JAEA-Data/Code
2007-003

JP0750044

JAEA-Data/Code

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Keisuke OKUMURA

Reactor Physics Group
Nuclear Science and Engineering Directorate

February 2007

Japan Atomic Energy Agency

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〒319-1195　茨城県那珂郡東海村白方白根2番地4
日本原子力研究開発機構　研究技術情報部　研究技術情報課
電話 029-282-6387, Fax 029-282-5920

*〒319-1195　茨城県那珂郡東海村白方白根2番地4　日本原子力研究開発機構内

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Tel +81-29-282-6387, Fax +81-29-282-5920

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COREBN: A Core Burn-up Calculation Module for SRAC2006

Keisuke OKUMURA

Division of Nuclear Data and Reactor Engineering
Nuclear Science and Engineering Directorate
Japan Atomic Energy Agency
Tokai-mura, Naka-gun, Ibaraki-ken

(Received January 4, 2007)

COREBN is an auxiliary code of the SRAC system for multi-dimensional core burn-up calculation based on the diffusion theory and interpolation of macroscopic cross-sections tabulated to local parameters such as burn-up degree, moderator temperature and so on. The macroscopic cross-sections are prepared by cell burn-up calculations with the collision probability method of SRAC. SRAC and COREBN have wide applicability for various types of cell and core geometries. They have been used mainly for the purpose of core burn-up management of research reactors in Japan Atomic Energy Agency.

The report is a revision of the users manual for the latest version of COREBN served with the SRAC released in 2006.

Keywords: Core Burn-up Calculation, Diffusion Theory, COREBN, SRAC, Burn-up Management of Research Reactor

COREBN:SRAC2006 の炉心燃焼計算モジュール

日本原子力研究開発機構 原子力基礎工学研究部門
核工学・炉工学ユニット
奥村　啓介

(2007年1月4日受理)

COREBN は拡散理論と燃焼度や減速材温度などの局所パラメータに対してテーブル化された巨視的断面積の内挿法に基づく多次元炉心燃焼計算のための SRAC システムの補助計算コードである。巨視的断面積は SRAC の衝突確率法による格子燃焼計算により作成される。SRAC と COREBN は多様な格子及び炉心形状に対して幅広い適用性がある。これらのコードは主に日本原子力研究開発機構の研究炉の燃焼管理のために利用されてきた実績を有する。

本報告書は、2006 年版 SRAC とともに提供される最新版の COREBN に対する改訂版ユーザー利用マニュアルである。

原子力科学研究所（駐在）：〒319-1195 茨城県那珂郡東海村白方白根 2-4

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1. COREBN and HIST; Core Burn-up Calculation and Fuel Management Codes

1.1 History of COREBN and HIST

COREBN is an auxiliary code of the SRAC system[1)-4)] for multi-dimensional core burn-up calculation based on the diffusion theory. HIST is a utility code to manage binary data of COREBN I/O file which is called "history file". Detailed burn-up information of the core and loaded fuels obtained from COREBN is recorded in the history file.

The first versions[2)] of the COREBN and HIST codes were developed in the 1980s and implemented into the SRAC system. After that, they have been mainly used for the purpose of core burn-up management of research and test reactors such as JRR-2, JRR-3, JRR-4 and JMTR in Japan Atomic Energy Agency. Furthermore, they were used for the study of the reduced enrichment program[5)-7)] for the research reactors and the program was successfully completed by 1999.

As the first versions had many restrictions in usage, several improvements have been made together with revision of SRAC. Details of the improvements are described in Sect.2. It is recommended to use the COREBN and HIST codes served with the latest SRAC2006[4)] to avoid inconsistency of interface data among SRAC, COREBN and HIST.

Hereafter, we assume that a user of COREBN has enough knowledge and experience at least for the cell burn-up calculation of SRAC.

1.2 Cell Burn-up Calculation and Macroscopic Cross-Section Table

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

In the SRAC system, core burn-up calculation process is divided into two steps. First, a cell burn-up calculation with SRAC yields few group homogenized macroscopic cross-sections at each burn-up step. The cross-sections are tabulated on the discrete values of fuel temperature, coolant temperature, and degree of burn-up for the cell modeling an actual fuel assembly in the core. In the next core burn-up calculation step with COREBN, necessary macroscopic cross-sections are prepared

by interpolation of the macroscopic cross-section tables on the core parameters such as fuel and moderator temperatures and burn-up degrees distributed in the core.

As the result of the cell burn-up calculation by SRAC, macroscopic cross-sections of every burn-up step are stored in the MACRO file. The member is named following the rules of SRAC, as '*case*A*bx*0' by 8 characters, where *case*-tag is to identify fuel type, *b*-tag (=0,1,2,…., 9,A,B,…) for burn-up step and *x*-tag for homogenized region (cf. Secton 3.1.5 of SRAC manual[4]). Degree of burn-up, atomic number densities of depleting nuclides, the data for Xe and Sm accumulation, *etc.* at every burn-up step are stored on the member *case*DN*x*T (*x*: X-Region number) for every X-Region. A set of these members takes a role of tabulation of macroscopic cross-sections as shown in Fig.1.2-1.

Macroscopic Cross Section Table

Burn-up Step	0	1	2	--------	10	11	--------
Member Name	caseA0xp	caseA1xp	caseA2xp	--------	caseAAxp	caseABxp	--------
$\Sigma_{f,g}$							
$\nu\Sigma_{f,g}$							
$\Sigma_{a,g}$							
χ_g							
$\Sigma_{t,g}$							
⋮							

Nuclide Density Table (Member Name: caseDNxT)

Burn-up Step	0	1	2	--------	10	11	--------
Burn-up(MWd/t)	0	100	500	--------	30000	35000	--------
U-235 (n/cm^3)							
U-236							
U-238							
Pu-238							
⋮							

Fig.1.2-1 Interpolation tables for macroscopic cross-sections (upper) and atomic number densities (lower), using the PDS file of SRAC

The types of tabulated cross-sections are as follows: 1) the cross-sections for diffusion or transport calculation ($\Sigma_{f,g}$, $\nu\Sigma_{f,g}$, $\Sigma_{a,g}$, χ_g, $\Sigma_{s,g\to g'}$, $D_{\perp,g}$, $D_{//,g}$, $\Sigma_{t,g}$), 2) $\Sigma_{(n,2n),g}$ for (n,2n) correction, 3) $\Sigma_{a,g}^{fiss}$ (absorption cross-sections of fissile nuclides) and $\Sigma_{c,g}^{fert}$ (capture cross-sections of fertile nuclides) for conversion ratio calculation, 4) delayed neutron data ($\chi_{d,g}$, $\beta_j\nu\Sigma_{f,g}$, $\beta_j\nu\Sigma_{f,g}/\lambda_j$) and $\Sigma_{v,g}$ (constant on neutron velocity) for point kinetics parameters, and 5) effective microscopic cross-sections and effective fission yields ($\sigma_{a,g}^{Xe}, \sigma_{a,g}^{I}, \sigma_{a,g}^{Sm}, \sigma_{a,g}^{Pm}, y^{Xe}, y^{I}, y^{Sm}, y^{Pm}$) to evaluate equilibrium Xe and Sm concentrations. While the COREBN does not calculate depletion of constituent nuclides, it obtains $N_k(\vec{r},t)$: the atomic number density of depleting nuclide (k) at burn-up time (t) by interpolation of the tabulated compositions with burn-up as argument that are stored in the member *case*DN*x*T.

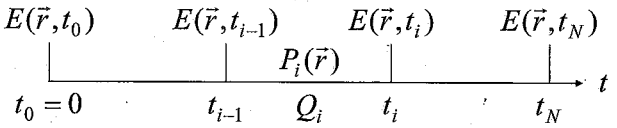
1.3 COREBN Code

(1) Interpolation of macroscopic cross-sections

The COREBN is a code for multi-dimensional core burn-up calculation based on macroscopic cross-section interpolation and finite difference diffusion. As the CITATION code is incorporated for the solution of diffusion calculation, the 12 geometries available in CITATION are also available in COREBN.

Before executing core burn-up calculation by COREBN, it is necessary to prepare macroscopic cross-section table for every type of fuel assembly by executing cell burn-up calculation with SRAC. By executing cell burn-up calculation, cross-sections homogenized over the cell are automatically tabulated with respect to burn-up in the member on MACRO file under the prescribed format (See Fig.1.2-1). When spatial distribution of fuel temperature and moderator temperature is considered, cross-section tables on several fuel temperature and/or moderator temperature are required.

After preparing cross-section tabulation, the core burn-up calculation is achieved. At first, we divide target burn-up exposure into N steps.



During a core burn-up step $t_{i-1} \le t \le t_i$,(i=1,2,...N), the total thermal power of the core Q_i and power

distribution $P_i(\vec{r})$ are assumed unchanged. The initial burn-up distribution $E(\vec{r},t_0)$ is given by the input of HIST. At the burn-up time $t=t_{i-1}$, COREBN forms the cross-sections of every spatial region by linear interpolation with specified three parameters; burn-up $E(\vec{r},t_{i-1})$, fuel temperature $T_f(\vec{r},t_{i-1})$ and moderator temperature $T_m(\vec{r},t_{i-1})$ from the cross-sections tabulated three-dimensionally with respect to burn-up, fuel temperature and moderator temperature.

$$\Sigma_{x,g}(\vec{r},t_{i-1}) = \Sigma_x \mathrm{TABLE}_m \left\{E(\vec{r},t_{i-1}),\ T_f(\vec{r},t_{i-1}),\ T_m(\vec{r},t_{i-1})\right\} \tag{1.3-1}$$

where,

x	: reaction type,
$\Sigma_x \mathrm{TABLE}_m$	: table-look-up of macroscopic cross-sections of reaction type x of fuel type m located at region $\vec{r}$,
$E(\vec{r},t_{i-1})$	: burn-up (MWd/cc) at time t_{i-1}, region $\vec{r}$,
$T_f(\vec{r},t_{i-1})$	: fuel temperature at time t_{i-1}, region $\vec{r}$,
$T_m(\vec{r},t_{i-1})$	: moderator temperature at time t_{i-1}, region $\vec{r}$,

Fuel composition is evaluated by interpolation of atomic number density table.

$$N_k(\vec{r},t_{i-1}) = \mathrm{N}_k \mathrm{TABLE}_m \left\{E(\vec{r},t_{i-1}),\ T_f(\vec{r},t_{i-1}),\ T_m(\vec{r},t_{i-1})\right\} \tag{1.3-2}$$

where,

$\mathrm{N}_k \mathrm{TABLE}_m$	: table-look-up of atomic number density table for the k-th depleting nuclide in the fuel type m located at region $\vec{r}$,
$N_k(\vec{r},t_{i-1})$	: atomic number density of nuclide k at time t_{i-1}, region $\vec{r}$, (10^{24} /cm^3)

After preparing the cross-sections for every region, relative neutron flux in the core $\varphi_g(\vec{r},t_{i-1})$ is obtained by COREBN. The relative flux distribution is normalized to give the whole thermal power Q_i (MW) into $\Phi_g(\vec{r},t_{i-1})$ as Eqs.(1.3-3) and (1.3-4).

$$\Phi_g(\vec{r},t_{i-1}) = F_{norm} \times \varphi_g(\vec{r},t_{i-1}) \tag{1.3-3}$$

$$F_{norm} = \frac{Q_i}{\int dV \sum_g \gamma(\vec{r},t_{i-1}) \Sigma_{f,g}(\vec{r},t_{i-1}) \varphi_g(\vec{r},t_{i-1})} \tag{1.3-4}$$

In Eq.(1.3-4), γ is the averaged release energy per fission. The power distribution $P(\vec{r},t_{i-1})$ [MW/cm^3] is expressed by Eq.(1.3-5). If we assume the power distribution is unchanged until $t=t_i$, multiplying burn-up period $\Delta t_i = t_i - t_{i-1}$ (days) gives burn-up increment (MWd/t) as Eq.(1.3-6). Updating burn-up distribution by Eq.(1.3-7), then the process moves to the next burn-up step.

$$\mathrm{P}(\vec{r},t_{i-1}) = \gamma(\vec{r},t_{i-1})\sum_{g}\left\{\Sigma_{f,g}(\vec{r},t_{i-1})\Phi_{g}(\vec{r},t_{i-1})\right\}V(\vec{r}) \tag{1.3-5}$$

$$\Delta E(\vec{r},t_{i}) = P(\vec{r},t_{i})(t_{i}\text{-}t_{i-1})/INV(\vec{r}) \tag{1.3-6}$$

where, INV is the conversion factor for unit of burn-up degree from MWd/cm^3 to MWd/t, that is the initial heavy metal inventory in the unit volume (ton/cm^3).

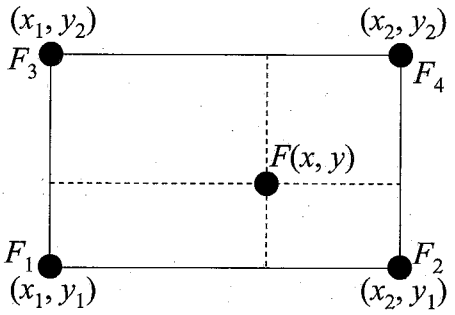
$$E(\vec{r},t_{i}) = E(\vec{r},t_{i-1}) + \Delta E(\vec{r},t_{i}) \tag{1.3-7}$$

In the burn-up calculation in COREBN, the process expressed from Eq.(1.3-1) to Eq.(1.3-7) is repeated for i=1,2,…, N. However, at the final step: i=N+1, the process after Eq.(1.3-2) is not executed. It means that at the final step, the burn-up distribution and fuel composition are calculated, but either neutron flux or collapsed homogenized cross-sections are not obtained.

The COREBN interpolates the cross-sections with respect to three parameters as described above, two parameters except burn-up can be replaced by arbitrary measures by user's selection. For example, instead of fuel temperature and moderator temperature, boron concentration for chemical shim, moderator void fraction or fraction of control rod insertion can be substituted. Hereafter, description will be given by assuming that the two parameters are fuel and moderator temperatures.

As COREBN has no function of coupled neutronics and thermal hydraulics calculation, spatial distribution of fuel temperature and moderator temperature has to be specified by user. The way of specifications are (1) to set constant value over the whole fuel assembly in the core, or (2) to give the axial distribution (by axial node) for every fuel assembly.

Cross-sections are interpolated with respect to burn-up, fuel temperature and moderator temperature by the method as follows;



In the above figure, let x to be fuel temperature, y moderator temperature and F_1 though F_4 the temperature points on which cross-sections are tabulated by execution cell burn-up calculation with SRAC. The interpolated value $F(x, y)$ on the point (x, y) is obtained by the following relations;

$$R_x = \frac{x - x_1}{x_2 - x_1}, \qquad R_y = \frac{y - y_1}{y_2 - y_1} \tag{1.3-8a}$$

$$FX_1 = F_1 + (F_2 - F_1)R_x, \qquad FX_2 = F_3 + (F_4 - F_3)R_x \tag{1.3-8b}$$

$$F = FX_1 + (FX_2 - FX_1)R_y \tag{1.3-8c}$$

The above shows a scheme for two-dimensional interpolation. Finally, the value $F(x, y, z)$ is obtained by the liner interpolation between $F(x, y, z_1)$ and $F(x, y, z_2)$ where burn-up z lies between z_1 and z_2. When combination (x, y) coincides either of (x_1, y_1), (x_2, y_2), (x_3, y_3) or (x_4, y_4), interpolation is not necessary. But when x coincides either of x_1 or x_2, or y coincides either of y_1 or y_2, the one-dimensional interpolation is executed. As the code does not extrapolate, cross-sections have to be prepared so that the parameter range cover the value required in the core burn-up calculation.

When burn-up characteristics of irradiated sample inserted in the core of test reactor such as JMTR at JAEA is required, if the sample is composed by non-fissionable material such as Gd_2O_3, interpolation with respect to burn-up (MWd/t) is not achievable. The COREBN interpolates cross-sections with respect to integrated absorption reaction rate defined by Eq.(1.3-9), instead of burn-up.

$$E_a = \int_0^t \sum_g \Sigma_{a,g} \Phi_g \, dt \tag{1.3-9}$$

$$E_a(\vec{r}, t_i) \approx E_a(\vec{r}, t_{i-1}) + \sum_g \Sigma_{a,g}(\vec{r}, t_{i-1}) \Phi_g(\vec{r}, t_{i-1})(t_i - t_{i-1}) \tag{1.3-10}$$

The COREBN does interpolate atomic number densities of depleting nuclides, average release energy per fission, the constants necessary to evaluate conversion ratio and kinetics parameters, (n,2n) reaction cross-sections and microscopic cross-sections and fission yields of I-135 and Xe-135 by the process described by Eq.(1.3-8) as well as macroscopic cross-sections.

(2) Treatment and drive of control rod

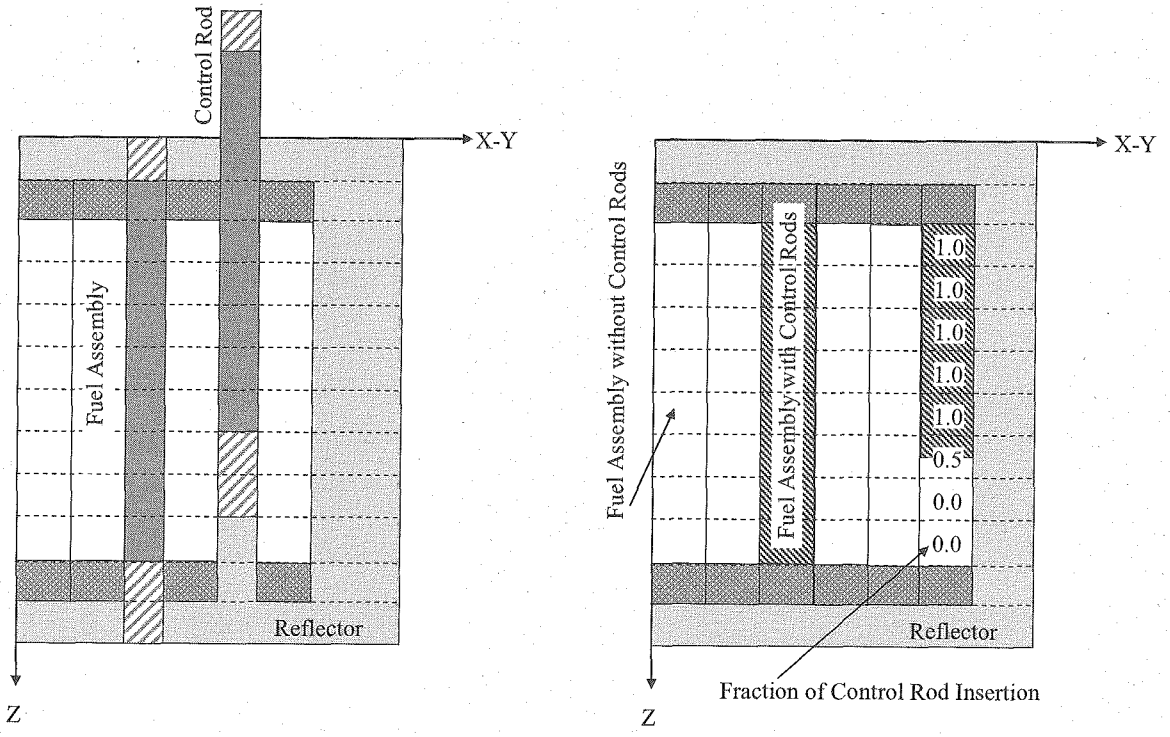
The treatment of control rod by COREBN is different by the following two cases.

(a) Control rod region is spatially separated from fuel region, then cross-sections of control rod region is separately prepared as occurs in research reactors or in FBRs.

(b) Control rod is homogenized in fuel assembly as in PWR clustered control rods.

In the former case (a), as shown in Fig.1.3-1(a), material composition is allocated to the nodes divided along the axial direction of control rod, and the allocation can be axially shifted along the core burn-up proceeds. The projection from the upper surface of the core is neglected in the diffusion calculation. Contrary, the region where the control rod is withdrawn is automatically substituted by the material registered beforehand. However, the drive of control rod is not continuous

but discrete by the unit of axial node.



(a) Control rods separated from fuel region (b) Control rods homogenized in fuel region

Fig.1.3-1 Two different treatments of control rods in COREBN

When the control rods consists of strongly absorbing lump, the diffusion calculation can not yield accurate worth. In such a case, use of the blackness option to the material corresponding to control rod region is recommended. In the blackness option, not only perfect black absorber but also logarithmic differential boundary condition (zero flux at extrapolated distance) for certain thermal groups and usual cross-sections for fast groups can be accepted. The group dependent constants for extrapolated distance may be specified by the user. These values are decided in comparison with the results of transport calculation or the measured data. The control assembly including black absorber can be driven axially. The blackness option is often used for JAEA core burn-up calculation of research reactors.

In the latter case (b), as shown in Fig.1.3-1(b), the region where the control rods are inserted and the region where the rods are withdrawn are discriminated by the replacement of cross-sections of fuel assembly. In this case, the continuous drive of control rod can be expressed by the fraction of control of insertion in an axial node by using the interpolation function for temperature interpolation. For example, a cell burn-up calculation by SRAC makes cross-section table with the condition of control rod withdrawal. These cross-sections correspond to the fraction of control rod insertion =0.

Next, the branch-off calculation makes cross-section table with the condition of control rod insertion. These correspond to the fraction of control rod insertion = 1. Replacing fuel temperature (x) by fraction of control rod insertion (f_{cr}) in Eq.(1.3-8) makes x_1=0.0, x_2=1.0. and R_x=x=f_{cr}, then Eq.(1.3-8b) is rewritten by Eq.(1.3-11).

$$FX_1 = F_1 + (F_2 - F_1)f_{cr} = (1 - f_{cr})F_1 + f_{cr}F_2 \tag{1.3-11}$$

where, F_1: Cross-sections corresponding to control rod withdrawal
F_2: Cross-sections corresponding to control rod insertion

That is, along the axial nodes, if f_{cr} =0.0 is specified, cross-section for control rod withdrawal is used and if f_{cr} =1.0, those for control rod insertion is used. And if the intermediate value 0< f_{cr} <1.0 is specified to a node, cross-sections averaged with volume weight by inserted and withdrawn cross-sections are used. The continuous movement of control rods can be simulated if the fractions of control rod insertion 0.0 and 1.0 are set in the place of fuel or moderator temperature points, and fractions of control rod insertions are specified by input on every axial node at the control rod position. In this case, there is not such a restriction on control rod movement as posed in case (a).

When an analysis of control rod worth is required, repeat the core burn-up calculation of zero exposure period with driving the control rods step by step, then reactivity curve (S-curve) of control rods will be obtained.

As the COREBN does not have any criticality search function, if burn-up calculation under critical condition is required, depth of control rod insertion or soluble boron concentration has to be beforehand estimated by some preliminary calculations.

(3) Correction of equilibrium xenon concentration and its extension

The equilibrium xenon concentration depends generally on thermal power (neutron flux level). When some local power greatly differs from the average thermal power which was assumed to form cross-section table by SRAC, as the local equilibrium Xe-135 concentration differs from that calculated by SRAC, it may cause deviation of whole reactor characteristics.

The time behavior of concentration of Xe-135 and its parent nuclide I-135 is expressed by Eq.(1.3-12) and (1.3-13). All the parameters except the decay constants (λ_X, λ_I) are space dependent.

$$\frac{dN_{Xe}}{dt} = -\lambda_{Xe}N_{Xe} - N_{Xe}\sum_g \{\sigma_{Xe,g}\Phi_g\} + y_{Xe}\sum_g \{\Sigma_{f,g}\Phi_g\} + \lambda_I N_I \tag{1.3-12}$$

$$\frac{dN_I}{dt} = -\lambda_I N_I - N_I\sum_g \{\sigma_{I,g}\Phi_g\} + y_I\sum_g \{\Sigma_{f,g}\Phi_g\} \tag{1.3-13}$$

As the absorption cross-section of I-135 is enough small ($\sigma_{I,g}\cong 0$), with assumption that the

concentration of I-135 reaches equilibrium (dN_I/dt=0), The equilibrium concentration of I-135 ($N_{I,\infty}$) is written by Eq.(1.3-14).

$$N_{I,\infty} = \frac{y_I \sum_g \{\Sigma_{f,g}\Phi_g\}}{\lambda_I} \tag{1.3-14}$$

Substituting it into N_I in Eq.(1.3-12) with letting dN_{Xe}/dt=0, we obtain the equilibrium concentration of Xe-135 by

$$N_{Xe,\infty} = \frac{(y_{Xe} + y_I)\sum_g \{\Sigma_{f,g}\Phi_g\}}{\lambda_{Xe} + \sum_g \{\sigma_{Xe,g}\Phi_g\}} \tag{1.3-15}$$

The COREBN has an option to correct absorption cross-sections upon assuming Xe-135 concentration is in equilibrium condition at each burn-up step. This correction is performed by Eq.(1.3-16).

$$\Sigma_{a,g} = \Sigma^0_{a,g} + (N_{Xe,\infty} - N^0_{Xe})\sigma_{Xe,g} = \Sigma^0_{Xe,g} + \left[\frac{(y_{Xe} + y_I)\sum_g \{\Sigma_{f,g}\Phi_g\}}{\lambda_{Xe} + \sum_g \{\sigma_{Xe,g}\Phi_g\}} - N^0_{Xe}\right]\sigma_{Xe,g} \tag{1.3-16}$$

$\Sigma_{a,g}$: Absorption cross-section after correction

$\Sigma^0_{a,g}$: Interpolated absorption cross-section before correction

$N_{Xe,\infty}$: Local Xe-135 concentration in equilibrium

N^0_{Xe} : Interpolated Xe-135 concentration before correction

$\sigma_{Xe,g}$: Interpolated microscopic absorption cross-section of Xe-135

$\Sigma_{f,g}$: Interpolated fission cross-section

y_x : Interpolated fission yield of Xe-135

y_I : Interpolated fission yield of I-135

λ_x : Decay constant of Xe-135 (2.1205E-5 s^{-1})

Φ_g : Local neutron flux at the previous step

The local neutron flux Φ_g should be that of the current step. However, it is substituted by that of the previous step to avoid iterative process due to feed back. Therefore, at the first step, this option is ineffective.

The xenon buildup occurs after reactor shutdown, but it diminishes after sufficient elapsed time by decay. As the half-life of Xe-135 is about 9 hours, almost of Xe-135 does not exist at the restart after refueling. Therefore, at the reactor start up loaded with shuffled fuel, Xe-135 should be zero. The COREBN installs the option to set Xe-135 concentration zero at the initial step by using Eq.(1.3-17) instead of Eq.(1.3-16). On and after the second step, the correction of equilibrium xenon concentration is performed by Eq.(1.3-16).

$$\Sigma_{a,g} = \Sigma^0_{a,g} - N^0_{Xe}\sigma_{Xe,g} \tag{1.3-17}$$

The value of $\sigma_{Xe,g}$ in Eq.(1.3-17) is obtained by homogenization and collapsing by using the spectrum evaluated in the cell including Xe-135. Therefore, this method is an approximation valid in the low xenon concentration. When more exact treatment is required for the core calculation at the start up, cross-section table with zero xenon condition should be formed by the combination of the option to feed number densities of specified nuclides (cf. option IBC9 in Sect.2.11 of SRAC manual[4]) and the branch-off calculation option (IBC3) in SRAC. By replacing the cross-section table for COREBN by this table, the core calculation with zero xenon concentration is available.

In the evaluation of void coefficient, as the change of void fraction is instantaneous, the xenon concentration should not be that at the accidental void condition but that at the nominal operation condition. The equilibrium xenon concentration at the nominal operation condition is stored in the history file described in the next section when the xenon concentration correction option is applied. The COREBN has an option to correct cross-section with xenon concentration N^{Hist}_{Xe} read from the history file as described in Eq.(1.3-18) only at the first step. On and after the second step, the correction is performed by Eq.(1.3-16). By modifying the xenon concentration written in the history file, the user can start burn-up calculation with xenon distribution specified by the user.

$$\Sigma_{a,g} = \Sigma^0_{a,g} + \left(N^{Hist}_{Xe} - N^0_{Xe}\right)\sigma_{Xe,g} \tag{1.3-18}$$

Another option is available to change the xenon concentration from the history file by multiplying an input value (*Fact*) as follows.

$$\Sigma_{a,g} = \Sigma^0_{a,g} + \left(Fact \times N^{Hist}_{Xe} - N^0_{Xe}\right)\sigma_{Xe,g} \tag{1.3-19}$$

This option is convenient, for example, to get reactivity on xenon concentration. The value N^{Hist}_{Xe} is kept as constant during burn-up. The burn-up calculation with zero exposure period can yield the above reactivity table by giving various *Fact* at each step.

(4) Treatment of (n,2n) reaction

The effect of (n,2n) reaction may be neglected in thermal reactors. However, it can not be neglected in hard spectrum core and in the system including beryllium. The diffusion equation to include (n,2n) reaction is given by Eq.(1.3-20). As usual deterministic code such as transport and diffusion codes does not treat explicitly (n,2n) reaction, the contribution of (n,2n) reaction is added to scattering matrix as shown in Eq.(1.3-21). The Equation (1.3-20) is reduced to Eq.(1.3-22) for CITATION code which does not treat in-group scattering. That is, the absorption term is corrected as shown in Eq.(1.3-23).

$$-\nabla D_g \nabla \varphi_g + \left\{ \Sigma_{a,g} + \sum_{g' \neq g} \Sigma_{s,g->g'} + \sum_{g' \neq g} \Sigma_{2n,g->g'} \right\} \varphi_g = \frac{\chi_g}{k_{eff}} \left\{ \sum_{g'} \nu \Sigma_{f,g'} \varphi_{g'} \right\} + \sum_{g' \neq g} \Sigma_{s,g'->g} \varphi_{g'} + \sum_{g' \neq g} 2\Sigma_{2n,g'->g} \varphi_{g'} \tag{1.3-20}$$

$$\Sigma'_{s,g->g'} = \Sigma_{s,g->g'} + 2\Sigma_{2n,g->g'} \tag{1.3-21}$$

$$-\nabla D_g \nabla \varphi_g + \left\{ \Sigma'_{a,g} + \sum_{g' \neq g} \Sigma'_{s,g->g'} \right\} \varphi_g = \frac{\chi_g}{k_{eff}} \left\{ \sum_{g'} \nu \Sigma_{f,g'} \varphi_{g'} \right\} + \sum_{g' \neq g} \Sigma'_{s,g'->g} \varphi_{g'} \tag{1.3-22}$$

$$\Sigma'_{a,g} = \Sigma_{a,g} - \sum_{g'} \Sigma_{2n,g->g'} \tag{1.3-23}$$

(5) Conversion ratio and its extension

The COREBN does interpolation process to the cross-sections of pair nuclides required for the evaluation of instantaneous conversion ratio as well as macroscopic cross-sections for diffusion calculation. These are absorption cross-sections of fissile nuclides ($\Sigma_{a,g}^{fiss}$) and capture ones of fertile nuclides ($\Sigma_{c,g}^{fert}$), and they are used to evaluate reaction rate of each fuel element defined by Eqs. (1.3-24) and (1.3-25) and local conversion ratio defined by Eq.(1.3-26) at every step. The conversion ratio of the whole core is evaluated by Eq.(1.3-27).

$$R_a^{fiss}(i) = \int_{Fuel\ Element} \sum_g \Sigma_{a,g}^{fiss}(\vec{r}) \varphi_g(\vec{r}) dV \tag{1.3-24}$$

$$R_c^{fert}(i) = \int_{Fuel\ Element} \sum_g \Sigma_{c,g}^{fert}(\vec{r}) \varphi_g(\vec{r}) dV \tag{1.3-25}$$

$$CR(i) = R_c^{fert}(i) / R_a^{fiss}(i) \tag{1.3-26}$$

$$CR = \int_{Core} \sum_g \Sigma_{c,g}^{fert}(\vec{r}) \varphi_g(\vec{r}) dV \Big/ \int_{Core} \sum_g \Sigma_{a,g}^{fiss}(\vec{r}) \varphi_g(\vec{r}) dV \tag{1.3-27}$$

By using the option to redefine the conversion ratio (cf. IBC5 in Sect.2.11 of the SRAC manual[4]), the contents of $\Sigma_{a,g}^{fiss}$, $\Sigma_{c,g}^{fert}$ can be replaced to obtain at every step the ratio of arbitrary pair reaction ratio (macroscopic or microscopic) specified by the user.

(6) Inventory

The COREBN has a function to edit the inventory of the specified heavy nuclide (j) by fuel element (i) and of the whole core at the final step, and also the total inventory of fissile nuclides by fuel element and of the whole core. For the fissile nuclides, the inventory ratio of the final and initial is edited by element and for the whole core. They are defined as follows;

$$Inventory(j,i) = N_j(E_X) M_j V_i / N_A \tag{1.3-28}$$

$$Inventory(j) = \sum_{i \in Core} Inventory(j,i) \tag{1.3-29}$$

$$Fissile\ Inventory\ Ratio(i) = \sum_{j \in fissile} Inventory(j,i) / \sum_{j \in fissile} Initial\ Inventory(j,i) \tag{1.3-30}$$

$$Fissile\ Inventory\ Ratio = \sum_{i \in Core} \sum_{j \in fissile} Inventory(j,i) / \sum_{i \in Core} \sum_{j \in fissile} Initial\ Inventory(j,i) \tag{1.3-31}$$

where,

j : Specified heavy nuclide (available for general depleting nuclide)

fissile : Fissile nuclide fixed by code (U-233, U-235, Pu-239, Pu-241)

$N_j(E_X)$: Atomic number density of the nuclide j (n/cm^3) at the final exposure (E_X)

M_j : Atomic mass of the nuclide j

N_A : Avogadro number

Initial Inventory : Inventory at the beginning of calculation

(7) One-point kinetics parameters

The macroscopic cross-sections defined by Eqs.(1.3-32), (1.3-33), (1.3-34) are tabulated with respect to burn-up, fuel temperature and moderator temperature as well as other macroscopic cross-sections necessary for the diffusion calculation.

$$\Sigma_{i,g}^{BVF} \equiv \sum_m \left(\beta_m^i \nu_{g,m} \sigma_{f,g,m} N_m \right) \tag{1.3-32}$$

$$\Sigma_{i,g}^{BVL} \equiv \sum_m \left(\beta_m^i \nu_{g,m} \sigma_{f,g,m} N_m / \lambda_m^i \right) \tag{1.3-33}$$

$$\Sigma_{\upsilon,g} \equiv \sqrt{\frac{1}{\overline{E}_g}} \approx \frac{const.}{\upsilon_g} \tag{1.3-34}$$

Space dependent values for the above cross sections are obtained by table-look-up method of macroscopic cross-section tables. Thus, one-point kinetics paramnters β_{eff}, λ_{eff} and l are calculated by COREBN as follows:

- Effective delayed neutron fraction

$$\beta_{eff} = \sum_i \beta_{eff,i} = \sum_i \left[\frac{\int \sum_g \phi_g^*(\vec{r}) \chi_{g,i}^d(\vec{r}) \sum_{g'} \Sigma_{i,g'}^{BVF}(\vec{r}) \phi_{g'}(\vec{r}) dV}{\int \sum_g \phi_g^*(\vec{r}) \chi_g^t(\vec{r}) \sum_{g'} \nu\Sigma_{f,g'}(\vec{r}) \phi_{g'}(\vec{r}) dV} \right] \tag{1.3-35}$$

- Effective decay constant for the i-th family of delayed neutron precursor.

$$\frac{1}{\lambda_{eff,i}} = \frac{\int \sum_g \phi_g^*(\vec{r}) \chi_{g,i}^d(\vec{r}) \sum_{g'} \Sigma_{i,g'}^{BVL}(\vec{r}) \phi_{g'}(\vec{r}) dV}{\int \sum_g \phi_g^*(\vec{r}) \chi_{g,i}^d(\vec{r}) \sum_{g'} \Sigma_{i,g'}^{BVF}(\vec{r}) \phi_{g'}(\vec{r}) dV} \tag{1.3-36}$$

- Effective delayed neutron life time

$$l = \frac{\int \sum_g \phi_g^*(\vec{r}) \frac{1}{\upsilon_g(\vec{r})} \phi(\vec{r}) dV}{\int \sum_g \phi_g^*(\vec{r}) \frac{\chi_g^t(\vec{r})}{k_{eff}} \sum_{g'} \nu\Sigma_{f,g'}(\vec{r}) \phi_{g'}(\vec{r}) dV} = k_{eff}\Lambda \tag{1.3-37}$$

1.4 HIST Code

(1) Definition of terms

This section starts with the definition of terms used in COREBN and HIST codes.

Material

A material has its proper cross-section table supplied by SRAC. It may be depleting or non-depleting. As for depleting materials, irrespective of parameters for interpolation such as burn-up step, temperature etc., the first two and seventh characters of the member name for the cross-sections identify the material

For example, light water reflector or fuel elements of a fixed enrichment with the same geometry consist of a material.

Fuel element

It corresponds to an individual fuel element of research reactor or a fuel assembly of LWR. It has to contain depleting material and may partly include non-depleting material such as gas plenum and nozzle components. The depleting material may be non-fissionable of which cross-sections are interpolated by integrated absorption rate.

Depending on geometry or usage of the code, fuel element does not correspond to fuel assembly. For example, as one- or two-dimensional calculation can not apply the concept of node division the following, the axial division of an assembly may be assumed as a composition of different fuel element (cf. Sect.7.1)

Node

It is an axial region of fuel element or non-fuel element divided by a plane to which cross-sections are allocated in three-dimensional calculation.

The node can be applied only in axial (z-direction) division in three-dimensional calculation by CITATION routine. In one- or two-dimensional calculation, number of axial nodes is treated as one, update of burn-up is made for a fuel element. Even in R-Z calculation, as R-direction is treated as x-direction (column) and Z as y (row) in CITATION, division in Z-direction is not taken as node but as fuel element.

Fuel element type

It is a group of fuel elements which can be assumed as identical at the fresh (zero burn-up) condition. An individual fuel element belongs to any of fuel element type. The same cross-section table is allocated to the fuel elements that belong to the same fuel element type.

Non-fuel element

It corresponds to an in-core structure composed by non-depleting material(s) such as reflector, black absorber, control rods, etc. Non-fuel element does not share the space with fuel elements. The cross-section interpolation is not performed to non-fuel element. In three-dimensional calculation, a non-fuel element is divided into axial nodes, and some non-depleting material is allocated to each node. The node boundary has to be identical with region boundary by the "plane" of CITATION.

Non-fuel element type

It is a group of elements that can be identical. As the characteristics of non-fuel element does not depends on burn-up history, an individual non-fuel element is not classified into non-fuel element type in the treatment of HIST code. In COREBN, only the control rod element is classified.

Control element

Among non-fuel elements, it is specially called as control element which is used as control rod. When several control elements of the same type are disposed in a core, this group is registered as a non-fuel element type. Each individual element may be driven independently.
To drive a control element axially in a three-dimensional calculation, the node boundaries of the control element have to coincide with Region boundaries (Plane by the term of CITATION) in the input specification.
A certain research reactor adopts control element with follower fuel. When fuel follower and absorber part are treated as an element, this belongs to fuel element.

Background material

To allocate cross-sections to regions in a core by use of OVERLAY function in CITATION routine, a background material is once allocated throughout the core. Then, particular elements are allocated in proper regions. In usual case, reflector or black absorber (with vacuum outer boundary condition) is used as background material. When control rod is withdrawn by control rod drive, the background material is allocated in the region withdrawn. If undefined allocation occurs by miss-loading, the region there, is filled by the background material.

Batch

The COREBN defines batch as a run beginning with the first core burn-up calculation until completing the calculation. In the history file described later, operation record such as fuel allocation etc. is written by batch.

Cycle

In general, an operation starting with a fuel pattern until shutdown for the next reloading is called as cycle. The operation cycle number conventionally used in irradiation reactors does not always correspond to the above definition. In COREBN input and on the history file, cycle number has merely a role of comment. It is assumed that one cycle includes several batches.

(2) Function of HIST code

The COREBN requires as input principally, the information related to operation condition such as thermal power, operation period, loading pattern of fuel elements and control elements etc. The information such as the geometry of the core and the fuel elements and non-fuel elements which are loaded into the core are registered in the history file edited by HIST code. The COREBN reads necessary information from the history file and writes at the end of calculation the updated burn-up and composition by element into the history file.

The HIST code is the code mainly to control the history file. It has the following functions. The concrete contents of the history file will be described in Sect.4.1.

(a) Creation and update of the history file

- Registration and update of reactor geometry and materials
- Registration and update of fuel element type
- Registration and update of non-fuel element type
- Registration, update and deletion of individual fuel element
- Deletion of operation record

(b) Print output of the history file

- Print output of reactor geometry and registered depleting nuclides
- Print output of operation record (operation condition)
- Print output of information on fuel element type
- Print output of information on non-fuel element type
- Print output of information on individual fuel element
- List of operation record
- List of fuel elements
- Print output of number densities of depleting nuclides table versus burn-up (content of member `caseDNxT`)
- List of burn-up of fuel elements loaded in core
- list of axial structure of fuel element types and non-fuel element types

(c) Conversion of MACRO file (PDS) into PS file

- The COREBN reads in cross-sections not from MACRO file of PDS format but from sequential (PS) file of which contents are converted from MACRO file. When MACRO file is created or updated, the contents of MACRO file have to be converted into PS file.

(3) Rules of naming of members on MACRO file (PDS)

When the HIST code converts the contents of MACRO file into PS file, the naming of members on MACRO file has to meet the rules defined by COREBN. It is because the member name indicates burn-up point and temperature point.

The member name on MACRO file consists of eight characters. Among the members, the member related to burn-up has to follow the rules below, and that for non-depleting material may has arbitrary character except fifth and eighth character.

Position	Meaning
1,2	Material identification, arbitrary characters
3	One of eleven alphanumerical characters to denote fuel temperature (1,2,3,.....9,A,B) Each of these characters defaultedly correspond to fuel temperatures for interpolation as shown below (user's setting is available by input specification)

1	2	3	4	5	6	7	8	9	A	B
300	325	350	400	450	500	550	600	900	1200	1600(K)

4	One of eleven alphanumeric characters to denote moderator temperature (1,2,3, ...9, A,B) Each of these characters defaultedly correspond to moderator temperatures for interpolation as same as fuel temperature. Uses's setting is available by input specification.
5	Energy range indicator. 'A' (all) denotes whole energy range, 'F' fast. As COREBN always works on whole energy range, this indicator is not significant. Only when thermal energy calculation is skipped (IC4=0: Sect.2.2 in the SRAC manual[4]), 'F' appears in this indicator.
6	One of 62 alphanumeric characters to denote burn-up step number among (0, 1, 2, 3,....., 9, A, B,....Z, a, b, c,……,z). This indicator is written after cell burn-up calculation. The corresponding measures of burn-up are stored in the member `caseDNxT` (cf. Sect.3.1.6 in the SRAC manual[4]).
7	One alphanumeric character to denote homogenized region number among (1, 2, 3, ..., 9, A, B,.....,Z). That is to say, X-Region number in the SRAC calculation.
8	One of three alphanumeric characters to denote the kind of cross-sections: (0 / N / Z), '0' for cross-sections used in diffusion calculation, 'N' for (n,2n) reaction cross-sections used (if any) to correct absorption cross-sections, and Z for delayed neutron data.

Note that the fifth to eighth characters are, as far as SRAC is used, automatically given. As the first to fourth characters are input characters for SRAC, the user specifies the input considering the above rules, it is not necessary to modify the member name on MACRO file. If necessary, a utility code to modify member names of a group of members by one command is available.

There exist a lot of kinds of members on MACRO file to restart or to execute branch-off calculation. These members are not directly used by HIST or COREBN code, but it is recommended to preserve them for reactivity analysis and for extension of maximum burn-up.

1.5 Process for Core Burn-up Calculation

To execute core burn-up calculation, use of SRAC, HIST and COREBN codes is required. The relation among these codes and I/O files is shown in Fig.1.5-1.

The process for core burn-up calculation is explained by using an example to execute it from the initial loading until the end of second cycle where it is assumed, for simplicity, that critical adjustment by control absorber is not done.

(1) Cell burn-up calculation by SRAC

The cross-sections of fuel element type and of non-fuel element type are provided by SRAC. The cell burn-up calculation is required specially for depleting material. If spatial distribution of fuel and moderator temperature is considered, the cell burn-up calculations are necessary at the temperatures to cover their variation. Here, the temperature (historical temperature) is assumed as constant during burn-up period. Therefore, the tabulation on temperature is not to follow the instantaneous change of temperature distribution.

(2) Creation of history file (HIST)

By using HIST code, the history file is created and registration work is made. The work necessary at least is as follows;

- Geometry of core and registration of materials (initialization)
- Registration of fuel element type
- Registration of non-fuel element type
- Registration of fuel elements loaded in the initial core

When the calculation of a partial core is done by using rotational symmetry or reflective boundary condition, if necessary, the fuel element which is not included in the partial core is registered. It is because the reloading may not be realized by the fuel elements in the partial core in the multiple batch refueling (cf. (6) of Sect. 3.3).

As all the fuel elements in the initial core are fresh, give zero value of burn-up of nodes in

the registration.

[OUTPUT file : HIST1B.DATA]

(3) Conversion of MACRO file from PDS organization to PS organization (HIST)

MACRO file provided by SRAC is converted from PDS organization into PS organization for COREBN.

[READ_IN file : MACRO PDS file]
[OUTPUT file : MACRO PS file]

(4) Core burn-up calculation for the first cycle core (COREBN)

By using the history file created by HIST code for the initial core of the first cycle (BO1C), the core burn-up calculation is executed by COREBN until the excess reactivity reaches the target value (ex. =0). At the end of calculation (end of the first cycle: EO1C), the COREBN writes the operation condition of the first cycle, burn-up and composition of depleting nuclides by individual fuel element in the core into the history file.

[READ_IN file : HIST1B.DATA]
[OUTPUT file : HIST1E.DATA]

(5) Registration of fuel elements loaded in the second cycle core (HIST)

The fuel elements newly loaded in the second cycle core are added into the history file which was edited at the end of the first cycle. If a new fuel element type or a material is used, it must be also added into the file.

[READ_IN file : HIST1E.DATA]
[OUTPUT file : HIST2B.DATA]

(6) Burn-up calculation of the second cycle core (COREBN)

By using the file edited by the step (5), the burn-up calculation of the second cycle is executed as same as the step (4).

[READ_IN file : HIST2B.DATA]
[OUTPUT file : HIST2E.DATA]

The process of the core burn-up calculation is completed by the step (6). The final output history file HIST2E.DATA keeps the operation record from the beginning of the first cycle until the end of the second cycle, burn-up and composition of every fuel element including those unloaded at the end of the first cycle. As the information on operation record and fuel elements is accumulated as cycle number increases unless it is deleted, care should be taken to excessive quantity.

By using the history file edited at an end of cycle, reactivity calculation is available. An example to evaluate the full core void reactivity at the end of the second cycle is shown below.

(7) Branch-off cell burn-up calculation (SRAC)

By using the branch-off option of the cell burn-up calculation with SRAC, the cross-section tables along the burn-up at several void fractions are composed assuming instantaneous change of void fraction.

[OUTPUT file : MACRO PDS file]

(8) Replacement of material names and file conversion from PDS into PS (HIST)

By using HIST code, the material names registered in the history file are replaced by those with voided condition. Finally, the MACRO file with voided condition is converted into PS file.

[READ_IN file : HIST2E.DATA]

[OUTPUT file : HIST2EV.DATA, MACRO PS file]

(9) Core calculation for the voided core

By using the history file and the cross-section table obtained by the step (8), a zero burn-up calculation (one step with zero exposure period) is executed at a void fraction to yield the multiplication factor. If this zero burn-up calculation is executed on multi-steps with changing void fraction, one job can yield multiplication factors at several void fractions. If the option to correct equivalent xenon concentration is applied in the nominal condition, this option should be applied to the voided core.

Apart from the above process to replace the cross-section table, if an interpolation parameter is left as unused, this parameter can be used to indicate void fraction instead of temperature. By utilizing interpolation function, the void reactivity at the void fraction lying between tabulated void fraction and also the reactivity caused by local voidage can be evaluated.

The Doppler reactivity, reactivity of soluble boron, and control rod worth at the point of time to which the history file is preserved can be evaluated as well as void reactivity. However, the cost of computation for cell calculation increases by proportional to the product of depleting materials, number of burn-up steps, number of fuel temperature points, number of moderator temperature points, and kinds of reactivity calculation.

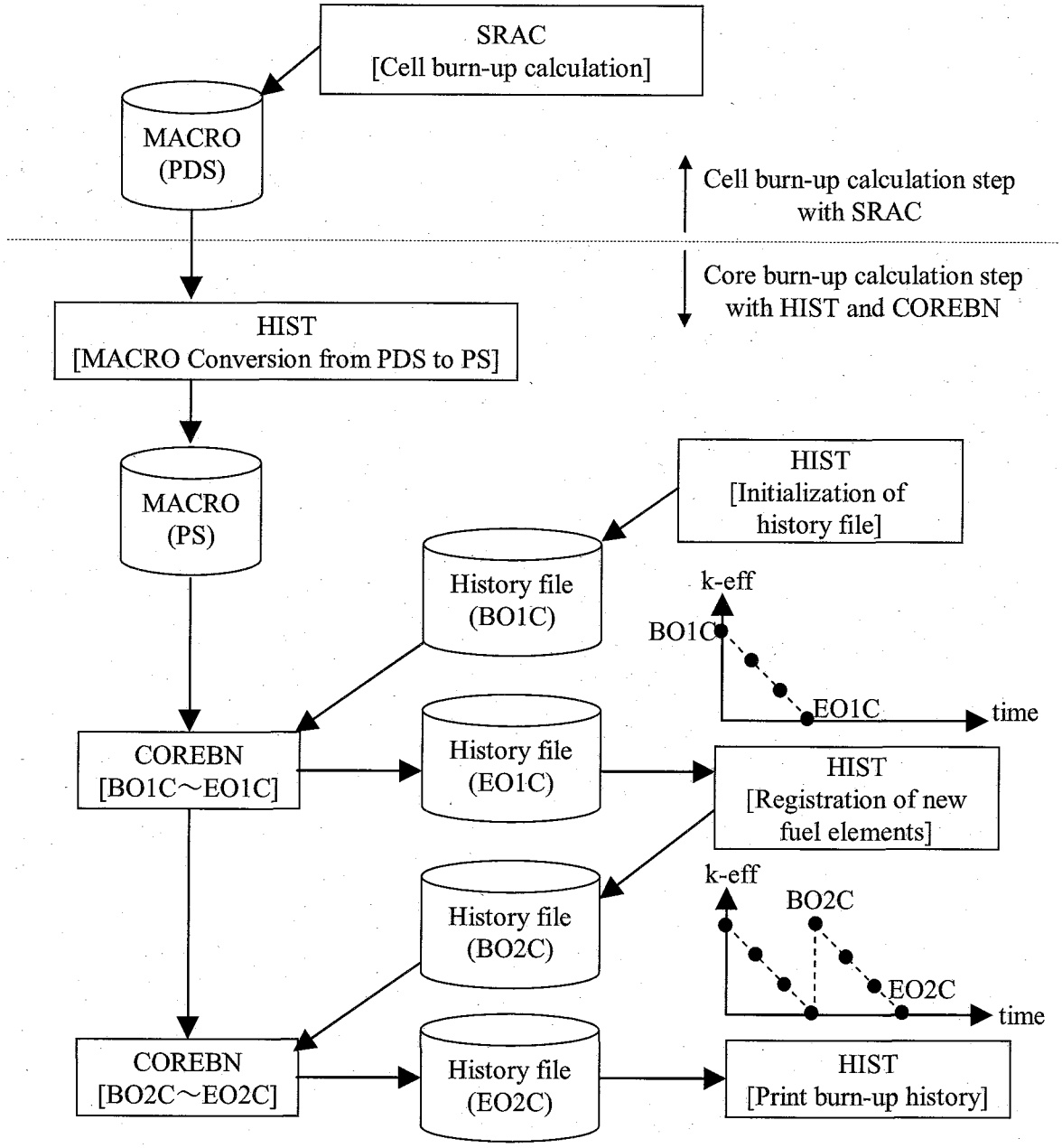


Fig.1.5-1 Example of burn-up calculation process for the core from beginning of the 1st cycle (BO1C) to end of the 2nd cycle (EO2C)

2. Improvements of COREBN and HIST Codes

In the history of development of the SRAC system, a major revision for function enhancement was done in 1995. [3] This revision led to format incompatibility between new and old I/O files including standard input data. After 1995, minor revisions have been made continuously to care for troubles pointed out by SRAC users, however, apparent compatibility of I/O files have been held. In this Chapter, we describe 21 items of improvements given to the COREBN and HIST in the 1995 revision (COREBN95, HIST95) [3]. The same features as the COREBN95 and HIST95 have been passed to the COREBN and HIST codes released after 1995. Therefore, in this chapter, "old" COREBN and HIST codes means the codes before 1995, and COREBN95 and HIST95 represents all newer codes released after 1995.

(1) Acceptance of cross-section tables of different number of burn-up steps

On the old version[2] of COREBN and HIST before 1995, number of burn-up steps had to be identical among cross-section tables of all depleting material. That is, when both of fuel material containing burnable poison and fuel material not containing poison are included in the core, the cell calculations by SRAC to prepare cross-section tables had to be executed on the same burn-up steps. If a local burn-up exceeds the maximum burn-up on the tabulated one, additional cell calculations to cover the local burn-up were required for not only the corresponding material but also other depleting materials to adjust the burn-up steps. The COREBN95 and HIST95 allow the different burn-up steps by material to overcome such inconvenience.

(2) Acceptance of different burn-up chain libraries by material

The old version of COREBN required the same kinds and order of depleting nuclides among the materials to interpolate number densities of depleting nuclides with respect to burn-up. Therefore, the cell burn-up calculation by SRAC had to use a common burn-up chain model for all the depleting materials. The COREBN interpolates the number densities with regard to burn-up only of the specified nuclides. It is not necessary to use the common burn-up chain model (library).

(3) Acceptance of cell burn-up results of plural depleting materials in a cell

The COREBN assumes one set of cross-section table to a homogenized material. The old version of SRAC outputs the member `caseNDEN` for composition table even a cell contains plural depleting materials which was not accepted by the COREBN.

The SRAC95[3] (and unpublished newer SRAC after 1995) divides the output into the member `caseBNUP` for composition by depleting material and the member `caseDNxT` (x: X-Region number) for averaged composition in the *x*-th Region. Therefore, the member `caseDNxT` keeps only one cross-section table. The COREBN95 utilizes the `caseDNxT` provided by SRAC95, instead of

caseNDEN for composition table. Thus, the result of cell burn-up calculation where plural depleting materials are included, can be accepted by COREBN.

(4) Acceptance of cell burn-up result of plural X-Regions

When the old SRAC executes a cell burn-up calculation with plural X-Regions, a part of homogenized cross-sections were nor correctly accepted by COREBN. It is because these data were not stored in members separated by X-Regions but in caseNDEN).

The SRAC95 store these data in each separate member caseDNxT by X-Region (x: X-Region number), and the COREBN obtains the information necessary from the member caseDNxT which keeps the averaged cross-sections of the x-th X-Region.

(5) Extension of HIST code

A few new functions are added to the HIST95 code. First, registration of fuel elements of the same burn-up and composition is simplified.

An option to modify burn-up by adjusting the final burn-up period written into the history file after the core burn-up calculation, and another option to modify burn-up by specifying the fuel element name with masking character (meta-character) are newly added.

Added is an option to modify the concentrations of I-135, Xe-135, Pm-149 and Sm-149 in the specified fuel elements due to reactor cooling by pursuing their chain decay. Especially for Xe-135 concentration, its cooling effect can be reflected in the core burn-up calculation by using the newly extended function to correct equilibrium xenon concentration in COREBN. This option may be used for evaluate the xenon build up reactivity after reactor shut down.

(6) Addition of one-dimensional core burn-up calculation

The old version of COREBN could treat two- or three-dimensional core. One-dimensional core (slab, sphere and cylinder) become available in COREBN95. Thus, all the geometries accepted in CITATION routine are available in COREBN95.

(7) Addition of cross-section interpolation by integrated absorption reaction rate

The COREBN form the cross-sections of depleting material by interpolation in regard to burn-up (MWd/cm^3). Among irradiated specimen, there is a material such as Gd_2O_3 which can not be treated by burn-up interpolation. To treating such a material, the COREBN95 interpolates cross-sections in respect to integrated absorption rate (absorption/cm^3) (cf. Eqs. (1.3-9) to (1.3-10)).

(8) Extension of temperature interpolation

The COREBN interpolates cross-sections in regard to fuel temperature and moderator temperature as well as burn-up. The old version of SRAC made the output cross-sections only on any of the 11 fixed tabulated temperature points lying from 300K to 1600K. The old COREBN made

interpolation assuming that tabulated cross-sections are on any of these fixed temperatures expressed by a simple indicator. As SRAC95 is able to run on arbitrary temperature, COREBN95 has an option to accept the temperature points for tabulation specified by the user.

Because the parameter values for interpolation can be flexibly specified, the fuel temperature and moderator temperature as parameters may be substituted by arbitrary quantities such as moderator void fraction, boron density and control rod insertion fraction. Thus, cross-section interpolation can be widely applied to various reactivity calculation and new treatment of control rod.

(9) Modification and extension of equilibrium boron concentration correction

In the old COREBN, the correction of equilibrium xenon concentration depending on the neutron flux level was performed by considering the difference of the average flux supposed in the cell burn-up calculation by SRAC and the flux obtained in the core burn-up calculation. As the correction was made actually to the absorption cross-section of the lowest group, when thermal range was treated by multi-groups, exact correction was not made.

The COREBN modified the way to correct the equilibrium xenon concentration into a general method based on Eq.(1.3-14) and made correction to the cross-sections in all the groups in thermal range.

The function to correct equilibrium xenon concentration is extended to include several new functions such as the burn-up calculation starting after long reactor shut-down (zero xenon concentration), burn-up calculation starting with specified xenon concentration by (cf. Eq.(1.3-18)) and the reactivity caused by xenon concentration change (cf. Eq.(1.3-19)).

(10) Allowance to replace the cross-section table with that of different group structure

The history file used by the old COREBN and HIST codes stored the information depending on the energy group structure (neutron flux necessary for equilibrium xenon concentration correction). Thus the replacement of cross-section table by that of different group structure was not allowed.

The history file used by COREBN95 and HIST95 does not include the information depending on the group structure. Therefore, it becomes available to evaluate the kinetics parameters and reactivity coefficients by using the history file provided by a two-group core burn-up calculation, together with the cross-section table expressed by different group structure.

(11) Heat generation by material

The old COREBN used the fixed heat generation per fission (J/fission) (based on U-235 value) for normalizing neutron flux level to the thermal power. The COREBN95 uses the values interpolated by material as the more exact treatment.

(12) Automatic conversion of the unit of burn-up

The COREBN evaluate the burn-up distribution in term of MWd/cm^3. However, the old

SRAC tabulated cross-sections in regard to burn-up in term of MWd. The constant to covert the burn-up in MWd into MWd/cm^3 was required in input of HIST. This constant seemed difficult to evaluate as it depends on the specification of thermal power in the cell burn-up calculation by the old SRAC. The SRAC95 tabulates cross-sections in regard to burn-up in term of MWd/t, and the constant necessary to convert burn-up in term of MWd/cm^3 into MWd/t is stored in the member in PDS file related to burn-up. The COREBN95 reads this constant by material from the PDS file via the history file, then the user is not any longer bothered to feed these constants.

(13) Removal of the limit on number of registered nuclides

The old COREBN and HIST codes gave a restriction on the number and nuclide of heavy nuclides for inventory calculation. That is, they were selected among Th-232, Pa-233, U-234, U-235, U-236, U-238, Pu-239, Pu-240, Pu-241, Pu-242. The all defaulted sets of heavy nuclides in the old HIST were derived from the obsolete chain models.

The HIST95 receive the defaulted set of depleting nuclides and the set of heavy nuclides for inventory calculation from an external file (cf. Sect.4.2), and passes them to COREBN95 via the history file. Thus, the above restriction is removed, and the defaulted set may be changed by the user.

(14) Suppression of print output for mesh map

The old COREBN printed out the mesh map at every step as it called CITATION routine repeatedly by step. The COREBN95 prints out the mesh map only at the initial step, or an option to suppress thoroughly printing is added.

(15) Removal of restrictions caused by fixed dimensioning

As the old COREBN installed the original CITATION as diffusion calculation, where many arrays of fixed dimension were adopted, some large-scale computations were limited. For example, number of every direction (211), number of regions of every direction (200), and total number of zones were limited. The declarations by fixed dimension are replaced by the description with PARAMETER and INCLUDE statements. The contents and defaulted values of PARAMETERs are shown in Sect.5.3 and Sect.6.3.

(16) Edit of effective decay constants of delayed neutron precursors

Whereas the old COREBN edited effective delayed neutron fraction and prompt neutron lifetime, effective decay constants of delayed neutron precursors (λ_{eff}) were not evaluated. The COREBN95 evaluates λ_{eff} in the same way to the SRAC-CITATION.

(17) Correction of power distribution output

The binary output of power distribution into logical unit 32 (NGC7=0) was executed with the specification of print point-wise power (IEDG=1). The COREBN95 accepts the binary output

specification unconditionally.

(18) Correction for vectorized computation

It happened that the old COREBN did not yield correct result when the system included an internal black absorber which applied the logarithmic differential condition to certain thermal groups. This treatment is often used for the control rod in the fuel management of research reactors. The scalar version was used to avoid the trouble. In the vectorized version of COREBN95, the debugging is completed. Also modifications were made to the routines for the geometries of (θ-R, hexagonal mesh X_H-Y_H, triangular mesh X_T-Y_T, θ-R-Z) to correct incomplete vectorization that caused slow conversion. Thus, the vectorized version of COREBN95 can be applied to all geometries, and occasional use of scalar version is no more necessary.

(19) Run on UNIX system

The COREBN95 and HIST95 are available on the machines working with UNIX operating system.

(20) Common use of CITATION routine

The CITATION routine in SRAC and COREBN are based on the same CITATION. They were maintained as different codes. It happened that a modification was not reflected to the other routine, and different vectorizations were applied.

The SRAC95 and COREBN95 codes use commonly the unified CITATION routines that are given flags to indicate in which code they are used. By these flags, they are processed to meet the proper code. By this modification, only one set of the CITATION routines are maintained and their control and modification become simple.

(21) Elevation of portability

The old COREBN and HIST codes used everywhere the built-in functions proper to FACOM machines. They had to be rewritten to meet the other system or machines when they were transferred.

In the revised versions, these system-depending functions are called through each user routine. By this modification, these codes can be easily transferred to another system or machine.

3. Input Requirements of COREBN and HIST

3.1 Input Requirements of HIST

Input data for HIST are, in general, entered by the free format of SRAC (cf. Sect.2.1 of the SRAC manual[4]). However, a part of character type input is entered by the free format proper to HIST where there is no column allocation and strings of specified length separated by a blank or a comma may be entered in a record. Repetition of character string is available by entering "n(abcd)" which corresponds to enter "abcd" n times.

The HIST code has 16 function as shown below. The user has to specify first one of these function by IMOD and then I/O files (NHIS1, NHIS2) used by the specified function.

When a history file is created, the initialization of file (IMOD=1) is required. A new core burn-up calculation needs at least the processes IMODE=1 through 4 and =99.

Block-n-1 (n=0-15 or 99 corresponding to input value for IMOD) /3/

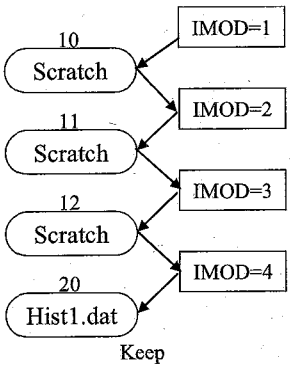
IMOD Selection of function

= 0 End of job
= 1 Initialization of history file (registration or update of core geometry and materials)
= 2 Registration or update of fuel element type
= 3 Registration or update of non-fuel element type
= 4 Registration, update, or deletion of individual fuel element
= 5 Deletion of operation record
= 6 Print out of core geometry and registered depleting nuclides
= 7 Print out of operation record (operation condition of core burn-up calculation)
= 8 Print out of information on fuel element type
= 9 Print out of information on non-fuel element type
= 10 Print out of information on individual fuel element
= 11 List of operation records
= 12 List of fuel elements
= 13 Print out of nuclide number densities tabulated versus burn-up
= 14 List of burn-up by fuel elements loaded in the core
= 15 List of axial structure of fuel element type or non-fuel element type
= 99 Conversion of MACRO file from PDS into PS

NHIS1 Logical number of input history file
Specify two digits except 1, 5, 6, 7, 49, 51, 52, 90, 97, which are used for special purpose. (number from 10 to 19 is recommended). Enter zero value when it is not used like in case of initialization.

NHIS2 Logical number of output history file
Specify two digits except 1, 5, 6, 7, 49, 51, 52, 90, 97 which are used for special purpose and the number used for NHIS1. (20 is recommended for the file to be preserved). Enter zero value when it is not used like in case of print out.

Specify the function and I/O file numbers, then enter the following input required by the function. Repeat these input data and complete by specifying IMOD=0. When plural functions are consecutively used, as shown in the figure below, use the history files in the intermediate steps as scratch and preserve the final history file.



(1) Initialization or update of the history file

(Registration of core geometry and materials)

Block-1-1 /3/

MOD = 1

NHIS1 Logical number of history file for read-in
To update the existing history file, specify the number of the file. To create a new history file, enter zero

NHIS2 Logical number of output history file.

Block-1-2 2*/A72/

HEADER Comment in two lines

Block-1-3 Control data /7/

1 IGEOM Geometry

			X(columns) direction	Y(rows) direction	Z(planes) direction
=	1	1D slab (X)	X	Y	Z
=	2	1D cylinder (R)	R		
=	3	1D sphere (R_S)	Rs		
=	4	not used			
=	5	not used			
=	6	2D slab (X-Y)	X	Y	
=	7	2D cylinder (R-Z)	R	Z	
=	8	2D disk (θ-R)	θ	R	
=	9	2D hexagonal (X_H-Y_H)	X_H	Y_H	
=	10	2D triangular (X_T-Y_T)	X_T	Y_T	
=	11	3D slab (X-Y-Z)	X	Y	Z
=	12	3D cylinder (θ-R-Z)	θ	R	Z
=	13	3D hexagonal (X_H-Y_H-Z)	X_H	Y_H	Z
=	14	3D triangular (X_T-Y_T-Z)	X_T	Y_T	Z

To update the geometry in the existing history file, enter IGEOM with negative value. In this case, input data Block-1-4 to Block-1-9 are required.

2 NREGI Number of Region divisions in X-direction (cf. Fig.3.1-1 through 3.1-6)

3 NREGJ Number of Region divisions in Y-direction (cf. Fig.3.1-2 through 3.1-6)

4 NREGKB Number of Region divisions in Z-direction (cf. Fig.3.1-6)
Enter NREGKB=0 if 1D or 2D (IGEOM≤10)

5 NMAT Number of materials used in core burn-up calculation
To update the information on materials, enter NMAT with negative sign. In this case, only Block-1-13 and Block-1-13' are required.

6 NTNUC Number of depleting nuclides of which number densities are chased in the core burn-up calculation
If NTNUC=0 is specified, the defaulted depleting nuclides specified in the text file allocated to logical unit 52 are used.

The contents of unit 52 may be modified by the user. The current defaulted set is organized by the following 70 nuclides (Heavy nuclides and fission products based on the burn-up chain model u4cm6fp50bp16):

XU04(U-234)	XU05(U-235)	XU06(U-236)	XU07(U-237)	XU08(U-238)
XNP7(Np237)	XNP9(Np239)	XPU8(Pu238)	XPU9(Pu239)	XPU0(Pu240)
XPU1(Pu241)	XPU2(Pu242)	XAM1(Am241)	XAMG(Am242)	XAMM(Am242m)
XAM3(Am243)	XCM2(Cm242)	XCM3(Cm243)	XCM4(Cm244)	XCM5(Cm245)
XCM6(Cm246)	XKR3(Kr83)	XZR5(Zr95)	XNB5(Nb95)	XMO5(Mo95)
XTC9(Tc99)	XRU1(Ru101)	XRU3(Ru103)	XRH3(Rh103)	XRH5(Rh105)
XPD5(Pd105)	XPD7(Pd107)	XPD8(Pd108)	XAG7(Ag107)	XAG9(Ag109)
XI05(I-135)	XXE1(Xe131)	XXE3(Xe133)	XXE5(Xe135)	XCS3(Cs133)
XCS4(Cs134)	XCS5(Cs135)	XCS7(Cs137)	XBA0(Ba140)	XLA0(La140)
XPR3(Pr143)	XND3(Nd143)	XND5(Nd145)	XND7(Nd147)	XND8(Nd148)
XPM7Pm147)	XPMM(Pm148m)	XPMG(Pm148)	XPM9(Pm149)	XSM7(Sm147)
XSM8(Sm148)	XSM9(Sm149)	XSM0(Sm150)	XSM1(Sm151)	XSM2(Sm152)
XEU3(Eu153)	XEU4(Eu154)	XEU5(Eu155)	XEU6(Eu156)	XGD4(Gd154)
XGD5(Gd155)	XGD6(Gd156)	XGD7(Gd157)	XGD8(Gd158)	XGD0(Gd160)

7 NHVNUC Number of heavy nuclides for inventory (gram) evaluation in the core burn-up calculation

If NHVNUC=0 is specified, the defaulted depleting nuclides specified in the text file allocated to logical unit 52 are used.

The contents of unit 52 may be modified by the user. The current defaulted set is organized by the following 9 nuclides;

XU05(U-235)	XU06(U-236)	XU07(U-237)	XU08(U-238)	XPU8(Pu238)
XPU9(Pu239)	XPU0(Pu240)	XPU1(Pu241)	XPU2(Pu242)	

Block-1-4 /NREGI/

NMESHX(*i*) Number of Mesh divisions in X-direction by Region

Enter from left to right

Block-1-5 /NREGI/

XX(*i*) Size (cm) of Region in X-direction

Enter from left to right

Block-1-6 Required for 2D or 3D (IGEOM ≥ 6) /NREGJ/

NMESHY(*j*) Number of Mesh divisions in Y-direction by Region

Enter from top to bottom

Block-1-7 Required for 2D or 3D (IGEOM ≥ 6) /NREGJ/

YY(*j*) Size (cm) of Region in Y-direction

Enter from top to bottom

Block-1-8 Required for 3D (IGEOM ≥ 11) /NREGKB/

NMESHZ(*k*) Number of Mesh divisions in Z-direction by Region

Enter from front to back

Block-1-9 Required for 3D (IGEOM ≥ 11) /NREGKB/

ZZ(k) Size (cm) of Region in Z-direction

Enter from front to back

Block-1-10 Registration of materials |NMAT| */A8,1,0/

MATNM Enter member name by 8 characters of the material stored in MACRO file

Specify non-depleting materials (IFORS=0) before depleting materials. As the third, fourth, and sixth characters for depleting material are Tags for interpolation, the input characters are ineffective. (usually use 'X' as F1XXAX10).

IFORS Material type

= 0 non-depleting material

= ±1 fissionable depleting material (usual fuel)

= ±2 non-fissionable depleting material (interpolation by integrated absorption rate)

= 3 black absorber (blackness)

Specify IFORS=3 for perfect absorbing material in whole energy groups for which cross-sections are not applied (cf. NUAC17 and XMIS2 in Sect.3.2).

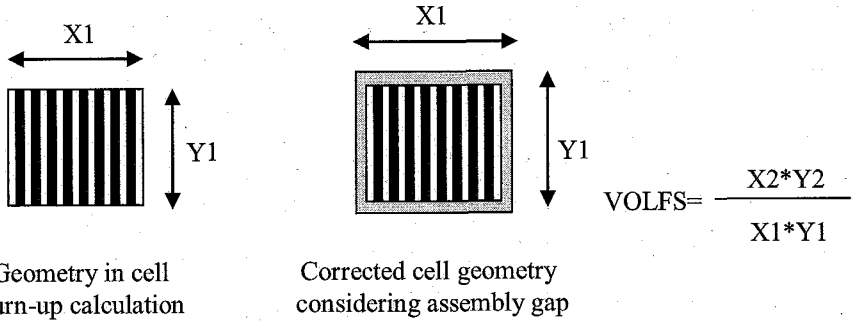
If IFORS<0 is specified, the conversion factor of burn-up unit for interpolation in respect to burn-up may be modified by VOLFS entered in Block-1-10'. As this is a special treatment, use IFORS ≥ 0 usually.

Block-1-10' Required if IFORS<0 /1/

VOLFS Correction factor to the conversion of the burn-up unit (ton/cm^3)

The conversion factor is to convert the tabulated burn-up point given by MWd/ton into that expressed by MWd/cm^3 : the unit in COREBN. This factor is used to correct the factor by multiplying 1/VOLFS. It is applied when the system assumed in the cell burn-up calculation does not match with the homogenized region treated in the core burn-up calculation. For example, as shown in the figure below, when the cell burn-up calculation is achieved by neglecting the gap between fuel element

and the correction to the cross-sections is made separately by a two-dimensional calculation with considering the gap at every burn-up step, this correction factor is applied as VOLFS=(volume of homogenized region including gap/volume of cell burn-up calculation) >1.0.



As this option to correct the factor to convert the burn-up unit is directly related to the interpolation of cross-sections, careful use should be made.

Even though this option is used, the equilibrium xenon concentration stored in composition table (`caseDNxT`) and cross-sections for conversion ratio calculation are not corrected. Therefore, the use of the model in the cell burn-up calculation by SRAC matched with the homogenized region treated in core burn-up calculation is recommended.

Repeat NMAT times input Block-1-10 through Block-1-10'.

Block-1-11 Required if NTNUC ≠ 0 (Block-1-3) /NTNUC/

NISO(*i*) Name of depleting nuclide of which number density is chased in the core burn-up calculation by 4 characters under the naming rule of SRAC. That is, the first character may be any (e.g. 'X'), and the second to the fourth characters indicate nuclide. The nuclide has to be any of those included in the burn-up calculation by SRAC. The input format is the character type free format of HIST. That is, entries are separated by blank or comma.

Sample input: `XU05 XU06 XU07 XU08 XNP7 XNP9 ...`

Block-1-12 Required if NHVUC ≠ 0 (Block-1-3) /NHVNUC/

IHVNUC(*i*) Name of heavy nuclide included in inventory calculation by 4 characters under the naming rule of SRAC. The first character may be 'X', and the second to the fourth characters indicate nuclide. The heavy nuclides are chosen from those entered by Block-1-11. (The nuclide is not necessarily heavy.) The nuclides used in evaluation of fissionable fuel inventory ratio (cf. Eqs.(1.3-30) and (1.3-31)) or depletion

fraction (%) of number density (U-235 is defaulted) have to be registered.
The input format is the character type free format of HIST. That is, entries are separated by blank or comma.

Block-1-13 Required if NMAT<0 |NMAT| */2A8,1,0/

MATMO Old material name by 8 characters (cf. Block-1-10)

MATMN New material name by 8 characters (cf. Block-1-10)

IFORS Material type of new material

= 0 no-depleting material

= ±1 fissionable depleting material (usual fuel)

= ±2 non-fissionable depleting material (interpolated by integrated absorption rate)

= 3 black absorber

Block-1-13' Required if NMAT<0 and IFORS<0 /1/

VOLFS Correction to burn-up unit conversion factor (cf. Block-1-10')

If NMAT<0, repeat Block-1-13 and Block-1-13' |NMAT| times.

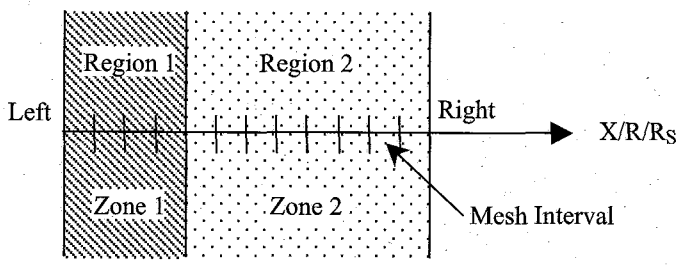


Fig.3.1-1 Sample geometrical mode, type: 1D slab (X), 1D cylinder (R), 1D Sphere(Rs).

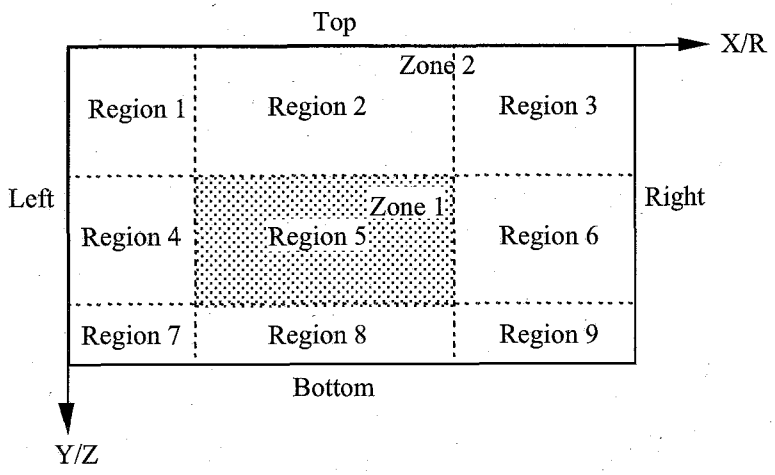


Fig. 3.1-2 Sample geometrical mode, type: 2D slab (X-Y), 2D cylinder (R-Z).

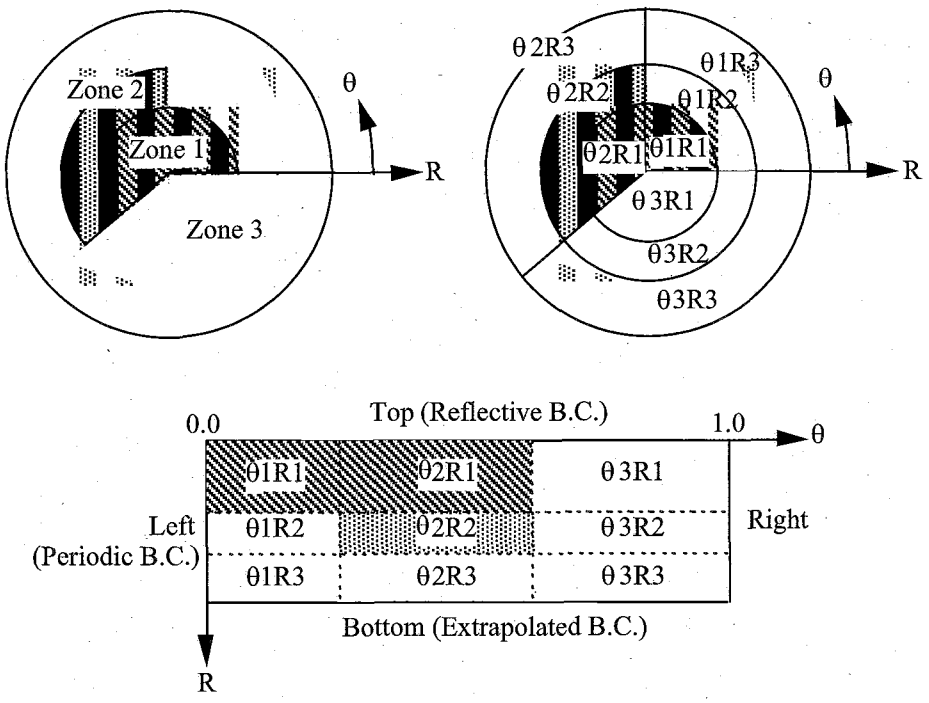


Fig. 3.1-3 Sample geometrical mode, type: 2D circle (θ-R).

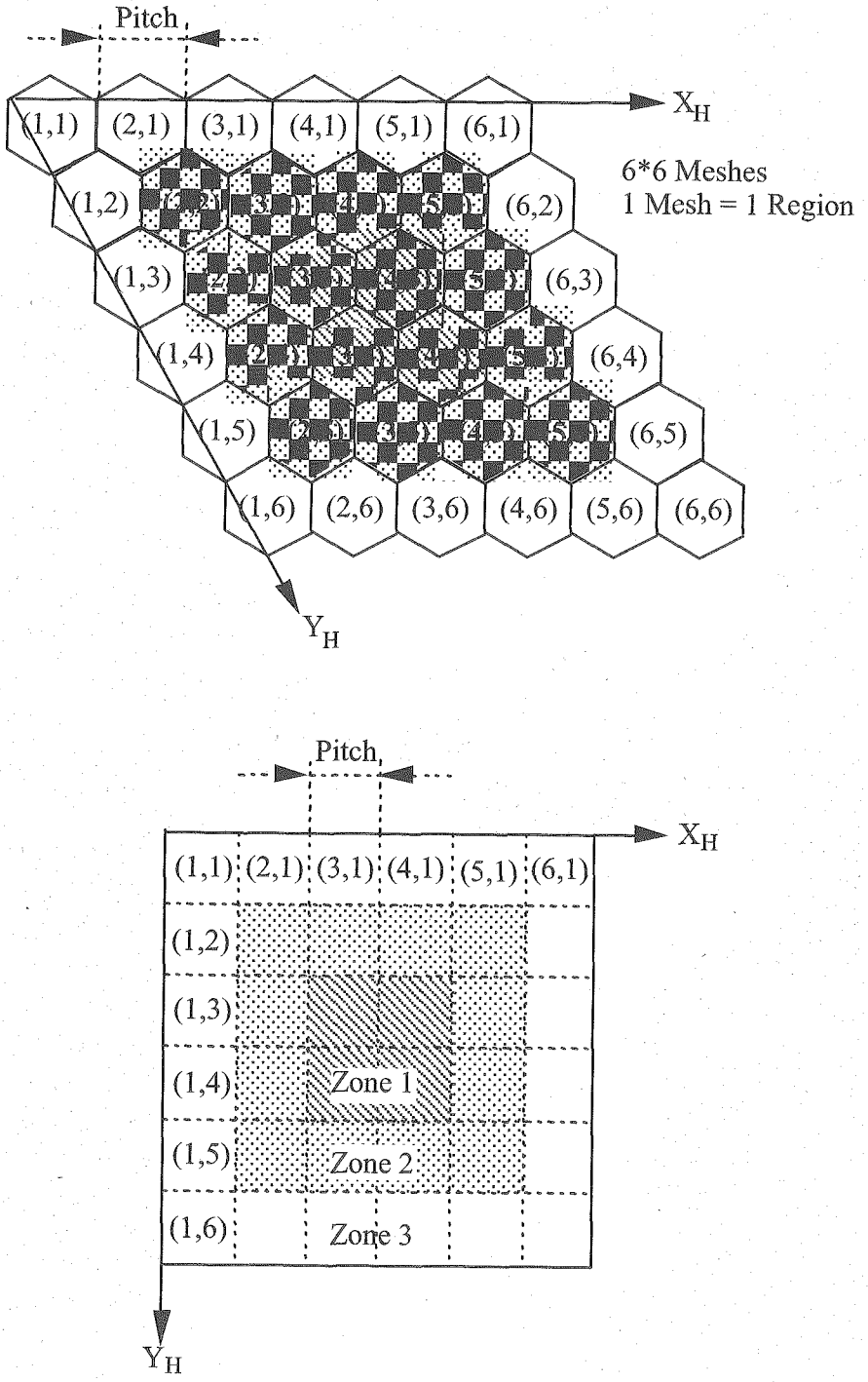


Fig.3.1-4 Sample geometrical mode, type: 2D Hexagonal (X_H-Y_H).

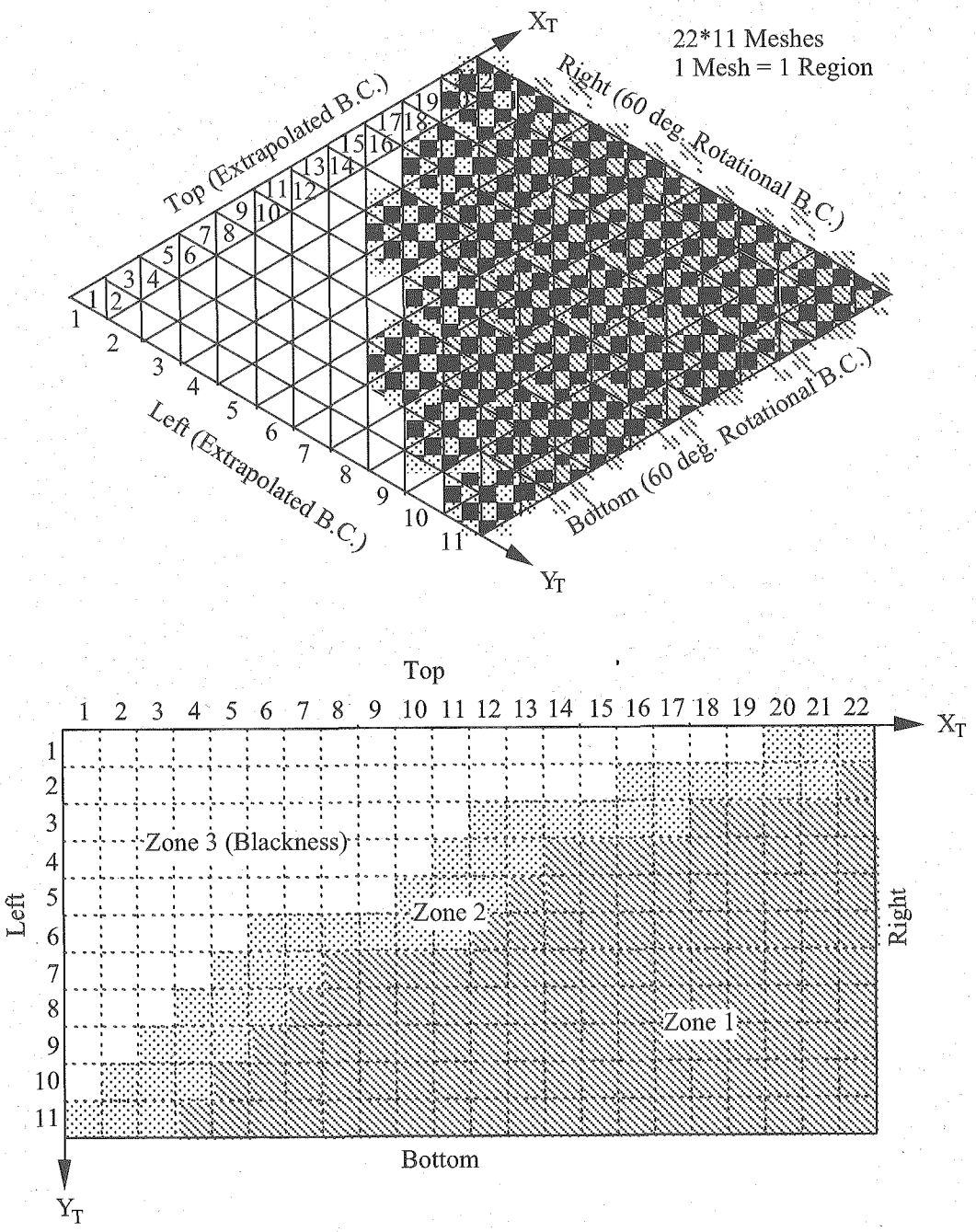


Fig. 3.1-5 Sample geometrical mode, type: 2D Triangular (X_T-Y_T).

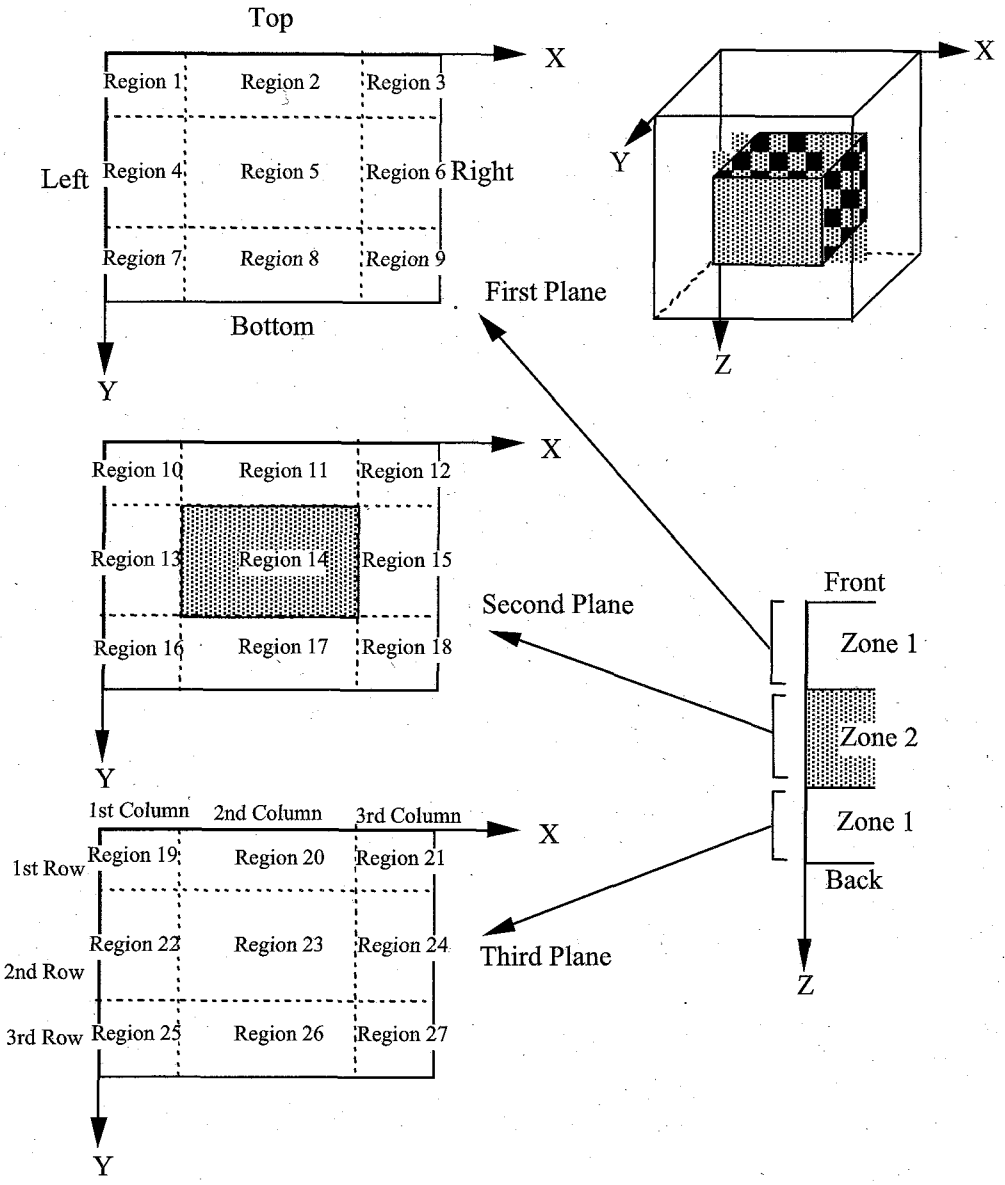


Fig.3.1-6 Sample geometrical mode, type: 3D slab (X-Y-Z).

(2) Registration or update of fuel element type

Block-2-1 /3/

IMOD = 2

NHIS1 Logical number of input history file

NHIS2 Logical umber of output history file

Block-2-2 /1/

IOPT Selection of option

= 1 new or additional registration of fuel element type (entry by Block-2-3)

= 2 update of registered fuel element type (entry by Block-2-4)

= 0 end of entries of Block-2

The following Block-2-3-1 through Block-2-3-4 are required if IOPT=1 (Block-2-2)

Block-2-3-1 /A8,2,0/

FTCOM1 Name of fuel element type by 8 characters

NRKF Number of total node divisions in Z-direction

For one- or two-dimensional calculation (IGEOM ≤ 10), enter NRKF=1

NRK1 Number of depleting nodes among NRKF

For one- or two-dimensional calculation (IGEOM ≤ 10), enter NRK1=1

Block-2-3-2 /NRK1/

VOLF1(k) Volumes (cm^3) of depleting nodes from front to back direction.

VOLF1(k) is used to calculate heavy nuclide inventory and depleting fraction of U-235 number density (no direct relation to burn-up calculation).

The old HIST code required the volume of fuel meat part, but the revised does the volume of node as this treat homogenized composition.

Block-2-3-3 /NHVNUC*NRK1/

(TZINV1(i, k), i=1,NHVNUC), k=1,NRK1)

Inventory (gram) of the heavy nuclide i in the burn-up node k.

The order of entry follows the order of registration of heavy nuclides in Block-1-12.

When NHVNUC=0 is entered in Block-1-12, the order of registration specified in logical number 52 (cf. Sect. 4.2) should be kept.

When number densities of depleting nuclides in every fuel element are entered by Block-4-3-3, all the entries in this item may be given as zero values as the code converts internally.
Contrary, entries of this item are correctly given, number densities of heavy nuclides belonging to this type of fuel elements are converted internally from these entries.
While the numerical values of this entries do not affect the calculation of eigenvalue nor power distribution, these are used to evaluate fissile fuel inventory and depleting fraction of U-235 number density.

Block-2-3-4 Material allocation by node /NRKF/

MATSP(k) Allocation of material by specifying material names (cf. Block-1-10) to axial node by 8 characters from front to back direction. The input format is the character type free format. Material names are separated by comma or blank where the repetition symbol ( ) is available.

After the entries of Block-2-3-1 through Block-2-3-4, enter Block-2-2 for the next option.

The following Block-2-4-1 through Block-2-4-4 are required if IOPT=2 (Block-2-2) is specified.

Block-2-4-1 /2A8,2,0/

FTCOMN Name of new fuel element type by 8 characters.
If name is not changed, enter 8 blanks.

FTCOMO Name of old fuel element type by 8 characters

NRKF Number of total node divisions in Z-direction
If not changed, enter NRKF=0

NRK1 Number of depleting nodes among NRKF
If not changed, enter NRK1=0

Block-2-4-2 /NRK1/

VOLF1(k) Volumes (cm^3) of depleting nodes from front to back direction.
If not changed at k, enter VOLF1(k)=0.
If any of VOLF1(k) is changed, the related quantities such as heavy nuclide inventory (g/node), burn-up (MWd/node), nuclide number densities (n/cm^3) are automatically modified by volume average.

Block-2-4-3 /NHVNUC*NRK1/

(TZINV1(i, k), i=1,NHVNUC), k=1,NRK1)

Inventory (gram) of the heavy nuclide i in the burn-up node k.

If not change at k for nuclide i, enter TZINV1(i, k)=0.

Block-2-4-4 Material allocation by node /NRKF/

MATSP(k) Allocation of material by specifying material names (cf. Block-1-10) to axial node by 8 characters from front to back direction. The input format is the character type free format.

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

Block-3-1 /3/

IMOD = 3

NHIS1 Logical unit of input history file

NHIS2 Logical unit of output history file

Block-3-2 /1/

IOPT Selection of option

= 1 new or additional registration of non-fuel element type (entries by Block-3-3)

= 2 update of registered non-fuel element type (entries by Block-3-4)

= 0 end of Block-3

The following Block-3-3-1 and Block-3-3-2 are required if IOPT=1 (Block-3-2) is specified

Block-3-3-1 /A8,1,0/

OTCOM1 Name of non-fuel element type by 8 characters

A non-fuel element type may be placed at several positions in the core. However, the nodes of non-fuel element type placed at several positions are assumed to belong the same Zone by COREBN. In editing neutron flux by Zone, if a non-fuel element of different positions is needed to edit separately, register it by another name.

A material region used for background has not to be registered.

NRKO Number of axial node divisions

For one- or two-dimensional calculation (IGEOM ≤ 10), enter NRKO=1

Block-3-3-2 Material allocation by node /NRKO/

MATSP(k) Allocation of material by specifying material names (cf. Block-1-10) to axial node by 8 characters from front to back direction. The input format is the character type free format. Material names are separated by comma or blank where the repetition symbol ( ) is available.

After entering Block-3-3-1 and Block-3-3-2, enter again Block-3-2 for the next option.

The following Block-3-4-1 and Block-3-4-2 are required if IOPT=2 (Block-3-2) is specified.

Block-3-4-1 /2A8,1,0/

OTCOMN New name of non-fuel element type by 8 characters
If name is not changed, enter 8 blanks.

OTCOMO Old name of non-fuel element type by 8 characters

NRKO Number of axial node divisions
If not changed, enter NRKO=0

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

MATSP(k) Allocation of material by specifying material names (cf. Block-1-10) to axial node by 8 characters from front to back direction. The input format is the character type free format. Material names are separated by comma or blank.
To the node of which allocation is not changed, enter 8 blanks.

After entering Block-3-4-1 and Block-3-4-2, enter again Block-3-2 for the next option.

(4) Registration, deletion, and update of individual fuel element

Block-4-1 /3/

IMOD = 4

NHIS1 Logical unit of input history file

NHIS2 Logical unit of output history file

Block-4-2 /1/

IOPT Selection of option

= 1 new and additional registration of fuel element (input by Block-4-3)
= 2 update of registered fuel element (input by Block-4-4)
= 3 deletion of fuel element (input by Block-4-5)
= 4 Modification of final burn-up of fuel element loaded in the core (input by Block-4-6)
= 5 Modification of concentrations of I-135, Xe-135, Pm-149, and Sm-149 due to decay (input by Block-4-7)

The following Block-4-3-1 through Block-4-3-3 are required if IOPT=1 (Block-4-2) is specified.

Block-4-3-1 /A8,4X,A8,2,2/

IDENT Name of fuel element by 8 characters
8 blanks for IDENT terminate Block-4-3

NTYPE Fuel element type to which the IDENT belongs to by 8 characters registered by Block-2-3.

NOPT1 Option to specify burn-up by node
= 0 specify by Block-4-3-2
= 1 use the same value of the previously registered fuel element
Entries of Block-4-3-2 are skipped

INOPT2 Option to specify number densities by node
= 1 specify by Block-4-3-3
= 2 use the same values of the previously registered fuel element
Entries of Block-4-3-3 are skipped

BURNUP(1) Burn-up per a fuel element (MWd/element)
This value is used as the initial value for the interpolation of cross-sections. Enter 0.0 for new fuel element.

BURNUP(2) Burn-up per fuel element (depleting fraction of U-235 number density (0 to 100%). The target nuclide may be switched by using an option installed in SRAC. The data of this item is stored as the initial values in the history file, but not used in the cross-section interpolation.

Block-4-3-2 Required only if 3D calculation (IGEOM ≥ 11) and INOPT=0 /NRK1*2/

BURNUZ(1,1) Burn-up (MWd/node) of the first node

BURNUZ(1,2) Burn-up (depleting fraction of U-235 number density) of the first node
BURNUZ(2,1) Burn-up (MWd/node) of the second node
BURNUZ(2,2) Burn-up (depleting fraction of U-235 number density) of the second node
....
BURNUZ(NRK1,1)
Burn-up (MWd/node) of the NRK1-th node
BURNUZ(NRK1,2)
Burn-up (depleting fraction of U-235: %/node) of NRK1-th node
The order of nodes is numbered from front to back.
While BURNUZ(k,1) is used as the initial value in the cross-section interpolation in respect to burn-up, BURNUZ(k,2) is not used but is registered as the initial value in the history file.

Block-4-3-3 Required if INOPT=0 /NTNUC*NRK1/
(ZINV(i,k), i=1,NTNUC), k=1,NRK1)
Number density (10^{24} particle/cm^3) of the nuclide i at the node k.
The density is not in the fuel meat but is the average value in the node. The order of nuclide i follows that in the registration by Block-1-11. If NTNUC=0 is entered in Block-1-11, the order follows that for defaulted nuclides in logical unit 52 (cf. Sect.4.2).
In one- or two-dimensional calculation (IGEOM≤10), NRK1=1 is assumed.
The number densities entered in this item are used for the calculations of inventory ratio to the initial loading of fissionable depleting nuclides (U-233, U-235, Pu-239, Pu-241), depleting fraction of U-235, an optional correction of Xe-135 concentration (Eqs.(1.3-18) and (1.3-19)) in COREBN. While the number densities of other nuclides are not directly used in core burn-up calculation, their values interpolated irrespective of input values at the final burn-up step are printed out (the value for Xe-135 depends on the option).
If zero values are entered to heavy nuclides, number densities evaluated from heavy nuclide inventory entered by Block-2-3-3 are given.

Enter Block-4-3-1 through Block-4-3-3 repeatedly until blank IDENT (Block-4-3-1) is entered.

The following Block-4-4-1 through Block-4-4-6 are required if IOPT=2 (Block-4-2) is specified.

Block-4-4-1 /A8,4,0/
IDENT Name of the fuel element to be updated by 8 characters.
Blank IDENT terminates the entry of Block-4-4.

When plural fuel elements are updated by repeating Block-4-4, IDENT has to be entered in the order of registration in the history file.

ICOR(1)

= 0 no change in fuel element name

> 0 rename fuel element (available only for fresh fuel element)

ICOR(2)

= 0 no change in fuel element type

> 0 replace fuel element type (available only for fresh fuel element)

ICOR(3)

= 0 no change in burn-up

> 0 update burn-up

ICOR(4)

= 0 no change in number densities of depleting nuclides

> 0 update number densities of depleting nuclides (available only for fresh fuel element)

Block-4-4-2 Required if ICOR(1)>0 /A8/

IDENT1 Name of fuel element

Block-4-4-3 Required if ICOR(2)>0 /A8/

NFTYPE1 Name of fuel element type

Block-4-4-4 Required if ICOR(3)>0 /2/

BURNU1(1) Burn-up per fuel element (MWd/element)
If no change, enter zero value

BURNU2(2) Burn-up per element (depleting fraction of U-235 by %)
If no change, enter zero value.

Block-4-4-5 Required if 3D calculation (IGEOM ≥ 11) and ICOR(3)>0

BURNUZ(1,1) Burn-up (MWd/node) of the first node

BURNUZ(1,2) Burn-up (depleting fraction of U-235 number density) of the first node

BURNUZ(2,1) Burn-up (MWd/node) of the second node

BURNUZ(2,2) Burn-up (depleting fraction of U-235 number density) of the second node

....

BURNUZ(NRK1,1)

Burn-up (MWd/node) of the NRK1-th node

BURNUZ(NRK1,2)

Burn-up (depleting fraction of U-235: %/node) of NRK1-th node

Block-4-4-6 Required if INOPT2=0 /NTNUC*NRK1/

(ZINV(i,k), i=1,NTNUC), k=1,NRK1)

Number density(10^{24} particle/cm^3) of the nuclide i at the node k.

The order of nuclide i follows that in the registration by Block-1-11. If no change is required, enter zero value.

In one- or two-dimensional calculation, NRK1=1 is assumed.

Repeat Block-4-4-1 through Block-4-4-6 until blank IDENT in BLOCK-4-4-1 is encountered.

The following Block-4-5 is required if IOPT=3 (Block-4-2).

Block-4-5 /A8,4,0/

IDENT1 Name of fuel element to be deleted by 8 characters

NSEQ Position of registration in the history file.

Position can be confirmed by IMOD=12.

Repeat Block-4-5 until blank IDENT1 and zero NSEQ value are encountered. The entry of Block-4-2 follows for the next option.

The following Block-4-6-1 through Block-4-6-3 are required if IOPT=4 (Block-4-2)

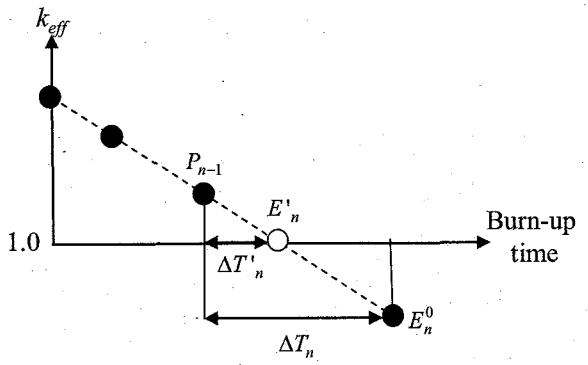
Block-4-6-1 /1/

IOPTB = 1 Modification of burn-up (the final burn-up period) of fuel element loaded in the core

= 2 Modification of burn-up of fuel element specified by masking (meta character)

Block-4-6-2 Required if IOPTB=1 /0,0,1/

HOUR The updated final burn-up period : $\Delta T'_n$ (hour)



Shown in the above figure is the decay of the effective multiplication factor around the final burn-up step as the result of COREBN. At the end of estimated period ΔT_n, k_{eff} goes down below 1.0, where power distribution $P_{n-1}(\vec{r})$, burn-up distribution $E_n^0(\vec{r})$ are recorded in the history file. Required are these quantities at the end of $\Delta T'_n$: the period when k_{eff} reaches 1.0. By specifying $\Delta T'_n$ by HOUR, burn-up distribution $E'_n(\vec{r})$ is obtained by the following equation.

$$E'_n(\vec{r}) = E_n^0(\vec{r}) + P_{n-1}(\vec{r})\{\Delta T'_n - \Delta T_n\}$$

By this procedure, the burn-up distribution at the end of cycle can be obtained without recalculation of core burn-up. However, number densities of depleting nuclides, depleting fraction of U-235 etc. are not modified unless the succeeding core burn-up calculation is achieved. This option is effective to evaluate simply the equilibrium cycle in the design calculation.

Block-4-6-3 Required if IOPTB=2 /A8,1X,A1,0,1/

FID Name of fuel elements to modify burn-up (MWd/element) by 8 characters including masking character(s). If all 8 characters are masked, all the fuel elements are specified.

MASK A character used for masking

FACT A factor to modify the burn-up

Sample input : `FUELNOXX X 2.0`

where `X` is masking character and burn-up (MWd/element, MWd/node) of all the fuel elements of which first 6 characters are FUELNO is modified by factor 2.0. If masking character is specified as blank, only the fuel element specified by FID is modified.

The information except burn-up is not modified unless the interpolation is achieved in respect to burn-up in the succeeding core burn-up calculation.

Repeat Block-4-6-3 until blank FID is encountered. At that time, dummy input for MASK and FACT (e.g. MASK=X, FACT=0.0) is necessary.

Block-4-7 Required if IOPT=5 (Block-4-2) /A8,1X,A1,0,1/

FID Name of fuel elements to give cooling time after reactor shut down by 8 characters including masking character(s). If all 8 characters are masked, all the fuel elements are specified.

MASK A character used for masking. If masking character is specified as blank, only the fuel element specified by FID is modified.

DTIME Cooling time after reactor shut down by hour.

Concentrations of I-135, Xe-135, Pm-149, and Sm-149 in the fuel element are modified by considering chain decay during cooling time. For example, concentrations of I-135 and Xe-135 are evaluated by the following equations.

$$\text{I-135} \xrightarrow{\lambda_I} \text{Xe-135} \xrightarrow{\lambda_{Xe}}$$

$$N^I = N_0^I e^{-\lambda_I t}$$

$$N^{Xe} = N_0^{Xe} e^{-\lambda_{Xe} t} + \frac{N_0^I \lambda_I}{\lambda_{Xe} - \lambda_I} \left\{ e^{-\lambda_I t} - e^{-\lambda_{Xe} t} \right\}$$

where N_0^I and N_0^{Xe} are concentrations stored in the history file before modification. Concentrations of Pm-149 and Sm-149 are evaluated similarly. As Sm-149 is stable, whereas Xe-135 vanishes after long cooling time, Sm-149 is saturated. Number densities of nuclides which are not registered are treated as zero. Modification of Xe-135 concentration may be reflected on the core burn-up calculation by using the extended function to correct equilibrium Xe concentration in COREBN (ITCAL =3, 4 in Block-2 of Sect.3.2). Modification of other nuclide concentration can not be reflected to the core burn-up calculation.

Repeat Block-4-7 until blank FID is encountered. At that time, dummy input for MASK and DTIME (e.g. MASK=X, DTIME=0.0) is necessary.

(5) Deletion of Operation Record

Block-5-1 /3/

IMOD = 5

NHIS1 Logical number of the input history file

NHIS2 Logical number of the output history file

Block-5-2 /1/

NBACH1 Batch number to begin deletion

NBACH2 Batch number to finish deletion

Operation record and fuel burn-up history corresponding to the batch numbers NBACH1 through NBACH2 are deleted. If the latest batch is deleted, the final burn-up has to be modified because the maximum burn-up of the batches remaining is not modified.

(6) Print out of core geometry and registered depleting nuclides

Block-6-1 /3/

IMOD = 6

NHIS1 Logical number of the input history file

NHIS2 = 0 dummy number (no output history file)

(7) Print out of operation record

Block-7-1 /3/

IMOD = 7

NHIS1 Logical number of the input history file

NHIS2 =0 dummy number (no output history file)

Block-7-2 /2/

IOPT Selection of options

= 1 whole operation record

= 2 operation record of one cycle (all batches included)

= 3 operation record of one batch (a series of core burn-up calculations)

= 0 end of Block-7 entry

ICBNO zero for IOPT=1, cycle number for IOPT=2, and batch number for IOPT=3, dummy input (e.g. ICBNO=0) for IOPT=0

Repeat Block-7-2 as necessary.

(8) Print out of information on fuel element type

Block-8-1 /3/

IMODE = 8

NHIS1 Logical number of the input history file

NHIS2 = 0 dummy number (no output history file)

(9) Print out of information on non-fuel element type

Block-9-1 /3/

IMODE = 9

NHIS1 Logical number of the input history file

NHIS2 = 0 dummy number (no output history file)

(10) Print out of information on individual fuel element

Block-10-1 /3/

IMODE = 10

NHIS1 Logical number of the input history file

NHIS2 = 0 dummy number (no output history file)

Block-10-2 /A8,1,0/

IDENTL Fuel element name (8 character) registered in the history file

IOPT = 0 end of Block-10
= ±1 all fuel element
= ±2 only the fuel element specified by IDENTL

If IOPT<0, burn-up history is not printed out

Repeat Block-10-2 until blank IDENTL and zero IOPT are encountered.

(11) List of operation records

Block-11-1 /3/

IMOD = 11

NHIS1 Logical number of the input history file

NHIS2 = 0 dummy number (no output history file)

(12) List of fuel elements

Block-12-1 /3/

IMOD = 12

NHIS1 Logical number of the input history file

NHIS2 = 0 dummy number (no output history file)

(13) Print out of burn-up dependent tabulation of number densities of depleting nuclides

Block-13-1 /3/

IMOD = 13

NHIS1 = 0 do not make plot file

> 0 make plot file

Number densities of depleting nuclides are read in from the member `caseDNxT` in MACRO file, then the history file is not referred. The contents of printing out are the information on burn-up and homogenization stored in this member. The plot file is written by the binary format exclusive for GPLP. If another plotter is used, use of the utility code for PDS editor (cf. Sect.6.2 of the SRAC manual[4]) is preferred to HIST code. It is more convenient to read directly the contents of member `caseDNxT` from MACRO file and edit them into the input form of plotter or table calculation software.

Block-13-2 /A8/

MEMBER Member name (8 characters) of the composition table stored in MACRO file. The last four characters of the member name must be DNxT (x: X-Region number)

Repeat Block-13-2 until blank MEMBER is encountered.
Note that the plot file is made only for the last MEMBER.

(14) List of burn-up of fuel element loaded in the core

Block-14-1 /3/

IMOD = 14

NHIS1 Logical number of the history file storing the information before the operation Allocate the history file used as the input of COREBN

NHIS2 Logical number of the history file storing the information after the operation Allocate the history file used as the output of COREBN.

This option permits the edit of burn-up before and after the core burn-up calculation and burn-up increment during the operation by every fuel element in the core. Both of NHIS1 and NHIS2 are used as input and their contents are not changed.

(15) List of axial composition of fuel element type and non-fuel element type

Block-15-1 /3/

IMOD = 15

NHIS1 Logical number of the input history file

NHIS2 = 0 dummy number (no output file)

(99) Conversion of MACRO file from PDS into PS organization

Block-99-1 /3/

IMOD = 99

NHIS1 = 0 dummy number (no use of history file)

NHIS2 Logical number of output PS file (usually NHIS2=90 is used)

The COREBN does not directly use the MACRO file provided by SRAC, but the file of sequential organization (PS) which includes the cross-sections. Therefore, when the COREBN is newly used or the materials are additionally registered or updated, MACRO file has to be converted.

3.2 Input Requirements of COREBN

Input requirements of COREBN are composed by control of core burn-up calculation and input for CITATION. The former is entered by the free format of SRAC, and the latter by the fixed column type format as the original CITATION requires.

Block-1 2*/A72/

OCOM Comment in two lines

Block-2 Control information on operation /0,14,2/

1 NOB Batch number

One job (until the end of a core burn-up calculation) is assumed as one batch, and the operation record by batch is accumulated in the history file. When the old operation record is not stored to avoid excessive amount of file, enter NOB=1 always. Contrary, to store the operation record of every batch, enter consecutive number at each batch (NOB=1,2,3...)

2 NOC Cycle number

This has no meaning in computation, but is recorded as comment in the history file.

3 NOFCOR Number of fuel elements loaded in the core

The elements which are loaded but are not explicitly treated due to symmetry in the diffusion calculation may be included.

4 NOCCOR Number of control elements in the core

5 NSBSTP Number of burn-up steps

6 IDATE(1) Reactor start-up date (YYMMDD)

YY: year, MM: month, DD: day

7 IDATE(2) Reactor shut down date (YYMMDD)

8 IREST0 Restart option

= 0 new calculation (cold start)

> 0 restart calculation (logical units 13 and 98 are required for restart)

9 IREST1 Preparation for next restart

= 0 do not prepare restart

> 0 write restart information into logical units 13 and 98

10 ITCAL Option for xenon concentration correction

= 0 make correction on equilibrium Xe concentration (ineffective for initial step)

= 1 do not make correction

= 2 start by zero Xe concentration (equilibrium correction after second step)

= 3 reactivity calculation due to Xe concentration based on the concentration in the history file (Block-2-1 is required)

= 4 start up by using Xe concentration in the history file (equilibrium correction after second step)

When other reactivity calculation than Xe concentration is achieved by using burn-up calculation function, enter ITCAL=3 and FACXE(i)=1.0 for every i to equalize the condition on Xe concentration at every step.

11 ILCAL Optional eigenvalue calculation at the final step

= 0 no eigenvalue calculation at the final burn-up step.
At the final step, update of burn-up and number densities of depleting nuclides is achieved, then calculation is terminated without diffusion calculation.

= 1 eigenvalue calculation at the final step.
Dummy entries are required for Block-3 and Block-4.

12 ICFLOW Specification of temperatures for interpolation (cf. Sect.1.3)

= 0 use the defaulted temperatures

= 1 use the temperatures entered by Block-2-3 and Block-2-4

The defaulted temperatures are as follows;

Temperature Tag	1	2	3	4	5	6	7	8	9	10	11
Temperatures (K)	300	325	350	400	450	500	550	600	900	1200	1600

If the cross-section tabulation is performed by using the other set of temperatures by SRAC, the user must enter the same set for the temperature interpolation by specifying ICFLOW=1.
If the interpolation by the other parameter than temperature (void fraction, boron concentration, fraction of control rod insertion), the user also must enter the set of

parameters instead of temperatures.

13 ID1 Specification of directional diffusion coefficients and kinetics parameters

= ±1 use the value on D1 position of macroscopic cross-section table as diffusion coefficient

= ±2 use the value on D2 position

= ±3 switch D1 or D2 position by material by Block-2-2

Negative entry specifies calculation of kinetics parameters. It requires the adjoint flux which is specified by NGC12=1 on Card-2 in 001 section described later.

14 IXKI Allocation of fission spectrum

= 0 use a spectrum commonly for every material. The spectrum of the first material on logical unit 31 will be used.

> 0 use the spectrum of each own material (recommended)

15 AVFTMP Average fuel temperature (K), cf. Block-5-1

16 AVMTMP Average moderator temperature (K), cf. Block-5-1

Block-2-1 Required if ITCAL=3 (Block-2) /NSBSTP+ILCAL/

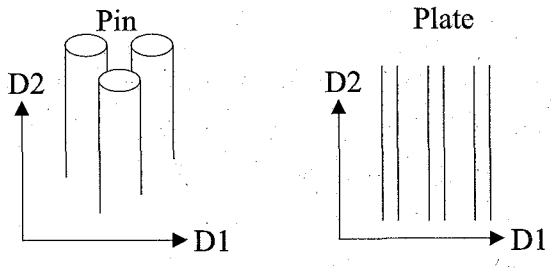
FACXE(*i*) Adjusting factors of Xe concentration (*Fact* in Eq.(1.3-17))

Block-2-2 Required if ID1=±3 /NMAT/

IXYZ(*m*) Specify the correspondence between D1, D2 and Dx, Dy, Dz by material in the order of registration in the history file.

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

The values on D1 and D2 positions depend on the option specification how to form diffusion constants when cross-sections are provided by SRAC. If directional diffusion coefficients are specified, they are stored as shown in the figure below.



On the coordinate systems other than Cartesian Dx, Dy, Dz are assigned to the direction shown below. In one- or two-dimensional calculation, Dz is used for the evaluation of leakage term (DB^2).

Geometry	Dx	Dy	Dz
1D slab	X		
1D cylinder	R		
1D sphere	Rs		
2D slab	X	Y	
2D cylinder	R	Z	
2D disk	θ	R	
2D hexagonal	X_H	Y_H	
2D triangular	X_T	Y_T	
3D slab	X	Y	Z
3D cylinder	θ	R	Z
3D hexagonal	X_H	Y_H	Z
3D triangular	X_T	Y_T	Z

Block-2-3 Required if ICFLOW=1 (Block-2) /11/

IFTMP(*i*) Fuel temperatures at which cross-sections are tabulated in ascending order. While 11 entries are always required, cross-sections don't have to be prepared at every temperature. In principle, average fuel temperatures at which cross-sections are formed by SRAC should be entered. If the interpolation in respect to fuel temperature is not performed, the entries may be the same as the defaulted values. If the cross-sections are interpolated by a parameter (moderator void fraction, boron concentration, etc.) assumed as fuel temperature, a set of parameter values are entered on this item.

Block-2-4 Required if ICFLOW=1 (Block-2) /11/

IMTMP(*i*) Moderator temperatures at which cross-sections are tabulated in ascending order. While 11 entries are always required, cross-sections don't have to be prepared at every temperature. In principle, average moderator temperatures at which cross-sections are formed by SRAC should be entered. If the interpolation in respect to moderator temperature is not performed, the entries may be the same as

the defaulted values. If the cross-sections are interpolated by a parameter (moderator void fraction, boron concentration, etc.) assumed as moderator temperature, a set of parameter values are entered on this item.

Block-3 /NSBSTP+ILCAL/

PERIO(*i*) Operation period (hour) at each step

Block-4 /NSBSTP+ILCAL/

AVRPOW(*i*) Reactor thermal power (MWt) during each step.
Enter the value by considering the symmetry in diffusion calculation. For example, the calculation is achieved on a half core by using reflective boundary condition, enter half of the whole power. In one- or two-dimensional calculation, enter the power by assuming heating length as unit length (1cm) in the direction not considered.

Block-5 Fuel loading: Repeat Block-5-1 through Block-5-3 NOFCOR (Block-2) times

Block-5-1 /A8,8,0/

ID Fuel element name (fuel element name by 8 characters registered in the history file)

LX(1) Region number in X direction to start loading
If the loading range is not expressed by this single Block as occurs in a triangular coordinates, give negative sign to LX(1) and this Block is followed by Block-5-1-1 to add the range.

LX(2) Region number in X direction to finish loading

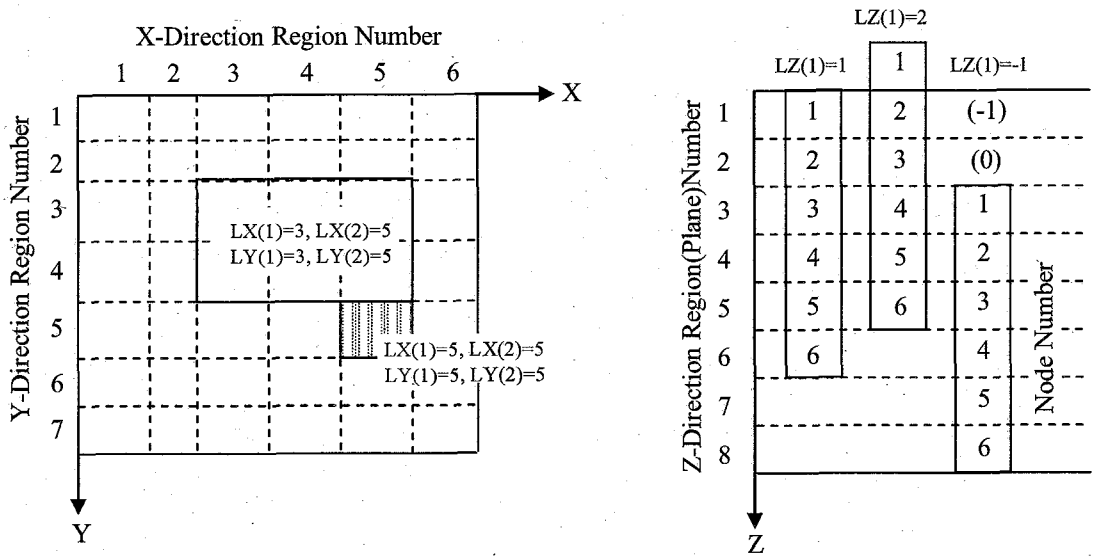
LY(1) Region number in Y direction to start loading
Enter LY(1)=1 in one-dimensional calculation.

LY(2) Region number in Y direction to finish loading
Enter LY(2)=1 in one-dimensional calculation.

LZ(1) Node number located at the first plane (region number in Z direction)
Enter LZ(1)=1 in one- or two-dimensional calculation.

How to enter LX(1) through LZ(1) is shown in the figure below. For three-dimensional calculation, if a fuel element type is registered with axial

reflector, enter LZ(1)=1. If axial reflector is treated as non-fuel element, enter LZ(1)≤0. If LZ(1)>0 is entered, the top part of the fuel element bumps out from the front boundary of calculation. This bumped part is neglected in the diffusion calculation, and the burn-up of this part is not updated.



ITEMP Specification of temperature input

= 0 give average fuel temperature by step and by burn-up node by using Block-5-2 and Block-5-3

= -1 use commonly AVFTMP and AVMTMP throughout entire steps and all axial nodes

The interpolation of cross-sections in respect to temperature is performed against the temperature specified by this item.

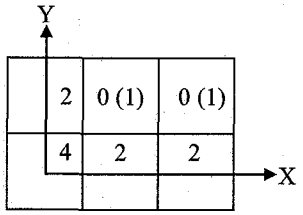
If no interpolation in respect to temperature is done, specify ITEMP=-1, and give AVFTMP and AVMTMP the temperatures corresponding to the temperature Tag.

IFACT Correction factor to volumes of fuel element

= 0 no correction

= *N* multiply factor *N* to the volume

If a fuel element is located at the center located symmetry, the quantities depending on the volume such as heavy nuclide inventory is multiplied by IFACT. An input example for 1/4 symmetric core is shown below.

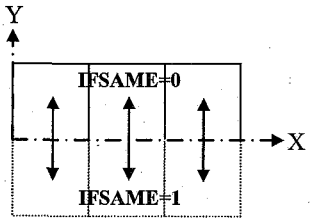


IFSAME

= 0 fuel element included in the diffusion calculation

= 1 fuel element not included in the diffusion calculation but existing in the core

By considering symmetry or shuffling, give IFSAME=1 to a fuel element which may be assumed to have the same burn-up history as the fuel element previously entered. This element is not treated in the diffusion calculation, but is given the same burn-up information as the just previously specified element.



Block-5-1-1 Required if LX(1)<0 (Block-5-1 or Block-5-1-1 itself) /4/

LX(1) Region number in X direction to start loading
If LX(1)<0, the loading range may be extended by repeating Block-5-1-1.

LX(2) Region number in X direction to finish loading

LY(1) Region number in Y direction to start loading
In one-dimensional calculation, enter LY(1)=1.

LY(2) Region number in Y direction to finish loading
In one-dimensional calculation, enter LY(2)=1.

Repeat Block-5-1-1 until LX(1)>0 is encountered.

Block-5-2 Required if ITEMP=0 (Block-5-1) /NSBSTP1*NREGKB/
(ZFTMP(i,k), i=1,NSBSTP1), k=1,NREGKB)

where NSBSTP1=NSBSTP+ILCAL (cf. Block-2), and NREGKB is number of Regions in Z direction (plane) and in one- or two-dimensional calculation, NREGKB=1.
Fuel temperature distribution (K) in axial direction. Temperatures have to be covered by those to which cross-sections are tabulated.
If the parameter other than fuel temperature is set in Block-2-3, enter parameters corresponding in this item.

Block-5-3 Required if ITEMP=0 (Block-5-1) /NSBSTP1*NREGKB/
(ZMTMP(i,k), i=1,NSBSTP1), k=1,NREGKB)
where NSBSTP1=NSBSTP+ILCAL (cf. Block-2), and NREGKB is number of Regions in Z direction (plane) and in one- or two-dimensional calculation, NREGKB=1.
Moderator temperature distribution (K) in axial direction. Temperatures have to be covered by those to which cross-sections are tabulated.
If the parameter other than moderator temperature is set in Block-2-3, enter parameters corresponding in this item.

Block-6 Allocation of control elements
Repeat Block-6-1 through Block-6-2 NOCCOR times (Block-2). If NOCCOR=0, Block-6-1 through Block-6-2 are not required.
The control element with fuel follower has to be entered in Block-5. If fuel part and absorber part is separately treated, absorber part is specified in this Block.

Block-6-1 /A8,4X,A8,4,NSBSTP+ILCAL/
IDC Name of control element by 8 characters
Individual control element is named first in this item as HIST code does not treat. When allocation and insertion depth of individual control elements are recorded in the history file, the name specified by this item is used.

MATNOC Name of non-fuel element type to which the control element belongs.
Name of non-fuel element type is registered in the history file by 8 characters.

LCX(1) Region number in X direction to start loading
If the loading range is not expressed by this single Block as occurs in a triangular coordinates, give negative sign to LCX(1) and this Block is followed by Block-6-1-1 to add the range.

LCX(2) Region number in X direction to finish loading

LCY(1) Region number in Y direction to start loading
Enter LCY(1)=1 in one-dimensional calculation.

LCY(2) Region number in Y direction to finish loading
Enter LY(2)=1 in one-dimensional calculation.

(CLOCZ(i), i=1,NSBSTP1)
where NSBSTP1=NSBSTP+ILCAL.
Control rod insertion depth (cm): which are not used in the calculation but recorded in the history file as comment.

Block-6-1-1 Required if LCX(1)<0 (Block-6-1 or Block-6-1-1 itself) /4/

LCX(1) Region number in X direction to start loading
If LCX(1)<0, the loading range may be extended by repeating Block-6-1-1.

LCX(2) Region number in X direction to finish loading

LCY(1) Region number in Y direction to start loading
In one-dimensional calculation, enter LCY(1)=1.

LCY(2) Region number in Y direction to finish loading
In one-dimensional calculation, enter LCY(2)=1.

Repeat Block-6-1-1 until LCX(1)>0 is encountered.

Block-6-2 Required if 3D calculation /NSBSTP+ILCAL/
(LCZ(i), i=1,NSBSTP1)
Node number of control element located at the first Plane (Region in Z direction). Specification of allocation and insertion depth of control element is the same as that for fuel element shown in Block-5. However, when the insertion depth is changed during exposure period, the Plane boundary (region boundary in Z direction) has to coincide always with node division boundary. The region where the control element is withdrawn is automatically substituted by the background material specified by Block-7-1 (cf. Sect.1.4).

In 2D burn-up calculation where no control rod is treated, allocation of control rod given by Block-6 may be recorded in the history file if blank MATNOC (Block-6-1) is entered. However, the control rod

with blank MATNOC is excluded from the diffusion calculation.

Block-7 Specification of background material and non-fuel element

Block-7-1 Specification of background material /A8/

MATNO1 Name of material used for background material

Enter name of non-depleting material registered in the history file by 8 characters.

The black absorber may be specified as background material.

Block-7-2 Specification of non-fuel element type

MATNO Name of non-fuel element type loaded in the core

Enter name of non-fuel element type registered in the history file by 8 characters.

Block-7-3 Allocation of non-fuel element type /5/

I1 Region number in X direction to start loading

I2 Region number in X direction to finish loading

I3 Region number in Y direction to start loading

Enter I3=1 for one-dimensional calculation

I4 Region number in Y direction to finish loading

Enter I4=1 for one-dimensional calculation

I5 Node number located at the first Plane (region in Z direction)

Enter I5=1 for one- or two-dimensional calculation

How to allocate non-fuel element type is the same as that for fuel element shown in Block-5-1.

Repeat Block-7-3 until I1=0 is encountered. (Enter dummy input 0 for I2-I5)

Then, repeat Block-7-2 through Block-7-3 until blank card is encountered.

Note. Allocation of materials to regions in the core are performed first by allocating the background material to whole regions, then non-fuel element type, fuel elements, and control elements are overwritten in this order.

The following is input requirements for CITATION part. They are identical with 001, 003, and 024 sections in SRAC-CITATION. The format is not the free format of SRAC, but the fixed column type to follow each CARD requires.

========== Section 001 : General Control ==========

Card-001-1 Section name (I3)

Enter section name '001'

Card-001-2 Control options (24I3)

NGC1 = 0

NGC2 = 0

NGC3 = 0

NGC4 = 0

NGC5 = 0

NGC6 Option to dump point-wise neutron flux

= 0 skip

= 1 write neutron flux into logical unit 9

NGC7 Option to dump point-wise power distribution

= 0 skip

= 1 write mesh information and power distribution into logical unit 32

The specification IEDG14=1 to activate this function is no more necessary.

NGC8 = 0

NGC9 = 0

NGC10 = 0

NGC11 = 0

NGC12 Adjoint flux option

= 0 skip

= ±1 calculate adjoint flux

Negative value causes print out of adjoint flux. Evaluation of kinetics parameters (ID1<0 Block-2) requires calculation of adjoint flux.

NGC13 = 0

NGC14 = 0

NGC15 = 0

NGC16 = 0

NGC17 = 0

NGC18 Residual option

= 0 evaluate multiplication factor and relative macroscopic absorption cross-sections so as to minimize summation of squared residual of point neutron balance after the eigenvalue calculation.

< 0 skip

NGC19 = 1 internally set

NGC20 = 0

NGC21 = 0

NGC22 = 0

NGC23 = 0

NGC24 = 0

Card-001-3 Edit options (24I3)

IEDG1 = 0 skip

> 0 print out iteration information

IEDG2 = 0

IEDG3 = 0 skip
> 0 print out macroscopic scattering cross-sections

IEDG3 = 0 skip
> 0 print out macroscopic reaction cross-sections

Removal cross-sections printed out do not include absorption cross-sections. Absorption cross-sections are corrected for (n,2n) reaction by Eq.(1.3-20).

IEDG5 = 0 skip
> 0 print out integrated neutron balance by group

IEDG6 = 0 skip
> 0 print out neutron balance by zone and by group

IEDG7 = 0

IEDG8 = 0

IEDG9 = 0 skip
> 0 print out zone averaged neutron flux by group

IEDG10 = 0 skip
> 0 print out point-wise neutron flux by group

IEDG11 = 0

IEDG12 = 0 skip
> 0 print out zone averaged power density (internally set)

IEDG13 = 0 skip
> 0 print out relative power distribution on the plane across the peak value

IEDG14 = 0 skip
> 0 print out point power densities

IEDG15 = 0

IEDG16 = 0 skip
> 0 print out energy integrated neutron densities

IEDG17 = 0

IEDG18 = 0

IEDG19 = 0

IEDG20 = 0

IEDG21 = 0

IEDG22 = 0

IEDG23 = 0

IEDG24 = 0 print out zone number by mesh
> 0 skip (save printing out if mesh number is excessive)

Card-001-4 General iteration count and machine time limits (24I3)

If any of the following items exceeds the limit, the iteration is terminated and the process specified by NGC15 is taken. If zero (=blank) value is entered, the defaulted value shown in < > is applied. In COREBN, other items than ITMX1, ITMX19 are ineffective.

ITMX1 Limit of outer iteration count at each burn-up step <200>

ITMX2 through ITMX18
= 0

ITMX19 Limit of CPU time in minute <60>

ITMX20 through ITMX24
= 0

Card-001-5 General restrictions (6E12.5)

If any of the following restrictions is not satisfied, the calculation is terminated. If zero (=blank) value is entered, the defaulted value in < > is applied.

GLIM1 Upper limit of multiplication factor <1.5>

GLIM2 Lower limit of multiplication factor <0.5>

GLIM3 = 0

GLIM4 = 0

GLIM5 = 0

GLIM6 = 0

When the multiplication factor fluctuates violently, set GLIM1=2.0 and GLIM2=0.001.

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

Card-003-1 Section name (I3)

Enter section name '003'

Card-003-2 General description (24I3)

NUAC1 = 0

NUAC2 Initial values of neutron flux at the restart calculation

= 0 refer neutron flux, multiplication factor, acceleration factor of the previous case

= 1 refer only neutron flux of the previous case

= 2 take installed initialization process (no reference)

NUAC3 = 0

NUAC4 = 0

NUAC5 = 0 Geometry model automatically taken from the history file

NUAC6 = 0

NUAC7 = 0

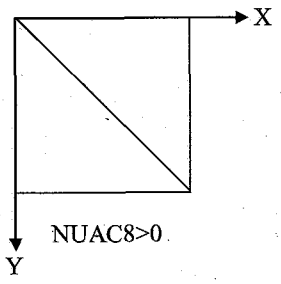
NUAC8 Option for diagonal symmetry on XY plane to accelerate the convergence by utilizing symmetry (ineffective on the vectorized version)

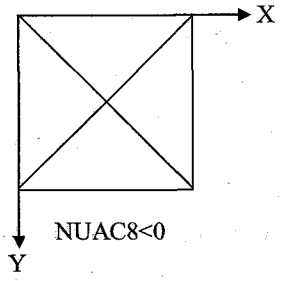
= 0 skip

> 0 number of meshes in X direction and in Y direction are identical, and symmetric on the diagonal line from upper left to lower right

< 0 number of meshes in X direction and in Y direction are identical, and point symmetric on the central point on XY plane.

If NUAC11= -1, enter NUAC8=0.



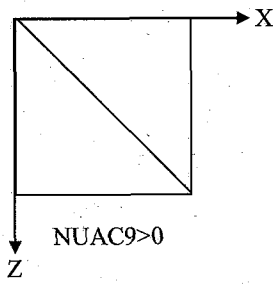


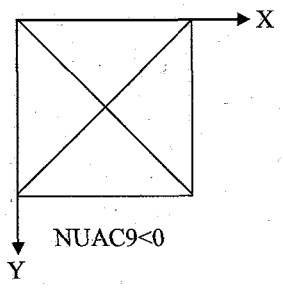
NUAC9 Option for diagonal symmetry on XZ plane (effective only for 3D calculation) to accelerate the convergence by utilizing symmetry (ineffective on the vectorized version)

= 0 skip

> 0 diagonal symmetry on the line from upper left to lower right

< 0 point symmetric on the central point on XZ plane





NUAC10 = 0

NUAC11 Left boundary condition (always effective)

= -1 periodic (effective if NUAC5=1, 6, 8, 11, 12 and left and right boundary surfaces are closed)

= 0 extrapolated (zero flux at the extrapolated distance)

= 1 reflective

NUAC12 Top boundary condition (effective if 2D or 3D calculation)

= 0 extrapolated

= 1 reflective

NUAC13 Right boundary condition (always effective)

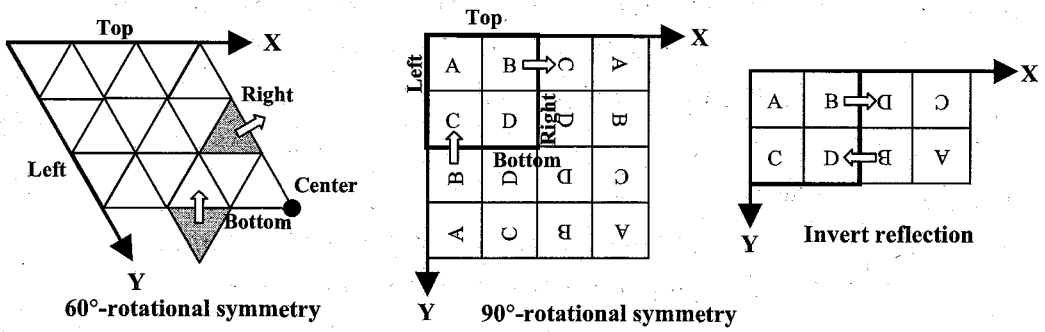
= -1 periodic (effective if NUAC11=-1)

= 0 extrapolated

= 1 reflective

= 2 90° rotational symmetry (for NUAC5=6,11)
60° rotational symmetry (for NUAC5=10,14)

= 3 inverse reflective: 180° rotational symmetry (for NUAC5= 6, 10, 11, 14)



NUAC14 Bottom boundary condition (effective for 2D or 3D)

= 0 extrapolated

= 1 reflective

= 2 if NUAC13=2

NUAC15 Front boundary condition (effective for 3D)

= 0 extrapolated

= 1 reflective

NUAC16 Back boundary condition (effective for 3D)

= 0 extrapolated

= 1 reflective

NUAC17 Internal black absorber option

= 0 no internal black absorber

> 0 zone number of internal black absorber. N-th material is treated as black absorber

An internal black absorber has the logarithmic differential (non-return) boundary condition applied at its surfaces. This zone will be black to all groups unless additional data are supplied to certain groups) by XMIS2 on Card-003-4. The cross-sections given to the zone may be accepted to the specified groups where the slowing-down neutrons into the groups specified as absorber are entirely absorbed in the group. Only one zone is allowed as an internal black absorber.

NUAC18 Option to allow negative neutron flux

= 0 negative flux is not allowed

> 0 negative flux is allowed

NUAC19 Override use of Chebychev polynomials in adjusting the parameters

= 0 Chebychev polynomials are used

> 0 Chebychev polynomials are not used

NUAC20 Line relaxation option

= -2 apply on Y- and X-direction only

= -1 apply on Y-, X- and Z-direction alternatively

= 0 for the problem involving up-scattering, apply on Y-direction only with one inner iteration. If no up-scattering, on X- and Y- alternatively in 2-D problem, and at three inner iterations for 3-D problems without I/O and at five with data I/O during iteration

> 0 apply only on Y-direction

NUAC21 = 0

NUAC22 = 0

NUAC23 Specified number of inner iterations.

= 0 use defaulted value

> 0 specify the value

NUAC24 = 0

Card-003-3 Iteration convergence criteria (6E12.5)

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

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 = 0. Not used

EPSI4 Replaces EPSI1 for all eigenvalue problems except those for initialization or station calculations. <0.0001>
(ineffective in SRAC, enter 0.0)

EPSI5 = 0. Not used

EPSI6 = 0. Not used

Card-003-4 Miscellaneous data (6E12.5)

XMIS1 External extrapolated boundary constant

> 0 the constant for all extrapolated boundaries (see NUAC11-16) for all groups

< 0 this is the total number of energy groups and Card-003-5 are to follow this card, which give the extrapolated boundary constants for problem boundaries for all energies

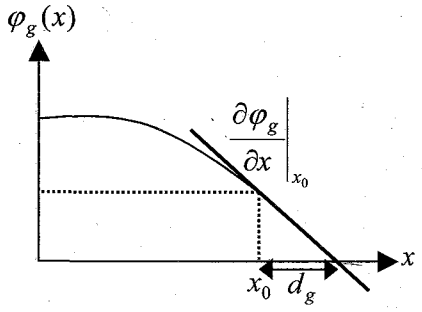
= 0 the code will use the built-in value for all extrapolated boundaries <0.4692>

The extrapolated boundary constant C_g of energy group g is defined as

$$C_g = \left[\frac{-D_g}{\varphi_g} \frac{\partial \varphi_g}{\partial x} \right]_{X=X_0}$$

and as the extrapolated length d_g is defined by

$$\frac{1}{d_g} \equiv \left[\frac{-1}{\varphi_g} \frac{\partial \varphi_g}{\partial x} \right]_{X=X_0}$$



thus the relation $C_g = D_g/d_g$ holds, where D_g is diffusion coefficient.

As the diffusion coefficient D_g is defined as

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

the constant C_g has the relation below;

$$C_g = \lambda_{tr,g} / 3d_g$$

If the extrapolation length of the Milne's problem $d_g = 0.7105\lambda_{tr,g}$ is applied, the extrapolation constant C_g becomes the defaulted value (0.4692) which does not depend on neutron energy. It is the reason to apply XMIS1=0.0 (defaulted value for all groups). If a sufficiently large value of extrapolation constant is applied, $d_g = D_g/C_g$ tends to zero so that the zero flux at the boundary is assumed.

XMIS2 The extrapolation constant for internal black absorber boundary specified by NUAC17

> 0 the constant for all groups applying

< 0 this is the total number of groups and Card-003-6 is to follow, after any required above, which gives the internal black absorber boundary constants for each energy group. Any zero values indicate that the rod condition is not applied to those groups.

= 0 the code will use the built-in value for all groups and the absorber will be black over all energy <0.4692>

For the case that involves strong absorber like control rod where the diffusion theory can not be applied, the internal black absorber may be introduced. The extrapolation constant is given by the comparison with the results of a transport calculation or/and experiments. If the cross-sections are used for certain groups, specify XMIS2<0. (cf Card-003-6).

XMIS3 Core power level, MWt

> 0 this value is used to normalize the power distribution and flux level.
For 1-D or 2-D calculation, give the thermal power by assuming the whole core of unit length (cm) for the transverse direction(s). For example, to the X-Y calculation of the core with thermal power 3,000MWt by the effective core height 100 cm, give XMIS3 = 3000/100 = 30.

= 0 defaulted value <1.0> is used.

XMIS4 Fission to power Conversion factor

> 0 ratio of thermal energy to fission energy (XMIS3 is divided by this) Normally set 1.0

= 0 defaulted value <1.0> is used

XMIS5 Core symmetry factor

The mass balances and the thermal power given by XMIS3 are divided by this number. Only the latter is effective in SRAC.

> 0 core symmetry factor i.e. fraction of the core considered; ex. for 1/4 symmetric core, enter XMIS5=0.25

Ex. to the 3-D 1/4 symmetric core calculation, if XMIS3=3,000 is given as whole core power, enter XMIS5=0.25. To the same case, if XMIS3=3000*0.25=750 is given, enter XMIS5=1.0

= 0 defaulted value <1.0> is used

To avoid confusion, it is recommended to set XMIS5=1.0 and give reactor thermal power with considering the symmetry.

XMIS6 Initial over-relaxation factor

> 0 input value is used

= 0 defaulted value <1.0> is used

Card-003-5 Required if XMIS1<0 (6E12.5)

The extrapolated boundary constants for problem boundaries; beginning with those for all energies for the left boundary; thus data is required then by beginning card refresh for right boundaries for 1-D problems.

For 2-D, give the constants in the order left, top, right, and bottom with each by card refresh.

For 3-D problems, give them for six boundaries in the order left. top, right, bottom, front and back.

For the periodic boundary condition, NUAC11=-1, skip the left and right boundaries.

For 90° rotational symmetry, NUAC13=2, skip the right and bottom..

For 180° rotational symmetry, NUAC13=3, skip the right.

Even the reflective condition is specified for the left, enter the constants of value 0.0.

Card-003-6 Required if XMIS2<0 (6E12.5)
The internal black absorber boundary constants for each energy group. Any zero value indicates that the rod condition is not applied to that group.

========== Section 024 : Buckling specification ==========

Card-024-1 Section name (I3)
Enter section name '024'

Card-024-2 Indicator and buckling value (I3,E9.0)

1 IND Specification of buckling value
= 1 specify a constant buckling in columns 4-12 (E9.0) on this card and no additional data is required.
= 2 specify values of group dependent buckling starting with Card-024-4. In this case, the KMAX value in Section 008 must be proper.
= 3 specify the group depending buckling to the set of consecutive zones specified on Card-024-3. In this case, the KMAX value in Section 008 must be proper. Continue with Card-024-3 and Card-24-4 for as many zones as required. A blank card (zero zone number) must be used to end this data.

2 BKLE Effective only if IND=1
Constant buckling value on columns 4 through12 in cm^{-2}

Card-024-3 Required if IND=3 (2I3)
Initial and final zone indicators; to the set of consecutive zones, the group dependent buckling on Card-024-4 will be applied

Card-024-4 Required if IND=2,3 (6E12.5)
Group dependent buckling values,

Card-999-1 Input section name (I3)
Enter '999'.

3.3 Notice for Usage

(1) By the modification of HIST code, it is no more necessary to use only one burn-up chain model in the cell burn-up calculation. However, to register in the history file by HIST depleting nuclides of which number densities are traced, each of them has to exist in any of composition tables (on member `caseDNxT`) of depleting materials.
By the modification of SRAC, depleting fraction of any fissile other than U-235 may be treated as the secondary measure of burn-up. If the reference nuclide for depleting fraction is replaced by another fissile in the cell burn-up calculation by SRAC, it has to be common among the depleting materials treated by COREBN. This reference nuclide has to be included in depleting nuclides and also in heavy nuclides at the initialization of the history file.

(2) The COREBN does interpolation, but does not extrapolation. Therefore, if the parameter for the interpolation exceeds the tabulated range, the calculation is terminated with message.

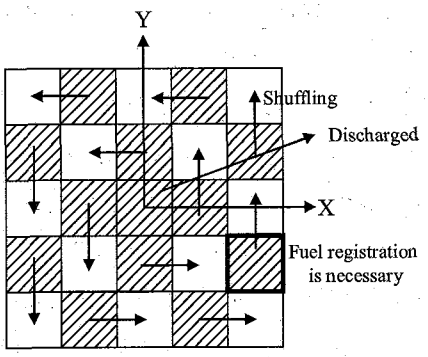
(3) The optional xenon reactivity calculation specified by ITCAL=3 in Block-2 of Sect.3.2 is not for burn-up calculation but for obtaining the reactivity curve versus Xe concentration at the condition when the history file is edited. Therefore the exposure period of every burn-up step is set zero to fix degree of burn-up.

(4) The optional kinetics parameter calculation specified by ID1<0 in Block 2 of Sect.3.2 is not available for plural burn-up steps. This option has to be specified by a separate job after finishing a series of burn-up calculations.

(5) While the number densities of depleting nuclides were treated as those in fuel meat in the old version of HIST and COREBN codes, they are treated as those in the homogenized region (an X-Region) in the revised version. That is, if materials other than fuel meat such as cladding and moderator are included in the X-Region specified in a cell burn-up calculation by SRAC, the number densities in the volume including those materials are treated in I/O of HIST code. Therefore, the initial number densities of depleting nuclides fed to HIST code for registration are not those for fuel meat fed to SRAC but for average values in the X-Region yielded by SRAC.

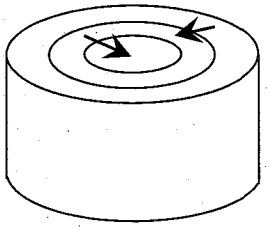
(6) In case that a partial core in some symmetric system (reflective or rotational) is treated in a core burn-up calculation with shuffling of reloaded fuel, some fuel elements not included in the diffusion calculation may be requested to register. For example, as shown in the figure below, when the 1/4 core composed by two types of fuel elements is solved, unless the elements not included in the 1/4 core are registered, the allocation of the elements for the core burn-up

calculation in the second cycle is not possible. By specifying IFSAME=1 in Block-5-1 of Sect. 3.2, COREBN gives the same information of the fuel elements in the 1/4 core to the elements located at the symmetric positions out of 1/4 core.



Care should be taken to whether or not the symmetric condition is preserved when reloaded fuel elements are moved by shuffling in a multi-batch core. Because; any unrealistic fuel shuffling in the whole core can be accepted in the partial core calculation.

(7) The information of fuel elements registered in the history file includes the quantities not depending on geometry such as burn-up and number densities of nuclides except node division in axial direction and their volumes. HIST code does not define detailed dimension in horizontal direction nor geometry. That is, until the fuel elements are allocated in the core by input specification of COREBN, their geometry is not yet defined. Especially in one- or two-dimensional calculation where there is no concept of node division, a fuel element may be assumed as liquid fuel of fixed volume. Therefore, as shown above, in cylindrical or spherical system, even though the fuel shuffling is actually not capable because of different geometry and dimension, burn-up calculation with shuffling is capable as far as the volumes are preserved.



(8) When the interpolation in respect to temperature is not performed, the fuel temperature or the moderator temperature used to form cross-sections by SRAC are not necessarily the same with those specified in the input of COREBN. For example, the cell burn-up calculation of case name FU98 by SRAC is performed with fuel temperature 880K and moderator one 610K. If the defaulted temperatures (cf. (3) in Sect.1.4) are used, the cross-sections are recognized as those with fuel temperature 900K and moderator temperature 600K by their temperature Tag. Therefore, by specifying fuel temperature AVFTMP and moderator temperature AVMTMP as 900K and 600K, respectively, instead of replacing the defaulted temperatures, the cross-sections formed with fuel temperature 880K and moderator one 610K are used only for the interpolation in respect to burn-up.

3.4 Error Messages in COREBN

Major error messages edited by COREBN and procedures for remedy are shown below. E-level messages show abnormal termination unrecoverable and W-level do caution to be able to continue but unreasonable. The italic digits denote computer output digits and small letters like 'xxxxxxxx' does output character string.

(1) E-level `LACK OF WORKING AREA (SIZEX) --------9999 xxxxxxxx STEP`
content Size of variable length array defined in MAIN routine is short by *9999* words at xxxxxxxx step
location SIZEX
remedy Enlarge parameter MXSIZE defined in include file MAINCINC by more than current value + *9999* and recompile the load modules of HIST and COREBN

(2) E-level `EXCEED NSBSMX (CRBNIN) 9999`
`RESET NSBSMX IN SUB-ROUTINE CRBN0 AND HISTORY FILE EDIT PROGRAM`
content Burn-up step number *9999* in core burn-up calculation exceeds the maximum value
location CRBNIN
remedy Enlarge parameter NSBSMX defined in include file CRBPMINC by more than *9999* and recompile the load modules of HIST and COREBN

(3) E-level `EXCEED NOFCMX (CRBNIN) 9999`
`RESET NOFCMX IN SUB-ROUTINE CRBN0 AND HISTRY FILE EDIT PROGRAM`
content Number of fuel elements *9999* in the core exceeds the maximum value
location CRBNIN
remedy Enlarge parameter NOFCMX defined in include file CRBPMINC by more than *9999* and recompile the load modules of HIST and COREBN

(4) E-level `EXCEED NOCCMX (CRBNIN) 9999`
`RESET NOCCMX IN SUB-ROUTINE CRBN0 AND HISTRY FILE EDIT PROGRAM`
content Number of control elements *9999* in the core exceeds the maximum value
location CRBNIN
remedy Enlarge parameter NOCCMX defined in include file CRBPMINC by more than *9999* and recompile the load modules of HIST and COREBN

(5) E-level EXCEED MAXNO (CRBNI2) *9999*
RESET MAXNO IN INCLUDE PARAMETER
content Number of non-fuel elements *9999* in the core exceeds the maximum value
location CRBNI2
remedy Enlarge parameter MAXNO defined in include file CRBPMINC by more than *9999* and recompile the load module of COREBN

(6) E-level EXCEED MAQXII (CRBNI2) *9999*
RESET MAXII IN INCLUDE PARAMETER
content Number of input for non-fuel elements *9999* in the core exceeds the maximum value
location CRBNI2
remedy Enlarge parameter MAXII defined in include file CRBPMINC by more than 9999 and recompile the load module of COREBN

(7) E-level X-Y LOADING POSITION (FUEL) IS OUT OF RANGE (*9,8,7,6*)
content Loading position of a fuel element on X-Y plane is out of system or not ascending order
location CRBNI2
remedy Verify LX(1)=*9*, LX(2)=*8*, LY(1)=7, LY(2)=*6* in input Block-5-1 of COREBN

(8) E-level X-Y LOADING POSITION (CONTROL) IS OUT OF RANGE (*9,8,7,6*)
content Loading position of a control element on X-Y plane is out of system or not ascending order
location CRBNI2
remedy Verify LCX(1)=*9*, LCX(2)=*8*, LCY(1)=7, LCY(2)=*6* in input Block-6-1 of COREBN

(9) E-level X-Y LOADING POSITION (NON-FUEL) IS OUT OF RANGE (*9,8,7,6*)
content Loading position of a non-fuel element on X-Y plane is out of system or not ascending order
location CRBNI2
remedy Verify LX(1)=*9*, LX(2)=*8*, LY(1)=7, LY(2)=*6* in input Block-7-3 of COREBN

(10) E-level NUMBER OF FUEL IS ZERO IN HISTORY FILE
content No fuel element is registered in the history file
location CRBN1

remedy Verify by list of fuel elements by HIST code and if not registered, register fuel elements necessary for core burn-up calculation

(11) E-level `FUEL TYPE NUMBER IS NOT FOUND IN HISTORY FILE`
`IDENT= xxxxxxxx`
`FUEL TYPE NO.=99 IF FUEL TYPE NO.=-1 FUEL ELEMENT IS NOT REGISTERED IN HISTORY FILE`

content Fuel element (xxxxxxxx) of type no. *99* is not found in the history file. If type no. is -1, fuel element (xxxxxxxx) is not registered.

location CRBN1

remedy Give correct fuel element type name to fuel element (xxxxxxxx), or register fuel element in the history file

(12) E-level `SYMMETRIC FUEL POSITION USED, BUT FUEL TYPE IS NOT EQUAL TO THE ORIGINAL FUEL`
`FUEL ID=xxxxxxxx FUEL TYPE=99 N.E. TYPE(0)=88`

content While fuel element (xxxxxxxx) of fuel element type *99* located out of computation range is specified as in symmetric position with the fuel element of type *88* in the range, element type is not consistent.

location CRBN1

remedy Verify name and type of the fuel element specified by IFSAME=1 and the previous one in Block-5-1 of COREBN. If fuel element type is wrong, correct by HIST code

(13) W-level `SYMMETRIC FUEL POSITION USED, BUT FUEL(xxxxxxxx) HAS NOT SAME HISTORY OF ORIGINAL FUEL`

content While fuel element (xxxxxxxx) located out of computation range is specified as in symmetric position with the fuel element in the range, degree of burn-up by node is different.

location CRBN1

remedy Degree of burn-up and number densities are determined by degree of burn-up recorded in the history file. Power distribution is determined by the fuel element in the core.

(14) E-level `BURN-UP INFORMATION RECORD IS NOT FOUND IN MACRO FILE (FUEL TYPE IDENT=(xxxxxxxx))`

content Member xxxxDNxT is not found in MACRO file

location CRBN12

remedy Verify member names on MACRO (PDS) file. If member xxxxDNxT is not found,

it may be wrongly deleted or renamed. The conversion of MACRO file from PDS into PS may not be correctly done. Verify also the naming following the rule of COREBN (cf. (3) in Sect.1.4)

(15) E-level `BURN-UP STEP EXCEED MAXIMUM BURN-UP STEPS FUEL TYPE (xxxxxxxx) SET=8888 NEEDED=9999`

content Number of burn-up steps in the cell burn-up calculation of material xxxxxx exceeds the maximum value *8888*. The maximum value is needed as *9999*.

location CRBN12

remedy Enlarge parameter NBSTPM defined in include file CRBBPMINC and recompile the load modules of HIST and COREBN

(16) E-level `ZONE NUMBER IS NULL, X-REGION=9, Y-REGION=8, Z-REGION=7 CHECK INPUT (CRBN22)`

content Zone specification of Region (*9, 8, 7*) is undefined

location CRBN22

remedy Verify and correct mapping of fuel and non-fuel elements in the input for CITATION

(17) E-level `99-th FUEL'S LOAD POSITION IS ALREADY LOADED OTHER FUEL ELEMENT OR NON-FUEL MATERIAL POSITION IS BAD X-REGION=9, Y-REGION=8, Z-REGION=7`

content The position (*9, 8, 7*) intended to load the *99*-th fuel element is already loaded by the other fuel element

location CRBN22

remedy Verify the input to load at the position (*9, 8, 7*)

(18) E-level `xxxxxxxx IS NOT FOUND IN HISTORY FILE. REGISTERED TABLE IS FOLLOWINGS.....`

content Material `xxxxxxxx` is not found in the history file

location CRBN24

remedy Verify registered materials by HIST code and correct material name or register it.

(19) E-level `MEMBER xxxxxxxx IS NOT FOUND IN MACRO FILE`

content Member `xxxxxxxx` is not found in MACRO file

location CRB24

remedy Verify member names on MACRO (PDS) file. If member `xxxxxxxx` is not found,

it may be wrongly deleted or renamed. The conversion of MACRO file from PDS into PS may not be correctly done. Verify also the naming following the rule of COREBN (cf. (3) in Sect.1.4).

(20) E-level `RESET LENGB=999 IN SUBROUTINE CRBN`

content On storing cross-sections, shortage of size of variable dimensioning defined by MAIN routine occurred or too small length of LENGB defined in CRBN routine is found

location CRBN3

remedy Enlarge parameter MXSIZE defined in include file MAINCINC or LENGB defined in CRBN routine and recompile the load module of COREBN

(21) E-level `MEMBER (xxxxxxxx) READ BUFFER SIZE OVER REQUIRED=888 SET=999. CHANGE SUBROUTINE CRBN (LBUFFS=999)`

content Shortage of memory to read member `xxxxxxxx` for cross-sections occurred

location CRBN4

remedy Enlarge current length of LBUFFS 999 by more than *888* and recompile the load module of COREBN

(22) E-level `BURN-UP IS OUTSIDE BURNUP TABLE. MEMBER (xxxxxxxx) BURN-UP (MWD/CC)=99.99 FUEL=(yyyyyyyy)`

content Local burn-up value (*99.99* MWd/cm^3) of fuel element `yyyyyyyy` exceeds the upper limit defined by member `xxxxxxxx`

location CRBN46, CRBN61

remedy Enlarge the upper limit of burn-up of fuel element type by restart burn-up calculation by SRAC

(23) E-level `FUEL TEMPERATURE IS OUTSIDE FUEL TEMPERATURE TABLE. MEMBER (xxxxxxxx) MODERATOR TEMP(K)=99.99 FUEL(yyyyyyyy)`

content At the material(`xxxxxxxx`) of a part of fuel element (`yyyyyyyy`), moderator temperature (*99.99*K) exceeds the upper limit of table

location CRBN46, CRBN61

remedy Verify naming of member name of the material whether or not it obeys the rule of COREBN. Also verify the temperature set for moderator (Block-2-4 of COREBN), or the specification of temperature distribution (AVMTMP in Block-2 or Block-5-3). If correct, remake the cross-section tabulation to include higher temperature by SRAC.

4. I/O Files of COREBN and HIST

4.1 Contents of History File

The history file stores the information necessary for fuel management. This file is initialized and created by HIST code beforehand the core burn-up calculation. The contents of the history file are updated, if necessary, by HIST code.

(1) Definition of COMMON variables

The labeled COMMON variables are summarized as shown below. Setting of dimension size by "parameter statements" will be described in Sect.5.3.

```
      PARAMETER (MAXII=50,    MAXNO=100)
      PARAMETER (NBSTPM=30,   NSBSMX=20,  MAXMTN=100)
      PARAMETER (NOFCMX=800,  NOCCMX=50,  MXYZF=4,  MXYZC=4)
      PARAMETER (MXISO=110,   MXNUC=110)
      PARAMETER (MAXMEM=800)
      PARAMETER (MAXX=250,    MAXY=250,   MAXZ=250,
     &           MXHVIS=25,
     &           MAXFT=50,    MAXKF=50 ,  MAXK1=50,
     &           MAXOT=50,    MAXKO=50           )
C
      CHARACTER*4 HEADER, XDATE, NISO, OCOM, ID, IDC, FTCOM, OTCOM, STATUS
      CHARACTER*8 MTNAME
C
      COMMON /REC1/  HEADER(18,2),XDATE(2)
      COMMON /REC2/ IGEOM,NREGI,NREGJ,NREGKB,NMAT,NTNUC,NHVNUC,
     1              NMESHX(MAXX),XX(MAXX),NMESHY(MAXY),YY(MAXY),
     2              NMESHZ(MAXZ),ZZ(MAXZ),MTNAME(MAXMTN),IFORS(MAXMTN),
     3              VOLFS(MAXMTN),VOLFR(MAXMTN),L235,LXE5,NISO(MXISO),
     4              IHVNUC(MXHVIS),AMASS(MXHVIS)
      COMMON /REC3/  NBATCH
      COMMON /REC4/  OCOM(18,2),NOB,NOC,NSBSTP,IDATE(2),PERIO(NSBSMX),
     1              AVRPO(NSBSMX),ACPOW,AVFTPH,AVMTPH,NOFCOR,ID(2,NOFCMX),
     2              LXYZ(2,3,MXYZF,NOFCMX),IFSAME(NOFCMX),NOCCOR,IDC(2,NOCCMX),
     3              LCXY(2,2,MXYZC,NOCCMX),CLOCZ(NSBSMX,NOCCMX),LCZ(NSBSMX,
     4              NOCCMX)
      COMMON /REC5/  NFE,NFT,NOT,NRKFMX,NRK1MX,NRKOMX
      COMMON /REC6/  FTCOM(2,MAXFT)     ,NREGKF(MAXFT),NREGK1(MAXFT),
     1              VOLF(MAXK1,MAXFT)  ,TZINV(MAXK1,MXHVIS,MAXFT),
     2              MATSPC(MAXKF,MAXFT)
      COMMON /REC7/  OTCOM(2,MAXOT),NREGKO(MAXOT),MATSPO(MAXKO,MAXOT)
      COMMON /REC8/  IDENT(2),NFTYPE,BURNUP(2),BURNUZ(2,MAXK1),
     1              ZINV(MXISO,MAXK1),STATUS,IBCORR,IDATEF(2),NLOAD,
     2              LXYZF(2,3,MXYZF),NHIS
      COMMON /REC9/  NOB1,NOC1,NSBST1,IDATEH(2),PERIO1(NSBSMX),
     1              POWZ(NSBSMX,MAXK1),ZFTEMP(NSBSMX,MAXZ),ZMTEMP(NSBSMX,MAXZ),
     2              ACBURN,ZABURN(MAXK1),NHLOAD,LXYZ1(2,3,MXYZF)
```

(2) Output format of the history file

The write statements of the contents of the history file are summarized below. If the user wants to edit by his own program, the contents may be read by replacing WRITE by READ with the labeled COMMON statements above.

```
      REWIND IFILE
C REC1
      WRITE(IFILE)    ((HEADER(I,J),I=1,18),J=1,2), (XDATE(J),J=1,2)

C REC2
      WRITE(IFILE)    IGEOM,NREGI,NREGJ,NREGKB,(NMESHX(I),XX(I),I=1,NREGI),
     1                (NMESHY(I),YY(I),I=1,NREGJ),(NMESHZ(I),ZZ(I),I=1,
     2               NREGKB),NTNUC,L235,LXE5,(NISO(I),I=1,NTNUC),NHVNUC,
     3                (IHVNUC(I),AMASS(I),I=1,NHVNUC),NMAT
      WRITE(IFILE)    (MTNAME(I),IFORS(I),VOLFS(I),VOLFR(I),I=1,NMAT)

C REC3
      WRITE(IFILE)    NBATCH

C REC4
      IF(NBATCH.EQ.0) GOTO 110
      DO 100 NB=1,NBATCH
      WRITE(IFILE)    ((OCOM(J,I),J=1,18),I=1,2),NOB,NOC,NSBSTP,
     1                (IDATE(I),I=1,2),(PERIO(I),I=1,NSBSTP),
     2                (AVRPO(I),I=1,NSBSTP),ACPOW,AVFTPH,AVMTPH,NOFCOR,
     3                ((ID(J,I),J=1,2),(((LXYZ(K,J,N,I),K=1,2),J=1,3),
     4               N=1,MXYZF),IFSAME(I),I=1,NOFCOR),
     5               NOCCOR,((IDC(J,I),J=1,2),(((LCXY(K,J,N,I),K=1,2),
     6               J=1,2),N=1,MXYZC),(CLOCZ(J,I),J=1,NSBSTP),(LCZ(J,I),
     7               J=1,NSBSTP),I=1,NOCCOR)
 100  CONTINUE
 110  CONTINUE

C REC5
      WRITE(IFILE)    NFE,NFT,NOT,NRKFMX,NRK1MX,NRKOMX

C REC6
      IF(NFT.GT.0) THEN
      WRITE(IFILE)    ((FTCOM(J,I),J=1,2),NREGKF(I),NREGK1(I),I=1,NFT),
     1                ((VOLF(K,I),(TZINV(K,J,I),J=1,NHVNUC),
     2               K=1,NREGK1(I)),I=1,NFT),
     3                ((MATSPC(J,I),J=1,NREGKF(I)),I=1,NFT)
      ENDIF

C REC7
      IF(NOT.GT.0) THEN
      WRITE(IFILE)    ((OTCOM(J,I),J=1,2),NREGKO(I),I=1,NOT),
     1                ((MATSPO(K,I),K=1,NREGKO(I)),I=1,NOT)
      ENDIF

C REC8
      IF(NFE.LE.0) GOTO 999
      DO 200 NF=1,NFE
      WRITE(IFILE)    (IDENT(I),I=1,2),NFTYPE,(BURNUP(I),I=1,2),((BURNUZ(J
     1                ,K),J=1,2),K=1,NREGK1(NFTYPE)),
     2                ((ZINV(J,K),J=1,NTNUC),K=1,NREGK1(NFTYPE)),
     3                STATUS,IBCORR,(IDATEF(J),J=1,2),NLOAD,
     4                (((LXYZF(J,I,K),J=1,2),I=1,3),K=1,NLOAD),NHIS

C REC9
      IF(NHIS.LE.0) GOTO 200
      DO 210 NH=1,NHIS
```

```
      WRITE(IFILE)   NOB1,NOC1,NSBST1,(IDATEH(J),J=1,2),(PERIO1(J),J=1,
     1               NSBST1),((POWZ(J,K),J=1,NSBST1),K=1,NREGK1(NFTYPE)),
     2               ((ZFTEMP(J,K),J=1,NSBST1),K=1,NREGKB),
     3               ((ZMTEMP(J,K),J=1,NSBST1),K=1,NREGKB),
     4               ACBURN,(ZABURN(J),J=1,NREGK1(NFTYPE)),NLOADH,
     5               (((LXYZ1(J,I,K),J=1,2),I=1,3),K=1,NLOADH)
  200  CONTINUE
  210  CONTINUE
  999  CONTINUE
```

(3) Detailed contents of the history file with variable names

Record 1 Header

HEADER(1-18,1-2)

Comment (2*72 characters=2*18A4)

XDATE(1-2) Date of the final update by 8 characters (2A4)

Record 2 Control variables for diffusion calculation

IGEOM Geometry type

=1 1D slab (X)

=2 1D cylinder (R)

=3 1D sphere (R_S)

=4 not used

=5 not used

=6 2D slab (X-Y)

=7 2D cylinder (R-Z)

=8 2D disk (θ-R)

=9 2D hexagonal (X_H-Y_H)

=10 2D triangular (X_T-Y_T)

=11 3D slab (X-Y-Z)

=12 3D cylinder (θ-R-Z)

=13 3D hexagonal (X_H-Y_H-Z)

=14 3D triangular (X_T-Y_T-Z)

NREGI Region division number in X direction

NREGJ Region division number in Y direction

NREGKB Region division number in Z direction

NMESHX(*i*) Mesh division number in each Region in X direction

XX(*i*) Region size (cm) in X direction, *i*=1,NREGI

NMESHY(*j*) Mesh division number in each Region in Y direction

YY(*j*) Region size (cm) in Y direction, *j*=1,NREGJ

NMESHZ(*k*) Mesh division number in each Region in Z direction

ZZ(k)	Region size (cm) in Z direction, k=1,NREGKB
NTNUC	Number of depleting nuclides registered
L235	Position of the reference nuclide in the depleting nuclide table. The reference nuclide is used for measure of burn-up by the depleting fraction of number density. U-235 is used as default, but it may be replaced (cf. Sect.2.11 in the SRAC manual[4])
LXE5	Position of Xe-135 in the depleting nuclide table.
NISO(l)	Name of depleting nuclide by 4 characters, l=1,NYNUC
NHVNUC	Number of heavy nuclides registered
IHVNUC(m)	Position of heavy nuclide in the depleting nuclide table, m=1,NHVNUC
AMASS(m)	Mass of heavy nuclide, m=1,NHVNUC
NMAT	Number of materials registered
MTNAME(n)	Name of material by 8 characters, n=1,NMAT
IFORS(n)	Material type =0 non depleting material =1 fissionable depleting material (usual fuel) =2 non-fissionable depleting material =3 blackness
VOLFS(n)	Correction factor used to convert burn-up unit (cf. Block-1-10 in Sect.3.1)
VOLFR(n)	Fraction of fuel meat in the material (always 1.0 in COREBN)

Record 3 Control variables on operation record

NBATCH	Number of operation batch

Record 4 Operation record at each batch

OCOM(1-18,1-2)	Comment on this operation batch (2*72 characters: 2*18A4)
NOB	Serial batch number
NOC	Cycle number (one cycle consists of one or more batch(es))
NSBSTP	Number of burn-up steps of this batch
IDATE(1)	Date to start operation by 6 digits : YYMMDD
IDATE(2)	Date to shut down operation by 6 digits : YYMMDD
PERIOD(i)	Operation period (hour), i=1,NSBSTP
AVRPO(i)	Average thermal power (MWt) during the i-th step
ACPOW(i)	Integrated thermal power by burn-up (MWd)
AVFTPHH	Average fuel temperature (K)
AVMTPH	Average moderator temperature (K)
NOFCOR	Number of fuel elements in the core

ID(*m*)	Name of individual fuel element by 8 characters, *m*=1,NOFCR
LXYZ(1,1,*,*m*)	Region number in X direction to start loading *m*-th fuel element '*' denotes serial specification number (cf. Block-5-1 in Sect.3.2), *m*=1,NOFCOR
LXYZ(2,1,*,*m*)	Region number in X direction to finish loading the *m*-th fuel element
LXYZ(1,2,*,*m*)	Region number in Y direction to start loading the *m*-th fuel element
LXYZ(2,2,*,*m*)	Region number in Y direction to finish loading the *m*-th fuel element
LXYZ(1,3,*,*m*)	Node number located in the first Plane of the *m*-th fuel element
LXYZ(2,3,*,*m*)	Volume correction factor for the *m*-th fuel element located at the center of symmetry (cf. Block-5-1 in Sect.3.2)
IFSAME(*m*)	Specification of symmetric condition (cf. Block-5-1 in Sect.3.2)
NOCCOR	Number of control elements in the core
IDC(*n*)	Name of individual control element by 8 characters, *n*=1,NOCCOR
LCXY(1,1,*,*n*)	Region number in X direction to start loading the *n*-th control element '*' denotes serial specification number (cf. Block-6-1 in Sect.3.2), n=1,NOFCOR
LCXY(2,1,*,*n*)	Region number in X direction to finish loading the *n*-th control element
LCXY(1,2,*,*n*)	Region number in Y direction to start loading the *n*-th control element
LCXY(2,2,*,*n*)	Region number in Y direction to finish loading the *n*-th control element
CLOCZZ(*i*, *n*)	Insertion depth of the *n*-th control element at the *i*-th burn-up step This item is merely users comment and not used in the calculation
LCZ(*i*, *n*)	Node number of the *n*-th control element located at the first plane. (At the full insertion from front, it makes LCZ=1)

Record 5 Control variables for fuel element and non-fuel element

NFE	Number of fuel elements registered
NFT	Number of fuel element types
NOT	Number of non-fuel element types
NRKFMK	Maximum number of nodes among whole fuel elements
NRK1MX	Maximum number of nodes in effective fuel part among whole fuel elements
NRKOMX	Maximum number of nodes among whole non-fuel elements

Record 6 Specification of individual fuel element type

FTCOM(1-2,*m*)	Name of each fuel element type by 8 characters (2A4), *m*=1,NFT
NREGKF(*m*)	Total number of nodes, (if 1D or 2D, NREGKF(*m*)=1)
NREGK(*m*)	Number of depleting nodes (if 1D or 2D, NREGK(*m*)=1)
VOLF(*k*1,*m*)	Volumes of depleting nodes *k*1=1,NREGK(*m*)

TZINV($k1$,j,i)	Initial inventories (gram) of heavy nuclides (j=1,NHVNUC) in node i on region $k1$
MATSPC(k,m)	Material number of each node k=1,NREGKF(m) Material is numbered in the order of registration into Record 2

Record 7 Specification of non-fuel type

OTCOM(1-2,n)	Name of each non-fuel element type (n=1,NOT) by 8 characters (2A4)
NREGKO(n)	Total number of nodes (if 1D or 2D, NREGKO(n)=1)
MATSPO(k,n)	Material number of each node k=1,NREGKO(n) Material is numbered in the order of registration into Record 2

Record 8 Specifications of individual fuel element (NFE elements)

IDENT(1-2)	Name of fuel element by 8 characters (2A4)
NTYPE	Fuel element type number Type is numbed in the order of registration of MTNAME into Record2
BURNUP(1)	Average burn-up (MWd/element) per element
BURNUP(2)	Depleting fraction of number density of the reference nuclide (%)
BURNUZ(1,$k1$)	Axial distribution of burn-up (MWd/node), $k1$=1,NREGK1(NFTYPE)
BURNUZ(2,$k1$)	Axial distribution of depleting fraction of the reference nuclide (%)
ZINV(i,$k1$)	Number densities (particle 10^{24}/cm^3)of depleting nuclides (i=1,NTNUC) in each node : They are updated after the termination of core burn-up calculation while Xe-135 concentration is determined by following the correction option (cf. Sect.1.3)
STATUS	Status of fuel element expressed by 4 characters (A4) =_NEW(the first character is blank): new fuel =CORE : loaded in the core =COOL : cooled out of the core =WEST : spent fuel
IBCORR	Status of update by HIST code =0 no update =1 updated except number densities =2 fully updated
IDATEF(1)	Date for initialization of the history file (YYMMDD)
IDATEF(2)	Date for delete unnecessary record (YYMMDD)

The followings show loading position of fuel elements at the latest irradiation.

NLOAD	Number of loading position specifications (cf. Block-5-1 in Sec.3.2) of this fuel element

LXYZF(1,1,*k*)	Region number in X direction to start loading, *k*=1,NLOAD
LXYZF(2,1,*k*)	Region number in X direction to finish loading
LXYZF(1,2,*k*)	Region number in Y direction to start loading
LXYZF(2,2,*k*)	Region number in Y direction to finish loading
LXYZF(1,3,*k*)	Node number located at the first Plane (cf. Block-5-1 in Sect.3.2)
LXYZF(2,3,*k*)	Volume correction factor for the element located at the center of symmetry (cf. Block-5-1 in Sect.3.2)
NHIS	Number of repetitions of Record 9

Record 9 Burn-up history specified by Record 8, /NHIS/*NFE times

NOB1	Serial batch number
NOC1	Cycle number
NSBST1	Number of burn-up steps in this batch
IDATEH(1)	Date to start operation (6 digits: YYMMDD)
IDATEH(2)	Date to shut down (YYMMDD)
PERIO1(*j*)	Operation period (hour) of each step
POWZ(*j*,*k*)	Thermal power(MWt/node) of each node (*k*=1,NREGKB) by step (*j*=1,NSBBBST1)
ZFTEMP(*j*,*k*)	Fuel temperature of each node by step
ZMTEMP(*j*,*k*)	Moderator temperature of each node by step
ACBURN	Average burn-up (MWd/element) at the end of operation
ZABURN(*j*,*k*)	Average burn-up (MWd/node) of each node at the end of operation
NLOADH	Number of specifications of loading position
LXYZ1(1,1,*k*)	Region number in X direction to start loading, *k*=1,NLOADH
LXYZ1(2,1,*k*)	Region number in X direction to finish loading
LXYZ1(1,2,*k*)	Region number in Y direction to start loading
LXYZ1(2,2,*k*)	Region number in Y direction to finish loading
LXYZ1(1,3,*k*)	Node number located at the first Plane (cf. Block-5-1 Sect.3.2)
LXYZ1(2,3,*k*)	Volume correction factor for the element located at the center of symmetry

4.2 I/O Files of HIST

(1) List of I/O files

Logical unit	Data format	Contents
1	VB	scratch (always required)
5	FB	standard input
6	FB	standard output
(10 through 20)	VB	I/O interface of the history file, Unit is specified, but use 10 through 20.
49	VB	This unit is always used as an interface to open / close / read / write / delete a PDS member
50	FB	scratch to keep standard input data
51	FB	scratch to keep directory path name and list of members of MACRO file
52	FB	Defaulted registration of depleting nuclides and their masses (can be updated by user)
(90)	VB	PS organization of MACRO file. Unit number (usually 90) is specified in input.
97	FB	Error message in reading PDS file

FB: formatted text file, VB: unformatted binary file

(2) Contents of file unit 51

The file assigned to unit 51 is required to access MACRO file (PDS). This file is created by the following shell-script before starting HIST code, and deleted after the job (cf. Sect.5.2).

```
#  XDR  : Directory name of MACRO(PDS) file
#  TMP  : Name of PDS member list file
   Set XDR = /home/okumura/crbn/smpl/macro
   Set TMP = /home/okumura/tmp.dat
   echo  $XDR >  $TMP
   cd    $XDR
#  ls -1 ???????? >> $WKDR/tmp.$DATE
   ls -1 ????[AFD]??? >> $WKDR/tmp.$DATE
```

In the file created by the above shell script, the path name and member list are written as shown below. The HIST code access to PDS file by referring these information.

```
  /home/okumura/crbn/smpl/macro
 FU98A010 FU98A01N FU98A01Z FU98A110 FU98A11N FU98A11Z
 FU98A210 FU98A21N FU98A21Z FU98A310 FU98A31N FU98A31Z
 FU98A410 FU98A41N FU98A41Z ...
```

Note that the members are not used in HIST or COREBN unless the fifth character is any of A, F, D. They may stay in the file, but they waste the memory.

(3) Contents of file unit 52

The file assigned to unit 52 is read at the initialization of the history file. This file stores the defaulted set of depleting nuclides and their atomic masses used for calculation of heavy nuclide inventory. The contents of file may be prepared by the user. Data should be prepared in the SRAC free format (cf. Sect.2.1 in the SRAC manual[4]). The structure of the file is as follows;

Block-1 /3/

NHV	Total number of heavy nuclides on the file
NFP	Total number of FP nuclides on the file
NBP	Total number of burnable poison nuclides

Block-2 /1/

ANMU	Conversion factor AMASS entered by Block-3 into atomic mass unit (amu). If AMASS is described by amu unit, enter 1.0.

Block-3 /A4,1,1/

ISOTB	Nuclide name by 4 characters Enter nuclide name under the rule of SRAC like `XU05`, `XU08`.
IDEFT	= 0 not use as defaulted depleting nuclide > 0 use as defaulted depleting nuclide Especially if IDEFT=2, use as defaulted heavy nuclide
AMASS	Relative atomic mass While unit is arbitrary, enter the value to make it in amu unit by multiplying ANMU. Enter usually neutron mass unit relative to reduced neutron mass as described in data book.

Repeat Block-3 NHV+NFP+NBP times. Ordering is arbitrary.

The registration of this set of nuclides becomes effective by specifying NTNUC=0 or NHVNUC=0 in Block-1-3 of Sect.3.1 for the use of the defaulted set of nuclides. Even if depleting nuclides or heavy nuclides are specified by Block-1-11 or Block-1-12 without using the defaulted set, atomic masses for heavy nuclides are read from this file.

The nuclides registered as heavy nuclide are given the atomic masses from this file into the history file, and they are used in the heavy nuclide inventory calculation. If the total weight of FP nuclides is necessary, non-heavy nuclides may be registered as heavy nuclides.

The following is a sample file (current standard set) which cover nuclides treated in the burn-up chain models of SRAC. In this sample, the defaulted set of heavy nuclides includes 9 nuclides of U-235, U-236, U-237, U-238, Pu-238, Pu-239, Pu-240, Pu-241, and Pu-242.

```
&******** Mass Table for HIST code (Read from the 52th Device) *********
29  66  1     / Number of Registered Nuclide in This Table(Heavy,FP,BP)
1.008665      / Conversion Factor from AMASS to Atomic Mass Unit(amu)
&-------------------------------------------------------------------
&    ISOTB  : Nuclide Name(A4) on SRAC Rule
&    IDEFT  : =0 Not Use as default Burnable Nuclide Table
&           >0 Use as Default Burnable Nuclide Table
&******** Mass Table for HIST code (Read from the 52th Device) *********
29 185 0      / Number of Registered Nuclide in This Table(Heavy,FP,BP)
1.008665      / Conversion Factor from AMASS to Atomic Mass Unit(amu)
&-------------------------------------------------------------------
&    ISOTB  : Nuclide Name(A4) on SRAC Rule
&    IDEFT  : =0 Not Use as default Burnable Nuclide Table
&           >0 Use as Default Burnable Nuclide Table
&           =2 Use as Default Heavy Nuclide Table
&           IDEFT>0 for u4cm6fp50bp16 chain
&    AMASS  : Relative Atomic Mass (Usually Neutron Mass Unit)
&-------------------------------------------------------------------
&ISOTAB IDEFT   AMASS   (Free Format of SRAC System)
&-----------( Heavy Nuclide )-----------------------------------------
XTH0     0  228.057  /    1   TH230
XTH2     0  230.045  /    2   TH232
XPA1     0  229.051  /    3   PA231
XPA3     0  231.038  /    4   PA233
XU02     0  230.044  /    5   U-232
XU03     0  231.038  /    6   U-233
XU04     1  232.030  /    7   U-234
XU05     2  233.025  /    8   U-235
XU06     2  234.018  /    9   U-236
XU07     2  235.013  /   10   U-237
XU08     2  236.006  /   11   U-238
XNP6     0  233.973  /   12   NP236
XNP7     1  235.012  /   13   NP237
XNP9     1  236.999  /   14   NP239
XPU6     0  234.018  /   15   PU236
XPU8     2  236.005  /   16   PU238
XPU9     2  236.999  /   17   PU239
XPU0     2  237.992  /   18   PU240
XPU1     2  238.986  /   19   PU241
XPU2     2  239.979  /   20   PU242
XAM1     1  238.986  /   21   AM241
XAMG     1  239.981  /   22   AM242
XAMM     1  239.981  /   23   AM242M
XAM3     1  240.973  /   24   AM243
XCM2     1  239.980  /   25   CM242
XCM3     1  240.972  /   26   CM243
XCM4     1  241.966  /   27   CM244
XCM5     1  242.961  /   28   CM245
XCM6     1  243.953  /   29   CM246
&---------- FP Nuclides ----------------------
XGE3     0   72.373  /    1   GE073
XGE4     0   73.364  /    2   GE074
XGE6     0   75.347  /    3   GE076
```

```
XAS5     0   74.356  /    4   AS075

  :      :             :

XI05     1  133.84   /  107   I-135
XXEG     0  124.918  /  108   XE126
XXE8     0  126.9    /  109   XE128
XXE9     0  127.892  /  110   XE129
XXE0     0  128.883  /  111   XE130
XXE1     1  129.875  /  112   XE131
XXE2     0  130.866  /  113   XE132
XXE3     1  131.857  /  114   XE133
XXE4     0  132.849  /  115   XE134
XXE5     1  133.84   /  116   XE135
XXE6     0  134.832  /  117   XE136
XCS3     1  131.857  /  118   CS133
XCS4     1  132.849  /  119   CS134
XCS5     1  133.84   /  120   CS135
XCS6     0  134.832  /  121   CS136
XCS7     1  135.823  /  122   CS137

  :      :             :

&---------- BP Nuclides ----------------------
XB00     0    9.927  /    1   B-010

  :      :             :

XHF6     0  174.488  /   12   HF176
XHF7     0  175.479  /   13   HF177
XHF8     0  176.471  /   14   HF178
XHF9     0  177.462  /   15   HF179
XHF0     0  178.454  /   16   HF180
&-----------------------------------------
```

4.3 I/O Files of COREBN

Unit	Data Format*	Content
1	VB	Scratch (always necessary)
2	VB	Scratch (always necessary)
3	VB	Scratch (always necessary)
5	FB	Standard Input
6	FB	Standard Output(Error Message only, Results are on Unit 99)
9	VB	Point-wise neutron flux (NGC6>0), or forward and adjoint fluxes used for perturbation calculation.
10	VB	Scratch (always necessary)
11	VB	Scratch (always necessary)
13	VB	Required if IREST0>0 or IREST1>0. I/O of restart data
14	VB	Scratch (always necessary) for macroscopic cross-sections
15	VB	Equation constants used in the iteration. Use of fast devise is effective for fast convergence
16	VB	Scratch (always necessary)
18	VB	Scratch for kinetics parameter calculation
19	VB	Scratch (always necessary)
26	VB	Scratch for kinetics parameter calculation and storage of restart data
31	FB	Scratch to provide macroscopic cross-sections from COREBN to CITATION
32	VB	Required if NGC7>0 for point-wise thermal power
50	FB	Scratch (always necessary) for storage of standard input
89	VB	Scratch (always necessary)
90	VB	PS file converted from MACRO (PDS) file
91	FB	Scratch (always necessary) for storage of CITATION input
92	VB	Input history file
93	VB	Output history file
94	FB	Scratch (always necessary) for storage of CITATION input
95	FB	Scratch (always necessary)
96	VB	Scratch (always necessary)
97	VB	Scratch (always necessary)
98	VB	Required if IREST0>0 or IREST1>0 for I/O of restart data
99	FB	Print output for whole calculation

FB: formatted text file, VB: unformatted binary file

5. Job Control Statements for HIST

5.1 Shell-script for Creation of Executable

An example of shell-script is shown below for creation of an HIST executable for the LINUX machine using g77/gcc compilers. The Fortran sources of SRAC part are commonly used by HIST code.

```
#!/bin/csh
#######################################################################
#
#  << Create HIST executable >>
#
#######################################################################
#
   alias  rm    rm
   alias  cp    cp
   alias  cd    cd
   alias  mkdir mkdir
   alias  echo  echo
#
#---------- Set Directory Name and Compile Driver & Options
#
   set CRBN_DIR = /home/okumura/code/SRAC2K6/COREBN
   set SRAC_DIR = /home/okumura/code/SRAC2K6/SRAC
#
   set    F77     = g77
   set    FFLAGS   = "-fno-automatic -finit-local-zero -O2 -funroll-loops"
#
   set    CC      = gcc
   set    CFLAGS   = "-DPOSIX_C"
   set    LDFLAGS  = -static
#
#---------- Set Load Module Name & Directory Name of Include Statement
#
#  The same include file used for corebn code should be used for hist code.
   set    LMN     = $CRBN_DIR/bin/HIST.100m
   set    INC     = $CRBN_DIR/src/inc/crbn100m
#
#---------- Make Working Directory -------------------------------
#
   set    DATE    = `date +%Y.%m.%d.%H.%M.%S`
   set    WKDIR   = $CRBN_DIR/tmp/tmpHIST.$DATE
   mkdir  $WKDIR
#
#---------- Copy Source Programs into Working Directory ----------
#
   cd $CRBN-DIR/src
   cp hist/*.f              $WKDIR
   cd $SRAC_DIR/src
   cp extnl/*.f             $WKDIR
   cp extnl/*.c             $WKDIR
   cp common/*.f            $WKDIR
   cp $INC/*INC             $WKDIR
   echo "--- all source programs are copied into temporary directory ---"
```

```
#
#---------- Compile C programs ----------------
#
   cd $WKDIR
   $CC   -c $CFLAGS *.c
   echo "--- end compile process for C-programs ---"
#
#---------- Compile F programs ----------------
#
   $F77  -c $FFLAGS *.f
   echo "--- end compile process for Fortran-programs ---"
#
#---------- Link all objects --------
#
   $F77 -o $LMN $FFLAGS *.o $LDFLAGS
   echo "--- end linking process ---"
#
#---------- Remove Working Directory ----------
#
   cd ..
   rm -r $WKDIR
   echo "--- temporary directory deleted ---"
#
#---------- End Process -----------------------
```

Without use of the above shell-script, user can easily install the HIST code by using an interactive installation command (@PunchMe) equipped in the COREBN files.

5.2 Shell-script for HIST Execution

The following is an example of the shell-script to execute the HIST code.

```
#!/bin/csh
######################################################################
#
#  <<  run HIST  >>
#
######################################################################
#  sample problem HTinit : Initialization of a History File
######################################################################
#
   alias  mkdir mkdir
   alias  cat   cat
   alias  ls    ls
   alias  echo  echo
   alias  rm    rm
   alias  cd    cd
#
#=============  Set by user ==================================================
#
#  LMN    : load module name of HIST
#  ODR    : directory name in which output data will be stored
#  IDR    : directory name in which input data(case.inp) is located
#  XDR    : directory name of MACRO PDS file
#  HTO    : directory name and file name of old history
```

```
#          (not necessary if initialization)
#  HTN    : directory name and file name of new history
#  PSX    : directory name and file name of PS converted MACRO
#           to be made by option-99
#  CASE   : case name which is referred as names of output files
#  WKDR   : directory name in which scratch PS files will be made and deleted
#
   set CRBN_DIR = $HOME/SRAC2k6/COREBN
   set SMPL_DIR = $CRBN_DIR/smpl
   set LMN  = HIST.100m
   set ODR  = $SMPL_DIR/X1D/outp
   set XDR  = $SMPL_DIR/macroPDS
   set HTO  =
   set HTN  = $SMPL_DIR/X1D/outp/HIST.BO1C
   set PSX  = $SMPL_DIR/macroPS.dat
   set CASE = HTinit
#
#============= Change if you like ============================================
#
   set LM   = $CRBN_DIR/bin/$LMN
   set TAB  = $CRBN_DIR/lib/standard
   set DATE = `date +%Y.%m.%d.%H.%M.%S`
   set WKDR = $HOME/HISTtmp.$CASE.$DATE
   mkdir $WKDR_
#
   set OUTLST =  $ODR/$CASE.HFT06.$DATE
#  setenv  fu10  $HTO
   setenv  fu20  $HTN
   setenv  fu52  $TAB
#  setenv  fu90  $PSX
#
#============= Generate a member list file for PDStoPS =======================
#
   echo  $XDR >  $WKDR/tmp.$DATE
   cd    $XDR
#  ls -1 ???????? >> $WKDR/tmp.$DATE
   ls -1 ????[AFD]??? >> $WKDR/tmp.$DATE
   setenv  fu51  $WKDR/tmp.$DATE
#
#============= Exec HIST code with the following input data ==================
#
cd $WKDR
cat - << END_DATA | $LM >& $OUTLST
*******************************
*   Input data for HIST code   *
*******************************
1   0  10     / Block-1-1
Sample for 1-Dimensional Slab Core
Burnup for 3.0w/o Enriched Fuel Assembly
1  21  0  0  6  71  7  / Block-1-3
  :
  :
END_DATA
#
#============= Remove temporary file =========================================
#
   cd $HOME
   rm -r $WKDR
```

5.3 Modification of Fixed Size of Arrays

The HIST is the code to manage the history file and makes I/O process in a short CPU time. For this reason, it does not adopt variable length array but fixed length arrays. Most of sizes of arrays are defined by parameter statements. If modification of sizes of fixed length arrays is required, it is achieved by modifying the parameter values described in the include files and recompiling to create a new executable. The include files setting the standard values are under the following directory.

$CRBN_DIR/crbn/src/inc/usrinc/

where, $CRBN_DIR is the installed top directory : e.g. /home/okumura/COREBN

This directory involves the following 12 files.

File name	Code applied	Contents
REC01NC	HIST	Dimensioning of COMMON/REC1/
REC02NC	HIST	Dimensioning of COMMON/REC2/
REC03NC	HIST	Dimensioning of COMMON/REC3/
REC04NC	HIST	Dimensioning of COMMON/REC4/
REC05NC	HIST	Dimensioning of COMMON/REC5/
REC06NC	HIST	Dimensioning of COMMON/REC6/
REC07NC	HIST	Dimensioning of COMMON/REC7/
REC08NC	HIST	Dimensioning of COMMON/REC8/
REC09NC	HIST	Dimensioning of COMMON/REC9/
MAINCINC	COREBN	Dimensioning of a variable array
CITPMINC	COREBN	Dimensioning of a fixed array
CRBPMINC	HIST & COREBN	Dimensioning of fixed length arrays in HIST & COREBN

The contents of REC01INC through REC09INC are dimensioning of COMMON arrays for organizing the history file as shown in Sect.4.1. For example, the content of REC02INC is as follows;

```
      COMMON /REC2/ IGEOM,NREGI,NREGJ,NREGKB,NMAT,NTNUC,NHVNUC,
     1              NMESHX(MAXX),XX(MAXX),NMESHY(MAXY),YY(MAXY),
     2              NMESHZ(MAXZ),ZZ(MAXZ),MTNAME(MAXMTN),IFORS(MAXMTN),
     3              VOLFS(MAXMTN),VOLFR(MAXMTN),L235,LXE5,NISO(MXISO),
     4              IHVNUC(MXHVIS),AMASS(MXHVIS)
```

What decides the sizes of the arrays is the contents of the file CRBPMINC where the sizes of fixed length arrays are defined by the following parameter statements.

```
C FOR COREBN-------------------------------------------------------------------
      PARAMETER ( MAXII=50,    MAXNO=100 )
C FOR COREBN & HIST ------------------------------------------------------------
      PARAMETER ( NBSTPM=30,   NSBSMX=20,  MAXMTN=100 )
      PARAMETER ( NOFCMX=800,  NOCCMX=50,  MXYZF=4,  MXYZC=4 )
```

```
      PARAMETER ( MXISO=110,   MXNUC=110 )
      PARAMETER ( MAXMEM=800 )
C FOR HIST-------------------------------------------------------------------
      PARAMETER ( MAXX=250,    MAXY=250,   MAXZ=250,
     &            MXHVIS=25,
     &            MAXFT=50,    MAXKF=50 ,  MAXK1=50,
     &            MAXOT=50,    MAXKO=50          )
      PARAMETER ( MAXWRK=500000 )
      PARAMETER ( MAXK1I=MAXK1*MXISO, MAXK1H=MAXK1*MXHVIS )
C ----------------------------------------------------------------------------
```

This include file is commonly used in HIST and COREBN codes. The first statement is exclusively for COREBN, the succeeding 4 statements for common use of both codes, and lower 6 statements are for uniquely HIST. The meaning of these parameters is as follows;

Parameter name	Standard value	Code* applied	Contents (upper limit)
MAXII	50	C	Number of inputs for non-fuel elements into overlay zones
MAXNO	100	C	Number of overlay zones for non-fuel elements
NBSTPM	30	C&H	Number of burn-up steps on cross-section table
NBSSMX	20	C&H	Number of burn-up steps in core burn-up calculation
NOFCMK	800	C&H	Number of fuel elements loaded in an operation
NOCCMK	50	C&H	Number of control elements loaded in an operation
MXYZF	4	C&H	Number of position specifications of an fuel element
MXYZC	4	C&H	Number of position specification of a control element
MXISO	110	C&H	Number of depleting nuclides on the composition table (caseDNxT)
MXNUC	110	C&H	Number of depleting nuclides of which number densities are traced
MAXMEM	800	C&H	Number of members on MACRO file on PS organization
MAXMTN	100	C&H	Number of materials
MAXX	250	H	Number of Regions in X direction
MAXY	250	H	Number of Regions in Y direction
MAXZ	250	H	Number of Regions in Z direction
MXHVIS	25	H	Number of heavy nuclides for inventory calculation
MAXFT	50	H	Number of fuel element types
MAXKF	50	H	Number of nodes in Z direction in Z direction
MAXOT	50	H	Number of non-fuel element types
MAXK1	50	H	Number of depleting nodes
MAXWRK	500000	H	Work area size to store whole members on PDS file to convert into PS organization
MAXK1I		H	MAXK1I=MAXK1*MXISO
MAXK1H		H	MAXK1H=MAXK1*MXHVIS

* C: COREBN, H: HIST

6. Job Control Statements for COREBN

6.1 Shell-script for Creation of Executable

An example of shell-script is shown below for creation of an COREBN executable for the LINUX machine using g77/gcc compilers. The Fortran sources of SRAC part are commonly used by COREBN code.

```
#!/bin/csh
######################################################################
#
#  <<  Create COREBN executable  >>
#
######################################################################
#
   alias  rm    rm
   alias  cp    cp
   alias  cd    cd
   alias  mkdir mkdir
   alias  echo  echo
#
#---------- Set Directory Name and Compile Driver & Options
#
   set CRBN_DIR = /home/okumura/code/SRAC2K6/COREBN
   set SRAC_DIR = /home/okumura/code/SRAC2K6/SRAC
#
  set    F77     = g77
  set    FFLAGS   = "-fno-automatic -finit-local-zero -O2 -funroll-loops"
#
  set    CC      = gcc
  set    CFLAGS   = "-DPOSIX_C"
  set    LDFLAGS  = -static
#
#---------- Set Load Module Name & Directory Name of Include Statement
#
   set    LMN     = $CRBN_DIR/bin/CRBN.100m
   set    INC     = $CRBN_DIR/src/inc/crbn100m
#
#----------.Make Working Directory ------------------------------
#
   set    DATE    = `date +%Y.%m.%d.%H.%M.%S`
   set    WKDIR   = $CRBN_DIR/tmp/tmpCRBN.$DATE
   mkdir  $WKDIR
#
#---------- Copy Source Programs into Working Directory ----------
#
   cd $SRAC_DIR/src
   cp extnl/*.f      $WKDIR
   cp extnl/*.c      $WKDIR
   cp common/*.f     $WKDIR
   cp cit/*.f        $WKDIR
   cd $CRBN_DIR/src
   cp crbn/*.f       $WKDIR
   cp $INC/*         $WKDIR
#
```

```
#---------- Compile C programs ----------------
#
  cd $WKDIR
  $CC   -c $CFLAGS *.c
  echo "--- end compile process for C-programs ---"
#
#---------- Compile F programs ----------------
#
  $F77  -c $FFLAGS *.f
  echo "--- end compile process for Fortran-programs ---"
#
#---------- Link all objects --------
#
  $F77 -o $LMN $FFLAGS *.o $LDFLAGS
  echo "--- end linking process ---"
#
#---------- Remove Working Directory ----------
#
  cd ..
  rm -r $WKDIR
  echo "--- temporary directory deleted ---"
#
#---------- End Process -----------------------
```

Without use of the above shell-script, user can eacilly install the COREBN code by using an interactive installation command (@PunchMe) equipped in the COREBN files.

6.2 Shell-script for COREBN Execution

```
#!/bin/csh
#
######################################################################
#
#  <<  run COREBN  >>
#
######################################################################
#  sample problem CRBNgo : Execute COREBN
######################################################################
#
  alias  mkdir mkdir
  alias  cat   cat
  alias  rm    rm
  alias  cd    cd
#
#=============  Set by user ========================================================
#
#  LMN    : load module name of COREBN
#  ODR    : directory name in which all output data will be stored
#  HTO    : directory and file name of old history (read only)
#  HTN    : directory and file name of new history
#  PSX    : directory and file name of PS converted MACRO (read only)
#  CASE   : case name which is referred as names of output files
#  WKDR   : directory name in which scratch PS files will be made and deleted
#
  set CRBN_DIR = $HOME/code/SRAC2K6/COREBN
  set SMPL_DIR = $CRBN_DIR/smpl
```

```
   set LMN  = CRBN.100m
   set ODR  = $SMPL_DIR/X1D/outp
   set HTO  = $ODR/HIST.BO1C
   set HTN  = $ODR/HIST.EO1C
   set PSX  = $SMPL_DIR/macroPS.dat
   set CASE = CRBNgo
#
#============  Change if you like ============================================
#
   set LM   = $CRBN_DIR/bin/$LMN
   set DATE = `date +%Y.%m.%d.%H.%M.%S`
   set WKDR = $HOME/CRBNtmp.$CASE.$DATE
   mkdir $WKDR
#
   set OUTLST =  $ODR/$CASE.CFT06.$DATE
   setenv  fu99  $ODR/$CASE.CFT99.$DATE
   setenv  fu90  $PSX
   setenv  fu92  $HTO
   setenv  fu93  $HTN
#  setenv  fu09  $ODR/$CASE.FLUX.dat
#  setenv  fu32  $ODR/$CASE.POWR.dat
#  setenv  fu13  $ODR/$CASE.REST1.dat
#  setenv  fu98  $ODR/$CASE.REST2.dat
#
#============  Exec COREBN code with the following input data ============
#
cd $WKDR
cat - << END_DATA | $LM >& $OUTLST
*********************************
*   Input data for COREBN code   *
*********************************
Sample for 1-D Slab Core, Upper:Water(10cm)+Barrel(10cm)+Nozzle(10cm)
+Plenum(10cm)+Fuel(140cm)+Nozzle(10cm)+Barrel(25cm)+Water(10cm):Lower
1  1  14  0  3  960101  960201  0  0  2  1  0  1  1  900.  600.
0.0  240.0  720.0  0.0  / Block-3 : Period (hour)
4(1.44E-2)              / Block-4 : Power (MWt)=q"'(MW/cc)*140cm*1cm*1cm
** Loading of fuel elements
FUEL-001    5  5   1  1   1      -1    0   0  /  Block-5-1
FUEL-002    6  6   1  1   1      -1    0   0
  :
  :
024
  1 1.062E-3
999
END_DATA
#
#========  Remove scratch PS files ========================================
#
   cd $HOME
   rm -r $WKDR
```

6.3 Modification of Size of Arrays

The COREBN adopts variable length arrays for major calculation. However, several fixed length arrays are used such that CITATION part originally uses fixed length arrays. If modification of sizes of fixed length arrays is required, it is achieved by modifying the parameter values described in the include files and recompiling to create a new executable. The include files setting the standard values are under the following directory.

$CRBN_DIR/crbn/src/inc/usrinc/

where, $CRBN_DIR is the installed top directory : e.g. /home/okumura/COREBN

This directory involves the 12 files, as already described, commonly used with HIST code. The include files used by COREBN are the following three.

(1) MAINCINC

On MAINCINC file, parameter to define the total memory size of variable length arrays is described as 'PARAMETER (MXSIZE=7000000)'.

In case where this memory size is short, modify this size for remedy.

Parameter Name	Standard value	Content
MZSIZE	7,000,000	Total memory size of variable length arrays

(2) CRBPMINC

The contents of the include file CRBPMINC is already described in Sect.5.3 as it is commonly used with HIST. This file is used in the routines READ, ADRSET, CRBN01, and CRBN12.

(3) CITMINC

On CITMINC file, parameters to define upper limits of fixed length arrays related to CITATION part are defined. If number of meshes, regions, or zones exceeds the limit in CITATION calculation, modify the parameter values.

Parameter Name	Standard value	Content
INTEGER RG$, MS$,ZNE$,ZD$, WZ$		
RG$	310	Upper limit of number of regions in every direction
MS$	311	Upper limit of number of meshes in every direction
XNW$	1000	Upper limit of number of zones (it must be even number)
ZD$	1000	Work area
WZ$	1000	Work area

7. Sample Input for HIST and COREBN

Sample input data will be introduced for three virtual cores. Each sample data is there under the directory;

- $CRBN_DIR/smpl/X1D/ for 1D slab core,
- $CRBN_DIR/smpl/TRI2D/ for 2D core with triangular meshes
- $CRBN_DIR/smpl/XYZ3D/ for 3D core in Cartesian coordinates.

Before executions of HIST and COREBN for these sample problems, cell burn-up calculations must be carried out with SRAC to prepare necessary macroscopic cross-sections. The followings are shell-scripts to prepare the cross-sections, which are commonly used for the above three samples.

- $CRBN_DIR/smpl/makeXS/U1.sh, U2.sh, U3.sh : shell-scripts to execute cell burn-up calculation with SRAC for three different fuels.
- $CRBN_DIR/smpl/makeXS/PDStoPS.sh : shell-scripts to execute HIST to convert PDS(MACRO) file to PS file.

Standard output files obtained by JAEA are kept in the directory outp/ under each sample directory. (e.g. $CRBN_DIR/smpl/X1D/outp/CRBNgo.CFT99.Sample)

7.1 One-dimensional Core Burn-up Calculation (infinite slab)

The followings are sample inputs of HIST and COREBN for a one-dimensional infinite slab core shown in Fig.7.1-1. As one-dimensional geometry has no concept of node division, each axial division has to be treated as individual fuel element. Thermal power is given by average power density (102.81E-6 MW/cm^3)× 1cm× 1cm× effective core height (140cm)=1.44E-2MWt where unit lengths are assumed in transverse directions.

(1) Input of HIST code (Initialization of the history file and registration of fuel elements)

```
1   0  10                / Block-1-1
Sample for 1-Dimensional Slab Core
Burnup for 3.0w/o Enriched Fuel Assembly
1  21  0  0  6  70  7     / Block-1-3
2 2 2 3   14(2)  3 5 2   / Block-1-4
10.0 10.0 10.0 10.0   14(10.0)  10.0 25.0 10.0 / Block-1-5
H2ORA0H0   0   / Water Reflector (Background Material)
BARLA0B0   0   / Core Barel
UNZLA0U0   0   / Upper Nozule
PLNMA0P0   0   / Upper Gas Plenum
LNZLA0L0   0   / Lower Nozule
```

```
U3XXAX10   1   /  3.0w/o Fule
XU04 XU05 XU06 XU07 XU08 XNP7 XNP9 XPU8 XPU9 XPU0
XPU1 XPU2 XAM1 XAMG XAMM XAM3 XCM2 XCM3 XCM4 XCM5
XCM6
XKR3 XZR5 XNB5 XMO5 XTC9 XRU1 XRU3 XRH3 XRH5 XPD5
XPD7 XPD8 XAG7 XAG9 XI05 XXE1 XXE3 XXE5 XCS3 XCS4
XCS5 XCS7 XBA0 XLA0 XPR3 XND3 XND5 XND7 XND8 XPM7
XPMM XPMG XPM9 XSM7 XSM8 XSM9 XSM0 XSM1 XSM2 XEU3
XEU4 XEU5 XEU6 XGD4 XGD5 XGD6 XGD7 XGD8 XGD0       / Dep.Nuclide
XU05 XU06 XU08 XPU9 XPU0 XPU1 XPU2                  / Heavy Metal
*********************************************************************
2  10  11            /  Block-2-1
1                    /  Block-2-2  : Registration of Fuel Element Type
FUELTYP3  1  1       /  Block-2-3-1
10.0                 /  Block-2-3-2  : 10cm*1cm*1cm
7(0.0)               /  Block-2-3-3
U3XXAX10             /  Block-2-3-4
0                    /  End of Block-2
*********************************************************************
3  11  12            /  Block-3-1  : Registration of Non-Fuel Element Type
1                    /  Block-3-2  : Upper Core Barrel
UPER-BAR   1
BARLA0B0
1                    /  Block-3-2  : Lower Core Barrel
LOWR-BAR   1
BARLA0B0
1                    /  Block-3-2  : Upper Nozzle
UPER-NZL   1
UNZLA0U0
1                    /  Block-3-2  : Lower Nozzle
LOWR-NZL   1
LNZLA0L0
1                    /  Block-3-2  : Gass Plenum
GAS-PLNM   1
PLNMA0P0
0                    /  End of Block-3
*********************************************************************
4  12  20            /  Block-4-1  : Registration of Individual Fuel Element
1                    /  Block-4-2
FUEL-001    FUELTYP3   0   0   2(0.0)              /  Block-4-3-1
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)      /  Block-4-3-3
FUEL-002    FUELTYP3   1   1   2(0.0)
FUEL-003    FUELTYP3   1   1   2(0.0)
FUEL-004    FUELTYP3   1   1   2(0.0)
FUEL-005    FUELTYP3   1   1   2(0.0)
FUEL-006    FUELTYP3   1   1   2(0.0)
FUEL-007    FUELTYP3   1   1   2(0.0)
FUEL-008    FUELTYP3   1   1   2(0.0)
FUEL-009    FUELTYP3   1   1   2(0.0)
FUEL-010    FUELTYP3   1   1   2(0.0)
FUEL-011    FUELTYP3   1   1   2(0.0)
FUEL-012    FUELTYP3   1   1   2(0.0)
FUEL-013    FUELTYP3   1   1   2(0.0)
FUEL-014    FUELTYP3   1   1   2(0.0)
                    0   0   2(0.0)      /  End of Block-4
*********************************************************************
6   20   0   /  Block-6-1
8   20   0   /  Block-8-1
9   20   0   /  Block-9-1
0    0   0   /  End of All
```

(2) Input of COREBN

```
Sample for 1-D Slab Core, Upper:Water(10cm)+Barrel(10cm)+Nozzle(10cm)
+Plenum(10cm)+Fuel(140cm)+Nozzle(10cm)+Barrel(25cm)+Water(10cm):Lower
1  1  14  0  3  960101  960201  0  0  2  1  0  1  1  900.  600.
0.0  240.0  720.0  0.0  / Block-3 : Period (hour)
4(1.44E-2)            / Block-4 : Power (MWt)=q"'(MW/cc)*140cm*1cm*1cm
** Loading of fuel elements ****************************************
FUEL-001    5  5   1  1   1      -1    0   0  /  Block-5-1
FUEL-002    6  6   1  1   1      -1    0   0
FUEL-003    7  7   1  1   1      -1    0   0
FUEL-004    8  8   1  1   1      -1    0   0
FUEL-005    9  9   1  1   1      -1    0   0
FUEL-006   10 10   1  1   1      -1    0   0
FUEL-007   11 11   1  1   1      -1    0   0
FUEL-008   12 12   1  1   1      -1    0   0
FUEL-009   13 13   1  1   1      -1    0   0
FUEL-010   14 14   1  1   1      -1    0   0
FUEL-011   15 15   1  1   1      -1    0   0
FUEL-012   16 16   1  1   1      -1    0   0
FUEL-013   17 17   1  1   1      -1    0   0
FUEL-014   18 18   1  1   1      -1    0   0
** loading of non-fuel elements ************************************
H2ORA0H0           /  Block-7-1 : Background Material = Water Reflector
UPER-BAR           /  Block-7-2
  2  2   1  1    1  /  Block-7-3
  0  0   0  0    0  /  End of Block-7-3
UPER-NZL
  3  3   1  1    1
  0  0   0  0    0
GAS-PLNM
  4  4   1  1    1
  0  0   0  0    0
LOWR-NZL
 19 19   1  1    1
  0  0   0  0    0
LOWR-BAR
 20 20   1  1    1
  0  0   0  0    0
                /  End of Block-7
** Input for CITATION ************************************************
001
  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  1  0  0  0  0  0
  1  0  0  0  0  0  0  0  0  0  0  0  1  0  0  0  0  0  0  0  0  0  0  0
200                                                 30
1.5        0.001
003
  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0
  0.00010   0.00001
  0.0
024
  1 1.062E-3
999
```

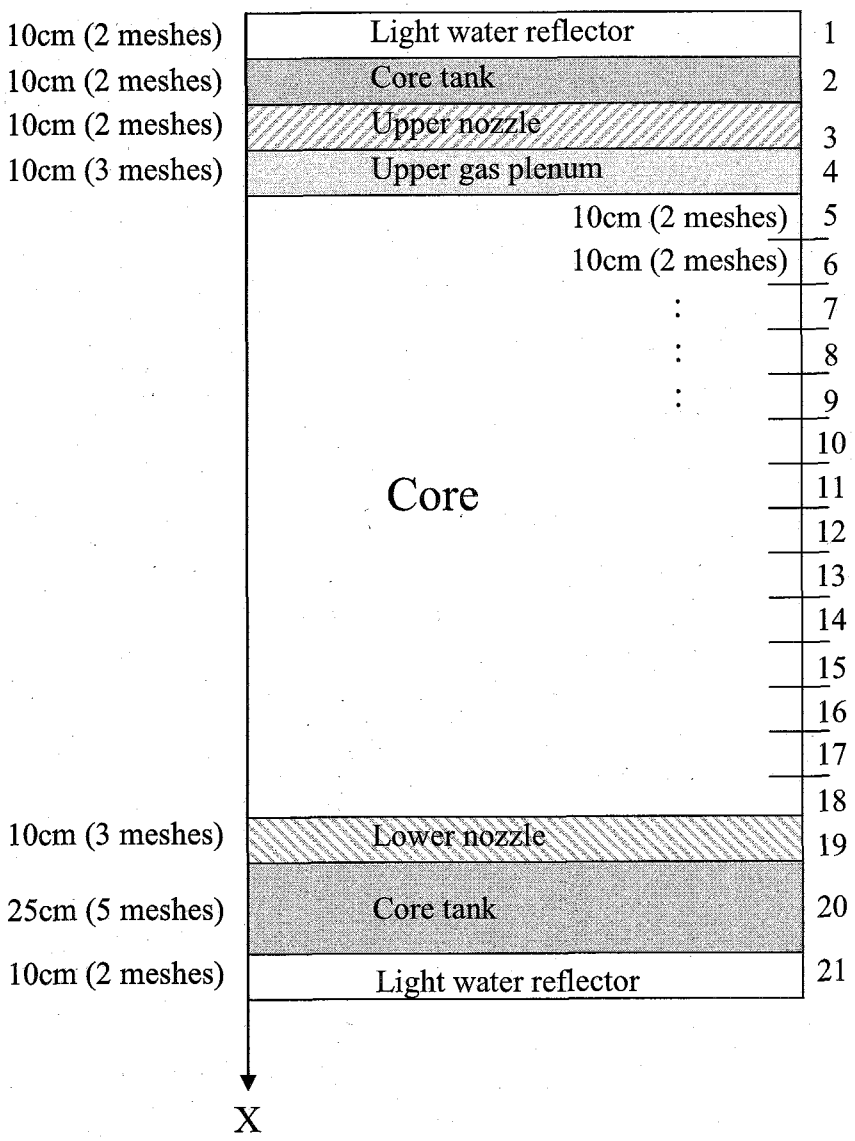


Fig.7.1-1 Burn-up calculation model for 1-dimensional slab core problem

7.2 Two-dimensional Core Burn-up Calculation (Triangular Mesh)

The followings are sample inputs of HIST and COREBN for a core composed by hexagonal fuel assemblies expressed by two-dimensional triangular meshes as shown in Fig.7.2-1. The initial loading core is composed by three fuel element types of different enrichments. It is surrounded by one layer (one assembly thickness) of light water reflector. The outer boundary is treated as vacuum boundary by placing perfect absorber (black absorber). Whereas the core calculation applies 60° rotational boundary condition, fuel elements occupying at least 1/3 of core as shown in Fig.7.2-2 have to be registered to continue burn-up calculations after the second cycle by applying refueling of 3 batch out-in system. Thermal power is given by the actual total power (3411 MWt) divided by the effective core height (223 cm) by assuming unit length of reactor height. This value is further divided by 6 because 1/6 core is under consideration.

(1) Input of HIST code (Initialization of the history file and registration of fuel elements)

```
1   0  10      / Block-1-1
Sample for 2-Dimensional 1/6 Core with 3 Types of Hexagonal Fuel
Assemblies Divided by Triangular Meshes. (Assembly Pitch=20.6cm)
10  34  17  0  5  70  7 / Block-1-3
34(1)                 / Block-1-4 NMESHX
34(5.94671)            / Block-1-5 XX
17(1)                 / Block-1-6 NMESHY
17(11.89342)           / Block-1-7 YY
H2ORA0H0  0   / Radial Water Reflector
BLACKABS  3   / Black Absorber (Background Material)
U3XXAX10  1   / 3.0w/o Fuel
U2XXAX10  1   / 2.0w/o Fuel
U1XXAX10  1   / 1.0w/o Fuel
XU04 XU05 XU06 XU07 XU08 XNP7 XNP9 XPU8 XPU9 XPU0
XPU1 XPU2 XAM1 XAMG XAMM XAM3 XCM2 XCM3 XCM4 XCM5
XCM6
XKR3 XZR5 XNB5 XMO5 XTC9 XRU1 XRU3 XRH3 XRH5 XPD5
XPD7 XPD8 XAG7 XAG9 XI05 XXE1 XXE3 XXE5 XCS3 XCS4
XCS5 XCS7 XBA0 XLA0 XPR3 XND3 XND5 XND7 XND8 XPM7
XPMM XPMG XPM9 XSM7 XSM8 XSM9 XSM0 XSM1 XSM2 XEU3
XEU4 XEU5 XEU6 XGD4 XGD5 XGD6 XGD7 XGD8 XGD0      / Block-1-11 Dep.Nuclide
XU05 XU06 XU08 XPU9 XPU0 XPU1 XPU2                 / Block-1-12 Heavy Metal
************************************************************************
2  10  11      / Block-2-1 (1.0w/o Fuel Element Type)
1              / Block-2-2
FUELTYP1  1  1  / Block-2-3-1
367.5           / Block-2-3-2  (SQRT(3)/2*P*P,P=20.6cm)
7(0.0)          / Block-2-3-3
U1XXAX10        / Block-2-3-4
1              / Block-2-1 (2.0w/o Fuel Element Type)
FUELTYP2  1  1
367.5
7(0.0)
U2XXAX10
1              / Block-2-1 (3.0w/o Fuel Element Type)
FUELTYP3  1  1
367.5
```

```
7(0.0)
U3XXAX10
0              / End of Block-3
*********************************************************************
3  11  12       / Block-3-1 (Non-Fuel Element Type)
1              / Block-3-2
REFLECTR  1     / Block-3-3-1
H2ORA0H0        / Block-3-3-2
0              / End of Block-3
*********************************************************************
4  12  20       / Block-4-1 (Fuel Elements)
1              / Block-4-2
FUELA001    FUELTYP1  0 0 2(0.0)          / Block-4-3-1 (Type-1)
0.0  7.081E-5  2(0.0)  6.921E-3  65(0.0)  / Block-4-3-3
FUELA002    FUELTYP1  1 1 2(0.0)
FUELA003    FUELTYP1  1 1 2(0.0)
FUELA004    FUELTYP1  1 1 2(0.0)
FUELA005    FUELTYP1  1 1 2(0.0)
FUELA006    FUELTYP1  1 1 2(0.0)
FUELA007    FUELTYP1  1 1 2(0.0)
FUELA008    FUELTYP1  1 1 2(0.0)
FUELA009    FUELTYP1  1 1 2(0.0)
FUELA010    FUELTYP1  1 1 2(0.0)
FUELA011    FUELTYP1  1 1 2(0.0)
FUELA012    FUELTYP1  1 1 2(0.0)
FUELA013    FUELTYP1  1 1 2(0.0)
FUELA014    FUELTYP1  1 1 2(0.0)
FUELA015    FUELTYP1  1 1 2(0.0)
FUELA016    FUELTYP1  1 1 2(0.0)
FUELA017    FUELTYP1  1 1 2(0.0)
FUELA018    FUELTYP1  1 1 2(0.0)
FUELA019    FUELTYP1  1 1 2(0.0)
FUELA020    FUELTYP1  1 1 2(0.0)
FUELA021    FUELTYP1  1 1 2(0.0)
FUELA022    FUELTYP1  1 1 2(0.0)
FUELA023    FUELTYP1  1 1 2(0.0)
FUELA024    FUELTYP1  1 1 2(0.0)
FUELA025    FUELTYP1  1 1 2(0.0)
FUELA026    FUELTYP1  1 1 2(0.0)
FUELA027    FUELTYP1  1 1 2(0.0)
FUELA028    FUELTYP1  1 1 2(0.0)
FUELA029    FUELTYP1  1 1 2(0.0)
FUELA030    FUELTYP1  1 1 2(0.0)
FUELA031    FUELTYP1  1 1 2(0.0)
FUELA032    FUELTYP1  1 1 2(0.0)
FUELA033    FUELTYP1  1 1 2(0.0)
*********************************
FUELB001    FUELTYP2  0 0 2(0.0)          / Block-4-3-1 (Type-2)
0.0  1.416E-4  2(0.0)  6.851E-3  65(0.0)  / Block-4-3-3
FUELB002    FUELTYP2  1 1 2(0.0)
FUELB003    FUELTYP2  1 1 2(0.0)
FUELB004    FUELTYP2  1 1 2(0.0)
FUELB005    FUELTYP2  1 1 2(0.0)
FUELB006    FUELTYP2  1 1 2(0.0)
FUELB007    FUELTYP2  1 1 2(0.0)
FUELB008    FUELTYP2  1 1 2(0.0)
FUELB009    FUELTYP2  1 1 2(0.0)
FUELB010    FUELTYP2  1 1 2(0.0)
FUELB011    FUELTYP2  1 1 2(0.0)
FUELB012    FUELTYP2  1 1 2(0.0)
FUELB013    FUELTYP2  1 1 2(0.0)
FUELB014    FUELTYP2  1 1 2(0.0)
```

```
FUELB015    FUELTYP2  1 1 2(0.0)
FUELB016    FUELTYP2  1 1 2(0.0)
FUELB017    FUELTYP2  1 1 2(0.0)
FUELB018    FUELTYP2  1 1 2(0.0)
FUELB019    FUELTYP2  1 1 2(0.0)
FUELB020    FUELTYP2  1 1 2(0.0)
FUELB021    FUELTYP2  1 1 2(0.0)
FUELB022    FUELTYP2  1 1 2(0.0)
FUELB023    FUELTYP2  1 1 2(0.0)
FUELB024    FUELTYP2  1 1 2(0.0)
FUELB025    FUELTYP2  1 1 2(0.0)
FUELB026    FUELTYP2  1 1 2(0.0)
FUELB027    FUELTYP2  1 1 2(0.0)
FUELB028    FUELTYP2  1 1 2(0.0)
FUELB029    FUELTYP2  1 1 2(0.0)
FUELB030    FUELTYP2  1 1 2(0.0)
FUELB031    FUELTYP2  1 1 2(0.0)
FUELB032    FUELTYP2  1 1 2(0.0)
FUELB033    FUELTYP2  1 1 2(0.0)
FUELB034    FUELTYP2  1 1 2(0.0)
*********************************
                  0 0 2(0.0)     / End of Block-4
*************************************************************************
6  20  0       / Block-6-1 : Print Geometry & Dep. Nuclide
8  20  0       / Block-8-1 : Print Fuel Element Type Information
9  20  0       / Block-9-1 : Print Non-Fuel Element Type Information
10 20  0       / Block-10-1: Print Individual Fuel Element Information
FUELA001 2
FUELB001 2
FUELC001 2
       0      / End of Block-10
0  0  0        / End of All
```

(2) Input of COREBN code (Initial loading core)

```
Sample for 2-D Triangular Mesh (HCLWR)
The First Cycle
1 1 101 0 4   000001 000335 0 1 2   1 0 3 1   900. 600. / Block-2
1 1 5 5 5                / Block-2-2  Benoist-D for Fuel
1200. 1200. 2400. 3240. 0.0 / Block-3 Period=(0),50,100,200,335 Day
5(2.549327)               / Block-4 Power=3411MW/6/223cm
** Loading of fuel elements ***************************************
FUELA001   34 34  17 17  1    -1  6  0    / 1  (1.0w/o Fuel)
FUELA002  -32 34  16 16  1    -1  1  0    / 2
          31 33  17 17
FUELA021  -32 34  16 16  1    -1  1  1    / 3
          31 33  17 17
FUELA003   28 30  17 17  1    -1  2  0    / 4
FUELA004  -30 32  15 15  1    -1  1  0    / 5
          29 31  16 16
FUELA022  -30 32  15 15  1    -1  1  1    / 6
          29 31  16 16
FUELA005  -34 34  14 14  1    -1  2  0    / 7
          33 34  15 15
FUELA006  -26 28  16 16  1    -1  1  0    / 8
          25 27  17 17
FUELA023  -26 28  16 16  1    -1  1  1    / 9
          25 27  17 17
```

```
FUELA007  -28 30  14 14  1    -1  1  0    /10
          27 29  15 15
FUELA024  -28 30  14 14  1    -1  1  1    /11
          27 29  15 15
FUELA008  -32 34  13 13  1    -1  1  0    /12
          31 33  14 14
FUELA025  -32 34  13 13  1    -1  1  1    /13
          31 33  14 14
FUELA009   22 24  17 17  1    -1  2  0    /14
FUELA010  -24 26  15 15  1    -1  1  0    /15
          23 25  16 16
FUELA026  -24 26  15 15  1    -1  1  1    /16
          23 25  16 16
FUELA011  -26 28  13 13  1    -1  1  0    /17
          25 27  14 14
FUELA027  -26 28  13 13  1    -1  1  1    /18
          25 27  14 14
FUELA012  -30 32  12 12  1    -1  1  0    /19
          29 31  13 13
FUELA028  -30 32  12 12  1    -1  1  1    /20
          29 31  13 13
FUELA013  -34 34  11 11  1    -1  2  0    /21
          33 34  12 12
FUELA014  -20 22  16 16  1    -1  1  0    /22
          19 21  17 17
FUELA029  -20 22  16 16  1    -1  1  1    /23
          19 21  17 17
FUELA015  -22 24  14 14  1    -1  1  0    /24
          21 23  15 15
FUELA030  -22 24  14 14  1    -1  1  1    /25
          21 23  15 15
FUELA016  -24 26  12 12  1    -1  1  0    /26
          23 25  13 13
FUELA031  -24 26  12 12  1    -1  1  1    /27
          23 25  13 13
FUELA017  -28 30  11 11  1    -1  1  0    /28
          27 29  12 12
FUELA032  -28 30  11 11  1    -1  1  1    /29
          27 29  12 12
FUELA018  -32 34  10 10  1    -1  1  0    /30
          31 33  11 11
FUELA033  -32 34  10 10  1    -1  1  1    /31
          31 33  11 11
FUELA019   16 18  17 17  1    -1  2  0    /32
FUELA020  -34 34   8  8  1    -1  2  0    /33
          33 34   9  9
*****************************************
FUELB001  -18 20  15 15  1    -1  1  0    / 1  (2.0w/o Fuel)
          17 19  16 16
FUELB019  -18 20  15 15  1    -1  1  1    / 2
          17 19  16 16
FUELB002  -20 22  13 13  1    -1  1  0    / 3
          19 21  14 14
FUELB020  -20 22  13 13  1    -1  1  1    / 4
          19 21  14 14
FUELB003  -22 24  11 11  1    -1  1  0    / 5
          21 23  12 12
FUELB021  -22 24  11 11  1    -1  1  1    / 6
          21 23  12 12
FUELB004  -26 28  10 10  1    -1  1  0    / 7
          25 27  11 11
FUELB022  -26 28  10 10  1    -1  1  1    / 8
```

```
          25 27  11 11
FUELB005  -30 32   9  9  1    -1  1  0    / 9
          29 31  10 10
FUELB023  -30 32   9  9  1    -1  1  1    /10
          29 31  10 10
FUELB006  -14 16  16 16  1    -1  1  0    /11
          13 15  17 17
FUELB024  -14 16  16 16  1    -1  1  1    /12
          13 15  17 17
FUELB007  -16 18  14 14  1    -1  1  0    /13
          15 17  15 15
FUELB025  -16 18  14 14  1    -1  1  1    /14
          15 17  15 15
FUELB008  -18 20  12 12  1    -1  1  0    /15
          17 19  13 13
FUELB026  -18 20  12 12  1    -1  1  1    /16
          17 19  13 13
FUELB009  -20 22  10 10  1    -1  1  0    /17
          19 21  11 11
FUELB027  -20 22  10 10  1    -1  1  1    /18
          19 21  11 11
FUELB010  -24 26   9  9  1    -1  1  0    /19
          23 25  10 10
FUELB028  -24 26   9  9  1    -1  1  1    /20
          23 25  10 10
FUELB011  -28 30   8  8  1    -1  1  0    /21
          27 29   9  9
FUELB029  -28 30   8  8  1    -1  1  1    /22
          27 29   9  9
FUELB012  -32 34   7  7  1    -1  1  0    /23
          31 33   8  8
FUELB030  -32 34   7  7  1    -1  1  1    /24
          31 33   8  8
FUELB013   10 12  17 17  1    -1  2  0    /25
FUELB014  -12 14  15 15  1    -1  1  0    /26
          11 13  16 16
FUELB031  -12 14  15 15  1    -1  1  1    /27
          11 13  16 16
FUELB015  -14 16  13 13  1    -1  1  0    /28
          13 15  14 14
FUELB032  -14 16  13 13  1    -1  1  1    /29
          13 15  14 14
FUELB016  -26 28   7  7  1    -1  1  0    /30
          25 27   8  8
FUELB033  -26 28   7  7  1    -1  1  1    /31
          25 27   8  8
FUELB017  -30 32   6  6  1    -1  1  0    /32
          29 31   7  7
FUELB034  -30 32   6  6  1    -1  1  1    /33
          29 31   7  7
FUELB018  -34 34   5  5  1    -1  2  0    /34
          33 34   6  6
*****************************************
FUELC001  -16 18  11 11  1    -1  1  0    / 1  (3.0w/o Fuel)
          15 17  12 12
FUELC019  -16 18  11 11  1    -1  1  1    / 2
          15 17  12 12
FUELC002  -18 20   9  9  1    -1  1  0    / 3
          17 19  10 10
FUELC020  -18 20   9  9  1    -1  1  1    / 4
          17 19  10 10
FUELC003  -22 24   8  8  1    -1  1  0    / 5
```

```
          21 23   9  9
FUELC021  -22 24   8  8  1    -1  1  1     / 6
          21 23   9  9
FUELC004   -8 10  16 16  1    -1  1  0     / 7
           7  9  17 17
FUELC022   -8 10  16 16  1    -1  1  1     / 8
           7  9  17 17
FUELC005  -10 12  14 14  1    -1  1  0     / 9
           9 11  15 15
FUELC023  -10 12  14 14  1    -1  1  1     /10
           9 11  15 15
FUELC006  -12 14  12 12  1    -1  1  0     /11
          11 13  13 13
FUELC024  -12 14  12 12  1    -1  1  1     /12
          11 13  13 13
FUELC007  -14 16  10 10  1    -1  1  0     /13
          13 15  11 11
FUELC025  -14 16  10 10  1    -1  1  1     /14
          13 15  11 11
FUELC008  -16 18   8  8  1    -1  1  0     /15
          15 17   9  9
FUELC026  -16 18   8  8  1    -1  1  1     /16
          15 17   9  9
FUELC009  -20 22   7  7  1    -1  1  0     /17
          19 21   8  8
FUELC027  -20 22   7  7  1    -1  1  1     /18
          19 21   8  8
FUELC010  -24 26   6  6  1    -1  1  0     /19
          23 25   7  7
FUELC028  -24 26   6  6  1    -1  1  1     /20
          23 25   7  7
FUELC011  -28 30   5  5  1    -1  1  0     /21
          27 29   6  6
FUELC029  -28 30   5  5  1    -1  1  1     /22
          27 29   6  6
FUELC012  -32 34   4  4  1    -1  1  0     /23
          31 33   5  5
FUELC030  -32 34   4  4  1    -1  1  1     /24
          31 33   5  5
FUELC013    4  6  17 17  1    -1  2  0     /25
FUELC014   -6  8  15 15  1    -1  1  0     /26
           5  7  16 16
FUELC031   -6  8  15 15  1    -1  1  1     /27
           5  7  16 16
FUELC015   -8 10  13 13  1    -1  1  0     /28
           7  9  14 14
FUELC032   -8 10  13 13  1    -1  1  1     /29
           7  9  14 14
FUELC016  -26 28   4  4  1    -1  1  0     /30
          25 27   5  5
FUELC033  -26 28   4  4  1    -1  1  1     /31
          25 27   5  5
FUELC017  -30 32   3  3  1    -1  1  0     /32
          29 31   4  4
FUELC034  -30 32   3  3  1    -1  1  1     /33
          29 31   4  4
FUELC018  -34 34   2  2  1    -1  2  0     /34
          33 34   3  3                     / Total 101 Bandles(2/6 Core)
** loading of non-fuel elements ***************************************
BLACKABS  / Block-7-1 Background Material (Black-Absorber)
REFLECTR  / Block-7-2 Radial Reflector
32 34  1  1  1
```

```
28 33  2  2  1
24 29  3  3  1
23 25  4  4  1
22 24  5  5  1
18 23  6  6  1
14 19  7  7  1
13 15  8  8  1
12 14  9  9  1
11 13 10 10  1
10 12 11 11  1
 6 11 12 12  1
 5  7 13 13  1
 4  6 14 14  1
 3  5 15 15  1
 2  4 16 16  1
 1  3 17 17  1
 0  0  0  0  0 / End of Block-7-3
            / End of Block-7
** Input for CITATION ************************************************
001
  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  1  0  0  0  0  0
  1  0  0  0  0  0  0  0  1  0  0  1  1  0  0  0  0  0  0  0  0  0  1
300                                                 25
2.0         0.001
003
  0  0  0  0  0  0  0  0  0  0  0  0  2  2  0  0  2  0  0  0  0  0  0  0
0.00001     0.00001
0.0
024
  1 1.985E-4
999
```

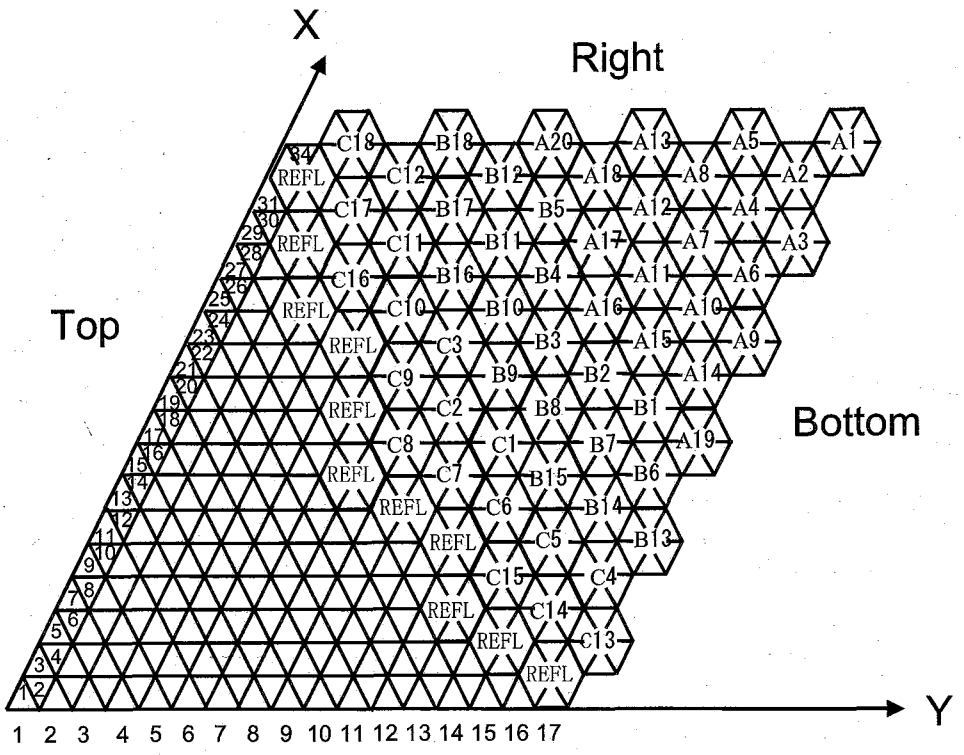


Fig.7.2-1 Burn-up calculation model for 2-dimensional core problem (triangular mesh division)

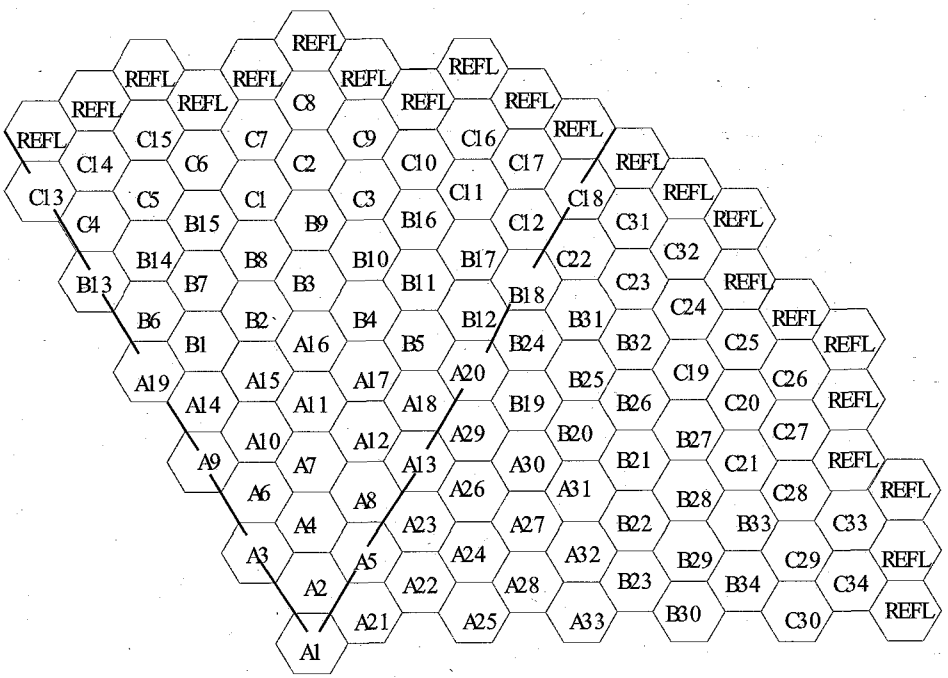


Fig.7.2-2 Burn-up calculation model for 2-dimensional core problem (registration of fuel elements)

7.3 Three-dimensional Core Burn-up Calculation (X-Y-Z)

The followings are sample inputs of HIST and COREBN for a core expressed by three-dimensional Cartesian coordinate system shown in Fig.7.3-1. While the initial loading core (the first cycle) can be solved on a quarter core, because the refueling after the second cycle becomes asymmetric, a full core model is used. The fuel element type consists of upper nozzle part, upper gas plenum part, fuel part of 14 nodes, and lower nozzle part, to simulate actual fuel assembly. Upper and lower core tanks, radial steel reflector are treated as non-fuel element types, respectively. Light water is used as background material. For the sake of convenience, Light water of upper and lower reflector, and core tank may be incorporated in the fuel element type.

(1) Input of HIST code (Initialization of the history file & registration of fuel elements)

```
1   0  10      / Block-1-1
Sample for 3-Dimensional (XYZ) Core
Full Core with 2-Types of Fuel Assemblies
11  11  11  21  8  70  7  / Block-1-3
11(5)          / Block-1-4
11(21.5)       / Block-1-5
11(5)          / Block-1-6
11(21.5)       / Block-1-7
2 2 2 3   14(2)  3 5 2 / Block-1-8
10.0 10.0 10.0 10.0   14(10.0)  10.0 25.0 10.0 / Block-1-9
H2ORA0H0   0   / Water Reflector (Background Material)
BARLA0B0   0   / Core Barel
UNZLA0U0   0   / Upper Nozule
PLNMA0P0   0   / Upper Gas Plenum
LNZLA0L0   0   / Lower Nozule
SUSRA0S0   0   / Radial SUS Reflector
U1XXAX10   1   / 1.0w/o UO2 Fuel
U3XXAX10   1   / 3.0w/o UO2 Fuel
XU04 XU05 XU06 XU07 XU08 XNP7 XNP9 XPU8 XPU9 XPU0
XPU1 XPU2 XAM1 XAMG XAMM XAM3 XCM2 XCM3 XCM4 XCM5
XCM6
XKR3 XZR5 XNB5 XMO5 XTC9 XRU1 XRU3 XRH3 XRH5 XPD5
XPD7 XPD8 XAG7 XAG9 XI05 XXE1 XXE3 XXE5 XCS3 XCS4
XCS5 XCS7 XBA0 XLA0 XPR3 XND3 XND5 XND7 XND8 XPM7
XPMM XPMG XPM9 XSM7 XSM8 XSM9 XSM0 XSM1 XSM2 XEU3
XEU4 XEU5 XEU6 XGD4 XGD5 XGD6 XGD7 XGD8 XGD0      / Block-1-11 Dep.Nuclide
XU05 XU06 XU08 XPU9 XPU0 XPU1 XPU2                 / Block-1-12 Heavy Metal
*****************************************************************
2  10  11      / Block-2-1
1             / Block-2-2  : Registration of Fuel Element Type(1.0w/o)
FUELTYP1 17 14 /  Block-2-3-1
14(4622.5)     /  Block-2-3-2  : 21.5cm*21.5cm*10cm
98(0.0)        /  Block-2-3-3    7 Heavy Metals * 14 Nodes
UNZLA0U0 PLNMA0P0 14(U1XXAX10) LNZLA0L0    /  Block-2-3-4
1             / Block-2-2  : Registration of Fuel Element Type(3.0w/o)
FUELTYP3 17 14 /  Block-2-3-1
14(4622.5)     /  Block-2-3-2  : 21.5cm*21.5cm*10cm
98(0.0)        /  Block-2-3-3    7 Heavy Metals * 14 Nodes
UNZLA0U0 PLNMA0P0 14(U3XXAX10) LNZLA0L0    /  Block-2-3-4
0             / End of Block-2
```

```
**********************************************************************
3  11  12      / Block-3-1  : Registration of Non-Fuel Element Type
1             / Block-3-2  : Upper Core Barrel
UPER-BAR   1
BARLA0B0
1             / Block-3-2  : Lower Core Barrel
LOWR-BAR   1
BARLA0B0
1             / Block-3-2  : Radial SUS Reflector
RADL-SUS  17
17(SUSRA0S0)
0             / End of Block-3
**********************************************************************
4  12  20      / Block-4-1  : Registration of Individual Fuel Element
1             / Block-4-2
FUEL1-01    FUELTYP1   0   0   2(0.0)      / Block-4-3-1
14(0.0 0.0)                               / Block-4-3-2
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
0.0  1.416E-04  0.0  0.0  6.851E-03  65(0.0)
FUEL1-02    FUELTYP1   1   1   2(0.0)      / Block-4-3-1
FUEL1-03    FUELTYP1   1   1   2(0.0)
FUEL1-04    FUELTYP1   1   1   2(0.0)
FUEL1-05    FUELTYP1   1   1   2(0.0)
FUEL1-06    FUELTYP1   1   1   2(0.0)
FUEL1-07    FUELTYP1   1   1   2(0.0)
FUEL1-08    FUELTYP1   1   1   2(0.0)
FUEL1-09    FUELTYP1   1   1   2(0.0)
FUEL1-10    FUELTYP1   1   1   2(0.0)
FUEL1-11    FUELTYP1   1   1   2(0.0)
FUEL1-12    FUELTYP1   1   1   2(0.0)
FUEL1-13    FUELTYP1   1   1   2(0.0)
FUEL1-14    FUELTYP1   1   1   2(0.0)
FUEL1-15    FUELTYP1   1   1   2(0.0)
FUEL1-16    FUELTYP1   1   1   2(0.0)
FUEL1-17    FUELTYP1   1   1   2(0.0)
FUEL2-01    FUELTYP3   0   0   2(0.0)      / Block-4-3-1
14(0.0 0.0)                               / Block-4-3-2
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)
```

```
0.0  2.124E-04  0.0  0.0  6.781E-03  65(0.0)    /  Block-4-3-3
FUEL2-02    FUELTYP3   1   1   2(0.0)      /  Block-4-3-1
FUEL2-03    FUELTYP3   1   1   2(0.0)
FUEL2-04    FUELTYP3   1   1   2(0.0)
FUEL2-05    FUELTYP3   1   1   2(0.0)
FUEL2-06    FUELTYP3   1   1   2(0.0)
FUEL2-07    FUELTYP3   1   1   2(0.0)
FUEL2-08    FUELTYP3   1   1   2(0.0)
FUEL2-09    FUELTYP3   1   1   2(0.0)
FUEL2-10    FUELTYP3   1   1   2(0.0)
FUEL2-11    FUELTYP3   1   1   2(0.0)
FUEL2-12    FUELTYP3   1   1   2(0.0)
FUEL2-13    FUELTYP3   1   1   2(0.0)
FUEL2-14    FUELTYP3   1   1   2(0.0)
FUEL2-15    FUELTYP3   1   1   2(0.0)
FUEL2-16    FUELTYP3   1   1   2(0.0)
FUEL2-17    FUELTYP3   1   1   2(0.0)
FUEL2-18    FUELTYP3   1   1   2(0.0)
FUEL2-19    FUELTYP3   1   1   2(0.0)
FUEL2-20    FUELTYP3   1   1   2(0.0)
                    0   0   2(0.0)      /  End of Block-4
0    0   0   /  End of All
```

(2) Input of COREBN code (Initial loading core)

```
Sample for 3-D Full Core Burnup Calculation
Cycle No. : 1
1  1  37  0  3  960101  960201  0  0  2  1  0  1  1  900.  600.
0.0  24.0  720.0  0.0  /  Block-3 : Period (hour)
4(246.0)              /  Block-4 : Power (MWt)
** Loading of fuel elements ******************************************
FUEL1-01    6  6   3  3  -1      -1    0   0  /  Block-5-1
FUEL1-02    6  6   4  4  -1      -1    0   0
FUEL1-03    5  5   5  5  -1      -1    0   0
FUEL1-04    6  6   5  5  -1      -1    0   0
FUEL1-05    7  7   5  5  -1      -1    0   0
FUEL1-06    3  3   6  6  -1      -1    0   0
FUEL1-07    4  4   6  6  -1      -1    0   0
FUEL1-08    5  5   6  6  -1      -1    0   0
FUEL1-09    6  6   6  6  -1      -1    0   0
FUEL1-10    7  7   6  6  -1      -1    0   0
FUEL1-11    8  8   6  6  -1      -1    0   0
FUEL1-12    9  9   6  6  -1      -1    0   0
FUEL1-13    5  5   7  7  -1      -1    0   0
FUEL1-14    6  6   7  7  -1      -1    0   0
FUEL1-15    7  7   7  7  -1      -1    0   0
FUEL1-16    6  6   8  8  -1      -1    0   0
FUEL1-17    6  6   9  9  -1      -1    0   0
FUEL2-01    7  7   3  3  -1      -1    0   0
FUEL2-02    5  5   3  3  -1      -1    0   0
FUEL2-03    4  4   5  5  -1      -1    0   0
FUEL2-04    4  4   4  4  -1      -1    0   0
FUEL2-05    7  7   4  4  -1      -1    0   0
FUEL2-06    3  3   7  7  -1      -1    0   0
FUEL2-07    3  3   5  5  -1      -1    0   0
FUEL2-08    4  4   8  8  -1      -1    0   0
FUEL2-09    5  5   4  4  -1      -1    0   0
FUEL2-10    8  8   4  4  -1      -1    0   0
```

```
FUEL2-11    9  9   7  7  -1      -1    0   0
FUEL2-12    9  9   5  5  -1      -1    0   0
FUEL2-13    5  5   8  8  -1      -1    0   0
FUEL2-14    8  8   8  8  -1      -1    0   0
FUEL2-15    8  8   7  7  -1      -1    0   0
FUEL2-16    7  7   9  9  -1      -1    0   0
FUEL2-17    5  5   9  9  -1      -1    0   0
FUEL2-18    8  8   5  5  -1      -1    0   0
FUEL2-19    4  4   7  7  -1      -1    0   0
FUEL2-20    7  7   8  8  -1      -1    0   0
** loading of non-fuel elements ***************************************
H2ORA0H0           /  Block-7-1 : Background Material = Water Reflector
UPER-BAR           /  Block-7-2
  4  8   2  2    0  /  Block-7-3
  3  9   3  3    0
  2 10   4  8    0
  3  9   9  9    0
  4  8  10 10    0
  0  0   0  0    0  /  End of Block-7-3
LOWR-BAR           /  Block-7-2
  4  8   2  2  -18  /  Block-7-3
  3  9   3  3  -18
  2 10   4  8  -18
  3  9   9  9  -18
  4  8  10 10  -18
  0  0   0  0    0  /  End of Block-7-3
RADL-SUS           /  Block-7-2
  4  8   2  2   -1  /  Block-7-3
  3  4   3  3   -1
  8  9   3  3   -1
  2  3   4  4   -1
  9 10   4  4   -1
  2  2   5  7   -1
 10 10   5  7   -1
  2  3   8  8   -1
  9 10   8  8   -1
  4  8  10 10   -1
  0  0   0  0    0  /  End of Block-7-3
                 /  End of Block-7
** Input for CITATION ************************************************
001
  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  1  0  0  0  0  0
  1  0  0  0  0  0  0  0  0  0  0  0  1  0  0  0  0  0  0  0  0  0  0  1
900                                                  45
1.5         0.001
003
  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0  0
  0.00010   0.00001
  0.0
999
```

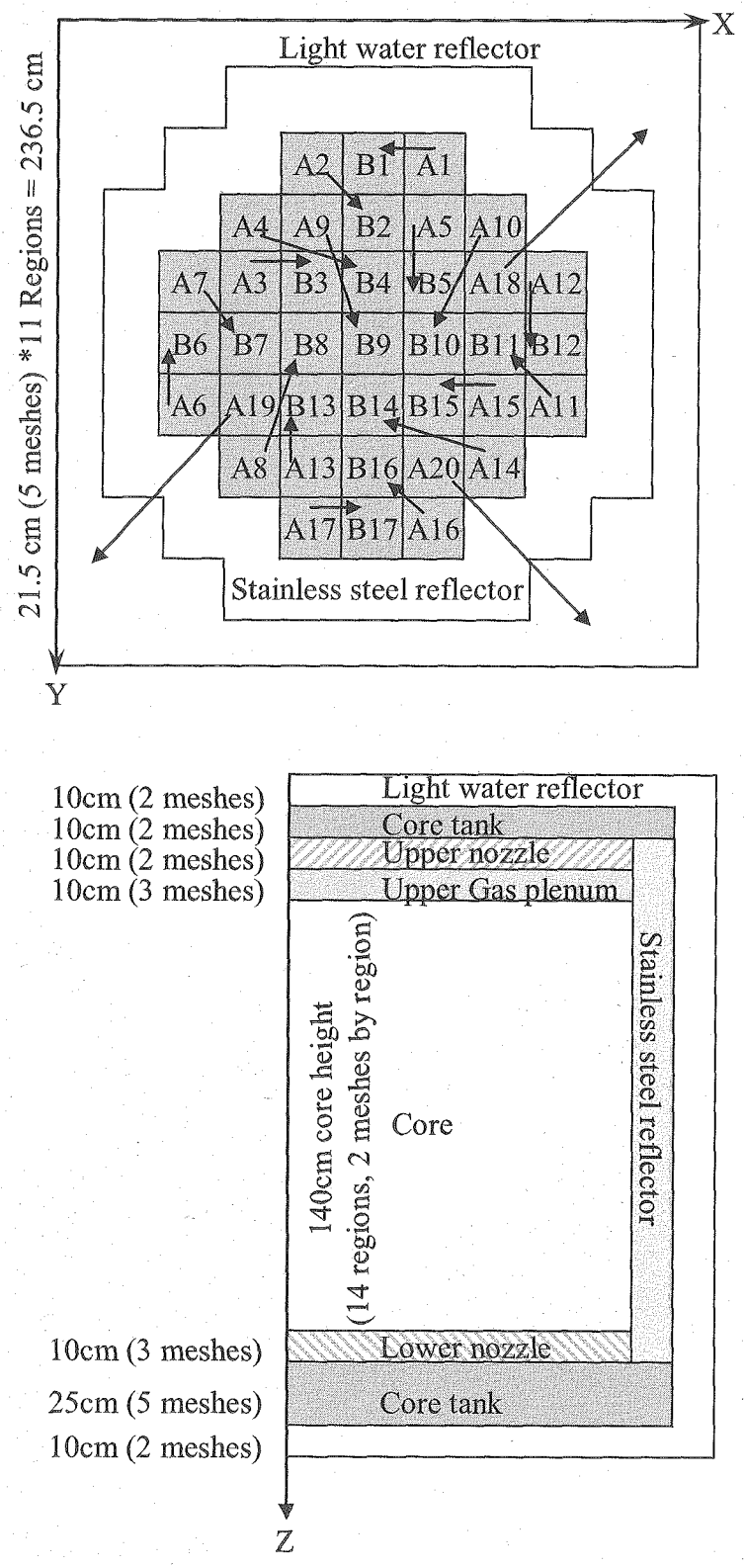


Fig.7.3-1 Burn-up calculation model for 3-dimensional core problem (X-Y-Z)

Acknowledgements

The author is grateful to Dr. Kiyonobu YAMASHITA, Tomoaki KATO, Yasuhiro ISHIKURO, Daisuke NIO for their encouragements and supports to write this report. The author also appreciate Yoshihiro NAKANO for his useful comments to improve COREBN and HIST codes.

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国際単位系（SI）

表１．SI 基本単位

基本量	SI 基本単位	
	名称	記号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd

表２．基本単位を用いて表されるSI組立単位の例

組立量	SI 基本単位	
	名称	記号
面積	平方メートル	m^2
体積	立法メートル	m^3
速さ，速度	メートル毎秒	m/s
加速度	メートル毎秒毎秒	m/s^2
波数	毎メートル	m^{-1}
密度（質量密度）	キログラム毎立法メートル	kg/m^3
質量体積（比体積）	立法メートル毎キログラム	m^3/kg
電流密度	アンペア毎平方メートル	A/m^2
磁界の強さ	アンペア毎メートル	A/m
（物質量の）濃度	モル毎立方メートル	mol/m^3
輝度	カンデラ毎平方メートル	cd/m^2
屈折率	（数の）1	1

表５．SI 接頭語

乗数	接頭語	記号	乗数	接頭語	記号
10^{24}	ヨタ	Y	10^{-1}	デシ	d
10^{21}	ゼタ	Z	10^{-2}	センチ	c
10^{18}	エクサ	E	10^{-3}	ミリ	m
10^{15}	ペタ	P	10^{-6}	マイクロ	μ
10^{12}	テラ	T	10^{-9}	ナノ	n
10^{9}	ギガ	G	10^{-12}	ピコ	p
10^{6}	メガ	M	10^{-15}	フェムト	f
10^{3}	キロ	k	10^{-18}	アト	a
10^{2}	ヘクト	h	10^{-21}	ゼプト	z
10^{1}	デカ	da	10^{-24}	ヨクト	y

表３．固有の名称とその独自の記号で表されるSI組立単位

組立量	SI 組立単位			
	名称	記号	他のSI単位による表し方	SI基本単位による表し方
平面角	ラジアン(a)	rad		$m \cdot m^{-1}=1$(b)
立体角	ステラジアン(a)	sr(c)		$m^2 \cdot m^{-2}=1$(b)
周波数	ヘルツ	Hz		s^{-1}
力	ニュートン	N		$m \cdot kg \cdot s^{-2}$
圧力，応力	パスカル	Pa	N/m^2	$m^{-1} \cdot kg \cdot s^{-2}$
エネルギー，仕事，熱量	ジュール	J	N・m	$m^2 \cdot kg \cdot s^{-2}$
工率，放射束	ワット	W	J/s	$m^2 \cdot kg \cdot s^{-3}$
電荷，電気量	クーロン	C		s・A
電位差（電圧），起電力	ボルト	V	W/A	$m^2 \cdot kg \cdot s^{-3} \cdot A^{-1}$
静電容量	ファラド	F	C/V	$m^{-2} \cdot kg^{-1} \cdot s^4 \cdot A^2$
電気抵抗	オーム	Ω	V/A	$m^2 \cdot kg \cdot s^{-3} \cdot A^{-2}$
コンダクタンス	ジーメンス	S	A/V	$m^{-2} \cdot kg^{-1} \cdot s^3 \cdot A^2$
磁束	ウエーバ	Wb	V・s	$m^2 \cdot kg \cdot s^{-2} \cdot A^{-1}$
磁束密度	テスラ	T	Wb/m^2	$kg \cdot s^{-2} \cdot A^{-1}$
インダクタンス	ヘンリー	H	Wb/A	$m^2 \cdot kg \cdot s^{-2} \cdot A^{-2}$
セルシウス温度	セルシウス度(d)	℃		K
光束	ルーメン	lm	cd・sr(c)	$m^2 \cdot m^{-2} \cdot cd=cd$
照度	ルクス	lx	lm/m^2	$m^2 \cdot m^{-4} \cdot cd=m^{-2} \cdot cd$
（放射性核種の）放射能	ベクレル	Bq		s^{-1}
吸収線量，質量エネルギー分与，カーマ	グレイ	Gy	J/kg	$m^2 \cdot s^{-2}$
線量当量，周辺線量当量，方向性線量当量，個人線量当量，組織線量当	シーベルト	Sv	J/kg	$m^2 \cdot s^{-2}$

(a)ラジアン及びステラジアンの使用は、同じ次元であっても異なった性質をもった量を区別するときの組立単位の表し方として利点がある。組立単位を形作るときのいくつかの用例は表４に示されている。
(b)実際には、使用する時には記号rad及びsrが用いられるが、習慣として組立単位としての記号“１”は明示されない。
(c)測光学では、ステラジアンの名称と記号srを単位の表し方の中にそのまま維持している。
(d)この単位は、例としてミリセルシウス度m℃のようにSI接頭語を伴って用いても良い。

表４．単位の中に固有の名称とその独自の記号を含むSI組立単位の例

組立量	SI 組立単位		
	名称	記号	SI 基本単位による表し方
粘度	パスカル秒	Pa・s	$m^{-1} \cdot kg \cdot s^{-1}$
力のモーメント	ニュートンメートル	N・m	$m^2 \cdot kg \cdot s^{-2}$
表面張力	ニュートン毎メートル	N/m	$kg \cdot s^{-2}$
角速度	ラジアン毎秒	rad/s	$m \cdot m^{-1} \cdot s^{-1}=s^{-1}$
角加速度	ラジアン毎平方秒	rad/s^2	$m \cdot m^{-1} \cdot s^{-2}=s^{-2}$
熱流密度，放射照度	ワット毎平方メートル	W/m^2	$kg \cdot s^{-3}$
熱容量，エントロピー	ジュール毎ケルビン	J/K	$m^2 \cdot kg \cdot s^{-2} \cdot K^{-1}$
質量熱容量（比熱容量），質量エントロピー	ジュール毎キログラム毎ケルビン	J/(kg・K)	$m^2 \cdot s^{-2} \cdot K^{-1}$
質量エネルギー（比エネルギー）	ジュール毎キログラム	J/kg	$m^2 \cdot s^{-2} \cdot K^{-1}$
熱伝導率	ワット毎メートル毎ケルビン	W/(m・K)	$m \cdot kg \cdot s^{-3} \cdot K^{-1}$
体積エネルギー	ジュール毎立方メートル	J/m^3	$m^{-1} \cdot kg \cdot s^{-2}$
電界の強さ	ボルト毎メートル	V/m	$m \cdot kg \cdot s^{-3} \cdot A^{-1}$
体積電荷	クーロン毎立方メートル	C/m^3	$m^{-3} \cdot s \cdot A$
電気変位	クーロン毎平方メートル	C/m^2	$m^{-2} \cdot s \cdot A$
誘電率	ファラド毎メートル	F/m	$m^{-3} \cdot kg^{-1} \cdot s^4 \cdot A^2$
透磁率	ヘンリー毎メートル	H/m	$m \cdot kg \cdot s^{-2} \cdot A^{-2}$
モルエネルギー	ジュール毎モル	J/mol	$m^2 \cdot kg \cdot s^{-2} \cdot mol^{-1}$
モルエントロピー，モル熱容量	ジュール毎モル毎ケルビン	J/(mol・K)	$m^2 \cdot kg \cdot s^{-2} \cdot K^{-1} \cdot mol^{-1}$
照射線量（X線及びγ線）	クーロン毎キログラム	C/kg	$kg^{-1} \cdot s \cdot A$
吸収線量率	グレイ毎秒	Gy/s	$m^2 \cdot s^{-3}$
放射強度	ワット毎ステラジアン	W/sr	$m^4 \cdot m^{-2} \cdot kg \cdot s^{-3}=m^2 \cdot kg \cdot s^{-3}$
放射輝度	ワット毎平方メートル毎ステラジアン	$W/(m^2 \cdot sr)$	$m^2 \cdot m^{-2} \cdot kg \cdot s^{-3}=kg \cdot s^{-3}$

表６．国際単位系と併用されるが国際単位系に属さない単位

名称	記号	SI 単位による値
分	min	1 min=60s
時	h	1h =60 min=3600 s
日	d	1 d=24 h=86400 s
度	°	1° =(π/180) rad
分	’	1’ =(1/60)° =(π/10800) rad
秒	”	1” =(1/60)’ =(π/648000) rad
リットル	l、L	1l=1 $dm^3=10^{-3}m^3$
トン	t	1t=10^3 kg
ネーパ	Np	1Np=1
ベル	B	1B=(1/2)ln10(Np)

表７．国際単位系と併用されこれに属さない単位でSI単位で表される数値が実験的に得られるもの

名称	記号	SI 単位であらわされる数値
電子ボルト	eV	$1eV=1.60217733(49) \times 10^{-19}J$
統一原子質量単位	u	$1u=1.6605402(10) \times 10^{-27}kg$
天文単位	ua	$1ua=1.49597870691(30) \times 10^{11}m$

表８．国際単位系に属さないが国際単位系と併用されるその他の単位

名称	記号	SI 単位であらわされる数値
海里		1海里=1852m
ノット		1ノット=1海里毎時=(1852/3600)m/s
アール	a	1a=1 $dam^2=10^2m^2$
ヘクタール	ha	1ha=1 $hm^2=10^4m^2$
バール	bar	1bar=0.1MPa=100kPa=1000hPa=10^5Pa
オングストローム	Å	1 Å=0.1nm=10^{-10}m
バーン	b	1b=100$fm^2=10^{-28}m^2$

表９．固有の名称を含むCGS組立単位

名称	記号	SI 単位であらわされる数値
エルグ	erg	1 erg=10^{-7} J
ダイン	dyn	1 dyn=10^{-5}N
ポアズ	P	1 P=1 dyn・s/cm^2=0.1Pa・s
ストークス	St	1 St =1$cm^2/s=10^{-4}m^2/s$
ガウス	G	1 G ^10^{-4}T
エルステッド	Oe	1 Oe ^(1000/4π)A/m
マクスウェル	Mx	1 Mx ^10^{-8}Wb
スチルブ	sb	1 sb =1$cd/cm^2=10^4cd/m^2$
ホト	ph	1 ph=10^4lx
ガル	Gal	1 Gal =1$cm/s^2=10^{-2}m/s^2$

表10．国際単位に属さないその他の単位の例

名称	記号	SI 単位であらわされる数値
キュリー	Ci	$1 Ci=3.7 \times 10^{10}Bq$
レントゲン	R	$1 R = 2.58 \times 10^{-4}C/kg$
ラド	rad	1 rad=1cGy=10^{-2}Gy
レム	rem	1 rem=1 cSv=10^{-2}Sv
X線単位		1X unit=1.002×10^{-4}nm
ガンマ	γ	1γ=1 nT=10^{-9}T
ジャンスキー	Jy	1 Jy=$10^{-26}W \cdot m^{-2} \cdot Hz^{-1}$
フェルミ		1 fermi=1 fm=10^{-15}m
メートル系カラット		1 metric carat = 200 mg = 2×10^{-4}kg
トル	Torr	1 Torr = (101 325/760) Pa
標準大気圧	atm	1 atm = 101 325 Pa
カロリー	cal	
ミクロン	μ	1 μ =1μm=10^{-6}m

（第７版，1998年改訂）