approximation could be covered by sub-dividing the system into so many concentric regions. The formulation given in this chapter is a special case of the general two-dimensional cylinder. For an annular geometry, no integration over the azimuthal angle is needed. The main difference between the present formulation and the formers' is in the sequence and the coordinates for the integration. Owing to the variable transformation, the angular integration appearing in the formers' is replaced by the integration over the volume element $d\rho$ where ρ is the distance from the center to one of the parallel lines drawn across the system. The number of lines drawn for the integration is quite small compared with that of the THERMOS code to have the same measure of accuracy, partly because no repeated integration by source region is required, partly because the analytic integration along the line in the source region does not need so many sub-division of the system.

The formulation for collision probabilities in the spherical system which can be sub-divided into an arbitrary number of spherical shells has been given by Tsuchihashi & Gotoh[4] in the course of studying the resonance absorption of coated particles in the HTGR, while the escape probability for an isolated sphere by Case *et al.*[5] and for a special case of the pebble bed type HTGR by Teuchert[6] have been derived. It will be seen that the final form of the present formulation is quite similar to that of the cylindrical case.

Two-dimensional geometry has been studied, first, by Fukai[7] in his exact expression for collision probabilities in a regular lattice of

unit pin rod cell. His expression has been applied to the cluster system with regular array of pin rods by Amyot & Benoist in the PROCOPE code[8]. Fukai's expression is characterized by the numerical process to organize the ranges of double integration for each pair of pin rods between which the interaction is computed. This method requires so much sophisticated programming technique that any sub-division of coolant region has not yet been realized.

An alternate formulation has been presented by Takahashi & Nakayama[9] for the collision probabilities in square and hexagonal lattice cells which was incorporated in a thermalization code GRAFA[10], and also independently by Carlvik[11] for an annularly arrayed pin rod cluster within a cylindrical channel which was realized in a code CLUCOP and an English code PIJ[12] (both codes have not been released). In their integral formulation to compute the collision probabilities, sets of parallel lines are drawn across the system, and all intersections with region boundaries are determined and the segments of the lines in each region are measured.

Their formulation has been generalized in the treatment of outer boundary conditions, and a numerical scheme for the multi-group calculation, named "Ray-Trace" method has been established by the present author during his works to develop the PATH-C code[13] for square and hexagonal array of pin rod cells with a two-dimensional sub-division. This method has been successively applied to the CLUP code[14] for an annularly arrayed pin rod cluster such as used in the ATR, the CLUP77 code[15] for the cluster of BWR geometry, and the CLUPH code[16] for the fuel block of the VHTR.

VI.1.1 General theory

We start at the general expression to describe the behavior of neutrons in a steady state. The neutron angular flux $\varphi(\boldsymbol{r},\bar{\Omega},E)$ at position $\boldsymbol{r}$, along direction $\bar{\Omega}$ with energy E satisfies the integral form of the Boltzmann transport equation,

$$\varphi(\boldsymbol{r},\bar{\Omega},E) = \int_0^\infty dR \exp(-\overline{\Sigma R})* \left[\int_0^\infty dE' \int_{4\pi} d\bar{\Omega}' \Sigma_s(\boldsymbol{r}',\bar{\Omega}'\to\bar{\Omega},E'\to E)\varphi(\boldsymbol{r}',\bar{\Omega}',E') + S(\boldsymbol{r}',\bar{\Omega},E)\right], \qquad (VI.1\text{-}1)$$

where R is the distance between point $\boldsymbol{r}$ and $\boldsymbol{r}'$, $\bar{\Omega}$ is the directional vector given by $\bar{\Omega} = (\boldsymbol{r}-\boldsymbol{r}')/R$, $\overline{\Sigma R} = \int_0^R \Sigma(s,E)ds$ is the optical distance between point $\boldsymbol{r}$ and $\boldsymbol{r}'$, $\Sigma_s(\boldsymbol{r},\bar{\Omega}'\to\bar{\Omega},E'\to E)$ is the scattering kernel at point $\boldsymbol{r}$ from direction $\bar{\Omega}'$ at energy E' to direction $\bar{\Omega}$ at energy E, and $S(\boldsymbol{r}',\bar{\Omega},E)$ is the neutron source at point $\boldsymbol{r}'$ of direction $\bar{\Omega}$ with energy E. In the above equation, while the fission neutron source is not expressed explicitly, it may be included in the scattering term or in the source term.

Here, we assume that the scattering and the source are isotropic;

$$\Sigma_s(\boldsymbol{r},\bar{\Omega}'\to\bar{\Omega},E'\to E) = \frac{1}{4\pi}\Sigma_s(\boldsymbol{r},E'\to E), \qquad (VI.1\text{-}2a)$$

$$S(\boldsymbol{r},\bar{\Omega},E'\to E) = \frac{1}{4\pi}S(\boldsymbol{r},E'\to E). \qquad (VI.1\text{-}2b)$$

Integrating Eq. (VI.1-1) over the whole angle of $\bar{\Omega}$, we obtain,

$$\varphi(\boldsymbol{r},E) = \int_{4\pi} d\bar{\Omega}\ \frac{1}{4\pi}\int_0^\infty dR \exp(-\overline{\Sigma R})$$

$$[\int_0^\infty dE' \Sigma_s(\boldsymbol{r}',E'\to E)\varphi(\boldsymbol{r}',E') + S(\boldsymbol{r}',E)]\ , \qquad (VI.1\text{-}3)$$

where $\varphi(\boldsymbol{r},E)$ is the neutron flux at point $\boldsymbol{r}$ with E, and is defined by

$$\varphi(\boldsymbol{r},E) = \int_{4\pi} d\bar{\Omega}\ \varphi(\boldsymbol{r},\bar{\Omega},E)\ . \qquad (VI.1\text{-}4)$$

Equation (VI.1-3) can be rewritten by the relation $d\boldsymbol{r}'=R^2 dR d\bar{\Omega}$,

$$\Sigma(\boldsymbol{r},E)\varphi(\boldsymbol{r},E) = \int d\boldsymbol{r}' P(\boldsymbol{r}'\to\boldsymbol{r},E)$$

$$[\int_0^\infty dE' \Sigma_s(\boldsymbol{r}',E'\to E)\varphi(\boldsymbol{r}',E') + S(\boldsymbol{r}',E)]\ , \qquad (VI.1\text{-}5)$$

where

$$P(\boldsymbol{r}'\to\boldsymbol{r},E) = \frac{\Sigma(\boldsymbol{r})}{4\pi R^2}\exp(-\int_0^R \Sigma(s)ds]\ . \qquad (VI.1\text{-}6)$$

By the form of Eq. (VI.1-6), the reciprocity relation holds,

$$\Sigma(\boldsymbol{r}',E)P(\boldsymbol{r}'\to\boldsymbol{r},E) = \Sigma(\boldsymbol{r},E)P(\boldsymbol{r}\to\boldsymbol{r}',E)\ . \qquad (VI.1\text{-}7)$$

We divide the whole system under consideration into several regions. Each region is assumed homogeneous with respect to its nuclear properties, but different region are not necessarily of different materials. The region is the spatial variable in the collision probability method. We denote space dependent cross sections with subscript i which are associated to the region i. Integrating Eq. (VI.1-5) over V_j, we obtain

$$\Sigma_j(E)\int_{V_j}\varphi(\boldsymbol{r},E)d\boldsymbol{r} = \sum_i \int_{V_j} d\boldsymbol{r}\int_{V_i} d\boldsymbol{r}'$$

$$[\int_0^\infty \Sigma_s(E'\to E)\varphi(\boldsymbol{r}',E')dE' + S(\boldsymbol{r}',E)] * P(\boldsymbol{r}'\to\boldsymbol{r},E)\ . \qquad (VI.1\text{-}8)$$

We make the flat flux approximation so that the neutron flux $\varphi(\boldsymbol{r},E)$ is assumed constant in each region, for example, it is expressed by $\varphi_i(E)$ in the region i. This assumption leads the equation,

$$\Sigma_j(E)V_j\ \varphi_j(E) = \sum_i P_{ij}(E)V_i\ [\int_0^\infty \Sigma_{si}(E'\to E)\varphi_i(E') + S_i(E)]\ , \qquad (VI.1\text{-}9)$$

where the collision probability $P_{ij}(E)$ is defined by

$$P_{ij}(E) = \frac{\Sigma_j(E)}{4\pi V_i}\int_{V_j} d\boldsymbol{r}\int_{V_i} d\boldsymbol{r}'\cdot\frac{\exp(-\overline{\Sigma R})}{R^2}\ . \qquad (VI.1\text{-}10)$$

It is explained as the probability that a neutron emitted uniformly and isotropically in the region i has its next collision in the region j. We divide the neutron energy range into multi-groups. The average flux

in the energy interval ΔE_g is denoted by φ_{ig}. Then from Eq. (VI.1-9), we obtain the simultaneous equation,

$$\Delta E_g \Sigma_{jg} V_j \varphi_{jg} = \sum_i P_{ijg} V_i \left[\sum_{g'} \Delta E_{g'} \Sigma_{sig' \to g} \varphi_{ig'} + \Delta E_g S_{ig} \right] , \qquad (VI.1\text{-}11)$$

where ΔE_g and $\Delta E_{g'}$ are the energy width of the group g and g' and $\Sigma_{sig'g}$ is the scattering cross section in the region i from the group g' to g, and is defined by

$$\Sigma_{sig' \to g} = \int_{\Delta E_{g'}} dE' \int_{\Delta E_g} dE \Sigma_{si}(E' \to E) \varphi_i(E') \Big/ \int_{\Delta E_{g'}} dE' \varphi_i(E') . \qquad (VI.1\text{-}12)$$

As seen in the above derivation, once we obtain the collision probabilities, the neutron flux can be easily obtained by solving the simultaneous equation Eq. (VI.1-11) by means of a matrix inversion or an iterative process.

Now we focus our considerations on the collision probability. The definition of the collision probability Eq. (VI.1-10) can be expressed equivalently by

$$P_{ij} = \frac{1}{4\pi V_i} \int_{V_i} dr \int_{4\pi} d\bar{\Omega} \int_{R_{j-}}^{R_{j+}} dR \Sigma_j \exp\left\{ -\int_0^R \Sigma(s) ds \right\} , \qquad (VI.1\text{-}13)$$

where the subscript to indicate the energy group is, hereafter, dropped for simplicity, and R_{j-} and R_{j+} denote the distances from point r to the inner and outer boundaries of the region j along the line through the points r and r'.

From the form of Eq. (VI.1-10), it can be seen easily that the reciprocity relation holds,

$$P_{ji} \Sigma_j V_j = P_{ij} \Sigma_i V_i . \qquad (VI.1\text{-}14)$$

This relation is, as shown later, utilized to reduce the angular range of numerical integration.

The integration by R between R_{j-} and R_{j+} in Eq. (VI.1-13) can be performed analytically in the homogeneous region j,

$$\int_{R_{j-}}^{R_{j+}} dR \Sigma_j \exp(-\overline{\Sigma R}) = \exp(-\overline{\Sigma R}_{j-}) * (1 - \exp(-\Sigma_j (R_{j+} - R_{j-}))$$
$$= \exp(-\overline{\Sigma R})_{R=R_{j-}} - \exp(-\overline{\Sigma R})_{R=R_{j+}} .$$

The summation over j along the direction $\bar{\Omega}$ leaves only the first term of $\exp(-\overline{\Sigma R})_{R=0} = 1$, then the conservation law is easily shown as,

$$\sum_j P_{ij} = \frac{1}{4\pi V_i} \int_{V_i} dr \int_{4\pi} d\bar{\Omega} = 1 . \qquad (VI.1\text{-}15)$$

Similarly, the directional probabilities P_{ijk} defined by Benoist[17] which is used to provide the Behrens term of the anisotropic diffusion coefficients is expressed by

$$P_{ijk} = \frac{1}{4\pi V_i}\int_{V_i} dr \int_{4\pi} d\bar{\Omega}\ 3\Omega_k^2 \int_{R_{j-}}^{R_{j+}} dR\Sigma_j \exp\{-\int_0^R \Sigma(s)ds\} \quad , \qquad (VI.1\text{-}16)$$

where k stands for direction, for example the parallel or perpendicular to the boundary plane in the case of plane lattice, and Ω_k denotes the directional cosine of $\bar{\Omega}$ in the direction k. The following relation holds:

$$\sum_k \Omega_k^2 = 1 \quad . \qquad (VI.1\text{-}17)$$

The extension to include surfaces given by Beardwood[12] is as follows: If S is any surface (not necessarily closed) such that no line drawn outwards from a surface point $\bar{S}$ crosses S more once,

$$P_{is} = \frac{1}{4\pi V_i}\int_{V_i} dr \int d\bar{S}\ \frac{(r-\bar{S})}{R_s^3} \exp\{-\int_0^{R_s} \Sigma(s)ds\} \quad , \qquad (VI.1\text{-}18)$$

is the probability that a neutron emitted from the region i crosses the outer boundary S, or an alternative expression,

$$P_{is} = \frac{1}{4\pi V_i}\int_{V_i} dr \int_{4\pi} d\bar{\Omega}\ \exp\{-\int_0^{R_s} \Sigma(s)ds\} \quad , \qquad (VI.1\text{-}19)$$

is given, where R_S is a distance from the emitting point r to the surface point $\bar{S}$.

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 using the explanation given by Bonalumi[18].

We assume the system with neither 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 is divided into N regions. We consider the balance of the collision rate in the region i

$$\Sigma_i \varphi V_i = \frac{\varphi}{4} S G_i + \sum_{j=1}^{N} P_{ji}\Sigma_j \varphi V_j \quad , \qquad (VI.1\text{-}20)$$

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

Σ_i ; total macroscopic cross section,
φ ; uniform scalar flux,
V_i ; volume,
S ; area of the surface,
G_i ; probability that a neutron impinging on the surface has its first collision in the region i,
P_{ji} ; probability that a neutron emitted in the region j has its first collision in the region i.

The term on the left hand side (L.H.S.) denotes the collision rate in the region i. 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 Eq.(VI.1-14) and the conservation theorem;

$$\sum_{j=1}^{N} P_{ij}+P_{is} = 1 \quad , \tag{VI.1-21}$$

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

$$G_i = \frac{4V_i}{S}\Sigma_i P_{is} \quad . \tag{VI.1-22}$$

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

$$G_s = 1-\sum_{i=1}^{N} G_i \quad . \tag{VI.1-23}$$

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:

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

which can be rewritten as

$$P_{ij}(lattice) = P_{ij}(isolated)+P_{is}\frac{G_j}{1-G_s} \quad . \tag{VI.1-24}$$

VI.1.2 Collision probabilities for slab lattice

In a one-dimensional slab geometry shown in Fig.VI.1-1, we have

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

$$dr=dx,$$
$$d\Omega=2\pi\sin\theta d\theta .$$

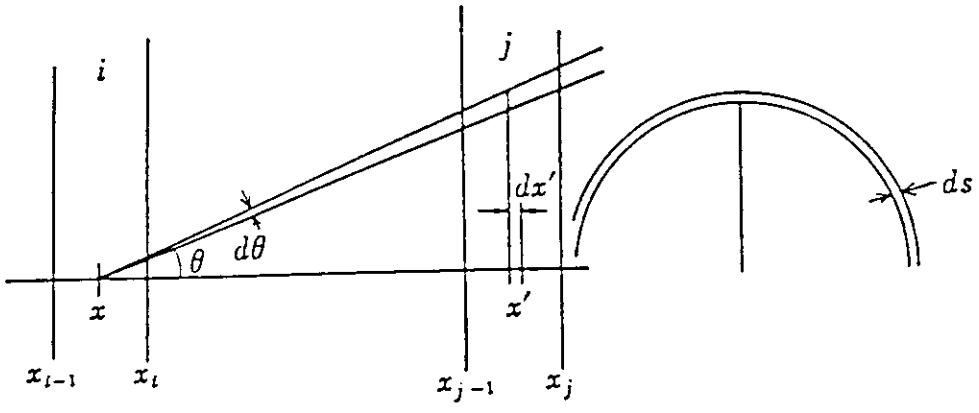


Fig.VI.1-1 Coordinates in slab geometry

We assume that the system is divided into an array of slabs. The slab i has its left edge at x_{i-1} and its total cross section denoted by Σ_i. Then we have

$$P_{ij} = \frac{1}{2(x_i-x_{i-1})}\int_{x_{i-1}}^{x_i} dx \int_{x_{j-1}}^{x_j} dx' \int_0^{\pi/2} \Sigma(x') \frac{\sin\theta}{\cos\theta} \\ *\exp\{-\int_x^{x'} \Sigma(t)dt/\cos\theta\}\}d\theta \ , \qquad (VI.1\text{-}25)$$

for the case $x_i < x_{j-1}$, and the optical distance which appears in the exponential term in Eq. (VI.1-25) 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 exponential integral function, E_{in} defined by Schloemich[19]

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

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

$$P_{ij} = \frac{1}{2\lambda_i}\{E_{i3}(\lambda_{ij})-E_{i3}(\lambda_{ij}+\lambda_i)-E_{i3}(\lambda_{ij}+\lambda_j)+E_{i3}(\lambda_{ij}+\lambda_i+\lambda_j)\} , \qquad (VI.1\text{-}26)$$

where

$$\lambda_{ij} = \sum_{k=i+1}^{j-1}\lambda_k , \quad \text{for} \quad x_i<x_j . \qquad (VI.1\text{-}27)$$

Equation (VI.1-26) is identical with the formulation derived by Honeck[2] started by the plane transport kernel expressed by the E_{i1} function.

Next we shall consider the case where $x_i>x_j$, 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 $x_i < x_j$ we get the same expression as Eq. (VI.1-26) except for the definition of λ_{ij}. In this case we have

$$\lambda_{ij} = \sum_{k=j+1}^{i-1}\lambda_k , \quad \text{for} \quad x_i>x_j . \qquad (VI.1\text{-}28)$$

In the last case where $x_i = x_j$, the optical distance in Eq.(VI.1-25) 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 P_{ii} by

$$P_{ii} = 1-\frac{1}{\lambda_i}\{E_{i3}(0)-E_{i3}(\lambda_i)\} \quad . \tag{VI.1-29}$$

If the λ_i's are so small that the differences in Eq.(VI.1-26) and in Eq.(VI.1-29) can not be obtained accurately in the numerical calculation, we should use the following differential forms instead of Eqs.(VI.1-26) and (VI.1-29), respectively,

$$P_{ij} = \frac{\lambda_j}{2} E_{i1}(\lambda_{ij}+\lambda_i/2+\lambda_j/2) \ , \tag{VI.1-30}$$

$$P_{ii} = \lambda_i E_{i1}(\lambda_i/2) \ . \tag{VI.1-31}$$

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

We consider now the lattice cell system where a unit cell is divided into N regions and the collision region j lies periodically. A sum of such terms as Eq.(VI.1-26) gives

$$\begin{aligned} P_{ij} = \frac{1}{2\lambda_i} \sum_{l=0} \{ & E_{i3}(\lambda_{ij}^{l1})-E_{i3}(\lambda_{ij}^{l1}+\lambda_i)-E_{i3}(\lambda_{ij}^{l1}+\lambda_j)+E_{i3}(\lambda_{ij}^{l1}+\lambda_i+\lambda_j) \\ & +E_{i3}(\lambda_{ij}^{l2})-E_{i3}(\lambda_{ij}^{l2}+\lambda_i)-E_{i3}(\lambda_{ij}^{l2}+\lambda_j)+E_{i3}(\lambda_{ij}^{l2}+\lambda_i+\lambda_j)\} , \end{aligned} \tag{VI.1-32}$$

where

$$\begin{aligned} \lambda_{ij}^{l1} &= \sum_{k=i+1}^{j-1} \lambda_k + l*\sum_{k=1}^{N} \lambda_k , \\ \lambda_{ij}^{l2} &= \sum_{k=1}^{i-1} \lambda_k + \sum_{k=j+1}^{N} \lambda_k + \sum_{k=1}^{i-1} \lambda_k + l*\sum_{k=1}^{N} \lambda_k \ . \end{aligned} \tag{VI.1-33}$$

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, say, 6, respectively. For the case where $j<i$, i and j in Eq.(VI.1-33) must be replaced by j and i, respectively.

For the case $i=j$ we have

$$\begin{aligned} P_{ii} = 1-\frac{1}{\lambda_i}\{E_{i3}(0)-E_{i3}(\lambda_i)\} \\ +\frac{1}{\lambda_i} \sum_{l=0} \{E_{i3}(\lambda_{ij}^{l})-E_{i3}(\lambda_{ij}^{l}+\lambda_i)-E_{i3}(\lambda_{ij}^{l}+\lambda_j)+E_{i3}(\lambda_{ij}^{l}+\lambda_i+\lambda_j), \end{aligned} \tag{VI.1-34}$$

where $$\lambda_{ij}^{l} = (l+1)*\sum_{k=1}^{N} \lambda_k - \lambda_i \ . \tag{VI.1-35}$$

When the lattice cell is arranged symmetrically, we may reduce the

computing time into half using the reciprocity relation, while it is not utilized in the SRAC code.

Now we consider the explicit form of the directional probabilities. For the perpendicular direction to the boundary plane, we have $3\cos^2\theta$ as $3\Omega_\perp^2$, c.f. Fig.VI.1-1, by which the integrand in Eq.(VI.1-25) has to be multiplied. The similar procedure gives us the expression of $P_{ij\perp}$ corresponding to Eq.(VI.1-26) as follows:

$$P_{ij\perp} = \frac{3}{2\lambda_i}\{E_{i5}(\lambda_{ij}) - E_{i5}(\lambda_{ij}+\lambda_i) - E_{i5}(\lambda_{ij}+\lambda_j) + E_{i5}(\lambda_{ij}+\lambda_i+\lambda_j)\}, \qquad (VI.1\text{-}36)$$

and corresponding to Eq.(VI.1-29)

$$P_{ii\perp} = 1 - \frac{3}{2\lambda_i}\{E_{i5}(0) - E_{i5}(\lambda_i)\} . \qquad (VI.1\text{-}37)$$

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

$$P_{ij} = \frac{1}{3} P_{ij\perp} + \frac{2}{3} P_{ij\parallel} . \qquad (VI.1\text{-}38)$$

The relation is derived 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 P_{is}:

$$P_{is} = \frac{1}{2\lambda_i}\{E_{i3}(\lambda_{is}^1) - E_{i3}(\lambda_{is}^1+\lambda_i) + E_{i3}(\lambda_{is}^2) - E_{i3}(\lambda_{is}^2+\lambda_i)\}, \qquad (VI.1\text{-}39)$$

where

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

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

VI.1.3 Collision probabilities for one-dimensional cylindrical lattice

We consider the infinitely long cylinder which is divided into several annular shells. The outer radius of the shell i is r_i. We suppose that a neutron emitted at the point P in the shell i has it first collision at the point Q in the shell j. 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.1-2a, we have the collision probability P_{ij} as,

$$P_{ij} = \frac{2}{V_i}\int_{r_{i-1}}^{r_i} r dr \int_0^{\pi} d\beta \int_0^{\pi/2} \sin\theta d\theta \int_{R_{j-}}^{R_{j+}} dR \frac{\Sigma_j}{\sin\theta} \exp\left\{-\int_0^R \frac{\Sigma(s)ds}{\sin\theta}\right\}, \qquad (VI.1\text{-}40)$$

where

$$V_i = \pi (r_i^2 - r_{i-1}^2). \qquad (VI.1\text{-}41)$$

Then we transform the variables r, β and R into new ones ρ, x and x' as illustrated in Fig.VI.1-2b. We define the perpendicular distance OM from O to the line PQ' by ρ, the distance between P and M by x, and the distance between Q' and M by x'. There are three relations among variables:

$$r^2 = \rho^2 + x^2,$$
$$r \sin\beta = \rho, \qquad (VI.1\text{-}42)$$
$$R = x' - x.$$

Using these relations we have the Jacobian

$$\frac{\partial(r,\beta,R)}{\partial(\rho,x,x')} = \frac{1}{r}. \qquad (VI.1\text{-}43)$$

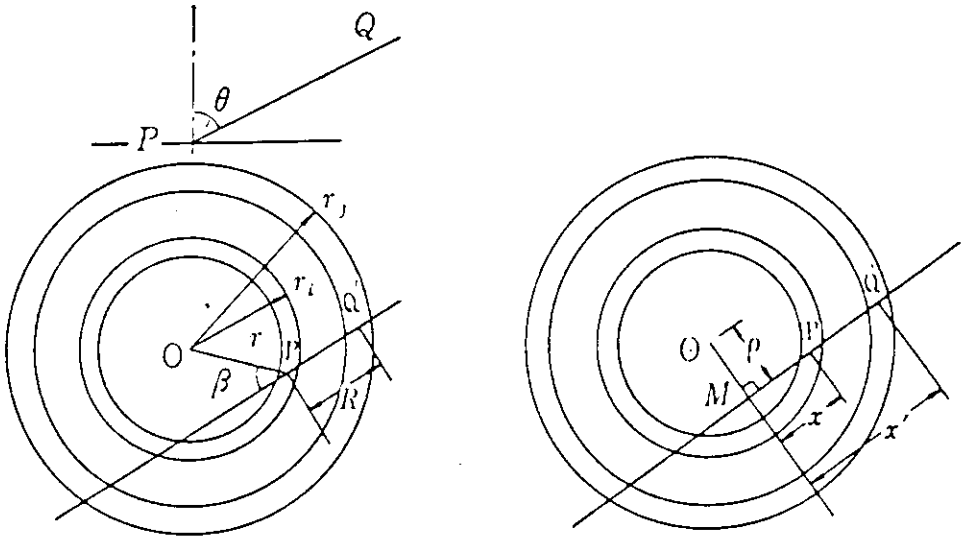


Fig.VI.1-2 Cylindrical coordinates

Then we can rewrite Eq.(VI.1-40) using new variables by

$$P_{ij} = \frac{2}{V_i}\int_0^{r_i} d\rho \int_0^{\pi/2} d\theta \int_{x_{i-1}}^{x_i} dx \int_{x_{j-1}}^{x_j} dx' \Sigma_j \{\exp\{-|\int_x^{x'} \Sigma(t)dt|/\sin\theta\}$$
$$+\exp\{-|\int_{-x}^{x'} \Sigma(t)dt|/\sin\theta\}\}, \qquad (VI.1\text{-}44)$$

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 Σ_i denotes the total macroscopic cross section of the shell i.

For the case $r_j>r_i$, the optical distances which appear in the exponential terms in Eq. (VI.1-44) 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 the Bickley-Naylor function[20]

$$K_{in}(x) = \int_0^{\pi/2}d\theta\sin^{n-1}\theta\ \exp(-\frac{x}{\sin\theta}). \qquad (VI.1\text{-}45)$$

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

$$P_{ij} = \frac{2}{\Sigma_i V_i}\int_0^{r_i}d\rho\ \{K_{i3}(\lambda_{ij}^1)-K_{i3}(\lambda_{ij}^1+\lambda_i)-K_{i3}(\lambda_{ij}^1+\lambda_j)+K_{i3}(\lambda_{ij}^1+\lambda_i+\lambda_j)$$
$$+K_{i3}(\lambda_{ij}^2)-K_{i3}(\lambda_{ij}^2+\lambda_i)-K_{i3}(\lambda_{ij}^2+\lambda_j)+K_{i3}(\lambda_{ij}^2+\lambda_i+\lambda_j)\}, \qquad (VI.1\text{-}46)$$

where

$$\lambda_{ij}^1 = \sum_{k=i+1}^{j-1}\lambda_k,$$

$$\lambda_{ij}^2 = \sum_{k=1}^{i-1}\lambda_k + \sum_{k=1}^{j-1}\lambda_k \qquad \text{for} \quad r_i<r_j. \qquad (VI.1\text{-}47)$$

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

$$\left|\int_x^{x'}\Sigma(t)dt\right| = \int_{x'}^{x}\Sigma_i(t)dt = \Sigma_j(x_j-x')+(\Sigma_i(x-x_{i-1})+\sum_{k=j+1}^{i-1}\lambda_k.$$

Integration over x and x' gives the final form of P_{ij} for the case $r_j<r_i$, which is exactly same expression as in Eq. (VI.1-46) but λ_{ij}^1 in Eq. (VI.1-47) must be replaced by

$$\lambda_{ij}^1 = \sum_{k=j+1}^{i-1}\lambda_k \qquad \text{for} \quad r_i>r_j. \qquad (VI.1\text{-}48)$$

We have not yet considered the case where the shell i coincides

with the shell j. In this case the optical distances are 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.1-44), we must divide the range into (x_{i-1},x) and (x,x_i) and then we have

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

Using the K_{in} function we get the final form of P_{ii} as follows:

$$P_{ii} = \frac{2}{\Sigma_i V_i}\int_0^{r_{i-1}}d\rho\{2\lambda_i-2K_{i3}(0)+2K_{i3}(\lambda_i)$$
$$+K_{i3}(\lambda_{ii})-2K_{i3}(\lambda_{ii}+\lambda_i)+K_{i3}(\lambda_{ii}+2\lambda_i)\}$$
$$+\frac{2}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i}d\rho\{2\lambda_i-K_{i3}(0)+K_{i3}(2\lambda_i)\}\ , \qquad (VI.1\text{-}49)$$

where

$$\lambda_{ii} = 2\sum_{k=1}^{i-1}\lambda_k. \qquad (VI.1\text{-}50)$$

If the λ_i's are so small that the differences in the brackets of Eq.(VI.1-46) and (VI.1-49) can not be obtained accurately in numerical calculation, we should use instead of Eqs.VI.1-46 and VI.1-49 the following differential forms:

$$P_{ij} = \frac{2}{\Sigma_i V_i}\int_0^{r_i}d\rho\lambda_i\lambda_j\{K_{i1}(\lambda_{ij}^1)+K_{i1}(\lambda_{ij}^2)\}\ , \qquad (VI.1\text{-}51)$$

$$P_{ii} = \frac{1}{\Sigma_i V_i}\int_0^{r_{i-1}}d\rho\{\lambda_i^2K_{i1}(\frac{\lambda_i}{2})+\lambda_i^2K_{i1}(\lambda_{ii})\}$$
$$+\frac{2}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i}d\rho\lambda_i^2K_{i1}(\lambda_i). \qquad (VI.1\text{-}52)$$

If we assume the cylindricalized cell with the perfect reflective outer boundary, more terms like those in Eq.(VI.1-46) are required as follows;

$$K_{i3}(\lambda_{ij}^3)-K_{i3}(\lambda_{ij}^3+\lambda_i)-K_{i3}(\lambda_{ij}^3+\lambda_j)+K_{i3}(\lambda_{ij}^3+\lambda_i+\lambda_j)$$

$$+K_{i3}(\lambda_{ij}^4)-K_{i3}(\lambda_{ij}^4+\lambda_i)-K_{i3}(\lambda_{ij}^4+\lambda_j)+K_{i3}(\lambda_{ij}^4+\lambda_i+\lambda_j)$$

$$+\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots$$

$$+K_{i3}(\lambda_{ij}^n)-K_{i3}(\lambda_{ij}^n+\lambda_i)-K_{i3}(\lambda_{ij}^n+\lambda_j)+K_{i3}(\lambda_{ij}^n+\lambda_i+\lambda_j)$$

$$+\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots,$$

where

$$\lambda_{ij}^3 = \lambda_{ij}^1 + \lambda_j + 2\sum_{k=i+1}^{N}\lambda_k,$$

$$\lambda_{ij}^4 = \lambda_{ij}^2 + \lambda_j + 2\sum_{k=i+1}^{N}\lambda_k,$$

$$\lambda_{ij}^n = \lambda_{ij}^{n-2} + \lambda_j + 2\sum_{k=i+1}^{N}\lambda_k.$$

These terms to be used in the integrand of Eq. (VI.1-46) are the generalized form of those appearing in the expression of $P_{1\to 2}$ given by Takahashi[3], while the integration variable has not yet been transformed to ρ.

As regards the directional probabilities in the cylindrical coordinates, we know for the axial direction $3\Omega_z^2=3\cos^2\theta$ and for the radial direction $3\Omega_r^2=(3/2)\sin^2\theta$. For the latter, P_{ijr} is obtained by multiplying the integrand in Eq. (VI.1-40) by $(3/2)\sin^2\theta$. It is not worth while to repeat here the whole expressions for each condition. It is enough for us to know only the fact that all the terms expressed by $K_{in}(x)$ must be replaced by $(3/2)K_{i(n+2)}(x)$. Similarly to the slab system, the following relation holds:

$$P_{ij} = \frac{1}{3}P_{ijz} + \frac{2}{3}P_{ijr} . \qquad (VI.1\text{-}53)$$

We know that the isotropic boundary condition brings more accurate result and is less time-consuming than the perfect reflective boundary condition to evaluate the flux distribution in the real cell by the cylindricalized model. In this case the probability, P_{is} that a neutron emitted in the shell i escapes from the outer boundary without suffering any collision is required. It is easily obtained as

$$P_{is} = \frac{2}{\Sigma_i V_i}\int_0^{r_n} d\rho\,\{K_{i3}(\lambda_{is}^1)-K_{i3}(\lambda_{is}^1+\lambda_i)+K_{i3}(\lambda_{is}^2)-K_{i3}(\lambda_{is}^2+\lambda_i)\} , \qquad (VI.1\text{-}54)$$

where

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

$$\lambda_{is}^2 = \sum_{k=1}^{i-1}\lambda_k + \sum_{k=1}^{N}\lambda_k. \qquad (VI.1\text{-}55)$$

VI.1.4 Collision probabilities for spherical system

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

$$P_{ij} = \frac{1}{4\pi V_i} \int_{V_i} dV \int_{4\pi} d\bar{\Omega} \int_{R\subset V_j} dR \Sigma_j \exp\left\{-\int_0^R \Sigma(s)ds\right\} \quad . \qquad (VI.1\text{-}56)$$

The integrand on R.H.S. of Eq.(VI.1-56) is interpreted as follows by referring Fig.VI.1-3a. A neutron emitted at a point P in the region i moves toward a point Q which is in distance R from the 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 of the system, C. The position of the point Q is defined by the distance R from the point P, and the angle θ made by the lines PQ and PC (see Fig.VI.1-3a). In this coordinate system,

$$dV = 4\pi r^2 dr \qquad 0<r<R_N,$$

$$d\bar{\Omega} = 2\pi \sin\theta d\theta \qquad 0 \leqq \theta \leqq \pi \ ,$$

and Eq.(VI.1-56) is rewritten by a triple integral form:

$$P_{ij} = \frac{4\pi\Sigma_j}{V_i} \int_0^{R_n} r^2 dr \int_0^{\pi} \sin\theta d\theta \int_{R\subset V_j} dR \exp\left\{-\int_0^R \Sigma(s)ds\right\} \ , \qquad (VI.1\text{-}57)$$

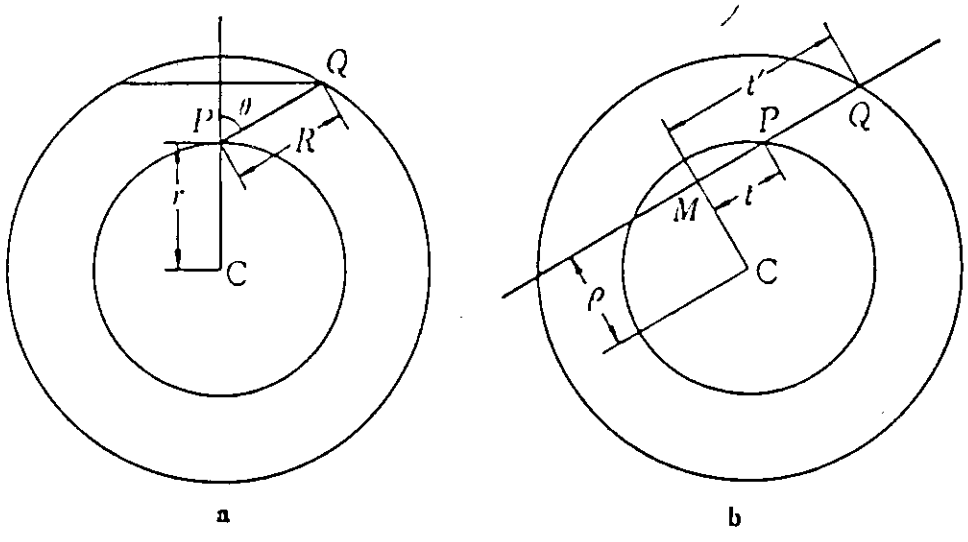


Fig.VI.1-3 Spherical coordinates

To perform analytically the integration as far as possible, the coordinates shown in Fig.VI.1-3a are transformed into the new coordinates shown in Fig.VI.1-3b 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:

$$\begin{aligned} r^2 &= t^2 + \rho^2, \\ r\sin\theta &= \rho, \\ R &= t' - t. \end{aligned} \qquad (2\text{-}58)$$

The Jacobian is then obtained as follows:

$$\frac{\partial(r,\theta,R)}{\partial(\rho;t,t')} = -\frac{1}{r}. \qquad (VI.1\text{-}59)$$

The probability is rewritten using the new variables by

$$P_{ij} = \frac{2\pi\Sigma_j}{V_i}\int_0^{R_n}\rho d\rho\int_{t\subset V_i}dt\int_{t'\subset V_j}dt'\exp\{-\int_0^{t-t'}\Sigma(s)ds\}\ . \qquad (VI.1\text{-}60)$$

As the nuclear cross section in each shell is uniform, we can integrate Eq.(VI.1-60) over t and t'. Finally the shell-to-shell collision probabilities are given in the form of single integral (see Fig.VI.1-4);

$$P_{ij} = \frac{2\pi}{\Sigma_i V_i}\int_0^{r_{i-1}}\rho d\rho\ \{1-\exp(-\lambda_i)\}\ \{1-\exp(-\lambda_j)\}$$
$$*\{\exp\{-\sum_{k=i+1}^{j-1}\lambda_k\}+\exp\{-2\sum_{k=i}^{i-1}\lambda_k-\lambda_i-\sum_{k=i+1}^{j-1}\lambda_k\}\}$$
$$+\frac{2\pi}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i}\rho d\rho\{\{1-\exp(-2\lambda_i)\}\ \{1-\exp(-\lambda_j)\}\exp\{-\sum_{k=i+1}^{j-1}\lambda_k\}\}$$
$$\text{for}\quad i<j, \qquad (VI.1\text{-}61)$$

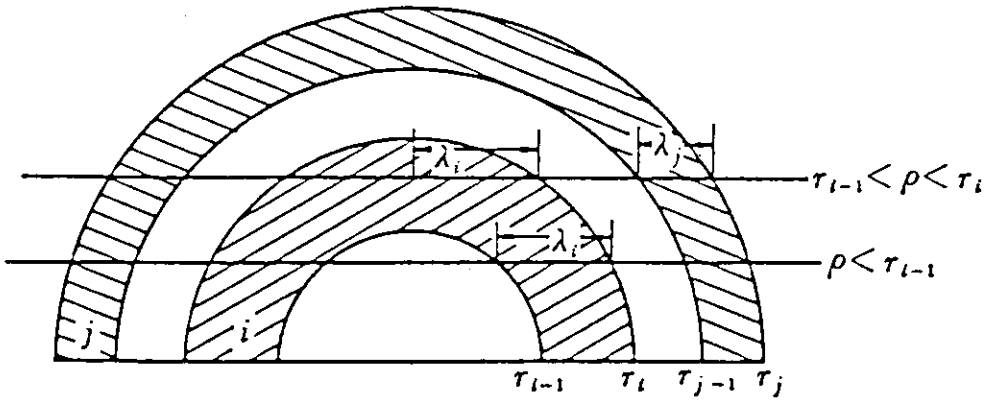


Fig.VI.1-4 Neutron lines in case i<j

$$P_{ii} = \frac{2\pi}{\Sigma_i V_i}\int_0^{r_{i-1}}\rho d\rho\{\{\lambda_i-1+\exp(-\lambda_i)\}\ \{1-\exp(-\lambda_j)\}^2\{\exp\{-2\sum_{k=1}^{i-1}\lambda_k\}\}$$
$$+\frac{2\pi}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i}\rho d\rho\{2\lambda_i-1+\exp(-2\lambda_i)\}\quad \text{for}\quad i=j, \qquad (VI.1\text{-}62)$$

where

$$t_i = \sqrt{r_i^2-\rho^2}\quad \text{for}\quad r_i\geqq\rho,$$

$$t_i = 0 \qquad \text{for} \quad r_i < \rho, \tag{VI.1-63}$$

$$\lambda_i = \Sigma_i (t_i - t_{i-1}). \tag{VI.1-64}$$

For $i>j$, the similar expression to Eq.(VI.1-61) can easily be obtained, but the reciprocity theorem gives P_{ji} directly from P_{ij}.

Now we have the escape probability P_{iS} as

$$P_{is} = \frac{2\pi}{\Sigma_i V_i}\int_0^{r_{i-1}} \rho d\rho \{1-\exp(-\lambda_i)\} * \{\exp\{-\sum_{k=i+1}^{N}\lambda_k\} \{1+\exp(-2\sum_{k=i}^{i-1}\lambda_k - \lambda_i)\}\} + \frac{2\pi}{\Sigma_i V_i}\int_{r_{i-1}}^{r_i} \rho d\rho \{1-\exp(-2\lambda_i)\}\exp(-\sum_{k=i+1}^{N}\lambda_k). \tag{VI.1-65}$$

In the SRAC code, the integrands in Eqs.(VI.1-61) and (VI.1-62) to possible pairs of (i,j) are, first calculated for a fixed ρ. Then, the integration over ρ is accomplished by changing the value of ρ .

VI.1.5 Collision probabilities for two-dimensional cylindrical lattice

In the cylindrical system with general shape of its cross section and of infinite height, the collision probability from a region i to another region j, P_{ij}, is expressed by the following Eq.(VI.1-66) in the coordinate system of Fig.VI.1-5, assuming flat flux in each region and isotropic emission in the laboratory system;

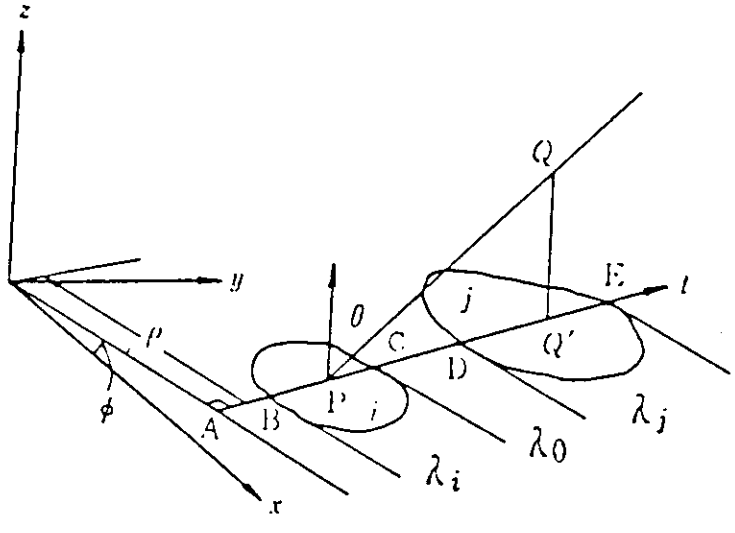


Fig.VI.1-5 Cylindrical coordinate system

$$P_{ij} = \{\int_{-\infty}^{\infty}\rho d\rho\int_0^{2\pi}d\varphi\int_0^{\pi/2}\sin\theta d\theta\int_{AB}^{AC}dt\exp\{-\frac{\Sigma_i|AC-t|}{\sin\theta}\}\int_{AD}^{AF}dt'\frac{\Sigma_j}{\sin\theta}$$

$$*\exp\{-\frac{\Sigma_j|t'-AD|}{\sin\theta}\}\exp\{-|\int_{AC}^{AD}\Sigma(s)ds|/\sin\theta\}\}$$

$$/\int_{-\infty}^{\infty}\rho d\rho\int_0^{2\pi}d\varphi\int_0^{\pi/2}\sin\theta d\theta\int_{AB}^{AC}dt \quad . \qquad (VI.1\text{-}66)$$

In Fig.VI.1-5, the line PQ' defined by ρ and φ is the projection of the neutron line 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 the region j more than once, a sum of Eq.(VI.1-66) is required.

The self collision probability, P_{ii} is expressed by the following Eq.(VI.1-67), where the point Q is in the region i.

$$P_{ii} = \int_{-\infty}^{\infty}\rho d\rho\int_0^{2\pi}d\varphi\int_0^{\pi/2}\sin\theta d\theta\int_{AB}^{AC}dt\int_t^{AC}dt'\frac{\Sigma_i}{\sin\theta}\exp\{-\frac{\Sigma_i|t-t'|}{\sin\theta}\}$$

$$/\int_{-\infty}^{\infty}\rho d\rho\int_0^{2\pi}d\varphi\int_0^{\pi/2}\sin\theta d\theta\int_{AB}^{AC}dt \quad . \qquad (VI.1\text{-}67)$$

If the line PQ' reenters the region i, a sum of such a term as Eq.(VI.1-66) is also required for obtaining P_{ii}.

The six-fold integrals of Eqs.(VI.1-66) and (VI.1-67) are reduced to the double integrals as follows:

$$P_{ij} = \frac{1}{2\pi\Sigma_i V_i}\int_{-\infty}^{\infty}d\rho\int_0^{2\pi}d\varphi\{K_{i3}(\lambda_0)-K_{i3}(\lambda_0+\lambda_i)$$

$$-K_{i3}(\lambda_0+\lambda_j)+K_{i3}(\lambda_0+\lambda_i+\lambda_j)\}, \qquad (VI.1\text{-}68)$$

$$P_{ii} = \frac{1}{2\pi\Sigma_i V_i}\int_{-\infty}^{\infty}d\rho\int_0^{2\pi}d\varphi\{\lambda_i-K_{i3}(0)+K_{i3}(\lambda_i)\}, \qquad (VI.1\text{-}69)$$

where λ_i and λ_j denote the optical path lengths (the physical path multiplied by the macroscopic total cross section), λ_i=BC*Σ_i, and λ_j=DE*Σ_i and λ_0 stands for the sum of optical path lengths between C and D; and K_{i3} is the third order Bickley-Naylor function.

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

$$P_{is} = \frac{1}{2\pi\Sigma_i V_i}\int_{-\infty}^{\infty}d\rho\int_0^{2\pi}d\varphi\{K_{i3}(\lambda_{is})-K_{i3}(\lambda_{is}+\lambda_i)\}, \qquad (VI.1\text{-}70)$$

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

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

$$P_{ijr} = \frac{3}{4\pi\Sigma_i V_i}\int_{-\infty}^{\infty} d\rho \int_0^{2\pi} d\varphi \{K_{i5}(\lambda_0) - K_{i5}(\lambda_0+\lambda_i) - K_{i5}(\lambda_0+\lambda_j) + K_{i5}(\lambda_0+\lambda_i+\lambda_j)\}, \quad (VI.1\text{-}71)$$

$$P_{iir} = \frac{1}{4\pi\Sigma_i V_i}\int_{-\infty}^{\infty} d\rho \int_0^{2\pi} d\varphi \{2\lambda_i - 3K_{i3}(0) + 3K_{i3}(\lambda_i)\}. \quad (VI.1\text{-}72)$$

Thus we have the double integration of the linear combination of K_{in} function as a final form of the collision probabilities for the two-dimensional cylindrical system.

VI.1.6 Ray-Trace method for integration of collision probabilities

The integrations by ρ and φ in Eqs.(VI.1-66) and (VI.1-67) are carried out by the trapezoidal integration formula with equal width and weight in a general two-dimensional geometry. As seen, a pair of ρ and φ determines a neutron line across the cell.

In one-dimensional geometries such as cylinder and sphere, no integration over the azimuthal angle φ is needed as the geometry is invariant about the azimuthal angle φ. In this case, the range of the integration over ρ; $(0, r_N)$ is sub-divided into N regions by r_i where r_i is the outer boundary of the i-th annular region, in order to perform an efficient Gaussian quadrature in each sub-division, so that we can avoid the singularity in the integrand. That is, the argument λ_i vanishes as ρ approaches r_i and this causes the integrand to have an undefined derivative at this point. The efficiency of the Gaussian quadrature is shown by an example that the 10-point Gaussian integration for $(r^2-\rho^2)^{1/2}$ gives the area of a circle by an accuracy of 0.1%.

For the two-dimensional cell of complex geometry which includes a number of pin rods where the same situation occurs as in a one-dimensional cell, we, however, have no means than to apply the trapezoidal rule. We know that the finer interval of $\Delta\rho$ and $\Delta\varphi$ gives the more accurate results.

Implementation of this integration scheme requires the development of a tracing routine to calculate the intersections traversed by the line of integration. To maximize the computing efficiency, specialized routines are prepared for a variety of geometries which has been shown in II.3.

The calculation of collision probabilities is performed in two steps. First, the tracing routine is used to get the geometrical information called "trace table" by each line and accumulate on a large scale storage, say, disc. In the second step, these data, together with the cross sections, are used to perform the integration of collision probabilities. The second step is repeated for every energy group.

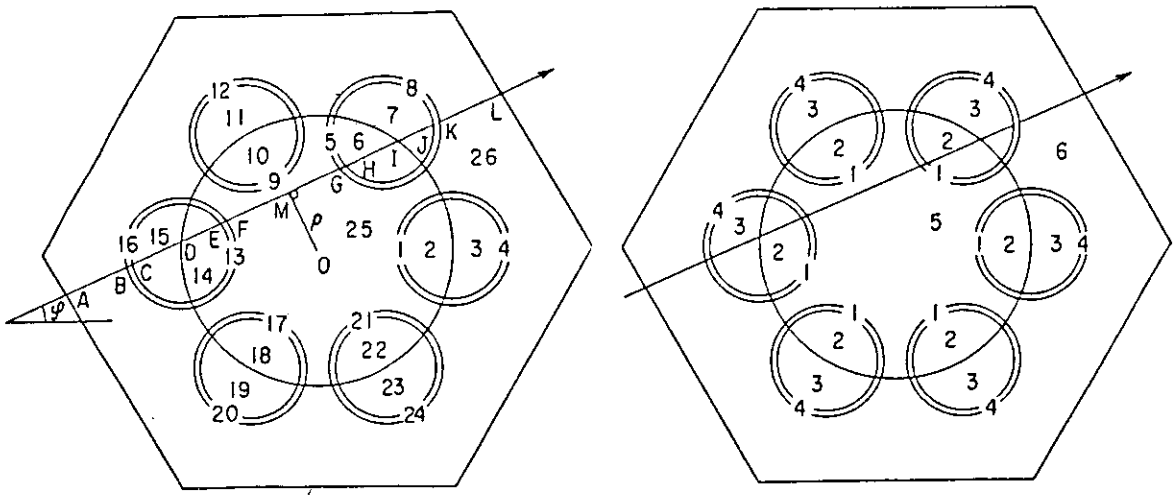
VI.1.6.1 How to compose trace table

We shall describe how to compose the "trace table" using a sample geometry which is a hexagonal cell including six fuel rods equally spaced on a circular ring, as shown in Fig.VI.1-6a in which the purely geometrical region (S-region) numbers are indicated and the corresponding physical region (T-region) numbers in Fig.VI.1.6b. The latters are treated as the spatial variables after considering any symmetric condition. The rods consist of two concentric layers and they, together with the coolant, are divided further by the circle through the centers of the rods into inner and outer regions. Hence a pin rod are divided into four regions. Since each rod is discriminated in the primary stage, the number of regions for rods amounts to 24.
The inner and outer coolant regions are assigned by the succeeding numbers; 25 and 26.

A neutron line defined by a pair of (ρ,φ) is drawn across the cell where ρ is the signed distance OM from the center of the cell, O to the line, and φ is the angle made by this line and the horizontal line. The line intersects with the region boundaries at A,B,C,.....L. The point

M (OM $\perp$ AL) is the origin of the measure on the line AL.

The preliminary trace table consists of an array T of which element is the signed distance from the point M to each intersecting point, an integer array IR for the region number of the right hand side of the point, and another array IL for the region number of the left hand side of the point.



a. S-region map b. T-region map

Fig.VI.1-6 A hexagonal cell with six pin rods on a circle crossed by a line defined by a pair of parameters (ρ, φ)

The computing process is organized to find, first, the intersections A and L on the hexagonal boundary, then D and I on the circle through the rods centers. At this step, the preliminary trace table is made up as shown in Table VI.1-1.

Table VI.1-1 Preliminary trace table after scanning annular regions

Position	1	2	3	4
T	l_{AM}	l_{DM}	l_{MI}	l_{ML}
IR	26	25	26	-
IL	-	26	25	26

The next step is to scan whether the line intersects or not with any rod. This scanning is done, first, on the outermost radius of a rod. If judged to intersect, the information for the two crossing points with the circle is inserted into the table by a routine "INSERT" which finds the positions in the table for the new two points and rearranges the arrays T, IR and IL so that the elements may be in the geometrical order.

After the insertion, then, the scanning on the inner radius of the same pin rod is continued until the innermost radius if judged to intersect, otherwise the scanning is transferred to the next pin rod. After scanning all pin rods, the preliminary trace table is organized for the neutron line illustrated in Fig.VI.1.6a as follows:

Table VI.1-2 Preliminary trace table after scanning all pin rods

	1	2	3	4	5	6	7	8	9	10	11	12
T	l_{AM}	l_{BM}	l_{CM}	l_{DM}	l_{EM}	l_{FM}	l_{MG}	l_{MH}	l_{MI}	l_{MJ}	l_{MK}	l_{ML}
IR	26	16	15	14	13	25	5	6	7	8	26	-
IL	-	26	16	15	14	13	25	5	6	7	8	26

The final table is further rearranged as;

1) Compose a new array II which keeps the T-region numbers reduced from the array IR which keeps S-region numbers using the correspondence specified in the input. Because of the 60° rotational symmetry, the common T-region numbers are assigned to each pin rod region.
2) Replace the element of T for the distance between the intersecting point and the point M by the distance between the two adjacent intersecting points.

Table VI.1-3 Final trace table at return step of geometry routine

	1	2	3	4	5	6	7	8	9	10	11
T	l_{AB}	l_{BC}	l_{CD}	l_{DE}	l_{EF}	l_{FG}	l_{GH}	l_{HI}	l_{IJ}	l_{JK}	l_{KL}
II	6	4	3	2	1	5	1	2	3	4	6

The information obtained by the geometry routine is the common length of arrays, LO, the arrays T and II and the modified ρ value; ρ' measured from the center of the adjacent cell which will be traced next if the perfect reflective boundary (mirror) condition is selected.

For the case of the mirror condition, the root routine calls repeatedly the same geometry routine with the arguments (ρ' and φ) until the fixed number of cells are traced, and concatenates the arrays T and II, respectively.

Finally, a record per line keeps the following information:
W, LO, LLL, (T(k),k=1,2,...,LLL), (II(k),k=1,2,...,LLL)
where W is the weight of a line given by

$$W = 1 \quad : \text{slab,} \tag{VI.1-73a}$$

$$W = \omega_g(r_i - r_{i-1}) \quad \text{for } r_{i-1}<\rho<r_i \quad : \text{cylindrier,} \tag{VI.1-73b}$$

$$W = \rho*\omega_g(r_i - r_{i-1}) \quad \text{for } r_{i-1} < \rho<r_i \quad : \text{sphere,} \tag{VI.1-73c}$$

$$W = \delta\rho*\delta\varphi = constant \quad : \text{two-dimensional cell,} \tag{VI.1-73d}$$

ω_g is the weight for the Gaussian quadrature at the point r_g in the range between r_{i-1} and r_i, and

LO : number of intersects of the source cell,
LLL : total number of intersects on a line (Unless the mirror or periodic boundary condition is taken, LLL = LO),
T(k) : distance between intersects which produces the optical thickness λ_k by multiplying the macroscopic cross section of the region II(k),

II(k): region numbers of k-th intersect.

The total number of lines LCOUNT (records) required to achieve the integration varies widely depending on the complexity of the geometry, for example, LCOUNT = 2 for slab, = several tens for one-dimensional cylinder and sphere, and = a few thousands for the most complicated two-dimensional case.

After storing the trace tables for all lines, a numerical integration of region volumes and an array of integrated to exact volume ratios is printed as an indicator of the adequacy of the integration mesh. The numerical integration is performed by

$$V(II(k)) = V(II(k)) + W * T_k \quad \text{for } k=1,2,\ldots,L0 \quad \text{and accumulated by line.} \qquad (VI.1\text{-}74)$$

The resultant array V gives the numerically integrated region volumes.

VI.1.6.2 Process for numerical integration

Being given the cross sections of an energy group, the integration of collision probabilities is performed line by line. Actually, the symmetric element Δ_{ij} $(=V_iP_{ij}/\Sigma_j)$ and $\Delta_{is}=V_iP_{is}$ are integrated instead of P_{ij} and P_{is}, respectively.

We shall show the computer process repeated by each line in the computational flow diagram shown in Fig.VI.1-7. The function $F(\lambda)$ appearing in the integrands in Fig.VI.1.7 is given by

$$F(\lambda) = E_{i3}(\lambda) \quad \text{for slab,} \qquad (VI.1\text{-}75a)$$

$$F(\lambda) = \exp(-\lambda) \quad \text{for sphere,} \qquad (VI.1\text{-}75b)$$

$$F(\lambda) = K_{i3}(\lambda) \quad \text{for one- and two-dimensional cylinder.} \qquad (VI.1\text{-}75c)$$

Note that among four terms appearing in the expression of $\Delta_{ij}(\rho,\varphi)$, the first two terms have been calculated as the last two terms of the previous k'. The calculation of $\Delta_{ij}(\rho,\varphi)$ in the geometrical order reduces the number of transcendental functions to be evaluated into half.

Care is taken when λ_i is so small compared with unity so that differential approximations are used, for example,

$$K_{i3}(\lambda_{ij})-K_{i3}(\lambda_{ij}+\lambda_i) \fallingdotseq \lambda_i * K_{i2}(\lambda_{ij}) \qquad (VI.1\text{-}76)$$

On integrating Δ_{ij}, the symmetric relation, $\Delta_{ij}=\Delta_{ji}$ validates to eliminate the loop of the source region in the reverse direction or to reduce the range of angular integration into half. The simple process to replace the off-diagonal element Δ_{ij} by $(\Delta_{ij}+\Delta_{ji})/2$ covers the above saving.

Information contained in a record;
W, L0, LLL, (T(k),k=1,2,...,LLL), (II(k),k=1,2,...,LLL)

The optical length λ_k is calculated by,
λ_k = T_k * Cross section of II(k)
for k=1,2,..,LLL

------- Loop of source region repeated for k=1,2,...,L0
: i = II(k) ; source region number
: $\lambda_i = \lambda_k$; optical thickness of source region
: Set $\lambda_{ij} = 0$; optical distance between source
: region and collision region
: Contribution to the diagonal element
: $\Delta_{ii}(\rho,\varphi) = \{F(0)*\tau_i - F(0) + F(\tau_i)\} / \Sigma_i \Sigma_i$
:
: $\Delta_{ii} = \int W * \Delta_{ii}(\rho,\varphi) d\rho d\varphi$
:
: ---- Loop of collision region repeated for k'=k+1,...,LLL
: : j = II(k') ; collision region number
: : $\lambda_j = \lambda_{k'}$; optical thickness of collision region
: : Contribution to the off-diagonal element
: : $\Delta_{ij}(\rho,\varphi) = \{F(\tau_{ij}) - F(\tau_{ij}+\tau_i) - F(\tau_{ij}+\tau_j) + F(\tau_{ij}+\tau_i+\tau_j)\} / \Sigma_i \Sigma_j$
: :
: : $\Delta_{ij} = \int W * \Delta_{ij}(\rho,\varphi) d\rho d\varphi$
: :
: : Prepare optical distance for the next k'
: : $\lambda_{ij} + \lambda_j \rightarrow \lambda_{ij}$
: ---- End loop for collision region k'
: unless the mirror condition (LLL > L0), then
: Contribution to the escape element
: $\Delta_{is}(\rho,\varphi) = \{F(\tau_{ij}) - F(\tau_{ij}+\tau_i)\} / \Sigma_i$
:
: $\Delta_{is} = \int W * \Delta_{is}(\rho,\varphi) d\rho d\varphi$
:
:------ End loop for source region k

Fig.VI.1.7 Computational flow-diagram for numerical integration by "Ray-Trace" method

A normalization so that the sum of P_{ij} over j be unity is effective to reduce the error caused by coarse integration mesh, and also by truncated optical distance which is terminated by the fixed number of cells to be traced by neutron line for the perfect reflective boundary condition.

The numerical calculation of K_{in} functions has yet to be explained. Although some rational approximations are developed for the Bickley-Naylor functions[20], they would be very time consuming because they have to be used so frequently as $10^6 \sim 10^7$ times. In the SRAC, a quadratic interpolation is performed numerically by using tables of **a**, **b**, **c** ; the coefficients of three terms for the quadratic expression of the Bickley-Naylor function. These tables list **a**, **b** and **c** as a function of x and n, where

$$a_m = \frac{y_{m-1} - 2y_{m-1/2} + y_m}{2\Delta x^2} , \qquad (VI.1\text{-}77a)$$

$$b_m = \frac{y_m - y_{m-1}}{\Delta x} - a_m (x_{m-1} + x_m) \quad , \qquad (VI.1\text{-}77b)$$

$$c_m = y_{m-1} - b_m x_{m-1} - a_m x_{m-1}^2 \quad , \qquad (VI.1\text{-}77c)$$

$$y_m = K_{in}(x_m) \quad , \qquad (VI.1\text{-}78)$$

$$\Delta x = (x_m - x_{m-1}) \quad . \qquad (VI.1\text{-}79)$$

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,

in the range x>9.6, K_{in} (x) is set to be zero, while the practical usage assumes K_{in} (x) vanishes if x>6.0.

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

$$K_{in}(x) = (a_{nm}x + b_{nm})x + c_{nm} \quad , \qquad (VI.1\text{-}80)$$

where $x_{m-1} \leqq x \leqq x_m$.

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

VI.2 Optional Processes for Microscopic Total and Macroscopic Transport Cross Sections

This section is concerned with the options for the specification of the input data IC15 and IC16 in Sect.II.1. As these are related with transport property, care should be taken in processing moderating material. Discussions will be made on the optional treatment for microscopic total cross section and for macroscopic transport cross sections used in isotropic transport calculation and in evaluating diffusion coefficient.

Microscopic 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 depend on all the compositions concerned.

** IC15 = 1 **

One way is based on the concept of the Bondarenko[21] type cross section set, i.e.,

$$\sigma_{t,g} = \sigma_{t\infty,g} \cdot f_{t,g}(\sigma_0), \tag{VI.2-1}$$

where the subscript g stands for the group number, and $f_t(\sigma_0)$ is the self-shielding factor for the total cross section. The total cross section thus defined is efficient to maintain the diffusion coefficient evaluated in ultra-fine group structure in the multi-group constants.

** IC15 = 2 **

On another way, the total cross section is defined as the summation of all partial reaction cross sections, i.e.,

$$\sigma_{t,g} = \sum_z \sigma_{z\infty,g} \cdot f_{z,g}(\sigma_0), \tag{VI.2-2}$$

where $f_{z,g}(\sigma_0)$ is the self-shielding factor for reaction type z.

Macroscopic transport cross section

We start with the P_1 equations in multigroup representation[22]. The P_1 multigroup equations can be written as

$$\nabla \cdot J_g(r) + \Sigma_{t,g}\Phi_g(r) = \sum_{g'} \Sigma_{s0,g'\to g}\Phi_{g'}(r) + Q_g(r) \tag{VI.2-3}$$

$$\nabla\Phi_g(r) + 3\,\Sigma_{t,g}J_g(r) = 3\sum_{g'} \Sigma_{s1,g'\to g}J_{g'}(r) \tag{VI.2-4}$$

If a group-dependent form of Fick's law is postulated, i.e.,

$$J_g(r) = -D_g(r)\nabla\Phi_g(r), \tag{VI.2-5}$$

then from Eq.(VI.2-4) the diffusion coefficient can be formally expressed by

$$D_g(\boldsymbol{r}) = \frac{1}{3\Sigma_{tr,g}(\boldsymbol{r})} \qquad (VI.2\text{-}6)$$

with

$$\Sigma_{tr,g}(\boldsymbol{r}) \equiv \Sigma_{t,g} - \sum_{g'} \Sigma_{s1,g'\to g} J_{g'}(\boldsymbol{r})/J_g(\boldsymbol{r}). \qquad (VI.2\text{-}7)$$

If the group width is wider compared with the maximum energy degradation of neutron by elastic collision and the quantity $EJ(\boldsymbol{r}, E)$ can be assumed to be energy-independent in the width, we can show[22]

$$\sum_{g'} \Sigma_{s1,g'\to g} J_{g'}(\boldsymbol{r})/J_g(\boldsymbol{r}) \sim \sum_{g'} \Sigma_{s1,g\to g'} = (\bar{\mu})_g \Sigma_{s,g} \qquad (VI.2\text{-}8)$$

in this case, we have

$$\Sigma_{tr,g}(\boldsymbol{r}) = \Sigma_{t,g} - \sum_{g'} \Sigma_{s1,g\to g'} \qquad (VI.2\text{-}9)$$

Four methods are prepared to define the transport cross section to be used as total cross sections in the isotropic transport calculation or diffusion calculation.

** IC16 = 0 **

The first one treats simply the anisotropic scattering effect in the isotropic transport calculation by the extended transport approximation.[23] Here, the macroscopic transport cross section is defined by

$$\Sigma_{tr,g} = \Sigma_{t,g} - \sum_{g'} \Sigma_{s1,g'\to g}, \qquad (VI.2\text{-}10)$$

It should be noted that 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[24]. By choice of the option IC16 = 1 or 2, the system under study is, first, homogenized by simple smearing of the atomic number densities in the cell, if IC2>0 is specified, and the P_1 or B_1 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 material is calculated by

$$\Sigma_{tr,g,M} = \gamma_g \Sigma_{t,g,M} - \sum_{g'} \Sigma_{s1,g'\to g,M} \bar{\Phi}_{1,g'}/\bar{\Phi}_{1,g} \qquad (VI.2\text{-}11)$$

where

$$\gamma_g = 1 \quad : \quad \text{for} \quad P_1 \text{ approximation} \tag{VI.2-12a}$$

$$\gamma_g = \alpha_g tan^{-1}\alpha_g / [\ 3\{1-(tan^{-1}\alpha_g)/\alpha_g\}\ ] :$$

$$\text{for } B_1 \text{ approx. with } \alpha_g^2 = B^2/\Sigma_{t,g}^2, \text{ if } B^2>0 \tag{VI.2-12b}$$

$$\gamma_g = \alpha_g tanh^{-1}\alpha_g / [\ 3\{(tanh^{-1}\alpha_g)/\alpha_g - 1\}\ ] :$$

$$\text{for } B_1 \text{ approx. with } \alpha_g^2 = -B^2/\Sigma_{t,g}^2, \text{ if } B^2<0. \tag{VI.2-12c}$$

Here, $\bar{\Phi}_{0,g}$ and $\bar{\Phi}_{1,g}$ is, respectively, the P_0 and P_1 component of the angular flux, and B^2 is the geometrical buckling entered in BLOCK 4 of II.1. This approach is recommended for the calculation of the lattice cell or homogeneous medium including hydrogeneous materials.

For an isolated material not used in the cell calculation or a new material used with an 'old' material which has already transport cross sections, the above equation is solved in a system composed purely by the material.

** 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 defined, following to Eq. (VI.2-7), by

$$\Sigma_{tr,g}(r) = \Sigma_{t,g} - \sum_{g'} \Sigma_{s1,g'\to g}\ \Phi_{1,g'}(r)/\Phi_{1,g}(r) \tag{VI.2-13}$$

This quantity is averaged over each volume or region in the system under consideration to give the representative value

VI.3 Optional Processes for Resonance Absorption

Concerning thermal and intermediate 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 fixed to 130.07 eV in the SRAC system, as previously mentioned. In this resonance energy range (called the second resonance range), three optional methods are available; the first one is based on a table-look-up method of resonance shielding factors, assuming the NRA (narrow resonance approximation) to the slowing down due to moderators, while the IRA (intermediate resonance approximation) is assumed for the slowing down in the second one. In the third method, the effective cross sections are exactly calculated by solving neutron slowing-down equation in heterogeneous system. The choice of the methods depends on the input data specification of the user.

On the first step of these calculations, the whole energy range is dealt with the table-look-up method assuming the NRA for moderator slowing-down. On the next step, the effective cross sections in the second resonance energy range are replaced by the more accurate ones based on the IRA or direct numerical method. Hence, in the higher energy range above 130.07eV, available is only the first method.

In the followings, descriptions will be given to the methods which are incorporated in the SRAC system for calculating the effective resonance cross sections.

VI.3.1 Table-look-up method of f-tables based on NR approximation in homogeneous systems

At first, consider an infinite homogeneous mixture consisting of one kind of resonance nuclide and of nonabsorbing moderator. When the NRA is applied for the slowing down due to the moderator nuclide, the slowing down equation can be written as

$$(\sigma_t(u) + \sigma_b)\varphi(u) = K(\sigma_s\varphi) + \sigma_b \qquad (VI.3.1\text{-}1)$$

with

$$\sigma_t(u) = \sigma_a(u) + \sigma_s(u) = \sigma_r(u) + \sigma_p \qquad (VI.3.1\text{-}2)$$

where $\varphi(u)$ is flux per unit lethargy, K, the slowing down operator, $\sigma_a(u)$,$\sigma_s(u)$, microscopic absorption and scattering cross sections of the resonance absorber, respectively; $\sigma_r(u)$ and σ_p, the resonance part of total cross section $\sigma_t(u)$ and the potential scattering cross section, respectively, and σ_b is the scattering cross section of the moderator.

Equation (VI.3.1-1) is just the basic one that is used in the construction of the SRAC library for heavy resonance nuclides. That is, Eq.(VI.3.1-1) with the Doppler broadened cross sections by temperature T is rigorously solved with ultra-fine meshes for a given σ_b by a revised version of the TIMS-1 code[25]. The calculated effective cross sections are arranged by the infinite dilute cross sections and the self-shielding factors $f(\sigma_b,T)$ tabulated by two parameters σ_b and T. Let us describe the way to use the shielding factors in practical situations.

For a special case where the NRA is applicable to the slowing down

of absorber, the first-order solution can be written in the conventional form

$$\varphi(u) \sim \frac{\sigma_p + \sigma_b}{\sigma_t(u) + \sigma_b} \qquad (VI.3.1\text{-}3)$$

This first-order solution is usually adopted to construct a cross section set of the Bondarenko type[21] with the resonance shielding factors for use in fast reactor analysis. For the higher energy region, say E > 130 eV, the NRA is considered to be a reasonable approximation for heavy nuclides. That is, the flux of Eq. (VI.3.1-3) does not so much deviate from the exact one of Eq. (VI.3.1-1) in the meaning of the weighting function for cross section averaging. On the other hand, for the light and intermediate mass nuclides with resonance structure, the SRAC library has been generated assuming the spectrum of Eq. (VI.3.1-3). In fact, no simple and convenient spectrum has been proposed to treat these nuclides. Since the resonance structure of light and intermediate mass nuclide is of minor importance in thermal and intermediate reactors, the present treatment will in practice be sufficient. Hence, we can think of Eq. (VI.3.1-3) as representing a standard form of weighting spectrum for the higher energy regions.

For a homogeneous system including many moderator nuclides, the slowing down equation can be written as

$$\{\sigma_t(u) + \sigma_m\}\varphi(u) = K(\sigma_s\varphi) + \frac{1}{n_f}\sum_j n_j\sigma_j K_j(\varphi) \qquad (VI.3.1\text{-}4)$$

where $$\sigma_m = \frac{1}{n_f}\sum_j n_j\sigma_j \qquad (VI.3.1\text{-}5)$$

and K_j is the slowing down kernel for moderator j; n_f and n_j are the atomic number density of absorber and moderator nuclides, respectively.

When the NRA is applicable to the slowing down of moderators (referred to as NRA moderator), we have

$$(\sigma_t(u) + \sigma_m)\varphi(u) = K(\sigma_s\varphi) + \sigma_m \qquad (VI.3.1\text{-}6)$$

Consequently, a homogeneous system with the NRA moderators has the same effective cross section as the homogeneous system described by Eq. (VI.3.1-1). That is, the effective cross sections can be calculated by calculating σ_m as to be used for the table-look-up of the resonance shielding tables. Here, it should be noted that the slowing down of absorber is still accurately estimated in the present treatment.

VI.3.2 Table-look-up method of f-tables based on IR approximation

We start with the IRA of resonance absorption in homogeneous systems, to give an insight into the relationship with the table-look-up method.

From the two extreme cases representing the limits of NR and wide resonance (WR) for the slowing down kernel, the first-order solution for $\varphi(u)$ of Eq. (VI.3.1-1) can be written as[26),27)]

$$\varphi(u) \sim \frac{\lambda\sigma_p + \sigma_b}{\sigma_a + \lambda\sigma_s + \sigma_b} \qquad (VI.3.2\text{-}1)$$

where λ is the IRA parameter for the absorber. The value of λ can be determined by solving a transcendental equation for λ[26),27)].

For a homogeneous system including many moderator nuclides described by (VI.3.1-4), the corresponding first-order solution can be given by

$$\varphi(u) \sim \frac{\lambda\sigma_p + \sigma_b'}{\sigma_a + \lambda\sigma_s + \sigma_b'} \qquad (VI.3.2\text{-}2)$$

with $$\sigma_b' = \frac{1}{n_f}\sum_j \lambda_j (n_j\sigma_j), \qquad (VI.3.2\text{-}3)$$

where λ_j is the IR parameter for moderator j and can be again determined by solving a coupled set of transcendental equations[27].

Here, it should be noted that both the fluxes obtained from a numerical integration of Eq.(VI.3.1-1) and given by Eq.(VI.3.2-1) or (VI.3.2-2) 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.3.1-1). 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 case of letting all the λ_js equal to unity corresponds just to the NRA applied to the slowing down of moderators ($\sigma_b'=\sigma_m$), as discussed in the previous subsection.

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[28]. A simple way to take account of this dependence is to multiply the interference scattering term by a factor with temperature dependence[28].

Next consider the IR treatment of resonance absorption in heterogeneous systems. Assuming a flat flux in each spatial region, the slowing down equation in the two-region system, consisting of an absorbing lump (f) and a nonabsorbing moderator (m), can be written as

$$\sigma_f\varphi_f = p_{ff}\{\sigma_{cm}K_{cm}(\varphi_f) + K_f(\sigma_s\varphi_f)\} + (1-p_{ff})\sigma_f\sum_k\{R_kK_k(\varphi_m)\} \qquad (VI.3.2\text{-}4)$$

$$\sigma_f\varphi_f + \sigma_m\varphi_m = \sigma_{cm}K_{cm}(\varphi_f) + K_f(\sigma_s\varphi_f) + \sigma_m\sum_k\{R_kK_k(\varphi_m)\} \qquad (VI.3.2\text{-}5)$$

where

φ_f, φ_m = flux per unit lethargy in the lump and moderator region, respectively

$\sigma_f(u)=\sigma_a(u)+\sigma_s(u)+\sigma_{cm}$ = microscopic total cross section of the lump

$\sigma_{cm}=\Sigma_{cm}/n_f$ = 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} = collision probability in the fuel lump.

The other notation is conventional.

We make use of the simple interpolation formula for the collision probability as proposed by Wigner:

$$p_{ff} = \frac{X}{X + g(C)(1 - C)} = \frac{\sigma_f}{\sigma_f + s} \qquad (VI.3.2\text{-}6)$$

with $$X = l_f n_f \sigma_f = l_f \Sigma_f \qquad (VI.3.2\text{-}7)$$

$$s = g(C)(1 - C)/(l_f n_f) \quad \text{and} \quad g(C) = \frac{a}{1 + (a-1)C} \qquad (VI.3.2\text{-}8)$$

where l_f the lump mean chord length, C, the Dancoff factor and a is a purely geometrical quantity.

Generalized collision probability theory shows

$$p_{ff} \sim 1 - \frac{1 - C}{X} \quad \text{for} \quad X \to \infty, \qquad (VI.3.2\text{-}9)$$

where the Dancoff factor, C, is zero for isolated lumps. Being based on Eq. (VI.3.2-9), the Dancoff factor C is calculated by using the value of p_{ff} for a sufficiently large value of Σ_f, as $\Sigma_f = 300 cm^{-1}$

$$1 - C = \{1 - p_{ff}(\Sigma_f)\} X \,|_{\Sigma_f \to \infty} . \qquad (VI.3.2\text{-}10)$$

A rational interpolation of p_{ff} leads to g(C) = 1 in Eq. (VI.3.2-6) when use is made only of the behavior of p_{ff} at $\Sigma_f = \infty$ given by Eq. (VI.3.2-9). 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[29),30),31)]. Since the quantity, a, usually referred to as the Bell or Levine factor , somewhat fluctuates among resonances, there might be some minor problems with the choice. The exact choice of this quantity is not thought to be important, considering from the results of many studies done in this field[31)]. The values adopted in the SRAC system for the geometric quantities are shown in the following table:

Table VI.3.2-1 Levine or Bell Factors

Geometry	l_f	a	Remarks
Sphere of Radius r	$4r/3$	1.4	Ref. (5,32)
Slab of Thickness r	$2r$	1.2	Ref. (5,33)
Infinite Cylinder of Radius r	$2r$	1.2	Ref. (29,30)
Infinite Hollow Cylinder of Inner Radius a and Outer Radius b, $\sin\theta$=a/b	$2r\cos^2\theta$	1.2	Ref. (32,34)

The Dancoff factor and associated quantities will be in more general form discussed for multiregion problems in the next subsection.

Substituting Eq.(VI.3.2-6) into Eq.(VI.3.2-4) and subtracting Eq.(VI.3.2-5) from the resulting equation, we obtain the following set of equations for neutron balance:

$$(\sigma_f + s)\varphi_f = \sigma_{cm}K_{cm}(\varphi_f) + K_f(\sigma_s\varphi_f) + s\sum_k \{R_kK_k(\varphi_m)\} \qquad (VI.3.2\text{-}11)$$

$$s\varphi_f = \sigma_m\varphi_m + (s-\sigma_m)\sum_k \{R_kK_k(\varphi_m)\} \qquad (VI.3.2\text{-}12)$$

Then, from the two extreme cases representing the limits of NR and WR, respectively, for the slowing down kernels, the first-order solution for φ_f and φ_m can be written as[35]

$$\varphi_f^{(1)}(u) = \frac{\lambda\sigma_p + \kappa\sigma_{cm} + \mu^* s}{\sigma_a + \lambda\sigma_s + \kappa\sigma_{cm} + \mu^* s} \qquad (VI.3.2\text{-}13)$$

$$\varphi_m^{(1)}(u) = 1 - s\{1-\varphi_f^{(1)}(u)\}/\{\mu\sigma_m + (1-\mu)s\} \qquad (VI.3.2\text{-}14)$$

with

$$\mu^* = \mu\sigma_m/\{\mu\sigma_m + (1-\mu)s\} \quad \text{and} \quad \mu = \sum_k R_k\mu_k \quad , \qquad (VI.3.2\text{-}15)$$

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[35].

Equation(VI.3.2-13) can be written in the standard form of Eq.(VI.3.2-1):

$$\varphi_f^{(1)}(u) = \frac{\lambda\sigma_p + \sigma_b}{\sigma_a + \lambda\sigma_s + \sigma_b} \qquad (VI.3.2\text{-}16)$$

where

$$\sigma_b = \kappa\sigma_{cm} + \mu^* s$$

$$= \frac{\kappa\Sigma_{cm}}{n_f} + \frac{\mu^* q(C)(1-C)}{(n_f l_f)} \qquad (VI.3.2\text{-}17)$$

Particularly for the NRA (κ=1, μ^*=1), the above equation can generally be written as

$$\sigma_b = \sigma_{cm} + s = \frac{1}{n_f}\sum_{j\neq f}(n_j\sigma_j) + \frac{g(C)(1-C)}{(n_f l_f)} \qquad (VI.3.2\text{-}18)$$

The simple result of Eqs.(VI.3.2-17) and (VI.3.2-18) again reveals the following equivalence relation with practical usefulness : A heterogeneous system with σ_b has the same effective resonance cross section as a homogeneous system of Eq.(VI.3.1-1) 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[21] which is calculated based on Eq.(VI.3.1-1).

Several groups of the SRAC multigroup structure around 100 eV include more than one resonance levels of ^{238}U. Since the background cross section σ_b somewhat fluctuates among the resonance levels, the arithmetic average value of σ_b's in each group is in practice taken as the background cross section for the table-look-up. In the SRAC system, the IR method is applied only to ^{238}U, ^{232}Th, and ^{240}Pu.

VI.3.3 Generalized Dancoff factor

The effective resonance cross sections in heterogeneous systems have been calculated by combined use of an equivalence theorem[36] and a cross section set of the Bondarenko type[21]. The equivalence theorem between homogeneous and heterogeneous systems is usually established through a Dancoff factor. This factor has been derived under the assumption that the system under study consists of an infinite array of two-region cells. Meneghetti[37] derived a generalized Dancoff factor for a more general situation of infinite array of multi-region plate cell. In a complex situation as encountered in the lattice of the PROTEUS-LWHCR (Light Water High Converter Reactor) cores[38] where MOX pin and depleted U pin are alternatively placed in a hexagonal array, however, it is generally difficult to see how one can construct a unit cell for which the Dancoff factor can be calculated. A generalized Dancoff factor was derived for infinite array of multi-region cells including several absorber lumps with different absorber concentration[33]. The equivalence theorem based on this Dancoff factor is incorporated in the SRAC system.

The neutron slowing-down equation in a cell may be written by using the collision probabilities

$$\varphi_i(u) = \sum_j P_{ij}(u) W_j(u) X_{oj}/X_j(u) \qquad (VI.3.3\text{-}1)$$

$$W_j(u) = S_j(u)/\Sigma_{oj}, \qquad (VI.3.3\text{-}2)$$

$$X_{oj}(u) = \overline{l}_j \Sigma_{oj} \qquad X_j = \overline{l}_j \Sigma_j(u), \qquad (VI.3.3\text{-}3)$$

where the subscript j denotes a spatial region j, $S_j(u)$ the slowing-down source, $\overline{l}_j$, the mean chord length, Σ_{oj}, the non-resonance part of $\Sigma_j(u)$ and the other notation is conventional.

Now, let us consider the limit at which the resonance cross section of one resonant isotope, say σ_t, tends to be infinite. This black limit corresponds to a physical situation encountered near a resonance energy. Then, all the macroscopic cross sections of the region with the resonance isotope under consideration will also tend to be infinite. We denote these regions by the symbol R.

General arguments on asymptotic behaviors of the collision probability, P_{ij}, at the black limit[5),39),40] show

$$P_{ij}(\sigma_t) \equiv P_{ij}(u) \Rightarrow \delta_{ij} - \gamma_{ij}/X_i \quad (i \in R) \quad for\ X_i \Rightarrow \infty \qquad (VI.3.3\text{-}4)$$

$$\text{with} \quad \gamma_{ij} \equiv \lim_{\sigma_t \to \infty} \{\delta_{ij} - P_{ij}(\sigma_t)\} X_i. \qquad (VI.3.3\text{-}5)$$

From the conservation law

$$\sum_j P_{ij} \equiv 1, \qquad (VI.3.3\text{-}6)$$

we have

$$\sum_j \gamma_{ij} \equiv 0 \quad \text{or} \quad \gamma_{ii} = -\sum_{j \neq i} \gamma_{ij}. \qquad (VI.3.3\text{-}7)$$

Therefore, we have for the flux $\varphi_i(u)(i \in R)$

$$\varphi_i(u) \Rightarrow (W_i^{\infty} X_{oi} - \sum_{j \in R} W_j^{\infty} X_{oj} \gamma_{ij}/X_j)/X_i + O(X_i^{-2}) \quad for\ X_i \Rightarrow \infty \qquad (VI.3.3\text{-}8)$$

$$W_j^{\infty} \equiv \lim_{\sigma_t \to \infty} W_j \ . \qquad (VI.3.3\text{-}9)$$

Here, since we try to treat the higher energy region, say E ≧ several hundred eV, the NRA is a reasonable approximation, *i.e.*,

$$W_j(u) = const. \equiv 1. \qquad (VI.3.3\text{-}10)$$

Moreover, we assume that accidental overlapping between different resonance sequences is negligible, *i.e.*,

$$X_{oj}/X_j = 1 \quad (j \in R). \qquad (VI.3.3\text{-}11)$$

Under these assumptions, Eq. (VI.3.3-8) can be rewritten as

$$\varphi_i(u) \Rightarrow (X_{oi} - \sum_{j \in R} \gamma_{ij})/X_i \qquad for\ X_i \Rightarrow \infty. \qquad (VI.3.3\text{-}12)$$

On the other hand, an equivalent theorem between homogeneous and heterogeneous systems means that the flux in the absorber lump should be expressed by the spectrum in homogeneous medium

$$\varphi_i(u) = \frac{X_{oi} + b_i}{X_i + b_i} \Rightarrow (X_{oi} + b_i)/X_i \qquad for\ X_i \Rightarrow \infty, \qquad (VI.3.3\text{-}13)$$

where the flux is normalized to be unity at off-resonance energy.

For Eqs. (VI.3.3-12) and (VI.3.3-13) to be held, the following identity must be satisfied:

$$b_i = -\sum_{j \in R} \gamma_{ij} \ . \qquad (VI.3.3\text{-}14)$$

For a special case where the unit cell under study consists of fuel region (f) and moderator region (m), we can prove the conventional relation[36]

$$b_i = -\gamma_{fm} = 1 - C, \qquad (VI.3.3\text{-}15)$$

where C is the Dancoff factor and $-\gamma_{fm}$ corresponds to the first-flight blackness for neutrons leaving the fuel region[36]. Figure VI.3.3-1 shows the expression of b_i for other simple geometries.

Consequently, from the above examples, Eq. (VI.3.3-14) is considered to be a generalization of the previous works introduced in the previous section. Equation (VI.3.3-14) shows that the Dancoff factor for complex geometry can be obtained by calculating the generalized blackness γ_{ij}. Here, the blackness γ_{ij} is calculated by using the collision probability package in the SRAC system, that is, by using the value of P_{ij} for a sufficiently large value of $\Sigma_j (j \in R)$, as Σ_j=300 cm^{-1}, based on Eq. (VI.3.3-5).

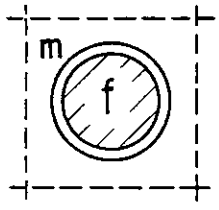
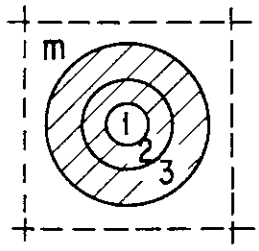
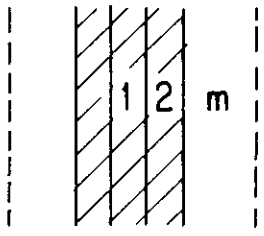
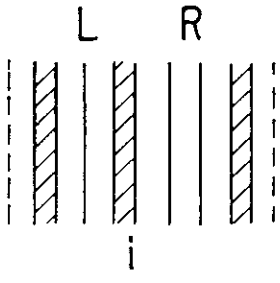
Geometry	b_i
	$b_f = -(\gamma_{fc} + \gamma_{fm}) > 0$
	$b_i = \begin{cases} 0 & i = 1 \\ 0 & i = 2 \\ \gamma_{33} = -\gamma_{3m} > 0 & i = 3 \end{cases}$
 (plate)	$b_i = \begin{cases} 0 & i = 1 \\ \gamma_{22} = -\sum_m \gamma_{2m} > 0 & \\ & i = 2 \end{cases}$
	$b_i = 1 - E_3(X_L) - E_3(X_R)$ $X \equiv \sum_m d_m \Sigma_m$ Meneghetti's two-sided formula

Fig.VI.3.3-1 Dancoff factors for simple geometries

The symbol (m) means moderator region and the shaded portions show the R-region: The quantites d_m stands for the width of moderator plate m.

The equation (VI.3.3-13) can be rewritten in a standard form of weighting spectrum

$$\varphi_i(u) = \frac{\sigma_p + \sigma_{bi}}{\sigma_t(u) + \sigma_{bi}} \qquad (VI.3.3\text{-}16)$$

$$\sigma_{bi} = \frac{1}{n_i} \sum_{j \neq i} (n_j \sigma_j) + \frac{b_i}{(n_i l_i)} \qquad (VI.3.3\text{-}17)$$

where n_i and n_j are the atomic number densities of the absorber under consideration and of admixed moderators, respectively.

The standard spectrum of Eq.(VI.3.3-16) was obtained again from the behavior of the collision probability at $\sigma_t \Rightarrow \infty$. So we need some corrections for the present approach, as done in the previous section. For this purpose we at first define the generalized Dancoff factor of an absorber in the region i by

$$C_i = 1 - b_i \qquad (VI.3.3\text{-}18)$$

Then the two-region problem in the previous subsection suggests the replacement of $(1 - C_i)$ by $(1-C_i)g(C_i)$ with

$$g(C_i) = \frac{a}{1 + (a-1)C_i} \qquad (VI.3.3\text{-}19)$$

Accordingly we can generally define the background cross section, σ_{bi}, including the heterogeneity by

$$\sigma_{bi} = \frac{1}{n_i} \sum_{j \neq i} (n_j \sigma_j) + \frac{g(C_i)(1-C_i)}{(n_i l_i)}. \qquad (VI.3.3\text{-}20)$$

Here, the value of the Bell factor, a, of Eq.(VI.3.3-19) is assumed to take the respective value of Table VI.3.2-1 corresponding to the geometry under consideration.

We again obtain the equivalence relation: The effective cross sections of absorber nuclides in each region can be calculated by using a cross section set of the Bondarenko type.

In the first resonance energy range of the SRAC system ($E >=$ 130.07 eV), the effective cross sections are obtained by the table-look-up method of a Bondarenko type cross section set, where the heterogeneity is treated by the established equivalence relation of Eq.(VI.3.3-20) between heterogeneous and homogeneous mixtures. The cross sections in the second region (130.07 eV$\geqq E \geqq$ 0.414 eV) are generally calculated by the IRA or a direct numerical method using collision probability and ultra-fine groups(Δu= 0.00125), as described in the next subsection. Hence, the energy range concerning the present improvement is mainly the first resonance region.

As known through the present derivation, each resonant nuclide in one absorber lump may take a different value for the generalized Dancoff factor. One example of this type of problems will be seen in the reference[40]. Moreover, there might be a problem whether or not we should treat a region with small amount of resonance absorbers as the R-region; this problem would not be essential, since the effective cross sections in such a region should be nearly infinite dilution cross sections. Anyway, a complex heterogeneity can be treated consistently without introducing any simplification of geometry.

VI.3.4 Direct method for calculating neutron flux distribution (the method adopted in routine '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

$$V_i\Sigma_i(u)\Psi_i(u) = \sum_{j=1}^{J} P_{ji}(u)V_j \sum_{k=1}^{K} S_{jk}(u) \qquad (VI.3.4\text{-}1)$$

$$S_{jk}(u) = \frac{1}{1-\alpha_k}\int_{u-\varepsilon_k}^{u} \exp\{-(u-u')\}\Sigma_{sjk}(u')\Psi_j(u')du' \qquad (VI.3.4\text{-}2)$$

with

$$\alpha = (\frac{A-1}{A+1})^2 \quad \text{and} \quad \varepsilon = -\ln\alpha. \qquad (VI.3.4\text{-}3)$$

Here, the subscript i or j stands for the subregion number and the k corresponds to 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

$$\Sigma_i(u)\psi_i(u) = \sum_{j,k} P_{ji}(u)\, S^0{}_{jk}(u) \qquad (VI.3.4\text{-}4)$$

with

$$S^0{}_{ik}(u) = \frac{1}{1-\alpha_k}\int_{u-\varepsilon_k}^{u} F_{jk}(u')du' \qquad (VI.3.4\text{-}5)$$

$$F_{jk}(u) = \Sigma_{sjk}(u)\psi_j(u). \qquad (VI.3.4\text{-}6)$$

Here, note that the equations (VI.3.4-4) and (VI.3.4-5) for $\psi_i(u)$ is more simple than Eqs. (VI.3.4-1) and (VI.3.4-2).

For the computation of the neutron spectrum $\psi_i(u)$ on discrete lethargy meshes, we use the RABBLE Method developed by Kier[42),43)]. 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

$$\psi^m{}_i = \int_{u_0}^{u_+} \psi_i(u)du \qquad (VI.3.4\text{-}7)$$

$$F^m{}_{jk} = \int_{u_0}^{u_+} F_{jk}(u)du = \Sigma^m{}_{sjk}\psi^m{}_j \qquad (VI.3.4\text{-}8)$$

where u_+ and u_0 are upper and lower lethargy bounds, respectively, corresponding to the fine group m. Then, the integration of Eq. (VI.3.4-

4) over the lethargy range from u_0 to u_+ gives

$$\Sigma^m{}_i \psi^m{}_i = \sum_{j,k} P^m{}_{ji} Q^m{}_{jk} \qquad (VI.3.4\text{-}9)$$

where

$$Q^m{}_{jk} = \frac{1}{1-\alpha_k}\int_{u_0}^{u_+} du \int_{u-\varepsilon_k}^{u} F_{jk}(u')du' \qquad (VI.3.4\text{-}10)$$

$$\sim \frac{\Delta u_m}{1-\alpha_k}\int_{u_0-\varepsilon_k}^{u_0} F_{jk}(u)du$$

$$= Q^{m-1}{}_{jk} + \frac{\Delta u_m}{1-\alpha_k}\{F^m{}_{jk} - (F_{jk})^{m-L^m_k}\} \qquad (VI.3.4\text{-}11)$$

with

$$(F_{jk})^{m-L^m_k} = \int_{u_--\varepsilon_k}^{u_0-\varepsilon_k} F_{jk}(u)du. \qquad (VI.3.4\text{-}12)$$

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[44].

It is moreover assumed that the scattering rate given by Eq. (VI.3.4-12) can be approximated by using the intermediate group scattering rate[43]. 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[44]. 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.

The lattice cell under study may consist of several materials and each may be subdivided into several spatial regions. 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

$$\bar{\sigma}_l(E) = \frac{1}{N_l}\sum_k N_{lk}\sigma_k(E) \qquad (N_l \neq 0) \qquad (VI.3.4\text{-}13)$$

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 kth absorber, and N_l and N_{lk} are the atomic number densities of the absorber under consideration and of the k resonance absorber in the I-th material, respectively.

Some fuel materials may have the same relative densities N_{lk}/N_l, hence the same value for $\bar{\sigma}_l(E)$. In such materials, the value of $\bar{\sigma}_l(E)$ 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 material J belonging to the independent variable $\bar{\sigma}_J(E)$. Let us introduce new variables defined by

$$X_J = (N_J\bar{\sigma}_J + \Sigma_{0J})\bar{l}_J \qquad (J=1, \mathrm{JMAX},) \qquad (VI.3.4\text{-}14)$$

where JMAX is the number of the independent variables (JMAX $\leq$2) and $\bar{l}_J$ is a parameter with dimension of length introduced to define the nondimensional variable X_J and assumes 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.3.4-1) or (VI.3.4-5) can be considered to be a function of X_J (J=1, JMAX).

In the routine '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 the values of $X_1, X_2 \leqq 9$, $P_{ji}(X_1, X_2)$ needed for the interpolation are calculated by calling the routine 'PIJ2'. The second one uses the asymptotic expansion of $P_{ji}(X_1, X_2)$ when eitherof X_1 or $X_2 \geqq 9$.

(I) One resonance-absorbing composition problem (JMAX=1)

It is easy to show[5),40),45)]

$$P_{ij}(X) \rightarrow \eta_{ij} + \gamma_{ij}/X \qquad \text{for } X \rightarrow \infty \qquad (VI.3.4\text{-}15)$$

with

$$\eta_{ij} \equiv P_{ij}(X = \infty) \qquad (VI.3.4\text{-}16)$$

and

$$\gamma_{ij} \equiv \begin{cases} \{P_{ij}(X)-\eta_{ij}\}X|_{X\rightarrow\infty} & \text{if } i \in R \\ 0 & \text{otherwise,} \end{cases} \qquad (VI.3.4\text{-}17)$$

where R stands for the resonance-absorbing material.

The above equation (VI.3.4-15) is used for $X > 9$ and the generalized Dancoff factor γ_{ij} given by Eq.(VI.3.4-17) 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

$$Z \equiv \frac{X}{X+1} \quad \text{or} \quad X = \frac{Z}{1-Z} \ . \qquad (VI.3.4\text{-}18)$$

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 the increment δZ=0.1 and the Lagrangian three points interpolation formula[46)]

$$f(Z_0 + \delta Z) = \frac{1}{2}u(u-1)f_{-1} + (1-u^2)f_0 + \frac{1}{2}u(u+1)f_1 \qquad (VI.3.4\text{-}19)$$

with

$$u = \frac{\delta Z}{\Delta Z} \quad \text{and} \quad |u| \leqq 1, \tag{VI.3.4-20}$$

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.3.4-1

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 (JMAX=2)

We can prove also for the two resonance-absorbing mixtures, R_1,R_2

$$P_{ij}(X_1,X_2) \sim \begin{cases} P_{ij}(\infty\ ,X_2)-\gamma^1{}_{ij}(X_2)/X_1 & X_1>9,\ X_2 \leqq 9 \\ \quad (\gamma^1{}_{ij}(X_2) \equiv 0 \quad \text{if } i \notin R_1) & \\ & \\ P_{ij}(X_1,\infty)-\gamma^2{}_{ij}(X_1)/X_2 & X_1 \leqq 9,\ X_2 >9 \\ \quad (\gamma^2{}_{ij}(X_1) \equiv 0 \quad \text{if } i \notin R_2) & \end{cases} \tag{VI.3.4-21}$$

$$P_{ij}(X_1,X_2) \sim \begin{cases} P_{ij}(\infty,\infty)-\gamma^{\infty}_{ij}/X_1 \quad (\text{if } i \in R_1 \quad X_1,X_2>9) \\ \\ P_{ij}(\infty,\infty)-\gamma^{\infty}{}_{ij}/X_2 \quad (\text{if } i \in R_2 \quad X_1,X_2>9) \\ P_{ij}(\infty,\infty) \quad (\text{otherwise}). \end{cases} \tag{VI.3.4-22}$$

Here, P_{ij} and γ_{ij} in Eq.(VI.3.4-21) is 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

$$Z_I = \frac{X_I}{1+X_I} \quad \text{or} \quad X_I = \frac{Z_I}{1-Z_I} \qquad (I=1,2) \qquad (VI.3.4\text{-}23)$$

and $P_{ij}(X_1,X_2))$ is interpolated on the these variables by using one of the following formulae[46]

$$f(Z_{10}+\delta Z_1, Z_{20}+\delta Z_2)=\{1-\frac{1}{2}(3-u-v)(u+v)\}f_{00}$$
$$+(2-u-v)(uf_{10}+vf_{01})+uvf_{11}$$
$$+\frac{1}{2}u(u-1)f_{20}+\frac{1}{2}v(v-1)f_{02}\ , \qquad (VI.3.4\text{-}24)$$

with $u=\frac{\delta Z_1}{\delta Z}$, $v=\frac{\delta Z_2}{\delta Z}$, $|u| \leqq 1, |v| \leqq 1,$ $(VI.3.4\text{-}25)$

$$f(Z_{10}+\delta Z_1, Z_{20}+\delta Z_2)=\frac{1}{2}u(u-1)f_{-10}+\frac{1}{2}v(v-1)f_{0-1}$$
$$+(1-uv-u^2-v^2)f_{00}+\frac{1}{2}u(u-2v-1)f_{10}$$
$$+\frac{1}{2}v(v-2u+1)f_{01}+uvf_{11}, \qquad (VI.3.4\text{-}26)$$

$$f(Z_{10}+\delta Z_1, Z_{20}+\delta Z_2)$$
$$=(1-u)(1-v)f_{00}+u(1-v)f_{10}+v(1-u)f_{01}+uvf_{11}. \qquad (VI.3.4\text{-}27)$$

The choice of Eqs.(VI.3.4-24), (VI.3.4-26) or (VI.3.4-27) 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 probabilities

The first-flight collision probability must satisfy the two important relations, that is, the conservation law

$$\sum_j P_{ij} = 1 \qquad \text{for all } i \qquad (VI.3.4\text{-}28)$$

and the reciprocity relation[36]

$$V_i\Sigma_i P_{ij} = V_j\Sigma_j P_{ji} \quad \text{for all } i,j. \qquad (VI.3.4\text{-}29)$$

In the routine 'PEACO', at first, the values of P_{ij} are calculated only for $i \leqq j$ by Eqs.(VI.3.4-15) and (VI.3.4-27). Then, the collision probabilities satisfying Eqs.(VI.3.4-28) and (VI.3.4-29) are successively obtained by the following equations starting from i=1:

$$P_{ij}^* = \frac{1-\beta_1}{\beta_0} P_{ij} \ , \ P_{ji}^* = \frac{1-\beta_1}{\beta_0} P_{ji} \quad (j=i,J) \qquad (VI.3.4\text{-}30)$$

where

$$\beta_0 = \sum_{j=i}^{J} P_{ji} \quad \text{and} \quad \beta_1 = \sum_{i=1}^{j-1} P_{ij}^{*} \qquad (VI.3.4\text{-}31)$$

with

$$P_{ij} = V_j \Sigma_j P_{ji} / (V_i \Sigma_i) \qquad \text{for } j > i. \qquad (VI.3.4\text{-}32)$$

The collision probabilities P_{ij}^{*} given by Eq.(VI.3.4-30) will be readily known to satisfy Eqs.(VI.3.4-28) and (VI.3.4-29).

Using the interpolation and the asymptotic expansion, combined with the methods mentioned above, we can guarantee the accuracy of 0.1% for the calculation of the collision probability, including the one resonance-absorbing mixture 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.3.5 Resonance absorption in doubly heterogeneous system

An HTGR, currently under development at the JAERI, uses fuel in the form of small spherical coated particles. A coated particle consists of a fuel kernel of UO_2 with an $\sim$ 600 μm diameter and several layers of pyrolytic carbon and SiC of $\sim$ 300 μm thickness. Such coated particles, together with graphite diluent, are formed into hollow annular fuel pellets that are packed in a graphite sleeve, and then inserted into a graphite block (see Fig.VI.3.5-1).

One of the physical problems associated with this type of fuel is a double heterogeneity through the self-shielding of the grain and also of the lattice configuration of fuel rods on the resonance absorption.

Here, the "Accretion" method by Leslie & Jonsson[47] to calculate collision probabilities in a cluster-type fuel element is applied to evaluate the resonance absorption in the doubly heterogeneous system in the fuel block of the VHTR. The details will be found in the reference[48] together with some typical numerical examples.

We assume that the fuel grains in a fuel pellet are uniformly distributed so that each coated particle, together with graphite diluent, forms a two-region spherical cell (microscopic cell) containing only a fuel grain (region f) and the associated amount of graphite diluent (region m).

The neutron slowing-down equations can be written by using the collision probabilities under the assumption that the neutron flux is flat in each spatial region and the neutron scattering is isotropic and elastic:

$$V_j \Sigma_j(u) \varphi_j(u) = \sum_i P_{ij}(u) V_i S_i(u) \quad , \qquad (VI.3.5\text{-}1)$$

$$S_i(u) = \int_0^u \Sigma_i(u' \to u) \varphi_i(u') du' \quad , \qquad (VI.3.5\text{-}2)$$

where V_i, Σ_i, φ_i, $\Sigma_i(u' \to u)$, and S_i are the volume, the total cross section, the neutron flux, the differential scattering cross section, and the slowing down source of the ith region, respectively. The quantity P_{ij} is the collision probability that a neutron emitted in the

region i has the next collision in the region j, evaluated by assuming the flat flux in each region.

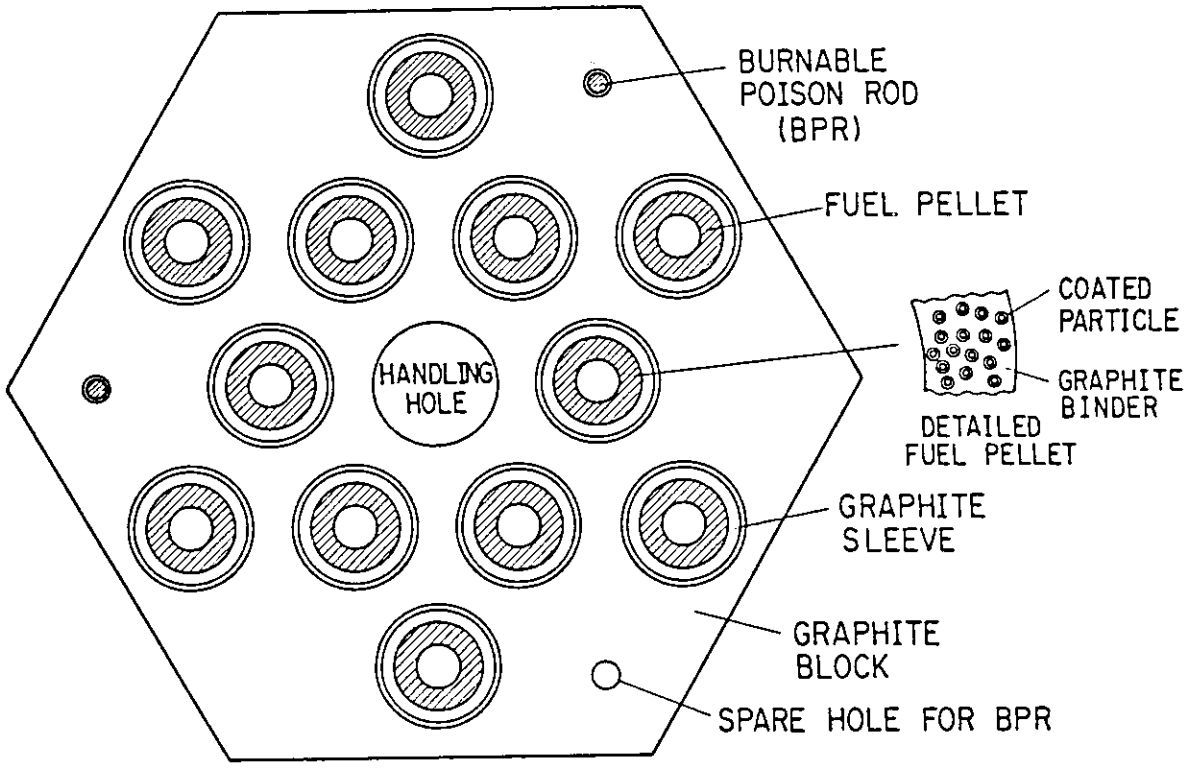


Fig.VI.3.5-1 Horizontal cross section of the standard fuel block of the VHTRC

When we consider the doubly heterogeneous case as seen in a VHTR block, each region must be specified as (l in I) to denote the microscopic region l (= f or m) in the macroscopic region I. Hence, the collision probability P_{ij} above defined takes the form, $P_{lJ}(l,k)$, where the index l indicates fuel grain f or diluent m in the macroscopic region I and the index k does fuel grain f or diluent m in the macroscopic region J. According to the derivation by Leslie & Jonsson[47], the off-diagonal element of the collision probability is given by

$$P_{lJ}(l,k) = \alpha_{lI}\ \alpha_{kJ}\ \frac{W_I}{W_{lI}}\ P_{IJ}\ , \qquad (VI.3.5\text{-}3)$$

and the diagonal element by

$$P_{lI}(l,k) = Q_I(l,k) - \frac{W_I}{W_{lI}}\ (1 - P_{II})\ \alpha_{lI}\ \alpha_{kI}\ , \qquad (VI.3.5\text{-}4)$$

where

$$W_{ll} = \Sigma_{ll} V_{ll}, \quad W_l = \Sigma_l V_l \quad \text{and} \quad \sum_l \alpha_{ll} = 1 \quad ,$$

α_{ll} , Σ_{ll} , and V_{ll} are the fraction of collision, the total cross section and the volume of the region (l in I), respectively, Σ_l and V_l are the corresponding quantities for the macroscopic region I. The quantity $Q_l(l,k)$ is the probability defined to an imaginary infinite lattice consisting of the microscopic cell of the region I, which can be evaluated numerically by the formalism presented in Appendix B of the reference[4] assuming a spherical cell with the white boundary condition. The quantity P_{IJ} denotes the collision probability between macroscopic regions I and J, which is assumed to be obtained using a homogenized cross section which substitutes for the heterogeneous fuel region.

If we can determine, independently of the macroscopic configuration of I, both of the quantity α_{ll} and the homogeneous-equivalent cross section Σ_l of the medium I, we can obtain $P_{IJ}(l,k)$ by the following procedure:

Now, we define the self-shielding factor f for the fuel grain so as to give an equivalent collision cross section to the macroscopic fuel region under the assumption of a uniform flux distribution through the microscopic cell. That is to say, the fraction of collision rate α_l in the region l is assumed to be given by the effective collision cross section Σ_F and the self-shielding factor f, *i.e.*,

$$\alpha_f = \frac{v_f f \Sigma_f}{v_F \Sigma_F} \quad , \qquad (VI.3.5\text{-}5)$$

$$\alpha_m = \frac{v_m \Sigma_m}{v_F \Sigma_F} \quad , \qquad (VI.3.5\text{-}6)$$

$$\alpha_f + \alpha_m = 1 \quad , \qquad (VI.3.5\text{-}7)$$

and $v_F = v_f + v_m$,

where Σ_m, is the macroscopic cross section of the graphite diluent, and v_f and v_m are the volumes of the grain and the associated diluent, respectively. Since we treat only one kind of medium with grain structure, we drop the subscript I for simplicity.

Insertion of Eqs.(VI.3.5-5) and (VI.3.5-6) into Eq.(VI.3.5-7) gives the equivalent cross sections of the fuel pellet, Σ_F as

$$\Sigma_F = \frac{1}{v_F} \left(v_f f \Sigma_f + v_m \Sigma_m \right) . \qquad (VI.3.5\text{-}8)$$

When one of four variables; f, Σ_F, α_f, or α_m is given, then the rest is determined by the relations expressed by Eqs.(VI.3.5-5), (VI.3.5-6), and (VI.3.5-7).

Using the newly defined variables, Eqs.(VI.3.5-3) and (VI.3.5-4) can be rewritten, respectively, as

$$P_{IJ}(l,k) = f_{ll} \alpha_{kJ} P_{lJ} \quad , \qquad (VI.3.5\text{-}9)$$

$$P_{ll}(l,k) = Q_l(l,k) - f_{ll} \alpha_{kl}(1 - P_{ll}) \quad , \qquad (VI.3.5\text{-}10)$$

where $f_{ll} = f$ for the grain, and $f_{ll} = 1$ for the diluent.

When the collision probability is given, Eqs.(VI.3.5-1) and (VI.3.5-2) can be solved recurrently on ultra-fine groups in the dominant resonance energy range, $E < 130$ eV, as described in the last subsection. That is, for computation of neutron flux on a uniform lethargy mesh, we use the RIFF-RAFF method developed by Kier[42,43]. Hence, the lethargy mesh Δu used is sufficiently narrow, say $\Delta u=0.00125$, compared with the maximum lethargy gain per collision with the heaviest nucleus in the system under consideration. This mesh is also narrow compared with the Doppler width at room temperature. The resonance cross sections are prepared at the mid-point of each ultra-fine group using a code MCROSS[49]. The time-consuming computation of collision probabilities for each ultra-fine group is economized using an interpolation scheme, where the values of P_{ij} are interpolated from the tabulated values on a dozen points of the absorber cross sections.

We present the following four models for the calculation of the self-shielding factor f:

(1) Smearing method

We assume that the grain medium is made up of an infinite number of microscopic cells and the neutron balance can be described by using the two-region collision probability

$$P_{ij} = Q_l(i,j). \qquad (VI.3.5\text{-}11)$$

Equations (VI.3.5-1) and (VI.3.5-2) are solved by following the same procedure as above described. Resonance cross sections of each resonant nuclide are spatially smeared without collapsing the ultra-fine groups to give their representative values for the medium. That is, the homogenized cross section σ_h is obtained by the flux-volume weighted average of the resonance cross section σ_f, *i.e.*,

$$\sigma_h(u) = \frac{\sigma_f(u) v_f \varphi_f(u)}{v_f \varphi_f(u) + v_m \varphi_m(u)} .$$

If we define the self-shielding factor f by

$$f = \frac{(v_f + v_m)\,\varphi_f(u)}{v_f \varphi_f(u) + v_m \varphi_m(u)} , \qquad (VI.3.5\text{-}12)$$

we can obtain σ_h by a simple volume-weighted average of the effective cross section, $f\sigma_f$ as

$$\sigma_h(u) = \frac{f\sigma_f(u)\, v_f}{v_f + v_m} .$$

Thus, the effective microscopic cross section $f\sigma_f$ can be treated as if it is the resonance cross section of one constituent in a homogeneous medium. The resulted variation of the cross-section with neutron energy is much smooth, compared with the original cross section, due to the spatial shielding. The shielded cross sections are prepared for all the reactions of the resonant nuclides under consideration.

(2) Collision rate method

As noted before, the fraction of collision rate α_l was introduced as to be independent of the origin of neutron. We can suppose several origins of neutron, for example, a neutron just having escaped from the absorber grain, a neutron emitted from the diluent, a neutron impinging on the outer surface of the imaginary cell, or a neutron beam injected

into the medium of grain structure *etc*.

Now we introduce a quasi-analytic expression of the collision probability for the imaginary cell to give a physical explanation. The following approximation was derived by Nordheim[50] :

$$Q(f,m)= P_e \left[\frac{1 - c}{1 - (1 - l_f\Sigma_f P_e) c} \right] , \qquad (VI.3.5\text{-}13)$$

where P_e (=1 - P_c) is the escape probability[5] from an individual grain, c, the Dancoff correction, and Σ_f is the macroscopic cross section of the absorber grain and its mean chord length l_f is given by

$$l_f = \frac{4v_f}{S_f} , \qquad (VI.3.5\text{-}14)$$

where S_f is the surface area of the absorber grain. The conservation and reciprocity relations give other elements of $Q(l,k)$.

We shall discuss the following two models of neutron origin for calculating the collision rates.

(2.1) Neutron emitted from absorber grain

Segev[51] proposed an expression for the self-shielding factor which was obtained by equating the term in the brackets of Eq. (VI.3.5-13) to $[1/(1+fv_f\Sigma_f/v_m\Sigma_m)]$ which is the probability that a neutron escaping from a lump will collide in the diluent of the homogenized medium. The resulted expression is

$$f = \frac{c}{1-c} \frac{v_m \Sigma_m l_f}{v_f} P_e . \qquad (VI.3.5\text{-}15)$$

Inserting this expression into Eqs. (VI.3.5-5), (VI.3.5-6) and (VI.3.5-8), we obtain

$$\alpha_f(f) = \frac{l_f\Sigma_f P_e c}{1 - (1 - l_f\Sigma_f P_e) c} , \qquad (VI.3.5\text{-}16)$$

$$\alpha_m(f) = \frac{1 - c}{1 - (1 - l_f\Sigma_f P_e)c} , \qquad (VI.3.5\text{-}17)$$

where the index f of $\alpha_f(f)$ denotes that the origin of neutron is fuel grain. It can be explained that the fraction of collision rate is measured for a neutron just having escaped from the absorber grain. That is, the probability that the neutron has the next collision with other grains is given from the expression of Eq. (VI.3.5-13) by

$$Q(f,f)\cdot P_c = 1 - Q(f,m) - (1 - P_e)$$

$$= \frac{l_f\Sigma_f P_e^2 c}{1 - (1 - l_f\Sigma_f P_e) c} . \qquad (VI.3.5\text{-}18)$$

On the other hand, the collision with the diluent is given by $Q(f,m)$, which can be calculated by the reciprocity relation. Dividing these quantities by the escape probability P_e, we have the fractions of collision rates; α_f and α_m just defined by Eqs. (VI.3.5-16) and (VI.3.5-17). Inserting Eqs. (VI.3.5-16) and VI.3.5-17 into Eqs. (VI.3.5-5) and (VI.3.5-6), we obtain the same expression of f as Eq. (VI.3.5-15).

(2.2) Neutron emitted from the diluent

We can define the collision rate for a neutron emitted from the diluent. It is given directly by either $Q(m,f)$ or $Q(m,m)$. A little algebra using the reciprocity relation gives

$$\alpha_f(m) = \frac{\upsilon_f\Sigma_f}{\upsilon_m\Sigma_m} \frac{P_e\ (1\ -\ c)}{1\ -\ (1\ -\ l_f\Sigma_f P_e)\ c}\ , \qquad (VI.3.5\text{-}19)$$

where the index m of $\alpha_f(m)$ denotes that the origin of neutron is the diluent.

We shall compare the α_f's above introduced. The ratio of $\alpha_f(f)$ to $\alpha_f(m)$ is given by

$$\frac{\alpha_f(f)}{\alpha_f(m)} = \frac{\upsilon_m\Sigma_m l_f}{\upsilon_f} \frac{c}{(1\ -\ c)}\ . \qquad (VI.3.5\text{-}20)$$

Lane *et al.*[52] gave a rational expression for the Dancoff correction in case of the small volume fraction of grain by

$$c = 1/(1\ +\ \frac{\upsilon_m\Sigma_m l_f}{\upsilon_f}\ )\ . \qquad (VI.3.5\text{-}21)$$

Insertion of Eq.(VI.3.5-21) into Eq.(VI.3.5-20) gives $\alpha_f(f)/\alpha_f(m)=1$. This fact means the two models identical in the limited case of the small volume fraction of grain. As shown later numerically, however, $\alpha_f(f)$ is always smaller than $\alpha_f(m)$ because the Dancoff correction accurately estimated is slightly lower than the value given by Eq.(VI.3.5-20). It can be physically explained: The neutron just having escaped from the absorber grain has to escape from the cell in order to collide with other grains, whereas the neutron emitted from the diluent may collide with the grain in the same cell and also it is at a shorter distance from the grains in the outside of the cell than the former. Hence, the latter has always the larger probability to collide with the grain than the former has.

Insertion of Eq.(VI.3.5-21) into Eq.(VI.3.5-15) gives $f=P_e$, then Eq.(VI.3.5-8) in this case is given by

$$\Sigma_F = \ (\ \upsilon_f P_e\ \Sigma_f\ +\ \upsilon_m\Sigma_m\ )/\ \upsilon_F\ , \qquad (VI.3.5\text{-}22)$$

which is just the expression given by Lane *et al.*[52]

The concrete form of P_e by Case *et al.*[5] gives readily the limiting values of Σ_F, *i.e.*

$$\Sigma_F = \begin{cases} (\ \upsilon_m\Sigma_m\ +\ \pi R_f^2\ )\ /\upsilon_F & for\ \ R_f\Sigma_f >> 1\ , \\ (\ \upsilon_m\Sigma_m\ +\ \upsilon_f\Sigma_f\ )\ /\upsilon_F & for\ R_f\Sigma_f << 1\ , \end{cases} \qquad (VI.3.5\text{-}23)$$

where R_f is the radius of the absorber grain. Here, it should be noted that the effective cross section, Σ_F is bounded to be finite even if Σ_f goes to infinite. This behavior is an essential feature of the grain effect.

(3) Transmission cross section method

Supposing the attenuation of a uniform current of neutron beam through a media having grain structure, Tsuchihashi & Gotoh[4] presented a formulation of the equivalent cross section, Σ_F given by the solution

of a following transcendental equation:

$$X \exp(-X) = C \ , \qquad (VI.3.5\text{-}24)$$

where

$$X = 2R_p(\Sigma_F - \Sigma_m) \ ,$$

$$C = 2R_p\pi\rho R_f^2 F(2R_f(\Sigma_f - \Sigma_m)) \ ,$$

$$F(x) = 1 - 2\,[\,1-(1+x)\exp(-x)\,]\,/x^2 \ ,$$

R_p, the radius of the coated particle, and ρ is the number density of particles. The equivalent cross section corresponds to the smaller root of Eq. (VI.3.5-24), and is always slightly larger than $\Sigma_m + C/2R_p$.

If account is taken of the fact that $F(x)$ is a monotonously increasing function for $x > 0$ and

$$F(x) = \begin{cases} 2/3\ x + O(x^2) & for\ |x| \ll 1 \ , \\ 1 + O(x^{-2}) & for\ x \gg 1 \ , \end{cases} \qquad (VI.3.5\text{-}25)$$

we can show that Eq. (VI.3.5-24) has the proper limit values for the black limit ($R_f\Sigma_f \gg 1$) and the homogeneous limit ($|(\Sigma_f - \Sigma_m)R_f| \ll 1$) just the same as given in Eq. (VI.3.5-23).

We have not described yet the treatment of the double heterogeneity in the higher resonance energy range; $E > 130$ eV. We take the table-look-up method for the resonance shielding factors. We adopted Segev's expression[51] based on the NR approximation for the background cross section presenting the heterogeneous effect, *i.e.*,

$$\Sigma_e = \frac{1-C}{l_f}\,\frac{a}{1 + C(a\beta - 1)} \ , \qquad (VI.3.5\text{-}26)$$

and

$$\beta = \frac{v_m / v_F}{v_m / v_F + (1-C)A / L(1 + C(A-1))} \ , \qquad (VI.3.5\text{-}27)$$

where a and A are the Bell or Levine factor for the microscopic and macroscopic cells, respectively, and C and L are the Dancoff correction and the mean chord length of the macroscopic cell, respectively.

The above treatment is incorporated into the SRAC code system together with the models described above for the lower energy range, to yield the resonance absorption in a doubly heterogeneous cell.

A sequence of study using the present method gives us confidence that the present approach is straightforwardly applicable to the doubly heterogeneous system with the realistic geometry such as LWR and LMFBR lattice cells since our treatment on the macroscopic geometry is fairly general. This method allowing three one-dimensional (sphere, plane and cylinder) cells as optional microscopic geometry has been incorporated in the SRAC code system

VI.4 Solution of Linear Equation

It is one of the features of the SRAC code system to execute the cell calculations by the collision probability method to cover the whole neutron energy range. In this section we shall describe how to solve the linear equation introduced in Sect.VI.1.

Because of the difference of physical characteristics, the different specialized equation is formulated separately by neutron energy range. Although the concatenation of the equations into a set of equations so as to describe the quantities in the whole neutron energy is available, the cell calculation is usually achieved separately by neutron energy range. Partly because there occurs no upscattering in the epi-thermal and fast neutron energy range, but does in the thermal neutron range where any iterative process among the energy group variables is required. Partly because the neutron flux distribution in the fast and epi-thermal neutron range is relatively flat, which allows coarse spatial division of the cell model or the overall flat flux assumption coupled with some suitable resonance shielding treatments, while the distribution in the thermal neutron range shows sharp spatial change due to small flight path in this energy range needs fine spatial division. It is to be noted that although there occurs the sharp flux depression due to the strong resonance structure of the fertile nuclides near the resonance energy, fine spatial division is not necessary to evaluate the overall resonance absorption. Because neutrons scarcely come out from the place where the depression occurs, the shape of the depression does not affect the absorption rate.

VI.4.1 General form of linear equation

When the system under consideration is divided into N regions and the neutron energy range is divided into G groups, Eq.(VI.1-9) is rewritten as

$$\Sigma_{jg}\varphi_{jg} = \sum_{i=1}^{N} P_{ijg} \left[\sum_{g'=1}^{G} \Sigma_{sig'\to g}\varphi_{ig'} + S_{ig} \right] . \qquad (VI.4.1\text{-}1)$$

The physical quantities are redefined as follows:

1. the volume of the region i; $V_i = \int_{V_i} dV$.
2. the integral flux over the region i for the energy group g;
$$\varphi_{ig} = \int_{V_i} dV \int_{\Delta E_g} dE\ \varphi(\boldsymbol{r},E) .$$
3. the fixed source;
$$S_{ig} = \int_{V_i} dV \int_{\Delta E_g} dE\ S(\boldsymbol{r},E) .$$
4. the collision probability from the region i to j for the group g; P_{ijg} .
5. the modified collision probability which has finite value even if the collision region j is vacuum;
$$P^{\sharp}_{ijg} = P_{ijg} / \Sigma_{jg} .$$
6. the emission rate of the region i for the group g; H_{ig} .
7. Nuclear constants of the material m;

Σ_{mg} = total cross section,
$\nu\Sigma_{fmg}$ = ν*fission cross section,
Σ_{cmg} = absorption cross section,
$\Sigma_{smg\to g'}$ = scattering cross section from the group g to the group g'

$$\Sigma_{rmg} = \sum_{g'\in G} \Sigma_{smg\to g'} + \sum_{g'\notin G} \chi_{mg'} \nu\Sigma_{fmg} \quad ,$$

scattering-out cross section which is used to evaluate the neutron balance of the system. It has non-zero value when a fixed source problem of a limited energy range is considered. The quantity χ_{mg} stands for the fission yield to the group g.

Using the above definitions, the equation to be solved is written by

$$\varphi_{jg} = \sum_{j} P^{\#}_{jig} H_{jg} \quad . \qquad (VI.4.1\text{-}2)$$

The emission rate for the fixed source problem is written by

$$H_{ig} = S_{ig} + \sum_{g'=1}^{G} \Sigma_{smg'\to g}\varphi_{ig'} + \chi_{mg} \sum_{g'=1}^{G} \nu\Sigma_{fmg'}\varphi_{ig'} \quad , \qquad (VI.4.1\text{-}3a)$$

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

$$H_{ig} = \sum_{g'=1}^{G} \Sigma_{smg'\to g}\varphi_{ig'} + \frac{\chi_{mg}}{\lambda} \sum_{g'=1}^{G} \nu\Sigma_{fmg'}\varphi_{ig'} \quad . \qquad (VI.4.1\text{-}3b)$$

Equation (VI.4.1-2) coupled with Eq. (VI.4.1-3a) forms inhomogeneous equations and that coupled with the Eq. (VI.4.1-3b) forms homogeneous equations. In both problems, the number of unknown is $N*G$. The general matrix of the same rank consists of $(N*G)^2$ elements, however, the computer storage required for the above equations is at most $N^2G + MG^2 < NG(N+G)$, [N^2G for the collision probability and MG^2 for scattering matrix, where M is the number of materials] . The size G^2 for the scattering matrix will be reduced if only down-scattering is considered or only heavy nuclides compose a material. In the following sections the techniques to reduce the computer time and storage for several particular problems will be described.

VI.4.2 Iterative procedure in thermal energy range

In the thermal neutron energy range, the emission rate Eq. (VI.4.1-3a) is rewritten as

$$H_{ig} = S_{ig} + \sum_{g'=1}^{G} \Sigma_{smg'\to g}\varphi_{ig'} \quad . \qquad (VI.4.2\text{-}1)$$

The fixed source S_{ig} is usually given by the slowing-down from the epi-thermal range. Since the fluxes are completely coupled each other by up- and down-scattering, the equations of the form of Eq. (VI.4.1-2) and (VI.4.2-1) are solved by an iterative procedure by using the method of Successive Over Relaxation (SOR) as used in the THERMOS code[2]. Some extension was made to consider an isolated cell and the outer iteration for the eigenvalue problem to the original routine. The procedure is as follows;

Step 1. Set the initial guess of φ_{ig}

Step 2. Obtain the normalization factor for the source term,

$$B = \sum_{g=1}^{G} \sum_{i=1}^{N} S_{ig} \sum_{j=1}^{N} P_{ijg} \quad .$$

After finding B, every S_{ig} is divided by B. Thus the total number of neutrons which will have the next collision in the system considered is set to unity.

Step 3. Calculate H_{ig} according to Eq. (VI.4.2-1) and simultaneously the scaling factor C;

$$C = \sum_{g=1}^{G} \sum_{j=1}^{N} \{ \Sigma_{mg}\varphi_{ig} - (H_{ig} - S_{ig}) \sum_{j=1}^{N} P_{ijg} \} ,$$

which is defined as the ratio of the removal reaction (absorption, scattering-up above the thermal cut off energy, and leakage) to the source which will have the next collision in the system. This factor must be unity in the converged state.

Step 4. Calculate the new fluxes $\varphi^{(m+1/2)}$ according to Eq. (VI.4.1-2) and the weighted residual,

$$R^{(m)} = \{ \sum_{g=1}^{G} \sum_{i=1}^{N} (\varphi_{ig}^{(m+1/2)}/C - \varphi_{ig}^{(m)})^2 R_{ig}^2 \; / \; \sum_{g=1}^{G} \sum_{i=1}^{N} R_{ig}^2 \}^{1/2} \quad .$$

The root mean square (RMS) residual will be used to estimate the converging slope as $\mu^{(m)} = R^{(m)}/R^{(m-1)}$. The weighting reaction R_{ig} is set to absorption $\Sigma_a\varphi_{ig}$, The superscript (m) is an iteration counter.

Step 5. Modify the over-relaxation factor ω. In the first L_e iterations, the initial value of ω_0 will be used. At each iteration, the value of $\omega_e = 1/(1 - \mu^{(m)})$ is tested. If all the values of ω_e in the last L_e iterations agree within the given extrapolation criterion ε_e, an extrapolation takes place using the most recent value of ω_e. The testing for a possible extrapolation is suppressed during the L_d iterations following the extrapolation. The value λ (the estimate of the eigenvalue of the matrix considered) is computed as $\lambda = (\mu^{(m)} - 1 + \omega_0)/\omega_0$ and a new ω is obtained as $\omega_1 = 2/(2-\lambda)$ to be used after the next iteration.

If an increase of the residual is detected during the iteration, a moderate value of ω is selected as $\omega_2 = (\omega_1 * f_{under})^{1/2}$.

Step 6. Obtain the new fluxes by the over-relaxation;

$$\varphi_{ig}^{(m+1)} = \varphi_{ig}^{(m)} + \omega(\varphi_{ig}^{(m+1/2)}/C - \varphi_{ig}^{(m)}) \quad .$$

The loop from Step 3 to Step 6 is repeated until the residual $R^{(m)}$ is less than ε or the iteration counter m exceeds L_{in}. The quantities ε, ω_0, ε_e, f_{under}, L_e, L_d, and L_{in} are input numbers.

In the practical use of this procedure, we find that the scaling by C is effective to accelerate the convergence, but, we encounter some difficulties in attaining the convergence. One problem occurs in a weakly absorbing case where a slow convergence rate is observed through

the iteration. Once an extrapolation is taken place, while it greatly reduces the RMS residual, the new over-relaxation factor ω_1 which takes the value close to 2.0, say, 1.8 after the extrapolation, causes growth of the residual in most cases. The following procedure which is activated when the increase of the residual is detected in order to have a moderate over-relaxation factor ω_2 helps to escape from such a catastrophe. Another problem happens in a strongly absorbing case where we encounter also growth of the residual. It is the case in which we can expect a rapid convergence. It is thought that because the secondary eigenvalue of the matrix, λ_1 is not far from the largest eigenvalue λ_0, the spectral radius of the higher mode in the modified matrix might exceed unity. We can escape from this trouble by feeding a relatively low f_{under}, say 0.5, which suppresses the new factor ω_2 below unity. In other word, in a strongly absorbing case, an under-relaxation is required.

As the computer time required for the iterative process is much shorter than that for the preparation of collision probabilities, the optimum use of the above procedure is not essential. We may suppress the extrapolation by feeding the strict criterion ε_e not to activate the extrapolation. Against the case where the divergence may occur, the low value of f_{under} can prevent the divergence.

VI.4.3 Solution by matrix inversion in fast neutron range

For the fixed source problem in the fast neutron range, the emission rate can be rewritten by

$$H_{ig} = S_{ig} + \sum_{g'=1}^{g} \Sigma_{smg'\to g}\varphi_{ig'} + \chi_{mg} \sum_{g'=1}^{G} \nu\Sigma_{fmg'}\varphi_{ig'} \quad . \qquad (VI.4.3\text{-}1)$$

The fixed source term S_{ig} usually consists of the thermal fission neutron. The scattering term is determined by the fluxes of the upper energy groups. Given the fast fission term, the fluxes can be successively solved starting at the highest energy group. The number of the unknowns to be simultaneously solved is the total number of regions, N.

We have a choice for the solution whether by an iterative method or by a matrix inversion. As far as N is less than about 40, the round off error due to the limited computer precision encountered in the matrix inversion can be negligible. The computer time required for the matrix inversion (proportional to N^3) does not so much exceed that for the iterative method (proportional to the *iteration count* $*$ N^2). In the SRAC code system, the matrix inversion is applied preferring its definitive solution.

After finding the flux distribution for all the energy groups, we have to modify the assumed fast fission source distribution by an iterative process. This power iteration converges rapidly for the case of a thermal reactor because the ratio of fast fission to thermal fission is small. Contrary, the procedure of this section can not be applied for the fast reactor where the matrix may have an eigenvalue greater than unity.

The procedure described in this section is summarized as follows;

Step 1. Normalize the fixed source S_{ig}.

Step 2. Set the initial guess for the fast fission distribution.
Step 3. Starting at the highest energy group g=1, calculate emission rates of a group H_{ig} by Eq. (VI.4.3-1).
Calculate fluxes of a group by a matrix inversion.
Repeat Step 3 for all groups.
Step 4. Calculate fast fission distributions and modify it by an SOR.

Repeat Step 3 and 4, until the fast fission distribution converges.

VI.4.4 Iterative procedure for eigenvalue problem in whole energy range

Considering the fact that while the iteration count in the thermal energy range amounts to several tens, in the fast energy range any iterative process is not needed, the treatment for the thermal group flux and for the fast group flux should be different when both have to be solved simultaneously in the whole energy range.

Our senario to solve the eigenvalue problem in the whole energy range in the cell calculation is a combination of the procedures described in the previous two sections, as follows;

Step 1. Set an initial guess of fission distribution and normalize it.
Step 2. Starting at the highest energy group g=1, calculate the emission rate of a group H_{ig} by Eq. (VI.4.3-1).
Calculate fluxes of a group by a matrix inversion.
Repeat Step 2 for all fast groups.
Step 3. Calculate the slowing-down source to the thermal groups.
Step 4. Calculate the thermal fluxes by repeating the thermal iteration as described in Section VI.4.1 until the convergence is attained or the fixed iteration count is reached.
Step 5 Calculate and renormalize the fission distribution using the new flux distribution and modify the distribution by an SOR.

Repeat Step 2 through Step 5 until the fission distribution converges. The eigenvalue is obtained as the renormalization factor of fission distribution calculated in Step 5 at the final power iteration.

VI.5 Smearing and Collapsing of Group Constants

It is one of the features of the SRAC code system to execute smearing and collapsing in separate steps and timing is controlled by the user. The smeared cross sections can be used by the succeeding cell calculation in the case of double heterogeneity. Smearing may be done by using the spatial distribution of the fluxes obtained by the cell calculation, over a whole cell or on partitioned regions which are specified by X-region. Description will be also made for the optional calculation of cell averaged diffusion coefficient specified by IC17 in Sect. II.1.

Next, discussions will be made on the optional use of neutron spectrum for collapsing, as we have several choice, as follows;

(1) the asymptotic spectrum characterized by Maxwellian temperature stored in FASTU and THERMALU files,
(2) the spectrum to a mixture provided by the user in FLUX file,
(3) the spectrum calculated by P_1 or B_1 approximation to an isolated mixture specified by IC16 which was used for the transport cross sections of the mixture,
(4) the spatially integrated spectrum of the flux obtained by the cell calculation,
(5) if the above is for an infinite array of lattice cell, the spectrum recalculated by the P_1 or B_1 approximation activated by IC9 to include the leakage effect, where attention is paid to reflect this leakage correction on the space-dependent spectrum in order to give the suitable reaction rate of depleting nuclide associated to the mixture to cell burn-up routine.

Collapsing of energy group structure for few group core calculation is described with emphasis on the diffusion parameter.

Description will be given using the following nomenclature.

Table VI.5-1 Nomenclature

Symbols	Meaning
g , g'	Multi-group number
G , G'	Few group number
i , j	A) In the collision probability method: T- or R-region number, which is used as spatial index. B) For Sn and diffusion calculation: fine mesh number
m	Material number for spatial region i or j, in other words, m-th M-region which includes regions i or j.
I	X-region number for editing region to which the average cross section is given (usually I=1 is given throughout the system, except for the case where a supercell model is used.)

(*Note*: $i,j \in m \in I$)

z	Nuclear reaction for process z (fission, capture, scattering etc.)
n	Nuclide.
l	Order of Legendre expansion (l=0 or 1).
$\chi_{g,m}$	Fission spectrum of material m.
$\sigma^n_{z,m,g}$	Effective multi-group cross section of nuclide n in material m (z = fission or capture), which is obtained by the interpolation of the self-shielding factors stored in FASTU and THERMALU files except for the second resonance energy range where the ultra-fine group calculation or IR method can be used. This cross section is kept in MICREF file in order to be used for cell burnup calculation or for activation calculation.
$\sigma^n_{z,m,G}$	Effective few-group cross section of nuclide n in material m (z = fission or capture) given to the cell burnup routine but not kept.
$\Sigma_{z,m,g}, \nu\Sigma_{f,m,g}$	Macroscopic multi-group cross section of material m.
$\Sigma_{tr,m,g}$	Macroscopic transport cross section of material m.
$\Sigma_{sl,m,g\to g'}$	Macroscopic energy transfer cross section of material m.
$\bar{\Sigma}_{z,I,g}$, $\nu\bar{\Sigma}_{f,I,g}$	Averaged multi-group cross section of I-th X-region. Usually one X-region is assigned to each lattice cell unless a supercell model is used.
$\bar{\Sigma}_{s0,I,g\to g'}$	Averaged energy transfer cross section of I-th X-region.
$\bar{\Sigma}_{tr,I,g}$	Averaged transport cross section of I-th X-region
$\bar{D}_{I,g}$	Averaged diffusion coefficient
$\bar{\chi}_{I,G}$	Few group fission spectrum
$\bar{\Sigma}_{z,I,G}$, $\nu\bar{\Sigma}_{f,I,G}$	Few group cross section of I-th X-region.
$\bar{\Sigma}_{s0,I,G\to G'}$	Few group energy transfer cross section of I-th X-region.
$\bar{\Sigma}_{tr,I,G}$	Few group transport cross section.
$\bar{D}_{I,G}$	Few group diffusion coefficient.

$\varphi_g(r) = \int_{\Delta u_g} \varphi(r,u)du$	Analytical expression of space dependent flux of group g.
$\varphi_{i,g} = \int_{V_i} \Phi_g(r)dr/V_i$	Average flux of region i and integrated in group g.
$\bar{\Phi}_{m,g} = \sum_{i\in m} \varphi_{i,g}V_i/V_m$	Average flux of region m of multi-group g
$\bar{\bar{\Phi}}_{m,G} = \sum_{g\in G} \bar{\Phi}_{m,g}$	Average flux of region m of few group G
$\bar{\Phi}_{I,g} = \sum_{i\in I} \varphi_{i,g}V_i/V_I$	Average flux of region I of multi-group g
$\bar{\Phi}_g = \sum_{i\in cell} \varphi_{i,g}V_i/V_{cell}$	Average flux of whole cell of multi-group g
$\bar{\Phi}_{I,G}=\sum_{g\in G} \varphi_{I,g}$	Average flux of region I of few group G

Smearing

Smearing or spatial average of cross sections is achieved in the MIXX routine by using the spatial distribution of the flux $\varphi_{i,g}$ obtained by the cell calculation as described in Sect.VI.4.

When the second resonance energy range below 130 eV is treated by the direct method described in Sect.VI.3 using the PEACO routine, the effective cross sections in each region and the fluxes with multi-group structure obtained by Eq.(VI.4.1-1) are replaced by those from the PEACO routine.

$$\bar{\bar{\Sigma}}_{z,I,g}= \sum_{i\in I}\Sigma_{z,m,g}V_i\varphi_{i,g}/(\bar{\Phi}_{I,g}V_i) \qquad (VI.5\text{-}1)$$

$$\nu\bar{\bar{\Sigma}}_{f,I,g}= \sum_{i\in I}\nu\Sigma_{f,m,g}V_i\varphi_{i,g}/(\bar{\Phi}_{I,g}V_i) \qquad (VI.5\text{-}2)$$

$$\bar{\chi}_{I,g}=\sum_{i\in I} \chi_{m,g} \sum_{g'} \nu\Sigma_{f,m,g'}V_i\varphi_{i,g'}/ \sum_{i\in I}\sum_{g'} \nu\Sigma_{f,m,g'}V_i\varphi_{i,g'} \qquad (VI.5\text{-}3)$$

$$\bar{\Sigma}_{sl,I,g\to g'}= \sum_{i\in I}\Sigma_{sl,m,g\to g'}V_i\varphi_{i,g}/(\bar{\Phi}_{I,g}V_I) \qquad (VI.5\text{-}4)$$

Benoist presented a theory of the diffusion coefficient in reactor lattice, leading to expressions valid in full generality[7]. For the diffusion coefficient of the direction k, omitting the absorption correction and angular terms, this theory gives

$$\bar{D}_{k,g}=\sum_i \sum_j V_i\varphi_{i,g}\lambda_{j,g}P_{ijk,g}/(3\sum_i V_i\varphi_{i,g}). \qquad (VI.5\text{-}5)$$

where i and j stand for the region number, $\lambda_{j,g}$ the transport mean free path for region j given by $1/\Sigma_{tr,m,g}$ and $P_{ij,k,g}$ is a directional first flight collision probability given by Eq.(VI.1-16).

Three options are provided for calculating the diffusion coefficient $\bar{D}_{k,g}$.

** IC17 = 1 **

The first option is a rather sophisticated one which has been proven by experience to be fairly accurate, and the diffusion coefficient is given by the inverse of the the cell averaged transport cross section, *i.e.*,

$$\bar{D}_{h,g} = \sum_i V_i \varphi_{i,g} / (3 \sum_i V_i \varphi_{i,g} \Sigma_{tr,m,g}). \qquad (VI.5\text{-}6)$$

When the flux $\varphi_{i,g}$ are flat in the whole system and we can assume

$$P_{ijk,g} = V_i \Sigma_{tr,m,g} / (\sum_j V_j \Sigma_{tr,m,g}) \qquad \text{(homogeneous limit)} \qquad (VI.5\text{-}7)$$

independent of k, we have

$$\bar{D}_{h,g} = \sum_i V_i / (3 \sum_i V_i \Sigma_{tr,m,g}). \qquad (VI.5\text{-}8)$$

Contrary, 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

$$P_{ij,k,g} = \delta_{ij} \qquad (VI.5\text{-}9)$$

and

$$\bar{D}_{t,g} = \sum_i V_i \varphi_{i,g} \lambda_{i,g} / (3 \sum_i V_i \varphi_{i,g}). \qquad (VI.5\text{-}10)$$

It follows at once from the fundamental theorem of algebra

$$\bar{D}_{t,g} \geqq \bar{D}_{h,g}. \qquad (VI.5\text{-}11)$$

The use of $\bar{D}_{t,g}$ to the cell with voided regions falls into the drawback that the diffusion coefficient is unreasonably large to be used in practical calculation.

** IC17 = 2 **

The isotropic diffusion coefficient is used and defined by

$$\bar{D}_{0,g} = \frac{1}{3} \sum_k \bar{D}_{k,g}. \qquad (VI.5\text{-}12)$$

Then, from the definition of the directional collision probability by Eq. (VI.1-16), this diffusion coefficient can be calculated using

the isotropic collision probability defined by Eq.(VI.1-13), i.e.,

$$\overline{D}_{0,g} = \sum_i \sum_j V_i \varphi_{i,g} \lambda_{j,g} P_{ij,g} / (3 \sum_i V_i \varphi_{i,g}). \qquad (VI.5\text{-}13)$$

** IC17 = 3 **

The anisotropic diffusion coefficients defined by Eq.(VI.5-5) are calculated. These coefficients are used for the 2-D or 3-D diffusion calculation using the CITATION code.

Spectrum for collapsing

The fluxes thus used are however not taken account of the neutron leakage from the system under study. So these fluxes should be renormalized to take account of the leakage. For this purpose, the homogenized effective cross sections are at first calculated without any group collapsing for the whole energy range to obtain the homogeneous P_1 or B_1 spectrum.

The P_1 or B_1 equation in multigroup form can be written in the conventional and unified way as

$$F_{1g} + \overline{\Sigma}_{tg} F_{0g} - \sum_{g'=1}^{G} \overline{\Sigma}_{s0g' \to g} F_{0g'} + \overline{X}_{0g}$$

$$3\alpha_g \overline{\Sigma}_{tg} F_{1g} - B^2 F_{0g} - \sum_{g'=1}^{G} \overline{\Sigma}_{s1g' \to g} F_{1G'} \qquad (VI.5\text{-}14)$$

where α_g equals unity for the P_1 approximation and is given for the B_1 approximation by

$$\alpha_g = (x_g \tan^{-1} x_g) / 3[1 - (\tan^{-1} x_g)/ x_g] \quad \text{for} \quad B^2 \geqq 0$$

$$\alpha_g = (x_g \tanh^{-1} x_g) / 3[(\tanh^{-1} x_g)/x_g - 1] \quad \text{for} \quad B^2 \leqq 0$$

$$\text{with} \quad x_g = \sqrt{|B^2|} / \overline{\Sigma}_{tg}. \qquad (VI.5\text{-}15)$$

where the quantity B^2 is the value entered in BLOCK 4 of II.1.

Assuming a fission spectrum as the source term. Eq.(VI.5-2) is solved by the P_1 or B_1 approximation for the fast energy range, while the solution is obtained in the thermal energy range assuming a slowing-down source. The linked spectrum for the whole energy range is optionally used to replace the spectrum of the flux of the cell calculation obtained by assuming an infinite array of lattice cell.

Using the solution, φ_{ig}, of Eq.(VI.4.1-1), the spatially averaged fluxes are calculated by

$$\overline{\Phi}_g = \sum_{i \in cell} V_i \varphi_{ig} / V_{cell}. \qquad (VI.5\text{-}16)$$

Then, the spectrum which includes the leakage effect can be given by

$$\varphi'_{i,g} = \varphi_{i,g} F_{0g} / \bar{\Phi}_g . \qquad (VI.5\text{-}17)$$

The resultant spectrum $\varphi'_{i,g}$ replaces the infinite spectrum $\varphi_{i,g}$, if the minus sign of IC9 is entered, and is used to give the microscopic effective cross section $\sigma^n_{z,m,G}$ to the cell burn-up routine. Above procedure is executed in the HOMOSP routine.

For the isolated mixture which is not used in the cell calculation, either of the spectra on the items (1), (2) and (3) in the second paragraph of this section will be chosen for the weight.

Collapsing of Group constants

Collapsed group cross sections for the few group calculations are obtained from the following prescriptions. Description will be also given for the transport cross section and diffusion coefficient in the next subsection.

$$\sigma^n_{z,m,G} = \sum_{g \in G} \sigma^n_{z,m,g} \bar{\Phi}_{m,g} / \bar{\Phi}_{m,G} \quad (z = f \text{ or } c) \qquad (VI.5\text{-}18)$$

$$\bar{\chi}_{l,G} = \sum_{g \in G} \bar{\chi}_{l,g} \qquad (VI.5\text{-}19)$$

$$\bar{\Sigma}_{z,l,G} = \sum_{g \in G} \bar{\Sigma}_{z,l,g} \bar{\Phi}_{l,g} / \bar{\Phi}_{l,G} \qquad (VI.5\text{-}20)$$

$$\nu\bar{\Sigma}_{f,l,G} = \sum_{g \in G} \nu\bar{\Sigma}_{f,l,g} \bar{\Phi}_{l,g} / \bar{\Phi}_{l,G} \qquad (VI.5\text{-}21)$$

$$\bar{\Sigma}_{s0,l,G \to G'} = \sum_{g \in G} \sum_{g' \in G'} \bar{\Sigma}_{s0,l,g \to g'} \bar{\Phi}_{l,g} / \bar{\Phi}_{l,G} \qquad (VI.5\text{-}22)$$

Here the multi-group diffusion coefficient are assumed for each homogenized X-region (zone).

The group-dependent diffusion equation in the I-th region can be written as

$$-\bar{D}_{l,g} \nabla^2 \Phi_g(\boldsymbol{r}) + \bar{\Sigma}_{t,l,g} \Phi_g(\boldsymbol{r}) = S_g(\boldsymbol{r}).$$

Let us sum up the above equation over g to collapse the multi-group equation into the few group and denote each few group by G, then we have for the diffusion term

$$-\sum_{g \in G} \bar{D}_{l,g} \nabla^2 \varphi_g \sim B^2 \sum_{g \in G} \bar{D}_{l,g} \varphi_g(\boldsymbol{r}) = B^2 \bar{D}_{l,G} \bar{\Phi}_{l,G} \sim -\bar{D}_{l,G} \nabla^2 \bar{\Phi}_{l,G}$$

where

$$\bar{\Phi}_{l,C} \equiv \sum_{g \in C} \bar{\Phi}_{l,g} \qquad (VI.5\text{-}23)$$

and

$$\bar{D}_{l,C} \equiv \sum_{g \in C} \bar{D}_{l,g} \bar{\Phi}_{l,g} / \bar{\Phi}_{l,C}. \qquad (VI.5\text{-}24)$$

Hence, the few group transport cross section can be defined by

$$\bar{\bar{\Sigma}}_{tr,l,C} = 3\bar{\Phi}_{l,C} / \sum_{g \in C} \bar{D}_{l,g} \bar{\Phi}_{l,g} = \bar{\Phi}_{l,C} / \sum_{g \in C} \frac{\bar{\Phi}_{l,g}}{\bar{\bar{\Sigma}}_{tr,l,g}}. \qquad (VI.5\text{-}25)$$

Here, an equivalent relation holds as

$$\bar{\bar{\Sigma}}_{tr,l,C} = \frac{1}{3\bar{D}_{h,l,C}} \qquad (VI.5\text{-}26)$$

For the isolated mixture which is not used in the cell calculation, by using either of the spectra on the items (1), (2) and (3) in the second paragraph of this section, the procedure mentioned above is taken.

A special treatment is available for a strongly absorbing material where the diffusion equation does not hold, if IC17=-1 is entered by

$$\bar{\bar{\Sigma}}_{tr,l,C} = \sum_{g \in C} \bar{\bar{\Sigma}}_{tr,l,g} \bar{\Phi}_{l,g} / \bar{\Phi}_{l,C} \qquad (VI.5\text{-}27)$$

The selection of the definition by IC17 results in the SRAC macroscopic format as shown in Table VI.5-1.

Table VI.5-1 Selection of Diffusion Coefficient by IC17

IC17	$\bar{\bar{\Sigma}}_{total}$		D1		D2	
1	$\bar{\bar{\Sigma}}_{tr,l,g}$	25*	$\bar{D}_{h,g}$	24	$\bar{D}_{t,g}$	24
2	$\bar{\bar{\Sigma}}_{tr,l,g}$	25	$\bar{D}_{0,g}$	24	$\bar{D}_{t,g}$	24
3	$\bar{\bar{\Sigma}}_{tr,l,g}$	25	$\bar{D}_{radial,g}$ or $\bar{D}_{\perp,g}$	24	$\bar{D}_{axial,g}$ or $\bar{D}_{\parallel,g}$	24
-1	$\bar{\bar{\Sigma}}_{tr,l,g}$	27	$\bar{D}_{h,g}$	24	$\bar{D}_{t,g}$	24

Note * Number following the definition denotes the equation number in this section to show how to collapse the quantity into the few group constant.

VI.6 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 nuclides and their chains. All necessary information is stored in a burn-up library so that the user can choose one of fission product (FP) 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[53] 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

$$\frac{dN_n(t)}{dt} = -A_n(t)N_n(t) + Y_n(t) + \sum_j G_{j,n-1\to n}(t)N_{n-1}(t) \qquad (VI.6\text{-}1)$$

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.6-1) chosen for an exposure time θ is

$$N_n(t) = N_n(t-\theta)\exp(-A_n\theta) + \frac{Y_n(\theta)\{1-exp(-A_n\theta)\}}{A_n} + \sum_j \sum_{i=1}^{n-1} \{N_i(t-\theta)Q_{j,n,i}(\theta) + Y_{j,i}(\theta)U_{j,n,i}(\theta)\}. \qquad (VI.6\text{-}2)$$

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 under the constant neutron fluxes). These factors are given by

$$Q_{j,n,i}(\theta) = \sum_{m=i}^{n-1} \frac{\exp(-A_m\theta) - \exp(-A_n\theta)}{A_n - A_m}$$

$$*G_{j,m\to m+1}(\theta) \qquad * \prod_{\substack{k=i \\ k\neq m}}^{n-1} \frac{G_{j,k\to k+1}(\theta)}{A_k - A_m} \qquad (VI.6\text{-}3)$$

$$U_{j,n,i}(\theta) = \frac{1-\exp(-A_n\theta)}{A_n} * \prod_{m=i}^{n-1} \frac{G_{j,m\to m+1}(\theta)}{A_m} - \sum_{m=i}^{n-1} \frac{\exp(-A_m\theta) - \exp(-A_n\theta)}{A_m(A_n - A_m)} * G_{j,m\to m+1}(\theta) * \prod_{\substack{k=i \\ k\neq m}}^{n-1} \frac{G_{j,k\to k+1}(\theta)}{A_k - A_m}. \qquad (VI.6\text{-}4)$$

In the above equations the term An represents the total specific loss rate of a nuclide, neglecting time change of neutron flux,

$$A_n = \lambda_n + 10^{-24} \sum_g \Phi_{g,z}\sigma_{a,n,g} \qquad (VI.6\text{-}5)$$

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

$$G_{j,n\to n+1}(\theta) = \begin{cases} 10^{-24} \sum_g \Phi_{g,z}\sigma_{c,n,g} & by\ capture \\ \lambda_{j,n} & by\ decay \\ H_n\lambda_{j,n} & by\ partial\ decay \\ 10^{-24}\, E_n \sum_g \Phi_{g,z}\sigma_{c,n,g} & by\ partial\ capture \\ 10^{-24} \sum_g \Phi_{g,z}\sigma_{n2n,n,g} & by\ n\text{-}2n\ \ reaction \end{cases} \qquad (VI.6\text{-}6)$$

depending on the chain route specifications, where Hn and En denote the fraction of each reaction to the nuclide n.

It is assumed that FP nuclides are arranged in the order of all fissile nuclides, and strong absorbers. The end-of-step concentrations are calculated by processing the chain relationship specified. When the first FP nuclide is encountered, all the yield rates in Eq. (VI.6-2) are calculated as

$$Y_n(\theta) = 10^{-24} \sum_m Y_{m\to n} \frac{N_m(t) + N_m(t+\theta)}{2} \sum_g \Phi_{g,z}\sigma_{f,n,g} \qquad (VI.6\text{-}7)$$

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.6-7).

For a non-depleting, one-nuclide chain, the equation solved is

$$N_n(t) = N_n(t-\theta) + \theta Y_n(\theta). \qquad (VI.6\text{-}8)$$

The code has provision for a strong absorbing, one-nuclide chain (*e.g.* burnable poison), to be solved as,

$$N_n(t) = N_n(t-0)\exp(-A_n\theta) \qquad (VI.6\text{-}9)$$

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.

VI.7 Reactivity Calculation by Perturbation Theory

The perturbation theory is used to estimate a small reactivity. Although the CITATION code is a component routine in the SRAC code system, the function installed in the original code[53] has not been utilized since it requires the microscopic cross sections which are not provided in the SRAC code system.

A new routine is made to feed the macroscopic cross sections into the first order perturbation calculation. The forward and adjoint fluxes have to be prepared by the installed CITATION routine.

A perturbed region can be specified by the following three modes;
(1) A whole volume specified by a zone number,
(2) A volume specified by mesh numbers of boundaries,
(3) A volume expressed by abscissa of boundaries,
and several volumes can be simultaneously accepted in a case. As shown in the input specification in Sect.II.10, a mixed mode of (2) and (3) accepted.

By specifying a new mixture to replace the original mixture located at the perturbed region, the change is automatically calculated from the difference of cross sections of two mixtures.

The following results will be printed;

1) The macroscopic cross sections of the new mixture.
2) The volume of the perturbed region and that of the zone including the region.
3) The whole reactivity change, reactivity changes by reaction and by energy group, and their fractions to the whole.
4) The reactivity due to the change of leakage on each outer surface and by energy group.
5) The summation over total perturbed regions

The following restrictions have to be noted.

1) A perturbed region can not extend over two zones. If necessary, a region should be split into two regions.
2) When the geometry is 3D triangular or 3D hexagonal, the region specification by abscissa is accepted only in Z-direction.

VI.7.1 First order perturbation

The reactivity change ρ by the first order perturbation theory is given by

$$\rho = \frac{\Delta F - \Delta A + \Delta S - \Delta L - \Delta DB^2}{F} \qquad (VI.7\text{-}1)$$

where F is the total fission, ΔF the fission term, ΔA the absorption term, ΔS the scattering term, ΔL the leakage term, ΔDB^2 the pseudo leakage term. These are given by

$$F = \frac{1}{K}\sum_{g}\int \chi_g(r)\varphi_g^*(r)\sum_{g'}\nu\Sigma_{fg'}(r)\varphi_{g'}(r)dV \qquad (VI.7\text{-}2)$$

$$\Delta F_g = \frac{1}{K}\int \varphi_g^*(r)\sum_{g'}\{X_g\Delta\nu\Sigma_{fg'}(r) + \Delta X_g\nu\Sigma_{fg'}(r)\}\varphi_{g'}(r)dV \quad (VI.7\text{-}3)$$

$$\Delta A_g = \int \varphi_g^*(r)\Delta\Sigma_{ag}(r)\varphi_g(r)dV \quad (VI.7\text{-}4)$$

$$\Delta S_g = \int\sum_{g}(\varphi_{g'}^*(r)-\varphi_g^*(r))\Delta\Sigma_{sg\to g'}(r)\varphi_{g'}(r)dV \quad (VI.7\text{-}5)$$

$$\Delta L_g = \int \nabla\varphi_g^*(r)\Delta D_g(r)\nabla\varphi_g(r)dV$$

$$= \int \{\frac{\partial\varphi_g^*(r)}{\partial x}\Delta D_{xg}(r)\frac{\partial\varphi_g(r)}{\partial x}$$

$$+ \frac{\partial\varphi_g^*(r)}{\partial y}\Delta D_{yg}(r)\frac{\partial\varphi_g(r)}{\partial y} + \frac{\partial\varphi_g^*(r)}{\partial z}\Delta D_{zg}(r)\frac{\partial\varphi_g(r)}{\partial z}\}\,dV \quad (VI.7\text{-}6)$$

$$\Delta DB_g^2 = \int \varphi_g^*(r)(\Delta D_{zg}(r)B_g^2 + D_{zg}(r)\Delta B_g^2(r))\varphi_g(r)dV \quad (VI.7\text{-}7)$$

$$\Delta F = \sum_g \Delta F_g \quad (VI.7\text{-}8)$$

$$\Delta A = \sum_g \Delta A_g \quad (VI.7\text{-}9)$$

$$\Delta S = \sum_g \Delta S_g \quad (VI.7\text{-}10)$$

$$\Delta L = \sum_g \Delta L_g \quad (VI.7\text{-}11)$$

$$\Delta DB^2 = \sum_g \Delta DB_g^2 \quad (VI.7\text{-}12)$$

$$\Delta\nu\Sigma_{fg}(r) = \nu\Sigma_{fg}^p(r) - \nu\Sigma_{fg}(r) \quad (VI.7\text{-}13)$$

$$\Delta X_g(r) = X_g^p(r) - X_g(r) \quad (VI.7\text{-}14)$$

$$\Delta\Sigma_{ag}(r) = \Sigma_{ag}^p(r) - \Sigma_{ag}(r) \quad (VI.7\text{-}15)$$

$$\Delta\Sigma_{sg\to g'}(r) = \Sigma_{sg\to g'}^p(r) - \Sigma_{sg\to g'}(r) \quad (VI.7\text{-}16)$$

$$\Delta D_g(r) = D_g^p(r) - D_g(r) \quad (VI.7\text{-}17)$$

$$\Delta B_g^2(r) = B_g^{2p}(r) - BD_g^2(r) \quad (VI.7\text{-}18)$$

The superscript p denotes the perturbed cross sections.

VI.7.2 Finite-Difference equation for the leakage term

The term appearing in the leakage term $\nabla\varphi_g^*(r)\nabla\varphi_g(r)\,dV$ is approximated as shown below.

$$\sum_j A_j \frac{1}{\Delta_i} \left\{ \frac{\Delta_i D_{jg}}{\Delta_i D_{jg} + \Delta_j D_{ig}} \right\}^2 \left\{ \varphi^*_{ig} - \varphi^*_{jg} \right\} \left\{ \varphi_{ig} - \varphi_{jg} \right\}$$

in the bulk of the system

$$\sum_j A_j \frac{1}{\Delta_i} \left\{ \frac{\Delta_i C_{sg}}{\Delta_i C_{sg} + \Delta_j D_{ig}} \right\}^2 \varphi^*_{ig} \varphi_{ig}$$

at the black boundary of the system

where the subscript i denotes a flux point, j the neighbouring flux point, A_j the leakage area, Δ_i the distance between the flux point i and the leakage surface, Δ_j the distance between the flux point j and the leakage surface, and C_{sg} the black boundary constant. At the reflective boundary the above term is null.

VI.7.3 Sample results

The reactivity effect of an irradiation sample of MOX pin enclosed in SUS capsule instead of the Al plug inserted into the central irradiation hole of the JRR-3M is calculated. It is to be noted that the reactivity effect in this sample case is too big to apply the first-order perturbation theory so that the results changes by the process. It is further emphasized by applying 2D core calculations while the actual sample length is much shorter than the core height.

The following five processes are examined;

Process A Calculate the forward and adjoint fluxes for the core with the Al-plug in the irradiation hole as the standard.
Process B Calculate the forward and adjoint fluxes for the core with the sample in the irradiation hole as the standard.
Process C Use the forward flux of Case B and the adjoint of Case A
Process D Use the forward flux of Case A and the adjoint of Case B

We can get the reactivity effect also by the direct calculation (difference of k_{eff}'s between Case A and Case B).

The results are compared in Table VI.7.4-1. The negative value of reactivity comes from the absorption by SUS which overrides the positive effect of MOX fuel. We can see the absorption effect by the sample depresses the forward and adjoint fluxes near the irradiation hole.

Table VI.7.3-1
The reactivity effect of a MOX pin in the central irradiation hole

Case	ρ (%Δk/k)	ratio
Direct	-0.7345	-------
A	-1.0721	1.46
B	-0.6916	0.94
C	-0.8627	1.17
D	-0.8704	1.18

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	Each Be O comp. given
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	Each U, O comp. given
Water light	H	H2O	Add O comp. by free
Water heavy	D	D2O	Add O comp. by free
Zirconium hydrate	Z	ZrH	Each Zr H comps. given
Simple	0	Free atom	

VII.3 Nuclides in SRAC Public Libraries

ENDF/B-IV Version

Nuclide	Key code	Thermal scatter	Thermal F-TAB	Fast F-TAB	Reso para	Orig File Mat No.	vol
H-001	XH01000A	free P1				ENDFB 1269	4
H-001H	XH01H008	S(αβ)P5		thermal only		ENDFB 1002	3
H-001Q	XH01Q00A	S(αβ)P1		thermal only		ENDFB 1095	3
H-001Z	XH01Z00A	S(αβ)P1		thermal only		ENDFB 1097	3
H-001P	XH01P004	S(αβ)P1		thermal only		ENDFB 1114	3
D-002	XD02000A	free P1		1-74		ENDFB 1120	4
D-002D	XD02D008	S(αβ)P1		thermal only		ENDFB 1004	3
T-003	XT03000A	free		1- 7		ENDFB 1169	4
HE003	XHE3000A	free		1-74		ENDFB 1146	4
LI006	XLI6000A	free		1-62		ENDFB 1271	4
LI007	XLI7000A	free		1-74		ENDFB 1272	4
BE009	XBE90009	free P1		1-74		ENDFB 1289	4
BE009E	XBE9E00A	S(αβ)P1		thermal only		ENDFB 1099	3
BE009B	XBE9B00A	S(αβ)P1		thermal only		ENDFB 1064	3
B-010	XB00000A	free		1-74		ENDFB 1273	4
B-011	XB01000A	free		1-74		ENDFB 1160	4
C-012	XC02000A	free P1		1-74		ENDFB 1274	4
C-012C	XC02C00A	S(αβ)P1		thermal only		ENDFB 1065	3
C-012Q	XC02Q001	S(αβ)P1		thermal only		ENDFB 1095	3
N-014	XN04000A	free P1		1-74		ENDFB 1275	4
O-016	XO06000A	free P1		1-16		ENDFB 1276	4
O-016E	XO06E001	S(αβ)P1		thermal only		ENDFB 1099	3
O-016W	XO06W00A	S(αβ)P1		thermal only		ENDFB 1167	3
F-019	XF090001	free				ENDFB 1277	4
NA023	XNA3000A	free		1-74		ENDFB 1156	4
MG000	XMGN000A	free		1-74		ENDFB 1280	4
AL027	XAL70009	free P1		1-74		ENDFB 1193	4
SI000	XSIN0001	cap		1-74		ENDFB 1194	4
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	:	:	1-74	:	:ENDFB 1191	4
MN055	: XMN50001	: cap	:	:	1-74	:	:ENDFB 1197	4
FE000	: XFEN000A	: free	:	:	1-74	:	:ENDFB 1192	4
CO059	: XCO90001	: free	:	:		:	:ENDFB 1199	4
NI000	: XNIN000A	: free	:	:	3-34	:	:ENDFB 1190	4
CU000	: XCUN0001	: cap	:	:	4-74	:	:ENDFB 1295	4
CU063	: XCU30001	: cap	:	:thermal only			:ENDFB 6411	4
KR083	: XKR30001	: cap	:	:		:	:ENDFB 1184	4
ZRal2	: XZRN000A	: free	:	:	2-74	:	:ENDFB 1284	4
ZR095	: XZR50001	: cap	:	:		:	:ENDFB	4
ZR000Z	: XZRNZ00A	: S(αβ)P1	:	:thermal only			:ENDFB 1096	3
NB093	: XNB30001	: free	:	:		:	:ENDFB 0236	4
MO000	: XMON000A	: free	:	:	2-74	:	:ENDFB	4
MO095	: XMO50001	: cap	:	:		:	:ENDFB	4
MO097	: XMO70001	: cap	:	:		:	:ENDFB	4
MO098	: XMO80001	: cap	:	:		:	:ENDFB	4
MO099	: XMO90001	: cap	:	:		:	:ENDFB 0269	4
TC099	: 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	:	:	1-74	: 1	:ENDFB 1138	4
AG109	: XAG9000A	: free	:	:	1-74	: A	:ENDFB 1139	4
CD000	: XCDN000A	: free	:	:	5-74	:	:ENDFB 1281	4
CD113	: XCD30001	: cap	:	:		:	:ENDFB 1282	4
IN113	: XIN3000A	: cap	:	:	1-74	: 1	:ENDFB 0445	4
IN115	: XIN5000A	: cap	:	:	1-74	: 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	:	:	4-74	:	:ENDFB 1294	4
XE136	: XXE60001	: cap	:	:		:	:ENDFB 1178	4
CS133	: XCS30001	: cap	:	:		:	:ENDFB 0613	4
CS134	: XCS40001	: cap	:	:		:	:TAKANO0434	S
CS135	: XCS50001	: cap	:	:		:	:ENDFB 0616	4
PR141	: XPR30001	: cap	:	:		:	:ENDFB	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	:	:	1-74	:	: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	XGDN000A	free			1-74		ENDFB 1030	4
GD154	XGD40001	cap					ENDFB 0791	4
GD155	XGD50001	cap			1-74	B	ENDFB 0792	4
GD156	XGD60001	cap					ENDFB 0793	4
GD157	XGD70001	cap			1-74	B	ENDFB 0794	4
GD158	XGD80001	cap					ENDFB 0795	4
DY164	XDY40001	cap					ENDFB 1031	4
ER166	XER60001	cap			1-74		ENDFB 0823	4
ER167	XER70001	cap			1-74		ENDFB 0824	4
LU175	XLU50001	cap					ENDFB 1032	4
LU176	XLU60001	cap					ENDFB 1033	4
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			2-74	1	ENDFB 1283	4
PB000	XPBN000A	free			1-74		ENDFB 1288	4
TH232	XTH2000A	free			23-74	B	ENDFB 1296	4
PA233	XPA3000A	free			1-74		ENDFB 1297	4
U-233	XU03000A	free	1	-48	51-74	B	ENDFB 1260	4
U-234	XU04000A	free			38-74	B	ENDFB 1043	4
U-235	XU05000A	free	1	-48	26-74	B	ENDFB 1261	4
U-236	XU06000A	free			38-74	B	ENDFB 1163	4
U-238	XU08000A	free			23-74	B	ENDFB 1262	4
U-238W	XU08W00A	S($\alpha\beta$)P1			thermal only		ENDFB 1167	3
PU239	XPU9000A	free	1	-48	26-74	B	ENDFB 1264	4
PU240	XPU0000A	free	1	-48	26-74	B	ENDFB 1265	4
PU241	XPU1000A	free	1	-48	23-74	B	ENDFB 1266	4
PU242	XPU2000A	free			29-74	B	ENDFB 1161	4
AM141	XAM10001	cap			1-74		ENDFB	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	for IIJIMA model				TAKANO0400	S
pseu25	XPS20001	cap	for VSOP25 model				VSOP 163	
pseu40	XPS40001	cap	for VSOP40 model				VSOP 160	

JENDL-2 Version

Nuclide	Key code	Thermal scatter	Thermal F-TAB	Fast F-TAB	Reso para	Orig File	Matn	Tape vol
H-001	XH01000A	fast only		1-74		JENDL 2011	201	2
LI006	XLI60001	free		thermal only		JENDL 2031		2
LI007	XLI70001	free		thermal only		JENDL 2032		2
B-010	XB000001	cap		1-74		JENDL 2051		2

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C-012  : XC02000A : fast only :   1-74 :      :JENDL 2061      2
F-019  : XF090001 : free      :   1-74 :      :JENDL 2091      2
NA023  : XNA30001 : free      :   1-74 :      :JENDL 2111      2
AL027  : XAL7000A : free      :   1-74 :      :JENDL 2131      2
SI000  : XSIN0001 : free      :   1-74 :      :JENDL 2140      2
S-032  : XS020001 : fast only :        :      :ENDL
CA000  : XCAN0001 : free      :thermal only :JENDL 2200      2
SC045  : XSC50001 : cap       :thermal only :JENDL 2211      2
V-051  : XV010001 : free      :thermal only :JENDL 2231      2
CR000  : XCRN0001 : free      :   1-74 :      :JENDL 2240 202  2
MN055  : XMN50001 : free      :   1-74 :      :JENDL 2251      2
FE000  : XFEN0001 : free      :   1-74 :      :JENDL 2260      2
CO059  : XC090001 : free      :thermal only :JENDL 2270      2
NI000  : XNIN0001 : free      :   1-74 :      :JENDL 2280      2
CU000  : XCUN0001 : free      :   1-74 :      :JENDL 2290 203  2
CU063  : XCUN0001 : cap       :   1-74 :      :JENDL 2291      2
CU065  : XCUN0001 : cap       :   1-74 :      :JENDL 2292      2
KR083  : XKR30001 : cap       :thermal only :JENDL 2361 207  2
KR084  : XKR40001 : cap       :thermal:only :JENDL 2362      2
KR085  : XKR50001 : cap       :thermal:only :JENDL 2363      2
KR086  : XKR60001 : cap       :thermal:only :JENDL 2364      2
RB085  : XRB50001 : cap       :thermal:only :JENDL 2371      2
RB087  : XRB70001 : cap       :thermal:only :JENDL 2372      2
SR000  : XSRN0001 : fast only :   1-74 :      :JENDL 2380      2
SR086  : XSR60001 : cap       :thermal:only :JENDL 2381      2
SR087  : XSR70001 : cap       :thermal:only :JENDL 2382      2
SR088  : XSR80001 : cap       :thermal:only :JENDL 2383      2
SR090  : XSR00001 : cap       :thermal:only :JENDL 2384      2
Y-089  : XY090001 : cap       :thermal:only :JENDL 2391      2
ZR090  : XZR00001 : cap       :thermal:only :JENDL 2401      2
ZR091  : XZR10001 : cap       :thermal:only :JENDL 2402      2
ZR092  : XZR20001 : cap       :thermal:only :JENDL 2403      2
ZR093  : XZR30001 : cap       :thermal:only :JENDL 2404      2
ZR094  : XZR40001 : cap       :thermal:only :JENDL 2405      2
ZR095  : XZR50001 : cap       :thermal:only :JENDL 2406      2
ZR096  : XZR60001 : cap       :thermal:only :JENDL 2407      2
NB093  : XNB30001 : cap       :thermal only :JENDL 2411      2
MO000  : XMON0001 : free      :   1-74 :      :JENDL 2420 203  2
MO092  : XMO20001 : cap       :thermal only :JENDL 2421      2
MO094  : XMO40001 : cap       :thermal only :JENDL 2422      2
MO095  : XMO50001 : cap       :thermal only :JENDL 2423      2
MO096  : XMO60001 : cap       :thermal only :JENDL 2424      2
MO097  : XMO70001 : cap       :thermal only :JENDL 2425      2
MO098  : XMO80001 : cap       :thermal only :JENDL 2426      2
MO100  : XMO00001 : cap       :thermal only :JENDL 2427      2
TC099  : XTC90001 : cap       :        :    :JENDL 2431 208  2
RU100  : XRU00001 : cap       :thermal:only :JENDL 2441      2
RU101  : XRU10001 : cap       :        :    :JENDL 2442      2
RU102  : XRU20001 : cap       :thermal:only :JENDL 2443      2
RU104  : XRU40001 : cap       :thermal:only :JENDL 2444      2
RU106  : XRU60001 : cap       :thermal:only :JENDL 2445      2
RH103  : XRH30001 : cap       :        :    :JENDL 2451      2
PD104  : XPD40001 : cap       :thermal:only :JENDL 2461      2
PD105  : XPD50001 : cap       :        :    :JENDL 2462      2
PD106  : XPD60001 : cap       :thermal:only :JENDL 2463      2
PD107  : XPD70001 : cap       :        :    :JENDL 2464      2
PD108  : XPD80001 : cap       :thermal:only :JENDL 2465      2
PD110  : XPD00001 : cap       :thermal:only :JENDL 2466      2
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AG107	XAG90001	cap		thermal	only	JENDL 2471		2
AG109	XAG90001	cap		1-74		JENDL 2472		2
CD110	XCD00001	cap		thermal	only	JENDL 2481		2
CD111	XCD10001	cap		thermal	only	JENDL 2482		2
CD112	XCD20001	cap		thermal	only	JENDL 2483		2
CD113	XCD30001	cap		thermal	only	JENDL 2484		2
CD114	XCD40001	cap		thermal	only	JENDL 2485		2
CD116	XCD60001	cap		thermal	only	JENDL 2486		2
IN115	XIN50001	cap		thermal	only	JENDL 2491		2
SB121	XSB10001	cap		thermal	only	JENDL 2511		2
SB123	XSB30001	cap		thermal	only	JENDL 2512		2
SB124	XSB40001	cap		thermal	only	JENDL 2513		2
TE128	XTE80001	cap		thermal	only	JENDL 2521		2
I-129	XI090001	cap		thermal	only	JENDL 2531	209	2
I-129	XI090001	cap		thermal	only	JENDL 2532		2
XE131	XXE10001	cap				JENDL 2541		2
XE132	XXE20001	cap		thermal	only	JENDL 2542		2
XE133	XXE30001	cap		thermal	only	JENDL 2543		2
XE134	XXE40001	cap		thermal	only	JENDL 2544		2
XE135	XXE50001	cap		thermal	only	JENDL 2545		2
XE136	XXE60001	cap		thermal	only	JENDL 2546		2
CS133	XCS30001	cap				JENDL 2551		2
CS135	XCS50001	cap				JENDL 2552		2
CS137	XCS70001	cap		thermal	only	JENDL 2553		2
BA134	XBA40001	cap		thermal	only	JENDL 2561		2
BA135	XBA50001	cap		thermal	only	JENDL 2562		2
BA136	XBA60001	cap		thermal	only	JENDL 2563		2
BA137	XBA70001	cap		thermal	only	JENDL 2564		2
BA138	XBA80001	cap		thermal	only	JENDL 2565		2
LA139	XLA90001	cap		thermal	only	JENDL 2571		2
CE140	XCE00001	cap		thermal	only	JENDL 2581		2
CE142	XCE20001	cap		thermal	only	JENDL 2582		2
CE134	XCE40001	cap		thermal	only	JENDL 2583		2
PR141	XPR10001	cap		thermal	only	JENDL 2591		2
ND142	XND20001	cap		thermal	only	JENDL 2601		2
ND143	XND30001	cap				JENDL 2602		2
ND144	XND40001	cap		thermal	only	JENDL 2603		2
ND145	XND50001	cap				JENDL 2604		2
ND146	XND60001	cap		thermal	only	JENDL 2605		2
ND148	XND80001	cap		thermal	only	JENDL 2606		2
ND150	XND00001	cap		thermal	only	JENDL 2607		2
PM147	XPM70001	cap				JENDL 2611	210	2
SM147	XSM70001	cap				JENDL 2621		2
SM148	XSM80001	cap		thermal	only	JENDL 2622		2
SM149	XSM90001	cap				JENDL 2623		2
SM150	XSM00001	cap		thermal	only	JENDL 2624		2
SM151	XSM10001	cap				JENDL 2625		2
SM152	XSM20001	cap		thermal	only	JENDL 2626		2
SM154	XSM40001	cap		thermal	only	JENDL 2627		2
EU151	XEU10001	cap		thermal	only	JENDL 2631		2
EU152	XEU20001	cap		thermal	only	JENDL 2632		2
EU153	XEU30001	cap				JENDL 2633		2
EU154	XEU40001	cap				JENDL 2634		2
EU155	XEU50001	cap		thermal	only	JENDL 2635		2
GD155	XGD50001	cap		thermal	only	JENDL 2641		2
GD156	XGD60001	cap		thermal	only	JENDL 2642		2
GD157	XGD70001	cap		thermal	only	JENDL 2643		2
GD158	XGD80001	cap		thermal	only	JENDL 2644		2

GD160	XGD00001	cap			thermal	only	JENDL 2645	2
TB159	XTB90001	cap			thermal	only	JENDL 2651	2
HF000	XHFN0001	free		38-74	B		TAKANO0872	S
HF174	XHF40001	free		48-74	B		JENDL 2721 204	S
HF176	XHF60001	free		38-74	B		JENDL 2722	S
HF177	XHF70001	free		41-74	B		JENDL 2723	S
HF178	XHF80001	free		38-74	B		JENDL 2724	S
HF179	XHF90001	free		38-74	B		JENDL 2725	S
HF180	XHF00001	free		41-74	B		JENDL 2726	S
TA181	XTA10001	cap			thermal only		JENDL 2731	2
PB000	XPBN0001	free			thermal only		JENDL 2820	2
TH232	XTH20001	free		23-74	B		JENDL 2903 205	2
PA233	XPA30001	cap			thermal only		JENDL 2911	2
U-233	XU030001	free		26-74	B		JENDL 2921	2
U-234	XU040001	free		48-74	B		JENDL 2922	2
U-235	XU050001	free		26-74	B		JENDL 2923	2
U-236	XU060001	free		38-74	B		JENDL 2924	2
U-238	XU080001	free		23-74	B		JENDL 2925	2
NP239	XNP90001	cap			thermal only		JENDL 2931	2
PU239	XPU90001	free		26-74	B		JENDL 2943 206	2
PU240	XPU0000A	free	1 -48	26-74	B		JENDL 2944	2
PU241	XPU10001	free		26-74	B		JENDL 2945	2
PU242	XPU20001	free		38-74	B		JENDL 2946	2
AM141	XAM10001	cap		48-74	B		JENDL 2951	2
AM143	XAM30001	cap			thermal only		JENDL 2954	2

****

Note 1. in Key code column
'n' denotes that the tabulation on the first n temperatures *ex.* '1' for no temperature dependency, and 'A' the temperature dependent matrices prepared on 10 temperatures.

Note 2. in Thermal scatter column

T	free	: scattering matrices calculated by free gas model
	$S(\alpha\beta)$	: scattering matrices calculated from tabulated $S(\alpha\beta)$
	cap	: only capture cross sections stored
T	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' group

Note 4. in Fast FTAB column
n n' : shielding factor tabulation starting from n to n' group

Note 5. in Res lib column
n : ultra-fine cross sections for n temperature points are stored in MCROSS librarye.
ex. '1' denotes on 300K only, 'B' on 11 points.

Note 5. in Orig File and Mat No. column
ENDFB nnnn m: material no. in ENDF/B version m
ENDL : material no. in ENDL
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	0.0852
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
   Uniform lethargy width         0.00125
   Number of ultra fine groups      4600
       on cross section table
   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.SEC/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 model
98	F3N	0	0	0.0	0.0 pseudo FP in Garrison model
99	F3S	0	0	0.0	0.0 pseudo FP in Garrison model
100	F3R	0	0	0.0	0.0 pseudo FP in Garrison model
101	F5N	0	0	0.0	0.0 pseudo FP in Garrison model
102	F5S	0	0	6.00000E-10	0.0 pseudo FP in Garrison model
103	F5R	0	0	2.10000E-10	0.0 pseudo FP in Garrison model
104	F9N	0	0	0.0	0.0 pseudo FP in Garrison model
105	F9S	0	0	4.40000E-10	0.0 pseudo FP in Garrison model
106	F9R	0	0	1.66000E-10	0.0 pseudo FP in Garrison model
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

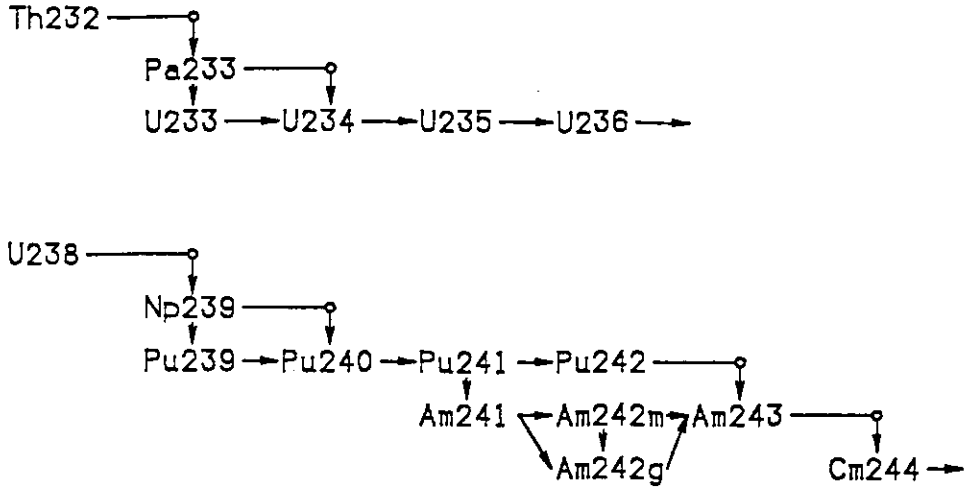
The routine is programed to process (n,2n) reaction which is, however, neglected in the present treatment because it is not so important in thermal reactors.

The simple series compliled in the initial version which is expresseed as by

Th232 => Pa233 => U233 => U234 => U235 => U236

U238 => Pu239 => Pu240 => Pu241 => Pu242

have been replaced in the latest version[1] in order to consider a high coversion and high burn-up LWR by those shown below



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 & Roos[2] 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.[3] 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.

F.P. chains in Iijima model

1) Kr83

2) Mo99 => Tc99

3) Ru101

4) Ru103 => Rh103

5) Ru105 => Rh105 => Pd105

6) Pd107 => Pd108 =>(Pd109)=> Ag109

7) Cd113

8) I 131 => Xe131

9) Xe133 => Cs133 => Cs134

10) I 135 => Xe135 => Cs135

11) Pr143 => Nd143

12) Nd145

13) Nd147

```
          |       Pmm148
Pm147 =>          => Pm149 =>(Pm150) Pm151
          |       Pmg148       |         |         |
          |         |           |         |         |
Sm147 => Sm148  => Sm149 => Sm150 => Sm151 => Sm152 :α

α: => Eu153 => Eu154 => Eu155 => Eu156
                 |
               Gd154 => Gd155 => Gd156 => Gd157 => Gd158
```

14) pseudo

4.3) V.S.O.P models

The code system V.S.O.P.[4] 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

1) Tc99

2) Ru103 => Rh103

3) Rh105

4) Xe131

5) Xe133 => Cs133 => Cs134

6) Pr143 =>(Pr144)
| |
Nd143 => Nd144 => Nd145 => Nd146 :α

Pmm148
α: => Pm147 => =>(Pm149)
| Pmg148 |
| | |
Sm147 => Sm148 => Sm149 => Sm150 => Sm151 => Sm152 :β

β: => Eu153 => Eu154 => Eu155

7) I 135 => Xe135 => Cs135

8) pseudo25

F.P. chains in VSOP 40

1) Kr83

2) Zr95 => Mo95

3) Mo97

4) Tc99

5) Ru101

6) Ru103 => Rh103

7) Rh105 => Pd105

8) Pd108 => Ag109

9) Cd113

10) I 131 => Xe131

11) Xe133 => Cs133 => Cs134

12) Xe136

13) Pr141

14) Pr143 =>(Pr144)
| |
Nd143 => Nd144 => Nd145 => Nd146 :α

Pmm148

```
α: => Pm147 =>          =>(Pm149)
        |       Pmg148       |
        |         |           |
      Sm147 => Sm148  => Sm149 => Sm150 => Sm151 => Sm152 :β

β: => Eu153 => Eu154 => Eu155 =>(Eu156)
                          |        |
                        Gd155 => Gd156 => Gd157 => Gd158
```

15) I 135 => Xe135 => Cs135

16) pseudo40

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 Auxiliary Programs

VIII.1 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 flux of a specified X-region, 3) Energy dependence of macroscopic cross sections for specified reactions.

Another code MACRPR permits printing the above quantities.

VIII.1.1 Program MFPLOT

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 n-th X-region. If the same member is found in the MACRO file, the corresponding macroscopic cross section will be plotted.

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

If MACRO(I) =0 Skip plotting
If MACRO(I) =1 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	$\nu\Sigma$f	ν*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 cross sections and the neutron spectra and the latter is for the plot of the spatial distribution of neutron fluxes.

The group flux in fast or whole energy range is plotted per unit lethargy and that in thermal range per unit energy.

The following example requires plotting of fine group information. If the coarse group one is simultaneously needed, concatenate the file which contains the coarse group cross sections after DD name MACRO. 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.SRACSC',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 I.1 case name,thermal, spatial distribution, fine
1 / I.2' no plot for cross sections
3     10 15 25 26 29/ I.3' cylinder , group numbers
TRX1T012 / Sample 2 I.1 case name, thermal,  x=1,  fine
0 1 0 1 0 1 0 1 / I.2 reactions for cross section plot
1/ I.3 plot neutron spectrum
TRX1A012 / Sample 3 I.1 case name, all, x=1, fine
0 1 0 1 0 1 0 1 / I.2 reactions for cross sections
1/ I.3 plot neutron spectrum
MOD4TM62 / Sample 4 I.1mixture name, thermal , fine
0 1 0 1 0 1 0 1 / I.2reactions for cross sections
0/ I.3  skip plot neutron spectrum
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.

VIII.1.2 Program MACRPR

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

++
//
```

VIII.2 PDS File Edit Programs

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

VIII.2.1 Program PDSEDT

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

```
    PRINT,PUNCH,.....,FIN :  command code
    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 for *PUT or *UPDATE
```

At the first of execution, an integer is required to notify the open mode of the USERPDS file; 1 for read only, 2 for write only, 3 for read/write. The character '*' is required to compose the command code on column 1 of input record . The 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 by 'list'.
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: copy the member NAME1 in DDN=USERPDS to DDN=USERPDS2 by the new member 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.SRACSC',PNM=PDSEDF
//USERPDS DD DSN=J1480.THERMLB2.DATA,DISP=MOD
//SYSIN DD *
   2  $  1 read only,  2 write only,  3 read/write
*UPD  CH01P000 I 2 2
  6 / update location 2 by the value 6
*FIN
//
```

Sample Macroscopic TSS Command for PDSEDT

```
PROC 1 NAME1 NAME2(OFF) NAME3(OFF) DISP(SHR)
/* NAME1 PDS FILE TO READ WRITE DELETE RENAME DDN=USERPDS
/* NAME2 PDS FILE TO TRANSFER                DDN=USERPDS2
/* NAME3 PS FILE TO READ WRITE IN BINARY MODE DDN=FT08F001
CONTROL PROMPT LIST MSG
FREE DA(&NAME1)
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)
WRITE REPLY INPUT OR OUTPUT FOR USERPDS FILE
READ IO
ATTR C &IO
ALLOC DA(&NAME1) F(USERPDS) USING(C) DISP(&DISP)
IF &NAME2 NE OFF THEN DO
FREE DA(&NAME2)
ALLOC DA(&NAME2) F(USERPDS2) DISP(OLD)
END
IF &NAME3 NE OFF THEN DO
FREE F(FT08F001)
FREE DA(&NAME3)
ALLOC DA(&NAME3) F(FT08F001)
END
CALL 'J1480.SRACSC.LOAD(PDSEDF)'
FREE DA(&NAME1)
FREE ATTRLIST(A)
IF &NAME2 NE OFF THEN FREE DA(&NAME2)
IF &NAME3 NE OFF THEN FREE DA(&NAME3)
END
```

VIII.2.2 Program PDSEDGRP

The program PDSEDGRP executes the commands as the PDSEDT does to a group of members.

At the beginning of the execution, an integer is required to specify the open mode of the USERPDS file; 1 to read only, 2 to write only, 3 to read/write.

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 into FT10F001, 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.

Sample Macroscopic TSS Command for 'PDSEDGRP'

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

```
PROC 1 NAME1 NAME2(OFF) NAME3(OFF) NAME4(OFF) DISP(SHR)
/* 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)
WRITE REPLY INPUT OR OUTPUT FOR USERPDS FILE
READ IO
ATTR C &IO
ALLOC DA(&NAME1) F(USERPDS) USING(C) DISP(&DISP)
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(PDSEDG)'
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(PDSEDG)'
   REPLY INPUT OR OUTPUT FOR USERPDS
     OUTPUT
   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
```

VIII.2.3 Program SPFREE

The program SPFREE liberates the old file for a new usage. After the execution, the directory and the whole extent of a file are cleaned

up.

This program is normally used by a TSS command "SPFREE" defined as follows;

```
/* Content of the member TSSMAC.CLIST(SPFREE) */
      PROC 1 FILENAME
      CONTROL PROMPT LIST MSG
      ATTR A  LRECL(133) RECFM(U)
      ATTR B  OUTPUT
      FREE F(FT06F001)
      ALLOC DA(*) F(FT06F001) USING(A)
      ALLOC DA(&FILENAME) F(USERPDS) OLD USING(B)
      X CALL 'J1480.SRACSC.LOAD(SPFREE)'
      FREE ATTRLIST(B)
      FREE ATTRLIST(A)
      FREE DA(&FILENAME)
      EXIT
```

The command is activated by a message as 'SPFREE MACFF.DATA', where MACFF.DATA is the file to be liberated by the user.

Appendix Sample Input and Output

We show a list of a sample input and this output. In the sample execution, three cases (in the term of SRAC) are included: first for a typical LWR pin rod cell, second the double spaced pin cell, third, a 2D diffusion calculation using the results of the previous cell calculations.

The output list is organized by two files: one on FT06F001 mainly for tracing the process and file management, the other on FT99F001 for the numerical results.

```
          **********************
          *                    *
          *  INPUT DATA LIST   *
          *                    *
          **********************

     ....*....1....*....2....*....3....*....4....*....5....*....6....*....7....*....8
 1   TCA1          ¥ Sample input for a TCA non-uniform core                    00000010
 2   TCA1 STANDARD CELL     CYLINDRICAL APPROX.                                00000020
 3    1 1 1 1 2  1 4 0 -2 1  0 0 0 0 1  2 2 0 0 0 / Block 3 II.1               00000030
 4    .0080 / Geometrical buckling to give Keff=1 at B1 one point calculation   00000040
 5   J1480              0 0/ PUBLIC FAST                                        00000050
 6   J1480              0 0/ PUBLIC THERMAL                                     00000060
 7   J1480              0 0/ USER FAST                                          00000070
 8   J1480              0 0/ USER THERMAL                                       00000080
 9   J1480              0 0/ NOT EFFECTIVE                                      00000090
10   J1480              0 0/ NOT EFFECTIVE                                      00000100
11   J1480              0 0/ NOT EFFECTIVE                                      00000110
12   J1480              0 0/ NOT EFFECTIVE                                      00000120
13   22 31 5  5  / NEF NET NERF NERT                                            00000130
14    7(2) 6(4) 8(3) 4       / FINE FAST                                       00000140
15   2(4) 2 2 1 1 2 2 23(1) / FINE THERMAL                                      00000150
16    4 4 4 5 5              / COARSE FAST                                     00000160
17     6 6 6 6 7             / COARSE THERMAL                                  00000170
18   XH01H001                                                                  00000180
19   X0060001                                                                  00000190
20   XU050001                                                                  00000200
21   XU080001                                                                  00000201
22   XAL70001                                                                  00000201
23                                                                             00000210
24     3 5 5 5 1  0 5 0 0 0 10 0 6 1 0  0 0 0   / COLLISION PROB. CONTROL     00250000
25     0 6(0) 6(0.) /  ITERATION PARAMETERS                                    00260000
26    1  1  1  1 1 / X-R                                                       00280000
27    1  1  2  3 3 /  M-R                                                      00290000
28    0. 0.4  0.625 0.701  0.85 1.0432     / RX 1.47529                       00300000
29     4 / NMAT                                                                00000460
30   FH1DX01X  0  3  300.  1.25    0.   / MAT1 UO2 2.6% pin rod                00000470
31   XU050001  2  0  0.5799 E-3 /                                             00000480
32   XU080001  2  0  0.0217     /                                             00000481
33   X0060001  0  0  0.04461                                                   00000500
34   CLADX00X  0  1  300.  0.14  0.      / Al Cladding                        00000510
35   XAL70001  0  0  0.0623     /                                             00000520
36   H2OCX00X  0  2  300.  0.0   0.   / MAT3 H2O Coolant                      00000550
37   XH01H001  0  0  0.06692                                                  00000560
38   X0060001  0  0  0.03346                                                  00000570
39   H2ORX00X  0  2  300.  0.0   0.   / MAT3 H2O Reflector                    00000550
40   XH01H001  0  0  0.06692                                                  00000560
41   X0060001  0  0  0.03346                                                  00000570
42   0/ for PEACO : End of the first cade                                      
43   TCA2           ¥ Case name for the second Cell                            00000010
44   TCA  2 DOUBLE SPACED PIN ROD ARRAY  CYLINDRICAL APPROX                    00000020
45    1 1 1 1 2  1 4 0 -2 1  0 0 0 0 1  2 2 0 0 0 / SRAC CONTROL               00000030
46    .0080 /  GEOMETRICAL BUCKLING                                           00000040
47     3 7 7 5 1  0 7 0 0 0 10 0 6 1 0  0 0 0   / PATH                        00250000
48   0 6(0) 6(0.) /                                                           00260000
49    1  2  3  4 5 5 5/ X-R                                                   00280000
50    1  1  1  1 1 / X-R                                                      00280000
51    1  1  2  3 3 /  M-R                                                     00290000
52    0. 0.4  0.625 0.701  0.85 1.0432  1.2 1.47529 /RX                       00300000
53     3 / NMAT                                                                00000460
54   FH2DX02X  0  3  300.  1.25  0.   / MAT2 FUEL                             00000510
55   XU050001  2  0  0.5799 E-3       /                                       00000520
56   XU080001  2  0  0.0217           /                                       00000521
57   X0060001  0  0  0.04461                                                   00000540
58   CLADX00X  0  1  300.  0.14  0.   / Cladding already give, required       00000510
59   XAL70001  0  0  0.0623           / by PEACO routine                      00000520
60   H2OCX00X  0  2  300.  0.0   0. / MAT3 H2O Coolant                        00000550
61   XH01H001  0  0  0.06692                                                  00000560
62   X0060001  0  0  0.03346                                                  00000570
63   0/  for PEACO : End of the second case                                    
64   CR19 ¥ Case name for the third case                                       00000010
65   2-D  DIFFUSION CALCULATION FOR TWO REGION CORE                            00000020
66    0 0 0 1 0  0 0 0  0 1  0 5 0 0 1  0 0 0 0 0 / SRAC CONTROL              00000030
67    .0080 /  GEOMETRICAL BUCKLING not used                                  00000040
68    3 0  3/ CIT1                                                            00000220
69    5  5  1/  SELECTION OF DIRECTIONAL DIFF COEF BY MATERIAL                 
70   CORE19 ORGANIZED BY CENTRAL 20 LINES BY DOUBLE SPACED ARRAY, BOTH SIDE    
71   5 LINES BY STANDARD: EACH LINE BY 18 RODS OF ACTUAL HEIGHT 60.95 CM       
72   001                                                                       00000260
73                                      0                                     00000270
74     1     0  0                  0     0  1  0     1                        00000280
```

```
100.                                                                    00000290
 1.5                                                                    00000300
003                                                                     00000310
                6                   1  1  0  0  0  0                  00000320
 0                                                                      00000330
 0.                                                                     00000340
004                                                                     00000350
 20  18.5    10  9.25    10   10.                                       00000360
 10  18.5    10  10.                                                   00000360
005                                                                     00000370
   2  1  3                                                            00000380
   3  3  3                                                            00000380
008                                                                     00000390
-24 24 13                                                               00000400
024                                                                     00000410
   1  0.00185                                                          00000420
999                                                                     00000430
                                                                        00000440
 1 2 3  / MATERIAL NO.                                                  00000450
   3 / NMAT                                                             00000460
TCA1X01X  0  0  0.   0.   0.   / MAT1 STANDARD PITCH                   00000510
TCA2X01X  0  0  0.   0.   0.   / MAT2 DOUBLE PITCH ARRAY               00000510
H2ORX00X  0  0  0.    0.    0. / MAT3 H2O  AS SIDE REFLECTOR           00000550
          /  Read as the casename of the next :No case follows          00000900
                                                                        00000910
....*....1....*....2....*....3....*....4....*....5....*....6....*....7....*....8

*** INPUT DATA END ***
```

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```
*********************************************************************************
***  USER  NAME  <<TSUCHIHASI.K>>       SECTION   NO  <<     G0434     >>   ***
*********************************************************************************
***  JOB   NUMBER      ...        4374     RUNNING   DATE   ...   1986/03/05  ***
***  JOB   NAME        ...    F1480015     USER-ID           ...        J1480  ***
***  JOB   CLASS       ...           B     JOB  PRIORITY     ...           02  ***
***  TOTAL JOB STEPS   ...           3     COMPLETION  CODE  ...         0000  ***
***  ------------------------------------------------------------------------ ***
***  CPU          TIME ...  0H 0M11S65 I  I/O    ACCESS       FILES   ACCESS  ***
***  SRB          TIME ...  0H 0M 0S49 I           DISK ...      42     1552  ***
***  JOB  START   DATE ... 1986/03/05  I           TAPE ...       0        0  ***
***               TIME ... 11H 7M52S85 I          TOTAL .....             1552  ***
***  JOB    END   DATE ... 1986/03/05  I  VIO  PAGE  IN ...                0  ***
***               TIME ... 11H28M33S52 I  VIO  PAGE OUT ...                0  ***
***  ELAPSED      TIME ...  0H20M40S67 I                                      ***
***  ------------------------------------------------------------------------ ***
***  PAGE         IN   ...           0     STORAGE REQ'D(MAX) ...    2112  KB  ***
***  PAGE        OUT   ...           0     STORAGE USED (MAX) ...    1500  KB  ***
*********************************************************************************
***  ROOM-LIMIT-CPU    ROOM-USED-CPU     USER-LIMIT-CPU    USER-USED-CPU     ***
***       XH XM XSXX        XH XM XSXX        XH XM XSXX        XH XM XSXX    ***
*********************************************************************************
                  Content FT06F006 follows                                      PAGE-0001
     *** === SRAC CODE SYSTEM RUN    DATE : 86-03-05 START TIME=11:25:49:

*** ENTRANCE STEP  1**INPUT 1 **    0 SEC ELAPSED I/O COUNT=     15 WITHIN    4000 AFTER STEP 0**
*** ENTRANCE STEP  2**USER F L**    0 SEC ELAPSED I/O COUNT=    118 WITHIN    4000 AFTER STEP 1**
         MEMBER FASTLIB  OF LENGTH     49 IS STORED IN FASTU     FILE
         MEMBER FISSYILD OF LENGTH     22 IS STORED IN FASTU     FILE
         MEMBER CH010000 OF LENGTH     42 IS STORED IN FASTU     FILE
         MEMBER MH010000 OF LENGTH   1276 IS STORED IN FASTU     FILE
         MEMBER QH010000 OF LENGTH    594 IS STORED IN FASTU     FILE
         MEMBER SH010000 OF LENGTH    594 IS STORED IN FASTU     FILE
         MEMBER TH010000 OF LENGTH    594 IS STORED IN FASTU     FILE
         MEMBER UH010000 OF LENGTH    594 IS STORED IN FASTU     FILE
         MEMBER C0060000 OF LENGTH     42 IS STORED IN FASTU     FILE
         MEMBER M0060000 OF LENGTH    229 IS STORED IN FASTU     FILE
         MEMBER F0060000 OF LENGTH    256 IS STORED IN FASTU     FILE
         MEMBER CU050000 OF LENGTH     42 IS STORED IN FASTU     FILE
         MEMBER MU050000 OF LENGTH    363 IS STORED IN FASTU     FILE
         MEMBER FU050000 OF LENGTH   2496 IS STORED IN FASTU     FILE
         MEMBER RU050000 OF LENGTH      6 IS STORED IN FASTU     FILE
         MEMBER PU050000 OF LENGTH   3767 IS STORED IN FASTU     FILE
         MEMBER PU050001 OF LENGTH   2987 IS STORED IN FASTU     FILE
         MEMBER BU05000C OF LENGTH    808 IS STORED IN FASTU     FILE
         MEMBER BU05000F OF LENGTH    878 IS STORED IN FASTU     FILE
         MEMBER BU05000E OF LENGTH   1122 IS STORED IN FASTU     FILE
         MEMBER YU050000 OF LENGTH    169 IS STORED IN FASTU     FILE
         MEMBER CU080000 OF LENGTH     42 IS STORED IN FASTU     FILE
         MEMBER MU080000 OF LENGTH    333 IS STORED IN FASTU     FILE
         MEMBER FU080000 OF LENGTH   2496 IS STORED IN FASTU     FILE
         MEMBER RU080000 OF LENGTH      6 IS STORED IN FASTU     FILE
         MEMBER PU080000 OF LENGTH    407 IS STORED IN FASTU     FILE
         MEMBER PU080001 OF LENGTH    497 IS STORED IN FASTU     FILE
         MEMBER BU08000C OF LENGTH     84 IS STORED IN FASTU     FILE
         MEMBER BU08000F OF LENGTH     10 IS STORED IN FASTU     FILE
         MEMBER BU08000E OF LENGTH     84 IS STORED IN FASTU     FILE
         MEMBER YU080000 OF LENGTH    169 IS STORED IN FASTU     FILE
         MEMBER CAL70000 OF LENGTH     42 IS STORED IN FASTU     FILE
         MEMBER MAL70000 OF LENGTH    295 IS STORED IN FASTU     FILE
         MEMBER FAL70000 OF LENGTH    704 IS STORED IN FASTU     FILE
```

```
        FILE DD=FASTP    CLOSED AFTER SEARCH  35 READ   36 WRITE    0 DELETE    0 OVRWRT    0 TIMES
        FILE DD=FASTU    CLOSED AFTER SEARCH   7 READ    0 WRITE   34 DELETE    0 OVRWRT    0 TIMES
*** ENTRANCE STEP  3**USER T L**   0 SEC ELAPSED I/O COUNT=   269 WITHIN   4000 AFTER STEP 2**
        MEMBER THERMAL1 OF LENGTH    64 IS STORED IN THERMALU FILE
        MEMBER KH01H001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER PH01H001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER QH01H001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER SH01H001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER TH01H001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER UH01H001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER CH01H000 OF LENGTH    30 IS STORED IN THERMALU FILE
        MEMBER KO060001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER PO060001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER CO060000 OF LENGTH    30 IS STORED IN THERMALU FILE
        MEMBER KU050001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER FU050001 OF LENGTH   744 IS STORED IN THERMALU FILE
        MEMBER CU050000 OF LENGTH    30 IS STORED IN THERMALU FILE
        MEMBER KU080001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER CU080000 OF LENGTH    30 IS STORED IN THERMALU FILE
        MEMBER KAL70001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER PAL70001 OF LENGTH  1085 IS STORED IN THERMALU FILE
        MEMBER CAL70000 OF LENGTH    30 IS STORED IN THERMALU FILE
        FILE DD=THERMALP CLOSED AFTER SEARCH  23 READ   24 WRITE    0 DELETE    0 OVRWRT    0 TIMES
        FILE DD=THERMALU CLOSED AFTER SEARCH  23 READ    0 WRITE   19 DELETE    0 OVRWRT    0 TIMES
        FILE DD=FLUX     OPENED BY MODE 3
*** ENTRANCE STEP  4**PIJ INPT**   0 SEC ELAPSED I/O COUNT=   351 WITHIN   4000 AFTER STEP 3**
 STRAGE USED   3919 FROM  60000
        ***    1 LOGICAL RECORDS OF THE SIZE  1020 WRITTEN ON THE FILE UNIT F81 ***
        ***    1 LOGICAL RECORDS OF THE SIZE  1020 WRITTEN ON THE FILE UNIT F83 ***
        RATIOS OF VOLUME OF T-REGIONS NUMERICALLY INTEGRATED TO ANALYTIC
          1)  1.0003     2)  0.9993     3)  0.9997     4)  0.9998     5)  0.9997
            30 LINES DRAWN *** ELAPSED TIME      0 SEC
*** ENTRANCE STEP  5**GEOMINPT**   0 SEC ELAPSED I/O COUNT=   359 WITHIN   4000 AFTER STEP 4**
*** ENTRANCE STEP  6**INPUT 2 **   0 SEC ELAPSED I/O COUNT=   359 WITHIN   4000 AFTER STEP 5**
*** ENTRANCE STEP  7**MACRO F **   1 SEC ELAPSED I/O COUNT=   362 WITHIN   4000 AFTER STEP 6**
        FILE DD=FASTU    OPENED BY MODE 1
        FILE DD=MACROWRK OPENED BY MODE 3
        FILE DD=MICREF   OPENED BY MODE 3
        MEMBER CONTF002 OF LENGTH    46 IS STORED IN MACROWRK FILE
        STRAGE USED   7580 WITHIN 60000 IN PIJ2-STEP
        MEMBER CU05F011 OF LENGTH    41 IS STORED IN MICREF   FILE
        MEMBER MU05F011 OF LENGTH   319 IS STORED IN MICREF   FILE
        MEMBER CU08F011 OF LENGTH    41 IS STORED IN MICREF   FILE
        MEMBER MU08F011 OF LENGTH   289 IS STORED IN MICREF   FILE
        MEMBER CO06F011 OF LENGTH    41 IS STORED IN MICREF   FILE
        MEMBER MO06F011 OF LENGTH   163 IS STORED IN MICREF   FILE
        MEMBER FH1DF014 OF LENGTH   307 IS STORED IN MACROWRK FILE
        MEMBER FH1DF01M OF LENGTH   261 IS STORED IN MACROWRK FILE
        MEMBER FH1DF01Y OF LENGTH   264 IS STORED IN MACROWRK FILE
        MEMBER FH1DF013 OF LENGTH   264 IS STORED IN MACROWRK FILE
        MEMBER CAL7F001 OF LENGTH    41 IS STORED IN MICREF   FILE
        MEMBER MAL7F001 OF LENGTH   229 IS STORED IN MICREF   FILE
        MEMBER CLADF004 OF LENGTH   298 IS STORED IN MACROWRK FILE
        MEMBER CLADF003 OF LENGTH   264 IS STORED IN MACROWRK FILE
        MEMBER CO06F001 OF LENGTH    41 IS STORED IN MICREF   FILE
        MEMBER MO06F001 OF LENGTH   163 IS STORED IN MICREF   FILE
        MEMBER H2OCF004 OF LENGTH   451 IS STORED IN MACROWRK FILE
        MEMBER H2OCF003 OF LENGTH   451 IS STORED IN MACROWRK FILE
        MEMBER H2ORF004 OF LENGTH   451 IS STORED IN MACROWRK FILE
        MEMBER H2ORF003 OF LENGTH   451 IS STORED IN MACROWRK FILE
*** ENTRANCE STEP  8**MACRO T **   1 SEC ELAPSED I/O COUNT=   485 WITHIN   4000 AFTER STEP 7**
        FILE DD=THERMALU OPENED BY MODE 1
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        MEMBER CONTT002 OF LENGTH    64 IS STORED IN MACROWRK FILE
        MEMBER FH1DT014 OF LENGTH   814 IS STORED IN MACROWRK FILE
        MEMBER FH1DT013 OF LENGTH   814 IS STORED IN MACROWRK FILE
        MEMBER FH1DT01Y OF LENGTH   372 IS STORED IN MACROWRK FILE
        MEMBER CLADT004 OF LENGTH   703 IS STORED IN MACROWRK FILE
        MEMBER CLADT003 OF LENGTH   703 IS STORED IN MACROWRK FILE
        MEMBER H2OCT004 OF LENGTH  1198 IS STORED IN MACROWRK FILE
        MEMBER H2OCT003 OF LENGTH  1198 IS STORED IN MACROWRK FILE
        MEMBER H2ORT004 OF LENGTH  1198 IS STORED IN MACROWRK FILE
        MEMBER H2ORT003 OF LENGTH  1198 IS STORED IN MACROWRK FILE
*** ENTRANCE STEP  9**GAM P1B1**   1 SEC ELAPSED I/O COUNT=   534 WITHIN   4000 AFTER STEP 8**
        MEMBER CONTA002 OF LENGTH   108 IS STORED IN MACROWRK FILE
        MEMBER H2ORF002 OF LENGTH   451 IS STORED IN MACROWRK FILE
        MEMBER H2ORT002 OF LENGTH  1198 IS STORED IN MACROWRK FILE
        MEMBER H2ORA002 OF LENGTH    53 IS STORED IN FLUX     FILE
        MEMBER TCA1F014 OF LENGTH   463 IS STORED IN MACROWRK FILE
        MEMBER TCA1F013 OF LENGTH   451 IS STORED IN MACROWRK FILE
        MEMBER TCA1T014 OF LENGTH  1198 IS STORED IN MACROWRK FILE
        MEMBER TCA1T013 OF LENGTH  1119 IS STORED IN MACROWRK FILE
        MEMBER FH1DF012 OF LENGTH   307 IS STORED IN MACROWRK FILE
        MEMBER FH1DT012 OF LENGTH   814 IS STORED IN MACROWRK FILE
        MEMBER CLADF002 OF LENGTH   298 IS STORED IN MACROWRK FILE
        MEMBER CLADT002 OF LENGTH   703 IS STORED IN MACROWRK FILE
        MEMBER H2OCF002 OF LENGTH   451 IS STORED IN MACROWRK FILE
        MEMBER H2OCT002 OF LENGTH  1198 IS STORED IN MACROWRK FILE
*** ENTRANCE STEP 12**CELLFAST**   1 SEC ELAPSED I/O COUNT=   587 WITHIN   4000 AFTER STEP 9**
        STRAGE USED   1305 WITHIN 60000 IN PIJ2-STEP
        STRAGE USED   1735 WITHIN 60000 IN PIJ3 STEP
        MEMBER TCA1F002 OF LENGTH   110 IS STORED IN FLUX     FILE
        MEMBER TCA1FVOL OF LENGTH     5 IS STORED IN FLUX     FILE
*** ENTRANCE STEP 13**PEACO   **   1 SEC ELAPSED I/O COUNT=   673 WITHIN   4000 AFTER STEP12**
        FILE DD=MCROSS   OPENED BY MODE 1
        STRAGE USED  11617 WITHIN 60000 IN PIJ2-STEP
        MEMBER TCA1F002                 IS DELETED FROM DD=FLUX
        MEMBER TCA1F002 OF LENGTH   110 IS STORED IN FLUX     FILE
        MEMBER FH1DF012                 IS DELETED FROM DD=MACROWRK
        MEMBER FH1DF012 OF LENGTH   307 IS STORED IN MACROWRK FILE
*** ENTRANCE STEP 14**MIXXFAST**   2 SEC ELAPSED I/O COUNT=   796 WITHIN   4000 AFTER STEP13**
        MEMBER TCA1F012 OF LENGTH   463 IS STORED IN MACROWRK FILE
        MEMBER TCA1F01M OF LENGTH   261 IS STORED IN MACROWRK FILE
        MEMBER TCA1F01Y OF LENGTH   264 IS STORED IN MACROWRK FILE
        MEMBER TCA1F012 OF LENGTH    22 IS STORED IN FLUX     FILE
*** ENTRANCE STEP 15**CELLTHML**   2 SEC ELAPSED I/O COUNT=   810 WITHIN   4000 AFTER STEP14**
        STRAGE USED   1350 WITHIN 60000 IN PIJ2-STEP
        STRAGE USED   4592 WITHIN 60000 IN PIJ3 STEP
        MEMBER TCA1T002 OF LENGTH   155 IS STORED IN FLUX     FILE
        MEMBER TCA1TVOL OF LENGTH     5 IS STORED IN FLUX     FILE
*** ENTRANCE STEP 16**MIXXTHML**   3 SEC ELAPSED I/O COUNT=   918 WITHIN   4000 AFTER STEP15**
        MEMBER TCA1T012 OF LENGTH  1198 IS STORED IN MACROWRK FILE
        MEMBER TCA1T01Y OF LENGTH   372 IS STORED IN MACROWRK FILE
        MEMBER TCA1T012 OF LENGTH    31 IS STORED IN FLUX     FILE
*** ENTRANCE STEP 17**HOMOSP  **   3 SEC ELAPSED I/O COUNT=   928 WITHIN   4000 AFTER STEP16**
        MEMBER TCA1A012 OF LENGTH    53 IS STORED IN FLUX     FILE
        MEMBER TCA1A002 OF LENGTH   265 IS STORED IN FLUX     FILE
        K-EFF= 1.01696 KINF= 1.34154 UNDER GEOMTRICAL BUCKLING= 0.80000E-02
*** ENTRANCE STEP 18**COND FIX**   3 SEC ELAPSED I/O COUNT=   937 WITHIN   4000 AFTER STEP17**
        FILE DD=MACRO    OPENED BY MODE 3
        MEMBER CONTA002 OF LENGTH   108 IS STORED IN FLUX     FILE
        MEMBER CONTA000 OF LENGTH    22 IS STORED IN MACRO    FILE
        MEMBER CONTA000 OF LENGTH    22 IS STORED IN FLUX     FILE
        MEMBER H2ORA000 OF LENGTH   144 IS STORED IN MACRO    FILE
        MEMBER H2ORA000 OF LENGTH    10 IS STORED IN FLUX     FILE
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        MEMBER TCA1A010 OF LENGTH   145 IS STORED IN MACRO     FILE
        MEMBER TCA1A01N OF LENGTH   112 IS STORED IN MACRO     FILE
        MEMBER TCA1A01Z OF LENGTH   120 IS STORED IN MACRO     FILE
        MEMBER TCA1A010 OF LENGTH    10 IS STORED IN FLUX      FILE
*** ENTRANCE STEP 28**END CASE**   3 SEC ELAPSED I/O COUNT=   959 WITHIN   4000 AFTER STEP18**

*** ENTRANCE STEP  1**INPUT 1 **   3 SEC ELAPSED I/O COUNT=   959 WITHIN   4000 AFTER STEP 0**
*** ENTRANCE STEP  4**PIJ INPT**   3 SEC ELAPSED I/O COUNT=   961 WITHIN   4000 AFTER STEP 1**
 STRAGE USED   3925 FROM  60000
        ***    1 LOGICAL RECORDS OF THE SIZE  1020 WRITTEN ON THE FILE UNIT F81 ***
        ***    1 LOGICAL RECORDS OF THE SIZE  1020 WRITTEN ON THE FILE UNIT F82 ***
        ***    1 LOGICAL RECORDS OF THE SIZE  1020 WRITTEN ON THE FILE UNIT F83 ***
*** ENTRANCE STEP  5**GEOMINPT**   3 SEC ELAPSED I/O COUNT=   970 WITHIN   4000 AFTER STEP 4**
*** ENTRANCE STEP  6**INPUT 2 **   3 SEC ELAPSED I/O COUNT=   970 WITHIN   4000 AFTER STEP 5**
*** ENTRANCE STEP  7**MACRO F **   3 SEC ELAPSED I/O COUNT=   972 WITHIN   4000 AFTER STEP 6**
        STRAGE USED   7532 WITHIN 60000 IN PIJ2-STEP
        STRAGE USED   7532 WITHIN 60000 IN PIJ2-STEP
        MEMBER CU05F021 OF LENGTH    41 IS STORED IN MICREF   FILE
        MEMBER MU05F021 OF LENGTH   319 IS STORED IN MICREF   FILE
        MEMBER CU08F021 OF LENGTH    41 IS STORED IN MICREF   FILE
        MEMBER MU08F021 OF LENGTH   289 IS STORED IN MICREF   FILE
        MEMBER C006F021 OF LENGTH    41 IS STORED IN MICREF   FILE
        MEMBER M006F021 OF LENGTH   163 IS STORED IN MICREF   FILE
        MEMBER FH2DF024 OF LENGTH   307 IS STORED IN MACROWRK FILE
        MEMBER FH2DF02M OF LENGTH   261 IS STORED IN MACROWRK FILE
        MEMBER FH2DF02Y OF LENGTH   264 IS STORED IN MACROWRK FILE
        MEMBER FH2DF023 OF LENGTH   264 IS STORED IN MACROWRK FILE
*** ENTRANCE STEP  8**MACRO T **   3 SEC ELAPSED I/O COUNT=  1045 WITHIN   4000 AFTER STEP 7**
        MEMBER FH2DT024 OF LENGTH   814 IS STORED IN MACROWRK FILE
        MEMBER FH2DT023 OF LENGTH   814 IS STORED IN MACROWRK FILE
        MEMBER FH2DT02Y OF LENGTH   372 IS STORED IN MACROWRK FILE
         ** MEMBER CLADT002 ALREADY EXISTS IN MACRO FILE FORMATION SKIPPED **
         ** MEMBER H2OCT002 ALREADY EXISTS IN MACRO FILE FORMATION SKIPPED **
*** ENTRANCE STEP  9**GAM P1B1**   3 SEC ELAPSED I/O COUNT=  1065 WITHIN   4000 AFTER STEP 8**
        MEMBER TCA2F014 OF LENGTH   463 IS STORED IN MACROWRK FILE
        MEMBER TCA2F013 OF LENGTH   451 IS STORED IN MACROWRK FILE
        MEMBER TCA2T014 OF LENGTH  1198 IS STORED IN MACROWRK FILE
        MEMBER TCA2T013 OF LENGTH  1119 IS STORED IN MACROWRK FILE
        MEMBER FH2DF022 OF LENGTH   307 IS STORED IN MACROWRK FILE
        MEMBER FH2DT022 OF LENGTH   814 IS STORED IN MACROWRK FILE
*** ENTRANCE STEP 12**CELLFAST**   3 SEC ELAPSED I/O COUNT=  1100 WITHIN   4000 AFTER STEP 9**
        STRAGE USED   1305 WITHIN 60000 IN PIJ2-STEP
        STRAGE USED   1735 WITHIN 60000 IN PIJ3 STEP
        MEMBER TCA2F002 OF LENGTH   110 IS STORED IN FLUX     FILE
        MEMBER TCA2FVOL OF LENGTH     5 IS STORED IN FLUX     FILE
*** ENTRANCE STEP 13**PEACO    **   4 SEC ELAPSED I/O COUNT=  1185 WITHIN   4000 AFTER STEP12**
        STRAGE USED  11617 WITHIN 60000 IN PIJ2-STEP
        MEMBER TCA2F002               IS DELETED FROM DD=FLUX
        MEMBER TCA2F002 OF LENGTH   110 IS STORED IN FLUX     FILE
        MEMBER FH2DF022               IS DELETED FROM DD=MACROWRK
        MEMBER FH2DF022 OF LENGTH   307 IS STORED IN MACROWRK FILE
*** ENTRANCE STEP 14**MIXXFAST**   5 SEC ELAPSED I/O COUNT=  1272 WITHIN   4000 AFTER STEP13**
        MEMBER TCA2F012 OF LENGTH   463 IS STORED IN MACROWRK FILE
        MEMBER TCA2F01M OF LENGTH   261 IS STORED IN MACROWRK FILE
        MEMBER TCA2F01Y OF LENGTH   264 IS STORED IN MACROWRK FILE
        MEMBER TCA2F012 OF LENGTH    22 IS STORED IN FLUX     FILE
*** ENTRANCE STEP 15**CELLTHML**   5 SEC ELAPSED I/O COUNT=  1286 WITHIN   4000 AFTER STEP14**
        STRAGE USED   1378 WITHIN 60000 IN PIJ2-STEP
        STRAGE USED   5650 WITHIN 60000 IN PIJ3 STEP
        MEMBER TCA2T002 OF LENGTH   155 IS STORED IN FLUX     FILE
        MEMBER TCA2TVOL OF LENGTH     5 IS STORED IN FLUX     FILE
        MEMBER TCA2SVOL OF LENGTH     7 IS STORED IN FLUX     FILE
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*** ENTRANCE STEP 16**MIXXTHML**   5 SEC ELAPSED I/O COUNT=  1395 WITHIN   4000 AFTER STEP15**
        MEMBER TCA2T012 OF LENGTH  1198 IS STORED IN MACROWRK FILE
        MEMBER TCA2T01Y OF LENGTH   372 IS STORED IN MACROWRK FILE
        MEMBER TCA2T012 OF LENGTH    31 IS STORED IN FLUX     FILE
*** ENTRANCE STEP 17**HOMOSP  **   5 SEC ELAPSED I/O COUNT=  1405 WITHIN   4000 AFTER STEP16**
        MEMBER TCA2A012 OF LENGTH    53 IS STORED IN FLUX     FILE
        MEMBER TCA2A002 OF LENGTH   265 IS STORED IN FLUX     FILE
        K-EFF= 1.00627 KINF= 1.27135 UNDER GEOMTRICAL BUCKLING= 0.80000E-02
*** ENTRANCE STEP 18**COND FIX**   5 SEC ELAPSED I/O COUNT=  1414 WITHIN   4000 AFTER STEP17**
        MEMBER TCA2A010 OF LENGTH   145 IS STORED IN MACRO    FILE
        MEMBER TCA2A01N OF LENGTH   112 IS STORED IN MACRO    FILE
        MEMBER TCA2A01Z OF LENGTH   120 IS STORED IN MACRO    FILE
        MEMBER TCA2A010 OF LENGTH    10 IS STORED IN FLUX     FILE
*** ENTRANCE STEP 28**END CASE**   6 SEC ELAPSED I/O COUNT=  1427 WITHIN   4000 AFTER STEP18**
*** ENTRANCE STEP  1**INPUT 1 **   6 SEC ELAPSED I/O COUNT=  1427 WITHIN   4000 AFTER STEP 0**
     **** CASE ID === CR19 ¥ C *** TITLE === 2-D  DIFFUSION CALCULATION FOR TWO REGION CORE          ***
*** CONDENSE OF MACRO SPECIFIED BY IC10 REQUIRES CALCULATION OF FLUXES SKIPPED BY IC6
*** ENTRANCE STEP  5**GEOMINPT**   6 SEC ELAPSED I/O COUNT=  1429 WITHIN   4000 AFTER STEP 1**
        STORAGE USED         4 FROM      4000 IN CIT1 STEP
*** ENTRANCE STEP  6**INPUT 2 **   6 SEC ELAPSED I/O COUNT=  1452 WITHIN   4000 AFTER STEP 5**
*** ENTRANCE STEP  7**MACRO F **   6 SEC ELAPSED I/O COUNT=  1454 WITHIN   4000 AFTER STEP 6**
*** ENTRANCE STEP  8**MACRO T **   6 SEC ELAPSED I/O COUNT=  1461 WITHIN   4000 AFTER STEP 7**
*** ENTRANCE STEP  9**GAM P1B1**   6 SEC ELAPSED I/O COUNT=  1464 WITHIN   4000 AFTER STEP 8**
*** ENTRANCE STEP 18**COND FIX**   6 SEC ELAPSED I/O COUNT=  1467 WITHIN   4000 AFTER STEP 9**
*** ENTRANCE STEP 22**EIGN VLE**   6 SEC ELAPSED I/O COUNT=  1470 WITHIN   4000 AFTER STEP18**
        MEMBER CR19AVOL OF LENGTH     6 IS STORED IN FLUX     FILE
        MEMBER CR19A000 OF LENGTH    60 IS STORED IN FLUX     FILE
*** ENTRANCE STEP 28**END CASE**  11 SEC ELAPSED I/O COUNT=  1550 WITHIN   4000 AFTER STEP22**
*** ENTRANCE STEP  1**INPUT 1 **  11 SEC ELAPSED I/O COUNT=  1550 WITHIN   4000 AFTER STEP 0**
     **** CASE ID ===          *** TITLE ===                                                        ***
        FILE DD=FASTU     CLOSED AFTER SEARCH  20 READ  138 WRITE    0 DELETE    0 OVRWRT    0 TIMES
        FILE DD=THERMALU  CLOSED AFTER SEARCH  43 READ   47 WRITE    0 DELETE    0 OVRWRT    0 TIMES
        FILE DD=MACROWRK  CLOSED AFTER SEARCH 189 READ  159 WRITE   59 DELETE    2 OVRWRT   10 TIMES
        FILE DD=FLUX      CLOSED AFTER SEARCH  34 READ   17 WRITE   27 DELETE    2 OVRWRT    9 TIMES
        FILE DD=MACRO     CLOSED AFTER SEARCH  16 READ    5 WRITE    8 DELETE    0 OVRWRT    0 TIMES
        FILE DD=MCROSS    CLOSED AFTER SEARCH   0 READ   34 WRITE    0 DELETE    0 OVRWRT    0 TIMES
        FILE DD=MICREF    CLOSED AFTER SEARCH  32 READ   12 WRITE   16 DELETE    0 OVRWRT    0 TIMES
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                Content of FT99F001 follows                                                    PAGE-0001
 ***  === SRAC CODE SYSTEM RUN    DATE : 86-03-05 START TIME=11:25:49:

 ****  CASE ID === TCA1      *** TITLE === TCA1 STANDARD CELL    CYLINDRICAL APPROX.                    ***

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)                             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 (P0,P1,B1,SN:0,1,2,3)                    2
DIFF COEF (INVERSE TR,BENOIST ISO,BENOIST ANISO:1,2,3)                2
REACTION RATE CALCULATION (0,1)                                       0
PRINT OF MACRO-X (0,1,2)                                              0
BURN-UP CALCULATION (0,1:SKIP,EXECUTE)                                0
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

               ### USERFL START ###
                        ENERGY GROUP STRUCTURE
      NEFL: TOTAL NBR OF PUBLIC FAST GROUPS -----   74
      NETL: TOTAL NBR OF PUBLIC THERMAL GROUPS --   48
      NEF : TOTAL NBR OF USERS  FAST GROUPS -----   22
      NET : TOTAL NBR OF USERS  THERMAL GROUPS --   31

          FINE   FAST    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    0.60653E+07       0.43740E+10    0.34065E+10          0.0          0.5000
  2       0.60653E+07    0.36788E+07       0.34065E+10    0.26529E+10          0.5000       1.0000
  3       0.36788E+07    0.22313E+07       0.26529E+10    0.20661E+10          1.0000       1.5000
  4       0.22313E+07    0.13534E+07       0.20661E+10    0.16091E+10          1.5000       2.0000
  5       0.13534E+07    0.82085E+06       0.16091E+10    0.12532E+10          2.0000       2.5000
  6       0.82085E+06    0.49787E+06       0.12532E+10    0.97596E+09          2.5000       3.0000
  7       0.49787E+06    0.30197E+06       0.97596E+09    0.76008E+09          3.0000       3.5000
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8	0.30197E+06	0.11109E+06	0.76008E+09	0.46101E+09	3.5000	4.5000
9	0.11109E+06	0.40868E+05	0.46101E+09	0.27962E+09	4.5000	5.5000
10	0.40868E+05	0.15034E+05	0.27962E+09	0.16960E+09	5.5000	6.5000
11	0.15034E+05	0.55308E+04	0.16960E+09	0.10287E+09	6.5000	7.5000
12	0.55308E+04	0.20347E+04	0.10287E+09	0.62391E+08	7.5000	8.5000
13	0.20347E+04	0.74852E+03	0.62391E+08	0.37842E+08	8.5000	9.5000
14	0.74852E+03	0.35357E+03	0.37842E+08	0.26009E+08	9.5000	10.2500
15	0.35357E+03	0.16702E+03	0.26009E+08	0.17875E+08	10.2500	11.0000
16	0.16702E+03	0.78893E+02	0.17875E+08	0.12286E+08	11.0000	11.7500
17	0.78893E+02	0.37266E+02	0.12286E+08	0.84437E+07	11.7500	12.5000
18	0.37266E+02	0.17604E+02	0.84437E+07	0.58033E+07	12.5000	13.2500
19	0.17604E+02	0.83153E+01	0.58033E+07	0.39885E+07	13.2500	14.0000
20	0.83153E+01	0.39279E+01	0.39885E+07	0.27413E+07	14.0000	14.7500
21	0.39279E+01	0.18554E+01	0.27413E+07	0.18841E+07	14.7500	15.5000
22	0.18554E+01	0.11253E+01	0.18841E+07	0.14673E+07	15.5000	16.0000

USERFL END

FINE THERMAL ENERGY GROUP STRUCTURE (M=1.67482E-24 GRAM, EV=1.60210E-12 ERG)

GROUP	ENERGY RANGE (EV)		VELOCITY RANGE (CM/SEC)		LETHARGY RANGE	
1	0.11254E+01	0.68256E+00	0.14673E+07	0.11427E+07	16.0000	16.5000
2	0.68256E+00	0.41399E+00	0.11427E+07	0.88996E+06	16.5000	17.0000
3	0.41399E+00	0.36528E+00	0.88996E+06	0.83597E+06	17.0000	17.1252
4	0.36528E+00	0.31961E+00	0.83597E+06	0.78196E+06	17.1252	17.2587
5	0.31961E+00	0.29792E+00	0.78196E+06	0.75496E+06	17.2587	17.3290
6	0.29792E+00	0.27699E+00	0.75496E+06	0.72796E+06	17.3290	17.4019
7	0.27699E+00	0.23742E+00	0.72796E+06	0.67396E+06	17.4019	17.5560
8	0.23742E+00	0.20090E+00	0.67396E+06	0.61996E+06	17.5560	17.7230
9	0.20090E+00	0.18378E+00	0.61996E+06	0.59296E+06	17.7230	17.8121
10	0.18378E+00	0.16743E+00	0.59296E+06	0.56597E+06	17.8121	17.9053
11	0.16743E+00	0.15183E+00	0.56597E+06	0.53896E+06	17.9053	18.0031
12	0.15183E+00	0.13700E+00	0.53896E+06	0.51196E+06	18.0031	18.1059
13	0.13700E+00	0.12293E+00	0.51196E+06	0.48496E+06	18.1059	18.2142
14	0.12293E+00	0.10963E+00	0.48496E+06	0.45797E+06	18.2142	18.3287
15	0.10963E+00	0.97080E-01	0.45797E+06	0.43096E+06	18.3287	18.4503
16	0.97080E-01	0.85397E-01	0.43096E+06	0.40420E+06	18.4503	18.5785
17	0.85397E-01	0.74276E-01	0.40420E+06	0.37696E+06	18.5785	18.7181
18	0.74276E-01	0.64017E-01	0.37696E+06	0.34996E+06	18.7181	18.8667
19	0.64017E-01	0.54520E-01	0.34996E+06	0.32296E+06	18.8667	19.0273
20	0.54520E-01	0.45785E-01	0.32296E+06	0.29596E+06	19.0273	19.2019
21	0.45785E-01	0.37813E-01	0.29596E+06	0.26897E+06	19.2019	19.3932
22	0.37813E-01	0.30602E-01	0.26897E+06	0.24196E+06	19.3932	19.6048
23	0.30602E-01	0.24154E-01	0.24196E+06	0.21497E+06	19.6048	19.8414
24	0.24154E-01	0.18467E-01	0.21497E+06	0.18796E+06	19.8414	20.1099
25	0.18467E-01	0.13543E-01	0.18796E+06	0.16097E+06	20.1099	20.4200
26	0.13543E-01	0.93805E-02	0.16097E+06	0.13396E+06	20.4200	20.7872
27	0.93805E-02	0.59804E-02	0.13396E+06	0.10696E+06	20.7872	21.2374
28	0.59804E-02	0.33423E-02	0.10696E+06	0.79965E+05	21.2374	21.8192
29	0.33423E-02	0.14663E-02	0.79965E+05	0.52965E+05	21.8192	22.6431
30	0.14663E-02	0.35238E-03	0.52965E+05	0.25965E+05	22.6431	24.0689
31	0.35238E-03	0.10010E-04	0.25965E+05	0.43761E+04	24.0689	27.6300

```
TCA1    TCA1 STANDARD CELL    CYLINDRICAL APPROX.                    *** STEP *** INPUT FOR PIJ ***
              GEOMETRY TYPE                                  3          ( CIRCULAR CYLINDER )
              NUMBER OF SUB - REGIONS                        5
              NUMBER OF  T  - REGIONS                        5
              NUMBER OF  R  - REGIONS                        5
              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

 ===X-REGION NO./R  ===
             1            1            1            1
 ===MATERIAL NO./R  ===
             1            1            2            3            3
 ===R-X DIVISION    ===
  0.0          0.40000E+00 0.62500E+00 0.70100E+00 0.85000E+00 0.10432E+01
 VOLUME OF (S)-REGION
   1) 5.0265E-01  2) 7.2453E-01  3) 3.1660E-01  4) 7.2602E-01  5) 1.1491E+00
 VOLUME OF (T)-REGION
   1) 5.0265E-01  2) 7.2453E-01  3) 3.1660E-01  4) 7.2602E-01  5) 1.1491E+00
 VOLUME OF (X)-REGION
   1) 3.4189E+00
 VOLUME OF (M)-REGION
   1) 1.2272E+00  2) 3.1660E-01  3) 1.8751E+00
 TOTAL VOLUME                    3.4189E+00
 RATIOS OF VOLUME OF T-REGIONS NUMERICALLY INTEGRATED TO ANALYTIC
   1)  1.0003     2)  0.9993     3)  0.9997     4)  0.9998     5)  0.9997
     30 LINES DRAWN *** ELAPSED TIME      0 SEC

   ======   MACROF  ======

  CASE I.D. : TCA1
  TITLE     : TCA1 STANDARD CELL    CYLINDRICAL APPROX.

              COMPOSITION DATA LIST
                        ## MATERIAL NAME -----  1   FH1DX01X ##
  NUMBER OF NUCLIDE ------------        3
  TEMPERATURE (K) --------------      300.00
  MEAN CHORD LENGTH ------------     1.25000
  DANCOFF FACTOR ---------------     0.0
  IDENTIFICATION OF NUCLIDE ---     U050001     U080001     0060001
  NUMBER DENSITY ---------------  5.79899E-04 2.17000E-02 4.46100E-02
  RESONANT INDICATOR -----------       2           2           0
  LXMICR -----------------------       0           0           0
  NUCLIDE-WISE DANCOFF FACTOR WILL BE CALCULATED AT SUBROUTINE(MAFSIG) !!

                        ## MATERIAL NAME -----  2   CLADX00X ##
  NUMBER OF NUCLIDE ------------        1
  TEMPERATURE (K) --------------      300.00
  MEAN CHORD LENGTH ------------     0.14000
  DANCOFF FACTOR ---------------     0.0
```

```
 IDENTIFICATION OF NUCLIDE ---     AL70001
 NUMBER DENSITY --------------  6.23000E-02
 RESONANT INDICATOR ----------       0
 LXMICR ----------------------       0
 NUCLIDE-WISE DANCOFF FACTOR WILL BE CALCULATED AT SUBROUTINE(MAFSIG) !!

                        ## MATERIAL NAME -----  3   H2OCX00X ##
 NUMBER OF NUCLIDE ------------         2
 TEMPERATURE (K) --------------       300.00
 MEAN CHORD LENGTH ------------      0.0
 DANCOFF FACTOR ---------------      0.0
 IDENTIFICATION OF NUCLIDE ---      H010001     0060001
 NUMBER DENSITY ---------------  6.69200E-02 3.34600E-02
 RESONANT INDICATOR -----------       0           0
 LXMICR -----------------------       0           0
 NUCLIDE-WISE DANCOFF FACTOR WILL BE CALCULATED AT SUBROUTINE(MAFSIG) !!

                        ## MATERIAL NAME -----  4   H2ORX00X ##
 NUMBER OF NUCLIDE ------------         2
 TEMPERATURE (K) --------------       300.00
 MEAN CHORD LENGTH ------------      0.0
 DANCOFF FACTOR ---------------      0.0
 IDENTIFICATION OF NUCLIDE ---      H010001     0060001
 NUMBER DENSITY ---------------  6.69200E-02 3.34600E-02
 RESONANT INDICATOR -----------       0           0
 LXMICR -----------------------       0           0

                        ## RESONANT NUCLIDE NAME LIST ##
 NUMBER OF RESONANT NUCLIDE --        2
 RESONANT NUCLIDE NAME -------  U05   U08

                        ## STANDARD TEMPERATUE ARRAY LIST (STND) ##
TAG         :    1        2        3        4        5        6        7        8        9        A        B
TEMPERATURE :    300.00   325.00   350.00   400.00   450.00   500.00   550.00   600.00   900.00  1200.00  1600.00
                        ******************************************
                        * CALCULATED NUCLIDE-WISE DANCOFF FACTOR *
                        ******************************************
                        ## MATERIAL NAME -----  1   FH1DX01X ##
 IDENTIFICATION OF NUCLIDE ----     U050001     U080001     0060001
 NUCLIDE-WISE DANCOFF FACTOR---     0.154768    0.154768    1.000000

                        ## MATERIAL NAME -----  2   CLADX00X ##
 IDENTIFICATION OF NUCLIDE ----     AL70001
 NUCLIDE-WISE DANCOFF FACTOR---     1.000000

                        ## MATERIAL NAME -----  3   H2OCX00X ##
 IDENTIFICATION OF NUCLIDE ----     H010001     0060001
 NUCLIDE-WISE DANCOFF FACTOR---     1.000000    1.000000
STRAGE USED   1305 WITHIN 60000 IN PIJ2-STEP

===PIJ3 STEP===
***TCA1     ***TCA1 STANDARD CELL     CYLINDRICAL APPROX.                              *** IN FAST      RANGE ***

ITERATION PARAMETERS
PRINT (1+2+4+8)(FLUX,XEC,PIJ,S)              0
MAX OF INNER ITERATIONS PER OUTER           20
MAX OF OUTER ITERATIONS                     50
EARLIST EXTRAPOLATION                        5
NUMBER OF ITERATIONS TESTED                  5
MINIMUM DELAY                                5
MONITOR PRINT(0,1)(SKIP,PRINT)              -1
CONVERGENCE CRITERION OF INNER     0.10000E-03
```

```
CONVERGENCE CRITERION OF OUTER    0.10000E-04
EXTRAPOLATION CRITERION           0.10000E-02
OVER-RELAXATION (INITIAL)         0.12000E+01
MAX EXTRAPOLATION                 0.10000E+03
BASE FACTOR OF OVER-RELAXATION    0.70000E+00

   ITERATION POWER-SCALING   RESIDUE CRITERON

    1         0.13077E+00 0.69288E-01 0.10000E-04
    2         0.14544E+00 0.33362E-02 0.10000E-04
    3         0.14473E+00 0.16002E-03 0.10000E-04
    4         0.14477E+00 0.76120E-05 0.10000E-04
=== ITERATION END IN PIJF STEP ===
=== FIXED SOURCE TYPE PROBLEM ===
TOTAL THERMAL INNER ITERATION COUNT            0
INITIAL SOURCE NORMALIZATION FACTOR   0.52049E+01
THERMAL SOURCE NORMALIZATION FACTOR   0.10000E+01
RENORMALIZATION FACTOR (INNER)        0.0
RESIDUE IN FINAL INNER ITERATION      0.0
FINAL FISSION SOURCE NORMALIZATION    0.14477E+00
RESIDUE IN FINAL FISSION RATE         0.76120E-05
STRAGE USED  11617 WITHIN 60000 IN PIJ2-STEP

*** MIX-X-SECTION STEP IN *** FAST ENERGY RANGE
*** TCA1    ***TCA1 STANDARD CELL    CYLINDRICAL APPROX.                    ***

X-REGION  1***
ONE GROUP CONSTANTS
* ACTIVATION CROSSSECTION      0.58345E-01
* FISSION    CROSSSECTION      0.25732E-02
* NU*FISSION CROSSSECTION      0.66301E-02
* TOTAL      CROSSSECTION      0.26677E+00
* DIFFUSION COEFFICIENT 1      0.12495E+01
* DIFFUSION COEFFICIENT 2      0.13601E+01
* ABSORPTION CROSSSECTION      0.94634E-02
* SCATTEROUT CROSSSECTION      0.23519E-01
* INTEGRATED FLUX-X-REGION     0.33510E+02
*** COMMENT *** DIFFUSION COEF. (BENOIST) NOT YET INCLUDED IN THE ABOVE LIST
STRAGE USED   1350 WITHIN 60000 IN PIJ2-STEP

===PIJ3 STEP===
***TCA1    ***TCA1 STANDARD CELL    CYLINDRICAL APPROX.                    *** IN THERMAL  RANGE ***

ITERATION PARAMETERS
PRINT (1+2+4+8)(FLUX,XEC,PIJ,S)            0
MAX OF INNER ITERATIONS PER OUTER        200
MAX OF OUTER ITERATIONS                    1
EARLIST EXTRAPOLATION                      5
NUMBER OF ITERATIONS TESTED                5
MINIMUM DELAY                              5
MONITOR PRINT(0,1)(SKIP,PRINT)            -1
CONVERGENCE CRITERION OF INNER    0.10000E-03
CONVERGENCE CRITERION OF OUTER    0.10000E-04
EXTRAPOLATION CRITERION           0.10000E-02
OVER-RELAXATION (INITIAL)         0.12000E+01
MAX EXTRAPOLATION                 0.10000E+03
BASE FACTOR OF OVER-RELAXATION    0.70000E+00
=== ITERATION END IN PIJF STEP ===
=== FIXED SOURCE TYPE PROBLEM ===
TOTAL THERMAL INNER ITERATION COUNT           13
INITIAL SOURCE NORMALIZATION FACTOR   0.78812E+00
THERMAL SOURCE NORMALIZATION FACTOR   0.10000E+01
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 RENORMALIZATION FACTOR (INNER)          0.10000E+01
 RESIDUE IN FINAL INNER ITERATION        0.89017E-04
 FINAL FISSION SOURCE NORMALIZATION      0.0
 RESIDUE IN FINAL FISSION RATE           0.79241E-08

*** MIX-X-SECTION STEP IN *** THER ENERGY RANGE
 *** TCA1     ***TCA1 STANDARD CELL    CYLINDRICAL APPROX.                          ***

 X-REGION  1***
 ONE GROUP CONSTANTS
 * ACTIVATION CROSSSECTION        0.40255E+01
 * FISSION    CROSSSECTION        0.62841E-01
 * NU*FISSION CROSSSECTION        0.15200E+00
 * TOTAL      CROSSSECTION        0.12210E+01
 * DIFFUSION COEFFICIENT 1        0.27299E+00
 * DIFFUSION COEFFICIENT 2        0.27300E+00
 * ABSORPTION CROSSSECTION        0.94843E-01
 * SCATTEROUT CROSSSECTION        0.16604E-03
 * INTEGRATED FLUX-X-REGION       0.82953E+01
 *** COMMENT *** DIFFUSION COEF. (BENOIST) NOT YET INCLUDED IN THE ABOVE LIST

    ***  BARE REACTOR SPECTRUM CALCULATION *** OF STEP 15 ***
 ***TCA1     ***TCA1 STANDARD CELL    CYLINDRICAL APPROX.                          ***
 *** BY B1 APPROXIMATION ***
 RATIO OF THERMAL SOURCE TO REMOVAL  0.10001E+01
 NEUTRON SPECTRUM PER LETHARGY
  0.46593E+00 0.19121E+01 0.41529E+01 0.44363E+01 0.42720E+01 0.41415E+01 0.30195E+01 0.20728E+01 0.14276E+01 0.11565E+01
  0.10373E+01 0.98463E+00 0.94837E+00 0.92246E+00 0.89730E+00 0.86031E+00 0.83408E+00 0.74318E+00 0.76346E+00 0.66214E+00
  0.71370E+00 0.70394E+00 0.71993E+00 0.76013E+00 0.75869E+00 0.77624E+00 0.77329E+00 0.78422E+00 0.82440E+00 0.89293E+00
  0.99594E+00 0.10728E+01 0.11913E+01 0.13181E+01 0.14891E+01 0.16847E+01 0.18695E+01 0.20537E+01 0.21849E+01 0.22611E+01
  0.22407E+01 0.21145E+01 0.18986E+01 0.16021E+01 0.12608E+01 0.91452E+00 0.60165E+00 0.34828E+00 0.17137E+00 0.66896E-01
  0.17994E-01 0.23480E-02 0.56450E-04
 NEUTRON CURRENT PER LETHARGY
  0.10088E-01 0.35816E-01 0.71793E-01 0.64776E-01 0.47996E-01 0.42836E-01 0.26489E-01 0.16263E-01 0.99528E-02 0.74314E-02
  0.62554E-02 0.57151E-02 0.54261E-02 0.52389E-02 0.50332E-02 0.48503E-02 0.46037E-02 0.41808E-02 0.42507E-02 0.37745E-02
  0.39783E-02 0.39186E-02 0.30442E-02 0.29205E-02 0.26531E-02 0.25891E-02 0.24790E-02 0.24533E-02 0.24985E-02 0.25871E-02
  0.27678E-02 0.28908E-02 0.31007E-02 0.33072E-02 0.35969E-02 0.39046E-02 0.41453E-02 0.43411E-02 0.43817E-02 0.42707E-02
  0.39511E-02 0.34458E-02 0.28322E-02 0.21707E-02 0.15426E-02 0.10062E-02 0.59315E-03 0.30617E-03 0.13288E-03 0.44517E-04
  0.96641E-05 0.88306E-06 0.10876E-07
 FAST FISSION              0.15703E+00
 FAST ABSORPTION           0.22145E+00
 FAST LEAKAGE              0.22957E+00
 TOTAL FISSION             0.10170E+01
 TOTAL ABSORPTION          0.75805E+00
 TOTAL LEAKAGE             0.24195E+00
 K-EFF= 1.01696 KINF= 1.34154 UNDER GEOMTRICAL BUCKLING= 0.80000E-02
```

COARSE WHOLE 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

```
**** CASE ID === TCA2     *** TITLE === TCA  2 DOUBLE SPACED PIN ROD ARRAY  CYLINDRICAL APPROX                    ***

 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)                             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 (P0,P1,B1,SN:0,1,2,3)                    2
 DIFF COEF (INVERSE TR,BENOIST ISO,BENOIST ANISO:1,2,3)                2
 REACTION RATE CALCULATION (0,1)                                       0
 PRINT OF MACRO-X (0,1,2)                                              0
 BURN-UP CALCULATION (0,1:SKIP,EXECUTE)                                0

TCA2     TCA  2 DOUBLE SPACED PIN ROD ARRAY  CYLINDRICAL APPROX                  *** STEP *** INPUT FOR PIJ ***
                 GEOMETRY TYPE                                        3          ( CIRCULAR CYLINDER )
                 NUMBER OF SUB - REGIONS                              7
                 NUMBER OF  T  - REGIONS                              7
                 NUMBER OF  R  - REGIONS                              5
                 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                                7
                 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          2          3          4          5          5          5
       ===X-REGION NO./R  ===
                 1          1          1          1
       ===MATERIAL NO./R  ===
                 1          1          2          3          3
       ===R-X DIVISION    ===
        0.0        0.40000E+00 0.62500E+00 0.70100E+00 0.85000E+00 0.10432E+01 0.12000E+01 0.14753E+01
       VOLUME OF (S)-REGION
         1) 5.0265E-01  2) 7.2453E-01  3) 3.1660E-01  4) 7.2602E-01  5) 1.1491E+00  6) 1.1050E+00  7) 2.3137E+00
       VOLUME OF (T)-REGION
         1) 5.0265E-01  2) 7.2453E-01  3) 3.1660E-01  4) 7.2602E-01  5) 1.1491E+00  6) 1.1050E+00  7) 2.3137E+00
       VOLUME OF (R)-REGION
         1) 5.0265E-01  2) 7.2453E-01  3) 3.1660E-01  4) 7.2602E-01  5) 4.5678E+00
       VOLUME OF (X)-REGION
         1) 6.8376E+00
       VOLUME OF (M)-REGION
```

```
  1) 1.2272E+00  2) 3.1660E-01  3) 5.2938E+00
 TOTAL VOLUME                   6.8376E+00
 RATIOS OF VOLUME OF T-REGIONS NUMERICALLY INTEGRATED TO ANALYTIC
   1)  1.0003     2)  0.9993     3)  0.9997     4)  0.9998     5)  0.9997     6)  0.9998     7)  0.9997
     42 LINES DRAWN *** ELAPSED TIME      3 SEC

    ======  MACROF =====

  CASE I.D. : TCA2
  TITLE     : TCA  2 DOUBLE SPACED PIN ROD ARRAY  CYLINDRICAL APPROX

            COMPOSITION DATA LIST

                        ## MATERIAL NAME -----  1   FH2DX02X ##
  NUMBER OF NUCLIDE ------------        3
  TEMPERATURE (K) --------------      300.00
  MEAN CHORD LENGTH ------------     1.25000
  DANCOFF FACTOR ---------------     0.0
  IDENTIFICATION OF NUCLIDE ---     U050001     U080001     0060001
  NUMBER DENSITY ---------------  5.79899E-04 2.17000E-02 4.46100E-02
  RESONANT INDICATOR -----------       2           2           0
  LXMICR -----------------------       0           0           0
  NUCLIDE-WISE DANCOFF FACTOR WILL BE CALCULATED AT SUBROUTINE(MAFSIG) !!

                        ## MATERIAL NAME -----  2   CLADXOOX ##
  NUMBER OF NUCLIDE ------------        1
  TEMPERATURE (K) --------------      300.00
  MEAN CHORD LENGTH ------------     0.14000
  DANCOFF FACTOR ---------------     0.0
  IDENTIFICATION OF NUCLIDE ---     AL70001
  NUMBER DENSITY ---------------  6.23000E-02
  RESONANT INDICATOR -----------       0
  LXMICR -----------------------       0
  NUCLIDE-WISE DANCOFF FACTOR WILL BE CALCULATED AT SUBROUTINE(MAFSIG) !!

                        ## MATERIAL NAME -----  3   H2OCX00X ##
  NUMBER OF NUCLIDE ------------        2
  TEMPERATURE (K) --------------      300.00
  MEAN CHORD LENGTH ------------     0.0
  DANCOFF FACTOR ---------------     0.0
  IDENTIFICATION OF NUCLIDE ---     H010001     0060001
  NUMBER DENSITY ---------------  6.69200E-02 3.34600E-02
  RESONANT INDICATOR -----------       0           0
  LXMICR -----------------------       0           0
  NUCLIDE-WISE DANCOFF FACTOR WILL BE CALCULATED AT SUBROUTINE(MAFSIG) !!

                        ## RESONANT NUCLIDE NAME LIST ##

  NUMBER OF RESONANT NUCLIDE --       2
  RESONANT NUCLIDE NAME -------  U05   U08

                        ## STANDARD TEMPERATUE ARRAY LIST (STND) ##
 TAG         :     1        2        3        4        5        6        7        8        9        A        B
 TEMPERATURE :   300.00   325.00   350.00   400.00   450.00   500.00   550.00   600.00   900.00  1200.00  1600.00
 ## MEMBER(CLADF002) ALREADY EXISTS IN MACRO LIB. ##
 ## MEMBER(H2OCF002) ALREADY EXISTS IN MACRO LIB. ##
         STRAGE USED   7532 WITHIN 60000 IN PIJ2-STEP
         STRAGE USED   7532 WITHIN 60000 IN PIJ2-STEP

                              ****************************************
                              * CALCULATED NUCLIDE-WISE DANCOFF FACTOR *
                              ****************************************
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```
                    ## MATERIAL NAME -----  1   FH2DX02X ##
 IDENTIFICATION OF NUCLIDE ----    U050001     U080001     0060001
 NUCLIDE-WISE DANCOFF FACTOR---    0.021768    0.021768    1.000000

                    ## MATERIAL NAME -----  2   CLADX00X ##
 IDENTIFICATION OF NUCLIDE ----    AL70001
 NUCLIDE-WISE DANCOFF FACTOR---    1.000000

                    ## MATERIAL NAME -----  3   H2OCX00X ##
 IDENTIFICATION OF NUCLIDE ----    H010001     0060001
 NUCLIDE-WISE DANCOFF FACTOR---    1.000000    1.000000
STRAGE USED   1305 WITHIN 60000 IN PIJ2-STEP

===PIJ3 STEP===
***TCA2    ***TCA  2 DOUBLE SPACED PIN ROD ARRAY  CYLINDRICAL APPROX                *** IN FAST     RANGE ***

ITERATION PARAMETERS
PRINT (1+2+4+8)(FLUX,XEC,PIJ,S)                0
MAX OF INNER ITERATIONS PER OUTER             20
MAX OF OUTER ITERATIONS                       50
EARLIST EXTRAPOLATION                          5
NUMBER OF ITERATIONS TESTED                    5
MINIMUM DELAY                                  5
MONITOR PRINT(0,1)(SKIP,PRINT)                -1
CONVERGENCE CRITERION OF INNER     0.10000E-03
CONVERGENCE CRITERION OF OUTER     0.10000E-04
EXTRAPOLATION CRITERION            0.10000E-02
OVER-RELAXATION (INITIAL)          0.12000E+01
MAX EXTRAPOLATION                  0.10000E+03
BASE FACTOR OF OVER-RELAXATION     0.70000E+00

   ITERATION POWER-SCALING   RESIDUE CRITERON

    1          0.62334E-01 0.20104E-01 0.10000E-04
    2          0.64377E-01 0.25598E-02 0.10000E-04
    3          0.64117E-01 0.32570E-03 0.10000E-04
    4          0.64150E-01 0.41405E-04 0.10000E-04
    5          0.64146E-01 0.52687E-05 0.10000E-04

=== ITERATION END IN PIJF STEP ===
=== FIXED SOURCE TYPE PROBLEM ===
TOTAL THERMAL INNER ITERATION COUNT              0
INITIAL SOURCE NORMALIZATION FACTOR   0.52049E+01
THERMAL SOURCE NORMALIZATION FACTOR   0.10000E+01
RENORMALIZATION FACTOR (INNER)        0.0
RESIDUE IN FINAL INNER ITERATION      0.0
FINAL FISSION SOURCE NORMALIZATION    0.64146E-01
RESIDUE IN FINAL FISSION RATE         0.52687E-05
STRAGE USED  11617 WITHIN 60000 IN PIJ2-STEP

*** MIX-X-SECTION STEP IN *** FAST ENERGY RANGE
 *** TCA2    ***TCA  2 DOUBLE SPACED PIN ROD ARRAY  CYLINDRICAL APPROX              ***

 X-REGION  1***
 ONE GROUP CONSTANTS
 * ACTIVATION CROSSSECTION      0.62854E-01
 * FISSION    CROSSSECTION      0.14474E-02
 * NU*FISSION CROSSSECTION      0.37667E-02
 * TOTAL      CROSSSECTION      0.26449E+00
 * DIFFUSION COEFFICIENT 1     0.12603E+01
 * DIFFUSION COEFFICIENT 2     0.14564E+01
```

```
* ABSORPTION CROSSSECTION       0.52615E-02
* SCATTEROUT CROSSSECTION       0.36286E-01
* INTEGRATED FLUX-X-REGION      0.25094E+02
*** COMMENT *** DIFFUSION COEF. (BENOIST) NOT YET INCLUDED IN THE ABOVE LIST
STRAGE USED   1378 WITHIN 60000 IN PIJ2-STEP

===PIJ3 STEP===
***TCA2    ***TCA  2 DOUBLE SPACED PIN ROD ARRAY  CYLINDRICAL APPROX                *** IN THERMAL  RANGE ***

ITERATION PARAMETERS
PRINT (1+2+4+8)(FLUX,XEC,PIJ,S)              0
MAX OF INNER ITERATIONS PER OUTER          200
MAX OF OUTER ITERATIONS                      1
EARLIST EXTRAPOLATION                        5
NUMBER OF ITERATIONS TESTED                  5
MINIMUM DELAY                                5
MONITOR PRINT(0,1)(SKIP,PRINT)              -1
CONVERGENCE CRITERION OF INNER     0.10000E-03
CONVERGENCE CRITERION OF OUTER     0.10000E-04
EXTRAPOLATION CRITERION            0.10000E-02
OVER-RELAXATION (INITIAL)          0.12000E+01
MAX EXTRAPOLATION                  0.10000E+03
BASE FACTOR OF OVER-RELAXATION     0.70000E+00

=== ITERATION END IN PIJF STEP ===
=== FIXED SOURCE TYPE PROBLEM ===
TOTAL THERMAL INNER ITERATION COUNT            18
INITIAL SOURCE NORMALIZATION FACTOR   0.91055E+00
THERMAL SOURCE NORMALIZATION FACTOR   0.10000E+01
RENORMALIZATION FACTOR (INNER)        0.10000E+01
RESIDUE IN FINAL INNER ITERATION      0.99108E-04
FINAL FISSION SOURCE NORMALIZATION    0.0
RESIDUE IN FINAL FISSION RATE         0.98224E-08

*** MIX-X-SECTION STEP IN *** THER ENERGY RANGE
 *** TCA2    ***TCA  2 DOUBLE SPACED PIN ROD ARRAY  CYLINDRICAL APPROX               ***

 X-REGION  1***
 ONE GROUP CONSTANTS
 * ACTIVATION CROSSSECTION      0.47547E+01
 * FISSION    CROSSSECTION      0.33088E-01
 * NU*FISSION CROSSSECTION      0.80034E-01
 * TOTAL      CROSSSECTION      0.17686E+01
 * DIFFUSION COEFFICIENT 1      0.18847E+00
 * DIFFUSION COEFFICIENT 2      0.18847E+00
 * ABSORPTION CROSSSECTION      0.59115E-01
 * SCATTEROUT CROSSSECTION      0.82517E-04
 * INTEGRATED FLUX-X-REGION     0.15381E+02
 *** COMMENT *** DIFFUSION COEF. (BENOIST) NOT YET INCLUDED IN THE ABOVE LIST

    ***  BARE REACTOR SPECTRUM CALCULATION *** OF STEP 15 ***
 ***TCA2    ***TCA  2 DOUBLE SPACED PIN ROD ARRAY  CYLINDRICAL APPROX               ***
 *** BY B1 APPROXIMATION ***
 RATIO OF THERMAL SOURCE TO REMOVAL  0.10000E+01
 NEUTRON SPECTRUM PER LETHARGY
  0.46151E+00 0.18313E+01 0.37749E+01 0.39331E+01 0.35877E+01 0.32220E+01 0.23241E+01 0.15751E+01 0.10839E+01 0.88462E+00
  0.80283E+00 0.76790E+00 0.74940E+00 0.73850E+00 0.72847E+00 0.71418E+00 0.70463E+00 0.66829E+00 0.67781E+00 0.63504E+00
  0.65826E+00 0.65358E+00 0.67891E+00 0.72875E+00 0.74009E+00 0.77360E+00 0.78351E+00 0.80644E+00 0.87384E+00 0.10184E+01
  0.12339E+01 0.14384E+01 0.17332E+01 0.20977E+01 0.25751E+01 0.31390E+01 0.37392E+01 0.43547E+01 0.48940E+01 0.53021E+01
  0.54805E+01 0.53796E+01 0.49976E+01 0.43547E+01 0.35307E+01 0.26330E+01 0.17772E+01 0.10573E+01 0.53416E+00 0.21398E+00
  0.59165E-01 0.79602E-02 0.19735E-03
 NEUTRON CURRENT PER LETHARGY
```

```
 0.10182E-01 0.34324E-01 0.64248E-01 0.57104E-01 0.40268E-01 0.32827E-01 0.20634E-01 0.12473E-01 0.75247E-02 0.54216E-02
 0.45471E-02 0.41333E-02 0.39293E-02 0.38410E-02 0.37782E-02 0.37016E-02 0.36683E-02 0.35385E-02 0.34564E-02 0.34484E-02
 0.33020E-02 0.32919E-02 0.24390E-02 0.23483E-02 0.21609E-02 0.21658E-02 0.21097E-02 0.21158E-02 0.22074E-02 0.24363E-02
 0.28143E-02 0.31690E-02 0.36773E-02 0.42772E-02 0.50424E-02 0.58851E-02 0.66923E-02 0.74165E-02 0.78928E-02 0.80392E-02
 0.77426E-02 0.70088E-02 0.59489E-02 0.47025E-02 0.34430E-02 0.23131E-02 0.14050E-02 0.75092E-03 0.33846E-03 0.11829E-03
 0.27002E-04 0.26216E-05 0.34438E-07
 FAST FISSION          0.74119E-01
 FAST ABSORPTION       0.10297E+00
 FAST LEAKAGE          0.19090E+00
 TOTAL FISSION         0.10063E+01
 TOTAL ABSORPTION      0.79149E+00
 TOTAL LEAKAGE         0.20851E+00
 K-EFF= 1.00627 KINF= 1.27135 UNDER GEOMTRICAL BUCKLING= 0.80000E-02
```

```
     **** CASE ID === CR19      *** TITLE === 2-D  DIFFUSION CALCULATION FOR TWO REGION 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)          0
    DIFF COEF (INVERSE TR,BENOIST ISO,BENOIST ANISO:1,2,3)      0
    REACTION RATE CALCULATION (0,1)                             0
    PRINT OF MACRO-X (0,1,2)                                    0
    BURN-UP CALCULATION (0,1:SKIP,EXECUTE)                      0

         ***** MULTI-DIMENSIONAL DIFFUSION CODE INPUT STEP-I *****

      NUMBER OF MATERIALS USED IN THIS ROUTINE                                  3
      NUMBER OF X-REGIONS ( =0 MIX CROSS SECTION STEP IS SKIPPED)               0
      OPTION TO SELECT DIFFUSION COEFFICIENT (1,2,3,<0) = (D1,D2,D1&D2,SPECIAL) 3
      SELECTION OF D1 AND D2 FOR EACH AXES AND EACH MATERIALS                   5          5          1

                    **********CIT1**********

CORE19 ORGANIZED BY CENTRAL 20 LINES BY DOUBLE SPACED ARRAY, BOTH SIDE
5 LINES BY STANDARD: EACH LINE BY 18 RODS OF ACTUAL HEIGHT 60.95 CM

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   0
   1   0   0   0   0   0   0   0   0   0   0   0   1   0   0   1   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  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   1   1   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  3.774998E+01  HEIGHT  2.849998E+01

 REGION SPECIFICATIONS
   PTS  REGION WIDTH
    20  1.850000E+01    10  9.250000E+00    10  1.000000E+01
   PTS  REGION HEIGHT
    10  1.850000E+01    10  1.000000E+01
 X-DIR. POINTS  40        Y-DIR. POINTS  20
 DISTANCES TO MESH INTERVAL INTERFACES
    J   DIST.
    2   0.925    3   1.850    4   2.775    5   3.700    6   4.625    7   5.550    8   6.475    9   7.400   10   8.325
```

```
  11   9.250   12  10.175   13  11.100   14  12.025   15  12.950   16  13.875   17  14.800   18  15.725   19  16.650
  20  17.575   21  18.500   22  19.425   23  20.350   24  21.275   25  22.200   26  23.125   27  24.050   28  24.975
  29  25.900   30  26.825   31  27.750   32  28.750   33  29.750   34  30.750   35  31.750   36  32.750   37  33.750
  38  34.750   39  35.750   40  36.750   41  37.750
   I   DIST.
   2   1.850    3   3.700    4   5.550    5   7.400    6   9.250    7  11.100    8  12.950    9  14.800   10  16.650
  11  18.500   12  19.500   13  20.500   14  21.500   15  22.500   16  23.500   17  24.500   18  25.500   19  26.500
  20  27.500   21  28.500
DISTANCES TO FLUX POINTS
   J   DIST.
   1   0.462    2   1.387    3   2.312    4   3.237    5   4.162    6   5.087    7   6.012    8   6.937    9   7.862
  10   8.787   11   9.712   12  10.637   13  11.562   14  12.487   15  13.412   16  14.337   17  15.262   18  16.187
  19  17.112   20  18.037   21  18.962   22  19.887   23  20.812   24  21.737   25  22.662   26  23.587   27  24.512
  28  25.437   29  26.362   30  27.287   31  28.250   32  29.250   33  30.250   34  31.250   35  32.250   36  33.250
  37  34.250   38  35.250   39  36.250   40  37.250
   I   DIST.
   1   0.925    2   2.775    3   4.625    4   6.475    5   8.325    6  10.175    7  12.025    8  13.875    9  15.725
  10  17.575   11  19.000   12  20.000   13  21.000   14  22.000   15  23.000   16  24.000   17  25.000   18  26.000
  19  27.000   20  28.000

ZONE INPUT BY REGION
   2   1   3
   3   3   3
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 26 27 28 29 30 31 32 33 34 35 36 37 38 39 40

  1   2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  1  1  1  1  1  1  1  1  1  1  3  3  3  3  3  3  3  3  3  3
  2   2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  1  1  1  1  1  1  1  1  1  1  3  3  3  3  3  3  3  3  3  3
  3   2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  1  1  1  1  1  1  1  1  1  1  3  3  3  3  3  3  3  3  3  3
  4   2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  1  1  1  1  1  1  1  1  1  1  3  3  3  3  3  3  3  3  3  3
  5   2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  1  1  1  1  1  1  1  1  1  1  3  3  3  3  3  3  3  3  3  3
  6   2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  1  1  1  1  1  1  1  1  1  1  3  3  3  3  3  3  3  3  3  3
  7   2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  1  1  1  1  1  1  1  1  1  1  3  3  3  3  3  3  3  3  3  3
  8   2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  1  1  1  1  1  1  1  1  1  1  3  3  3  3  3  3  3  3  3  3
  9   2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  1  1  1  1  1  1  1  1  1  1  3  3  3  3  3  3  3  3  3  3
 10   2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  2  1  1  1  1  1  1  1  1  1  1  3  3  3  3  3  3  3  3  3  3
 11   3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3
 12   3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3
 13   3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3
 14   3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3
 15   3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3
 16   3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3
 17   3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3
 18   3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3
 19   3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3
 20   3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3  3

MEMORY LOCATIONS RESERVED FOR DATA STORAGE---  60000
MEMORY LOCATIONS USED FOR THIS PROBLEM-------  47248
MEMORY LOCATIONS NOT USED--------------------  12752
         MATERIAL NO.
                  1         2         3

                **********CIT2 **********
           *********THIS JOB WAS RUN ON  86-03-05 ON THE FACOM M-380  **********

CORE19 ORGANIZED BY CENTRAL 20 LINES BY DOUBLE SPACED ARRAY, BOTH SIDE
5 LINES BY STANDARD: EACH LINE BY 18 RODS OF ACTUAL HEIGHT 60.95 CM

FISSION SOURCE DISTRIBUTION AND SUM  0.56219  0.42038  0.01739  0.00003  0.0      0.0      0.0      0.0      0.0      0.0
  0.99999
```

```
CORE STORAGE DIFFERENCE (WORDS) EQUATION CONSTANTS I/O INSTEAD OF STORED   20921

EQUATION CONSTANTS WILL BE STORED IN CORE
NUMBER OF---COLUMNS, ROWS, PLANES, GROUPS, UPSCAT, DOWNSCAT, REGIONS, AND ZONES      40  20   1  10   1   9   6   3
MEMORY LOCATIONS RESERVED FOR DATA STORAGE---  60000
MEMORY LOCATIONS USED FOR THIS PROBLEM-------  47248
MEMORY LOCATIONS NOT USED-------------------  12752

**********WARNING**********  INPUT SPECIFIED UPSCATTER =  1  HAS BEEN CHANGED TO ACTUAL UPSCATTER =  5

**********WARNING**********  INPUT SPECIFIED DOWNSCATTER =  9  HAS BEEN CHANGED TO ACTUAL DOWNSCATTER =  5
CORE19 ORGANIZED BY CENTRAL 20 LINES BY DOUBLE SPACED ARRAY, BOTH SIDE
5 LINES BY STANDARD: EACH LINE BY 18 RODS OF ACTUAL HEIGHT 60.95 CM

LINE RELAXATION WILL BE DONE ON ROWS -   1 INNER ITERATION(S)
ITERATION  FLUX CHANGE    BETA      MU-1      MU-2      MU-3          K
  ..... 46 lines deleted .....
   48     -1.04308E-04 1.439E+00-8.682E-01-7.503E-01 8.130D-01      0.996494
   49     -9.50098E-05 1.439E+00 9.108E-01 3.678E-01 7.082D-01      0.996495

END OF EIGENVALUE CALCULATION - ITERATION TIME  0.080 MINUTES

CONVERGENCE INDICATION BY MINIMIZING THE SUM OF THE SQUARES OF THE RESIDUES - RELATIVE ABSORPTION  1.0000048   K  0.9965034

    LEAKAGE  4.32118E+15  TOTAL LOSSES  7.85512E+16  TOTAL PRODUCTIONS  7.82759E+16  REACTOR POWER(WATTS)  1.00000E+06

CORE19 ORGANIZED BY CENTRAL 20 LINES BY DOUBLE SPACED ARRAY, BOTH SIDE

END OF CASE - TOTAL CPU TIME WAS   0.09 MINUTES   TOTAL CLOCK TIME WAS   0.64 MINUTES
**********************************************************************************************************
**********************************************************************************************************

                              *********THIS JOB WAS RUN ON  86-03-05  ON THE FACOM M-200  **********

JOB ENDED NORMALLY
```

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