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MOSRA-SRAC is a lattice calculation module of the Modular code System for nuclear Reactor Analyses (MOSRA). This module performs the neutron transport calculation for various types of fuel elements including existing light water reactors, research reactors, etc. based on the collision probability method with a set of the 200-group cross-sections generated from the Japanese Evaluated Nuclear Data Library JENDL-4.0. It has also a function of the isotope generation and depletion calculation for up to 234 nuclides in each fuel material in the lattice. In these ways, MOSRA-SRAC prepares the burn-up dependent effective microscopic and macroscopic cross-section data to be used in core calculations.

Keywords: MOSRA-SRAC, Lattice Calculation, Collision Probability, Neutron Transport, Burn-up Calculation, JENDL-4.0, Effective Cross Section

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MOSRA-SRAC:モジュラー型原子炉解析コードシステム MOSRA の格子計算モジュール

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奥村 啓介

(2015年8月6日受理)

MOSRA-SRAC は、モジュラー型原子炉解析コードシステム(MOSRA)の格子計算モジュールである。本モジュールは、既存の軽水炉や研究炉などを含む多様な原子炉の燃料要素に対し、日本の評価済み核データライブラリ JENDL-4.0 から作成した 200 群断面積セットを用いた衝突確率法に基づく中性子輸送計算を行う機能を有する。また、格子内の各燃料物質中の最大 234 核種に対する核種生成崩壊計算を行う。これらにより、MOSRA-SRAC は、炉心計算で使用する燃焼依存の実効微視的断面積及び巨視的断面積を提供する。

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1. General Descriptions

1.1 Functions and Features

MOSRA-SRAC is a lattice calculation module of the Modular code System for nuclear Reactor Analyses MOSRA[1]. It is designed to produce effective microscopic and macroscopic cross-sections for the succeeding core calculation by other modules such as MOSRA-Light[2] based on the nodal diffusion method. MOSRA-SRAC was developed by uncoupling the lattice calculation function from the old integrated code system SRAC2006[3]. However, a number of additions and modifications to the functions have been made to establish the more powerful code system MOSRA. The features of MOSRA-SRAC are in the following;

- A neutron cross-section library based on the latest and well validated Japanese Evaluated Nuclear Data Library JENDL-4.0[4] is available for more than 400 nuclides.
- The routine of the collision probability method applicable to 16 types of geometries in Fig. 1.1-1 covers lattice calculations for most of existing reactors.
- The effective cross-sections by the conventional table look-up method based on the narrow resonance (NR) approximation can be replaced with those by an optional routine PEACO[3,5] which solves a multi-region lattice problem by the collision probability method using an almost continuous (hyper-fine) energy group structure for the resonance energy region. The interaction of resonances can be accurately treated by the PEACO routine.
- Arbitrary temperature of composite materials is allowed by the interpolation of resonance shielding factors and thermal scattering matrices. For the PEACO routine, Doppler broaden cross-sections in hyper-fine energy structure are internally calculated from the point-wise cross-sections at room temperature.
- The Dancoff correction factor required in the interpolation of the self-shielding factors of resonant nuclides is automatically calculated by the installed collision probability routines. The factor is given not for an absorber lump but for each constituent nuclide in the lattice which contains a resonant nuclide in two or more materials with different compositions[6].
- A doubly heterogeneous system can be solved by successive lattice calculations since homogenizing and collapsing of macroscopic cross-sections is carried out separately. Especially, the resonance absorption of which double heterogeneity effect should be solved simultaneously, can be treated as far as the microscopic lattice can be approximated by one of 1-dimensional

lattices[7]. This method is effective, for example, in a fuel assembly lattice of the high-temperature gas cooled reactor including many coated-particle fuels.

- The option for the lattice burn-up calculation provides burn-up dependent microscopic/ macroscopic cross-sections and the change of nuclide composition during burn-up by a series of procedures to get a neutron spectrum, to get one-group effective cross-sections and to calculate generation and incineration of nuclides at each burn-up step. It can treat a detailed burn-up chain model which includes 28 heavy nuclides and 201 fission products or burnable poisons.
- Spatial homogenizations and energy-group collapsing (from 200-groups until one-group) can be carried out any number of times, although some functions (e.g. use of PEACO, burn-up calculation) may be suppressed.
- The I/O data files for group cross-sections, neutron fluxes and so on are written in the common format called PDS (Partitioned Data Set) files among the modules of MOSRA. The information written by a module can be read by the succeeding core calculation modules. For the effective I/O access of many PDS files, a new module MOSRA-PonPon was developed and equipped in MOSRA-SRAC.
- MOSRA-SRAC can be executed on most of computers with the UNIX operating system or its similar ones like Linux or Cygwin.

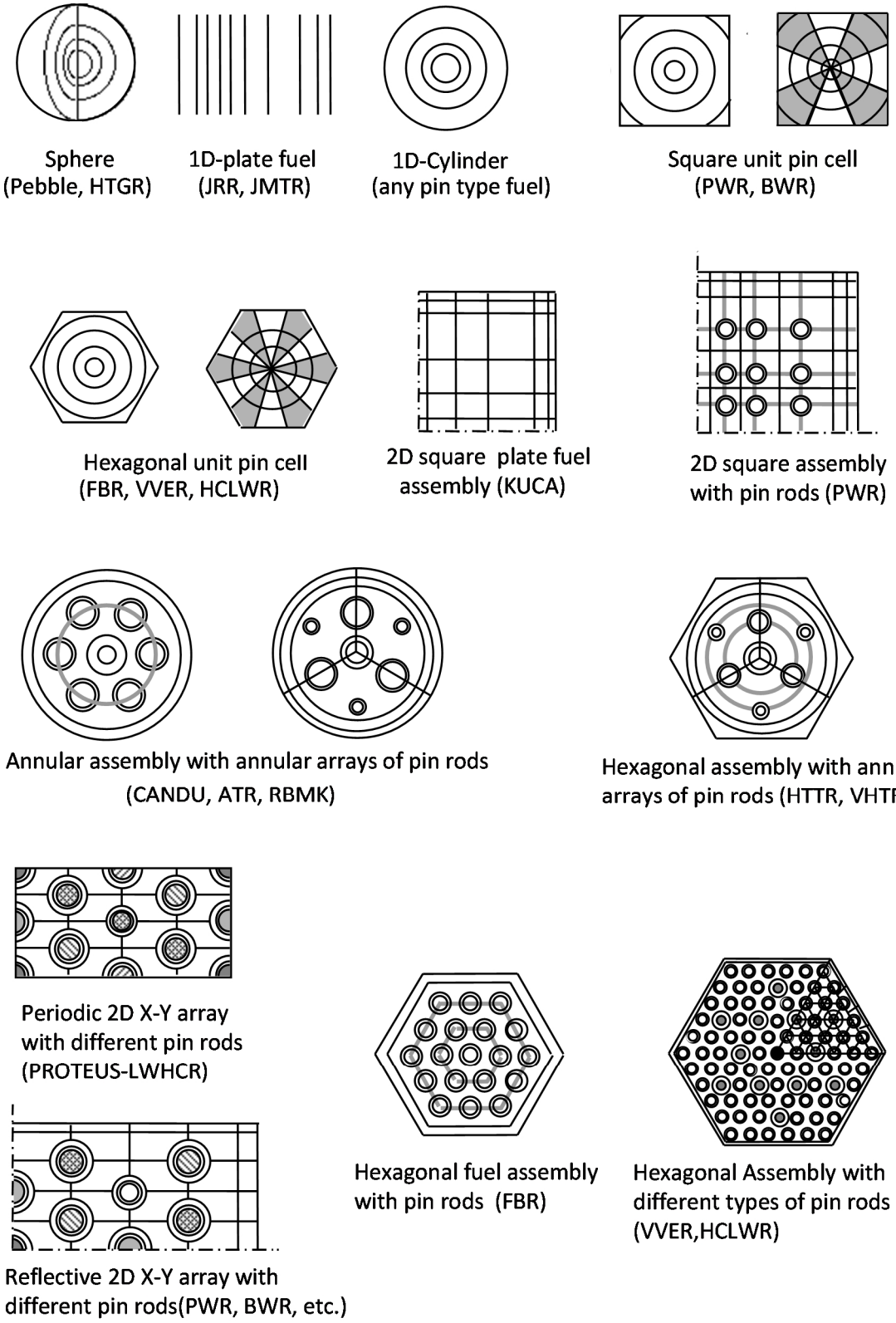


Fig.1.1-1 Lattice geometry available by MOSRA-SRAC

1.2 Data Libraries

The startup libraries of MOSRA-SRAC are organized by three cross-section libraries called Public Libraries. They consist of Public Fast Library to install fast group cross-sections, Public Thermal Library to install thermal group cross-sections and Public MCROSS Library to install point-wise cross-sections in the resonance energy region which is used by the PEACO routine.

The present system provides the Public Libraries produced from the recent Japanese Evaluated Nuclear Data Library JENDL-4.0. The Public Libraries provide the cross-sections which cover the energy region from 1.0×10^{-5} eV to 20 MeV for more than 400 nuclides required in neutronics calculation.

Before calculations, the User MCROSS Library is compiled from the Public MCROSS Library for saving the computer memory and having smooth data access. The User MCROSS Library contains the data of nuclides the user needs in his own calculation.

1.2.1 Energy Group Structure and Cut-off-energy

The energy group structure of the current Public Libraries consists of 200 groups (179 groups for fast and 82 groups for thermal energy regions, respectively, with 61 overlapping groups). The energy boundary of each group is shown in Table 1.2-1. The 200-group structure was determined with considerations as follows;

- Basic structure is referred from the 172-group structure of XMAS[8)] designed for thermal and intermediate energy reactor lattice calculations.
- Upper energy of all groups is 20 MeV, which is the maximum energy of the cross-sections evaluated in JENDL-4.0.
- The energy boundary must include the upper and lower boundaries of PEACO.
- The maximum lethargy width is less than or equal to 0.25.
- The energy boundary of the each fine-group in the energy region of PEACO must be equal to the boundary of PEACO.
- The structure of energy region from 1.5 keV to 300 keV was fined to represent the resonance windows of Fe and Na isotopes.

The Public Fast Library and the Public Thermal Library cover the cross-section data for neutron energy region $0.1523\ \text{eV} < E < 20$ MeV and $1.0\times10^{-5}\ \text{eV} < E < 4.00721$ eV, respectively. The thermal cut-off-energy is 4.00721 eV. The up-scattering of thermal neutron is considered up to the cut-off-energy. Optionally users can use only the Public Fast Library in 179-groups structure for fast reactors.

1.2.2 Public Fast Library

The fast neutron energy region for the Public Fast Library is defined as the range from 0.1523 eV to 20 MeV, although the lower energy boundary (thermal cut-off energy) for the fast neutron calculation is internally fixed as 4.00721 eV. The group constants in this energy region are arranged in the form of the cross-section set of the Bondarenko type; that is, the self-shielding factors are given for elastic scattering, elastic removal, capture, fission, and total cross-sections. The self-shielding factors are tabulated by temperature T and background cross-section σ_0. This form is widely used in fast reactor analysis. The angular dependence of elastic scattering is taken into consideration up to P_5 component depending on nuclide. The Public Fast Library has the asymptotic spectrum (a representative spectrum of thermal reactors) for collapsing the cross-sections of materials not defined in the collision probability calculation.

1.2.3 Public MCROSS Library

The effective resonance cross-sections in the energy region from 0.1523 eV to 1433.82 eV can be calculated by the PEACO routine with the collision probability method using an almost continuous energy group structure. The minimum and maximum lethargy meshes (Δu) are 0.000125 and 0.0005, respectively. The Public MCROSS Library has point-wise cross-section data at a room temperature. Before execution of the PEACO routine, the point-wise cross-section is Doppler broadened by the method of SIGMA1[9] for the temperature of resonant material specified by input. After that, the hyperfine-group cross-sections are composed into the User MCROSS Library. The PEACO routine solves a slowing-down equation with it. The MCROSS library data are given for dominant resonant nuclides.

1.2.4 Public Thermal Library

The thermal library installs the thermal neutron scattering matrices. The matrices are tabulated on 10 or fewer discrete temperatures ranging from 293 K to 2100 K. The self-shielding factors are also prepared for the fission and capture cross-sections of the nuclides whose resonance levels exist in the thermal neutron energy region. For some of such nuclides, a representative scattering matrix without temperature dependency is assigned for all the temperatures used in the tabulation of the thermal library, because of their smaller contributions to neutron energy transfer in practical reactor calculations. In the lattice calculation, the temperature interpolation is done not only for the resonance shielding factors, but also for the thermal scattering matrices. The interpolation of thermal scattering matrices is made by using the Lagrandian three point interpolation formula,

$$\begin{aligned}\sigma(T) = {} & (T-T_2)(T-T_3)\sigma(T_1) \ / (T_1-T_2)(T_1-T_3) \\ & + (T-T_1)(T-T_3)\sigma(T_2) \ / (T_2-T_1)(T_2-T_3) \\ & + (T-T_1)(T-T_2)\sigma(T_3) \ / (T_3-T_1)(T_3-T_2)\end{aligned} \tag{1.2-1}$$

where T is the specified material temperature, T_2 is the nearest temperature to T in the tabulated temperatures, T_1 and T_3 are the neighboring temperatures of T_2 in the tabulated temperatures. Extrapolation is not available. The scattering matrices are omitted for most of FP (Fission Product) nuclides.

Table 1.2-1(1/2) Energy group structure of Public Fast and Thermal Libraries

Group	Upper (eV)	Lower (eV)	Δu	Remark	Group	Upper (eV)	Lower (eV)	Δu	Remark
1	2.00000E+07	1.73325E+07	0.1431		51	3.43067E+04	3.18278E+04	0.0750	
2	1.73325E+07	1.49182E+07	0.1500		52	3.18278E+04	2.84982E+04	0.1105	
3	1.49182E+07	1.38403E+07	0.0750		53	2.84982E+04	2.70001E+04	0.0540	
4	1.38403E+07	1.16183E+07	0.1750		54	2.70001E+04	2.60584E+04	0.0355	
5	1.16183E+07	1.00000E+07	0.1500		55	2.60584E+04	2.47875E+04	0.0500	
6	1.00000E+07	8.18731E+06	0.2000		56	2.47875E+04	2.41755E+04	0.0250	
7	8.18731E+06	6.70320E+06	0.2000		57	2.41755E+04	2.35786E+04	0.0250	
8	6.70320E+06	6.06531E+06	0.1000		58	2.35786E+04	2.18749E+04	0.0750	
9	6.06531E+06	5.48812E+06	0.1000		59	2.18749E+04	1.93045E+04	0.1250	
10	5.48812E+06	4.49329E+06	0.2000		60	1.93045E+04	1.50344E+04	0.2500	
11	4.49329E+06	3.67879E+06	0.2000		61	1.50344E+04	1.17088E+04	0.2500	
12	3.67879E+06	3.01194E+06	0.2000		62	1.17088E+04	1.05946E+04	0.1000	
13	3.01194E+06	2.46597E+06	0.2000		63	1.05946E+04	9.11882E+03	0.1500	
14	2.46597E+06	2.23130E+06	0.1000		64	9.11882E+03	7.10174E+03	0.2500	
15	2.23130E+06	2.01896E+06	0.1000		65	7.10174E+03	5.53084E+03	0.2500	
16	2.01896E+06	1.65299E+06	0.2000		66	5.53084E+03	4.30743E+03	0.2500	
17	1.65299E+06	1.35335E+06	0.2000		67	4.30743E+03	3.70744E+03	0.1500	
18	1.35335E+06	1.22456E+06	0.1000		68	3.70744E+03	3.35463E+03	0.1000	
19	1.22456E+06	1.10803E+06	0.1000		69	3.35463E+03	3.03539E+03	0.1000	
20	1.10803E+06	1.00259E+06	0.1000		70	3.03539E+03	2.74654E+03	0.1000	
21	1.00259E+06	9.07180E+05	0.1000		71	2.74654E+03	2.61259E+03	0.0500	
22	9.07180E+05	8.20850E+05	0.1000		72	2.61259E+03	2.48517E+03	0.0500	
23	8.20850E+05	7.06512E+05	0.1500		73	2.48517E+03	2.24867E+03	0.1000	
24	7.06512E+05	6.08101E+05	0.1500		74	2.24867E+03	2.03468E+03	0.1000	
25	6.08101E+05	5.50232E+05	0.1000		75	2.03468E+03	1.58461E+03	0.2500	
26	5.50232E+05	4.97871E+05	0.1000		76	1.58461E+03	1.43382E+03	0.1000	
27	4.97871E+05	4.50492E+05	0.1000		77	1.43382E+03	1.23410E+03	0.1500	a)
28	4.50492E+05	4.07622E+05	0.1000		78	1.23410E+03	1.01039E+03	0.2000	
29	4.07622E+05	3.50844E+05	0.1500		79	1.01039E+03	9.14242E+02	0.1000	
30	3.50844E+05	3.01974E+05	0.1500		80	9.14242E+02	7.48518E+02	0.2000	
31	3.01974E+05	2.94518E+05	0.0250		81	7.48518E+02	6.77287E+02	0.1000	
32	2.94518E+05	2.87246E+05	0.0250		82	6.77287E+02	5.54516E+02	0.2000	
33	2.87246E+05	2.73237E+05	0.0500		83	5.54516E+02	4.53999E+02	0.2000	
34	2.73237E+05	2.47235E+05	0.1000		84	4.53999E+02	3.71703E+02	0.2000	
35	2.47235E+05	2.12797E+05	0.1500		85	3.71703E+02	3.04325E+02	0.2000	
36	2.12797E+05	1.83156E+05	0.1500		86	3.04325E+02	2.49160E+02	0.2000	
37	1.83156E+05	1.49956E+05	0.2000		87	2.49160E+02	2.03995E+02	0.2000	
38	1.49956E+05	1.22773E+05	0.2000		88	2.03995E+02	1.67017E+02	0.2000	
39	1.22773E+05	1.11090E+05	0.1000		89	1.67017E+02	1.48873E+02	0.1150	
40	1.11090E+05	9.80365E+04	0.1250		90	1.48873E+02	1.36742E+02	0.0850	
41	9.80365E+04	8.65170E+04	0.1250		91	1.36742E+02	1.11955E+02	0.2000	
42	8.65170E+04	8.25035E+04	0.0475		92	1.11955E+02	9.16609E+01	0.2000	
43	8.25035E+04	7.94987E+04	0.0371		93	9.16609E+01	7.57998E+01	0.1900	
44	7.94987E+04	7.19981E+04	0.0991		94	7.57998E+01	6.79040E+01	0.1100	
45	7.19981E+04	6.73795E+04	0.0663		95	6.79040E+01	5.55951E+01	0.2000	
46	6.73795E+04	5.65622E+04	0.1750		96	5.55951E+01	5.15780E+01	0.0750	
47	5.65622E+04	5.24752E+04	0.0750		97	5.15780E+01	4.83321E+01	0.0650	
48	5.24752E+04	4.63092E+04	0.1250		98	4.83321E+01	4.55174E+01	0.0600	
49	4.63092E+04	4.08677E+04	0.1250		99	4.55174E+01	4.01690E+01	0.1250	
50	4.08677E+04	3.43067E+04	0.1750		100	4.01690E+01	3.72665E+01	0.0750	

a) Upper energy limit of PEACO

Table 1.2-1(2/2) Energy group structure of Public Fast and Thermal Libraries

Group	Upper (eV)	Lower (eV)	Δu	Remark	Group	Upper (eV)	Lower (eV)	Δu	Remark
101	3.72665E+01	3.37202E+01	0.1000		151	9.98097E-01	9.88166E-01	0.0100	
102	3.37202E+01	3.05113E+01	0.1000		152	9.88166E-01	9.73454E-01	0.0150	
103	3.05113E+01	2.76077E+01	0.1000		153	9.73454E-01	9.49420E-01	0.0250	
104	2.76077E+01	2.49805E+01	0.1000		154	9.49420E-01	9.30620E-01	0.0200	
105	2.49805E+01	2.26033E+01	0.1000		155	9.30620E-01	9.12192E-01	0.0200	
106	2.26033E+01	1.94548E+01	0.1500		156	9.12192E-01	8.63377E-01	0.0550	
107	1.94548E+01	1.59283E+01	0.2000		157	8.63377E-01	8.54786E-01	0.0100	
108	1.59283E+01	1.37096E+01	0.1500		158	8.54786E-01	7.93022E-01	0.0750	
109	1.37096E+01	1.12245E+01	0.2000		159	7.93022E-01	7.81215E-01	0.0150	
110	1.12245E+01	9.90555E+00	0.1250		160	7.81215E-01	7.06873E-01	0.1000	
111	9.90555E+00	9.18981E+00	0.0750		161	7.06873E-01	6.30083E-01	0.1150	
112	9.18981E+00	8.31529E+00	0.1000		162	6.30083E-01	5.42317E-01	0.1500	
113	8.31529E+00	7.52398E+00	0.1000		163	5.42317E-01	5.00622E-01	0.0800	
114	7.52398E+00	6.16012E+00	0.2000		164	5.00622E-01	4.85826E-01	0.0300	
115	6.16012E+00	5.35535E+00	0.1400		165	4.85826E-01	4.33049E-01	0.1150	
116	5.35535E+00	5.04348E+00	0.0600		166	4.33049E-01	4.13994E-01	0.0450	
117	5.04348E+00	4.12925E+00	0.2000		167	4.13994E-01	3.99755E-01	0.0350	
118	4.12925E+00	4.00721E+00	0.0300	b)	168	3.99755E-01	3.89885E-01	0.0250	
119	4.00721E+00	3.38074E+00	0.1700	b)	169	3.89885E-01	3.49272E-01	0.1100	
120	3.38074E+00	3.29727E+00	0.0250		170	3.49272E-01	3.19211E-01	0.0900	
121	3.29727E+00	2.76792E+00	0.1750		171	3.19211E-01	3.14458E-01	0.0150	
122	2.76792E+00	2.72671E+00	0.0150		172	3.14458E-01	3.00621E-01	0.0450	
123	2.72671E+00	2.60673E+00	0.0450		173	3.00621E-01	2.80297E-01	0.0700	
124	2.60673E+00	2.55511E+00	0.0200		174	2.80297E-01	2.48601E-01	0.1200	
125	2.55511E+00	2.37049E+00	0.0750		175	2.48601E-01	2.20490E-01	0.1200	
126	2.37049E+00	2.13421E+00	0.1050		176	2.20490E-01	1.89777E-01	0.1500	
127	2.13421E+00	2.10243E+00	0.0150		177	1.89777E-01	1.80522E-01	0.0500	
128	2.10243E+00	2.02000E+00	0.0400		178	1.80522E-01	1.60108E-01	0.1200	
129	2.02000E+00	1.93111E+00	0.0450		179	1.60108E-01	1.52300E-01	0.0500	c)
130	1.93111E+00	1.84614E+00	0.0450		180	1.52300E-01	1.40000E-01	0.0842	
131	1.84614E+00	1.76490E+00	0.0450		181	1.40000E-01	1.34000E-01	0.0438	
132	1.76490E+00	1.67883E+00	0.0500		182	1.34000E-01	1.15000E-01	0.1529	
133	1.67883E+00	1.59695E+00	0.0500		183	1.15000E-01	1.00001E-01	0.1398	
134	1.59695E+00	1.50395E+00	0.0600		184	1.00001E-01	9.50000E-02	0.0513	
135	1.50395E+00	1.48156E+00	0.0150		185	9.50000E-02	8.00000E-02	0.1719	
136	1.48156E+00	1.44498E+00	0.0250		186	8.00000E-02	7.70000E-02	0.0382	
137	1.44498E+00	1.37451E+00	0.0500		187	7.70000E-02	6.70000E-02	0.1391	
138	1.37451E+00	1.34057E+00	0.0250		188	6.70000E-02	5.80000E-02	0.1442	
139	1.34057E+00	1.30095E+00	0.0300		189	5.80000E-02	5.00000E-02	0.1484	
140	1.30095E+00	1.23750E+00	0.0500		190	5.00000E-02	4.20000E-02	0.1744	
141	1.23750E+00	1.17128E+00	0.0550		191	4.20000E-02	3.50000E-02	0.1823	
142	1.17128E+00	1.15384E+00	0.0150		192	3.50000E-02	3.00000E-02	0.1542	
143	1.15384E+00	1.12535E+00	0.0250		193	3.00000E-02	2.50000E-02	0.1823	
144	1.12535E+00	1.10860E+00	0.0150		194	2.50000E-02	2.00000E-02	0.2231	
145	1.10860E+00	1.09757E+00	0.0100		195	2.00000E-02	1.50000E-02	0.2877	
146	1.09757E+00	1.07047E+00	0.0250		196	1.50000E-02	1.00000E-02	0.4055	
147	1.07047E+00	1.04404E+00	0.0250		197	1.00000E-02	6.90000E-03	0.3711	
148	1.04404E+00	1.03365E+00	0.0100		198	6.90000E-03	5.00000E-03	0.3221	
149	1.03365E+00	1.01826E+00	0.0150		199	5.00000E-03	3.00000E-03	0.5108	
150	1.01826E+00	9.98097E-01	0.0200		200	3.00000E-03	1.00000E-05	5.7038	

b) Thermal cut-off energy

c) Lower energy limit of PEACO

1.3 Data Storage in PDS (Partitioned Data Set) Files

In MOSRA, the modular codes exchange the information such as cross-sections and fluxes which are stored in the common format file called PDS file. The PDS file is a partitioned data set (of only one storied hierarchy) in binary format. As shown in Fig.1.3-1, one PDS file can contain a number of sub-files, each of which is called member. The member name is given by not more than 16 alphanumeric characters (usually 8 or 16). In MOSRA-SRAC, the role of each constituent character is defined. For example, the Public Fast Library keeps principal (vector type) cross-section data of U-235 and Pu-239 in the members named 'IU023500' and 'IPU23900', respectively. On the UNIX operating system, one PDS file is just a 'directory', and a member is a usual sequential access file.

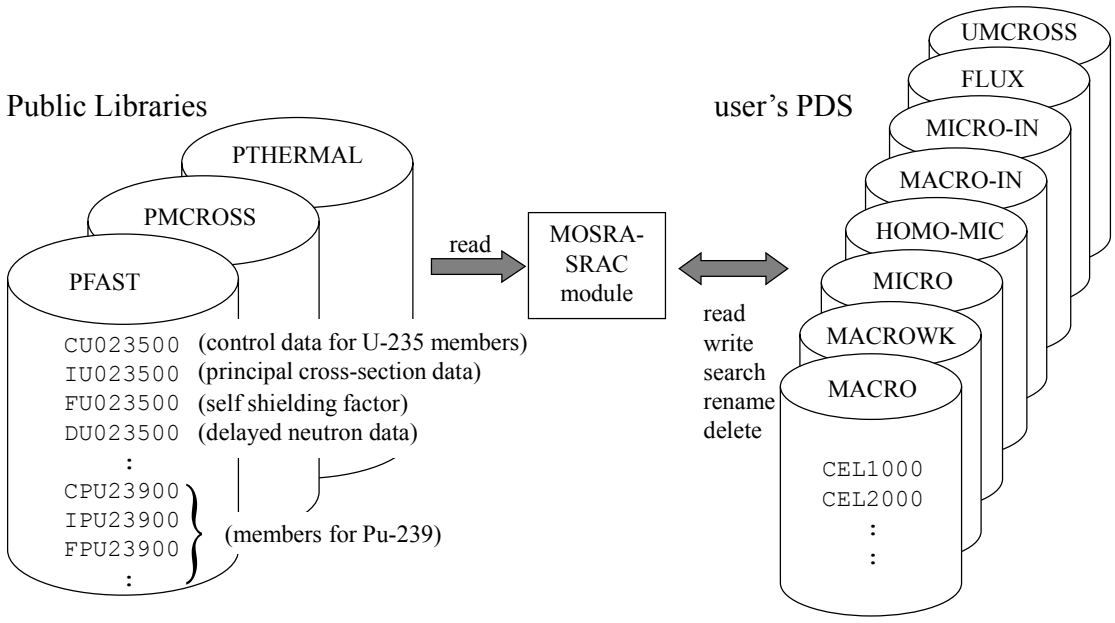


Fig.1.3-1 File I/O with Partitioned Data Set (PDS) files

The MOSRA-SRAC code uses the following eleven PDS files;

- PFAST : Public Fast Library (read only),
- PMCROSS : Public MCROSS Library for PEACO (read only),
- PTHERMAL : Public Thermal Library (read only),
- UMCROSS : User MCROSS Library for PEACO (internal use only),
- MACROWK : Macroscopic cross-sections of mixtures and/or those of

homogenized materials in the fine-group structure,

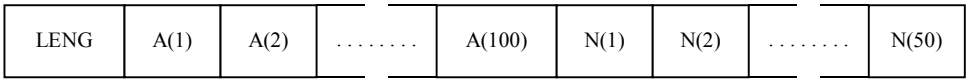
- FLUX : Flux distribution in the fine-group or collapsed group structure,
- MACRO : Homogenized macroscopic cross-sections in the collapsed group structure and fuel composition data in the burn-up calculation,
- MICRO : Effective microscopic cross-sections of nuclides in mixtures,
- HOMO-MIC : Homogenized effective microscopic cross-sections of nuclides in mixtures,
- MACRO-IN : Homogenized macroscopic cross-sections supplied by user (option),
- MICRO-IN : Effective microscopic cross-sections supplied by user (option).

In general, The PDS file has the following advantages:

- A large amount of data can be partitioned by attribution and stored in the member with user-friendly name. In addition, required data can be designated by the member name.
- As the data length of a member is relatively short, it is efficient to read/write data.
- File structure is extendable. It is easy to add new members or PDS files for new concept data.
- File management (deleting, copying, renaming, etc.) is easy. For example, deletion of all members on U-235 in the Public Fast Library can be done by the following UNIX command and masking characters (called meta-character) to specify target member files.

  rm ?U0235*
- Only one logical device is used to access many member files.

Record format of all members is common, although their contents differ. For example, we assume a member ‘TEST0000’ which has 100 real numbers and 50 integers. The data structure of this member is as follows.



Here, ‘LENG’ is the total data length in word unit (one word = 4 byte), thus LENG=150 in this case. Data of the member TEST0000 can be read by the following FORTRAN program.

```
C-------------------------------------------------------
      DIMENSION A(100),N(50),WORK(200),IWORK(200)
      EQUIVALENCE (WORK(1),IWORK(1))
      OPEN(UNIT=1,FILE='TEST0000',FORM='UNFORMATTED,ACCESS=SEQUENTIAL)
```

```
      READ(1) LENG, (WORK(I),I=1,LENG)
      DO I=1,100
        A(I)=WORK(I)
      END DO
      DO I=1,50
        N(I)=IWORK(I+100)
      END DO
      CLOSE(UNIT=1)
C-------------------------------------------------------------
```

Since MOSRA-SRAC treats so many members, frequent opening and closing of members cause much execution time. To reduce it, virtual PDS files on the core memory are used during the execution. The members once read by MOSRA-SRAC are kept in the virtual PDS files, and after that, the data access to the members is done on the memory. When the total amount of data exceeds the memory capacity secured for the virtual PDS, the virtual PDS files not frequently used are dumped into the real disk. When the dump of the virtual PDS files occurs frequently, the data access via virtual PDS is automatically switched to the direct access to the actual PDS. All of the managements of I/O with virtual or real PDS files are carried out by a set of programed routines (MOSRA-PonPon) equipped in the MOSRA-SRAC. MOSRA-PonPon is commonly used in the modules of the MOSRA system to exchange data among the modules.

Details on the contents of each member in user's PDS files and the rule of member name are described in Chapter 6.

1.4 Effective Resonance Cross-sections

The MOSRA-SRAC code installs two kinds of methods for calculation of effective resonance cross-sections, they are the NR approximation and the PEACO calculation. One of them can be selected depending on the characteristics of the problem, the required accuracy or computation cost. The NR approximation is employed unconditionally. If the PEACO option is specified, the cross-sections obtained by the NR approximation are replaced by the values calculated by the PEACO calculation, in the energy region where the PEACO option is active.

1.4.1 NR Approximation

The Public Fast and Thermal Libraries contain the infinite dilution cross-sections (σ_∞) of every nuclide and the tabulation of self-shielding factors for resonant nuclides (*f*-table). The effective microscopic cross-section of reaction *x* of nuclide *n* in resonant material (mixture with one or more resonant nuclides) *i* is given by Eq.(1.4-1) using the above quantities.

$$\sigma_{x,n}^{i} = \sigma_{\infty,x,n} f_x(\sigma_{0,n}^{i}, T^i) \tag{1.4-1}$$

where $f_x(\sigma_{0,n}^{i}, T^i)$ is the resonance self-shielding factor obtained by the interpolation of the *f*-table with a parameter: background cross-section $\sigma_{0,n}^{i}$ defined by Eq.(1.4-2) and another parameter: the temperature T^i of the mixture of which the nuclide is a constituent. The interpolation is carried out by using the third order spline function. The background cross-section is given by the equivalence theory between homogeneous and heterogeneous systems as

$$\sigma_{0,n}^{i} = \frac{1}{N_n^i} \sum_{m \neq n} (N_m^i \sigma_{t,m}^{i}) + \frac{g(C_n^i)(1 - C_n^i)}{N_n^i L^i} \tag{1.4-2}$$

$$g(C_n^i) = \frac{a}{1 + (a-1)C_n^i} \tag{1.4-3}$$

where N is the atomic number density, L the mean chord length, C_n the nuclide dependent Dancoff correction factor[6] and a the Bell factor defined in the MOSRA-SRAC code by the geometry of absorbing mixture[3].

The nuclide dependent Dancoff factor is specified optionally by the input value or the calculated value assuming the black limit. The nuclide dependence of Dancoff factor is effective for the lattice including two or more different kinds of resonant materials.

The infinite dilution cross-sections are given to the nuclide which has no *f*-table or to every nuclide in the energy group out of range of tabulation. The current available library of MOSRA-SRAC has *f*-tables for all resonant nuclides as far as resonance levels are evaluated in nuclear data files. The *f*-table is given for the reactions capture, fission, total, elastic scattering, and elastic removal in the fast energy region, and the reactions of capture and fission in the thermal energy region.

Dancoff correction by Tone's method[10] is also available. In this method, the background cross-section is calculated by the following formula;

$$\sigma_{0,n}^{i} = \frac{\sum_j \sum_{m \neq n} N_m^j P_{j \to i} \sigma_{t,m}^{j} V^j}{\sum_j N_n^j P_{j \to i} V^j}, \tag{1.4-4}$$

where $P_{j \to i}$ collision probability from region *j* to region *i*, and

V_j volume of region *j*.

The lattice heterogeneity is evaluated so as to consider contributions from each region of the lattice through collision probabilities. This method is expected to be effective for plate type fuel lattice, where a resonant nuclide exists in several composite mixtures with different atomic number densities. It should not be used for a normal pin type lattice.

1.4.2 Direct Calculation on Hyper-fine Energy Group Structure (PEACO)

A direct calculation on hyper-fine energy group structure for the heterogeneous lattice is available by the collision probability method. The following slowing-down equations on the ultra-fine lethargy mesh (Δu) of about 10^{-3}~10^{-4} are solved by the PEACO routine to obtain the hyper-fine spectra as shown in Fig.1.4-1.

$$V_i \Sigma_i(u)\varphi_i(u) = \sum_{j=1}^{J} P_{ji}(u) V_j \sum_{n=1}^{N} S_{jn}(u) \tag{1.4-5}$$

$$S_{jn}(u) = \frac{1}{1-\alpha_n} \int_{u-\varepsilon_n}^{u} \exp\{-(u-u')\}\Sigma_{sjn}(u')\varphi_j(u')du' \tag{1.4-6}$$

$$\alpha_n = \left\{\frac{A_n - 1}{A_n + 1}\right\}^2, \qquad \varepsilon_n = -\ln \alpha_n \tag{1.4-7}$$

where the subscripts i, j denote the region numbers, n the nuclide, $P_{ji}(u)$ the probability that a neutron scattered isotropically in the region j has the first collision in region i, $S_{jn}(u)$ the slowing-down source. The neutron spectrum $\varphi_i(u)$ is numerically calculated by the recurrence method developed by Kier[11].

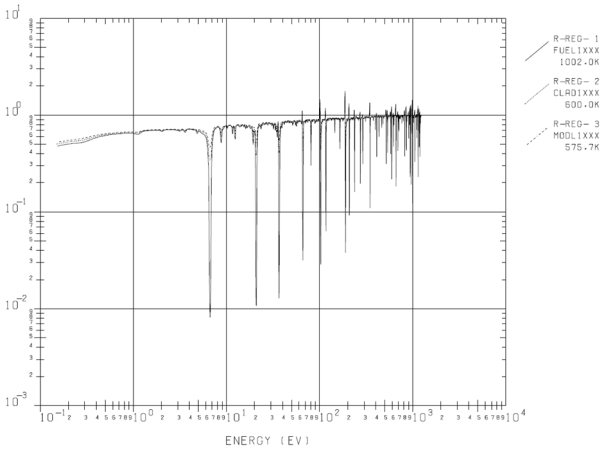


Fig.1.4-1 Neutron spectra by PEACO

The effective fine-group cross-sections are directly calculated with $\varphi_i(u)$ and hyper-fine cross-sections. The use of the PEACO routine is, in principle, limited to the system which includes one or two different resonant material(s) having resonant nuclides appearing in the MCROSS Library. It can be, however, extended to the system which includes more than two resonant materials as far as the resonant materials can be classified into two groups on the basis of absorbing characteristics. For the nuclides not compiled in the MCROSS Library, their cross-sections are not corrected by the PEACO routine.

1.5 Definition of Spatial Division

In the MOSRA-SRAC code, several spatial divisions called Sub-region, T-region, R-region, X-region and M-region are used. Various space regions and meshes are defined to enhance the calculation accuracy or to save the computer time, as the needs of the case demand. They are defined as follows.

1.5.1 Sub-region

The Sub-region is the purely geometrical sub-division that is bounded by the lines or circles used to identify the location of sub-division under consideration for the collision probability method. The rule of numbering of Sub-regions is fixed by geometry model. Sub-region is defined for convenience and does not have direct relation with the accuracy of flux calculation.

1.5.2 T-region

In the fixed source problem mode (cf. Sect.1.6), the fast and thermal energy regions are separately solved. Thermal neutron fluxes have steeper gradient than fast neutron fluxes. Therefore, the flux calculation in the thermal energy region requires finer spatial mesh division than in the fast energy region. A unit of spatial division used in thermal flux calculation is called T-region. A T-region is composed of one or more Sub-region(s) taking account of geometrical symmetry. Some adjacent Sub-regions with thin optical thickness can make a T-region.

1.5.3 R-region

Since the neutron spatial distribution in the fission or resonance energy region is flatter than in the thermal range, it is not always necessary to sub-divide the geometry into so many meshes as in the thermal energy region. In the fixed source problem mode, a unit of spatial division used in fast and resonance flux calculation is called R-region. The user forms an R-region by collecting several T-regions estimating the flatness of flux distribution. When the eigenvalue mode is chosen, R-region is the spatial division in the whole energy region. A material is allocated to each R-region.

1.5.4 X-region

A core calculation requires homogenized cross-sections in a lattice. Therefore, one or more lattice calculations on heterogeneous lattice are necessary beforehand the core calculation. Usually the homogenized cross-sections are obtained by the flux-volume weight of cross-sections of constituent materials. An X-region is the range of spatial integration. It is formed by gathering some of the

R-regions. The homogenized cross-sections and fluxes used are stored in PDS files.

For usual cases, one X-region corresponds to whole lattice of which homogenized cross-sections are provided to the core calculation. An example to use plural X-regions is the case to calculate the cross-sections of a control rod. As output of a heterogeneous lattice calculation of the system including fuel and control rods, one can obtain the cross-sections of the control rod region and those of the neighboring fuel region by allocating these two regions to two separate X-regions. Another kind of specification is that some of the R-regions may be excluded from any of the X-regions when they are added as extra regions to an isolated lattice to simulate the surrounding boundary condition by allocating zero value to these R-regions.

1.5.5 M-region

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

1.6 Fixed Source Problem Mode and Eigenvalue Problem Mode

The collision probability routine (PIJ) works on the following two alternative problem modes.

1.6.1 Fixed Source Problem Mode

The first mode is the fixed source problem mode where a flux calculation is carried out by separating fast groups and thermal groups. The Public Fast Library installs a fission neutron spectrum as a default source assuming thermal fission of U-235. In the fixed source problem mode, the fixed source having this spectrum is uniformly distributed in the fuel region, then the spatial and energy distribution of flux in fast groups is solved. After finishing the calculation of fast groups, the slowing-down source into the thermal energy region is calculated, then the thermal flux is iteratively solved. In this mode, the spatial division may be different between the fast and the thermal ranges. While the spectra are separately solved in this mode, an optional routine HOMOSP is prepared to calculate the spectrum over the whole energy region by using the homogenized cross-sections obtained in the above steps with considering the effect of leakage. In this process, the one-point eigenvalue problem is solved based on the B_1 approximation by using the buckling value specified in the input. The HOMOSP routine yields the infinite multiplication factor and the effective one corresponding to

the buckling, as well as the homogenized spectrum.

1.6.2 Eigenvalue Problem Mode

The second mode is the eigenvalue problem mode where a flux calculation is carried out over the whole energy region. In this mode, the spatial division (R-region) is common over the whole energy groups, and fluxes and fission neutron sources are iteratively solved. Therefore, the fission source distribution in this mode is more accurate than that in the first mode. As a special case, the lattice calculation in which a surface current source is given over the whole energy region is treated as the eigenvalue problem mode in the calculation flow of MOSRA-SRAC.

As the PEACO routine solves the slowing-down equation by the collision probability method only in the resonance energy region, it is not available in the eigenvalue calculation mode. In order to use the effective cross-sections obtained by PEACO in the eigenvalue problem mode, the cross-sections have to be prepared in advance by PEACO in the fixed source problem mode.

1.7 Lattice Burn-up Calculation

In the lattice burn-up calculation, burn-up changes of atomic number densities of nuclides in the burn-up chain model are obtained by solving the depletion equation with the reaction rates for (n,f), (n,γ) and (n,2n). The reactions (n,p), (n,α), and (n,3n) are also available if they are described in the burn-up chain model given in Sect.3.2. MOSRA-SRAC provides several burn-up chain modes, and a user can choose the most appropriate model by considering reactor type and his purpose.

The following optional treatments are available in the lattice burn-up calculation:

- Option to consider cooling time during burn-up (e.g. analysis of post irradiation examination)
- Option to consider changes of non-burnable material compositions during burn-up (e.g. change of boron concentration in coolant)
- Option to calculate instantaneous or integrated conversion ratio by user's definition
- Predictor-Corrector method[12] (e.g. accurate treatments of burnable poisoned rods)
- Burn-up calculation with constant flux level (e.g. burn-up of blanket fuel)
- Branch-off calculation (e.g. the Doppler or void reactivity in each burn-up step)
- Burn-up calculation to start with the initial composition from a burn-up calculation result in different conditions.
- Burn-up calculation with fixed atomic number densities for specified nuclides (e.g. zero

Xe in branch-off calculation, or on-line decontamination)

- Restart option to recover a burn-up calculation terminated for any reasons, which is available as far as the MACRO file is preserved.

As well as the atomic number densities along burn-up, the following items are edited for each M-region and X-region on the text file allocated to the 98-th logical device.

DAYS	Accumulated burn-up period in days
MWD/T	Exposure (MWt×days per metric-ton of initial heavy metal inventory)
U02350-%	Fraction of depleted U-235 (changeable by user) atomic number density to the fresh one (0～100%)
K-EFF	Effective neutron multiplication factor
K-INF	Infinite neutron multiplication factor
INST.-C.R.	Instantaneous conversion ratio defined by user
INTE.-C.R.	Integrated conversion ratio defined by user
MWD	Exposure (MWt×days)
POWER (MW)	Thermal power over the lattice
TON-HM	Heavy metal inventory in metric-ton ($=10^3$ kg)
FLUX- LEVEL	Absolute one-group flux level (n/cm^2-s)
FIS.-ABSOR	Macroscopic absorption rate of fissile nuclides defined by user (s^{-1})
FIS.-DECAY	Decay rate of fissile nuclides defined by user (s^{-1})
FER.-CAPTR	Macroscopic capture rate of fertile nuclides defined by user (s^{-1})
PRE.-DECAY	Decay rate of fertile nuclides defined by user (s^{-1})
POW (MW/CC)	Power density (MW/cm^3)
ENRGY/FIS.	Average energy release per fission (Joule/fission) weighted by nuclide-wise fission rates
XE-135-YD.	Average fission yield of X-135 weighted by nuclide-wise fission rates
I-135-YD.	Average fission yield of I-135 weighted by nuclide-wise fission rates
SM-149-YD.	Average fission yield of Sm-149 weighted by nuclide-wise fission rates
PM-149-YD.	Average fission yield of Pm-149 weighted by nuclide-wise fission rates

The above data are stored in the MACRO file together with macroscopic cross-sections to be used in other core calculation modules.

1.8 Input Data Structure

To overview how to make an input data of MOSRA-SRAC, we shall follow a typical example for the analysis of a critical assembly composed of a central mixed oxide (MOX) fuel zone and a surrounding UO_2 driver fuel zone. As shown in Fig.1.8-1, we make two sets of homogenized macroscopic cross-sections in a few energy groups by MOSRA-SRAC for the succeeding core calculation by other modules.

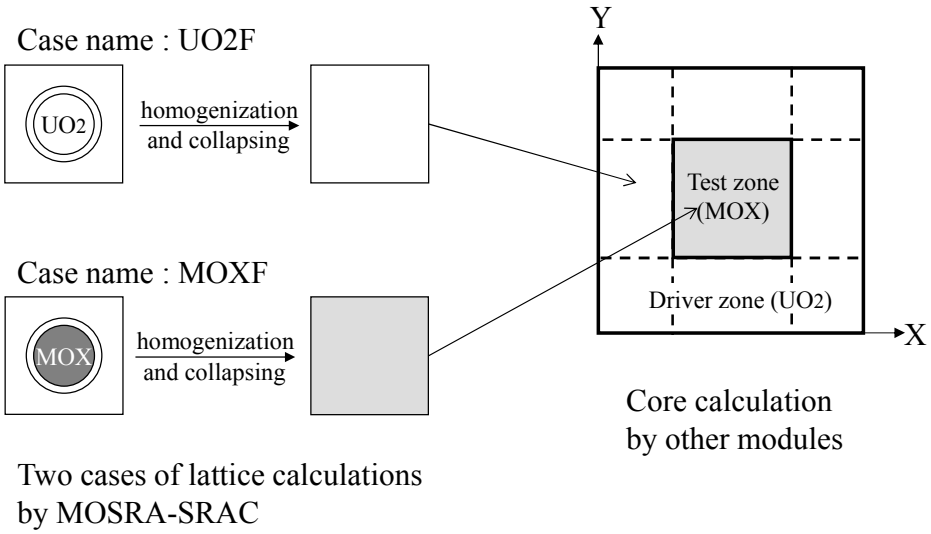


Fig.1.8-1 An image of generating macroscopic cross-sections by MOSRA-SRAC for the analysis of critical assembly composed of a MOX fueled test zone and a surrounding UO_2 driver zone.

In this sample, two cases of lattice calculations are carried out for the UO_2 and MOX pin cells. Figure 1.8-2 shows the input data corresponding to the above sample problem, although details are omitted to emphasize the input data structure. In the first case (case name UO2F), 200-group spectrum of the UO_2 pin cell is calculated as a fixed source problem by using PIJ and PEACO routines to obtain the homogenized and collapsed-group cross-sections (17-groups in this sample). The second case (MOXF) is the same process for the MOX pin cell as the first case.

As shown in this sample, a set of lattice calculation named by 4 characters (case name) is repeated until a blank case name is entered. The cases can be repeated as far as the computation cost allows.

```
UO2F   ← Case name for UO2 cell calculation
Cell calculation for TCA experiment  ← Comment for this case (two lines)
Pin cell calculation for UO2 fuel
0 0 0 0 1   1 0 0 0 0  10(0)  ← Option control
0 0 0 0 0   0 0 0 0 0         ← Print control
8.33E-3  / Buckling
*******************************************
PFAST
PTHERMAL
PMCROSS
UMCROSS
MACROWK
FLUX                                        ← Specification of PDS (first case only)
MACRO
MICRO
HOMO-MIC
            / MACRO-IN is not used
            / MICRO-IN is not used
*******************************************
200 118 17 10    / Fast(118g)+Thermal(82g) => Fast(10G)+Thermal(7G)
           :
           :     { Energy group structure }
           :
           :     { Geometry for PIJ }
           :
*******************************************
3 / Number of Materials                                  ←Material specification
FUEL100X   0 4   293.15  1.250    0.0 / 1 : UO2 Fuel
U02340     0 0   4.8872E-6     /1
U02350     0 0   6.0830E-4     /2
U02380     0 0   2.2531E-2     /3
O00160     0 0   4.7214E-2     /4
CLAD100X   0 1   293.15  0.3143   0.0  /2 : Cladding
AL0270     0 0   5.5137E-2     /1
WATR100X   0 2   293.15  0.0      0.0  /3 : MODERATOR
H0001H     0 0   6.6735E-2     /2
O00160     0 0   3.3368E-2     /2
           :
           :
0   / IPLOT in Block-1 of Sect.2.6 for PEACO
*******************************************
MOXF   ← Case name for MOX cell calculation
Cell calculation for TCA experiment  ← Comment for this case
Pin cell calculation for UO2 fuel
0 0 0 0 1   1 0 0 0 0  10(0)  ← Option control
0 0 0 0 0   0 0 0 0 0         ← Print control
8.33E-3  / Buckling
*******************************************          Input for MOXF case
           :
           :     { Geometry for PIJ }
           :
           :     { Material specification for MOX fuel }
           :
*******************************************
     / End job  ← Blank case name to terminate job
```

Fig.1.8-2 Sample input data structure of the MOSRA-SRAC

As a result of the above lattice calculations, the homogenized collapsed cross-sections are written in the macroscopic cross-section file (MACRO) by the member names UO2F1000 (P0 cross-sections), UO2F1001 (P1 cross-sections), UO2F100D (kinetics parameters), and MOXF1000

(P0 cross-sections), MOXF1001 (P1 cross-sections), MOXF100D (kinetics parameters). The member name consists of the first four characters for the case name, the homogenized region number by the fifth character, the burn-up step number by the sixth and seventh characters, and the data type by the eighth character. Rules of the member names are described in Chapter 6.

1.9 Output Information

Major calculated results of MOSRA-SRAC are edited on a text-formatted file allocated to the 99-th logical device. On the standard output file of the 6-th device, information is written to check whether a series of calculations has completed appropriately or not, for instance, echo of user's input, status of calculation progress, record of PDS file access, warning or error message. When the calculation finished normally, the following message will appear at the last of the above two files.

```
'======================== END OF MOSRA-SRAC CALCULATION ========================'
```

Otherwise, a user has to check the standard output carefully.

For PIJ or PEACO, plotting options are available to confirm lattice geometry under consideration or to draw hyper-fine neutron spectra obtained by PEACO. The plot data is written as a text-formatted Postscript file allocated to the 89th device. The user can see the figures of a Postscript file on a screen or printed matter, by using a free software on the market. On the UNIX operating system, a command 'lpr' is usually available to print out the figures.

In the case of the burn-up calculation, editing of the calculated results in each burn-up step is repeated up to the final burn-up step on the 99-th file. Its contents may be too enormous to extract necessary data. Therefore, major burn-up parameters are summarized and tabulated to the burn-up step on the 98-th text-formatted file.

The binary data of fluxes and effective microscopic or macroscopic cross-sections are written in their own PDS files (cf. Chapter 6). Although some of them can be printed on the 99-th file by options, their printed format may not be convenient for plotting them or editing reaction rates defined by each user. In order to support editing or managing contents of PDS files, several utility programs are equipped in MOSRA-SRAC as described in Chapter 7.

2. Input Data Requirements

The input data of MOSRA-SRAC consist of the following five input sections for a calculation case (cf. Sect. 1.8) in a job.

- General control and PDS files (always required at the first case)
- PIJ: Collision probability calculation (always required)
- Material specification (always required)
- Lattice burn-up calculation (option)
- Additional input (depending on the use of optional functions specified in the general control in Sect.2.2)

For a job with multiple-cases, a set of the above input sections are repeated necessary times.

2.1 MOSRA-SRAC Free Format

All input data for MOSRA-SRAC are read in a free format common to all modules of MOSRA. The features and usage are as follows.

(1) Three types of data array (character string of four bytes, integer, and floating point number of single precision) can be read. (cf. Material specification in Fig.1.8-2).

e.g. TEST 1 2 1.00 2.00 3.00

(2) Columns 1 to 130 of a line record is used as data field. Data out of the field is neglected.

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

e.g. 1,2 3+4-5 is accepted as 1 2 3 4 -5

(4) A floating number may be entered by *F*-type or *E*-type; the latter needs the exponent code 'E' or 'e' at the beginning of exponent. *D*-type is not accepted.

e.g. -12.543 00.00125 1.0E-4 -.4E12 2.9e-2

(5) A word must be completed in a line record.

A wrong example: -12.543 0.00125 1.0E
-4 -.4E12

(6) Any blank column should not be inserted between sign code and digit code.

A wrong example: 1 2 - 3 4 - 5

(7) For *E*-type, any blank column should not be inserted between the exponent code 'E' or 'e' and the succeeding character.

A wrong example: 1.000E -5

(8) For character type, the style for the free format is not applied. Column position of character type variables is, in general, organized to start at the first column of a line record.

A wrong example in the case /4(A4)/ : ABCD EFGH 2(IJKL)

A correct example in the case /4(A4)/ : ABCDEFGHIJKLIJKL

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

e.g. 1 3(2) 2 (1.E-4) is accepted as 1 2 2 2 1.0E-04 1.0E-04

The data string enclosed by '( )' can be written on more than one records. The closing ') ' must not be written on the first of the succeeding line.

e.g. 10 (1 2 3 4 5 4 3 2 1
5 4 3 2 1 2 3 4 5)

A wrong example: 2 (1 2 3 4
) 5 6 7 8

The duplicate use of '( )' like 2(1 2(3 4)) is not allowed.

(10) Accumulation function is also available. An integer before ' * ' is taken as the number of times of accumulation, and the data value after ' * ' is taken as increment to be added to the previous data. That is to say, 'a b*c' means 'a a+c a+2c a+3c ..... a+bc'.

e.g. 0.0 4*1.0 2*-2.0 is accepted as 0.0 1.0 2.0 3.0 4.0 2.0 0.0

The coupling of ' ( )' and ' * ' is not allowed.

A wrong example: 10(0 5*1)

(11) A series of strings for repetition or accumulation function must close within each type of array.

A wrong example: 10(1 1.0E-4)

A correct example: 10(1) 10(1.0E-4)

(12) A line which has the character ' * ' at the first column is taken as a comment line.

(13) The character ' / ' is taken as the termination code of required data. The ' / ' is not necessarily required. If the termination code character is encountered, a check whether or not the input data length meets with the required data length is likely done by the code. However the character ' / ' on the new record after entering required data on the previous record causes an error because the read-in is finished on the previous record without the termination code, then the code ' / ' is read at the beginning of the next call. The columns after ' / ' can be used as comment.

e.g. 5(0) 5(1) / Input data for Block-1

(14) The character '&' is taken as the end-of-record code character. The columns after the '&' can be used for comment. If the entries are not yet finished at this code, the remaining data follow on the

next record.

An example when ten integers are required:

```
1  2  3  4  5  &  Input for Block-1(1-5)
& Comment 1
& Comment 2
6  7  8  9  10  /  End of Block-1
```

Although data type (character, integer or floating) of variable or array in the following description is not always mentioned, the user can recognize character type data by finding the indicator for character length as '/A8/'. Concerning numerical data, the user can discriminate integer type or floating type by the first character of the variable name whether it is one of characters from I to N or not.

The term Block appearing in the descriptions denotes one or a series of data required by one FORTRAN read statement which may be entered on any number of lines. The use of the termination code ' / ' is recommended to have a suitable message if the data length is mismatched. The number of data required in a Block is shown as '/20/' or '/NRR/'. If mixed types of data are required in a Block, they are read in the order of character, integer, then floating type, and the data requirement is expressed by /A8,3,2/ for 8 characters, 3 integers and 2 floating numbers, respectively.

2.2 General Control and PDS Files

Block-1 /A4/

CASEID Case identification (*case*-tag)

It is used as the first four characters of the names of the members in the PDS files which store the effective cross-sections (MACRO, MACROWK, etc.) and fluxes (FLUX) homogenized in the X-region.

As plural cases can be run in one job, enter a blank line after the last case to terminate the job. The energy group structure for the collision probability calculations in the plural cases must be common in a job.

[cf.] Sect.1.8, Chapter 6

Block-2 Case description /2A72/

TITLE1 The first comment for the problem

TITLE2 The second comment for the problem

Block-3 Integers to specify the options to control calculation flow /20/

IOPT1 Selection of the calculation mode in the collision probability method (CPM) routine

= 0 Fixed source problem mode

= 1 Eigenvalue problem mode

Note:

The PEACO option is effective only when IOPT1=0. For the eigenvalue problem mode, effective cross-sections must be prepared beforehand by the fixed source problem mode.

IOPT2 Indicator for the energy region solved

= 0 Thermal energy region is included (for thermal reactors)

= 1 Thermal energy region is excluded (for fast reactors)

IOPT3 Indicator whether to enter or not the geometrical information required in Sect. 2.3 for this case

= 0 Read the new geometry

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

IOPT4 Selection of the correction for the heterogeneous effect on the admixture cross-sections in the interpolation of resonance shielding factors upon the NR approximation

= 0 Dancoff correction factor calculated by CPM[6)]

= 1 Tone's method[10)]

= 2 Stoker and Weiss's method[13)]

= 3 Kugo's method[14)]

= 4 Use the input value of Dancoff correction factor specified in the material specification (cf. Sect.2.4)

Note:

Tone's method is recommended for a plate type lattice with neighboring different fuels, but it is not recommended for a pin type lattice.

Stoker and Weiss's method is available when we consider the special dependent self-shielding effect in a fuel pellet that is divided into several M-regions. When IOPT4=2 is entered, an additional input block is required in Sect.2.6.

When the double heterogeneity is solved by the PEACO routine by specifying a negative value of MAR (Block-6 in Sect. 2.3), the Dancoff correction factor of the microscopic heterogeneity in the material specification is used in spite of any IOPT4 value while that of the macroscopic heterogeneity is controlled by the IOPT4 value.

IOPT5 Selection of the process for resonance absorption in the resonance energy region

= 0 Interpolation of Bondarenko type table by the NR approximation.

= 1 The PEACO routine (hyperfine-group calculation by CPM)

The PEACO routine runs only on the fixed source problem mode (IOPT1=0).

The number of resonant materials (which contains at least one nuclide(s) with the MCROSS library data) is limited to one or two.

= 2 The PEACO routine to treat more than two resonant materials by an approximation to assume two pseudo resonant materials.

Additional input (KCOREK in Sect.2.4) is required to assign the materials to which each resonant material belongs.

Note:

It is recommended to use the PEACO routine for the lattice of thermal reactors,

because the NR approximation is not so accurate in the lower energy region. However, the PEACO routine does not work for more than two resonant materials in a lattice, unless IOPT5=2 is entered.

When IOPT5=1 or 2 is entered to use the PEACO option, additional input (IPLOT) is required in Sect.2.6.

[cf.] Sect.1.4, Sect.2.4, Sect.2.6

IOPT6 Indicator to calculate the one point (bare) reactor neutron spectrum, k_∞ and k_{eff} in the fixed source problem mode (IOPT1=0). Enter IOPT6=4 in the eigenvalue problem mode (IOPT1=1).

= 0 B_1 approximation with no correction of neutron spectrum for energy collapsing

= 1 B_1 approximation with correction of neutron spectrum for energy collapsing

= 2 Critical buckling search by B_1 approximation to correct neutron spectrum for energy collapsing

= 3 (not used)

= 4 Skip one point reactor calculation

Note:

If IOPT6=1 or =2 is entered, P_0 components of the solution of B_1 equations are used as the weight to collapse the homogenized cross-sections; i.e. the leakage effect is reflected in the spectrum by the geometrical buckling value. If not, the spectrum obtained by the lattice calculation is used for collapsing.

The CPM calculation is carried out upon infinite lattice approximation. The geometrical buckling is used to reflect the leakage effect on the spectrum by use of the HOMOSP routine.

If IOPT6=0 or =1 is entered, the geometrical buckling given in Block-5 is used in the B_1 equations. If IOPT6=2 is entered, the geometrical buckling is obtained so that the k_{eff} is unity. The use of the critical buckling search option should be avoided for the core where the large excess reactivity is mainly suppressed by using control absorber.

IOPT7 Indicator to call the lattice burn-up calculation

= 0 Skip

= 1 Execute burn-up calculation

Note:

If the system contains a homogenized region which occurs in a super-cell calculation, the lattice burn-up calculation is not available.

[cf.] Sect.1.7, Sect.2.5

IOPT8 P_L scattering order of cross-sections to be stored in the PDS files

= 0 Use default value (L=1)

= L Upto L-th order ($1 \leq L \leq 5$)

IOPT9 Indicator to get the X-region averaged effective microscopic cross-sections in the PDF files.

= 0 Skip the averaging process

= 1 Call the averaging process for all nuclides

= 2 Call the averaging process for the nuclides specified in the material specification (cf. IXMICR in Sect.2.4)

= 3 Call the averaging process for all nuclides in each X-region and isolated materials. Matrix data for elastic scattering, inelastic scattering, (n,2n) cross-sections are also stored in the PDS files.

= 4 Call the averaging process for the nuclides specified in the material specification (cf. IXMICR in Sect.2.4) in each X-region and isolated materials. Matrix data for elastic scattering, inelastic scattering, (n,2n) cross-sections are also stored in the PDS files.

= 5 Call the averaging process for all nuclides. Matrix data for elastic scattering, inelastic scattering, (n,2n) cross-sections are also stored in the PDS files.

= 11 Call the averaging process for all nuclides. In addition to a collapsed-group cross-sections, fine-group cross-sections are also stored in the PDS (MICRO-OUT) file.

= 12 Call the averaging process for all nuclides specified in the material specification (cf. IXMICR in Sect.2.4). In addition to a collapsed-group cross-sections, fine-group cross-sections are also stored in the PDS (MICRO-OUT) file.

Note:

The isolated material is the material which is defined in the material specification but not allocated to any R-region of the CPM calculation (e.g. water reflector). It is usually prepared for the core calculation by other MOSRA modules.

When IOPT7=1 (burn-up calculation) and IOPT9=2 are entered and when a user requires the averaged cross-sections for the nuclide whose initial atomic number densities is zero (e.g. fission products), the user has to specify the nuclides in the material specification.

When IOPT7 is given by negative value, printing and PDS output of microscopic delayed neutron data (cf. Sect.6.2) is skipped.

IOPT10 Indicator to print debugging information for developers

= 0 No printing

= 1 Print simple information

= 2 Print detailed information

IOPT11 = 0 (not used)

IOPT12 = 0 (not used)

IOPT13 = 0 (not used)

IOPT14 = 0 (not used)

IOPT15 = 0 (not used)

IOPT16 = 0 (not used)

IOPT17 = 0 (not used)

IOPT18 = 0 (not used)

IOPT19 = 0 (not used)

IOPT20 Indicator for the additional options to control calculation flow

= 0 No additional options (default options are used)

= 1 Specify additional options in the following Block-3A

Block-3A Required if IOPT20=1 in Block-3 /10/

IOPT21 Indicator how to form the macroscopic transport cross-sections of each homogenized X-region or isolated material

= 0 Use the current component (J) obtained by the B_1 approximation[12,15,)]

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

= 1 The extended transport approximation[12)]

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

Note:

The macroscopic transport cross-section of each mixture defined by M-region is always formed with the extended transport approximation. On the other hand, the homogenized transport cross-section of each X-region can be formed with the above B_1 approximation, where the geometrical buckling given in Block-5 is used in the leakage term. The B_1 approximation is also applied to the isolated materials, which are defined in the material specification but not allocated to any R-region of the CPM calculation. This option (IOPT21=0) is effective when the energy group structure for the current case is the same as those of the Public Library (i.e. no energy collapsing).

IOPT22 Indicator how to compose and to collapse the averaged diffusion coefficients

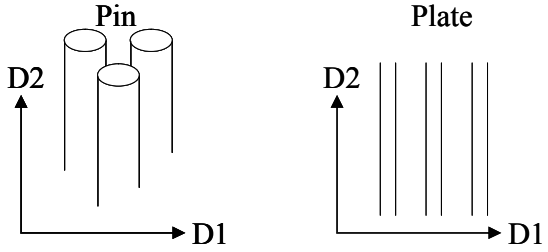
Three kinds of diffusion coefficients D1, D2 and D3 are stored in the MACRO and MACROWRK files to allow core calculations with direction dependent diffusion coefficients.

= 0 The anisotropic components of Behrens' term of the Benoist model[16)].

$$D_{k,g} = \left\{ \sum_i \varphi_{i,g} \sum_j P_{i->j,k,g} / \Sigma_{tr,j,g} \right\} / 3 \sum_i \varphi_{i,g}$$

The radial components in the cylindrical coordinate or the perpendicular components in the plate geometry are written into D1, and the axial components in the cylindrical or the parallel components in the plate are written into the D2 position. In the D3 position, the diffusion coefficients

are made from the inverse of the transport cross-sections as $D_g = 1/3\Sigma_{tr,g}$.



= 1 Inverse of the transport cross-sections.

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

The same values are stored in the D1, D2 and D3 positions.

= 2 The isotropic components of Behrens' term of the Benoist model[25)]

$$D_g = \left\{ \sum_i \varphi_{i,g} \sum_j P_{i->j,g} / \Sigma_{tr,j,g} \right\} / 3 \sum_i \varphi_{i,g}$$

The same values are stored in the D1, D2 and D3 positions.

Note:

When the number of X-regions is more than one, IOPT22=1 is automatically set regardless of the input value here.

IOPT23 Indicator how to prepare fixed sources for the fixed source problem mode.

= 0 A fixed source is prepared automatically by the code. A flat source with the fission spectrum of U-235 is assumed.

= 1 Input a fixed source distribution by R-region

= 2 A surface source prepared by the user is used. The source has to be beforehand prepared as a member file *caseid*FIXS in the FLUX file. (cf.

Note:

Usually enter IOPT23=0. If IOPT23=1, additional input data is required in Sect.2.6. When IOPT23=2 is entered, set IOPT1=1 in Block-3 and IBOUND=2 in the PIJ input section (Sect.2.3)

IOPT24 Indicator to specify the lower energy boundary of the PEACO calculation in the fixed source problem mode.

= *n* Energy group number in the PFAST library. ($118 \leq n \leq 179$)

The lower energy boundary is 4.00721 eV and 0.1523 eV when n=118 and n=179, respectively. When n is out of the above range (e.g. n=0), the lowest energy boundary (n=179) is taken internally.

IOPT25 Indicator to correct σ_0 value for interpolation of the self-shielding factor in Eq.(1.4-1).

= 0 No correction . The σ_0 value is calculated by Eq.(1.4-2).

= 1 Empirical correction factor (F^{het}) is applied to all resonant nuclides as follows:

$$\sigma_{0,g}(corrected) = F_g^{het} \times \sigma_{0,g}(original)$$

$$F_g^{het} = \begin{cases} 1.0 & (\overline{E}_g \geq 10\text{keV}) \\ 0.77778 + 2.2222 \times 10^{-5} \times \overline{E}_g & (1\text{keV} < \overline{E}_g < 10\text{keV}) \\ 0.80 & (\overline{E}_g \leq 1\text{keV}) \end{cases}$$

= 2 Empirical correction factor (F^{het}) given by the user is applied to all resonant nuclides. Additional input for F^{het} is required in Block-9.

Note:

Usually enter IOPT25=0. The IOPT25=1 or 2 is one of options for a research purpose. The correction factors of IOPT25=1 were determined by comparison with the PEACO results for typical thermal and fast reactor lattices.

IOPT26 = 0 (not used)

IOPT27 = 0 (not used)

IOPT28 = 0 (not used)

IOPT29 = 0 (not used)

IOPT30 = 0 (not used)

Block-4 Edit control for calculated results /10/

Printing of fine-group information is very voluminous. The user can skip printing here, because most of the calculated results can be printed after the execution of

job by using some utility programs as far as the related PDS files are remained.

IP1 Neutron flux

= 0 Skip print

= 1 Print fine-group flux

= 2 Print collapsed-group flux

= 3 Print both of fine-group and collapsed-group fluxes

IP2 Lattice averaged effective macroscopic cross-section

= 0 Skip print

= 1 Print fine-group cross-section

= 2 Print collapsed-group cross-section

= 3 Print both of fine-group and collapsed-group cross-sections

IP3 Lattice averaged effective microscopic cross-section

= 0 Skip print

= 1 Print fine-group cross-section

= 2 Print collapsed-group cross-section

= 3 Print both of fine-group and collapsed-group cross-sections

IP4 Effective macroscopic cross-section of isolated material

= 0 Skip print

= 1 Print fine-group cross-section

= 2 Print collapsed-group cross-section

= 3 Print both of fine-group and collapsed-group cross-sections

Note:

The isolated material is the material which is defined in the material specification but not allocated to any R-region of the CPM calculation (e.g. water reflector). It is usually prepared for the core calculation by other MOSRA modules.

IP5 Effective microscopic cross-section of the isolated material

= 0 Skip print

= 1 Print fine-group cross-section

= 2 Print collapsed-group cross-section

= 3 Print both of fine-group and collapsed-group cross-sections

IP6 Fine-group effective macroscopic cross-section of the material which constitute a lattice for the collision probability calculation

= 0 Skip print

= 1 Print cross-section

IP7 Fine-group effective microscopic cross-section of material which constitute a lattice for the collision probability calculation

= 0 Skip print

= 1 Print cross-section

IP8 = 0 (not used)

IP9 Print control for the PEACO results (effective when IOPT5>0)

= 0 Skip print

= 1 Print flux and effective macroscopic cross-section

= 2 Print flux, effective macroscopic and microscopic cross-sections

= 3 Print debug information for developers

IP10 Indicator for additional edit options

= 0 No additional options

= 1 Specify additional options in the following Block-4A

Block-4A Required if IP10=1 in Block-4 /10/

IP10 = 0 (not used)

IP11 = 0 (not used)

IP12 = 0 (not used)

IP13 = 0 (not used)

IP14 = 0 (not used)

IP15 = 0 (not used)

IP16 = 0 (not used)

IP17 = 0 (not used)

IP18 = 0 (not used)

IP19 = 0 (not used)

IP20 = 0 (not used)

Block-5 Geometrical buckling B^2 (cm^{-2}) /1/

BSQ Buckling value used in the B_1 approximation in the one-point reactor calculation specified by IOPT6. The negative value is accepted. The zero or too small value is internally replaced with 1.0×10^{-15} to avoid a numerical overflow.

The succeeding Block-6 through Block-8 are required only in the first case.

Block-6 Data set specification for PDS files /11*A130/

One PDS file is specified by PDSENV, MOACS, MOSTY within 130 columns. These are separated by one or more blank(s). The input for MOACS and MOSTY can be skipped by blank columns, which means the default setting is employed by the code. This type of input must be repeated 11 times for each PDS file in the order of 1) PFAST, 2) PTHERMAL, 3) PMCROSS, 4) UMCROSS, 5) MACROWK, 6) FLUX, 7) MACRO, 8) MICRO, 9) HOMO-MIC, 10) MACRO-IN and 11) MICRO-IN files.

PDSENV(*i*) Alphanumeric strings within 15 characters to indicate the *i*-th PDF file. PDSENV(*i*) is allocated to the real PDS file by a job control statement, i.e. shell-script on the UNIX system, as follows.

An example of PDSENV(1):

PFAST

An example of the file allocation of PFAST in the shell-scrip:

export PFAST=/home/okumura/MSRACLIB_J40/pds/pfast

Note:

The same name as PDSENV(*i*) must be defined for the shell variable in the shell-script, although any alphanumeric strings can be used for PDSENV(*i*) in this input Block.

Not only the allocation of PDS files, but also the creation, deletion and preservation of each PDS file is controlled by the description of shell-script. The directory of the PDS file to be used in the job should be prepared by the shell-script beforehand.

MOACS(*i*) Access mode to the *i*-th PDS file

Only the first character (capital letter) is effective.

= File Direct I/O access to file

= Core I/O access on virtual PDS file on core memory

Note:

If 'Core' is specified, all the information on a PDS file is read into the core memory at the first access. After that, I/O is carried out from/into the core. This mode is effective to reduce running time of the job which takes much I/O time. However, this mode has a risk that newly generated member files during a job may be lost when the job does not normally finished, because they are dumped into the disk at the end of the job.

When the overflow of reserved memory occurs on any PDS file, all member files are dumped into the disk and the reserved memory is refreshed. When the overflow occurs frequently, all the 'Core' specifications to the PDS files are switched to the 'File' mode automatically.

MOSTY (*i*) Option to treat the member files on the virtual *i*-th PDS file at the time of the file dump and refreshing memory during calculations

Only the first character (capital letter) "S" is effective.

= Stay Keep member files on virtual PDS

= blank* Dump all member files and refresh core memory

*No input (blank or any characters not starting from "S")

When the lattice burn-up calculation is executed, a sufficient amount of memory has to be

reserved as a lot of members are created. The memory capacity for virtual PDS files is defined by a parameter statement (MXVRF = 10000000) in the include-file 'INCPDS'.

Usually the user can skip input for MOACS and MOSTY. At that time, the following default setting is employed internally.

No.	PDS file	MOACS	MOSTY
1	PFAST	Core	Stay
2	PTHERMAL	File (IOPT7=0, MXVRF≦6000000) Core (IOPT7=0, MXVRF>6000000) File (IOPT7=1)	Stay
3	PMCROSS	File	Stay
4	UMCROSS	File (MXVRF≦10000000) Core (MXVRF>10000000)	blank
5	MACROWK	Core	blank
6	FLUX	File	blank
7	MACRO	Core	blank
8	MICRO	File (MXVRF≦15000000) Core (MXVRF>15000000)	blank
9	HOMO-MIC	File	blank
10	MACRO-IN	File	blank
11	MICRO-IN	File	blank

An example of Block-6 input:

```
PFAST                    / Public FAST Library
PTHERMAL                 / Public THERMAL Library
PMCROSS                  / Public MCROSS Library (for PEACO)
UMCROSS     Core  Stay   / User's MCROSS Library
MACROWK     File         / Fine-group macroscopic-XS
FLUX                     / Fine- and collapsed-group flux
MACRO                    / Collapsed-group Macro-XS
MICRO                    / Fine-group Micro-XS
HOMO-MIC                 / Collapsed-group homogenized Micro-XS
                         / MACRO-IN if necessary
                         / MICRO-IN if necessary
```

The member files of MACRO or MICRO generated in the previous jobs can be used as the macroscopic / microscopic cross-section data for another job by specifying MACRO-IN / MICRO-IN. If they are not necessary in the current job, enter blank lines for MACRO-IN and MICRO-IN.

In the job where MACRO-IN and/or MICRO-IN are specified in PDSENV, one of the following two procedures is available.

(1) The procedure when the energy group structure of MACRO-IN and/or MICRO-IN is the same as that of the Public Library, i.e. the case when the number of fine-energy group is 200 (IOPT2=0) or 179 (IOPT2=1).

The referred member files in MACRO-IN and MICRO-IN are transferred into MACROWK and MICRO, respectively. They can be treated in the same way as the members in MACROWK or MICRO. However, it should be noted that some functions, e.g. IOPT9>0 to make the X-region averaged microscopic cross-sections, are suppressed because the detailed information of the heterogeneous lattice is not always included in MACRO-IN.

(2) The procedure when the energy group structure of MACRO-IN is not the same as that of the Public Library, i.e. the case when the number of fine-energy group is neither 200 (IOPT2=0) nor 179 (IOPT2=1).

This procedure is aimed at the so-called super-cell calculation, where the lattice calculation is carried out with the homogenized or collapsed cross-sections prepared by other jobs. The cross-sections are supplied from MACRO-IN and transferred to MACROWK, whose member files are taken as the cross-section data with the fine-group structure in each job.

Homogenizations and collapsing of macroscopic cross-sections can be done any number of times by repeating the jobs with this procedure. However, it is not allowed to use any functions dealing with the microscopic cross-sections by specifying material compositions (e.g. MICRO-IN, PEACO, burn-up calculation, etc.). Enter a blank line for MICRO-IN, since the combined usage of MACRO-IN and MICRO-IN is not accepted in this procedure.

Block-7	Specification for energy group structure	/4/
1 NGF	Total number of fine-groups Note: Usually NGF=200 (when IOPT2=0) or NGF=179 (when IOPT2=1). Otherwise enter the number of groups in MACRO-IN.	
2 NEF	Number of the fast fine-groups (NEF≤NGF) Since the thermal cut-off energy (the energy boundary of PFAST and PTHERMAL cross-sections and upper limit of neutron up-scattering) is internally fixed as 4.00721 eV regardless of NEF, the value of NEF is not effective to calculation results; NEF is just a reminder for a user to recognize the thermal energy region of the fine-group cross-sections.	
3 NGB	Total number of collapsed-groups (1≤NGB≤NGF)	
4 NGBF	Number of the fast collapsed-groups (1≤NGBF≤NGB) The value of NGBF is not effective to calculation results; NGBF is just a reminder	

for a user to recognize the thermal energy region of the collapsed-group cross-sections.

Block-8 Group boundaries to collapse from fine-group into collapsed-group /NGB/

ICOLNG(i) The lowest fine-group number of each collapsed-group (i)

ICOLNG(i)<ICOLNG(i+1), ICOLNG(NGB)=NGF

An example of the Block-7 and Block-8

```
-----------------------------------------------------------------------------------------------------------------
200 118  2 1  / Block-7: 200(=118+82)fine-groups => 2(=1+1) collapsed-groups
118 200       / Block-8: lowest fine-group number of each collapsed-group
-----------------------------------------------------------------------------------------------------------------
```

Block-9 Required if IOPT25=2 in Block-3A /NGF/

FACHET(i) Correction factor for $\sigma_{0,g}$. ($F^{het}(g)$, g=1,NGF)

See the explanation on IOPT25 in Block-3A.

2.3 PIJ : Collision Probability Method (CPM)

The input of this section is always required to specify the control variables, geometry model, computation accuracy and options used in the calculation of collision probabilities.

Block-1 Control integers /20/

1 IGT Geometry type (See Figs.2.3-1 through 2.3-6)

= 1 One-dimensional sphere of multi-shells with the isotropically reflective condition at the outer boundary.

= 2 One-dimensional slab of multi-layers.

Care should be taken of the boundary condition. If IBOUND=1 is specified, not the perfect reflective (mirror) boundary condition but the periodic condition is applied for this geometry so as to treat an asymmetric lattice. On the other hand, if a symmetric lattice is considered, the full geometry must be given.

= 3 One-dimensional circular cylinder divided by concentric annuli.

= 4 Square cell with cylinder divided by concentric annuli.

A square lattice is divided by the concentric circles into several

regions. It is to be noticed that the lattice can be divided by the circle of which radius exceeds the distance from the center to the flat.

= 5 Square cell with circular cylinder of two-dimensional division.

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

= 6 Hexagonal cell with circular cylinder divided by concentric annuli.

= 7 Hexagonal cell with circular cylinder of two-dimensional division.

A hexagonal lattice is divided by the concentric circles and also by six lines crossing the central axis. Each line makes an angle of 75° with a flat of the hexagon. While an annular ring is divided into twelve pieces, because the 60° rotational symmetry is assumed, two adjacent pieces on an annular division remain as independent regions.

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

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

A pin rod cannot lie on the grid line specified by RX(*i*). Different radius by pin is accepted.

Sub-regions are numbered by the rules 1) coolant regions ahead of pin rod regions, 2) lower to upper, 3) left to right, 4) inner to outer in a pin rod.

= 10 Annular assembly with annular array of pin rods.

A circular cylindrical assembly is divided by concentric annuli specified by RX(*i*). A certain number NPIN(*j*) of pin rods are placed on circular annuli specified by RPP(*j*). They must be placed with equal spacing on their azimuthal angles because the pin rods on a circle are assumed equivalent. A pin rod is sub-divided into several concentric layers by RDP. All the pin rods have the same geometry. The pin rods may, together with the coolant, be divided further into inner and outer by the circles denoted by RPP(*j*) by an option indicator IDIVP.

Sub-region is numbered first to the inner-most pin rod starting from inner to outer of a pin rod if IDIVP<2, and from inner to outer measured

from the lattice center if IDIVP=2, then to the pin rod on the outer ring. After the outer-most pin rod, the coolant region follows from the inner to the outer.

= 11 Annular assembly with asymmetric array of pin rods.

The model IGT=10 is extended to permit an asymmetric disposition of pin rods. Any size of pin rod can be mounted at an arbitrary position as far as pin rods do not intersect each other. The coolant regions can be subdivided two-dimensionally by the concentric circles and by the radiating lines.

The numbering of Sub-regions obeys principally the same rule as IGT=10. In view of azimuthal angle, the region positioned at smaller angle comes ahead.

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

= 12 Hexagonal assembly with asymmetric array of pin rods.

A model is provided to permit a hexagonal block with asymmetrical array of pin rods. Except the shape of the outer surface, the input requirements and the rule of Sub-region numbering are same as those of IGT=11.

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

This type permits the placement of pin rods on any grid point of an X-Y division of a rectangular lattice. Every pin rod has its own radius with annular sub-division.

When IBOUND=1 is specified as the outer boundary condition, not the perfect reflective but the periodic condition is supposed in x- and y-direction. Though Sub-region numbers are purely geometrical, the user is requested to allocate T- and R-region numbers so as to satisfy the

periodic condition. For example, the region number allocated to a fuel pin located on the left edge must be coincided with to that on the right edge.

Sub-regions are numbered by the rules 1) coolant regions ahead of pin rod regions, 2) small ordinate to large ordinate, 3) small abscissa to large abscissa, 4) inner to outer in a pin rod.

= 14 Concentric layer of hexagons with pin rod array of 60° symmetry.

A hexagonal assembly is divided by several concentric hexagons. On the flat of the arbitrary hexagon, pin rods of the uniform size can be mounted. The number of pin rods on a hexagon must be a multiple of six, since the 60° rotational symmetry is assumed. They are placed with equal interval starting at a corner of a hexagon. Pin rods on a hexagon are treated to have the same fluxes.

Sub-regions are numbered by the rules 1) pin rod regions ahead of coolant regions, 2) inner to outer in an assembly, 3) inner to outer in a pin rod. But in case of IDIVP=2, pin rod regions are numbered by the distance from the central axis of the assembly.

= 15 Hexagonal assembly of 60° rotational symmetry with pin rods on triangular grid points.

Difference from IGT=14 is that 1) coolant regions are divided by triangular meshes, 2) each pin rod can have the particular size, and 3) Several trapezoidal shape regions near the outer boundary of the assembly simulate the wrapper tube and inter-assembly gap.

The isotropically reflective condition is always applied. The perfect reflective boundary condition is not supported.

Sub-regions are numbered by the rules: 1) coolant regions ahead of pin rod regions. For coolant regions, 2.1) from inner to outer measured from the central axis, 2.2) small ordinate to large ordinate, For pin rod regions, 3.1) inner to outer in a pin rod, 3.2) small ordinate to large ordinate, 3.3) inner to outer measured from the central axis.

= 16 Rectangular pillar divided by X-Y coordinates of quadrant symmetry with pin rods on grid points.

This type permits the placement of pin rods on any grid point of an x-y division of a rectangular lattice. Every pin rod has its own radius with annular sub-division. This model differs from the model IGT=13 by

applying the perfect reflective boundary conditions always on the left surface and on the lower surface.

When IBOUND=1 is specified as the outer boundary condition, the perfect reflective condition is applied on the right and the upper surfaces.

Sub-regions are numbered by the rules 1) coolant regions ahead of pin rod regions, 2) small ordinate to large ordinate, 3) small abscissa to large abscissa, 4) inner to outer in a pin rod.

2 NZ Total number of Sub-regions

3 NR Total number of T-regions

4 NRR Total number of R-regions

5 NXR Total number of X-regions

6 IBOUND Outer boundary condition of the lattice calculation

= 0 Isotropic (white) reflection

= 1 Perfect reflection (mirror)

For IGT=2 (1D slab) or IGT=13 (2D X-Y pillar) periodic condition is applied. For IGT=15, isotropic reflection is applied regardless of the input value of IBOUND.

= 2 Isolated (black)

= -1 60° rotational (applicable only for IGT=12)

Note:

It is recommended that reflective boundary condition at the outer surface for sphere (IGT=1) or cylinder (IGT=3, 10, 11) should be not perfect but white reflection.

If the fixed boundary source problem is solved by the specification of IOPT23=2, IBOUND is automatically set to isolated.

For the optically large assembly, IBOUND=0 is recommended to avoid much computation time.

7 NX Number of mesh intervals for X division (IGT=2,8,9,13,15,16)

Number of mesh intervals for R division (IGT=1,3,4,5,6,7,10,11,12,14)

8 NY Effective for IGT=11,12,13,15,16

Number of mesh intervals for Y division (IGT=13,16)

Number of mesh intervals for angular division (IGT=11,12)

Number of mesh intervals for the division along the flat of outer trapezoidal regions (IGT=15)

9 NTPIN Total number of pin rods (effective for IGT=10,11,12,13,14,15,16)

This item is calculated internally for IGT=9 by NAPIN.

For IGT=10,11,12, the pin rod on the central axis is not counted as a pin rod; then, the central pin has to be entered by RXs. For IGT=14,15, the pin rod on the central axis is counted as a pin rod. For IGT=15, enter total number of pin rods in a whole (6/6) assembly, although 60° rotational symmetry is assumed.

10 NAPIN Effective for IGT=9,10,14,15

Number of pin rods in an array on X-direction (for IGT=9).

Number of circles on which the pin rods are located (for IGT=10). The central axis is not counted for NAPIN.

Number of hexagons on which the pin rods are located (for IGT=14). The central axis is counted for NAPIN.

Number of triangular meshes on X-axis (for IGT=15) where (NX-NAPIN) are number of layers of outer trapezoidal regions.

11 NCELL Minimum number of lattices traced by a neutron path

This item is effective only for IBOUND=1 for geometries of non-circular outer shape. This is used to cut off a neutron path without knowing the actual optical path length. It is desirable to trace a neutron beyond an optical length of 6.0 if the computer time allows. Recommended value to this item is NCELL=2 for a lattice large enough in the sense of optical path, or NCELL=5 for a transparent or small lattice. The larger value causes the longer computer time. The user should not be afraid of the short cut of a path by insufficient number of NCELL while a certain amount of neutrons reach the end of the path and loose the contribution. It will be recovered by the later process of normalization and redistribution of collision probabilities.

If a negative value is entered, the computation algorithm suitable for vector

computers is employed for the numerical integration of collision probabilities.

12 IEDPIJ Edit control for collision probabilities

= 0 Skip print

= 1 Print collision probabilities

13 NGR Order of Gaussian integration for the numerical radial integration

This item is ineffective for one-dimensional slab (IGT=2). Recommended value is from 6 to 10. The computer time for the integration of collision probabilities is proportional to this item. For the geometries IGT=8,9,13,15 and 16, the Gaussian integration is replaced by the trapezoidal rule.

14 NDA Number of division of the range IBETM (described below) entered by the unit of degree for the numerical angular integration of the collision probabilities.

Required for two-dimensional integration for IGT=4 through 16. Sufficient accuracy will be attained if approximately IBETM/2 is entered as NDA.

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

15 NDPIN Number of annular division of a pin rod (effective for IGT=9 through 16)

16 IDIVP Control of sub-division by RPPs (effective for IGT= 9, 10, 11,12,14)

= 0 RPPs indicate the radial positions of pin rods.

= 1 RPPs also play the role of RX. i.e. positions of annular division.

= 2 RPPs further divide the pin rod regions into inner and outer regions.

Control of sub-division by RXs, TYs (effective for IGT=13 and 16)

= 0 with NTPIN≠0, RXs and TYs do not divide coolant region, then only one coolant region is allocated to the region except pin rod regions.

= 1 Division by RXs and TYs is effective. Usually enter IDIVP=1.

17 IBETM Range of angular integration in degree. (Effective for IGT=4 through 16)

Enter =45 in octant symmetric square geometry, =30 in hexagonal symmetry, Set a double value if IBOUND=1 is specified. Enter =360 if symmetric only on left and right planes. The value of IBETM is inefficient for one-dimensional geometry.

18 IPLOT Indicator to call plotter routine for geometry display

Plot data is stored as a PostScript file (cf. Sect.1.9, Chapter 4)

= 0 Skip plotting

= 1 Call plotter routine

Enter IPLOT=0 when IGT=13,15 or 16, because the plotter routine is not effective for these geometries.

19 IEPS Indicator for iteration control parameters and edit control

= 0 Skip input of iteration control parameters and edit control and default values are used.

= 1 Input iteration control parameters and edit control in Block-2

20 IDUM Not used

= 0

Block-2 Required if IEPS=1 in Block-1 /7,6/

Parameters for the iterative solution of linear equations for neutron fluxes by the CPM. The value in < > shows the default value used when ITMINN≤0 is specified.

When IEPS=0 is entered in Block-1, IEDIT=0 and the default value in < > are employed.

1 IEDIT Edit control

= 0 No edit

plus 1 Print reaction balance and flux distribution

plus 2 Print macroscopic cross-sections

plus 4 Print collision probabilities

plus 8 Print fixed source distribution

Note:

If the user wants to print out macroscopic cross-sections and collision probabilities, IEDIT=2+4=6.

2 ITMINN Maximum number of inner iterations per an outer iteration
<20> for the eigenvalue problem mode, and <200> for the fixed source problem mode in thermal energy region

3 ITMOUT Maximum number of outer iterations for the eigenvalue problem <60>

4 ITBG Minimum number of iterations before extrapolation for acceleration <5>

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

6 ITDM Minimum delay between extrapolation <5>

7 IPT Control of monitor print at each iteration <0>
= 0 suppress print
= 1 print record
= -1 print detailed record

1 EPSI Convergence criterion for inner iterations <0.0001>

2 EPSO Convergence criterion for outer iterations <0.00001>

3 EPSG Extrapolation criterion <0.001>

4 RELC Initial over-relaxation factor <1.2>

5 OVERX Maximum extrapolation <100.>

6 FACTOR Under extrapolation factor <0.8>

Requirement of the following input Blocks depends on the geometry type (IGT) specified in Block-1. It is summarized in Table 2.3-1.

Block-3 Required if NR<NZ /NZ/

NREG(*i*) T-region number by Sub-Region (*i*)

Block-4 Required if NRR<NR /NR/

IRR(*i*) R-region number by T-region (*i*)

Block-5 Required if NXR<NRR /NRR/

IXR(*i*) X-region number by R-region (*i*).

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

Block-6 /NRR/

MAR(*i*) Material number by R-Region (*i*)

Sequential order of a material appearing in the material specification in Sect.2.4 is used as material number.

=== Double heterogeneity problem option ===

A negative value of the material number indicates that this material is heterogeneous in the sense of the double heterogeneity[7]. For example, the coated particles dispersed in graphite matrix in the fuel compact of high temperature reactors compose the double heterogeneity. It is supposed that two composite materials of this material are specified in the preceding positions. That is, if the number -3 of an R-region appears in MAR's, the first and second materials specified in Sect.2.4 are the composites of the microscopic cell. The relevant input data are required in Block-14. By this specification, both of microscopic and macroscopic heterogeneities are treated. The microscopic heterogeneity treated in this option is restricted to two-region problems expressed by any of one-dimensional geometries (sphere, slab and cylinder) where the inner region is resonant and the outer is non-resonant. The double heterogeneity problem which does not satisfy this condition must be treated in two-step process (cf. IXMICR in Sect. 2.4).

Block-7 Required only if IGT=10 or 14 and NAPIN ≠ 0 /NAPIN/

NPIN(*i*) Number of pin rods on a circular ring (IGT=10) or on a hexagon (IGT=14).

If IGT=10, the pin rod on the central axis is not counted in NPIN. If IGT=14, the center pin rod is counted by NPIN. Default values are prepared if NPIN(1)=0 is entered for IGT=14 as 1,6,12,18,....

Block-8 /NX+1/

RX(i) abscissa, radii, or the distances from the center to the flat of hexagon or square in unit of cm. Enter RX(1)=0 always.

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

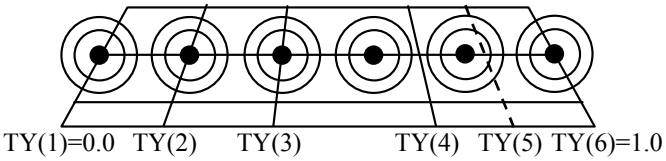
TY(i) Angular division by θ in degree

Block-9' Required if IGT=13 or 16 and if NY>1 /NY+1/

TY(i) ordinate in cm. Enter TY(1)=0 always.

Block-9" Required if IGT=15 /NY+1/

TY(i) Division in anti-clockwise direction along the flat of outer trapezoidal regions by fraction. The region boundary line is drawn from the point expressed by the fraction to the center of hexagonal assembly. TY(1)=0.0 and TY(NY+1)=1.0 always. If the user wants to divide the flat into four equal pieces, enter 0.0, 0.25, 0.5, 0.75, 1.0. If a line crosses a pin rod, the line has to pass through the center of the pin rod. In the figure below, the line specified by TY(5) is not proper.



Block-10 Required if IGT=9,10 or 14 and NAPIN≠0 /NAPIN/

RPP(i) Positions (cm) of pin rods on X-direction for IGT=9.
Radii (cm) of the circles on which pin rods are located for IGT=10.
Distances (cm) from the center to the flats of hexagons for IGT=14.

Block-10' Required if IGT=11, or 12 and NTPIN≠0 /NTPIN/

RPP(i) Radial position (cm) of each pin rod for IGT=11 or 12

Block-10" Required if IGT=13, 15, or 16 and NTPIN≠0 /NTPIN/

IXP(i) X-position of each pin rod on RX. Enter integers ranging from 1 to NX+1 for IGT=13 and =16, i.e. the lower left corner has the grid coordinate (1,1).
For IGT=15, the center of the assembly has the coordinate (0,0), then IXPs range from 0 to NX. While the input requires NTPIN entries, the entries for the first one-sixth part of a hexagon must be proper and others are dummy numbers because

the 60° rotational symmetry is assumed.

Block-11 Required if IGT=10,11, or 12 and NTPIN≠0 /NTPIN/

THETA(i) Angular position of each pin rod by θ in degree

Block-11" Required if IGT=13, 15, or 16 and NTPIN≠0 /NTPIN/

IYP(i) Y-position of each pin rod on TY. Enter integers ranging from 1 to NY+1 for IGT=13 and 16, from 0 to NY for IGT=15. While the input requires NTPIN entries, the entries for the first one-sixth part of a hexagon must be proper and others are dummy numbers because the 60° rotational symmetry is assumed.

Block-12 Required if IGT=10 or 14 /NDPIN+1/

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

Block-12' Required if IGT=9,11,12, 13, 15 or 16 and NTPIN≠0 /(NDPIN+1)*NTPIN/

RDP(i) Radii (cm) for annular sub-division of individual pin rod, where (RDP(1,j), j=1, NTPIN) =0 always.

For IGT=9, the input is required for an octant (1/8); that is, the code internally set NTPIN= NAPIN×(NAPIN+1)/2.

For IGT=15, the entries for the first one-sixth part of a hexagon must be proper and the others are dummy numbers because the 60° rotational symmetry is assumed.

Block-13 Plotter control integers, required if IPLOT in Block-1 is specified. /3/
Repeat Block-13 necessary times until ICONT=1 is entered.

1 IG Signed integer to specify the combination of required region maps; the integer is made of the summation of following integers corresponding to the kind of map.

= 0 None

plus 1 Sub-region

plus 2 T-region

plus 4 R-region

plus 8 Material number

plus 16 X-region

Note:

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

2 ISCAL Indicator of the scale of figures

= 1 One figure of diameter of 20 cm in a screen

= 2 Two figures of each diameter of 15 cm in a screen

= 3 Five figures of each diameter of 10 cm in a screen

3 ICONT Continuation indicator

= 0 Followed by the next Block-13

= 1 The last plotting

Block-14 Required if any negative MAR in Block-6 is entered /0,3,2/

Control integers for the treatment of the double heterogeneity

1 IDB Energy region indicator

= 1 Resonance region by the PEACO routine

= 2 Thermal region (not yet available)

= 3 Resonance and thermal region (not yet available)

2 IGEOM Geometry indicator of the microscopic heterogeneity

Restricted to the two-region problem where the inner region is resonant and the outer is non-resonant.

= 1 Slab

= 2 Cylinder

= 3 Sphere

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

= 1 Transmission (recommended)

= 2 Neutron from the outer region (e.g. moderator)

= 3 Neutron escaping from the inner region (e.g. absorber lump)

= 4 Simplified transmission

4 RF The thickness (cm) of absorber plate for IGEOM=1, and the outer radius (cm) of absorber lump for IGEOM=2,3.

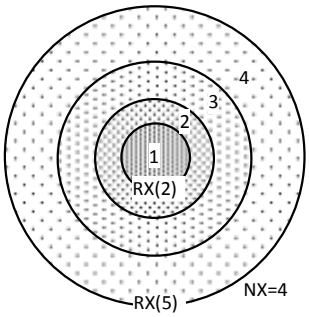
5 RM The outer radius of microscopic cell. As the escape probability is evaluated by the analytical expression by Case *et al.*,[17] the Dancoff correction factor must be fed in

the material specification in Sect.2.4 even if any is specified by IOPT4 in Sect.2.2.

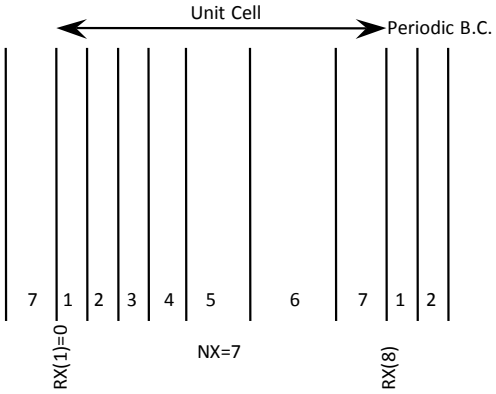
Table 2.3-1 List of input requirements for CPM routines by geometry model (IGT)

Block	IGT=	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
B-1	IGT	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○
	NZ	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○
	NR	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○
	NRR	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○
	NRX	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○
	IBOUND	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○
	NX	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○
	NY	●	●	●	●	●	●	●	●	●	●	○	○	○	●	○	○
	NTPIN	●	●	●	●	●	●	●	●	●	○	○	○	○	○	○	○
	NAPIN	●	●	●	●	●	●	●	●	○	○	●	●	●	○	○	●
	NCELL	●	○	●	○	○	○	○	○	○	●	●	○	●	○	●	○
	IEDPIJ	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○
	NGR	○	●	○	○	○	○	○	○	○	○	○	○	○	○	○	○
	NDA	●	●	●	○	○	○	○	○	○	○	○	○	○	○	○	○
	NDPIN	●	●	●	●	●	●	●	●	○	○	○	○	○	○	○	○
	IDIVP	●	●	●	●	●	●	●	●	○	○	○	○	○	○	●	○
	IBETM	●	●	●	○	○	○	○	○	○	○	○	○	○	○	○	○
	IPLOT	○	○	○	○	○	○	○	○	○	○	○	○	○	○	●	○
	IEPS	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○
	IDUM	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●
B-2	IEDT～FACTOR	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△
B-3	NREG	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△
B-4	IRR	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△
B-5	IXR	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△
B-6	MAR	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○
B-7	NPIN	×	×	×	×	×	×	×	×	×	△	×	×	×	△	×	×
B-8	RX	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○
B-9	TY	×	×	×	×	×	×	×	×	×	×	△	△	×	×	×	×
B-9'	TY	×	×	×	×	×	×	×	×	×	×	×	×	○	×	×	○
B-9"	TY	×	×	×	×	×	×	×	×	×	×	×	×	×	×	○	×
B-10	RPP	×	×	×	×	×	×	×	×	△	△	×	×	×	△	×	×
B-10'	RPP	×	×	×	×	×	×	×	×	×	×	×	△	△	×	×	×
B-10"	IXP	×	×	×	×	×	×	×	×	×	×	×	×	△	×	△	△
B-11	THETA	×	×	×	×	×	×	×	×	×	△	△	△	×	×	×	×
B-11'	IYP	×	×	×	×	×	×	×	×	×	×	×	×	△	×	△	△
B-12	RDP	×	×	×	×	×	×	×	×	×	△	×	×	×	△	×	×
B-12"	RDP	×	×	×	×	×	×	×	×	△	×	△	△	△	×	△	△
B-13	IG～ICONT	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△
B-14	IDB～RM	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△	△

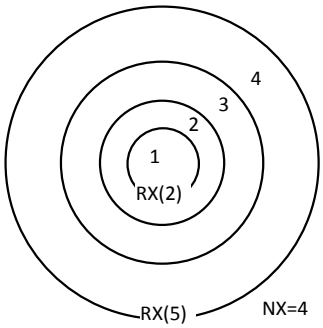
○: always required, ●: always required but ineffective, △: conditionally required, ×: not required



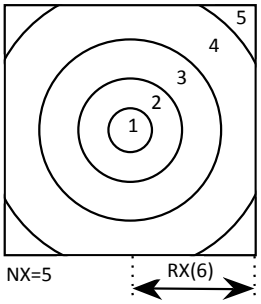
IGT=1 (Spherical cell)



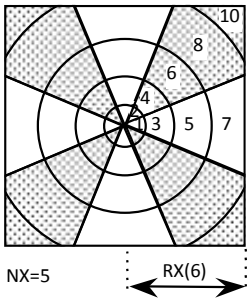
IGT=2 (Infinite slab cell)



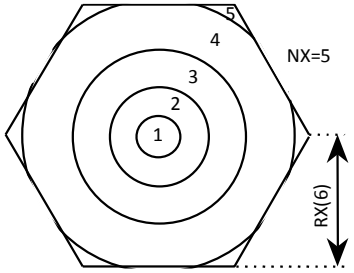
IGT=3 (Cylindrical cell)



IGT=4 (Square cell)

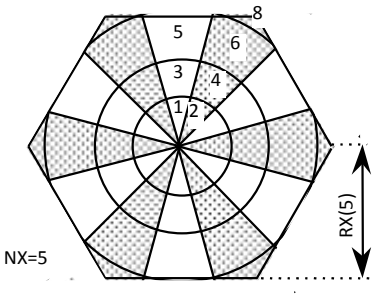


IGT=5 (2D square cell)

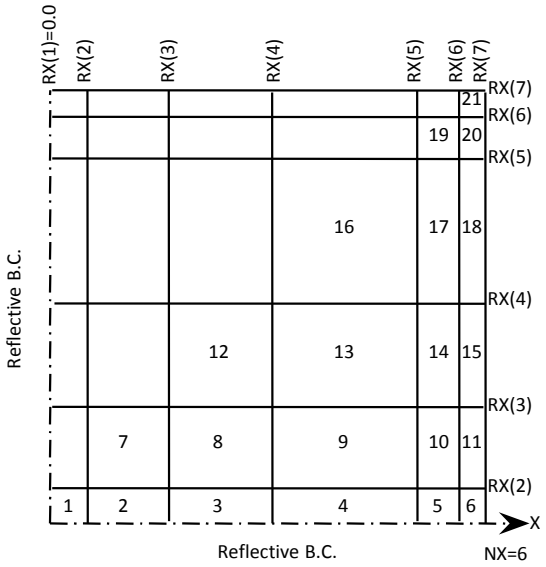


IGT=6 (Hexagonal cell)

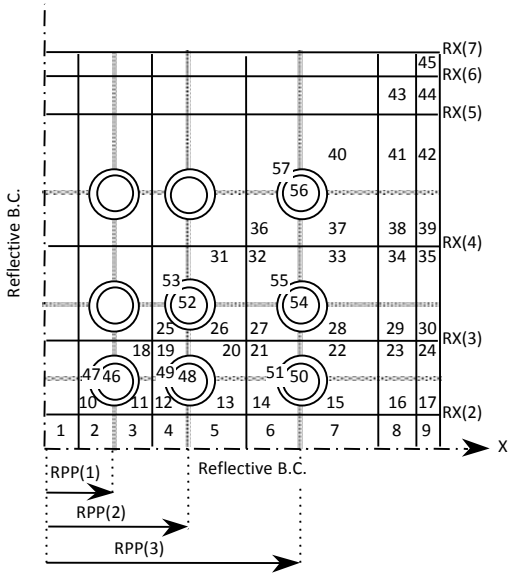
Fig.2.3-1 Geometries for PIJ (IGT=1~6)



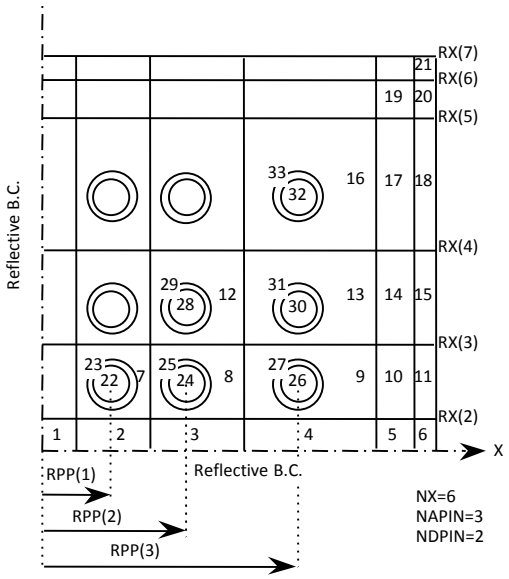
IGT=7 (2D hexagonal cell)



IGT=8 (Quadrant of an octant symmetric assembly)

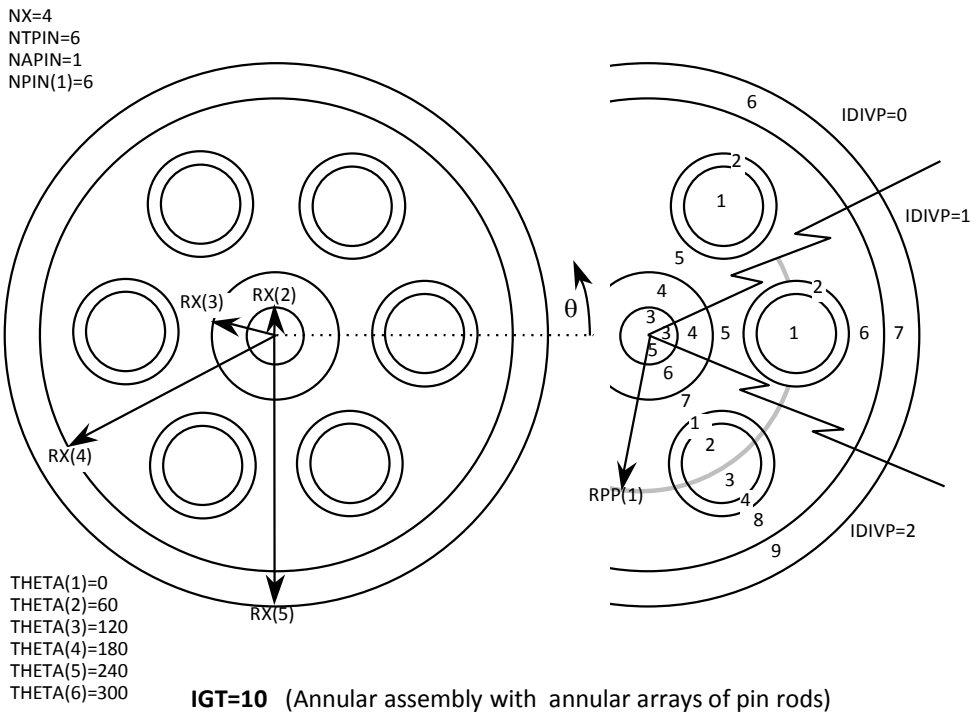


IGT=9 (Octant symmetric square assembly with pin rods) IDIVP=1

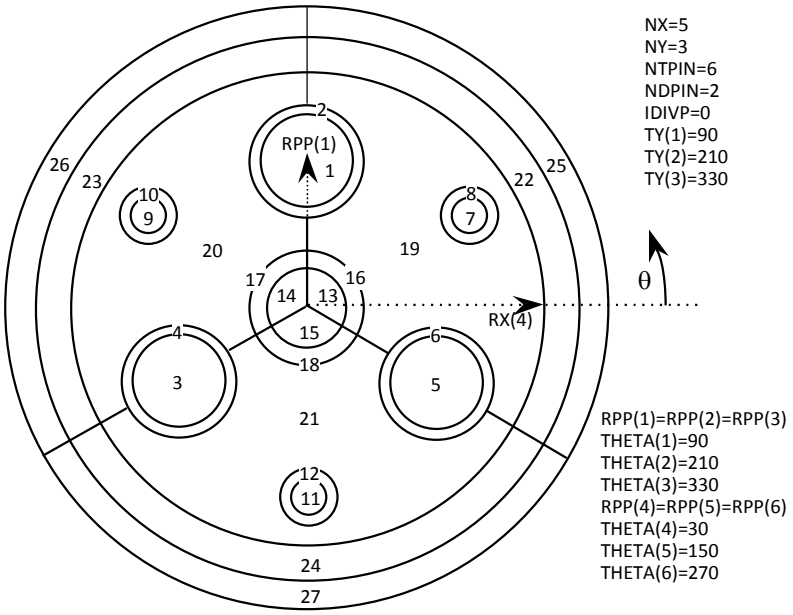


IGT=9 (Octant symmetric square assembly with pin rods) IDIVP=0

Fig.2.3-2 Geometries for PIJ (IGT=7~9)

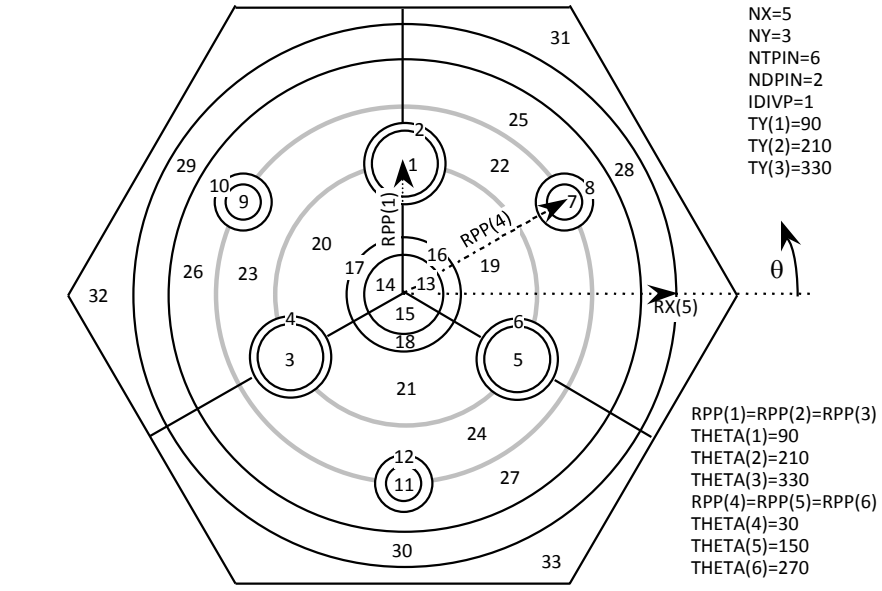


IGT=10 (Annular assembly with annular arrays of pin rods)



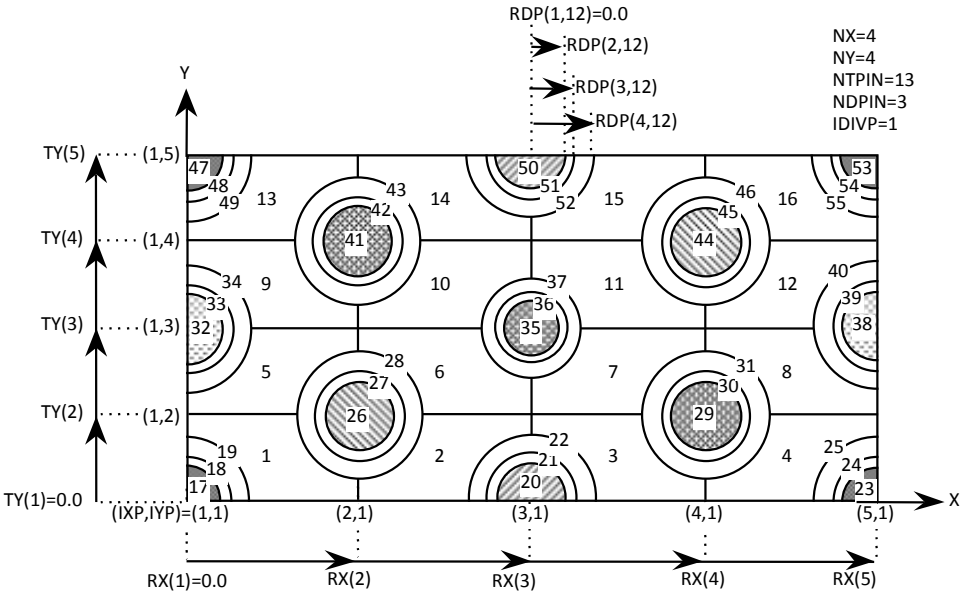
IGT=11 (Annular assembly with asymmetric pin rods)

Fig.2.3-3 Geometries for PIJ (IGT=10, 11)



If IDIVP=1, RPP(i) must not accross the pin rods on RPP(j) different from RPP(i).

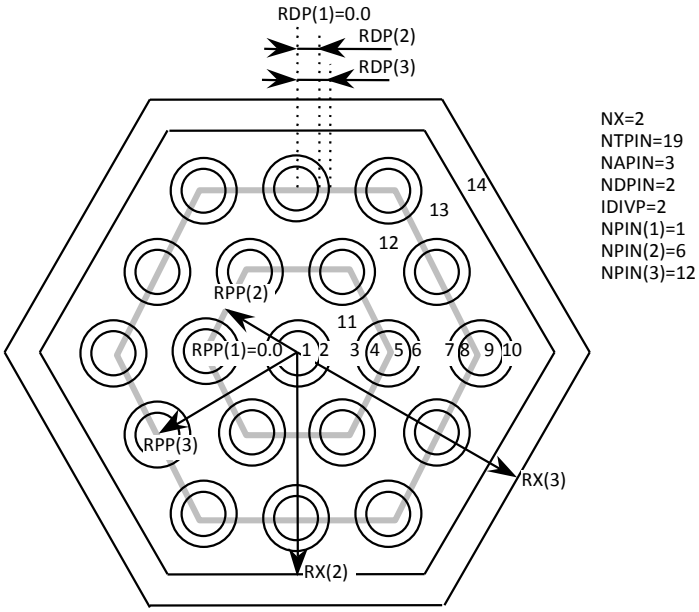
IGT=12 (Hexagonal assembly with asymmetric pin rods)



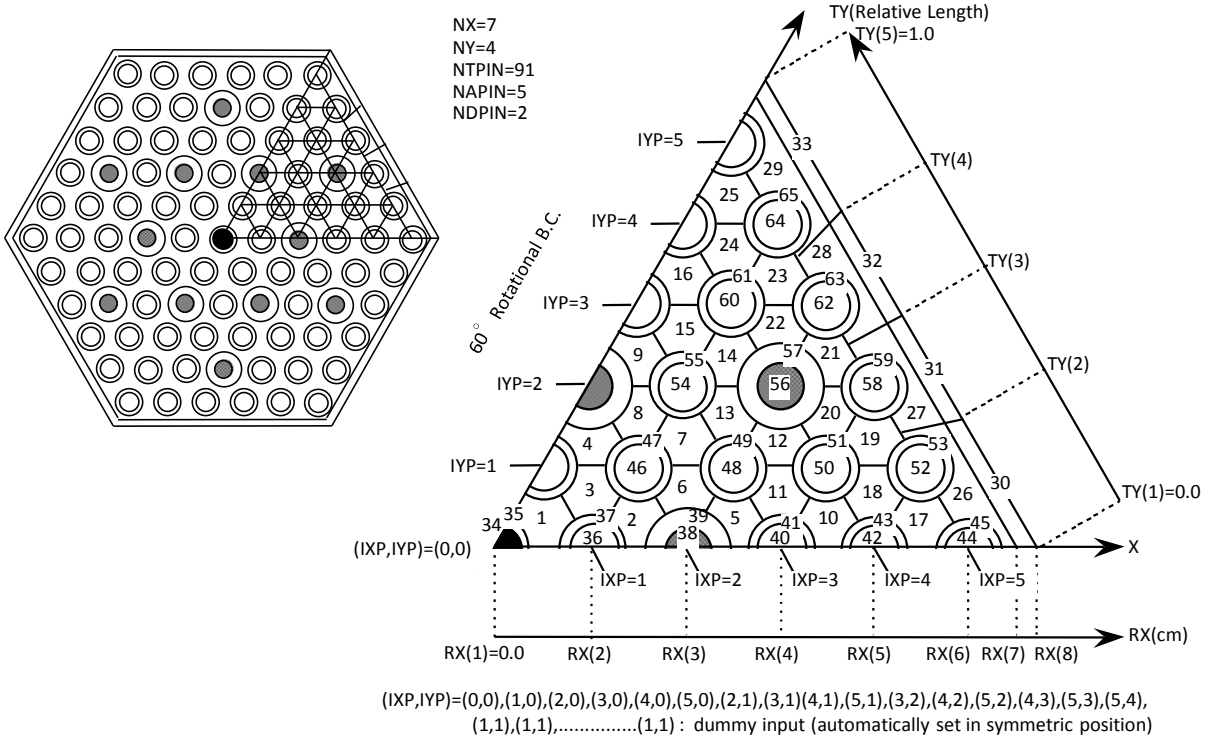
Sub-Region number for each pin rod is in order of (IXP,IYP).
(IXP,IYP)=(1,1), (3,1),(5,1),(2,2),(4,2),(1,3),(3,3),(5,3),(2,4),(4,4),(1,5),(3,5),(5,5)

IGT=13 (X-Y 2D cell with pin rods on arbitrary grid points)

Fig.2.3-4 Geometries for PIJ (IGT=12, 13)

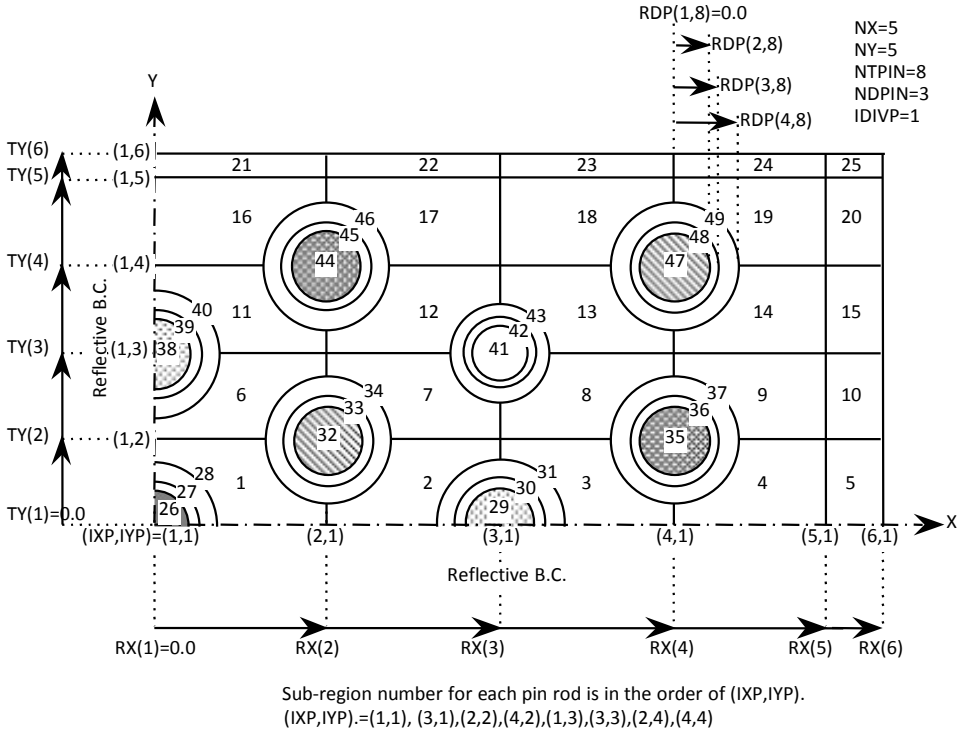


IGT=14 (Concentric layers of hexagons with equi-distant pin rod arrays)



IGT=15 (Hexagonal assembly with triangular pin rod arrangement)

Fig.2.3-5 Geometries for PIJ (IGT=14, 15)



IGT=16 (X-Y 2D symmetric cell with pin rods on arbitrary grid points)

Fig.2.3-6 Geometries for PIJ (IGT=16)

2.4 Material Specification

'Material' in this section has a set of its own inherent macroscopic cross-sections. Therefore, material name denoted by 8 characters expresses the name of material of which cross-sections have been already formed or will be formed on fine-group macroscopic cross-section files. The fine-group macroscopic cross-sections are supplied as member files of MACROWK or MACRO-IN, but not supplied from MACRO, which keeps the output data of collapsed-group macroscopic cross-sections from a current job. It should be noted that all materials in multiple cases in a job have the macroscopic cross-sections with the same energy group structure except for those in MACRO. The member files of MACRO can be treated as the materials for the next job by supplying them from MACRO-IN.

Several kinds of mixtures, as follows, are included in material.

- A mixture with composition of which macroscopic cross-sections will be formed in this case.
- A mixture of which macroscopic cross-sections were formed in the previous case and are kept in MACROWK. The composition is not given in this case.
- A mixture of which macroscopic cross-sections were formed in the previous job and are supplied from MACRO-IN in this job. The composition is not given in this case.
- A homogenized material with a CASEID label of which macroscopic cross-sections were formed in the previous case and kept in MACROWRK. The former half of the material name (member name) is the case name of the previous case.
- A homogenized material with a CASEID label of which macroscopic cross-sections were formed in the previous job and stored in MACRO, and are supplied from MACRO-IN in this job. The former half of the material name (member name) is the case name in the previous job.

Block-1 /1/

NMAT Number of materials to be specified hereafter.

Block-2 /A8, 2, 3/

1 MTNAME Material identification expressed by 8 characters, composed of three tags as '*matnm-bb-p*' or four tags as '*case-x-bb-p*' which appears as a member name in macroscopic cross-section files (MACROWK or MACRO-IN).

matnm-tag Material identification of 5 characters; the first character must be any alphabetic character.

case-tag Tag to specify a set of X-region averaged cross-sections formed in the previous case or job. Enter the CASEID (4 characters) of the previous case or job. (e.g. CELL, UO2A, PWR1, etc.), together with *x*-tag and *bb*-tag described below.

Otherwise, *matnm*-tag is used as the former 5 characters of member name on MACROWK or MACRO-IN to discriminate the member for a mixture. Therefore, do not use more than once the same *matnm*-tag with the same *bb*-tag.

x-tag *x*-tag is used only when it is used together with *case*-tag to specify a member file in MACROWK or MACRO-IN. Enter an alphanumeric character (1, 2,….,9, A, B, … Z) corresponding to X-region number (1, 2,….,9, 10, 11, … 35) to specify an X-region among a set of homogenized cross-sections formed in the previous case or job.

bb-tag Tag to indicate burn-up step as ‘00’ , ‘01’, ‘02’, ... and ’99’ corresponding to fresh, steps 1, 2,..... and ’99’, respectively, when the lattice burn-up calculation is done to make up this material. Otherwise this tag is filled by the character given in the input. When the user wants to refer the cross-sections obtained in a series of burn-up calculations, he can specify the burn-up step by using this tag.

p-tag Tag internally filled to specify Legendre component or delayed neutron data of the fine-group cross-section set. Any character given in the input is replaced by the code.

= ‘0’ P_0 component

= ‘1’ P_1 component

= ‘D’ delayed neutron data

2 KCOREK Indicator to assign this material to the resonant material in the PEACO calculation This term is effective when IOPT5=2.

The PEACO routine has a restriction that one or two independent resonant material(s) can be treated, where a resonant material is defined as a material containing at least one nuclide of which cross-sections are stored in the MCROSS file. When more than two resonant materials exist in the system, they must be classified into one of two groups by their characteristics.

= 0 Non-resonant material

= 1 Assigned to the first resonant material

= 2 Assigned to the second resonant material

Note:

The PEACO routine calculates the collision probabilities by interpolation of tabulated probabilities by using the total cross-section as argument[3),5)]. Such tabulation can be prepared for the combination of two different total cross-sections. This is the reason for the restriction. If the third resonant material has almost the same energy dependence of total cross-section as any of the two resonant materials, an approximation to assume the third is equivalent to one of the two holds. Most of resonance absorption is, as known, contributed by U-238. When the resonant materials are mainly composed by U-238, the extension to permit more than two resonant materials can be applied.

When a depletion problem is solved for a multi-region lattice, several compositions which have been uniform at the clean stage have to be considered to be different from each other. The similarity of cross-sections can permit the above-mentioned approximation.

[cf.] Sect.1.4, Sect.5.4

3 NISO Number of nuclides to compose the mixture (given by Block-3).
Enter NISO=0 to specify a set of cross-sections which were already formed in the MACROWK or MACRO-IN file. Therefore, NISO must be always zero when the energy group structure of fine-group cross-sections is different from that of the Public Library.

4 TEMP Physical temperature of the material in Kelvin (K).
Any temperature is accepted for the interpolations of 1) the THERMAL library and 2) the resonance self-shielding factor, and 3) Doppler broadening of the Public MCROSS Library, unless the extrapolation is required.
[cf.] Sect.1.2.4, Sect.1.4, Sect.8.2

5 XL The mean chord length l(cm) of the resonance absorber lump used to include heterogeneous effect in the interpolation of resonance self-shielding factor in the NR approximation, and also used in PEACO for the constant to yield a non-dimensional blackness for the interpolations of the collision probabilities.
Generally l is defined as

$l = 4\ V / S$,

where V is the volume of the material, and S is the surface area.

For a one dimensional slab of thickness a,

$l = 2\ a$.

For a one dimensional sphere of radius a,

$l = 4\ a / 3$.

For a one dimensional cylinder of radius a,

$l = 2\ a$.

For a hollow cylinder of inner radius a and outer radius b,

$l = 2\ b\ (\ 1 - (\ a / b\ )^2\ )$.

Even when a fuel pellet is divided into several material regions (e.g. segmented burn-up regions for Gd_2O_3-UO_2), l=2a (a: outer radius of pellet) is recommended. For more complicated geometry which contains several absorber lumps in a unit lattice, enter the mean chord length of a dominant absorber, for example, that of a pin rod in the case of a BWR fuel assembly.

Note:

When IOPT4=0, the calculation of the nuclide-dependent Dancoff correction factor[6)] is applied to the nuclides which have the MCROSS library data, because the library data are prepared for the nuclides whose resonance shielding effects are important. Even if the MCROSS library data is not prepared, the nuclide-dependent Dancoff correction factors are calculated to the constituent nuclides of the structural material such as fuel cladding when the following conditions are satisfied.

- IOPT4=0 and XL>0
- Isotopes of Al, Mn, Cr, Fe, Ni, Cu, Zr, Mo and W
- Atomic number density of the nuclide is greater than 5.0×10^{-3} (n/barn-cm)

If this is not required, enter XL=0.0.

6 DC The Dancoff correction factor C (cf. Sect.1.4)

(effective only if IOPT4=4 in Block-3 of Sect.2.2 is specified)

Enter DC=1.0 for the homogeneous approximation (no shadow effect).

Remember that l and C are used in a conventional table-look-up for the heterogeneous effect to the background cross-section σ_0 in the following form;

$$\sigma_{0,n}^{i} = \frac{1}{N_n^i}\sum_{m\neq n}(N_m^i \sigma_{t,m}^i) + \frac{g(C_n^i)(1-C_n^i)}{N_n^i l^i} \quad \text{for resonant material } (i) \qquad (2.4\text{-}1)$$

where $g(C)$ is a geometric correction factor to the simple NR approximation, n and m denote the resonant nuclide under consideration and any nuclides admixed in the material, respectively.

Block-3 Required if NISO > 0 NISO times /A6 or A13, 2, 1/

1 IDENT Nuclide identification expressed by six [or optionally 13] characters composed of three [or optionally five] tags like '*zz-mmm-c*[*-matnm-bb*]'.
e.g. U02350 or U02350FUEL100

zz-tag Element symbol of the nuclide (cf. Sect.8.1)
Enter 'PU' for plutonium. For a nuclide with single character element such as H for hydrogen, add character '0' to complete the tag as 'H0'.

mmm-tag The three digit of the mass number is used to discriminate the isotopes as '239' for Pu-239. '000' is used to specify the element with natural abundance like 'FE000' for natural iron.

c-tag Chemical compound status tag to select the proper thermal scattering law (cf. Sect.8.1). Use '0' or '1'for nuclide to treat with free gas model. The '0' denotes the nuclide in ground state and '1' does the nuclide in the first meta-stable state'.
e.g. H0001H for H in H_2O, AM2420 for Am-242g, AM2421 for Am-242m

The following tags can be optionally used to specify the effective microscopic cross-section set which has been already formed and stored in MICRO or MICRO-IN in the previous case or job.

[*matnm*-tag] Material identification of 5 characters of the previous case or job.

[*bb*-tag] Burn-up step tag of the previous case or job.

2 IPCO Resonance cross-section treatment indicator for PEACO
In the PEACO routine, the nuclide which has the MCROSS library data is taken as a resonant nuclide, and otherwise it is taken as a non-resonant nuclide. The nuclide whose atomic number density is less than or equal to 1.0×10^{-15} (10^{24} n/cm^3) is

neglected in the PEACO calculation, unless IPCO=3 is specified.

= 0 If this nuclide is a constituent of non-resonant material, the tabulation of collision probabilities (P_{ij}) will be made by assuming a constant cross-section of the highest energy group of the PEACO calculation. (popular option for all nuclides)

= 1 If this nuclide is a constituent of non-resonant material, the tabulation of collision probabilities (P_{ij}) will be made by each fine-group considering the change of cross-section of this nuclide by group.

= 2 If this nuclide is a constituent of non-resonant material. Two-dimensional P_{ij} tabulation will be made assuming the behavior of cross-section of this nuclide to be $1/\upsilon$.

= 3 If this nuclide is a resonant nuclide which has the MCROSS library data, it is treated as a resonant nuclide even if its atomic number density is less than or equal to 1.0×10^{-15} (10^{24} n/cm^3).

Note:

In the tabulation of P_{ij} for PEACO calculation, the cross-sections of non-resonant nuclides are assumed as constant by using the value of the highest energy group of PEACO calculation, unless IPCO=1 or IPCO=2 is specified.

For the systems involving coolant water with boron or control rod, the specification of IPCO=1 or IPCO=2 for B-10 gives more exact treatment. However, treatment by IPCO=1 requires the P_{ij} tabulation by energy group and that by IPCO=2 does two-dimensional tabulation. Both treatments need much computer time.

3 IXMICR Indicator to write the effective cross-sections into the effective microscopic cross-section file (MICRO)

= 0 No edit.

= 1 Write the effective microscopic cross-sections into the MICRO file.

= 2 Write the X-region averaged effective microscopic cross-sections into the HOMO-MIC file, when IOPT9=2 is specified in Block-3 of Sect.2.2.

= 3 both of functions of IXMICR=1 & 2.

Note:

When IOPT9=1 is specified, the X-region averaged effective microscopic cross-sections of all nuclides are written into HOMO-MIC, independently of

IXMICR. In the case of burn-up calculation, IOPT9=1 is effective to all nuclides described in the burn-up chain model (See Sect.8.9), e.g. fission products or minor actinides. To apply the function of IXMICR>0 to the selected burn-up nuclides by IOPT9=2, a user has to specify the nuclides in this Block even if their atomic number densities are zero. For the nuclide which has no cross-section data but is used in the burn-up calculation, give a negative value for IXMICR.

4 DN Atomic number density (10^{24} n/cm^3)

Note:

DN=0.0 is acceptable.

In the double heterogeneity problem, enter the atomic number density averaged in the microscopic cell to the nuclide to which the microscopic cross-sections of hyperfine-group structure are written in UMCROSS.

Repeat Block-3 NISO times.

Repeat Block-2 through Block-3 NMAT times.

An example of material specification to the system of UO_2 fuel, zircaloy cladding and coolant water is shown below, where the Dancoff correction factor is null as it is internally calculated by specifying IOPT4=0.

```
---------+---------+---------+---------+---------+--------
3   / NMAT
FUEL1XXX  0 4  849.12 0.82 0.0    / 1 :UO2 FUEL
U02350      0 1  1.05692E-03  / 1
U02380      0 1  2.21466E-02  / 2
XE1350      0 3  0.0          / 3 dummy (F.P.)
O00160      0 0  4.64071E-02  / 4
CLAD1XXX  0 3  586.88 0.13 0.0    / 2 : CLADDING
ZRN000      0 0  3.80326E-02  / 1
CRN000      0 0  6.71520E-05  / 2
FEN000      0 0  1.31290E-04  / 3
MODE1XXX  0 2  578.15 1.0  0.0    / 3 : WATER MODERATOR
H0001H      0 0  4.78704E-02  / 1
O00160      0 0  2.39352E-02  / 2
---------+---------+---------+---------+---------+--------
```

The following is an example of material specification in which the effective microscopic cross-sections formed in the previous case or job are used (less common).

```
---------+---------+---------+---------+---------+--------
3   / NMAT
FUEL2XXX  0 3  849.12 0.82 0.0    / 1 : UO2 FUEL
U02350FUEL1XX 0 1  1.05692E-03 / 1
```

```
U02380FUEL1XX 0 1  2.21466E-02 / 2
O00160FUEL1XX 0 0  4.64071E-02 / 3
CLAD2XXX  0 3  586.88 0.13 0.0    / 2 : CLADDING
ZRN000CLAD1XX 0 0  3.80326E-02 / 1
CRN000CLAD1XX 0 0  6.71520E-05 / 2
FEN000CLAD1XX 0 0  1.31290E-04 / 3
MODE2XXX  0 2  578.15 1.0  0.0    / 3 : WATER MODERATOR
H0001HMODE1XX 0 0  4.78704E-02 / 1
O00160MODE1XX 0 0  2.39352E-02 / 2
---------+---------+---------+---------+---------+--------
```

The ‘isolated material’, which is not allocated to any R-region, can be specified in Block-2 for later core calculations (e.g. water reflector surrounding a core). The macroscopic cross-section of the isolated material is formed from the infinite dilution cross-sections of component nuclides and it is collapsed with a neutron spectrum obtained by solving the B1 equation. The 4-th material of the following sample is an example for the isolated material to generate the macroscopic cross-section of water reflector.

```
---------+---------+---------+---------+---------+--------
4   / NMAT
FUEL1XXX  0 4  849.12 0.82 0.0    / 1 :UO2 FUEL
U02350      0 1  1.05692E-03  / 1
U02380      0 1  2.21466E-02  / 2
XE1350      0 3  0.0          / 3 dummy (F.P.)
O00160      0 0  4.64071E-02  / 4
CLAD1XXX  0 3  586.88 0.13 0.0    / 2 : CLADDING
ZRN000      0 0  3.80326E-02  / 1
CRN000      0 0  6.71520E-05  / 2
FEN000      0 0  1.31290E-04  / 3
MODE1XXX  0 2  578.15 1.0  0.0    / 3 : WATER MODERATOR
H0001H      0 0  4.78704E-02  / 1
O00160      0 0  2.39352E-02  / 2
REFL1XXX  0 2  578.15 1.0  0.0    / 4 : ISOLATED MATERIAL
H0001H      0 0  4.78704E-02  / 1
O00160      0 0  2.39352E-02  / 2
---------+---------+---------+---------+---------+--------
```

2.5 Lattice Burn-up Calculation

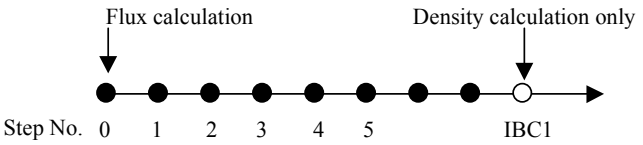
The input of this section is required if IOPT7=1 (Sect.2.2) is specified for lattice burn-up calculation. The information such as burn-up chain scheme, yield, decay constant, power per fission, etc. is read from the file of logical unit 50. Several files are prepared for the optional use of burn-up chain models. As these files are written in text format, the user can easily modify or compile his own file. The contents and structure are described in Section 3.2.

Block-1 Control integers for optional use /20/

IBC1 Final burn-up step number (<MXSTEP)

The value of IBC1 is the same as the number of burn-up steps, unless it is for restart burn-up calculation activated by negative IBC3 below. The MXSTEP(=100) is the upper limit defined by the code. In the restart calculation, IBC1 must be larger than the step number that the previous calculation stopped at.

A burn-up step number for edits starts from zero, which corresponds to fresh fuel condition. At each burn-up step point, flux calculation is carried out and results are printed and stored in the PDS files. At the last step point (step number=IBC1), flux calculation is skipped although fuel composition is calculated. If results like k_{eff} at the last step point are necessary, a dummy step point is required at the last.



The *bb*-tag (cf. Chapter 6) of each member in PDS files is named as follows:
00, 01, 02, 03,….., 09, 10, 11,…., 99.

IBC2 Unit of burn-up for the entries of Block-3

= ±1 Integrated burn-up in MWd/t (MWday per 10^3kg of initial heavy metals)

= ±2 Integrated burn-up in MWd

= ±3 Integrated burn-up in day

= ±4 Burn-up interval of each step in day

= ±5 Depletion fraction of U-235 atomic number density to the fresh one in percent (0.0~100.0)

Note:

If the negative number is entered for IBC2, the nuclide for depletion fraction in output edit is changed from U-235 by that specified in Block-4. Especially with IBC2=-5, the burn-up step can be defined by depletion fraction of the specified nuclide.

IBC3 Burn-up mode

= ±1 Normal burn-up calculation

= ±2 Branch-off calculation (entry Block-5 is required)

= ±3 Burn-up calculation under constant flux

= ±4 Burn-up calculation by reading initial composition from Block-6

Note:

If the negative number is entered for IBC3, restart calculation is executed to succeed the last step of the previous calculation of the same case name.

The following burn-up calculation modes can be selected here.

- Normal burn-up calculation: The depletion equation is solved by assuming constant power level in each burn-up step interval.
- Branch-off calculation: the depletion equation is not solved. After the normal burn-up calculation, lattice calculation is performed by using the fuel composition obtained by the normal burn-up mode but in the different lattice conditions (e.g. temperature, void fraction, boron concentration, etc.)
- Constant flux calculation: In the blanket fuel, power density increases in proportion to accumulation of produced fissile nuclides. In such a case, the normal burn-up calculation model may be not proper. In this mode, a burn-up calculation is performed with the constant flux level determined by the initial power or the constant flux level given by input. In the latter case, burn-up calculation is possible for the lattice including burnable nuclides but no fissionable ones.
- Burn-up calculation by reading initial composition: After a normal burn-up calculation, a new case of burn-up calculation can be done in different lattice conditions with the fuel composition taken from the normal calculation at an arbitrary burn-up step.

IBC4 Edit option

= 0 Brief edit (for usual case, summary is printed on the 98-th device)

= 1 Detailed edit (one group microscopic cross-sections are printed on the 99-th device)

= 2 Information for debugging

= 3 Detailed information for debugging

IBC5 Definition of instantaneous conversion ratio

= 0 Use default definition of conversion ratio

= 1 Conversion ratio is redefined by entries of Block-7-1 through -7-3

Note:

Generally, an instantaneous conversion ratio is defined as the ratio of production rate to depletion rate of fissile nuclides as follows:

$$\text{Instantaneous CR}(t) = \frac{\text{FISSP}(t)}{\text{FISSD}(t)} \tag{2.5-1}$$

The user can flexibly define the numerator and the denominator of Eq.(2.5-1). They are calculated by Eqs.(2.5-2) and (2.5-3), respectively. (Both are the same function forms)

$$\text{FISSP}(t) = \int \left[\sum_{g} \sum_{i \in Specified} \left\{ \alpha^i < N^i(t) > \sigma^i_{x(i),g}(t)\, \Phi_g(t) + \beta^i \lambda^i N^i \right\} \right] dV \tag{2.5-2}$$

$$\text{FISSD}(t) = \int \left[\sum_{g} \sum_{j \in Specified} \left\{ \alpha^j < N^j(t) > \sigma^j_{x(j),g}(t)\, \Phi_g(t) + \beta^j \lambda^j N^j \right\} \right] dV \tag{2.5-3}$$

where,

i , j : any nuclide selected from the burn-up chain library used,

α , β : arbitrary factors defined by the user, which are used to consider branching ratios or to cut off unnecessary terms (e.g. $\beta = 0.0$),

$< N^i >$: atomic number density of nuclide i for macroscopic or 1.0 for microscopic reaction rates,

$\sigma^i_{x(i)}$: microscopic cross-section of nuclide i for reaction type x. The reaction type is determined by the user depending on nuclide i,

λ : decay constant of each nuclide whose value is defined in the burn-up chain library. (cf. Sect.3.1)

Furthermore, the integrated conversion ratio is also calculated as follows:

$$\text{Integrated CR}(t) = \frac{\int_0^t \text{FISSP}(\tau)d\tau}{\int_0^t \text{FISSD}(\tau)d\tau} \quad (2.5\text{-}4)$$

Thus the user can calculate his own microscopic or macroscopic reaction rates and their ratio by the option IBC5=1.

The default definition of conversion ratio follows the description in the burn-up chain library. As the initial setting, it is defined as the ratio of the summation of capture reaction rates of Th-232 (if any in chain), U-234 (if any), U-238 and Pu-240 to the summation of absorption reaction rates of U-233 (if any), U-235, Pu-239 and Pu-241 with no decay effect.

IBC6 Option to write effective microscopic cross-sections into MICRO file

= 0 No cross-sections are written except for the nuclides specified by IXMICR (Sect.2.4). This option is suggested if MICRO file is not preserved.

= 1 Write cross-sections of the nuclides specified by Block-8

= 2 Write cross-sections of the whole depleting nuclides

= 3 Write cross-sections of the whole burnable resonant nuclides

Note:

In such a case that the numbers of depleting materials, burn-up steps and/or depleting nuclides are so large as to cause shortage of core memory for the virtual PDS file, keep the requirement to write cross-sections to a minimum.

[cf.] Block-3 in Sect.2.2, Block-3 in Sect.2.4

IBC7 Option to make up collision probability table for PEACO calculation (effective if IOPT5=1 or 2 is specified in Sect.2.2)

= 0 Update at each burn-up step (suggested)

= 1 Make up tables at first two steps and use the second one after that

Note:

Option IBC7=1 is used to reduce large CPU time.

IBC8 Maximum length (LCHA) of decay chain in linear analysis

= 0 Default value (LCHA=6) is used.

> 0 This input value is used for LCHA. If large LCHA is used, long CPU time and large core memory are required, as the number of chains under

consideration is enlarged.

IBC9 Option to fix atomic number densities of specified nuclides

= 0 Calculate atomic number densities of all nuclides

= *N* Replace atomic number densities of *N* nuclides by the values specified by Block-11.

Note:

This option is used, for example, to prepare Xenon free cross-sections for a branch-off calculation, or to take account of on-line decontamination in the normal burn-up calculation. $N \leq$ MXREPL must be satisfied. The value of MXREPL is defined by a parameter statement in the include file.

IBC10 Option to specify depleting material

= 0 Any material containing at least one nuclide appearing in burn-up chains is automatically assumed to be depleting material.

= 1 The user specifies depleting materials in Block-9

Some burn-up chain model assumes such a material as cladding, boric water or control rod to be depleting although the user does not want to treat as depleting. This option is used to avoid it.

Note:

Especially pay attention to boric water, because B-10 is treated as a burnable nuclide in any burn-up chain model.

IBC11 Option to produce data for pseudo fission products

= 0 Use the fission yield data described in the file of burn-up chain model, which is read from the file of logical unit 50.

= *N* Replace all fission yield data by the *N*-th type of data described in the burn-up chain model.

Note:

Enter IBC11=0 for usual usage. This option is intended to produce a special yield data for pseudo fission products by developers.

IBC12 Option for the Predictor-Corrector method (PC-method)

= 0 Not use the PC-method

≠ 0 Use the PC-method, which is applied to the burn-up steps specified in Block-12.

Note:

Computation time for a step of lattice calculation with the PC-method is about twice of that without the PC-method. However, the method is usually effective to reduce the number of burn-up steps to achieve required accuracy when burnable poisoned fuels such as UO_2-Gd_2O_3 are included in a lattice[12].

When IBC12 is given a negative value, the results of intermediate step (i.e. result before the PC-method is not applied) are printed on the 99-th device.

IBC13 Option for the blanch-off calculation (effective when IBC3=±2)

= 0 Blanch-off calculation is carried out at all burn-up steps of the normal burn-up calculation.

= *NBR* Blanch-off calculation is carried out at *NBR* burnup steps specified in Block-13.

IBC14 Option to change the number of time intervals (sub-steps) in each burn-up step

= 0 Use the default value (=20) for all burn-up steps

> 0 Use the input value specified in Block-14 by burn-up step.

IBC15 =0 (not used)

IBC16 =0 (not used)

IBC17 =0 (not used)

IBC18 =0 (not used)

IBC19 =0 (not used)

IBC20 =0 (not used)

Block-2 Thermal power (MW) /IBC1/

(POWERL(*i*), *i*=1,IBC1)

Thermal power (MW) in the whole lattice at each burn-up step interval

For example, in the case of the two-dimensional lattice containing 4 pin rods with average linear power of 170 W/cm, enter 170×4×1.0E-6=6.80E-4 as POWERL.

For a one-dimensional plane geometry, assume unit length for two transverse directions, and enter {(average thermal power density [MW/cm^3]) × (lattice thickness [cm]) × 1cm × 1cm } as POWERL.

The geometry model in the collision probability method should be taken care of. For example, as the models for IGT= 8, 9, 16 calculate the collision probabilities for a quarter of a fuel assembly, enter 1/4 of assembly power as POWERL. The model for IGT=15 calculates them for a whole assembly while it assumes 60 degree rotational symmetry. Then enter the whole assembly power.

If POWERL(*i*)=0.0 is entered, the cooling calculation (reactor stop) is executed, where only radioactive decay is considered.

If IBC3=±3 is entered, the average flux in fuel zone corresponding to the initial thermal power POWERL(1) is applied through the whole steps. The thermal power at each step POWERL(*i*) after the initial step is calculated by the code regardless of input values.

Block-3 /IBC1/

(PERIOD(*i*),*i*=1,IBC1)

Burn-up period (exposure) by step in unit specified by IBC2

If cooling calculation steps (POWERL(*i*)=0) are included is and if burn-up unit is not day (IBC2=±1, ±2, ±5), enter cooling period (day) by negative value for the step of POWERL(*i*)=0 regardless of IBC2.

Block-4 Required if IBC2<0 /A8/

STDNUC Name of the nuclide to calculate depletion fraction of the atomic number density. Enter 8 characters as ‘U02330__’ or ‘PU2390__’ (cf. Sect.8.1). Any character is accepted for the last two characters (usually two blanks are entered).

Note:

The default ‘U02350__’ must be changed, if U-235 is not included in the fresh fuel, to avoid zero division.

Block-5 Required if IBC3=±2 /A4/

CASBRN Enter case name by 4 characters to refer in the branch-off calculation.
In the branch-off calculation a series of lattice calculations are executed without depletion calculation by reading the composition of each burn-up step from the existing member; 'CASBRN'BNUP on MACRO file.
[cf.] Sect.6.2

Block-6 Required if IBC3=±4 /A4, 1, 0/

1 CASINT Enter case name with 4 characters to read the initial composition. The composition at each burn-up step is stored in the existing member; 'CASINT'BNUP on the MACRO file.

2 INTSTP Step number to read the composition. Enter 0 if the burn-up step tag (*bb*-tag) is '00'. The *bb*-tags (cf. Sect.6.2) correspond to burn-up step numbers as:
(*bb*-tag=00, 01, 02,…,99 ⇒ Step number=0, 1, 2,…,99)

Block-7-1 Required if IBC5=1 /2/

1 NFIS Number of nuclides contributing to depletion of fissile nuclides
See Eq.(2.5-3) and burn-up chain model (cf. Sect.3.1)

2 NFER Number of nuclides contributing to production of fissile nuclides
See Eq.(2.5-2) and burn-up chain model (cf. Sect.3.1)

Block-7-2 Required if IBC5=1 /A8, 1, 1/

1 NAMFIS Nuclide name and reaction type (with 8 characters) contributing to depletion of fissile nuclides. The first 7 characters are for nuclide name with one blank at the last and the eighth character for reaction type (Fission / Production / Capture / Absorption / N2N / Decay). Enter as 'U02350_A', 'U02380_C', 'PU2410_D'. If 'D' is specified, contribution of decay is included in the calculation of conversion ratio whereas this contribution is not taken into account for the tabulated absorption cross-sections of fissile material ($\Sigma_{a,g}^{fissile}$) stored in the MACRO file.
Note:

Nuclide name and reaction type should be appropriately selected by taking into account the burn-up chain model used in the calculation (cf. Sect.3.1).

The fissile absorption cross-section ($\Sigma_{a,g}^{fissile}$) is used by core burn-up calculation

modules of MOSRA to calculate conversion ratio together with fertile capture cross-section ($\Sigma_{c,g}^{fertile}$).

2 IFISFLG

= 0 Do not multiply atomic number density of this nuclide (microscopic cross-sections are used)

= 1 Multiply atomic number density of this nuclide (macroscopic cross-sections are used)

3 FISFACT The factor to indicate sign and branching ratio

If Block-7-2 are entered as below, the summation of the values in parentheses is defined as the depletion rate of fissile material (denominator of conversion ratio calculation: Eq.(2.5-3))

U02330_A	1	+1.0	$(+N^{\text{U-233}}\sigma_a^{\text{U-233}}\Phi)$
U02350_A	0	+0.5	$(+0.5\times\sigma_a^{\text{U-235}}\Phi)$
PA2330_A	1	-1.0	$(-N^{\text{Pa-233}}\sigma_a^{\text{Pa-233}}\Phi)$
PU2410_D	1	+0.2	$(+0.2\times\lambda^{\text{Pu-241}}N^{\text{Pu-241}})$

It should be noted that the tabulated values of fissile absorption cross-sections along burn-up are defined as:

$\Sigma_{a,g}^{fissile} = N^{\text{U-233}}\sigma_{a,g}^{\text{U-233}} + 0.5\times\sigma_{a,g}^{\text{U-235}} - N^{\text{Pa-233}}\sigma_{a,g}^{\text{Pa-233}}$,

where no contribution of decay is included.

Repeat Block-7-2 NFIS times.

Block-7-3 Required if IBC5=1 /A8, 1, 1/

1 NAMFRT Nuclide name and reaction type (with 8 characters) contributing to production of fissile nuclides. The first 7 characters are for nuclide name with one blank at the last and the eighth character for reaction type (Fission / Production / Capture / Absorption / N2N / Decay). Enter as ‘U02380_C’, ‘PU2400_C’, ‘PU2400_N’.

Note:

Nuclide name and reaction type should be appropriately selected by taking into account the burn-up chain model used in the calculation. (cf. Sect.3.1)

If ‘D’ is specified, contribution of decay is included in the calculation of conversion ratio whereas this contribution is not taken into account for the

tabulated capture cross-sections of fertile material ($\Sigma_{c,g}^{fertile}$) stored in the MACRO file.

The fertile capture cross-section ($\Sigma_{c,g}^{fertile}$) is used by core burn-up calculation modules of MOSRA to calculate conversion ratio together with fissile absorption cross-section ($\Sigma_{a,g}^{fissile}$).

2 IFRTFLG

= 0 Do not multiply atomic number density of this nuclide (microscopic cross-sections are used)

= 1 Multiply atomic number density of this nuclide (macroscopic cross-sections are used)

3 FRTFACT The factor to indicate sign and branching ratio. (same as FISFACT in Block 7-2)

Repeat Block-7-3 NFER times.

Block-8-1 Required if IBC6=1 /1/

NMICR Number of nuclides of which effective cross-sections are written on MICRO file

Block-8-2 Required if IBC6=1 /NMICR/, (9A8)

NAMMIC Names of nuclides of which effective cross-sections are written on MICRO file. Enter a nuclide name with 6 characters and two arbitrary characters at the last as a delimiter. The microscopic cross-sections of the specified nuclide are written for every material in which the nuclide is contained.
Sample input: U02380__B00100__AM2421__ (The 7-th and 8-th characters are ineffective and usually two blanks are entered.)

Block-9 Required if IBC10=1 /NMAT/

(IBTYPE(*i*), *i*=1,NMAT)

= 0 Non-depleting material

= 1 Depleting material

Enter IBTYPE(*i*) to material *i* to specify if it is depleting or not. NMAT is the number of materials, and material is numbered in the order appearing in the material specifications (cf. Sect.2.4). Specify IBTYPE(*i*)=0 if the material includes none of the nuclides in the selected burn-up chain model.

Block-10 Required if IBC3=±3 and no fissionable nuclide is contained in any material /1/

FLXLVL Constant flux level (n/cm^2/s)

When the system does not contain any fissionable nuclide but depleting nuclides like burnable poisons, depletion calculation is available under the fixed flux condition by specifying IBC3=±3 together with IBC2=±3 or ±4. Enter an average flux level (n/cm^2/s) to the burnable region. Unit of burn-up (MWd/t, MWd) is replaced with the integrated absorption reaction rate.

Block-11-1 Required if IBC9>0 /A8/

NAMFP Name of nuclides of which atomic number densities are specified by input values.

Sample: XE1350__ (the first 6 characters are effective)

Block-11-2-1 Required if IBC9>0 /1/

MPOS = 0 End of Block-11-2 entries

> 0 Material number for which the atomic number density of the nuclide is replaced with the input value. Material is numbered in the order appearing in the material specification (Sect.2.4).

Block-11-2-2 Required if IBC9>0 and MPOS>0 /IBC1+1/

DENFP(*i*) Atomic number density (×10^{24} n/cm^3) at each burn-up step (0,1, 2,…, IBC1)

Repeat Block-11-2 until MPOS=0 is encountered.

Repeat Block-11-1 through Block-11-2 IBC9 times.

<<Sample input of Block-11 for IBC9=2>>

```
---------+---------+---------+---------+---------+--------
XE1350             / NAMFP (Block-11-1)
1           / MPOS(Block-11-2-1) for 1st Fuel
11(0.0)            / DENFP (Block-11-2-2)
5           / MPOS for 2nd Fuel
11(0.0)            / DENFP
0           / MPOS=0 : End for Xe-135
SM1490             / NAMFP (Block-11-1)
1           / MPOS(Block-11-2-1) for 1st Fuel
11(1.23E-4 )       / DENFP (Block-11-2-2)
5           / MPOS for 2nd Fuel
11(1.23E-4) / DENFP : Peak Sm-149
0           / MPOS=0 : End for Sm-149
---------+---------+---------+---------+---------+--------
```

Block-12 Required if IBC12≠0 /IBC1/

(JPCFLG(*i*), *i*=1,IBC1)

= 0 Not apply the PC-Method to the *i*-th burn-up step interval

= 1 Apply the PC-method to the *i*-th burn-up step interval

Sample: 20(1) 30(0) / PC-method for first 20 steps among 50 steps

Note:

The PC-method is not applied to the blanch-off calculation (IBC3=±2). When it is applied, the calculation is terminated with an error message.

For the burn-up step with the cooling calculation, the PC-method is neglected by the code and the burn-up calculation is continued.

Block-13 Required if IBC13>0 /NBR/

(IBSTP (*i*), *i*=1,NBR)

Burnup step number (integer from 0 to 99) of the reference case to which the branch-off calculation is applied

NBR is the total number of branch-off steps specified by IBC13

Block-14 Required if IBC14>0 /IBC1/

(NSTP (*i*), *i*=1,IBC1)

Number of sub-steps in each burnup step (*i*)

The following notices and recommendations would be helpful for the use of burn-up calculation.

- Burn-up step interval

At every burn-up step, effective microscopic cross-sections and flux distribution are updated and output edit is prepared. As change of composition is calculated by an analytical method at every sub-step (20 sub-steps in each burn-up step interval), the interval has not necessarily to be short. Normally, 5,000 ~ 10,000 MWd/t is used. However, the initial Xenon buildup and the treatment of burnable poison of relatively short lifetime should be taken care of.

- Multiplication factor and cross-sections at the final step

At the end of final burn-up step, the composition of depleting material is edited, but the spectrum calculation for this composition is not executed. Therefore, values are not printed out for the multiplication factor and the conversion ratio corresponding to the final composition. Neither cross-sections nor fluxes are written into PDS files.

The user should keep in mind that the step before the last gives the upper limit of burn-up in the

tabulation of cross-sections for the succeeding use of core burn-up calculation.

- Plural X-region specification

When plural X-Regions are specified, one depleting material should not be contained in two different X-regions.

- Restart burn-up calculation

The restart burn-up calculation is available as far as members *case*HT*bb* (cf. Sect.6.2) of the same case name are preserved on MACRO file. The change of input data is limited, in principle, only on restart specification and addition of burn-up steps.

Even in the case of abnormal termination due to lack of CPU time or shortage of disk extent, the calculation can be continued by using the restart option, as long as the members *case*HT*jj* are preserved on the MACRO file. However, if the virtual PDS file is used, the members are not stored in the case of some abnormal terminations. To avoid such a trouble in a large-scale burn-up calculation, it is recommended to set MOACS (access mode of MACRO file) to 'File' (direct file access) in Block-6 of Sect.2.2.

- Branch-off calculation

The branch-off calculation along the burn-up steps is available if the member *case*BNUP of the reference case is preserved on MACRO file. The input for geometry, number of burn-up steps, burn-up intervals, etc. must be consistent with those of the reference case. The number of materials, the order of materials (=material number) and the burn-up chain model must be also kept in principle, whereas material temperatures, composition of non-depleting materials can be replaced. For example, to obtain the average cross-sections of the assembly with control rod insertion is obtained by the branch-off calculation, the region in which the control rod is inserted should be reserved as an R-region in the reference calculation without control rod insertion.

- Burn-up calculation by reading the initial composition

To execute the burn-up calculation by reading the initial composition, the member *case*BNUP (cf. Sect.6.2) of the reference case must be preserved on the MACRO file. The geometry, the number of materials, the order of materials (=material number), the burn-up chain model, etc. must be consistent with those in the reference case.

2.6 Additional Input

Depending on the use of the optional functions specified in Sect.2.2, additional input blocks are required here.

(1) Input data for Stoker and Weiss's method (Required if IOPT4=2)

The following Block-1 through Block-3 are required when the Stoker and Weiss's method is used by IOPT4=2 in Sect.2.2.

Block-1 Information on lattice geometry /2/

1 IGTSK geometry type

= 1 Slab (for a fuel plate)

= 2 Cylinder (for a fuel pellet)

2 NDIVSK Number of segmented regions in a fuel plate or fuel pellet

Block-2 Material assignment /NDIVSK/

(MATSK(*i*), *i*=1,NDIVSK)

Material number by segmented region from left to right (IGTSK=1) or from center to outer (IGTSK=2). Material is numbered in the order appearing in the material specifications (cf. Sect.2.4).

Block-3 Boundary position of segments /NDIVSK/

(RMESH(*i*), *i*=1,NDIVSK)

Position (cm) from left boundary (IGTSK=1) or center (IGTSK=2)

RMSSH(NDIVSK) should be the position of right or outer boundary.

<<Sample input for a pellet divided into 10 segments >>

```
---------+---------+---------+---------+---------+--------
**** Material specification (Sect.2.4)***
12   / NMAT
UO20100X  0 3  849.12 0.82 0.0    / 1 :UO2#1 Center of pellet
U02350      0 0  1.05692E-03  / 1
U02380      0 0  2.21466E-02  / 2
O00160      0 0  4.64071E-02  / 3
UO20200X  0 3  849.12 0.82 0.0    / 2 :UO2#2
U02350      0 0  1.05692E-03  / 1
U02380      0 0  2.21466E-02  / 2
O00160      0 0  4.64071E-02  / 3
UO20300X  0 3  849.12 0.82 0.0    / 3 :UO2#3
U02350      0 0  1.05692E-03  / 1
U02380      0 0  2.21466E-02  / 2
O00160      0 0  4.64071E-02  / 3
  :
  : (abbr. for UO2#4-UO2#9)
  :
UO21000X  0 3  849.12 0.82 0.0    / 10:UO2#10
U02350      0 0  1.05692E-03  / 1
U02380      0 0  2.21466E-02  / 2
O00160      0 0  4.64071E-02  / 3
ZRCLD00X  0 3  586.88 0.13 0.0    / 11 : Cladding
```

```
ZRN000      0 0  3.80326E-02  / 1
CRN000      0 0  6.71520E-05  / 2
FEN000      0 0  1.31290E-04  / 3
WATER00X  0 2  578.15 1.0  0.0    / 12 : Water moderator
H0001H      0 0  4.78704E-02  / 1
O00160      0 0  2.39352E-02  / 2
*
**** Input for Stoker & Weiss's method (Sect.2.4)****
*
2 10        / IGTSK NMAT Block-1 of Sect.2.6
1 2 3 4 5 6 7 8 9 10 / Block-2: MAT number from UO2#1 to UO2#10
1.29653E-01 1.83358E-01 2.24566E-01 &  Block-3 (
2.59307E-01 2.89914E-01 3.17585E-01 &  radius from center
3.43031E-01 3.66715E-01 3.88960E-01 &  with equal volume
4.10000E-01                         /  End of Block-3
---------+---------+---------+---------+---------+--------
```

(2) Input data for fixed source distribution (Required if IOPT1=0 and IOPT23=1)

The following Block-1 is required when the user gives a fixed source distribution in the case when there is no fissionable material in the lattice. If there are one or more fissionable materials, the input here is ineffective. The energy spectrum of the fixed source is assumed to be a typical fission spectrum of U-235 implemented in the Public FAST Library.

Block-1 Source distribution /NRR/

(SOURCE(i), i=1,NRR)

Relative source density per unit volume by R-region (n/s/cm^3). NRR is the number of R-regions specified in Block-1 of the PIJ input section (Sect.2.3).

(3) Input data for PEACO (Required if IOPT1=0 and IOPT5=1 or 2)

The following input Blocks are required when the PEACO routine is used for effective microscopic resonance cross-sections by specifying IOPT5=1 or 2 in Sect.2.2.

Block-1 /1/

IPLOT Plot control for spectra by PEACO in resonance energy region

Plot data is stored as a PostScript file (cf. Sect.1.9, Chapter 4).

= 0 Skip plot

= ±1 Plot neutron spectra by up-to five R-regions in one figure

= ±2 Plot neutron spectra by R-region per figure

= ±3 Plot neutron spectra by up-to five R-regions in one figure in the energy region from EL(i) to EH(i) which are specified in Block-2 and -3, respectively. (i=1,2,3,4: four energy regions for plotting)

Note:

If the negative value is entered, group cross-sections modified by PEACO

calculation are printed out in the fine-group structure.

Block-2 Required if IPLOT=±3 /4/

(EL(i),i=1,4) The lower energy boundaries of 4 energy regions for plotting the spectra. In the first figure, spectra are plotted in the first energy region from EL(1) to EH(1).

Block-3 Required if IPLOT=3 /4/

(EH(i),i=1,4) The upper energy boundaries of 4 energy regions for plotting the spectra.

3. Burn-up Chain Model

3.1 Equipped Chain Models

There are several burn-up chain library files as shown in Table 3.1 in text format. Each file keeps a burn-up chain scheme (model) where the decay chains and constants are described. The user can choose any of them considering the reactor type and his purpose, or he can edit his own chain scheme if necessary.

Table 3-1 Available burn-up chain models

Type of burn-up chain models (main purpose)	Name of chain model (=file name of burn-up chain data)	
	for thermal reactors	for fast reactors
Standard chain model (recommended model to reduce memory and I/O)	u4cm6fp119bp14T_J40	u4cm6fp119bp14F_J40
Detailed chain model (for Th fueled lattice and for detailed fission products)	th2cm6fp198bp8T_J40	th2cm6fp198bp8F_J40

Values of fission yield and isomeric ratio are different between models for thermal and fast reactors.

file name [MSRAC/chain/*] Contents

****

u4cm6fp119bp14T/F_J40

This model includes 21 heavy nuclides starting from U-234 to Cm-246 (Fig.3.1-1) and 118 explicit FPs and one pseudo FP (Fig.3.1-3). Burnable poisons or neutron absorbers such as boron, gadolinium, hafnium and erbium can be treated. This is a standard model for the lattice with UO_2 and/or MOX fuel.

u4cm6fp193bp14T/F_J40

This model includes 28 heavy nuclides from Th-232 to Cm-246 (Fig.3.1-2) and 198 explicit FPs and 8 burnable poisons (Fig.3.1-4). When U-233 and Thorium are involved or more detailed FPs are required, this model is suitable.

****

The constants of the burnup chain models are taken from the following nuclear data;

- Decay branching: JENDL FP Decay Data File 2000[18] for fission products and ENDF/B-VII.0[19] decay data sublibrary (file 8) for others.
- Half-life: ENSDF(2008)[20] except for Se-79 and Sn-126. For Se-79 and Sn-126 with long

half-lives, recent values are taken from literatures[21,22], and they are validated by burnup analyses of assay data of LWR spent fuels[23] .

- Energy release per fission: values from JNDC-V2 (Table 2.5.7 of JAERI 1320)[24] for major nuclides and values from fitting formula of ORIGEN2[25] for other minor actinides.
- Fission yield: JENDL-4.0 (thermal fission yield of Pu-241 is taken from the updated file: JENDL-4.0u[26].
- Isomeric ratio: JENDL-4.0 for ^{241}Am(n,γ) and ^{237}Np(n,2n) and JEFF-3.0/A[27] for others, i.e. Kr-84, Sr-86, Nb-93, Nb-94, Rh-103, Rh-105, Ag-109, Cd-112, Cd-114, Sn-118, Sn-120, Sn-122, Sb-123, Sb-125, Te-122, Te-124, Te-126, I-129, Ba-136, Pm-147, Eu-151, Ho-237.

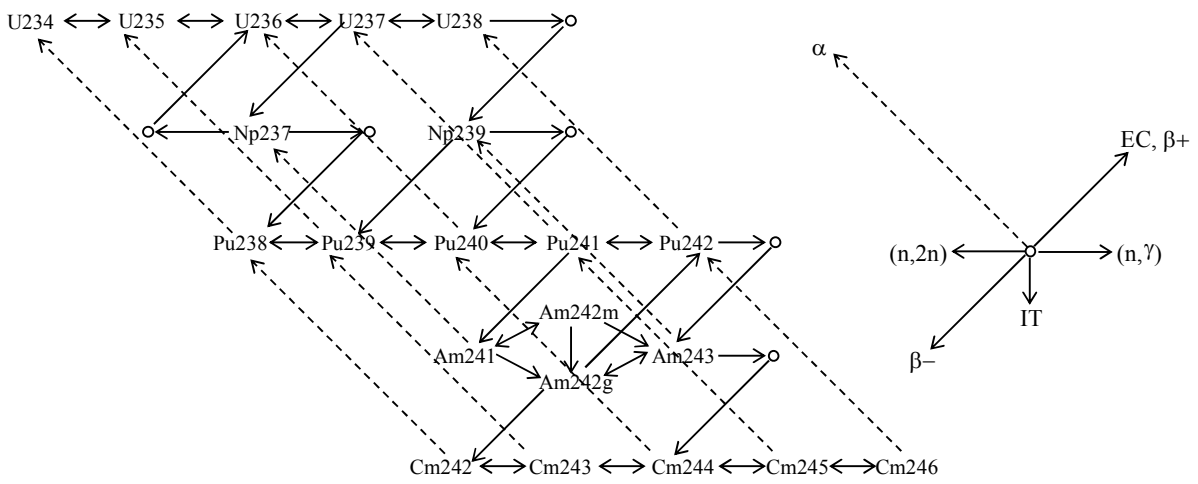


Fig. 3.1-1 Burn-up chain model for heavy nuclides (models : u4cm6~)

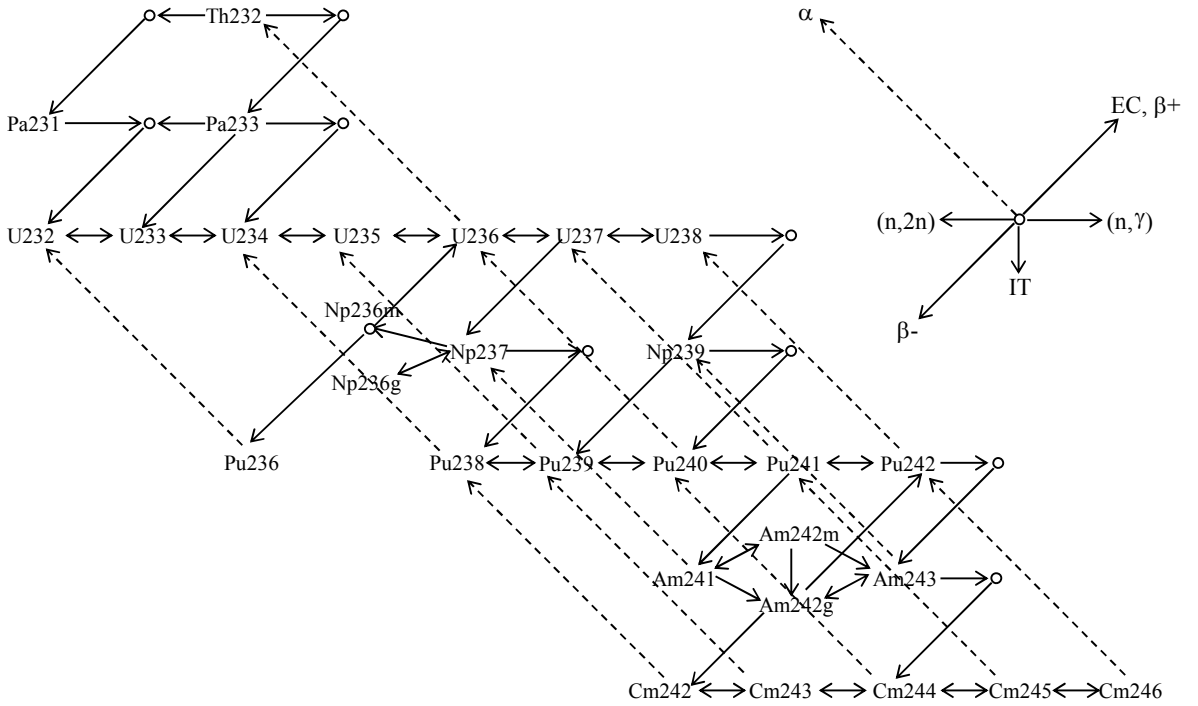


Fig. 3.1-2 Burn-up chain model for heavy nuclides (models : th2cm6~)

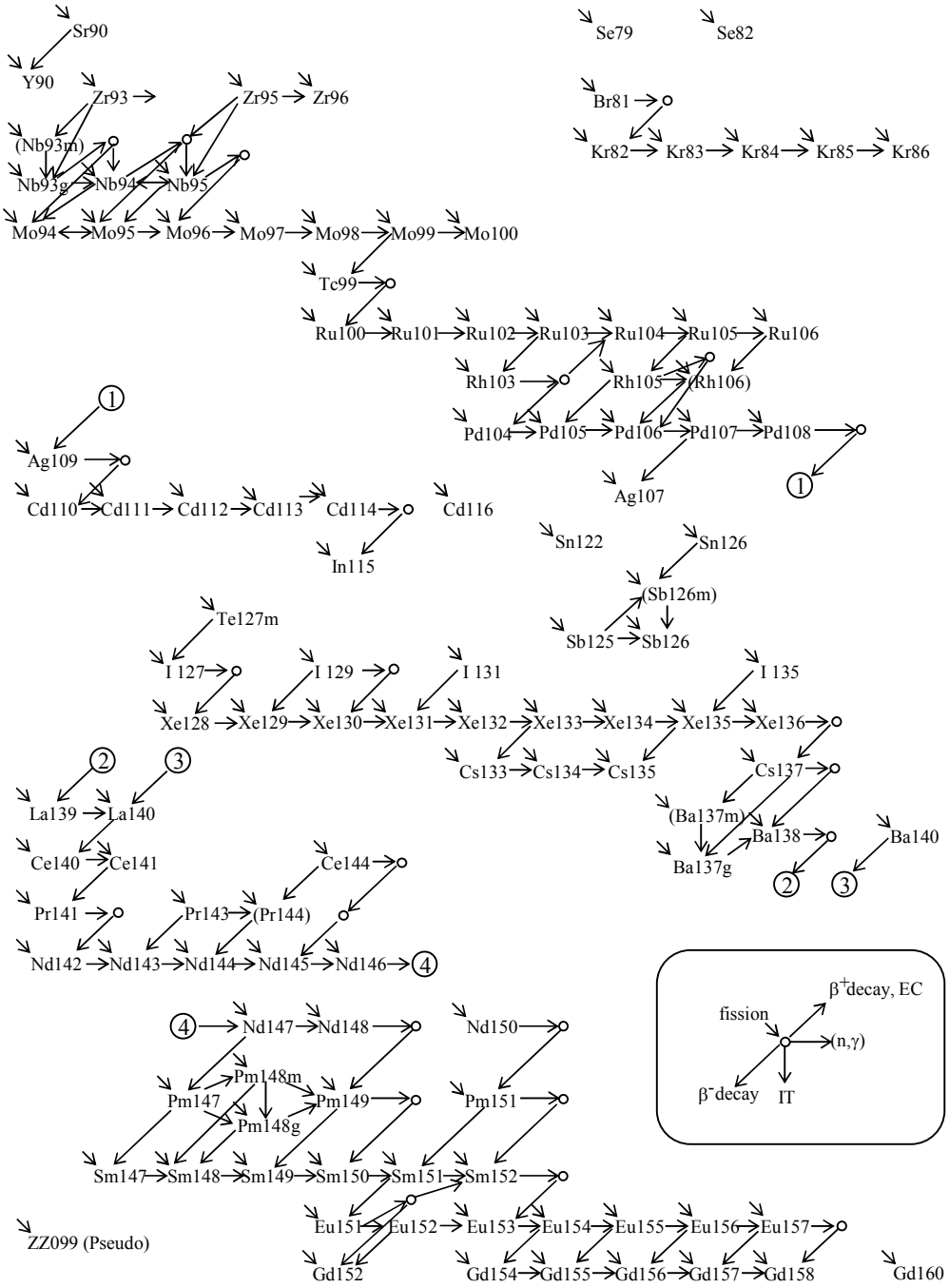


Fig. 3.1-3 Burn-up chain model for fission products and burnable poisons (models : ~ fp119bp14T/F)

Note: Nuclides in parentheses have no cross-section data, only time-decay is considered.

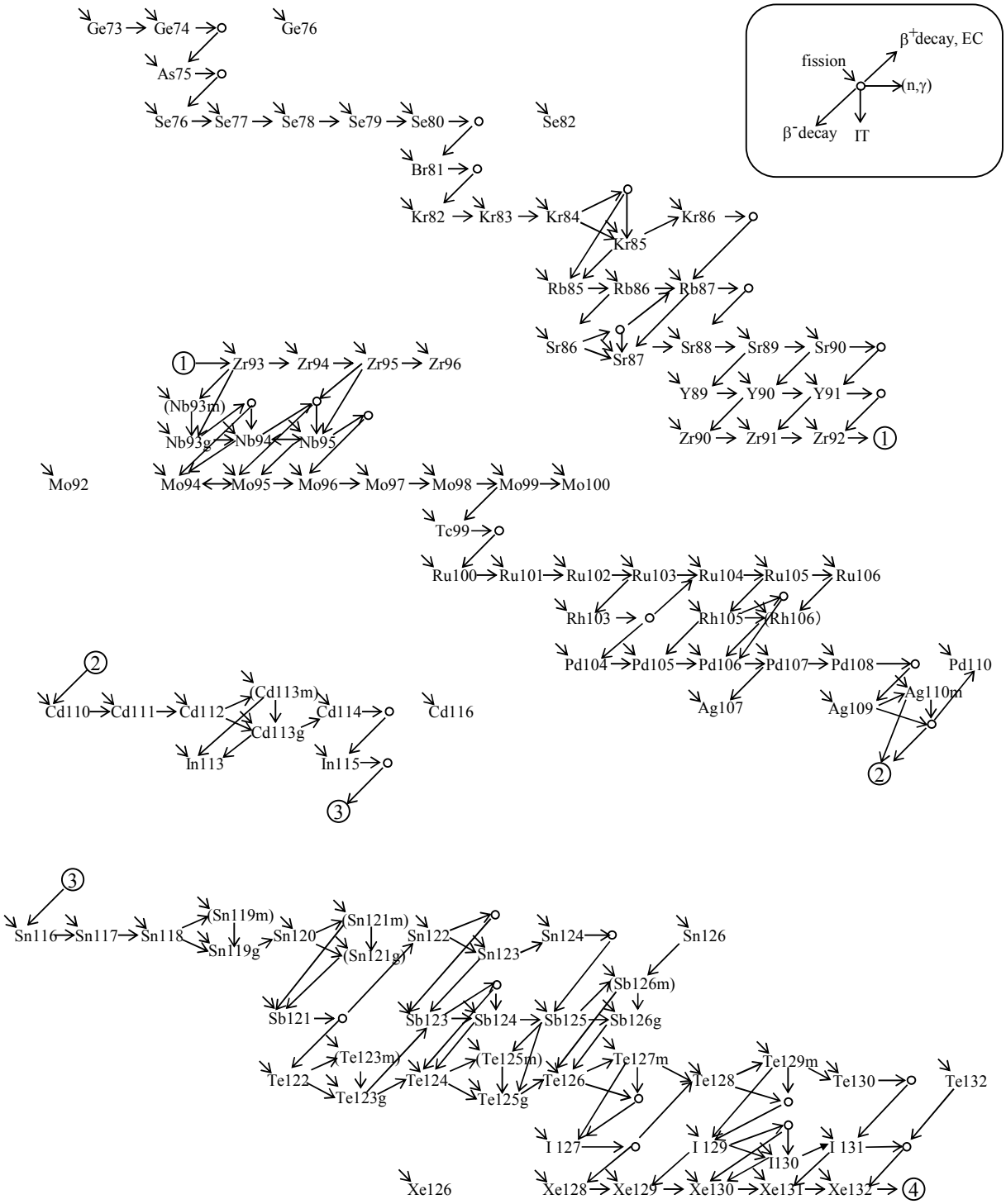
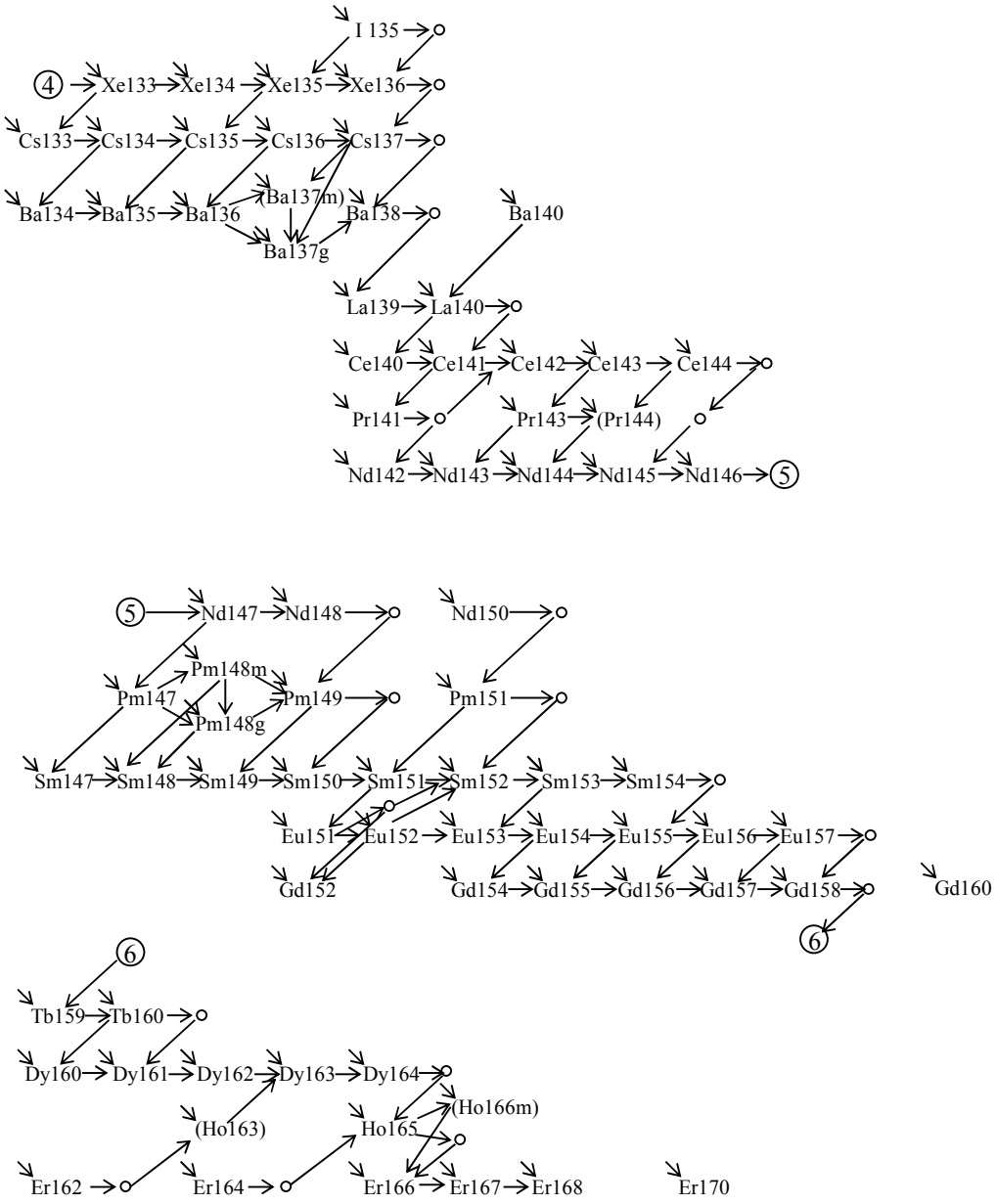


Fig. 3.1-4 (1/2) Burn-up chain model for fission products and burnable poisons
(models: ~fp198bp8T/F)



B 10(BP)

Hf176 → Hf177 → Hf178 → Hf179 → Hf180 → Hf181 → Hf182

Fig. 3.1-4 (2/2) Burn-up chain model for fission products and burnable poisons (models: ~fp198bp8T/F)

3.2 Structure of Chain Model

A user can generate new chain models or modify equipped chain models. The structure of burn-up chain model is as follows. Any comment line can be inserted with '*' at the first column.

Block-1-1	Number of nuclides	/5/
1 LNMAX	Number of total nuclides registered. Heavy nuclides, FP nuclides and burnable poison nuclides should be included. Non-depleting nuclides may be included, though they are ineffective in the burn-up calculation.	
2 NMAX	Number of nuclides related to depletion and/or decay (usually LNMAX=NMAX)	
3 NFIS	Number of fissionable nuclides (heavy nuclides)	
4 NPAR	Maximum number of parent nuclides from which a depleting nuclide is produced (cf. NCH in Block-3-1)	
5 NYLDTY	Number of types of FP yield data (cf. Block-4)	

Block-1-2	General constants	/4/
1 ANMU	Conversion factor to get atomic mass in *amu* for AMASS(*i*) in Block-2 If AMASS(*i*) is given in *amu*, enter 1.0.	
2 AVOGA	Avogadro number (e.g. 6.0221358E+23 n/mol)	
3 EVTOJ	Conversion factor to get energy in *Joule* unit for energy in eV unit (e.g. 1.160219E-19 J/eV)	
4 GASCON	Gas constant (e.g. 8.3148 J/mol/K)	

DO *i*=1,LNMAX	<Repeat Block-2 LNMAX times>	
Block-2	Nuclide list (Free format for the last 7 data in columns 19-80)	/A8,2X,A8,7/
1 IHOL(*i*)	Nuclide ID by 8 characters (e.g. `U-235___`)	
2 SRACID(*i*)	Nuclide ID by 8 characters in the SRAC2006 format[3]: X*zzmc*`---` (e.g. `XU050001`), where the last 3 characters are ineffective. SRACID(*i*) is not used in MOSRA-SRAC.	
3 NCODE(*i*)	Code number of nuclide (enter after column 19) Number is arbitrary, but not duplicate with that of another nuclide. e.g. the definition 'atomic number + mass number + (0: ground / 1: exited)' like 922350 gives reasonable numbering. For pseudo fission product, give the integer from 1 to 9999.	

4 AMASS(*i*) Mass of nuclide

Unit is arbitrary (in the equipped model, neutron mass unit is used). Multiplied by ANMU in Block-1-2, it makes mass in *amu*.

5 IFISS(*i*) Attribute of nuclide

> 1 heavy nuclide (fissionable) related to instantaneous conversion ratio

= 1 heavy nuclide (fissionable) not related to instantaneous conversion ratio

= 0 FP (burnable poison nuclide is included, if it has fission yield)

= -1 nuclide without cross-section data (decay is considered)

≤ -10 non-FP nuclide used for burnable poison (without fission yield)

Heavy nuclides assigned as IFISS(*i*)>1 by this item are used for the default definition of instantaneous conversion ratio. Unless the user redefines the conversion ratio by the option IBC5=1 in Sect.2.5, the ratio is defined as follows;

The instantaneous conversion ratio

=(production rate of fissile nuclides)/(depletion rate of fissile nuclides)

≈ (capture rate of fertile nuclides)/(absorption rate of fissile nuclides),

where IFISS(*i*)=2 should be given to fertile nuclides for the numerator, and IFISS(*i*)=3 to fissile nuclides for the denominator.

6 IRES(*i*) Indicator of nuclide type

= 0 non-resonant nuclide

= 2 resonant nuclide which has the MCROSS library

= -1 nuclide which has no cross-section data (e.g. short-lived FP whose cross-section is not evaluated in nuclear data)

7 EFISS(*i*) Emission energy per fission (MeV/fission)

8 ECAP(*i*) Emission energy by (n, γ) reaction (MeV/emission)

Give zero value (ineffective) for ECAP, because only EFISS is treated as heat generation in the current MOSRA-SRAC.

9 FACT2N(*i*) Reaction rate modifier for sensitivity analysis

Either (n,2n) or (n,γ) reaction rate is multiplied by FACT2N(*i*). Set 1.0 usually.

> 0 (n,2n) reaction rate is multiplied by FACT2N(*i*)

≤ 0 (n,γ) reaction rate is multiplied by absolute value of FACT2N(*i*)

Note: When zero value of cross-section is found by the code, the value is replaced with 1.0×10^{-20}.

END DO <end of repetition for Block-2>

DO *j*=1,NMAX <Repeat Block-3 NMAX times>

Block-3-1 Decay data /A8,I2,E10.3,A2/

1 NUCL(*j*) Nuclide ID by 8 characters as specified by IHOL in Block-2 of nuclide related to depletion and/or decay

2 NCH(*j*) Number of parent nuclides of NUCL(*j*)

NCH(*j*) should be less than or equal to NPAR in Block 1-1.

3 HALFT(*j*) Half-life of NUCL(*j*) in unit specified by the next item)

4 TUNIT(*j*) Unit of HALFT(*j*) by 8 (or less) characters (first character is effective)

= 'SECONDS' second

= 'MINUTES' minutes

= 'HOURS' hours

= 'DAYS' days

= 'YEARS' years

DO *k*=1,NCH(*j*) <Repeat Block-3-2 NCH(*j*) times>

Block-3-2 Decay path from parent nuclides /A8,2X,A8,2X,E12.5/

1 NUCLP(*k*) Nuclide ID by 8 characters as specified by IHOL in Block-2 of a parent nuclide of NUCL(*j*)

2 NBIC(*k*) Decay or reaction type from NUCLP(*k*) to NUCL(*j*) by 8 (or less) characters.

First five characters are effective.

= 'BETA-' β^- decay

= 'IT' isomeric transition

= 'BETA+' β^+ decay

= 'EC' orbital electron capture

= 'ALPHA' α decay

= 'DELAYED' delayed neutron emission

= 'CAPTURE' or '(N,G)' (n,γ) reaction

= '2N' or '(N,2N) (n,2n) reaction

= '3N' or '(N,3N) (n,3n) reaction

= '(N,P)' (n,p) reaction

= '(N,A)' (n,α) reaction

3 PBIC(*k*) Branching ratio from NUCLP(*k*) to NUCL(*j*) of this decay

END DO <end of repetition for Block-3-2>

END DO <end of repetition for Block-3>

DO *m*=1,NYLDTY <Repeat Block-4 NYLDTY times>
Block-4-1 /A8,2/

1 NAMYLD(*m*) ID name of *m*-th type FP yield data by 8 (or less) characters
(e.g. 'YU235T' for U-235 thermal fission)

2 NYNUCL(*m*) Number of heavy nuclides to commonly use *m*-th type FP yield data

3 NFP(*m*) Number of FP nuclides included in *m*-th type FP yield data

Block-4-2 /5(A8,2X)/

(NAMFIS(*i*), *i*=1,NYNUCL(*m*))
Nuclide ID names appearing in IHOL in Block-2 of heavy nuclides (IFISS>0) which commonly use *m*-th type FP yield data in 5*(A8,2X) format

DO *n*=1,NFP(*m*) <Repeat Block-4-3 NFP(*m*) times>
Block-4-3 /A8,1/

NAMFP(*n*) Nuclide ID names appearing in IHOL in Block-2 of FP (IFISS=0) included in *m*-th type FP yield data

YLDFP(*n*) Fission yield of the nuclide with ID name NAMFP(*n*)

END DO <end of repetition for Block-4-3>
END DO <end of repetition for Block-4>

The chain model 'u4cm6fp119bp14T_J40' is described as follows;

```
*LNMAX NMAX NFIS NPAR NYLDTY  (FREE FORMAT) for Block-1-1
  154  154   21    4   19
*  ANMU       AVOGA          EVTOJ    GASCON (FREE FORMAT) for Block-1-2
1.008665  6.0221358E+23  1.60219E-19  8.31480
* AMASS(neutron mass unit) x ANMU = Mass in Atomic Mass Unit
*IHOL      SRACID  NCODE   AMASS    IFISS  IRES   EFIS    ECAP  FACT2N
*21 Heavy Nuclide ------------------------------------------------
U-234     XU040001 922340  232.030     1     2    200.32  0.00   1.0
U-235     XU050001 922350  233.025     3     2    202.25  0.00   1.0
U-236     XU060001 922360  234.018     1     2    202.39  0.00   1.0
U-237     XU070001 922370  235.013     1     0    202.39  0.00   1.0
U-238     XU080001 922380  236.006     2     2    205.92  0.00   1.0
NP237     XNP70001 932370  235.012     1     2    206.05  0.00   1.0
NP239     XNP90001 932390  236.999     1     2    206.05  0.00   1.0
PU238     XPU80001 942380  236.005     1     2    206.05  0.00   1.0
PU239     XPU90001 942390  236.999     3     2    210.96  0.00   1.0
PU240     XPU00001 942400  237.992     2     2    211.01  0.00   1.0
PU241     XPU10001 942410  238.986     3     2    213.27  0.00   1.0
PU242     XPU20001 942420  239.979     1     2    214.29  0.00   1.0
AM241     XAM10001 952410  238.986     1     2    215.25  0.00   1.0
AM242     XAMG0001 952420  239.981     1     0    215.62  0.00   1.0
AM242M    XAMM0001 952421  239.981     1     2    215.62  0.00   1.0
AM243     XAM30001 952430  240.973     1     2    216.00  0.00   1.0
CM242     XCM20001 962420  239.980     1     2    219.48  0.00   1.0
CM243     XCM30001 962430  240.972     1     2    219.86  0.00   1.0
CM244     XCM40001 962440  241.966     1     2    220.25  0.00   1.0
```

```
CM245     XCM50001 962450  242.961     1     2    220.63  0.00   1.0
CM246     XCM60001 962460  243.953     1     0    221.02  0.00   1.0
*118 FP + 1 Pseudo Nuclide
SE079     XSE90001 340790   78.321     0     0      0.00  0.00   1.0
SE082     XSE20001 340820   81.213     0     0      0.00  0.00   1.0
BR081     XBR10001 350810   80.304     0     0      0.00  0.00   1.0
KR082     XKR20001 360820   81.210     0     0      0.00  0.00   1.0
KR083     XKR30001 360830   82.202     0     0      0.00  0.00   1.0
KR084     XKR40001 360840   83.191     0     0      0.00  0.00   1.0
KR085     XKR50001 360850   84.183     0     0      0.00  0.00   1.0
KR086     XKR60001 360860   85.173     0     0      0.00  0.00   1.0
SR090     XSR00001 380900   89.227     0     0      0.00  0.00   1.0
Y-090     XY000001 390900   89.227     0     0      0.00  0.00   1.0
ZR093     XZR30001 400930   92.201     0     0      0.00  0.00   1.0
ZR094     XZR40001 400940   93.192     0     0      0.00  0.00   1.0
ZR095     XZR50001 400950   94.184     0     0      0.00  0.00   1.0
ZR096     XZR60001 400960   95.175     0     0      0.00  0.00   1.0
NB093M    XNBM0001 410931   92.201     0    -1      0.00  0.00   1.0
NB093     XNB30001 410930   92.201     0     0      0.00  0.00   1.0
             :
             :
GD154     XGD40001 641540  152.677     0     0      0.00  0.00   1.0
GD155     XGD50001 641550  153.668     0     2      0.00  0.00   1.0
GD156     XGD60001 641560  154.660     0     0      0.00  0.00   1.0
GD157     XGD70001 641570  155.651     0     2      0.00  0.00   1.0
GD158     XGD80001 641580  156.643     0     0      0.00  0.00   1.0
GD160     XGD00001 641600  158.626     0     0      0.00  0.00   1.0
ZZ111     XZ990001   1110  101.718     0     0      0.00  0.00   1.0
*14 Burnable Poisons except FPs ------------------------------------
B-010     XB000001  50100    9.9269  -10     0      0.00  0.00   1.0
ER162     XER20001 681620  160.608   -10     0      0.00  0.00   1.0
ER164     XER40001 681640  162.591   -10     0      0.00  0.00   1.0
ER166     XER60001 681660  164.574   -10     2      0.00  0.00   1.0
ER167     XER70001 681670  165.565   -10     2      0.00  0.00   1.0
ER168     XER80001 681680  166.557   -10     2      0.00  0.00   1.0
ER170     XER00001 681700  168.540   -10     2      0.00  0.00   1.0
HF176     XHF60001 721760  174.488   -10     0      0.00  0.00   1.0
HF177     XHF70001 721770  175.479   -10     2      0.00  0.00   1.0
HF178     XHF80001 721780  176.471   -10     2      0.00  0.00   1.0
HF179     XHF90001 721790  177.462   -10     2      0.00  0.00   1.0
HF180     XHF00001 721800  178.454   -10     2      0.00  0.00   1.0
HF181     XHF10001 721810  179.395   -10     0      0.00  0.00   1.0
HF182     XHF20001 721820  180.388   -10     0      0.00  0.00   1.0
*-------------------------------------------------------------------
*NUCL   NCH   HALFT TUNIT (A8,I2,E10.3,A8) for Block-3-1
U-234    2  2.455E+5YEARS
*NUCLP  NBIC         PBIC (A8,2X,A8,2X,E12.5) for Block-3-2
U-235     2N          1.000
PU238     ALPHA       1.000
* Repeat from Block-3-1 to Block-3-2 NMAX Times
U-235    3  7.038E+8YEARS
U-234     CAPTURE     1.000
U-236     2N          1.000
PU239     ALPHA       1.000
*
U-236    4  2.342E+7YEARS
U-235     CAPTURE     1.000
NP237     2N          0.31161936
U-237     2N          1.000
PU240     ALPHA       1.000
*
U-237    3  6.75    DAYS
U-236     CAPTURE     1.000
U-238     2N          1.000
PU241     ALPHA       0.0000245
*
U-238    2  4.468E+9YEARS
U-237     CAPTURE     1.000
```

```
PU242     ALPHA       0.9999950
             :
             :
CM246     1   4760.0 YEARS
CM245      CAPTURE     1.0000
*  Block-3 for FP
SE079     0 3.770OE+5YEARS
SE082     0 8.300E+19YEARS
*
BR081     0   0.0    SECONDS
*
KR082     1   0.0    SECONDS
BR081      CAPTURE    1.0000
*
KR083     1   0.0    SECONDS
KR082      CAPTURE     1.0000
*
KR084     1   0.0    SECONDS
KR083      CAPTURE     1.0000
*
KR085     1  10.756  YEARS
KR084      CAPTURE     0.356556
*
KR086     1   0.0    SECONDS
KR085      CAPTURE     1.0000
*
SR090     0  28.79   YEARS
*
Y-090     1  64.00   HOURS
SR090      BETA-       1.0000
             :
             :
GD157     2   0.0    SECONDS
EU157      BETA-       1.0000
GD156      CAPTURE     1.0000
*
GD158     2   0.0    SECONDS
EU157      CAPTURE     1.0000
GD157      CAPTURE     1.0000
*
GD160     0   0.0    SECONDS
ZZ111     0   0.0    SECONDS
*  Block-3 for BP (nuclide with no fission yield)
B-010     0   0.0    SECONDS
ER162     0   0.0    SECONDS
ER164     0   0.0    SECONDS
ER166     0   0.0    SECONDS
*
ER167     1   0.0    SECONDS
ER166      CAPTURE     1.0000
*
ER168     1   0.0    SECONDS
ER167      CAPTURE     1.0000
             :
             :
HF182     1   8.90E+6YEARS
HF181      CAPTURE    1.0000
*
*****************************************************
*Repeat from Block-4-1 to Block-4-3 NYLDTY Times
*FP-Yield-Data for Fast Fission of U-234
*NAMYLD          NYNUCL     NFP (A8,*) for Block-4-1
*****************************************************
YU234F             1       119
*(NAMFIS(I),I=1,NYNUCL) (5(A8,2X)) for Block-4-2
U-234
*****************************************************
*NAMFP        YLDFP for Block-4-3, Repeat NFP Times
```

```
SE079       1.44355E-03
SE082       6.49521E-03
BR081       3.45597E-03
KR082       3.32281E-05
KR083       1.20541E-02
             :
             :
ZZ111       4.75726E-01
*
*****************************************************
*FP-Yield-Data for Thermal Fission of U-235
*****************************************************
YU235T              1       119
U-235
*****************************************************
*NAMFP         YLDFP for Block-4-3, Repeat NFP Times
SE079       4.46238E-04
SE082       3.23569E-03
             :
             :
ZZ111       4.25013E-01
*
*****************************************************
*FP-Yield-Data for Fast Fission of U-236
*****************************************************
YU236F              1       119
U-236
*****************************************************
*NAMFP         YLDFP for Block-4-3, Repeat NFP Times
SE079       1.13922E-03
SE082       3.83024E-03
             :
             :
ZZ111       4.01396E-01
*
*****************************************************
*FP-Yield-Data for Fast Fission of U-237
*****************************************************
YU237F              1       119
U-237
*****************************************************
*NAMFP         YLDFP for Block-4-3, Repeat NFP Times
SE079       7.51846E-04
SE082       3.06325E-03
             :
             :
ZZ111       3.64126E-01
*
*****************************************************
*FP-Yield-Data for Fast Fission of U-238
*****************************************************
YU238F              1       119
U-238
*****************************************************
*NAMFP         YLDFP for Block-4-3, Repeat NFP Times
SE079       3.26937E-04
SE082       2.13075E-03
             :
             :
             :
             :
*****************************************************
*FP-Yield-Data for Fast Fission of CM-246
*****************************************************
YCM246F             1       119
CM246
*****************************************************
*NAMFP         YLDFP for Block-4-3, Repeat NFP Times
```

```
SE079         1.60444E-04
SE082         6.81436E-04
                :
                :
ZZ111         2.63031E-01
*
**************************************************
```

4. Job Control Statements

In this chapter, job control statements will be described by widely used Bourne shell-script for the machine that works with the UNIX or similar operating system (e.g. Linux). A typical sample shell-script is shown below. In this sample, it is assumed that the user's home directory ($Home) is '/home/okumura', and the MOSRA-SRAC system is installed in the directory '/home/Administrator/'.

```
#!/bin/sh
#-----------------------------------------------------
#  <<  Sample shell script for MOSRA-SRAC execution  >>
#
#
#       At least, lines in bold face should be changed by each user
#
#
#-----------------------------------------------------
#
#============= to use native UNIX command
   alias  mkdir=mkdir
   alias  cat=cat                                    ..... (a)
   alias  cd=cd
   alias  rm=rm
#
#============= Set by user
#
#  MSRAC_DIR  : top directory name of MOSRA_SRAC
#  MSRAC_LIB  : top directory name of Public library
#  EXEC       : executable of MOSRA-SRAC
#  CHAIN      : burnup chain data file in ~MOSRA-SRAC/chain/
#  CASE       : arbitrary case name used for output file name
#  OUT_DIR    : directory for output of text files
#  PDS_DIR    : directory for output of PDS files
#  WKDR       : work directory for MOSRA-SRAC execution
#
   MSRAC_DIR=/home/Administrator/MOSRA-SRAC/MSRAC ..... (b)
   MSRAC_LIB=/home/Administrator/MOSRA-SRAC/MSRACLIB_J40 ..... (c)
   EXEC=$MSRAC_DIR/bin/MSRAC.exe ..... (d)
   CHAIN=$MSRAC_DIR/chain/u4cm6fp119bp14T_J40 ..... (e)
   CASE=Test-1 ..... (f)
   OUT_DIR=$HOME/Job/MyOutput ..... (g)
   PDS_DIR=$HOME/Job/Mypds/$CASE ..... (h)
#
   DATE=`date +%Y.%m.%d.%H.%M.%S` ..... (i)
   WKDR=$HOME/MSRACtmp.$CASE.$DATE ..... (j)
   mkdir $WKDR
#
#=======  mkdir for user's PDS
#
   if [ ! -e $PDS_DIR ]; then
     mkdir $PDS_DIR
   fi
   cd $PDS_DIR
   mkdir UMCROSS
   mkdir MACROWK                                     ..... (k)
   mkdir FLUX
   mkdir MACRO
   mkdir MICRO
   mkdir HOMOMIC
#
#=======  Set environment variables for all PDS files
#
```

```
   export PFAST=$MSRAC_LIB/pds/pfast
   export PTHERMAL=$MSRAC_LIB/pds/pthml
   export PMCROSS=$MSRAC_LIB/pds/pmcrs
   export UMCROSS=$PDS_DIR/UMCROSS
   export MACROWK=$PDS_DIR/MACROWK          ..... (l)
   export FLUX=$PDS_DIR/FLUX
   export MACRO=$PDS_DIR/MACRO
   export MICRO=$PDS_DIR/MICRO
   export HOMOMIC=$PDS_DIR/HMIC
#
#=======  Set environment variables for I/O devices
#
   export fu50=$CHAIN ..................................................... (m)
#  export fu97=$OUT_DIR/$CASE.FT97.$DATE ................................. (n)
#  export fu98=$OUT_DIR/$CASE.FT98.$DATE ................................. (o)
   export fu99=$OUT_DIR/$CASE.FT99.$DATE ................................. (p)
   OUTLST=$OUT_DIR/$CASE.FT06.$DATE ...................................... (q)
#
#======== Execution of MOSRA-SRAC with the following input data ==========
#
cd $WKDR
cat - << END_DATA | $EXEC >& $OUTLST................................... (r)
TEST
Sample input for UO2 unit pin cell                                  Input Data
by K. Okumura
0 0 0 0 0  0 0 0 0 0 / IOPT: General control
3 0 2 0 2  0 0 0 0 0 / IPRN:
0.0E0   / Buckling
**** Environment variable of each PDS (name defined in script)
PFAST
PTHERMAL
PMCROSS
UMCROSS
MACROWK                                                  ..... (s)
FLUX
MACRO
MICRO
HOMOMIC
           / MACRO-IN (not used)
           / MICRO-IN (not used)
**** energy group structure
200 118  2  1   /
118 200
         :
         :
         :
END_DATA................................................................ (t)
#
#========  Remove scratch files ==========================================
#
   cd $HOME
   rm -r $WKDR ........................................................... (u)
#
#========  Remove PDS files if you don't keep them =======================
#
#  rm -r $PDS_DIR
#
  rm -r $PDS_DIR/UMCROSS
  rm -r $PDS_DIR/MACROWRK
  rm -r $PDS_DIR/FLUX                    ..... (v)
# rm -r $PDS_DIR/MACRO
  rm -r $PDS_DIR/MICRO
  rm -r $PDS_DIR/HMIC
```

The following underlined statements indicate actions by the user when he uses the above shell-script.

(a) In the shell-script, several native UNIX commands are used, however, the user may customize their operations. By using 'alias' command, it is recovered to native ones during this process.

(b) Specify the MOSRA-SRAC module (~MSRAC) by full path directory name.

(c) Specify the Public Library (~MSRACLIB_J40) by full path directory name.

(d) Specify the executable (load module) of MOSRA-SRAC in the directory file ~MSRAC/bin/.

(e) Specify an appropriate burn-up chain library data that is used in burn-up calculation. [cf. Sect. 3.1]. Available burn-up chain data files (text data) are stored in ~MSRAC/chain/.

(f) Give arbitrary case name for current calculation process. This case name has no relation to the case name specified in MOSRA-SRAC input data 'CASEID' in Sect.2.2. The case name here is used to name output files, PDS files, and work directories described below [(h),(j),(l), (n)~(q)]. Therefore, too long name or name including inappropriate characters as file name should be avoided.

(g) Specify an existing user's directory for output files. All useful output files except PDS is stored in this directory.

(h) Specify user's directory, in which PDS files are created.

(i) The Unix command 'date' with formatted parameters gives current date and time as 'Dec25.11.22.33' (11h. 22m. 33s. on Christmas). This expression may be different by environment of user's system. The expression is connected to name of output file. If it is inappropriate on the system or dislikable, change it.

(j) The MOSRA-SRAC module is executed in the directory specified here. In this case, it is:
'/home/okumura/MSRACtmp.Test-1.Dec25.11.22.33'
Don't remove this directory during MOSRA-SRAC executing. Some scratch files may be made in it.

(k) Empty PDS files (=directories) are made here. In this sample, full path directory name of MACRO file is '/home/okumura/Job/Mypds/Test-1/MACRO'.

(l) Set environment variable for the full path directory name of each PDS. The variable is referred in the MOSRA-SRAC input data (s).

(m) The burn-up chain data specified in (e) is allocated to the 50th device.

(n) Set this line active (erase # in the first column), if plot option is used by PIJ or PEACO (cf. Sect. 2.3 and 2.6). Plot data is stored on the 89-th file. It is portable among machines, because it is a normal text file (PostScript file).

(o) Set this line active, when option for burn-up calculation is used (IOPT7=1 in Sect.2.2), and if summary table of burn-up calculation results is necessary.

(p) Major calculated results are printed on the file of 99-th device, but not on the standard output file.

Name of the file in this case is:
'/home/okumura/Job/MyOutput/Test-1.FT99. Dec25.11.22.33'.

(q) The name of the standard output file is set here in the same way as above (not FT99 but FT06). Information on progress of calculation and/or error messages are printed on this file. The user may browse it during MOSRA-SRAC execution but he must not edit and save it.

(r) The UNIX command 'cat' has a function to display files. Here, it is used to give the executable a standard input data of MOSRA-SRAC starting from next line to the line before 'END_DATA' [See (t)].
Pay attention to the fact that the character string starting from '$' is taken as an environmental variable even if it appears in the input data field. The character '$' must not be used, for instance, in comment line of MOSRA-SRAC input. As usual, an input data file can be separately specified by:
$EXEC < [input file] >& $OUTLST

(s) Specify the environmental variable defined at (l) in the standard input data.

(t) 'END_DATA' indicates end of the standard input data described above.

(u) Work directory is removed after the completion of MOSRA-SRAC calculation.

(v) Many member files are stored in User's PDS files. Remove them if they are not necessary. In this case, only MACRO file is preserved. If 'Core' is specified for a PDS at the MOSRA-SRAC input [cf. MOACS in Sect.2.2], all member files in the PDS are automatically removed by the code, although an empty directory is left.

5. Sample Input

Several typical examples for MOSRA-SRAC will be shown in this chapter. Input data files for the examples are stored in a directory: ~MSRAC/smpl/shr/.

5.1 Unit Pin Cell Calculation for UO_2 Fueled Rod

A sample input of the lattice calculation for the unit pin cell model of UO2 fueled rod (Fig.5.1-1) is shown below. This simple sample is suitable to confirm whether MOSRA-SRAC works well on user's computer or not. The execution time is a few seconds, and sample output data are stored in ~smpl/outp/Test.FT06.Sample and Test.FT99.Sample.

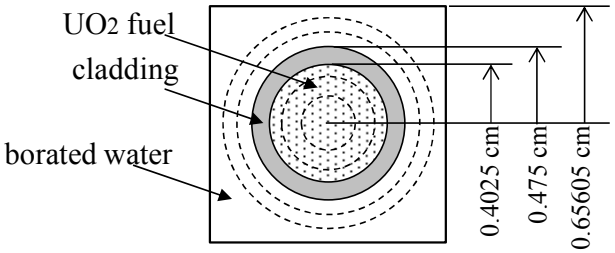


Fig.5.1-1 Unit pin cell geometry of the sample input for lattice calculation

An input data for this sample problem is shown below. [File name: Test.sh]

```
************************************************************ Start input
UO2F
unit pin cell calculation with PEACO
UO2 fuel rod of 17x17 PWR (Takahama-3)
0 0 0 0 1  0 0 0 0 0  10(0)  / IOPT(1-20)
1 9(0)                       / IP(1-10)
0.0                          / Buckling
************************************************************ PDS files
PFAST         / Public Fast Library
PTHERMAL      / Public Thermal Library
PMCROSS       / Public MCROSS Library
UMCROSS       / Users MCROSS Library
MACROWK       / Fine-group Macroscopic XS
FLUX          / Flux
MACRO         / Collapsed-group Macroscopic XS
MICRO         / Microscopic XS
HOMOMIC       / Homogenized Microscopic XS
              / Input MACRO (not used)
              / Input MICRO (not used)
************************************************************ Group structure
200  118  2   1   / NGF NEF NGB NGBF  (200g->2G)
     118    200   / ICOLNG
************************************************************ PIJ input
4 7 7 3 1   1 7 0 0 0   5 0 6 23 0   0 45 0 0 0   / Pij Control
1 1 1 2 3 3 3  /  R by T(=S)-region
```

```
3(1)           /  X by R-region
1 2 3          /  M by R-region
0.0 0.232383 0.328640 0.4025  0.475  0.5354 0.5957 0.65605 / RX
************************************************************ Materials
3 / NMAT
FUEL100X  0 5 1002.0   0.805   0.0  / 1 : Fuel (4.1w.t% U-235)
U02340    0 0  8.68037E-6      /1
U02350    0 0  9.64905E-4      /2
U02360    0 0  5.80013E-7      /3
U02380    0 0  2.22288E-2      /4
O00160    0 0  4.64060E-2      /5
CLAD100X  0 5  600.0   0.145   0.0  / 2 : Cladding
SN0000    0 0  4.2889E-4       /1
FE0000    0 0  1.3200E-4       /2
CR0000    0 0  5.9792E-5       /3
NI0000    0 0  3.7132E-5       /4
ZR0000    0 0  3.7784E-2       /5
MODER00X  0 4  559.14  1.0     0.0  / 3 : Moderator (borated water)
H0001H    0 0  5.04492E-2      /1
O00160    0 0  2.52246E-2      /2
B00100    0 0  5.65057E-6      /3  average boron density during burnup
B00110    0 0  2.31786E-5      /4
************************************************************ PEACO
   0          / IPLOT for PEACO
              / End of all case
************************************************************ End input
```

5.2 Burn-up Calculation for Assay Data of PWR Spent Fuel

This sample problem is based on the post irradiation examination for the spent fuel from Japanese PWR (Takahama-3)[28]. The lattice geometry is the same as that of the previous sample, i.e. the unit pin cell model of UO_2 whose cell pitch and other conditions are determined to represent the neutron spectrum at the position of the fuel sample (SF97-5) in a 17x17 type PWR fuel assembly. The histories of specific power and boron concentration are shown in Fig.5.2-1 and Fig.5.2-2, respectively. In the burn-up calculation, the fuel sample is exposed until 47.25 GWd/t according to the above burn-up histories and the spent fuel is cooled for 3.96 years after irradiation. At each burn-up step, neutron multiplication factor and fuel composition are evaluated and two group collapsed constants are preserved on the MACRO file.

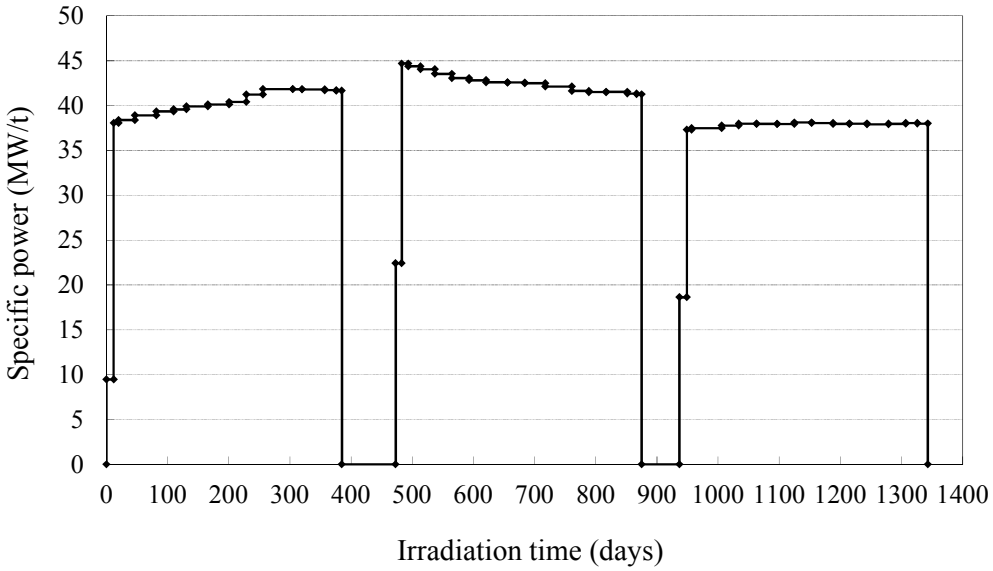


Fig.5.2-1 Power history of irradiated UO_2 fuel sample (SF97-5)

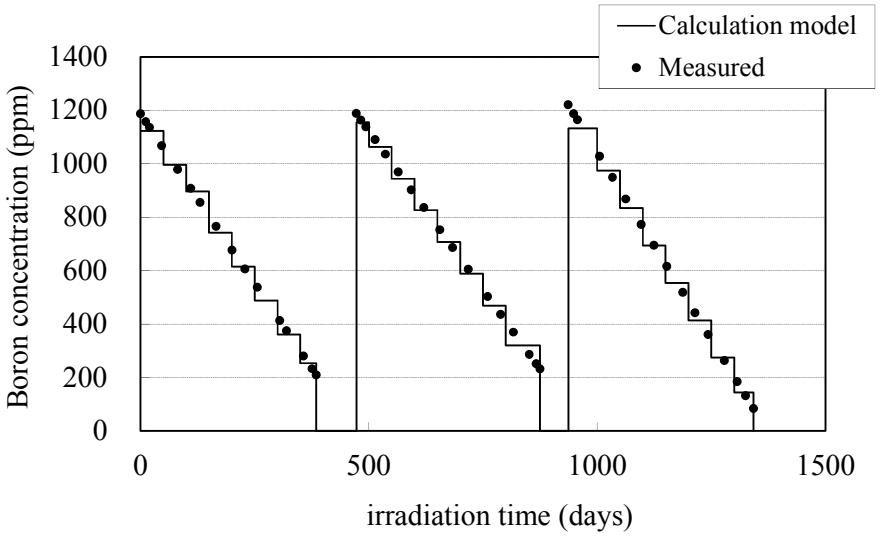


Fig.5.2-2 Born history assumed in the PIE analysis of UO_2 fuel sample (SF97-5).

An input data for this sample problem is shown below. [File name: PIE.sh]

```
************************************************************ Start input
S975
PIE analysis for sample fuel (SF97-5) irradiated in Takahama-3
Burnup calculation using a UO2 pin cell model
0 0 0 0 1  0 1 0 0 0  10(0)  / IOPT(1-20)
0 0 0 0 0  0 0 0 0 0         / IPRN(1-10)
0.0                          / Buckling
************************************************************ PDS files
```

```
PFAST         / Public Fast Library
PTHERMAL      / Public Thermal Library
PMCROSS       / Public MCROSS Library
UMCROSS       / Users MCROSS Library
MACROWK       / Fine-group Macroscopic XS
FLUX          / Flux
MACRO         / Collapsed-group Macroscopic XS
MICRO         / Microscopic XS
HOMOMIC       / Homogenized Microscopic XS
              / Input MACRO (not used)
              / Input MICRO (not used)
************************************************************ Group structure
200  118  2   1   / NGF NEF NGB NGBF  (200g->2G)
     118    200   / ICOLNG
************************************************************ PIJ input
4 7 7 3 1   1 7 0 0 0   5 0 6 23 0   0 45 0 0 0         / Pij Control
1 1 1 2 3 3 3  /  R-S
3(1)           /  X-R
1 2 3          /  M-R
0.0 0.232383 0.328640 0.4025  0.475  0.5354 0.5957 0.65605 / RX
************************************************************ Materials
3 / NMAT
FUEL100X  0 5 1002.0   0.805    0.0  / 1 : Fuel (4.1wt.% U-235)
U02340    0 0  8.68037E-6      /1
U02350    0 0  9.64905E-4      /2
U02360    0 0  5.80013E-7      /3
U02380    0 0  2.22288E-2      /4
O00160    0 0  4.64060E-2      /5
CLAD100X  0 5  600.0   0.145    0.0  / 2 : Cladding
SN0000    0 0  4.2889E-4       /1
FE0000    0 0  1.3200E-4       /2
CR0000    0 0  5.9792E-5       /3
NI0000    0 0  3.7132E-5       /4
ZR0000    0 0  3.7784E-2       /5
MODER00X  0 4  559.14  1.0      0.0  / 3 : Moderator (0.753957g/cc)
H0001H    0 0  5.04492E-2      /1
O00160    0 0  2.52246E-2      /2
** average boron density during burnup **
B00100    0 0  5.65057E-6      /3  Densities of B-10 & B-11 are replaced
B00110    0 0  2.31786E-5      /4  replaced by burnup input data
************************************************************ Cell burnup
* Burnup input for SF97-5
49 4 1 1 0  0 0 0 2 1  10(0)   / Control data for burnup calculation (IBC)
*** Power (MW/cm) : Initial heavy metal = 4.66567E-6(ton/cm)
4.41372E-05 1.77482E-04 1.79068E-04 1.81541E-04 1.83547E-04
1.84667E-04 1.86160E-04 1.87187E-04 1.88493E-04 1.92319E-04
1.95212E-04 1.95118E-04 1.94978E-04 1.94652E-04 1.94372E-04
0.00000E+00
1.04604E-04 2.08462E-04 2.07016E-04 2.05523E-04 2.03097E-04
2.00904E-04 1.99784E-04 1.98711E-04 1.98618E-04 1.98244E-04
1.96518E-04 1.94232E-04 1.93625E-04 1.93765E-04 1.92832E-04
1.92459E-04
0.00000E+00
8.69214E-05 1.73983E-04 1.74916E-04 1.76222E-04 1.77202E-04
1.77109E-04 1.77062E-04 1.77809E-04 1.77575E-04 1.77109E-04
1.77295E-04 1.76876E-04 1.77155E-04 1.77575E-04 1.77295E-04
0.00000E+00 / 47.25GWd/t
**** Burnup Period (step intervals in Day)
12.0  8.0 27.0 35.0 28.0   21.0 35.0 35.0 28.0 27.0
49.0 15.0 37.0 19.0  9.0
88.0   &  cooling
10.0 11.0 20.0 23.0 28.0   28.0 28.0 35.0 28.0 34.0
43.0 28.0 28.0 35.0 15.0    8.0
62.0   &  cooling
12.0  8.0 49.0 28.0 29.0   34.0 28.0 28.0 35.0 27.0
29.0 35.0 28.0 19.0 17.0   1445.4   / 3.96-years cooling
*** Burnup material indicator (when IBC10=1)
1 0 0 / No burnup for Boron in water
```

```
*** Change boron concentration in coolant during burnup
B00100  / Nuclide name
3     / Material position (moderator)
*-- B-10 atomic density from zero-step /IBC1+1/  : 0,1,2,...,IBC1
**********************
* Boron history (ppm)
**********************
*   1187 1156 1136 1067  978    907 854 765 676 605
*    537  412  374  280  232
*    209   & EO1C and cooling
*   1187 1163 1137 1089 1034    968 901 835 752 685
*    604  502  436  369  286    250
*    231   & EO2C and cooling
*   1220 1187 1164 1027  949    867 772 694 615 518
*    442  361  263  184  131    84
*     84
*   B-10 = 1.0925E-8*(water density)*ppm
*   B-11 = B-10 * 4.102
**********************
9.7732E-06 9.5223E-06 9.3550E-06 8.7904E-06 8.0586E-06
7.4731E-06 7.0340E-06 6.3022E-06 5.5704E-06 4.9849E-06
4.4203E-06 3.3958E-06 3.0821E-06 2.3085E-06 1.9112E-06
1.7230E-06 & EO1C and cooling
9.7765E-06 9.5807E-06 9.3653E-06 8.9736E-06 8.5232E-06
7.9749E-06 7.4267E-06 6.8784E-06 6.1930E-06 5.6447E-06
4.9789E-06 4.1369E-06 3.5886E-06 3.0403E-06 2.3550E-06
2.0613E-06
1.9046E-06 & EO2C and cooling
1.0050E-05 9.7733E-06 9.5888E-06 8.4590E-06 7.8134E-06
7.1448E-06 6.3608E-06 5.7152E-06 5.0696E-06 4.2626E-06
3.6401E-06 2.9715E-06 2.1645E-06 1.5189E-06 1.0808E-06
6.8882E-07 & EO3C
6.8882E-07  / dummy : keff is not calculated at the last step
0     / Material position = 0 : End of B-10 input
**********************
B00110  / Nuclide name
3     / Material position (moderator)
*-- B-11 atomic density from zero-step /IBC1+1/  : 0,1,2,...,IBC1
4.0090E-05 3.9060E-05 3.8374E-05 3.6058E-05 3.3056E-05
3.0655E-05 2.8854E-05 2.5852E-05 2.2850E-05 2.0448E-05
1.8132E-05 1.3929E-05 1.2643E-05 9.4693E-06 7.8396E-06
7.0677E-06 & EO1C and cooling
4.0103E-05 3.9300E-05 3.8416E-05 3.6810E-05 3.4962E-05
3.2713E-05 3.0464E-05 2.8215E-05 2.5404E-05 2.3155E-05
2.0424E-05 1.6970E-05 1.4721E-05 1.2471E-05 9.6601E-06
8.4553E-06
7.8127E-06 & EO2C and cooling
4.1225E-05 4.0090E-05 3.9333E-05 3.4699E-05 3.2051E-05
2.9308E-05 2.6092E-05 2.3444E-05 2.0796E-05 1.7485E-05
1.4932E-05 1.2189E-05 8.8786E-06 6.2304E-06 4.4334E-06
2.8255E-06 & EO3C
2.8255E-06  / dummy : keff is not calculated at the last step
0     / Material position = 0 : End of B-11 input
************************************************************ PEACO
   0          / IPLOT for PEACO
              / END OF ALL CASE
************************************************************ End input
```

Major results of the burn-up calculation are printed on the 98-th device (file name: PIE. FT98) A part of the output list is shown below.

```
------------------------------------------------------------------------------------------------------------------------
 <<<< RESULT-OF-DEPLETION-CALCULATION >>>>
 CASEID=S975
 TITLE: PIE analysis for sample fuel (SF97-5) irradiated in Takahama-3 :
```

```
 :
 * DAYS         0.00000e+00 1.20000e+01 2.00000e+01 4.70000e+01 8.20000e+01 1.10000e+02
 * MWD/TON      0.00000e+00 1.13520e+02 4.17841e+02 1.45410e+03 2.81595e+03 3.91747e+03
 * U02350-%     0.00000e+00 3.31403e-01 1.21532e+00 4.16192e+00 7.89172e+00 1.08027e+01

 * K-EFF        1.265021    1.245072    1.223782    1.219057    1.214904    1.210604
 * K-INF        1.265021    1.245072    1.223782    1.219057    1.214904    1.210604

 * INST.-C.R.   0.412428    0.417486    0.423066    0.423970    0.427489    0.432142
 * INTE.-C.R.   0.412428    0.416479    0.418452    0.421695    0.423551    0.425254
 * MWD          0.00000e+00 5.29646e-04 1.94950e-03 6.78434e-03 1.31383e-02 1.82776e-02
 * POWER(MW)    4.41372e-05 1.77482e-04 1.79068e-04 1.81541e-04 1.83547e-04 1.84667e-04
 * TON-HM       4.66566e-06 4.66511e-06 4.66364e-06 4.65865e-06 4.65208e-06 4.64678e-06
 * FLUX-LEVEL   6.44495e+13 2.62892e+14 2.69531e+14 2.74943e+14 2.79791e+14 2.83156e+14
 * FIS.-ABSOR   1.57717e+12 6.34458e+12 6.40951e+12 6.54366e+12 6.66822e+12 6.74540e+12
 * FIS.-DECAY   0.00000e+00 0.00000e+00 0.00000e+00 0.00000e+00 0.00000e+00 0.00000e+00
 * FER.-CAPTR   6.50471e+11 2.64877e+12 2.71165e+12 2.77432e+12 2.85059e+12 2.91497e+12
 * PRE.-DECAY   0.00000e+00 0.00000e+00 0.00000e+00 0.00000e+00 0.00000e+00 0.00000e+00

MATERIAL-NO.=  1   VOLUME= 5.08957e-01 (CC)   WEIGHT= 9.16710e-06 (TON/CC)   MATERIAL-TYPE IS FUEL


 * DAYS         0.00000e+00 1.20000e+01 2.00000e+01 4.70000e+01 8.20000e+01 1.10000e+02
 * MWD/TON      0.00000e+00 1.13520e+02 4.17841e+02 1.45410e+03 2.81595e+03 3.91747e+03
 * POW(MW/CC)   8.67208e-05 3.48717e-04 3.51833e-04 3.56692e-04 3.60633e-04 3.62834e-04
 * ENRGY/FIS.   3.24366e-11 3.24406e-11 3.24504e-11 3.24916e-11 3.25402e-11 3.25757e-11
 * XE-135-YD.   2.43428e-03 2.45251e-03 2.50301e-03 2.73833e-03 3.01214e-03 3.20803e-03
 *  I-135-YD.   6.29462e-02 6.29588e-02 6.29831e-02 6.30664e-02 6.31649e-02 6.32384e-02
 * SM-149-YD.   1.64335e-12 3.69207e-12 9.02694e-12 3.29046e-11 6.06821e-11 8.05830e-11
 * PM-149-YD.   1.10857e-02 1.10941e-02 1.11088e-02 1.11521e-02 1.12034e-02 1.12420e-02
 ---- -------- ----------- ----------- ----------- ----------- ----------- -----------
    1 U02340    8.68037e-06 8.66885e-06 8.63770e-06 8.53156e-06 8.39371e-06 8.28350e-06
    2 U02350    9.64905e-04 9.61707e-04 9.53178e-04 9.24746e-04 8.88757e-04 8.60670e-04
    3 U02360    5.80013e-07 1.17232e-06 2.75575e-06 8.03408e-06 1.46858e-05 1.98516e-05
    4 U02370    0.00000e+00 4.27483e-09 1.68940e-08 3.47753e-08 4.90668e-08 5.97324e-08
    5 U02380    2.22288e-02 2.22273e-02 2.22233e-02 2.22095e-02 2.21912e-02 2.21763e-02
    6 NP2370    0.00000e+00 3.12533e-09 1.23135e-08 8.73695e-08 2.34735e-07 3.84256e-07
    7 NP2390    0.00000e+00 3.64331e-07 1.41696e-06 1.55505e-06 1.58616e-06 1.61625e-06
      :
      :
-------------------------------------------------------------------------------------------------------------------------
```

5.3 Burn-up Calculation of a PWR Fuel Assembly

This sample provides the burnup calculation for a typical 17x17 type PWR fuel assembly, which consists of 264 low U-235 enriched UO_2 fuels (no poisoned rods) and 25 guide thimbles for cluster type control rods and central instrumentation. In the calculation, a quarter part of the fuel assembly surrounded by small water gap is modeled by using reflective boundary conditions. The region maps of the Sub-region, T-region and R-region corresponding to the sample input are shown in Figs.5.3-1, 5.3-2 and 5.3-3, respectively.

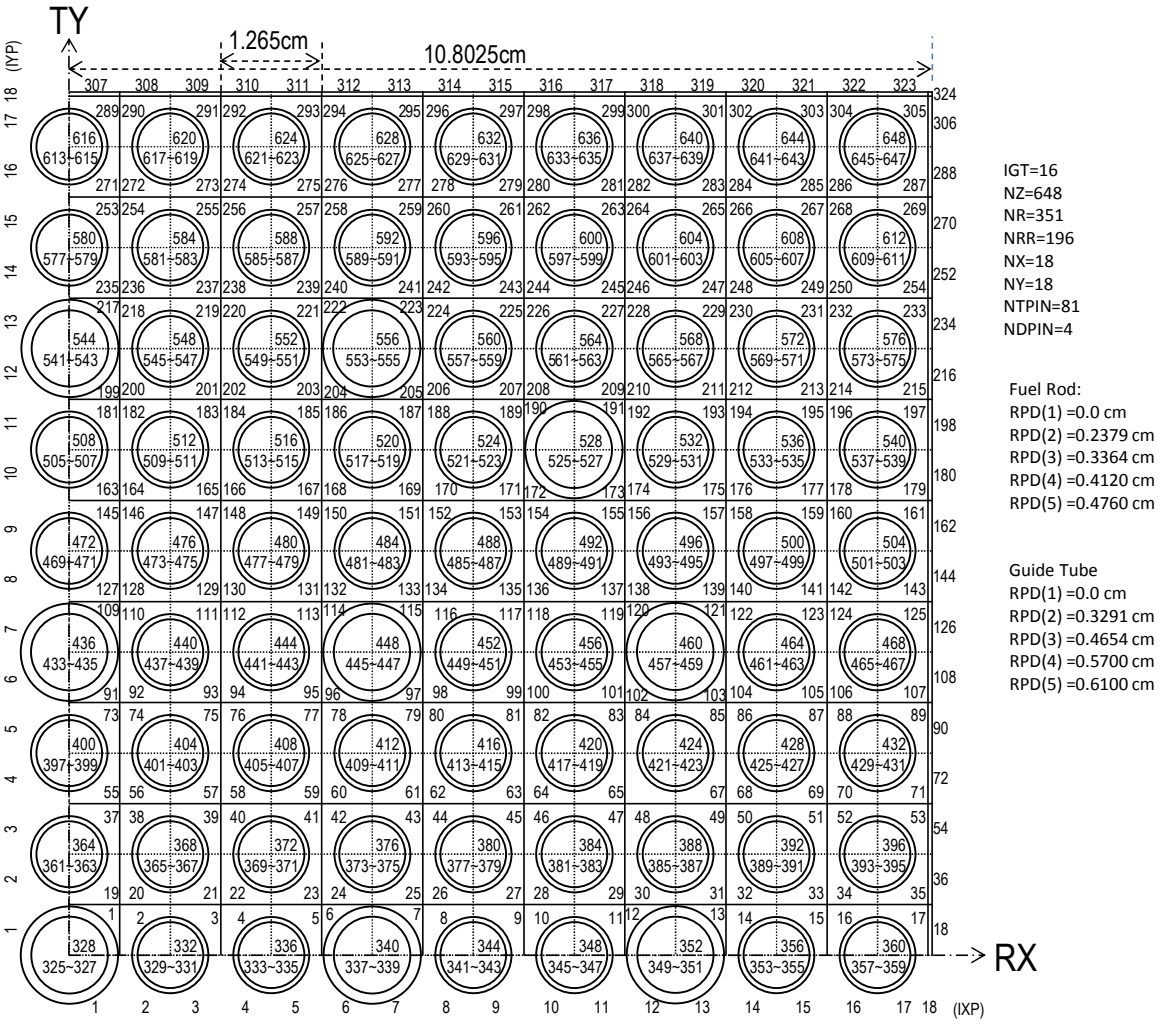


Fig.5.3-1 Sub-region map for a 17x17 type PWR fuel assembly

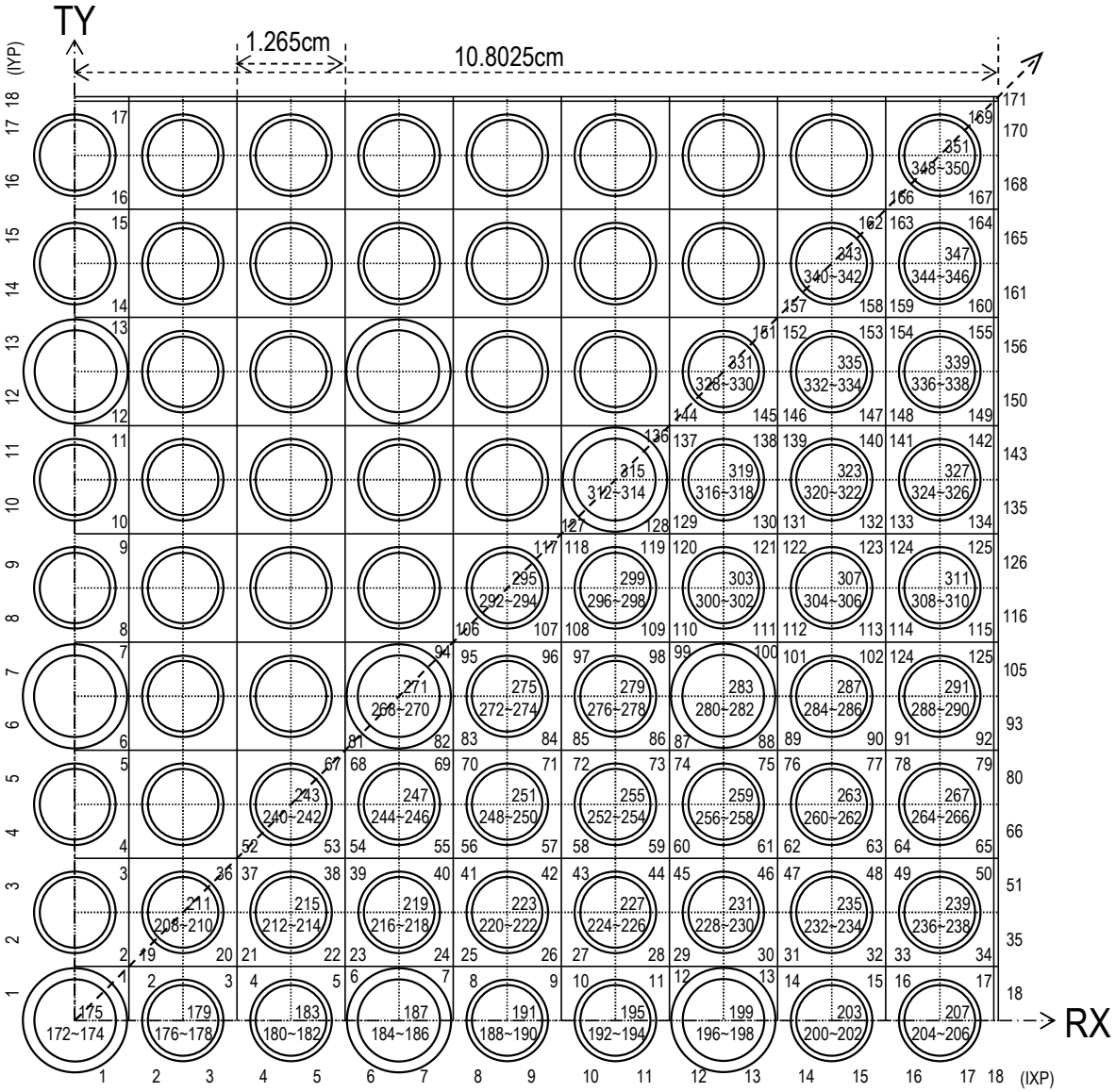


Fig.5.3-2 T-region map for a 17x17 type PWR fuel assembly

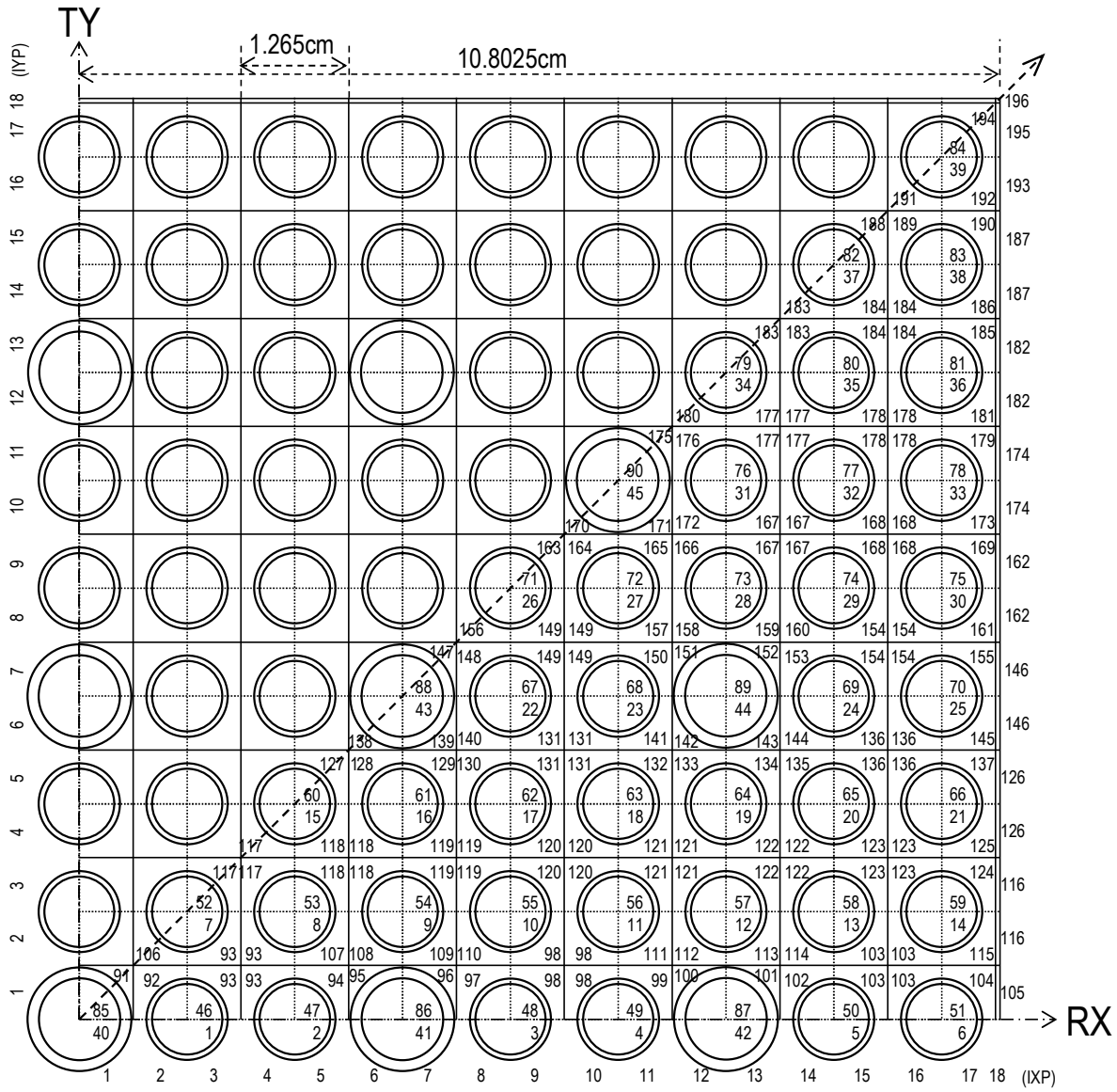


Fig.5.3-3 R-region map for a 17x17 type PWR fuel assembly

An input data for this sample problem is shown below. [File name: PWR17x17.sh]

```
************************************************************ Start input
PWRA
PWR Assembly Burnup Calc.
17x17 rod, 4.1wt% U-235
 0 0 0 0 2  0 1 0 0 0    9(0) 1 / IOPT(20)
 0 0 0 0 1  0 0 0 0 0          / IOPT(21)-IOPT(30)
 0 0 0 0 0  0 0 0 0 0          / IPRN(10)
 0.0      / Buckling
************************************************************ PDS files
PFAST         / PDS NAME FOR PUBLIC FAST LIBRAY
PTHERMAL      / PDS NAME FOR PUBLIC THERMAL LIBRAY
PMCROSS       / PDS NAME FOR PUBLIC MCROSS LIBRAY
```

```
UMCROSS        / PDS NAME FOR USER'S MCROSS LIBRAY
MACROWK        / PDS NAME FOR USER'S FINE GROUP MACRO. X-SECTION
FLUX           / PDS NAME FOR FLUX  (OUTPUT)
MACRO          / PDS NAME FOR BROAD GROUP EFF. MACRO. X-SECTION (OUTPUT)
MICRO          / PDS NAME FOR FINE  GROUP EFF. MICRO. X-SECTION (OUTPUT)
MICROHM        / PDS NAME FOR BROAD GROUP EFF. HOMOG. MICRO. XS (OUTPUT)
               / PDS NAME FOR MACRO-IN IF NEED
               / PDS NAME FOR MICRO-IN IF NEED
************************************************************ Group structure
 200  118    0 0   / NGF NEF  NGB NGBF
************************************************************** PIJ input
16 648 351 196  1   1 18 18 81 0   2 0 25 91 4   1 90 0 0 0         / Pij Control
*
*** Block 3 T-S
   1   2   3   4   5   6   7   8   9  10  11  12  13  14  15
  16  17  18   2  19  20  21  22  23  24  25  26  27  28  29
  30  31  32  33  34  35   3  20  36  37  38  39  40  41  42
  43  44  45  46  47  48  49  50  51   4  21  37  52  53  54
  55  56  57  58  59  60  61  62  63  64  65  66   5  22  38
  53  67  68  69  70  71  72  73  74  75  76  77  78  79  80
   6  23  39  54  68  81  82  83  84  85  86  87  88  89  90
  91  92  93   7  24  40  55  69  82  94  95  96  97  98  99
 100 101 102 103 104 105   8  25  41  56  70  83  95 106 107
 108 109 110 111 112 113 114 115 116   9  26  42  57  71  84
  96 107 117 118 119 120 121 122 123 124 125 126  10  27  43
  58  72  85  97 108 118 127 128 129 130 131 132 133 134 135
  11  28  44  59  73  86  98 109 119 128 136 137 138 139 140
 141 142 143  12  29  45  60  74  87  99 110 120 129 137 144
 145 146 147 148 149 150  13  30  46  61  75  88 100 111 121
 130 138 145 151 152 153 154 155 156  14  31  47  62  76  89
 101 112 122 131 139 146 152 157 158 159 160 161  15  32  48
  63  77  90 102 113 123 132 140 147 153 158 162 163 164 165
  16  33  49  64  78  91 103 114 124 133 141 148 154 159 163
 166 167 168  17  34  50  65  79  92 104 115 125 134 142 149
 155 160 164 167 169 170  18  35  51  66  80  93 105 116 126
 135 143 150 156 161 165 168 170 171 172 173 174 175 176 177
 178 179 180 181 182 183 184 185 186 187 188 189 190 191 192
 193 194 195 196 197 198 199 200 201 202 203 204 205 206 207
 176 177 178 179 208 209 210 211 212 213 214 215 216 217 218
 219 220 221 222 223 224 225 226 227 228 229 230 231 232 233
 234 235 236 237 238 239 180 181 182 183 212 213 214 215 240
 241 242 243 244 245 246 247 248 249 250 251 252 253 254 255
 256 257 258 259 260 261 262 263 264 265 266 267 184 185 186
 187 216 217 218 219 244 245 246 247 268 269 270 271 272 273
 274 275 276 277 278 279 280 281 282 283 284 285 286 287 288
 289 290 291 188 189 190 191 220 221 222 223 248 249 250 251
 272 273 274 275 292 293 294 295 296 297 298 299 300 301 302
 303 304 305 306 307 308 309 310 311 192 193 194 195 224 225
 226 227 252 253 254 255 276 277 278 279 296 297 298 299 312
 313 314 315 316 317 318 319 320 321 322 323 324 325 326 327
 196 197 198 199 228 229 230 231 256 257 258 259 280 281 282
 283 300 301 302 303 316 317 318 319 328 329 330 331 332 333
 334 335 336 337 338 339 200 201 202 203 232 233 234 235 260
 261 262 263 284 285 286 287 304 305 306 307 320 321 322 323
 332 333 334 335 340 341 342 343 344 345 346 347 204 205 206
 207 236 237 238 239 264 265 266 267 288 289 290 291 308 309
 310 311 324 325 326 327 336 337 338 339 344 345 346 347 348
 349 350 351 /
*
*** Block 4 R-T
  91  92  93  93  94  95  96  97  98  98  99 100 101 102 103
 103 104 105 106  93  93 107 108 109 110  98  98 111 112 113
 114 103 103 115 116 117 117 118 118 119 119 120 120 121 121
 122 122 123 123 124 116 117 118 118 119 119 120 120 121 121
 122 122 123 123 125 126 127 128 129 130 131 131 132 133 134
 135 136 136 137 126 138 139 140 131 131 141 142 143 144 136
 136 145 146 147 148 149 149 150 151 152 153 154 154 155 146
 156 149 149 157 158 159 160 154 154 161 162 163 164 165 166
```

```
 167 167 168 168 169 162 170 171 172 167 167 168 168 173 174
 175 176 177 177 178 178 179 174 180 177 177 178 178 181 182
 183 183 184 184 185 182 183 184 184 186 187 188 189 190 187
 191 192 193 194 195 196  40  40  40  85   1   1   1  46   2
   2   2  47  41  41  41  86   3   3   3  48   4   4   4  49
  42  42  42  87   5   5   5  50   6   6   6  51   7   7   7
  52   8   8   8  53   9   9   9  54  10  10  10  55  11  11
  11  56  12  12  12  57  13  13  13  58  14  14  14  59  15
  15  15  60  16  16  16  61  17  17  17  62  18  18  18  63
  19  19  19  64  20  20  20  65  21  21  21  66  43  43  43
  88  22  22  22  67  23  23  23  68  44  44  44  89  24  24
  24  69  25  25  25  70  26  26  26  71  27  27  27  72  28
  28  28  73  29  29  29  74  30  30  30  75  45  45  45  90
  31  31  31  76  32  32  32  77  33  33  33  78  34  34  34
  79  35  35  35  80  36  36  36  81  37  37  37  82  38  38
  38  83  39  39  39  84 /
*
*** Block 5 X-R
196(1) /
*
*** Block 6 M-R
  1  2  3  4  5  6  7  8  9 10 11 12 13 14 15 16 17 18 19 20
 21 22 23 24 25 26 27 28 29 30 31 32 33 34 35 36 37 38 39 41
 41 41 41 41 41 40 40 40 40 40 40 40 40 40 40 40 40 40 40 40
 40 40 40 40 40 40 40 40 40 40 40 40 40 40 40 40 40 40 40 40
 40 40 40 40 40 40 40 40 40 40 41 41 41 41 41 41 41 41 41 41
 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41
 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41
 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41
 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41
 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 41 /
*
*** Block 8 RX
 0.0000   0.6325   1.2650  1.8975  2.5300  3.1625  3.7950  4.4275
 5.0600   5.6925   6.3250  6.9575  7.5900  8.2225  8.8550  9.4875
10.1200  10.7525  10.8025  /
*
*** Block 9' TY
 0.0000   0.6325   1.2650  1.8975  2.5300  3.1625  3.7950  4.4275
 5.0600   5.6925   6.3250  6.9575  7.5900  8.2225  8.8550  9.4875
10.1200  10.7525  10.8025  /
*
*** Block 10'' IXP
  1  3  5  7  9 11 13 15 17  1  3  5  7  9 11 13 15 17  1  3
  5  7  9 11 13 15 17  1  3  5  7  9 11 13 15 17  1  3  5  7
  9 11 13 15 17  1  3  5  7  9 11 13 15 17  1  3  5  7  9 11
 13 15 17  1  3  5  7  9 11 13 15 17  1  3  5  7  9 11 13 15
 17 /
*** Block 11' IYP
  1  1  1  1  1  1  1  1  1  3  3  3  3  3  3  3  3  3  5  5
  5  5  5  5  5  5  5  7  7  7  7  7  7  7  7  7  9  9  9  9
  9  9  9  9  9 11 11 11 11 11 11 11 11 11 13 13 13 13 13 13
 13 13 13 15 15 15 15 15 15 15 15 15 17 17 17 17 17 17 17 17
 17 /
*
*** Block 12' RDP
0 0.3291 0.4654 0.5700 0.6100  &  1- 1 T
0 0.2379 0.3364 0.4120 0.4760  &  3- 1 F
0 0.2379 0.3364 0.4120 0.4760  &  5- 1 F
0 0.3291 0.4654 0.5700 0.6100  &  7- 1 T
0 0.2379 0.3364 0.4120 0.4760  &  9- 1 F
0 0.2379 0.3364 0.4120 0.4760  & 11- 1 F
0 0.3291 0.4654 0.5700 0.6100  & 13- 1 T
0 0.2379 0.3364 0.4120 0.4760  & 15- 1 F
0 0.2379 0.3364 0.4120 0.4760  & 17- 1 F
0 0.2379 0.3364 0.4120 0.4760  &  1- 3 F
0 0.2379 0.3364 0.4120 0.4760  &  3- 3 F
0 0.2379 0.3364 0.4120 0.4760  &  5- 3 F
```

```
0 0.2379 0.3364 0.4120 0.4760  &  7- 3 F
0 0.2379 0.3364 0.4120 0.4760  &  9- 3 F
0 0.2379 0.3364 0.4120 0.4760  & 11- 3 F
0 0.2379 0.3364 0.4120 0.4760  & 13- 3 F
0 0.2379 0.3364 0.4120 0.4760  & 15- 3 F
0 0.2379 0.3364 0.4120 0.4760  & 17- 3 F
0 0.2379 0.3364 0.4120 0.4760  &  1- 5 F
0 0.2379 0.3364 0.4120 0.4760  &  3- 5 F
0 0.2379 0.3364 0.4120 0.4760  &  5- 5 F
0 0.2379 0.3364 0.4120 0.4760  &  7- 5 F
0 0.2379 0.3364 0.4120 0.4760  &  9- 5 F
0 0.2379 0.3364 0.4120 0.4760  & 11- 5 F
0 0.2379 0.3364 0.4120 0.4760  & 13- 5 F
0 0.2379 0.3364 0.4120 0.4760  & 15- 5 F
0 0.2379 0.3364 0.4120 0.4760  & 17- 5 F
0 0.3291 0.4654 0.5700 0.6100  &  1- 7 T
0 0.2379 0.3364 0.4120 0.4760  &  3- 7 F
0 0.2379 0.3364 0.4120 0.4760  &  5- 7 F
0 0.3291 0.4654 0.5700 0.6100  &  7- 7 T
0 0.2379 0.3364 0.4120 0.4760  &  9- 7 F
0 0.2379 0.3364 0.4120 0.4760  & 11- 7 F
0 0.3291 0.4654 0.5700 0.6100  & 13- 7 T
0 0.2379 0.3364 0.4120 0.4760  & 15- 7 F
0 0.2379 0.3364 0.4120 0.4760  & 17- 7 F
0 0.2379 0.3364 0.4120 0.4760  &  1- 9 F
0 0.2379 0.3364 0.4120 0.4760  &  3- 9 F
0 0.2379 0.3364 0.4120 0.4760  &  5- 9 F
0 0.2379 0.3364 0.4120 0.4760  &  7- 9 F
0 0.2379 0.3364 0.4120 0.4760  &  9- 9 F
0 0.2379 0.3364 0.4120 0.4760  & 11- 9 F
0 0.2379 0.3364 0.4120 0.4760  & 13- 9 F
0 0.2379 0.3364 0.4120 0.4760  & 15- 9 F
0 0.2379 0.3364 0.4120 0.4760  & 17- 9 F
0 0.2379 0.3364 0.4120 0.4760  &  1-11 F
0 0.2379 0.3364 0.4120 0.4760  &  3-11 F
0 0.2379 0.3364 0.4120 0.4760  &  5-11 F
0 0.2379 0.3364 0.4120 0.4760  &  7-11 F
0 0.2379 0.3364 0.4120 0.4760  &  9-11 F
0 0.3291 0.4654 0.5700 0.6100  & 11-11 T
0 0.2379 0.3364 0.4120 0.4760  & 13-11 F
0 0.2379 0.3364 0.4120 0.4760  & 15-11 F
0 0.2379 0.3364 0.4120 0.4760  & 17-11 F
0 0.3291 0.4654 0.5700 0.6100  &  1-13 T
0 0.2379 0.3364 0.4120 0.4760  &  3-13 F
0 0.2379 0.3364 0.4120 0.4760  &  5-13 F
0 0.3291 0.4654 0.5700 0.6100  &  7-13 T
0 0.2379 0.3364 0.4120 0.4760  &  9-13 F
0 0.2379 0.3364 0.4120 0.4760  & 11-13 F
0 0.2379 0.3364 0.4120 0.4760  & 13-13 F
0 0.2379 0.3364 0.4120 0.4760  & 15-13 F
0 0.2379 0.3364 0.4120 0.4760  & 17-13 F
0 0.2379 0.3364 0.4120 0.4760  &  1-15 F
0 0.2379 0.3364 0.4120 0.4760  &  3-15 F
0 0.2379 0.3364 0.4120 0.4760  &  5-15 F
0 0.2379 0.3364 0.4120 0.4760  &  7-15 F
0 0.2379 0.3364 0.4120 0.4760  &  9-15 F
0 0.2379 0.3364 0.4120 0.4760  & 11-15 F
0 0.2379 0.3364 0.4120 0.4760  & 13-15 F
0 0.2379 0.3364 0.4120 0.4760  & 15-15 F
0 0.2379 0.3364 0.4120 0.4760  & 17-15 F
0 0.2379 0.3364 0.4120 0.4760  &  1-17 F
0 0.2379 0.3364 0.4120 0.4760  &  3-17 F
0 0.2379 0.3364 0.4120 0.4760  &  5-17 F
0 0.2379 0.3364 0.4120 0.4760  &  7-17 F
0 0.2379 0.3364 0.4120 0.4760  &  9-17 F
0 0.2379 0.3364 0.4120 0.4760  & 11-17 F
0 0.2379 0.3364 0.4120 0.4760  & 13-17 F
0 0.2379 0.3364 0.4120 0.4760  & 15-17 F
```

```
0 0.2379 0.3364 0.4120 0.4760 /& 17-17 F
************************************************************ Materials
41 / NMAT
FL001INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL002INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL003INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL004INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL005INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL006INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL007INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL008INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL009INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL010INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL011INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL012INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL013INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL014INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL015INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL016INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL017INI   1  3   968.8 0.824    0.0
```

```
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL018INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL019INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL020INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL021INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL022INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL023INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL024INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL025INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL026INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL027INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL028INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL029INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL030INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL031INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL032INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL033INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL034INI   1  3   968.8 0.824    0.0
```

```
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL035INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL036INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL037INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL038INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
FL039INI   1  3   968.8 0.824    0.0
U02350     0  0   9.349E-4
U02380     0  0   2.159E-2
O00160     0  0   4.505E-2
ZRCLYINI   0 28   604.0 0.000    0.0
ZR0900     0  0   2.168E-2
ZR0910     0  0   4.727E-3
ZR0920     0  0   7.225E-3
ZR0940     0  0   7.322E-3
ZR0960     0  0   1.180E-3
CR0500     0  0   3.271E-6
CR0520     0  0   6.308E-5
CR0530     0  0   7.153E-6
CR0540     0  0   1.780E-6
FE0540     0  0   8.604E-6
FE0560     0  0   1.351E-4
FE0570     0  0   3.119E-6
FE0580     0  0   4.151E-7
NI0580     0  0   2.497E-5
NI0600     0  0   9.619E-6
NI0610     0  0   4.181E-7
NI0620     0  0   1.333E-6
NI0640     0  0   3.395E-7
SN1120     0  0   4.638E-6
SN1140     0  0   3.156E-6
SN1150     0  0   1.626E-6
SN1160     0  0   6.952E-5
SN1170     0  0   3.672E-5
SN1180     0  0   1.158E-4
SN1190     0  0   4.107E-5
SN1200     0  0   1.558E-4
SN1220     0  0   2.214E-5
SN1240     0  0   2.768E-5
WATERINI   0  4   574.2 0.000    0.0
H0001H     0  0   4.844E-2
O00160     0  0   2.422E-2
B00100     0  0   4.419E-6
B00110     0  0   1.779E-5
************************************************************ Cell Burnup
  29 1 1 0 0  0 1 0 0 1  0 1 8(0)   / Control Data for Burnup Calculation
  25(1.1746E-02) 4(0)
1.00E+3 2.00E+3 3.00E+3 4.00E+3 5.00E+3
6.00E+3 7.00E+3 8.00E+3 9.00E+3 1.00E+4
1.25E+4 1.50E+4 1.75E+4 2.00E+4 2.25E+4
2.50E+4 2.75E+4 3.00E+4 3.25E+4 3.50E+4
3.75E+4 4.00E+4 4.50E+4 5.00E+4 5.50E+4  & Exp (MWd/t)
4(-1826.2)          /      Cooling time (in Day; = 5 years)
  39(1)   2(0)      /  Specify Burnable Material Region (required when IBC10 != 0) ---- Block-09
  25(1)   4(0)      /  Specify Steps with PC method     (required when IBC12 != 0) ---- Block-12
```

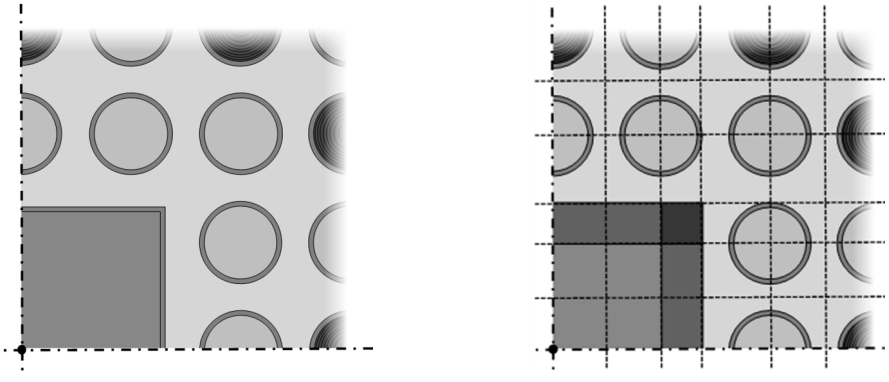
```
************************************************************ PEACO
   0          / IPLOT for PEACO
              / END OF ALL CASE
************************************************************ End input
```

5.4 Burn-up Calculation of a BWR Fuel Assembly

In this section, as a sample of the burn-up calculation for a typical BWR fuel assembly with burnable poisoned fuels (UO_2-Gd_2O_3), we employ the burnup benchmark "Burnup Credit Criticality Benchmark Phase IIIC"[29], which has been conducted by the Expert Group on Burnup Credit Criticality Safety of Working Party of Nuclear Criticality Safety (WPNCS) under the Nuclear Science Committee (NSC) of OECD/NEA. The fuel assembly model is based on the "9x9 B" type of BWR fuel called "STEP-3 BWR fuel" in Japan. Similar types of fuels were loaded in the cores of unit-1 and unit-2 of TEPCO's Fukushima Daiichi Nuclear Power Station.

There are six kinds of UO_2 fuels whose U-235 enrichments are different. One of them is poisoned fuel (UO_2-Gd_2O_3). There is a water channel filled by saturated water at the center of the fuel assembly as well as the water gap surrounding the fuel assembly. Although three voided moderator conditions (0%, 40%, 70%) are treated in the benchmark, only the case of 40% voided one is taken here as a sample problem.

From the restriction on geometry description in PIJ, the channel box and a part of moderator regions are smeared so that the total number of atoms in the smeared area is conserved (See Fig.5.4-1). The effect of this approximation was confirmed to be small[30] by comparison with the results of MVP-BURN[31] using a continuous-energy Monte Caro code MVP[32].



Benchmark geometry　　Calculation model

Fig.5.4-1　Benchmark geometry and calculation model

The region maps of the Sub-Region, T-region and R-region corresponding to the input data for analysis of the benchmark problem are shown in Figs.5.4-2, 5.4-3 and 5.4-4, respectively.

In the burnup calculation, the number of the burnup steps is 89 up to 50 GWd/t. For all time steps except for the first step, the Predictor-Collector method is applied. The calculated results are stored in the directory '~smp/outp/BWR9x9.FT06, BWR9x9.FT98 and BWR9x9.FT99.

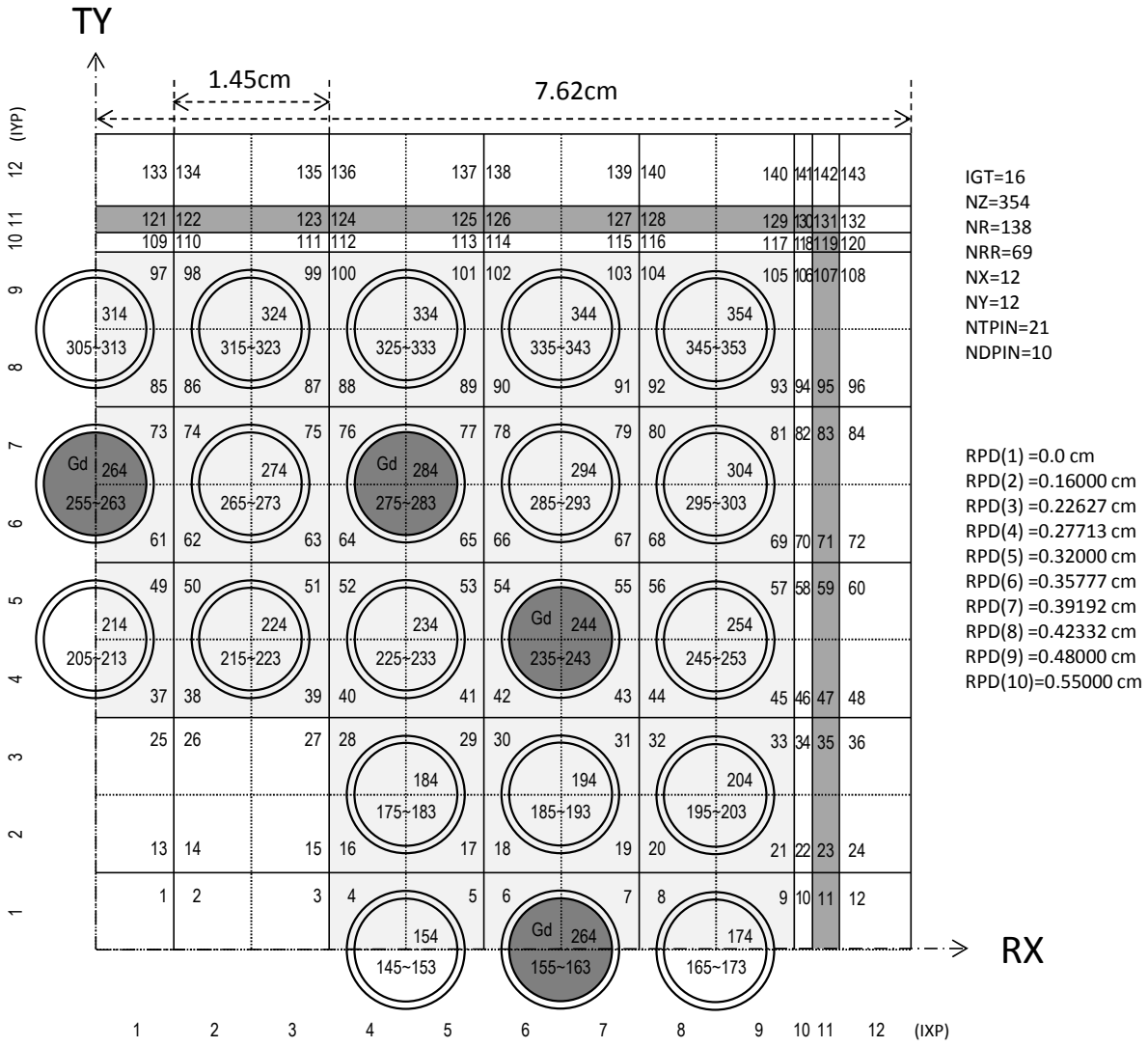


Fig.5.4-2 Sub-region map for a BWR fuel assembly (9x9 B, STEP-3)

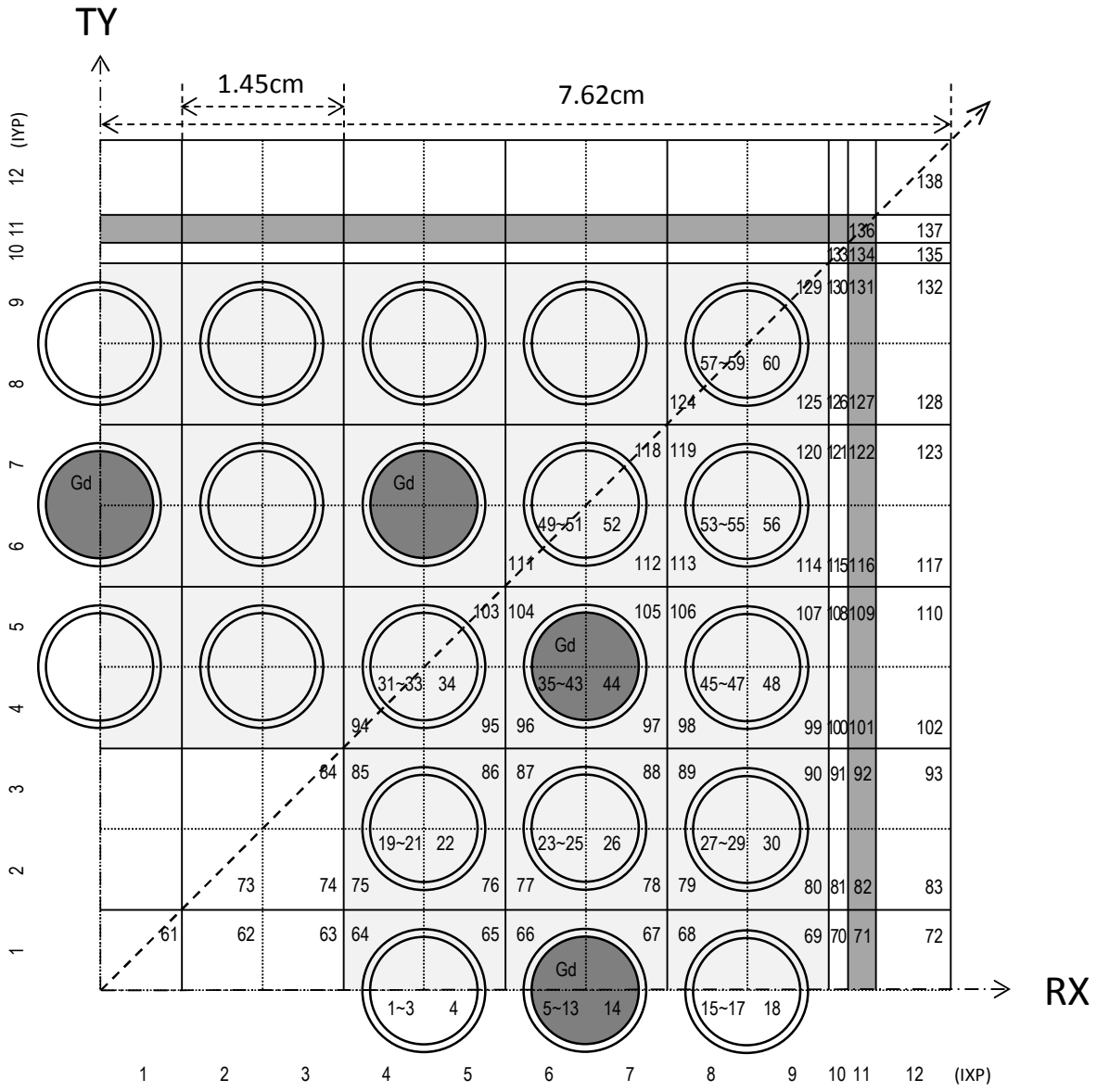


Fig.5.4-3 T-region map for a BWR fuel assembly (9x9 B, STEP-3)

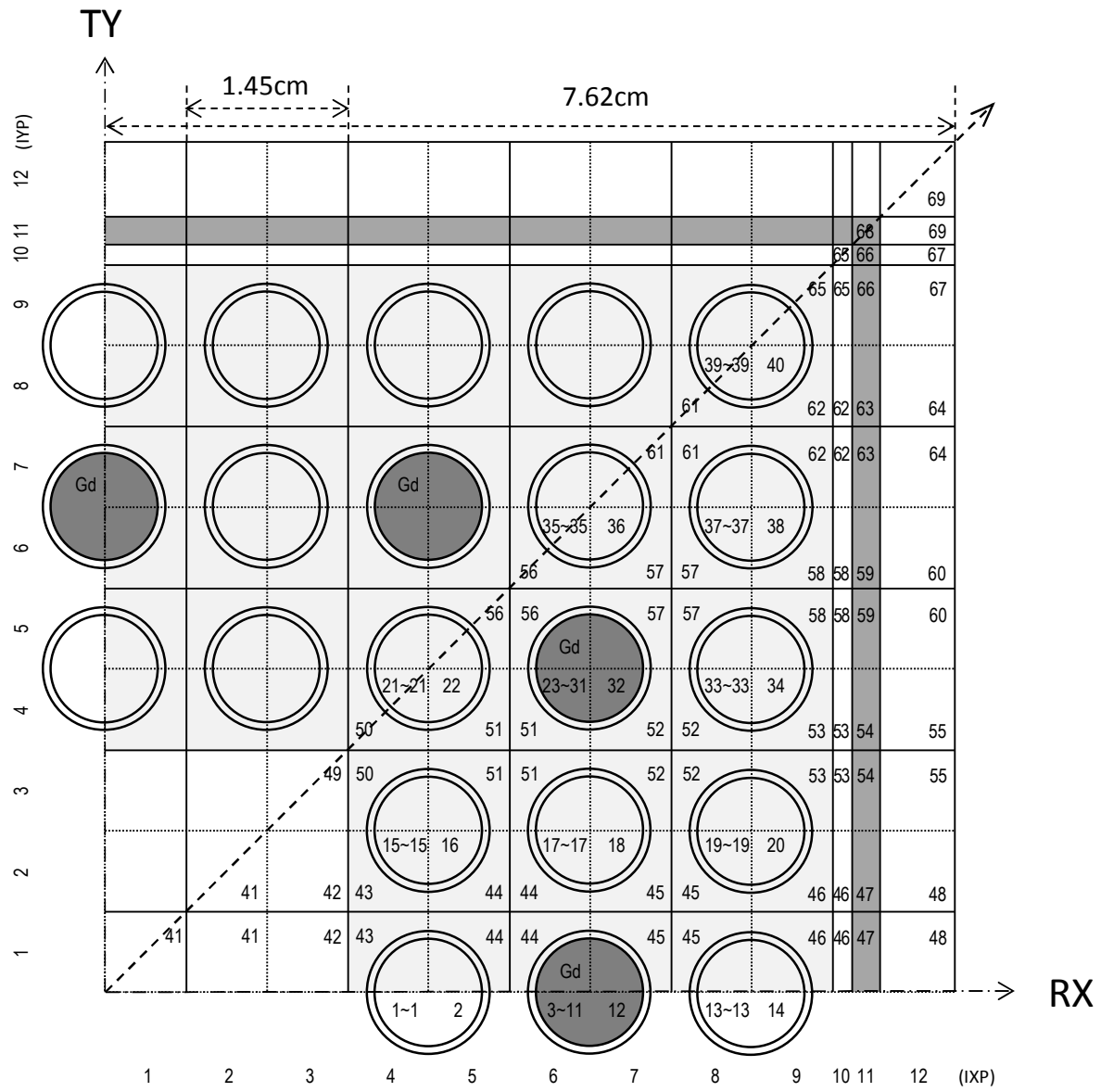


Fig.5.4-4 R-region map for a BWR fuel assembly (9x9 B, STEP-3)

An input data for this sample problem is shown below. [File name: BWR9x9.sh]

```
************************************************************ Start input
BWRA
Burnup Credit Criticality Benchmark Phase IIIC
Case 11 (Burn-up : 50 GWd/ton, Void Fraction : 40%)
 0 0 0 0 2  0 1 0 0 0    10(0)  / IOPT(20)
 0 0 0 0 0  0 0 0 0 0           / IPRN(10)
 0.0      / Buckling
************************************************************ PDS files
```

```
PFAST          / PDS NAME FOR PUBLIC FAST LIBRAY
PTHERMAL       / PDS NAME FOR PUBLIC THERMAL LIBRAY
PMCROSS        / PDS NAME FOR PUBLIC MCROSS LIBRAY
UMCROSS        / PDS NAME FOR USER'S MCROSS LIBRAY
MACROWK        / PDS NAME FOR USER'S FINE GROUP MACRO. X-SECTION
FLUX           / PDS NAME FOR FLUX  (OUTPUT)
MACRO          / PDS NAME FOR BROAD GROUP EFF. MACRO. X-SECTION (OUTPUT)
MICRO          / PDS NAME FOR FINE  GROUP EFF. MICRO. X-SECTION (OUTPUT)
MICROHM        / PDS NAME FOR BROAD GROUP EFF. HOMOG. MICRO. XS (OUTPUT)
               / PDS NAME FOR MACRO-IN IF NEED
               / PDS NAME FOR MICRO-IN IF NEED
************************************************************ Group structure
 200  118  0   0   / NGF NEF  NGB NGBF
*      118    200   / ICOLNG
************************************************************ PIJ input
16 354 138 69 1   1 12 12 21 0   2 0 25 91 10   1 90 0 0 0         / Pij Control
*
*** Block 3 T-S
 61  62  63  64  65  66  67  68  69  70  71  72  62  73  74  75  76  77
 78  79  80  81  82  83  63  74  84  85  86  87  88  89  90  91  92  93
 64  75  85  94  95  96  97  98  99 100 101 102  65  76  86  95 103 104
105 106 107 108 109 110  66  77  87  96 104 111 112 113 114 115 116 117
 67  78  88  97 105 112 118 119 120 121 122 123  68  79  89  98 106 113
119 124 125 126 127 128  69  80  90  99 107 114 120 125 129 130 131 132
 70  81  91 100 108 115 121 126 130 133 134 135  71  82  92 101 109 116
122 127 131 134 136 137  72  83  93 102 110 117 123 128 132 135 137 138
  1   1   1   2   2   2   3   3   3   4   5   6   7   8   9  10  11  12
 13  14  15  15  15  16  16  16  17  17  17  18  19  19  19  20  20  20
 21  21  21  22  23  23  23  24  24  24  25  25  25  26  27  27  27  28
 28  28  29  29  29  30   1   1   1   2   2   2   3   3   3   4  19  19
 19  20  20  20  21  21  21  22  31  31  31  32  32  32  33  33  33  34
 35  36  37  38  39  40  41  42  43  44  45  45  45  46  46  46  47  47
 47  48   5   6   7   8   9  10  11  12  13  14  23  23  23  24  24  24
 25  25  25  26  35  36  37  38  39  40  41  42  43  44  49  49  49  50
 50  50  51  51  51  52  53  53  53  54  54  54  55  55  55  56  15  15
 15  16  16  16  17  17  17  18  27  27  27  28  28  28  29  29  29  30
 45  45  45  46  46  46  47  47  47  48  53  53  53  54  54  54  55  55
 55  56  57  57  57  58  58  58  59  59  59  60 /
*** Block 4 R-T
  1   1   1   2   3   4   5   6   7   8   9  10  11  12  13  13  13  14
 15  15  15  16  17  17  17  18  19  19  19  20  21  21  21  22  23  24
 25  26  27  28  29  30  31  32  33  33  33  34  35  35  35  36  37  37
 37  38  39  39  39  40  41  41  42  43  44  44  45  45  46  46  47  48
 41  42  43  44  44  45  45  46  46  47  48  49  50  51  51  52  52  53
 53  54  55  50  51  51  52  52  53  53  54  55  56  56  57  57  58  58
 59  60  56  57  57  58  58  59  60  61  61  62  62  63  64  61  62  62
 63  64  65  65  66  67  65  66  67  68  69  69 /
*** Block 5 X-R
69(1) /
*** Block 6 M-R
 6 29 20 21 22 23 24 25 26 27 28 29  3 29  5 29  4 29  2 29  1 29 11 12
13 14 15 16 17 18 19 29  7 29  8 29  9 29 10 29 31 33 32 32 32 32 30 31
34 32 32 32 32 30 31 32 32 32 30 31 32 32 30 31 32 30 31 30 31 /
*** Block 8 RX
0.000  0.725  1.450  2.001  2.900    3.625  4.350  5.075  5.800  6.525
6.700  6.950  7.620  /
*** Block 9' TY
0.000  0.725  1.450  2.001  2.900    3.625  4.350  5.075  5.800  6.525
6.700  6.950  7.620  /
*** Block 10'' IXP & Block 11' IYP
5 7 9 5 7 9 1 3 5 7 9 1 3 5 7 9 1 3 5 7 9 /
1 1 1 3 3 3 5 5 5 5 5 7 7 7 7 7 9 9 9 9 9 /
*** Block 12' RDP
21 (0.00000  0.16000  0.22627  0.27713  0.32000
    0.35777  0.39192  0.42332  0.45255  0.48000  0.55000)
************************************************************ Materials
34 / NMAT
FL110INI   1  4   900.0 0.960    0.0  /  1, Rod type: 1, Rod ID:10
```

```
U02340     0  0   8.5649E-6        / 1
U02350     0  0   1.1221E-3        / 2
U02380     0  0   2.1495E-2        / 3
O00160     0  0   4.5252E-2        / 4
FL204INI   1  4   900.0 0.960    0.0  /  2, Rod type: 2, Rod ID: 4
U02340     0  0   7.6910E-6        / 1
U02350     0  0   1.0076E-3        / 2
U02380     0  0   2.1609E-2        / 3
O00160     0  0   4.5249E-2        / 4
FL205INI   1  4   900.0 0.960    0.0  /  3, Rod type: 2, Rod ID: 5
U02340     0  0   7.6910E-6        / 1
U02350     0  0   1.0076E-3        / 2
U02380     0  0   2.1609E-2        / 3
O00160     0  0   4.5249E-2        / 4
FL208INI   1  4   900.0 0.960    0.0  /  4, Rod type: 2, Rod ID: 8
U02340     0  0   7.6910E-6        / 1
U02350     0  0   1.0076E-3        / 2
U02380     0  0   2.1609E-2        / 3
O00160     0  0   4.5249E-2        / 4
FL211INI   1  4   900.0 0.960    0.0  /  5, Rod type: 2, Rod ID:11
U02340     0  0   7.6910E-6        / 1
U02350     0  0   1.0076E-3        / 2
U02380     0  0   2.1609E-2        / 3
O00160     0  0   4.5249E-2        / 4
FL212INI   1  4   900.0 0.960    0.0  /  6, Rod type: 2, Rod ID:12
U02340     0  0   7.6910E-6        / 1
U02350     0  0   1.0076E-3        / 2
U02380     0  0   2.1609E-2        / 3
O00160     0  0   4.5249E-2        / 4
FL303INI   1  4   900.0 0.960    0.0  /  7, Rod type: 3, Rod ID: 3
U02340     0  0   6.8170E-6        / 1
U02350     0  0   8.9315E-4        / 2
U02380     0  0   2.1723E-2        / 3
O00160     0  0   4.5247E-2        / 4
FL306INI   1  4   900.0 0.960    0.0  /  8, Rod type: 3, Rod ID: 6
U02340     0  0   6.8170E-6        / 1
U02350     0  0   8.9315E-4        / 2
U02380     0  0   2.1723E-2        / 3
O00160     0  0   4.5247E-2        / 4
FL402INI   1  4   900.0 0.960    0.0  /  9, Rod type: 4, Rod ID: 2
U02340     0  0   5.9431E-6        / 1
U02350     0  0   7.7865E-4        / 2
U02380     0  0   2.1838E-2        / 3
O00160     0  0   4.5244E-2        / 4
FL501INI   1  4   900.0 0.960    0.0  / 10, Rod type: 5, Rod ID: 1
U02340     0  0   3.6708E-6        / 1
U02350     0  0   4.8094E-4        / 2
U02380     0  0   2.2134E-2        / 3
O00160     0  0   4.5238E-2        / 4
GD071INI   2 10   900.0 0.960    0.0  / 11, Rod type: G, Rod ID: 7, R-Pos.:1
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
GD1600     0  0   3.6481E-4        /10
GD072INI   2 10   900.0 0.960    0.0  / 12, Rod type: G, Rod ID: 7, R-Pos.:2
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
```

```
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
GD1600     0  0   3.6481E-4        /10
GD073INI   2 10   900.0 0.960    0.0  / 13, Rod type: G, Rod ID: 7, R-Pos.:3
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
GD1600     0  0   3.6481E-4        /10
GD074INI   2 10   900.0 0.960    0.0  / 14, Rod type: G, Rod ID: 7, R-Pos.:4
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
GD1600     0  0   3.6481E-4        /10
GD075INI   2 10   900.0 0.960    0.0  / 15, Rod type: G, Rod ID: 7, R-Pos.:5
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
GD1600     0  0   3.6481E-4        /10
GD076INI   2 10   900.0 0.960    0.0  / 16, Rod type: G, Rod ID: 7, R-Pos.:6
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
GD1600     0  0   3.6481E-4        /10
GD077INI   2 10   900.0 0.960    0.0  / 17, Rod type: G, Rod ID: 7, R-Pos.:7
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
GD1600     0  0   3.6481E-4        /10
GD078INI   2 10   900.0 0.960    0.0  / 18, Rod type: G, Rod ID: 7, R-Pos.:8
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
```

```
GD1600     0  0   3.6481E-4       /10
GD079INI   2 10   900.0 0.960    0.0  / 19, Rod type: G, Rod ID: 7, R-Pos.:9
U02340     0  0   5.5429E-6       / 1
U02350     0  0   7.2622E-4       / 2
U02380     0  0   2.0367E-2       / 3
O00160     0  0   4.4678E-2       / 4
GD1540     0  0   3.6051E-5       / 5
GD1550     0  0   2.4475E-4       / 6
GD1560     0  0   3.3852E-4       / 7
GD1570     0  0   2.5881E-4       / 8
GD1580     0  0   4.1079E-4       / 9
GD1600     0  0   3.6481E-4       /10
GD091INI   2 10   900.0 0.960    0.0  / 20, Rod type: G, Rod ID: 9, R-Pos.:1
U02340     0  0   5.5429E-6       / 1
U02350     0  0   7.2622E-4       / 2
U02380     0  0   2.0367E-2       / 3
O00160     0  0   4.4678E-2       / 4
GD1540     0  0   3.6051E-5       / 5
GD1550     0  0   2.4475E-4       / 6
GD1560     0  0   3.3852E-4       / 7
GD1570     0  0   2.5881E-4       / 8
GD1580     0  0   4.1079E-4       / 9
GD1600     0  0   3.6481E-4       /10
GD092INI   2 10   900.0 0.960    0.0  / 21, Rod type: G, Rod ID: 9, R-Pos.:2
U02340     0  0   5.5429E-6       / 1
U02350     0  0   7.2622E-4       / 2
U02380     0  0   2.0367E-2       / 3
O00160     0  0   4.4678E-2       / 4
GD1540     0  0   3.6051E-5       / 5
GD1550     0  0   2.4475E-4       / 6
GD1560     0  0   3.3852E-4       / 7
GD1570     0  0   2.5881E-4       / 8
GD1580     0  0   4.1079E-4       / 9
GD1600     0  0   3.6481E-4       /10
GD093INI   2 10   900.0 0.960    0.0  / 22, Rod type: G, Rod ID: 9, R-Pos.:3
U02340     0  0   5.5429E-6       / 1
U02350     0  0   7.2622E-4       / 2
U02380     0  0   2.0367E-2       / 3
O00160     0  0   4.4678E-2       / 4
GD1540     0  0   3.6051E-5       / 5
GD1550     0  0   2.4475E-4       / 6
GD1560     0  0   3.3852E-4       / 7
GD1570     0  0   2.5881E-4       / 8
GD1580     0  0   4.1079E-4       / 9
GD1600     0  0   3.6481E-4       /10
GD094INI   2 10   900.0 0.960    0.0  / 23, Rod type: G, Rod ID: 9, R-Pos.:4
U02340     0  0   5.5429E-6       / 1
U02350     0  0   7.2622E-4       / 2
U02380     0  0   2.0367E-2       / 3
O00160     0  0   4.4678E-2       / 4
GD1540     0  0   3.6051E-5       / 5
GD1550     0  0   2.4475E-4       / 6
GD1560     0  0   3.3852E-4       / 7
GD1570     0  0   2.5881E-4       / 8
GD1580     0  0   4.1079E-4       / 9
GD1600     0  0   3.6481E-4       /10
GD095INI   2 10   900.0 0.960    0.0  / 24, Rod type: G, Rod ID: 9, R-Pos.:5
U02340     0  0   5.5429E-6       / 1
U02350     0  0   7.2622E-4       / 2
U02380     0  0   2.0367E-2       / 3
O00160     0  0   4.4678E-2       / 4
GD1540     0  0   3.6051E-5       / 5
GD1550     0  0   2.4475E-4       / 6
GD1560     0  0   3.3852E-4       / 7
GD1570     0  0   2.5881E-4       / 8
GD1580     0  0   4.1079E-4       / 9
GD1600     0  0   3.6481E-4       /10
GD096INI   2 10   900.0 0.960    0.0  / 25, Rod type: G, Rod ID: 9, R-Pos.:6
```

```
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
GD1600     0  0   3.6481E-4        /10
GD097INI   2 10   900.0 0.960    0.0  / 26, Rod type: G, Rod ID: 9, R-Pos.:7
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
GD1600     0  0   3.6481E-4        /10
GD098INI   2 10   900.0 0.960    0.0  / 27, Rod type: G, Rod ID: 9, R-Pos.:8
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
GD1600     0  0   3.6481E-4        /10
GD099INI   2 10   900.0 0.960    0.0  / 28, Rod type: G, Rod ID: 9, R-Pos.:9
U02340     0  0   5.5429E-6        / 1
U02350     0  0   7.2622E-4        / 2
U02380     0  0   2.0367E-2        / 3
O00160     0  0   4.4678E-2        / 4
GD1540     0  0   3.6051E-5        / 5
GD1550     0  0   2.4475E-4        / 6
GD1560     0  0   3.3852E-4        / 7
GD1570     0  0   2.5881E-4        / 8
GD1580     0  0   4.1079E-4        / 9
GD1600     0  0   3.6481E-4        /10
ZRYCLINI   0 28   559.0 0.0    0.0  / 29, Zircalloy-2 for Cladding
SN1120     0  0   4.8303E-6
SN1140     0  0   3.2866E-6
SN1150     0  0   1.6931E-6
SN1160     0  0   7.2405E-5
SN1170     0  0   3.8244E-5
SN1180     0  0   1.2061E-4
SN1190     0  0   4.2776E-5
SN1200     0  0   1.6224E-4
SN1220     0  0   2.3056E-5
SN1240     0  0   2.8832E-5
FE0540     0  0   5.3647E-6
FE0560     0  0   8.4214E-5
FE0570     0  0   1.9449E-6
FE0580     0  0   2.5883E-7
CR0500     0  0   3.2962E-6
CR0520     0  0   6.3563E-5
CR0530     0  0   7.2076E-6
CR0540     0  0   1.7941E-6
NI0580     0  0   2.7445E-5
NI0600     0  0   1.0572E-5
NI0610     0  0   4.5958E-7
NI0620     0  0   1.4650E-6
NI0640     0  0   3.7331E-7
ZR0900     0  0   2.1848E-2
```

```
ZR0910     0  0   4.7646E-3
ZR0920     0  0   7.2827E-3
ZR0940     0  0   7.3804E-3
ZR0960     0  0   1.1890E-3
ZRYCBINI   0 28   559.0 0.0    0.0  / 30, Zircalloy-2 for Channel Box
SN1120     0  0   4.8303E-6
SN1140     0  0   3.2866E-6
SN1150     0  0   1.6931E-6
SN1160     0  0   7.2405E-5
SN1170     0  0   3.8244E-5
SN1180     0  0   1.2061E-4
SN1190     0  0   4.2776E-5
SN1200     0  0   1.6224E-4
SN1220     0  0   2.3056E-5
SN1240     0  0   2.8832E-5
FE0540     0  0   5.3647E-6
FE0560     0  0   8.4214E-5
FE0570     0  0   1.9449E-6
FE0580     0  0   2.5883E-7
CR0500     0  0   3.2962E-6
CR0520     0  0   6.3563E-5
CR0530     0  0   7.2076E-6
CR0540     0  0   1.7941E-6
NI0580     0  0   2.7445E-5
NI0600     0  0   1.0572E-5
NI0610     0  0   4.5958E-7
NI0620     0  0   1.4650E-6
NI0640     0  0   3.7331E-7
ZR0900     0  0   2.1848E-2
ZR0910     0  0   4.7646E-3
ZR0920     0  0   7.2827E-3
ZR0940     0  0   7.3804E-3
ZR0960     0  0   1.1890E-3
WTRV0INI   0  2   559.0 1.000    0.0  / 31, Water at 0% of Void fraction
H0001H     0  0   4.9316E-2       / 1
O00160     0  0   2.4658E-2       / 2
WTRVXINI   0  2   559.0 1.000    0.0  / 32, Water at X% of Void fraction (X:0,40,70%)
H0001H     0  0   3.0588E-2       / 1
O00160     0  0   1.5294E-2       / 2
WTRCSINI   0 30   559.0 1.000    0.0  / 33, Smeared with Water & Zircalloy at Side of Water Channel
H0001H     0  0   3.7756E-2       / 1
O00160     0  0   2.0010E-2       / 2
SN1120     0  0   6.5748E-7
SN1140     0  0   4.4736E-7
SN1150     0  0   2.3046E-7
SN1160     0  0   9.8555E-6
SN1170     0  0   5.2056E-6
SN1180     0  0   1.6417E-5
SN1190     0  0   5.8225E-6
SN1200     0  0   2.2083E-5
SN1220     0  0   3.1383E-6
SN1240     0  0   3.9246E-6
FE0540     0  0   7.3022E-7
FE0560     0  0   1.1463E-5
FE0570     0  0   2.6473E-7
FE0580     0  0   3.5230E-8
CR0500     0  0   4.4866E-7
CR0520     0  0   8.6520E-6
CR0530     0  0   9.8106E-7
CR0540     0  0   2.4421E-7
NI0580     0  0   3.7356E-6
NI0600     0  0   1.4390E-6
NI0610     0  0   6.2556E-8
NI0620     0  0   1.9941E-7
NI0640     0  0   5.0813E-8
ZR0900     0  0   2.9739E-3
ZR0910     0  0   6.4854E-4
ZR0920     0  0   9.9130E-4
```

```
ZR0940     0  0   1.0046E-3
ZR0960     0  0   1.6184E-4
WTRCCINI   0 30   559.0 1.000    0.0  / 34, Smeared with Water & Zircalloy at Corner of Water Channel
H0001H     0  0   3.2470E-2        / 1
O00160     0  0   1.6235E-2        / 2
SN1120     0  0   1.0441E-6
SN1140     0  0   7.1042E-7
SN1150     0  0   3.6597E-7
SN1160     0  0   1.5651E-5
SN1170     0  0   8.2667E-6
SN1180     0  0   2.6070E-5
SN1190     0  0   9.2462E-6
SN1200     0  0   3.5069E-5
SN1220     0  0   4.9837E-6
SN1240     0  0   6.2323E-6
FE0540     0  0   1.1596E-6
FE0560     0  0   1.8203E-5
FE0570     0  0   4.2039E-7
FE0580     0  0   5.5946E-8
CR0500     0  0   7.1248E-7
CR0520     0  0   1.3740E-5
CR0530     0  0   1.5580E-6
CR0540     0  0   3.8781E-7
NI0580     0  0   5.9323E-6
NI0600     0  0   2.2851E-6
NI0610     0  0   9.9341E-8
NI0620     0  0   3.1667E-7
NI0640     0  0   8.0692E-8
ZR0900     0  0   4.7226E-3
ZR0910     0  0   1.0299E-3
ZR0920     0  0   1.5742E-3
ZR0940     0  0   1.5953E-3
ZR0960     0  0   2.5701E-4
************************************************************ Cell Burnup
  92 1 1 0 0  0 1 0 0 1  0 1 8(0)   / Control Data for Burnup Calculation
  88(2.9132E-03) 4(0)
1.000E+2 2.000E+2 3.500E+2 5.000E+2 7.500E+2
1.000E+3 1.250E+3 1.500E+3 1.750E+3
2.000E+3 2.250E+3 2.500E+3 2.750E+3
3.000E+3 3.250E+3 3.500E+3 3.750E+3
4.000E+3 4.250E+3 4.500E+3 4.750E+3
5.000E+3 5.250E+3 5.500E+3 5.750E+3
6.000E+3 6.250E+3 6.500E+3 6.750E+3
7.000E+3 7.250E+3 7.500E+3 7.750E+3
8.000E+3 8.250E+3 8.500E+3 8.750E+3
9.000E+3 9.250E+3 9.500E+3 9.750E+3
1.000E+4 1.025E+4 1.050E+4 1.075E+4
1.100E+4 1.125E+4 1.150E+4 1.175E+4
1.200E+4 1.225E+4 1.250E+4 1.275E+4
1.300E+4 1.325E+4 1.350E+4 1.375E+4
1.400E+4 1.425E+4 1.450E+4 1.475E+4
1.500E+4 1.525E+4 1.550E+4 1.575E+4
1.600E+4 1.625E+4 1.650E+4 1.675E+4
1.700E+4 1.725E+4 1.750E+4 1.775E+4
1.800E+4 1.825E+4 1.850E+4 1.875E+4
1.900E+4 1.925E+4 1.950E+4 1.975E+4
2.000E+4 2.500E+4 3.000E+4 3.500E+4 4.000E+4 4.500E+4 5.000E+4 & Exp (MWd/t)
4(-1826.2)            /     Cooling time (in Day; = 5 years)
  28(1)   6(0)            /  Specify Burnable Material Region (required when IBC10 != 0) ---- Block-09
    0   87(1)   4(0)      /  Specify Steps with PC method     (required when IBC12 != 0) ---- Block-12
************************************************************ PEACO
   0          / IPLOT for PEACO
              / END OF ALL CASE
************************************************************ End input
```

6. Contents of PDS Files

In this chapter, we describe contents and structures of major PDS files to which user may access. As mentioned in Sect.1.3, all data in a member file are stored in the following binary record.

LENG, (WORK(*i*), *i*=1, LENG),

where LENG is total data length of the one-dimensional array WORK.

6.1 FLUX

The neutron fluxes integrated spatially in R-region and those integrated in X-region are stored in FLUX file. For plotting and homogenization purpose, the volumes of each spatial region are also written. The following nine kinds of members are treated in this file.

Member name	Contents
CONT000*e*	energy group structure of fine-group or collapsed-group identified by *e*-tag
*case*0*bbe*	neutron fluxes by R-region by group
casexbbe	neutron fluxes of X-region by group
*case*XVOL	volumes of X-regions
*case*RVOL	volumes of R-regions
*case*TVOL	volumes of T-regions
*case*FIXS	boundary surface source (fed by a user)
WTFLUX	weighting spectrum to collapse the cross-sections of isolated materials (cf. IOPT9 in Sect.2.2). (fed by a user)
CHIFLUX	weighting spectrum to convert matrix-form fission spectrum to vector-form one (fed by a user)

case	Case identification (CASEID in Sect.2.2).
e	*e*=F for fine-group, *e*=C for course-group
bb	Burn-up step indicator (00, 01, 02,…….., 99). For non-depleting problem, '00' is given.
x	X-region number (1,2,……..,9,A,B,….,Z) is given.

Member **CONT000*e*** /2*(NG + 1)/

NG Number of energy groups (depending on *e*-tag)

(W(*g*), *g*=1,NG)

Lethargy width (asymptotic spectrum for collapsing)

(E(*g*), *g*=1,NG+1)

Energy boundaries (eV) starting at the highest energy

Member ***case0bbe*** /NRR*NG/

(PHI(*i*,*g*), *i*=1,NRR), *g*=1,NG)

Energy integrated neutron flux multiplied by volume of region *i* and lethargy width of group *g* (n/cm^2/s×cm^3), NRR is the number of R-regions.

Member ***casexbbe*** /NG/

(PHIX(*g*), *g*=1,NG)

Energy integrated neutron flux multiplied by volume of X-region *x* (n/cm^2/s×cm^3)

Member ***case*XVOL** /NX/

(VOLX(*i*), *i*=1,NX)

Volume of X-region *i* (cm^3)

Member ***case*RVOL** /NRR/

(VOLR(*i*), *i*=1,NRR)

Volume of R-region *i* (cm^3)

Member ***case*TVOL** /NR/

(VOLT(*i*), *i*=1,NR)

Volume of T-region *i* (cm^3)

Member ***case*FIXS** /ng (fine-group)/

The spectrum stored in this member will be used as the in-current flux at the outer boundary of the lattice. It is used in the eigenvalue problem mode specified by IOPT1=1 (as whole energy calculation) and IOPT23=2 in Sect.2.2. The member has to be prepared before the execution. Data structure is identical with that of the member `casexbbe`.

Member **WTFLUX** /NG (fine-group)/

The spectrum stored in this member will be used as the weight to collapse the fine-group cross-sections into the collapsed-group constants of isolated materials like a reflector material not allocated to the R-region of PIJ. The member has to be prepared before the execution. In the usual case, i.e. when this member is not given, the asymptotic spectrum stored in the Public Library is used as the weighting spectrum. The user can replace the spectrum by this member.

Member **CHIFLUX** /NG (fine-group)/

The spectrum stored in this member will be used to generate a problem-dependent fission spectrum in vector-form $\chi(g)$ from the general fission spectrum in matrix-form $\chi(g, g')$ in the Public

Library. The member has to be prepared before the execution. In the usual case, i.e. when this member is not given, the asymptotic spectrum stored in the Public Library is used to generate the vector-form fission spectrum. The user can replace the spectrum by this member.

6.2 MACRO

Homogenized and collapsed macroscopic cross-sections are stored in the MACRO file. In the case of burn-up calculation, information such as data printed on the 98-th device is also stored in members in the MACRO file.

Member name	Contents
CONT0000	energy group structure
casexbbp	the *p*-th order macroscopic cross-section of X-region
*casexbb*D	delayed neutron data of X-region which includes fuel materials
*case*BNUP	material-wise burn-up calculation results for all steps (IOPT7=1)
*case*HTbb	material-wise burn-up calculation results at the step indicated by *bb*-tag (IOPT7=1)
*case*xDNT	homogenized burn-up calculation results for all steps (IOPT7=1)

case	Case identification (CASEID in Sect.2.2).
x	X-region number (1,2,……..,9,A,B,….,Z) is given.
bb	Two digit number corresponding to burn-up steps (*bb*=00,01,02,03….). For non-depleting problem, '00' is given.
p	Legendre component indicator (the *p*-th component indicated by *p*-tag (*p* =0,1)

Member **CONT0000**

NG Number of energy groups

(W(*g*),*g*=1,NG)

Lethargy width (asymptotic spectrum for collapsing)

(E(*g*),*g*=1,NG+1)

Energy boundaries (eV) starting at the highest energy

Member ***casexbbp***

This member contains the cross-section data shown in Table 6.2-1.

Table 6.2-1 Macroscopic cross-sections stored in a member file *casexbbp* (1/2)

m	Cross-section type
1	Scattering matrix $\Sigma_s(g \rightarrow g')$
2	Absorption $\Sigma_a(g)$
3	Production $\nu\Sigma_f(g)$
4	Fission $\Sigma_f(g)$
5	Fission spectrum $\chi(g)$
6	Total cross-section $\Sigma_t(g)$
7	Transport cross-section $\Sigma_{tr}(g)$
8	Diffusion coefficient D1 defined by IOPT22 in Sect.2.2
9	Diffusion coefficient D2 defined by IOPT22 in Sect.2.2
10	Diffusion coefficient D3 defined by IOPT22 in Sect.2.2
11	Inverse of velocity $1/\upsilon(g)=\sqrt{\frac{m}{2\bar{E}(g)}}$ [s/cm]
12	Total (n,2n) $=\sum_{g'}\Sigma_{n2n,}(g \rightarrow g')/2$
13	P0 component of elastic scattering $\Sigma_e^{P0}(g)=\sum_{g'}\Sigma_e^{P0}(g \rightarrow g')$
14	P1 component of elastic scattering $\Sigma_e^{P1}(g)=\sum_{g'}\Sigma_e^{P1}(g \rightarrow g')$ Note: (2L+1) is not multiplied
15	Total inelastic scattering $\Sigma_{in}(g)=\sum_{g'}\Sigma_{in}(g \rightarrow g')$
16	Total (n,3n)$=\sum_{g'}\Sigma_{n3n,}(g \rightarrow g')/3$
17	(n,γ) cross-section $\Sigma_{n\gamma}(g)$ (defined by MT=102 in the ENDF-6 format[33])
18	(n,p) cross-section $\Sigma_{np}(g)$ (MT=103)
19	(n,α) cross-section $\Sigma_{n\alpha}(g)$ (MT=107)
20	(n,2n) cross-section $\Sigma_{n2n}(g)$ (MT=16)

Table 6.2-1 Macroscopic cross-sections stored in a member file *casexbbp* (2/2)

21	(n,3n) cross-section $\Sigma_{n3n}(g)$ (MT=17)
22-25	Not used (or arbitrary cross-sections defined and added by user)

The above cross-sections are stored in the one-dimensional array in accordance with the following structure, which consists of a header part (H1∼H6) and a cross-section data part (m=1∼25). The value in [ ] indicates the starting address in the one-dimensional array, and that in { } indicates the data length.

H1 NG Number of energy groups [1], {1}

H2 NOMT Number of cross-section types (NOMT=25 always) [2], {1}

H3 (IPMT(m), m=1, NOMT) [3], {NOMT}

Starting address of the m-th cross-section data in the one-dimensional array of a member

H4 (NGMT(m), m=1, NOMT) [NOMT+3], {NOMT}

The group number of the last group in which a non-zero value is given to the m-th cross-section type.

Note: To reduce the stored data of a member, the last successive zero values are not stored in the member. (e.g. fission spectrum, scattering matrix, cross-sections with high threshold energy, etc.) However, for the cross-section type (m) whose cross-section values are all zero (e.g. fission cross-section of a non-fissionable nuclide) or for the cross-section type which is not defined (e.g. m=22-25), a zero value is given only to the first group for convenience of data reading.

H5 (LS(g), g=1, NGMT(1)) [NOMT×2+3], {NGMT(1)}

The data storage of the scattering matrix of the g-th energy group $\Sigma_s(g \to g')$ starts from the g'=LS(g)-th group for the compact storage.

H6 (LE(g), g=1, NGMT(1)) [2×NOMT+NGMT(1)+3], {NGMT(1)}

The data storage of the scattering matrix of the g-th energy group $\Sigma_s(g \to g')$ ends at the g'=LE(g)-th group for the compact storage.

1 ((SIGS(g', g), g'=LS(g), LE(g)), g=1, NGMT(1))

[IPMT(1)≡2×(NOMT+NGMT(1))+3], {LN(1)≡ $\sum_g$[LE(g) − LS(g) + 1]}

Scattering matrix, $\Sigma_s(g \to g')$

2 (XS(g, 2), g=1, NGMT(2)) [IPMT(2)=IPMT(1)+LN(1)], {LN(2) ≡NGMT(2)}

The 2nd type of cross-section data, $\Sigma_a(g)$

:

m (XS(g, m), g=1, NGMT(m)) [IPMT(m)=IPMT(m-1)+LN(m-1)], {LN(m) ≡NGMT(m)}

The m-th type of cross-section data (cf. Table 6.2-1)

:

25 (XS(g, 25), g=1, NGMT(25)) [IPMT(25)=IPMT(24)+LN(24)], {LN(25) ≡NGMT(25)}

The 25-th type of cross-section data (Dummy data for future use)

Note: A zero value is given only to the 1st group.

Member **casexbbD**

The X-region averaged delayed neutron data to be used for the calculation of kinetics parameters (e.g. the effective delayed neutron fraction β_{eff}, etc) in core calculations are stored in the one-dimensional array in accordance with the following structure, which consists of a header part (H1~H5) and a delayed neutron data part (m=1~3). The value in [ ] indicates the top address in the one-dimensional array, and that in { } indicates the data length.

H1 NG Number of energy groups [1], {1}

H2 NOMTD Number of delayed neutron data types (NOMTD=3 always) [2], {1}

H3 NFAM Number of delayed neutron families (NFAM=6 or 15 if D_2O is concerned) [3], {1}

H4 (IPMT(m), m=1, NOMTD) [3], {NOMTD}

Starting address of the m-th delayed neutron data in the one-dimensional array of a member

H5 (NGMT(m), m=1, NOMTD) [NOMTD+3], {NOMTD}

The group number of the last group in which a non-zero value is given to the m-th delayed neutron data type

1 (XD(j,g), j=1,NFAM), g=1, NGMT(1))

[IPMT(1)≡2×NOMTD+4], {LN(1) ≡NFAM×NGMT(1)}

Delayed neutron spectrum by family j; $\chi_d(j,g)$

2 (BVFMAC(j,g), j=1,NFAM), g=1, NGMT(2))

[IPMT(2)≡IPMT(1)+LN(1)], {LN(2) ≡NFAM×NGMT(2)}

The parameter for delayed neutron fraction by family j

BVFMAC(g, j)≡ $\sum_n(\alpha_j^n \nu_{d,g}^n \sigma_{f,g}^n N^n)$,

where, n: nuclides, α:delayed neutron fraction normalized to unity.

3 (BVLMAC(j,g), j=1,NFAM), g=1, NGMT(3))

[IPMT(3)≡IPMT(2)+LN(2)], {LN(3) ≡NFAM×NGMT(3)}

The parameter for decay constant by family j

BVLMAC$(g,j)\equiv \sum_n(\alpha_j^n \nu_{d,g}^n \sigma_{f,g}^n N^n/\lambda_j^n)$

where, λ: decay constant of delayed neutron precursor.

Member **`caseBNUP`** Material-wise burn-up information

This member is generated at the time when the lattice burn-up calculation is finished. In this member material-wise burn-up information printed on the 98-th device are stored in the one-dimensional array with the following structure.

NOWSTP Number of burn-up steps including the initial step
If the burn-up calculation is completed, NOWSTP=1+IBC1 (Sect.2.5).

NTNUC Total number of depleting nuclides

NTDEPZ Number of depleting materials in a lattice

ST235 Initial atomic numbers (atoms) of the nuclide which is used for the measure of depletion fraction.

TWTHVY Initial inventory (ton) of all heavy nuclides in whole of a lattice with unit height (1cm)

IDUM6 = 0 not used

IDUM7 = 0 not used

IDUM8 = 0 not used

IDUM9 = 0 not used

CASEID Case identification (A4) (cf. Block-1 of Sect.2.2)

STDNUC Name (2A4) of the nuclide used for the measure of depletion fraction (cf. Block-4 of Sect.2.5, 'U02350__' is defaulted when IBC2>0 (cf. Block-1 of Sect.2.5)

(TITLE(i), i=1,18)
Comment of 72 characters (18A4)

(MTNM(m), m=1, NTDEPZ) (A4)
The first four characters of the depleting material name (A8) specified in Sect.2.5. NTDEPZ is the number of depleting materials (by M-region)

(MTYP(m), m=1, NTDEPZ)
Type of depleting materials
= 1 Fissionable material (e.g. UO_2)
= 2 Non-fissionable material (e.g. Gd_2O_3)

(VOLDPZ(*m*), *m*=1, NTDEPZ)

Volume of each depleting material (cm^3)

(NUCLID(*i*), *i*=1, NTNUC) (2A4)

Nuclide name by eight characters (cf. Sect.8.1). The last two characters are filled with two blanks. (e.g. U02350__, AM2421__).

(DAYS(*j*), *j*=1, NOWSTP)

Accumulated burn-up days (unit day) by step *j*

The step number *j*=1,2,3,4,…,100 corresponds to the burn-up tag of the member in PDS *bb*= 00,01,02,03,…,99, respectively.

(EXPST(*j*), *j*=1, NOWSTP)

Accumulated burn-up (MWd/t) of whole lattice

(U235F(*j*), *j*=1, NOWSTP)

Depletion fraction (%) of nuclide STDNUC (cf. Block-4 of Sect.2.5)

U-235 is defaulted when IBC2>0 (cf. Block-1 of Sect.2.5)

U235F(*j*) ≤ 100 (Negative values may appear for the nuclide with increasing burn-up, e.g. blanket fuel.)

(AKEFF(*j*), *j*=1, NOWSTP)

Effective multiplication factor by step. AKEFF(NOWSTP)=0.0, because the flux (eigenvalue) calculation is not done at the last step.

(AKINF(*j*), *j*=1, NOWSTP)

Infinite multiplication factor by step. AKINF(NOWSTP)=0.0

(INSCR(*j*), *j*=1, NOWSTP)

Instantaneous conversion ratio by step. INSCR(NOWSTP)=0.0

(INTCR(*j*), *j*=1, NOWSTP)

Integrated conversion ratio by step. INTCR(NOWSTP)=0.0

(POWERL(*j*), *j*=1, NOWSTP)

Thermal power per unit length of lattice (MWt/cm) by step.

POWERL (NOWSTP)=0.0

In the constant flux calculation, POWERL(*j* ≤ 1) are calculated by the code.

(FLXNRM(*j*), *j*=1, NOWSTP)

Normalizing factor of neutron flux by step

FLXNRM(NOWSTP)=0.0

Even in the restart calculation under constant flux level, the flux level is set to be FLXNRM(1) throughout the exposure.

((POWRZN(*j*,*m*), *j*=1, NOWSTP), *m*=1, NTDEPZ)

Power density (MW/cm^3) by step *j* by depleting material *m*

((EXPSZN(*j*,*m*), *j*=1, NPWSTP), *m*=1, NTDEPZ)

Burn-up (MWd/t) by step by depleting material

In the case of MTYPX(*m*)=2, time integrated absorption rate (n/cm^3) is given.

((HMINV(*j*,*m*), *j*=1, NOWSTP), *m*=1, NTDEPZ)

Heavy metal inventory (ton/cm^3) by step by depleting material

Initial inventory in the restart calculation is given by

$$\sum_{m} \{\mathrm{HMINV}(1,m)*\mathrm{VOLDPZ}(m)\}$$

((GAMAV(*j*,*m*), *j*=1, NOWSTP), *m*=1, NTDEPZ)

Heat generation (Joule/fission) by step by depleting material

((YDXE(*j*,*m*), *j*=1, NOWSTP), *m*=1, NTDEPZ)

Fission yield of Xe-135 by step by depleting material

((YDIO(*j*,*m*), *j*=1, NOWSTP), *m*=1, NTDEPZ)

Fission yield of I-135 by step by depleting material

((YDSM(*j*,*m*), *j*=1, NOWSTP), *m*=1, NTDEPZ)

Fission yield of Sm-149 by step by depleting material

((YDPM(*j*,*m*), *j*=1, NOWSTP), *m*=1, NTDEPZ)

Fission yield of Pm-149 by step by depleting material

(((DNSITY(*j*,*i*,*m*), *j*=1, NOWSTP), *i*=1, NTNUC), *m*=1, NTDEPZ)

Atomic number density (10^{24} n/cm^3) by step by nuclide by depleting material

(((SIGXE(*g*,*j*,*m*), *g*=1, NG), *j*=1, NOWSTP), *m*=1, NTDEPZ)

Microscopic capture cross-section of Xe-135 by collapsed-group *g* by step *j* by depleting material *m*

If no collapsing is done, the fine-group cross-section is given.

(((SIGIO(*g*,*j*,*m*), *g*=1, NG), *j*=1, NOWSTP), *m*=1, NTDEPZ)

Microscopic capture cross-section of I-135 by collapsed-group by step by depleting material

(((SIGSM(*g*,*j*,*m*), *g*=1, NG), *j*=1, NOWSTP), *m*=1, NTDEPZ)

Microscopic capture cross-section of Sm-149 by collapsed-group by step by depleting material

If no collapsing is done, the fine-group cross-section is given.

(((SIGPM(*g*,*j*,*m*), *g*=1, NG), *j*=1, NOWSTP), *m*=1, NTDEPZ)

Microscopic capture cross-section of Pm-149 by collapsed-group by step by depleting material

If no collapsing is done, the fine-group cross-section is given.

Member ***case*DNT** Homogenized burn-up information of each X-region

Contents of the member are similar to those of the member *case*BNUP; however, all data in this member are homogenized in the *x*-th X-region.

NOWSTP Number of burn-up steps including the initial step.

If the burn-up calculation is completed, NOWSTP=1+IBC1 (Sect.2.5).

NTNUC Total number of depleting nuclides

NG Number of collapsed-groups, If no collapsing is done, number of fine-groups is given.

NGT Number of collapsed-thermal-groups. If no collapsing is done, number of fine thermal groups is given.

IDUM5 = 0 Not used

IDUM6 = 0 Not used

IDUM7 = 0 Not used

IDUM8 = 0 Not used

IDUM9 = 0 Not used

CASEID Case identification (A4) (cf. Block-1 of Sect.2.2)

STDNUC Name (2A4) of the nuclide used for the measure of depletion fraction (cf. Block-4 of Sect.2.5, 'U02350__' is defaulted when IBC2>0 (cf. Block-1 of Sect.2.5)

MTYPX Type of depleting materials

= 1 Fissionable material (usual fuel)

= 2 Non-fissionable depleting material (e.g. Gd_2O_3)

VOLX Volume of this X-region (cm^3)

(NUCLID(*i*), *i*=1, NTNUC) (2A4)

Nuclide name by eight characters (cf. Sect.8.1). The last two characters are filled with two blanks. (e.g. U02350__, AM2421__).

(POWRX(*j*), *j*=1, NOWSTP)

Average power density (MW/cm^3) of this X-region by step

(EXPSX(*j*), *j*=1, NOWSTP)

Average accumulated burn-up (MWd/t) by step

In the case of MTYPX=2, time integrated absorption rate (n/cm^3) is given.

(U235FX(*j*), *j*=1, NOWSTP)

Depletion fraction (%) of nuclide STDNUC (U-235 is defaulted)

U235FX(*j*) ≤ 100.0

(HMINVX(*j*), *j*=1, NOWSTP)

Heavy metal inventory (ton/cm^3) of this X-region by step

(GAMAVX(*j*), *j*=1, NOWSTP)

Heat generation (Joule/fission) of this X-region by step

(YDXEX(*j*), *j*=1, NOWSTP)

Fission yield of Xe-135 by step

(YDIOX(*j*), *j*=1, NOWSTP)

Fission yield of I-135 by step

(YDSMX(*j*), *j*=1, NOWSTP)

Fission yield of Sm-149 by step

(YDPM(*j*), *j*=1, NOWSTP)

Fission yield of Pm-149 by step

((DENSX(*j*,*i*), *j*=1, NOWSTP), *i*=1, NTNUC)

Atomic number density (10^{24} n/cm^3) averaged in X-region by step by nuclide

((AFISSX(*g*,*j*), *g*=1, NG), *j*=1, NOWSTP)

X-region averaged macroscopic absorption cross-section of fissile nuclides by collapsed-group by step

((CFERTX(*g*,*j*), *g*=1, NG), *j*=1, NOWSTP)

X-region averaged macroscopic capture cross-section of fertile nuclides by collapsed group by step

((SIGXEX(*g*,*j*), *g*=1, NG), *j*=1, NOWSTP)

X-region averaged microscopic capture cross-section of Xe-135 by collapsed-group by step

((SIGIOX(*g*,*j*), *g*=1, NG), *j*=1, NOWSTP)

X-region averaged microscopic capture cross-section of I-135 by collapsed-group by step

((SIGSMX(*g*,*j*), *g*=1, NG), *j*=1, NOWSTP)

X-region averaged microscopic capture cross-section of Sm-149 by collapsed-group by step

((SIGPMX(*g*,*j*), *g*=1, NG), *j*=1, NOWSTP)

X-region averaged microscopic capture cross-section of Pm-149 by collapsed-group by step

Member ***caseHTbb*** Member for the restart burn-up calculation

This member is written at the end of every burn-up step (*bb*=00, 01, 02, ….). This keeps the burn-up information of the step to make the members *case*BNUP and *case*x*DNT*.

At the end of whole steps, the code edits these restart members and creates the members *case*BNUP and *case*x*DNT*. Therefore, even if an abnormal termination occurs, the restart is available by using the members *case*HT*bb*. As the contents of this member are included in the members *case*BNUP and *case*x*DNT*, description of data structure is skipped.

6.3 MACROWK

Material-wise and X-region averaged macroscopic cross-sections in the fine-group structure are stored in the MACROWK file. The contents and data structure are the same as those of the member in the MACRO file (cf. CONT0000, *casexbbp*, *casexbb*D in Sect.6.2).

Member name	Contents
CONT0000	energy group structure
matnmbbp	the *p*-th order macroscopic cross-section of M-region
*matnmbb*D	delayed neutron data of M-region which includes fuel materials
casexbbp	the *p*-th order macroscopic cross-section of X-region
casexbbD	delayed neutron data of X-region which includes fuel materials
*matnm*MIC	material-wise microscopic cross-section (this member is internally used and deleted when the calculation is completed)

matnm	The first five characters of the material identification (MTNAME in Sect.2.4)
case	Case identification (CASEID in Sect.2.2).
x	X-region number (1,2,……..,9,A,B,….,Z) is given.
bb	Two digit number corresponding to burn-up steps (*bb*=00,01,02,03….). For non-depleting problem or non-depleting material, the 6-th and 7-th characters of MTNAME are given.
p	Legendre component indicator (the *p*-th component indicated by *p*-tag (*p* =0,1)

6.4 MICRO

The fine-group effective microscopic cross-sections are stored in members named with 14 characters which specify data type (the first tag of ‘X’ or ‘T’), nuclide (*zzmmmc*), material (*matnm*)

or case identification (*case*) with the X-region number (*x*), and burnup step (*bb*).

Member name	Contents
CONT0000	energy group structure
X*zzmmmc*/*matnm*/*bb*	fine-group effective microscopic cross-sections by nuclide (*zzmmmc*) by material (*matnm*) by burn-up step (*bb*).
X*zzmmmc*/*case*/*x*/*bb*	fine-group homogenized effective microscopic cross-sections of X-region (*x*) by nuclide (*zzmmmc*) by burn-up step (*bb*).
T*zzmmmc*/*matnm*/*bb*	fine-group effective microscopic thermal scattering matrix by nuclide (*zzmmmc*) by material (*matnm*) by burn-up step (*bb*).
T*zzmmmc*/*case*/*x*/*bb*	fine-group homogenized effective microscopic thermal scattering matrix of X-region (*x*) by nuclide (*zzmmmc*) by burn-up step (*bb*).
*matnm*MIC	temporary data on material-wise microscopic cross-section (this member is internally used and deleted when the calculation completed)

matnm	The first five characters of the material identification (MTNAME in Sect.2.4)
case	Case identification (CASEID in Sect.2.2).
x	X-region number (1,2,……..,9,A,B,….,Z) is given.
bb	Two digit number corresponding to burn-up steps (*bb*=00,01,02,03….). For non-depleting problem or non-depleting material, the 6-th and 7-th characters of MTNAME are given.

Member **X*zzmmmcmatnmbb*** or **X*zzmmmccasexbb***

The vector data of the fine-group effective microscopic cross-sections are stored in the one-dimensional array in accordance with the following structure, which consists of a header part (H1~H4) and a cross-section data part (*m*=1~8). The value in [ ] indicates the starting address in the one-dimensional array, and that in { } indicates the data length.

H1	NG	Number of fine-energy groups (cf. NGF in Sect.2.2)	[1], {1}
H2	NOMT	Number of cross-section types (NOMT=8 always)	[2], {1}
H3	(IPMT(*m*), *m*=1, NOMT)		[3], {NOMT}
		Starting address of the *m*-th cross-section data in the one-dimensional array of a member	
H4	(NGMT(*m*), *m*=1, NOMT)		[NOMT+3], {NOMT}
		The group number of the last group in which a non-zero value is given to the *m*-th cross-section type	
		Note: To reduce the stored data of a member, the last successive zero values are not stored in the member. (e.g. fission spectrum, cross-sections with high threshold	

energy, etc.) However, for the cross-section type (m) whose cross-section values are all zero (e.g. fission cross-section of a non-fissionable nuclide), a zero value is given only to the first group for convenience of data reading.

1 (XS(g, 1), g=1, NGMT(1)) [IPMT(1)=2×NOMT+3], {NGMT(1)}

Capture ((n,γ) reaction) cross-section, $\sigma_c(g)$

2 (XS(g, 2), g=1, NGMT(2)) [IPMT(2)=IPMT(1)+ NGMT(1)], {NGMT(2)}

Fission cross-section, $\sigma_f(g)$

3 (XS(g, 3), g=1, NGMT(3)) [IPMT(3)=IPMT(2)+ NGMT(2)], {NGMT(3)}

(n,2n) cross-section, $\sigma_{n2n}(g)$

4 (XS(g, 4), g=1, NGMT(4)) [IPMT(4)=IPMT(3)+ NGMT(3)], {NGMT(4)}

Elastic scattering cross-section, $\sigma_e(g)$

5 (XS(g, 5), g=1, NGMT(5)) [IPMT(5)=IPMT(4)+ NGMT(4)], {NGMT(5)}

Elastic removal cross-section, $\sigma_{er}(g)$

6 (XS(g, 6), g=1, NGMT(6)) [IPMT(6)=IPMT(5)+ NGMT(5)], {NGMT(6)}

Number of neutrons produced per fission, $\nu(g)$

7 (XS(g, 7), g=1, NGMT(7)) [IPMT(7)=IPMT(6)+ NGMT(6)], {NGMT(7)}

Fission spectrum, $\chi(g)$

8 (XS(g, 8), g=1, NGMT(8)) [IPMT(8)=IPMT(7)+ NGMT(7)], {NGMT(8)}

Inelastic scattering cross-section, $\sigma_{in}(g)$

Member **Tzzmmmcmatnmbb** or **Tzzmmmccasexbb**

The matrix data of the fine-group effective microscopic thermal scattering cross-sections are stored in the one-dimensional array in accordance with the following structure, which consists of a header part (H1~H7) and a cross-section data part (m=1~2). The value in [ ] indicates the starting address in the one-dimensional array, and that in { } indicates the data length.

H1	NET	Number of fine-energy thermal groups	[1], {1}
H2	NOMT	Number of cross-section types (NOMT=2 always)	[2], {1}
H3	NORD	maximum order of the Legendre components (L of P_L)	[3], {1}
H4	(IPMT(m), m=1,NOMT)		[4], {NOMT}

Stating address of the m-th cross-section data in the one-dimensional array of a member

H5 (NGMT(m), m=1,NOMT) [NOMT+4], {NOMT}

The last group number in which a non-zero value is given to the *m*-th cross-section type

Note: To reduce the stored data of a member, the last successive zero values are not stored in the member.

H6 (LS(g), g=1, NGMT(m)) [2×NOMT +4], {NGMT(1)}

The data storage of the thermal scattering matrix of the g-th thermal energy group $\sigma_s^{th}(g \rightarrow g')$ starts from the g'=LS(g)-th thermal group for the compact storage.

H7 (LE(g), g=1, NGMT(m)) [2×NOMT+NGMT(1)+4], {NGMT(1)}

The data storage of the thermal scattering matrix of the g-th thermal energy group $\sigma_s^{th}(g \rightarrow g')$ ends at the g'=LE(g)-th thermal group for the compact storage.

1 (((SIGS(g', g, n), g'=LS(g), LE(g)), g=1, NGMT(1)), n=1, 1+NORD)

[IPMT(1)≡2×(NOMT+NGMT(1))+4],

{LN(1)≡ $(\mathrm{NORD}+1)\times\sum_g[\mathrm{LE}(g)-\mathrm{LS}(g)+1]$}

The (n-1)-th Legendre component of the thermal scattering matrix; $\sigma_s^{th}(g \rightarrow g', n)$

2 ((XSUP(g, n), g=1, NGMT(2)), n=1, 1+NORD)

[IPMT(2)≡IPMT(1)+LN(1)], {(NORD+1)×NGMT(2)}

The (n-1)-th Legendre component of the up-scattering cross-section from the g-th thermal group to fast energy region, $\sigma_{up}(g, n)$.

Note: The (n-1)-th Legendre component of the total scattering cross-section of the g-th thermal group is given by $\sum_{g'}[\mathrm{SIGS}(g', g, n)] + \mathrm{XSUP}\,(g, n)$.

6.5 HOMO-MIC

Homogenized effective microscopic cross-sections in collapsed-group structure are stored in the HOMO-MIC file. The collapsed-group microscopic cross-sections of the isolated materials (cf. Sect.2.4) are also stored in the HOMO-MIC file.

Member name	Contents
CONT0000	energy group structure
*casex*NLO	List of nuclides in X-region (index by integer)
*matnm*NLO	List of nuclides in isolated material (index by integer)
*casex*NLT	List of nuclides in X-region (index by character)
*matnm*NLT	List of nuclides in isolated material (index by character)
*casexbb*N	homogenized atomic number densities of nuclides in X-region
*matnmoo*N	atomic number densities of nuclides in isolated material

casexbbp the *p*-th order homogenized microscopic cross-sections of nuclides in X-region
matnmoop the *p*-th order microscopic cross-sections of nuclides in isolated material.
*casexbb*D homogenized microscopic delayed neutron data of X-region which includes fuel materials
*matnmoo*D microscopic delayed neutron data of nuclides in isolated material which includes fissionable nuclides
*casexbb*E homogenized microscopic elastic P0 scattering matrices of nuclides in X-region (IOPT9=±3 or ±4)
*matnmoo*E microscopic elastic P0 scattering matrices of nuclides in isolated material (IOPT9=±3 or ±4)
*casexbb*F homogenized microscopic elastic P1 scattering matrices of nuclides in X-region (IOPT9=±3 or ±4)
*matnmoo*F microscopic elastic P1 scattering matrices of nuclides in isolated material (IOPT9=±3 or ±4)
*casexbb*I homogenized microscopic inelastic P0 scattering matrices of nuclides in X-region (IOPT9=±3 or ±4)
*matnmoo*I microscopic inelastic P0 scattering matrices of nuclides in isolated material (IOPT9=±3 or ±4)
*casexbb*W homogenized microscopic (n,2n) reaction matrices of nuclides in X-region (IOPT9=±3 or ±4)
*matnmoo*W microscopic (n,2n) reaction matrices of nuclides in isolated material (IOPT9=±3 or ±4)
*casexbb*T homogenized microscopic (n,3n) reaction matrices of nuclides in X-region (IOPT9=±3 or ±4)
*matnmoo*T microscopic (n,3n) reaction matrices of nuclides in isolated material (IOPT9=±3 or ±4)

****

case Case identification (CASEID in Sect.2.2).
x X-region number (1,2,……..,9,A,B,….,Z) is given.
matnm The first five characters of the material identification (MTNAME in Sect.2.4) for the isolated material.
bb Two digit number corresponding to burn-up steps (*bb*=00,01,02,03….). For non-depleting problem, '00' is given.
oo The 6-th and 7-th characters of material identification (MTNAME in Sect.2.4) for the isolated material.
p Legendre component indicator (the *p*-th component indicated by *p*-tag (*p* =0,1)

****

Member **CONT0000**

NG Number of collapsed-energy groups

(W(*g*),*g*=1,NG)

Lethargy width (asymptotic spectrum for collapsing)

(E(*g*),*g*=1,NG+1)

Energy boundaries (eV) starting at the highest energy

Member ***casex*NLO** or ***matnm*NLO**

(INAME(*i*), *i* =1,NISO)

Integer to identify nuclide

NISO is the number of nuclides whose microscopic cross-sections are stored in other members. It is the same as the data length of this member (LENG). INAME(*i*) is an integer to identify the *i*-th nuclide. For the nuclide whose atomic number is *Z* and mass number *A*.

INAME(*i*) is set to be the integer 10000×Z+10×A+*n*, where *n*=0, 1, 2 for stable nuclides, meta-stable and second meta-stable nuclides, respectively.

(e.g. H-1:10010, U-235:922350, Am-242m:952421), The chemical compound status tag c-tag (cf. Sect.2.4, 8.1) is neglected here.

Member ***casex*NLT** or ***matnm*NLT**

(ANAME(*i*), *i* =1,NISO)

Characters (A8) to identify nuclide

The first six characters of ANAME(*i*) is given by zz-tag, mmm-tag, n-tag or c-tag (cf. Sect.8.1), and the last two characters are filled with blanks.

Member ***casexbb*N** or ***matnmoo*N**

(DNHOMO(*i*), *i* =1,NISO)

Atomic number density (10^{24} atoms/cm^3) of each nuclide in the order of INAME(*i*) or ANAME(*i*).

In the case of the member for the homogenized material (*casexbbn*), the following conservation is satisfied for the numbers of atoms in X-region.

$\sum_m \left(N_m^i V_m \right) = N_X^i V_X$,

where N_m^i is the atomic number density of nuclide (*i*) in the *m*-th R-region included in the *X*-th X-region; V_m and V_X are volumes (cm^3) of R-region and X-region, respectively.

Member ***casexbb*p** or ***matnmoo*p**

This member contains the microscopic cross-section data shown in Table 6.5-1.

Table 6.5-1 Microscopic cross-sections stored in a member file *casexbbp* or *matnmoop* (1/2)

m	Cross-section type
1	Scattering matrix $\sigma_s^i(g \rightarrow g')$ including elastic, inelastic, (n,2n), (n,3n) reactions
2	Absorption $\sigma_a^i(g)$
3	Production $\nu\sigma_f^i(g)$
4	Fission $\sigma_f^i(g)$
5	Fission spectrum $\chi^i(g)$
6	Total cross-section $\sigma_t^i(g)$
7	Transport cross-section $\sigma_{tr}^i(g)$
8	Diffusion coefficient d_1^i (cf. "Note" below)
9	Diffusion coefficient d_2^i (cf. "Note" below)
10	Diffusion coefficient d_3^i (cf. "Note" below)
11	Inverse of velocity $1/\upsilon(g)=\sqrt{\frac{m}{2\bar{E}(g)}}$ [s/cm]
12	Total (n,2n) $=\sum_{g'}\sigma_{n2n}^i(g \rightarrow g')/2$
13	P0 component of elastic scattering $\sigma_e^{i,P0}(g)=\sum_{g'}\sigma_e^{i,P0}(g \rightarrow g')$
14	P1 component of elastic scattering $\sigma_e^{i,P1}(g)=\sum_{g'}\sigma_e^{i,P1}(g \rightarrow g')$ Note: (2L+1) is not multiplied
15	Total inelastic scattering $\sigma_{in}^i(g)=\sum_{g'}\sigma_{in}^i(g \rightarrow g')$
16	Total (n,3n)$=\sum_{g'}\sigma_{n3n}^i(g \rightarrow g')/3$
17	(n,γ) cross-section $\sigma_{n\gamma}^i(g)$ (defined by MT=102 in ENDF-6 the format[33])
18	(n,p) cross-section $\sigma_{np}^i(g)$ (MT=103)
19	(n,α) cross-section $\sigma_{n\alpha}^i(g)$ (MT=107)
20	(n,2n) cross-section $\sigma_{n2n}^i(g)$ (MT=16)

Table 6.5-1 Microscopic cross-sections stored in a member file *casexbbp* or *matnmoop* (2/2)

21	(n,3n) cross-section $\sigma_{n3n}^{i}(g)$ (MT=17)
22-25	Not used (or arbitrary cross-sections defined and added by user)

Note: $d_n^i \equiv c_n^i \left(\frac{1}{3\sigma_{tr}^i}\right)$, where c_n^i is the normalization factor to satisfy $\sum_i N^i d_n^i = D_n$,

where $D_n$is the diffusion coefficient in Table 6.2-1 for the corresponding homogenized region.

The above cross-sections are stored in the one-dimensional array in accordance with the following structure, which consists of a header part (H1∼H6) and a cross-section data part (*m*=1∼25), which has a loop on nuclides. The value in [ ] indicates the top address in the one-dimensional array, and that in { } indicates the data length.

H1 NG Number of energy groups [1], {1}

H2 NOMT Number of cross-section types (NOMT=25 always) [2], {1}

H3 ((IPMT(*m*,*i*), *m*=1,NOMT), *i*=1,NISO) [3], {NOMT×NISO}

Starting address of the *m*-th cross-section data in the one-dimensional array of a member

H4 ((NGMT(*m*,*i*), *m*=1,NOMT), *i*=1,NISO) [NOMT×NISO +3], {NOMT×NISO}

The group number of the last group in which a non-zero value is given to the *m*-th cross-section type

Note: To reduce the stored data of a member, the last successive zero values are not stored in the member. (e.g. fission spectrum, scattering matrix, cross-sections with high threshold energy, etc.) However, for the cross-section type (*m*) whose cross-section values are all zero (e.g. fission cross-section of a non-fissionable nuclide) or for the cross-section type which is not defined (e.g. *m*=22-25), a zero value is given only to the first group for convenience of data reading.

H5 ((LS(g), g=1, NGMT(*m*)), *i*=1,NISO) [2×NOMT×NISO+3], {NGMT(1,*i*)×NISO}

The data storage of the scattering matrix of the g-th energy group $\sigma_s^i(g \to g')$ starts from the g'=LS(g)-th group for the compact storage.

H6 ((LE(g), g=1, NGMT(*m*)), *i*=1,NISO)

[2×NOMT×NISO+$\sum_i$ NGMT$(1,i)$+3], {NGMT(1,*i*)×NISO}

The data storage of the scattering matrix of the g-th energy group $\sigma_s^i(g \to g')$ ends at the g'=LE(g)-th group for the compact storage.

DO i=1, NISO The following nuclide-wise data are repeated NISO times. NISO is the number of nuclides whose cross-section data are stored in this member. It is the same as the data length (LENG) of the member `casexNLO` or `matnmNLO`.

1 ((SIGS^i(g', g), g'=LS(g), LE(g)), g=1, NGMT(1,i))

[IPMT(1,1)≡2×NOMT×NISO+2 × $\sum_i^{\mathrm{NISO}}$ NGMT$(1, i)$+3], when i=1

{LN(1,i)≡ $\sum_g$[LE(g, i) − LS(g, i) + 1]}

[IPMT(1,i)=IPMT(1,i-1)+LNT(i-1)], when i>1

LNT(i)≡ $\sum_g$[LE(g, i) − LS(g, i) + 1]+$\sum_{m=2,\mathrm{NOMT}}$(NGMT(m, i))}

Scattering matrix, $\sigma_s^i(g \to g')$

2 (XS^i $(g, 2)$,g=1, NGMT(2)) [IPMT(2,i)=IPMT(1,i)+LN(1,i)], {LN(2,i) ≡NGMT(2,i)}

The 2nd type of cross-section data, $\sigma_a(g)$

:

m (XS^i (g, m),g=1, NGMT(m)) [IPMT(m,i)=IPMT(m-1,i)+LN(m-1,i)], {LN(m,i) ≡NGMT(m,i)}

The m-th type of cross-section data for the i-th nuclide (cf. Table 6.5-1)

:

25 (XS^i $(g$,25), g=1, NGMT(25)) [IPMT(25,i)=IPMT(24,i)+LN(24,i)], {LN(25,i) ≡NGMT(25,i)}

The 25-th type of cross-section data. (Dummy data for future use)

Note: A zero value is given only to the 1st group.

END DO (End of the loop on nuclide i)

7. Utility for PDS File Management

Printed information on the 99-th device may be insufficient for the user's demand or may be not convenient for tabulation and plotting. In the PDS files such as FLUX, MACRO, MACROWK, and so on, a lot of information is stored as binary data. Here, several utilities to extracting necessary information from PDS files are shown.

(1) PDStoTXT [~util/smpl/shr/PDStoTXT.sh]

The PDStoTXT is a program to convert the contents of a member in PDS file into a text file. It is used to dump the contents of members in PDS file or to transfer the members into a machine of different data organization.

(2) TXTtoPDS [~util/smpl/shr/TXTtoPDS.sh]

The TXTtoPDS is a utility program to reconvert a text file converted by PDStoTXT into PDS file. By using PDStoTXT and TXTtoPDS, it is available to transfer PDS files between machines with different data organization. By these utilities, The Public Libraries or other user's PDS files can be easily transferred from a machine to others.

(3) MacroEdit [~util/smpl/shr/MacroEdit.sh]

The MacroEdit is a utility program to print out the contents of the member of macroscopic cross-section file MACRO or MACROWK. While the direct use of PDStoTXT for this purpose gives simply printout of one-dimensional array, the use of this MacroEdit gives tabulated form of macroscopic cross-sections.

(4) MicroEdit [~util/smpl/shr/MicroEdit.sh]

The MicroEdit is a utility program to print out the contents of the member of microscopic cross-section file MICRO. It is used for the same purpose as MacroEdit but for microscopic cross-sections.

(5) FluxEdit [~util/smpl/shr/FluxEdit.sh]

The FluxEdit is a utility program to print out the contents of the member storing neutron flux of fine or collapsed group structure. The flux in FLUX is multiplied by the region volume. The volume can be also printed out in addition to the neutron spectrum and its spatial distribution in a tabulated form. The usage is almost the same as to use MacroEdit.

(6) FluxPlot [~util/smpl/shr/FluxPlot.sh]

The FluxPlot is a utility program to provide a table in text form for plotting the neutron spectrum in histogram (log-log) style graph by reading the energy group structure and flux on the FLUX file. Actual plotting is done with a spreadsheet software on the market by feeding the table provided by FluxPlot.

(7) BnupEdit [~util/smplshr/BnupEdit.sh]

The BnupEdit is a utility program to print out the information on burn-up calculation (members `caseHTbb`) on the MACRO file. The major results of burn-up calculation written on logical unit 98 are reproduced by BnupEdit.

8. Tables on Cross-Section Library

8.1 Symbols to Denote Nuclide

Table 8.1-1 Chemical symbols (zz-tag)

Z	Element Name	zz-tag	Z	Element Name	zz-tag	Z	Element Name	zz-tag
1	Hydrogen	H0	35	Bromine	BR	71	Lutetium	LU
1	Deuterium	D0	36	Krypton	KR	72	Hafnium	HF
1	Tritium	T0	37	Rubidium	RB	73	Tantalum	TA
2	Helium	HE	38	Strontium	SR	74	Tungsten	W0
3	Lithium	LI	39	Yttrium	Y0	75	Rhenium	RE
4	Beryllium	BE	40	Zirconium	ZR	76	Osmium	OS
5	Boron	B0	41	Nioblium	NB	77	Iridium	IR
6	Carbon	C0	42	Molybdenum	MO	78	Platinium	PT
7	Nitrogen	N0	43	Technetium	TC	79	Gold	AU
8	Oxygen	O0	44	Ruthenium	RU	80	Mercury	HG
9	Fluorine	F0	45	Rhodium	RH	81	Thallium	TL
10	Neon	NE	46	Palladium	PD	82	Lead	PB
11	Sodium	NA	47	Silver	AG	83	Bismuth	BI
12	Magnesium	MG	48	Cadmium	CD	84	Polonium	PO
13	Aluminium	AL	49	Indium	IN	85	Astatine	AT
14	Silicon	SI	50	Tin	SN	86	Radon	RN
15	Phosphorus	P0	51	Antimony	SB	87	Francium	FR
16	Sulfur	S0	52	Tellurium	TE	88	Radium	RA
17	Chlorine	CL	53	Iodine	I0	89	Actinium	AC
18	Argon	AR	54	Xenon	XE	90	Thorium	TH
19	Potassium	K0	55	Cesium	CS	91	Protactinium	PA
20	Calcium	CA	56	Barium	BA	92	Uranium	U0
21	Scandium	SC	57	Lanthane	LA	93	Neptunium	NP
22	Titanium	TI	58	Cerium	CE	94	Plutonium	PU
23	Vanadium	V0	59	Praseodymium	PR	95	Americium	AM
24	Chromium	CR	60	Neodymium	ND	96	Curium	CM
25	Manganese	MN	61	Promethium	PM	97	Berkelium	BK
26	Iron	FE	62	Samarium	SM	98	Californium	CF
27	Cobalt	CO	63	Europium	EU	99	Einsteinium	ES
28	Nickel	NI	64	Gadolinium	GD	100	Fermium	FM
29	Copper	CU	65	Terbium	TB	101	Mendelevium	MD
30	Zinc	ZN	66	Dysprosium	DY	102	Nobelium	NO
31	Gallium	GA	67	Holmium	HO	103	Lawrencium	LR
32	Germanium	GE	68	Erbium	ER	104	Kurchatovium	KU
33	Arsenic	AS	69	Thulium	TM			
34	Selenium	SE	70	Ytterbium	YB			

Specification of nuclide in the input of MOSRA-SRAC is performed by 6 characters in the form of zzmmmn or zzmmmc, where,

- zz-tag denotes element by 2 characters of chemical symbol as shown in Table 8.1-1. As a special case, 'ZZ' is given to some pseudo fission products described in the burn-up chain model.
- mmm-tag is triple-digit to denote mass number of nuclide. For example, mmm-tag is '235' for U-235, '001' for H-1 and '010' for B-10. If the cross-section of natural element is compiled in the library, '000' is given (e.g. FE000 for Fe-natural).
- n-tag is used for the nuclides whose scattering kernel is given by the free gas model. It denotes the excited level of nuclide. That is, n-tag is 0 for the ground state, 1 for the 1st meta-stable, and 2 for the 2nd meta-stable.
- c-tag is used instead of n-tag for the nuclide whose thermal scattering law data is given by nuclear data in consideration of chemical binding effect. The c-tag is one character specified in Table 8.1-2.

Table 8.1-2 Chemical bounding symbols (c-tag)

Compound	*c*-tag	Chemical symbol	Remarks (combination)
Beryllium metal	B	Be	BE009B
Beryllium oxide	E	BeO	BE009E+O0016E
Benzene	Q	C_6H_6	H0001Q+C0000(free)
Graphite	C	C	C0000C
Polyethylene	P	$(CH_2)_n$	2(H0001P)+C00001(free)
Uranium carbite	V	UC	not used
Uranium oxide	W	UO2	U0238W+2(O0016W) (not used)
Light water	H	H2O	2(H0001H)+O0016(free)
Heavy water	D	D2O	2(D0002D)+O0016(free)
Zirconium hydrate	Z	ZrH	ZR*mmm*Z+H0001Z (*mmm*: mmm-tag of each isotope)
Free atom	0	*	e.g. U02350, Am2420
1st meta stable	1	*	e.g. Am2421 (Am-242m)

Note: Use with pairing nuclide(s) specified by "+" in Remarks column.

8.2 Nuclide List of Public Library

The present Public Library (MSRACLIB_J40) contains the cross-section data for 449 nuclides shown in Table 8.2-1. Most of them are generated from JENDL-4.0[4)] which has been well validated by integral testing for various fast and thermal fission reactors[34)]. Data for the nuclide with natural isotopes abundance are composed from the nuclear data of constituent isotopes, for which 'comp' is

specified in Table 8.2-1. Exceptionally, the data of Eu-156 and Ho-165 are generated from JENDL-3.3[35] and ENDF/B-VII.0[19], respectively. This is because the evaluation of Eu-156 of JENDL-3.3 gives better results for burn-up benchmark of LWR spent fuel compositions[36], and because cross-sections of Ho-165 have not been evaluated in JENDL.

The explanation of items in Table 8.2-1 follows.

****

FASTDATA	=NO	No self-shielding factor table for fast group constants
	=YES	Self-shielding factor table for fast group constants
F-TABLE NTEMP	Number of temperature points for self-shielding factor table	
	=0	No table
	=1	1 point (usually 293.15K)
	=4	4 points (usually 293.15K, 800K, 2100K, 4500K)
	Temperatures are described in member Czzmmm0 in PFAST	
MCROSS LIBRARY	=0	No MCROSS library data for PEACO
	=1	There is MCROSS library data for PEACO
THERMAL F-TAB.	=NO	No self-shielding factor table for thermal group constants
	=YES	Self-shielding factor table for thermal group constants
NTEMP	Number of temperature points for thermal group constants	
	Temperatures are described in member Czzmmmc0 in PTHERMAL	
THERMAL KERNEL	=P0	P_0 component of thermal scattering matrix
	=P1	P_1 component of thermal scattering matrix
	=P0-Pn	P_0 through P_n components of thermal scattering matrix
	=NO DATA	No thermal scattering matrix

****

Table 8.2-1(1/9) Nuclide list of the Public Library (MSRACLIB_J40)

NO	NUCLIDE	FAST DATA	F-TABLE NTEMP	MCROSS LIBRARY	THERMAL F-TAB	NTEMP	THERMAL KERNEL	REMARKS
1	AC2250	YES	1	NO	NO	1	P0-P3	
2	AC2260	YES	1	NO	NO	1	P0-P3	
3	AC2270	YES	1	NO	NO	1	P0-P3	
4	AG0000	YES	4	NO	NO	10	P0-P3	comp
5	AG1070	YES	4	YES	NO	10	P0-P3	
6	AG1090	YES	4	YES	YES	10	P0-P3	
7	AG1101	YES	4	NO	NO	1	P0-P3	
8	AG1110	YES	4	NO	NO	1	P0-P3	
9	AL0270	YES	4	NO	NO	10	P0-P3	
10	AM2400	YES	1	NO	NO	1	P0-P3	
11	AM2410	YES	4	YES	YES	10	P0-P3	
12	AM2420	YES	4	YES	YES	10	P0-P3	
13	AM2421	YES	4	YES	YES	10	P0-P3	
14	AM2430	YES	4	YES	YES	10	P0-P3	
15	AM2440	YES	1	NO	NO	1	P0-P3	
16	AM2441	YES	1	NO	NO	1	P0-P3	
17	AR0400	YES	4	NO	NO	10	P0-P3	
18	AS0750	YES	4	NO	NO	1	P0-P3	
19	AU1970	YES	4	YES	YES	10	P0-P3	
20	B00100	YES	1	NO	NO	10	P0-P3	
21	B00110	YES	1	NO	NO	10	P0-P3	
22	BA1300	YES	4	NO	NO	1	P0-P3	
23	BA1320	YES	4	NO	NO	1	P0-P3	
24	BA1340	YES	4	NO	NO	1	P0-P3	
25	BA1350	YES	4	NO	NO	1	P0-P3	
26	BA1360	YES	4	NO	NO	1	P0-P3	
27	BA1370	YES	4	NO	NO	1	P0-P3	
28	BA1380	YES	4	NO	NO	1	P0-P3	
29	BA1400	YES	4	NO	NO	10	P0-P3	
30	BE0090	YES	1	NO	NO	10	P0-P3	
31	BE009B	YES	1	NO	NO	8	P0-P3	
32	BE009E	YES	1	NO	NO	8	P0-P3	
33	BI2090	YES	4	NO	NO	10	P0-P3	
34	BK2450	YES	1	NO	NO	1	P0-P3	
35	BK2460	YES	1	NO	NO	1	P0-P3	
36	BK2470	YES	1	NO	NO	1	P0-P3	
37	BK2480	YES	1	NO	NO	1	P0-P3	
38	BK2490	YES	4	NO	NO	1	P0-P3	
39	BK2500	YES	1	NO	NO	1	P0-P3	
40	BR0790	YES	4	NO	NO	1	P0-P3	
41	BR0810	YES	4	NO	NO	1	P0-P3	
42	C00000	YES	1	NO	NO	10	P0-P3	
43	C0000C	YES	1	NO	NO	10	P0-P3	
44	CA0000	YES	4	NO	NO	10	P0-P3	comp
45	CA0400	YES	4	NO	NO	10	P0-P3	
46	CA0420	YES	4	NO	NO	10	P0-P3	
47	CA0430	YES	4	NO	NO	10	P0-P3	
48	CA0440	YES	4	NO	NO	10	P0-P3	
49	CA0460	YES	4	NO	NO	10	P0-P3	
50	CA0480	YES	4	NO	NO	10	P0-P3	

comp: composed from the nuclear data of constituent isotopes

Table 8.2-1(2/9) Nuclide list of the Public Library (MSRACLIB_J40)

NO	NUCLIDE	FAST DATA	F-TABLE NTEMP	MCROSS LIBRARY	THERMAL F-TAB	NTEMP	THERMAL KERNEL	REMARKS
51	CD0000	YES	4	NO	NO	10	P0-P3	comp
52	CD1060	YES	4	NO	NO	10	P0-P3	
53	CD1080	YES	4	NO	NO	10	P0-P3	
54	CD1100	YES	4	NO	NO	10	P0-P3	
55	CD1110	YES	4	NO	NO	10	P0-P3	
56	CD1120	YES	4	NO	NO	10	P0-P3	
57	CD1130	YES	4	YES	NO	10	P0-P3	
58	CD1140	YES	4	NO	NO	10	P0-P3	
59	CD1160	YES	4	NO	NO	10	P0-P3	
60	CE1400	YES	4	NO	NO	10	P0-P3	
61	CE1410	YES	4	NO	NO	10	P0-P3	
62	CE1420	YES	4	NO	NO	10	P0-P3	
63	CE1430	YES	4	NO	NO	1	P0-P3	
64	CE1440	YES	4	NO	NO	1	P0-P3	
65	CF2460	YES	1	NO	NO	1	P0-P3	
66	CF2480	YES	1	NO	NO	1	P0-P3	
67	CF2490	YES	4	NO	NO	1	P0-P3	
68	CF2500	YES	4	NO	NO	1	P0-P3	
69	CF2510	YES	4	NO	NO	1	P0-P3	
70	CF2520	YES	4	NO	NO	1	P0-P3	
71	CF2530	YES	1	NO	NO	1	P0-P3	
72	CF2540	YES	1	NO	NO	1	P0-P3	
73	CL0000	YES	1	NO	NO	10	P0-P3	comp
74	CL0350	YES	1	NO	NO	10	P0-P3	
75	CL0370	YES	1	NO	NO	10	P0-P3	
76	CM2400	YES	1	NO	NO	1	P0-P3	
77	CM2410	YES	1	NO	NO	1	P0-P3	
78	CM2420	YES	4	YES	NO	10	P0-P3	
79	CM2430	YES	4	YES	YES	10	P0-P3	
80	CM2440	YES	4	YES	NO	10	P0-P3	
81	CM2450	YES	4	YES	YES	10	P0-P3	
82	CM2460	YES	4	YES	YES	10	P0-P3	
83	CM2470	YES	4	NO	NO	1	P0-P3	
84	CM2480	YES	4	NO	NO	1	P0-P3	
85	CM2490	YES	1	NO	NO	1	P0-P3	
86	CM2500	YES	4	NO	NO	1	P0-P3	
87	CO0590	YES	4	NO	NO	10	P0-P3	
88	CR0000	YES	4	NO	NO	10	P0-P3	comp
89	CR0500	YES	4	NO	NO	10	P0-P3	
90	CR0520	YES	4	NO	NO	10	P0-P3	
91	CR0530	YES	4	NO	NO	10	P0-P3	
92	CR0540	YES	4	NO	NO	10	P0-P3	
93	CS1330	YES	4	YES	NO	10	P0-P3	
94	CS1340	YES	4	NO	NO	10	P0-P3	
95	CS1350	YES	4	YES	NO	10	P0-P3	
96	CS1360	YES	4	NO	NO	1	P0-P3	
97	CS1370	YES	4	NO	NO	10	P0-P3	
98	CU0000	YES	4	NO	NO	10	P0-P3	comp
99	CU0630	YES	4	NO	NO	10	P0-P3	
100	CU0650	YES	4	NO	NO	10	P0-P3	

Table 8.2-1(3/9) Nuclide list of the Public Library (MSRACLIB_J40)

NO	NUCLIDE	FAST DATA	F-TABLE NTEMP	MCROSS LIBRARY	THERMAL F-TAB	NTEMP	THERMAL KERNEL	REMARKS
101	DY1540	YES	4	NO	YES	10	P0-P3	
102	DY1560	YES	4	NO	YES	10	P0-P3	
103	DY1580	YES	4	NO	NO	10	P0-P3	
104	DY1590	YES	4	NO	YES	10	P0-P3	
105	DY1600	YES	4	NO	YES	10	P0-P3	
106	DY1610	YES	4	NO	YES	10	P0-P3	
107	DY1620	YES	4	NO	YES	10	P0-P3	
108	DY1630	YES	4	NO	YES	10	P0-P3	
109	DY1640	YES	4	NO	NO	10	P0-P3	
110	ER1620	YES	4	YES	NO	10	P0-P3	
111	ER1640	YES	4	YES	NO	10	P0-P3	
112	ER1660	YES	4	YES	NO	10	P0-P3	
113	ER1670	YES	4	YES	YES	10	P0-P3	
114	ER1680	YES	4	YES	NO	10	P0-P3	
115	ER1700	YES	4	YES	NO	10	P0-P3	
116	ES2510	YES	1	NO	NO	1	P0-P3	
117	ES2520	YES	1	NO	NO	1	P0-P3	
118	ES2530	YES	4	NO	NO	1	P0-P3	
119	ES2540	YES	1	NO	NO	1	P0-P3	
120	ES2541	YES	1	NO	NO	1	P0-P3	
121	ES2550	YES	1	NO	NO	1	P0-P3	
122	EU0000	YES	4	NO	NO	10	P0-P3	comp
123	EU1510	YES	4	YES	YES	10	P0-P3	
124	EU1520	YES	4	NO	YES	10	P0-P3	
125	EU1530	YES	4	YES	YES	10	P0-P3	
126	EU1540	YES	4	YES	YES	10	P0-P3	
127	EU1550	YES	4	YES	YES	10	P0-P3	
128	EU1560	YES	4	NO	NO	10	P0-P3	j33
129	EU1570	YES	4	NO	NO	1	P0-P3	
130	F00190	YES	4	NO	NO	10	P0-P3	
131	FE0000	YES	4	NO	NO	10	P0-P3	comp
132	FE0540	YES	4	NO	NO	10	P0-P3	
133	FE0560	YES	4	NO	NO	10	P0-P3	
134	FE0570	YES	4	NO	NO	10	P0-P3	
135	FE0580	YES	4	NO	NO	10	P0-P3	
136	FE0590	YES	4	NO	NO	10	P0-P3	
137	FM2550	YES	1	NO	NO	1	P0-P3	
138	GA0000	YES	4	NO	NO	10	P0-P3	
139	GA0690	YES	4	NO	NO	10	P0-P3	
140	GA0710	YES	4	NO	NO	10	P0-P3	
141	GD1520	YES	4	YES	YES	10	P0-P3	
142	GD1530	YES	4	NO	YES	10	P0-P3	
143	GD1540	YES	4	YES	NO	10	P0-P3	
144	GD1550	YES	4	YES	YES	10	P0-P3	
145	GD1560	YES	4	YES	NO	10	P0-P3	
146	GD1570	YES	4	YES	YES	10	P0-P3	
147	GD1580	YES	4	YES	NO	10	P0-P3	
148	GD1600	YES	4	YES	NO	10	P0-P3	
149	GE0000	YES	4	NO	NO	10	P0-P3	comp
150	GE0700	YES	4	NO	NO	10	P0-P3	

j33: generated from JENDL-3.3

Table 8.2-1(4/9) Nuclide list of the Public Library (MSRACLIB_J40)

NO	NUCLIDE	FAST DATA	F-TABLE NTEMP	MCROSS LIBRARY	THERMAL F-TAB	NTEMP	THERMAL KERNEL	REMARKS
151	GE0720	YES	4	NO	NO	10	P0-P3	
152	GE0730	YES	4	NO	NO	10	P0-P3	
153	GE0740	YES	4	NO	NO	10	P0-P3	
154	GE0760	YES	4	NO	NO	10	P0-P3	
155	H00010	NO	0	NO	NO	10	P0-P5	
156	H0001H	NO	0	NO	NO	8	P0-P5	
157	H0001P	NO	0	NO	NO	2	P0-P3	
158	H0001Q	NO	0	NO	NO	8	P0-P3	
159	H0001Z	NO	0	NO	NO	8	P0-P3	
160	H00020	NO	0	NO	NO	10	P0-P3	
161	H0002D	NO	0	NO	NO	8	P0-P3	
162	HE0030	YES	1	NO	NO	10	P0-P3	
163	HE0040	YES	1	NO	NO	10	P0-P3	
164	HF0000	YES	4	NO	NO	10	P0-P3	comp
165	HF1740	YES	4	YES	YES	10	P0-P3	
166	HF1760	YES	4	YES	NO	10	P0-P3	
167	HF1770	YES	4	YES	YES	10	P0-P3	
168	HF1780	YES	4	YES	NO	10	P0-P3	
169	HF1790	YES	4	YES	NO	10	P0-P3	
170	HF1800	YES	4	YES	NO	10	P0-P3	
171	HF1810	YES	4	NO	NO	10	P0-P3	
172	HF1820	YES	4	NO	NO	10	P0-P3	
173	HG0000	YES	4	NO	NO	10	P0-P3	comp
174	HG1960	YES	4	YES	NO	10	P0-P3	
175	HG1980	YES	4	YES	NO	10	P0-P3	
176	HG1990	YES	4	YES	NO	10	P0-P3	
177	HG2000	YES	4	NO	NO	10	P0-P3	
178	HG2010	YES	4	YES	NO	10	P0-P3	
179	HG2020	YES	4	NO	NO	10	P0-P3	
180	HG2040	YES	4	NO	NO	10	P0-P3	
181	HO1650	YES	4	YES	YES	10	P0-P3	b70
182	I01270	YES	4	YES	NO	10	P0-P3	
183	I01290	YES	4	YES	NO	10	P0-P3	
184	I01300	YES	4	NO	NO	1	P0-P3	
185	I01310	YES	4	NO	NO	10	P0-P3	
186	I01350	YES	4	NO	NO	10	P0-P3	
187	IN1130	YES	4	YES	YES	10	P0-P3	
188	IN1150	YES	4	YES	YES	10	P0-P3	
189	K00000	YES	4	NO	NO	10	P0-P3	comp
190	K00390	YES	4	NO	NO	10	P0-P3	
191	K00400	YES	4	NO	NO	10	P0-P3	
192	K00410	YES	4	NO	NO	10	P0-P3	
193	KR0780	YES	4	NO	NO	1	P0-P3	
194	KR0800	YES	4	NO	NO	1	P0-P3	
195	KR0820	YES	4	NO	NO	1	P0-P3	
196	KR0830	YES	4	NO	NO	10	P0-P3	
197	KR0840	YES	4	NO	NO	1	P0-P3	
198	KR0850	YES	4	NO	NO	1	P0-P3	
199	KR0860	YES	4	NO	NO	1	P0-P3	
200	LA1380	YES	4	NO	YES	10	P0-P3	

b70: generated from ENDF/B-VII.0

Table 8.2-1(5/9) Nuclide list of the Public Library (MSRACLIB_J40)

NO	NUCLIDE	FAST DATA	F-TABLE NTEMP	MCROSS LIBRARY	THERMAL F-TAB	NTEMP	THERMAL KERNEL	REMARKS
201	LA1390	YES	4	YES	NO	10	P0-P3	
202	LA1400	YES	4	NO	NO	10	P0-P3	
203	LI0060	YES	1	NO	NO	10	P0-P3	
204	LI0070	YES	1	NO	NO	10	P0-P3	
205	MG0000	YES	4	NO	NO	10	P0-P3	comp
206	MG0240	YES	4	NO	NO	10	P0-P3	
207	MG0250	YES	4	NO	NO	10	P0-P3	
208	MG0260	YES	4	NO	NO	10	P0-P3	
209	MN0550	YES	4	NO	NO	10	P0-P3	
210	MO0000	YES	4	NO	NO	10	P0-P3	comp
211	MO0920	YES	4	NO	NO	10	P0-P3	
212	MO0940	YES	4	NO	NO	10	P0-P3	
213	MO0950	YES	4	YES	NO	10	P0-P3	
214	MO0960	YES	4	NO	NO	10	P0-P3	
215	MO0970	YES	4	NO	NO	10	P0-P3	
216	MO0980	YES	4	NO	NO	10	P0-P3	
217	MO0990	YES	4	NO	NO	1	P0-P3	
218	MO1000	YES	4	NO	NO	10	P0-P3	
219	N00140	YES	1	NO	NO	10	P0-P3	
220	N00150	YES	1	NO	NO	10	P0-P3	
221	NA0230	YES	4	NO	NO	10	P0-P3	
222	NB0930	YES	4	NO	NO	10	P0-P3	
223	NB0940	YES	4	NO	NO	1	P0-P3	
224	NB0950	YES	4	NO	NO	10	P0-P3	
225	ND1420	YES	4	NO	NO	1	P0-P3	
226	ND1430	YES	4	YES	NO	10	P0-P3	
227	ND1440	YES	4	NO	NO	1	P0-P3	
228	ND1450	YES	4	YES	YES	10	P0-P3	
229	ND1460	YES	4	NO	NO	1	P0-P3	
230	ND1470	YES	4	NO	NO	10	P0-P3	
231	ND1480	YES	4	NO	NO	10	P0-P3	
232	ND1500	YES	4	NO	NO	1	P0-P3	
233	NI0000	YES	4	NO	NO	10	P0-P3	comp
234	NI0580	YES	4	NO	NO	10	P0-P3	
235	NI0590	YES	4	NO	NO	10	P0-P3	
236	NI0600	YES	4	NO	NO	10	P0-P3	
237	NI0610	YES	4	NO	NO	10	P0-P3	
238	NI0620	YES	4	NO	NO	10	P0-P3	
239	NI0640	YES	4	NO	NO	10	P0-P3	
240	NP2340	YES	1	NO	NO	1	P0-P3	
241	NP2350	YES	1	NO	NO	1	P0-P3	
242	NP2360	YES	4	YES	YES	10	P0-P3	
243	NP2370	YES	4	YES	YES	10	P0-P3	
244	NP2380	YES	4	NO	NO	1	P0-P3	
245	NP2390	YES	1	NO	NO	10	P0-P3	
246	O00160	YES	1	NO	NO	10	P0-P3	
247	O0016E	YES	1	NO	NO	8	NO DATA	
248	OS1840	YES	4	NO	NO	10	P0-P3	
249	OS1860	YES	4	NO	NO	10	P0-P3	
250	OS1870	YES	4	NO	NO	10	P0-P3	

Table 8.2-1(6/9) Nuclide list of the Public Library (MSRACLIB_J40)

NO	NUCLIDE	FAST DATA	F-TABLE NTEMP	MCROSS LIBRARY	THERMAL F-TAB	NTEMP	THERMAL KERNEL	REMARKS
251	OS1880	YES	4	NO	NO	10	P0-P3	
252	OS1890	YES	4	NO	NO	10	P0-P3	
253	OS1900	YES	4	NO	NO	10	P0-P3	
254	OS1920	YES	4	NO	NO	10	P0-P3	
255	P00310	YES	4	NO	NO	10	P0-P3	
256	PA2290	YES	1	NO	NO	1	P0-P3	
257	PA2300	YES	1	NO	NO	1	P0-P3	
258	PA2310	YES	4	YES	YES	10	P0-P3	
259	PA2320	YES	4	NO	NO	1	P0-P3	
260	PA2330	YES	4	YES	YES	10	P0-P3	
261	PB0000	YES	4	NO	NO	10	P0-P3	comp
262	PB2040	YES	4	NO	NO	10	P0-P3	
263	PB2060	YES	4	NO	NO	10	P0-P3	
264	PB2070	YES	4	NO	NO	10	P0-P3	
265	PB2080	YES	4	NO	NO	10	P0-P3	
266	PD1020	YES	4	NO	NO	1	P0-P3	
267	PD1040	YES	4	NO	NO	1	P0-P3	
268	PD1050	YES	4	YES	NO	10	P0-P3	
269	PD1060	YES	4	NO	NO	10	P0-P3	
270	PD1070	YES	4	YES	YES	10	P0-P3	
271	PD1080	YES	4	YES	YES	10	P0-P3	
272	PD1100	YES	4	NO	NO	1	P0-P3	
273	PM1470	YES	4	YES	YES	10	P0-P3	
274	PM1480	YES	4	NO	NO	10	P0-P3	
275	PM1481	YES	4	NO	YES	10	P0-P3	
276	PM1490	YES	4	NO	YES	10	P0-P3	
277	PM1510	YES	4	NO	YES	10	P0-P3	
278	PR1410	YES	4	YES	NO	10	P0-P3	
279	PR1430	YES	4	NO	NO	10	P0-P3	
280	PU2360	YES	4	YES	YES	10	P0-P3	
281	PU2370	YES	1	NO	NO	1	P0-P3	
282	PU2380	YES	4	YES	YES	10	P0-P3	
283	PU2390	YES	4	YES	YES	10	P0-P3	
284	PU2400	YES	4	YES	YES	10	P0-P3	
285	PU2410	YES	4	YES	YES	10	P0-P3	
286	PU2420	YES	4	YES	YES	10	P0-P3	
287	PU2440	YES	4	NO	NO	1	P0-P3	
288	PU2460	YES	1	NO	NO	1	P0-P3	
289	RA2230	YES	1	NO	NO	1	P0-P3	
290	RA2240	YES	1	NO	NO	1	P0-P3	
291	RA2250	YES	1	NO	NO	1	P0-P3	
292	RA2260	YES	4	NO	NO	1	P0-P3	
293	RB0850	YES	4	NO	NO	1	P0-P3	
294	RB0860	YES	4	NO	NO	1	P0-P3	
295	RB0870	YES	4	NO	NO	1	P0-P3	
296	RH1030	YES	4	YES	YES	10	P0-P3	
297	RH1050	YES	4	NO	YES	10	P0-P3	
298	RU0960	YES	4	NO	NO	1	P0-P3	
299	RU0980	YES	4	NO	NO	1	P0-P3	
300	RU0990	YES	4	NO	NO	1	P0-P3	

Table 8.2-1(7/9) Nuclide list of the Public Library (MSRACLIB_J40)

NO	NUCLIDE	FAST DATA	F-TABLE NTEMP	MCROSS LIBRARY	THERMAL F-TAB	NTEMP	THERMAL KERNEL	REMARKS
301	RU1000	YES	4	NO	NO	1	P0-P3	
302	RU1010	YES	4	YES	NO	10	P0-P3	
303	RU1020	YES	4	NO	NO	10	P0-P3	
304	RU1030	YES	4	NO	NO	10	P0-P3	
305	RU1040	YES	4	NO	NO	10	P0-P3	
306	RU1050	YES	4	NO	NO	10	P0-P3	
307	RU1060	YES	4	NO	NO	1	P0-P3	
308	S00000	YES	4	NO	NO	10	P0-P3	comp
309	S00320	YES	4	NO	NO	10	P0-P3	
310	S00330	YES	4	NO	NO	10	P0-P3	
311	S00340	YES	4	NO	NO	10	P0-P3	
312	S00360	YES	4	NO	NO	10	P0-P3	
313	SB0000	YES	4	NO	NO	10	P0-P3	comp
314	SB1210	YES	4	YES	NO	10	P0-P3	
315	SB1230	YES	4	YES	NO	10	P0-P3	
316	SB1240	YES	4	NO	YES	10	P0-P3	
317	SB1250	YES	4	NO	NO	1	P0-P3	
318	SB1260	YES	4	NO	NO	1	P0-P3	
319	SC0450	YES	4	NO	NO	10	P0-P3	
320	SE0740	YES	4	NO	NO	1	P0-P3	
321	SE0760	YES	4	NO	NO	1	P0-P3	
322	SE0770	YES	4	NO	NO	1	P0-P3	
323	SE0780	YES	4	NO	NO	1	P0-P3	
324	SE0790	YES	4	NO	NO	1	P0-P3	
325	SE0800	YES	4	NO	NO	1	P0-P3	
326	SE0820	YES	4	NO	NO	1	P0-P3	
327	SI0000	YES	4	NO	NO	10	P0-P3	comp
328	SI0280	YES	4	NO	NO	10	P0-P3	
329	SI0290	YES	4	NO	NO	10	P0-P3	
330	SI0300	YES	4	NO	NO	10	P0-P3	
331	SM1440	YES	4	NO	NO	10	P0-P3	
332	SM1470	YES	4	YES	YES	10	P0-P3	
333	SM1480	YES	4	YES	NO	10	P0-P3	
334	SM1490	YES	4	YES	YES	10	P0-P3	
335	SM1500	YES	4	YES	NO	10	P0-P3	
336	SM1510	YES	4	YES	YES	10	P0-P3	
337	SM1520	YES	4	YES	NO	10	P0-P3	
338	SM1530	YES	4	NO	NO	1	P0-P3	
339	SM1540	YES	4	NO	NO	1	P0-P3	
340	SN0000	YES	4	NO	NO	10	P0-P3	comp
341	SN1120	YES	4	NO	NO	10	P0-P3	
342	SN1140	YES	4	NO	NO	10	P0-P3	
343	SN1150	YES	4	NO	NO	10	P0-P3	
344	SN1160	YES	4	NO	NO	10	P0-P3	
345	SN1170	YES	4	NO	NO	10	P0-P3	
346	SN1180	YES	4	NO	NO	10	P0-P3	
347	SN1190	YES	4	NO	NO	10	P0-P3	
348	SN1200	YES	4	NO	NO	10	P0-P3	
349	SN1220	YES	4	NO	NO	10	P0-P3	
350	SN1230	YES	4	NO	NO	1	P0-P3	

Table 8.2-1(8/9) Nuclide list of the Public Library (MSRACLIB_J40)

NO	NUCLIDE	FAST DATA	F-TABLE NTEMP	MCROSS LIBRARY	THERMAL F-TAB	NTEMP	THERMAL KERNEL	REMARKS
351	SN1240	YES	4	NO	NO	10	P0-P3	
352	SN1260	YES	4	NO	NO	1	P0-P3	
353	SR0840	YES	4	NO	NO	1	P0-P3	
354	SR0860	YES	4	NO	NO	1	P0-P3	
355	SR0870	YES	4	NO	YES	10	P0-P3	
356	SR0880	YES	4	NO	NO	1	P0-P3	
357	SR0890	YES	4	NO	NO	1	P0-P3	
358	SR0900	YES	4	NO	NO	1	P0-P3	
359	TA1810	YES	4	YES	YES	10	P0-P3	
360	TB1590	YES	4	YES	YES	10	P0-P3	
361	TB1600	YES	4	NO	YES	10	P0-P3	
362	TC0990	YES	4	YES	NO	10	P0-P3	
363	TE1200	YES	4	NO	NO	1	P0-P3	
364	TE1220	YES	4	NO	NO	1	P0-P3	
365	TE1230	YES	4	NO	YES	10	P0-P3	
366	TE1240	YES	4	NO	NO	1	P0-P3	
367	TE1250	YES	4	NO	NO	1	P0-P3	
368	TE1260	YES	4	NO	NO	1	P0-P3	
369	TE1271	YES	4	NO	NO	1	P0-P3	
370	TE1280	YES	4	NO	NO	1	P0-P3	
371	TE1291	YES	4	NO	NO	1	P0-P3	
372	TE1300	YES	4	NO	NO	1	P0-P3	
373	TE1320	YES	4	NO	NO	1	P0-P3	
374	TH2270	YES	1	NO	NO	1	P0-P3	
375	TH2280	YES	4	NO	NO	1	P0-P3	
376	TH2290	YES	4	NO	NO	1	P0-P3	
377	TH2300	YES	4	YES	YES	10	P0-P3	
378	TH2310	YES	1	NO	NO	1	P0-P3	
379	TH2320	YES	4	YES	NO	10	P0-P3	
380	TH2330	YES	1	NO	NO	1	P0-P3	
381	TH2340	YES	1	NO	NO	1	P0-P3	
382	TI0000	YES	4	NO	NO	10	P0-P3	comp
383	TI0460	YES	4	NO	NO	10	P0-P3	
384	TI0470	YES	4	NO	NO	10	P0-P3	
385	TI0480	YES	4	NO	NO	10	P0-P3	
386	TI0490	YES	4	NO	NO	10	P0-P3	
387	TI0500	YES	4	NO	NO	10	P0-P3	
388	TM1690	YES	4	NO	YES	10	P0-P3	
389	U02300	YES	1	NO	NO	1	P0-P3	
390	U02310	YES	1	NO	NO	1	P0-P3	
391	U02320	YES	4	YES	NO	10	P0-P3	
392	U02330	YES	4	YES	YES	10	P0-P3	
393	U02340	YES	4	YES	NO	10	P0-P3	
394	U02350	YES	4	YES	YES	10	P0-P3	
395	U02360	YES	4	YES	NO	10	P0-P3	
396	U02370	YES	4	YES	YES	10	P0-P3	
397	U02380	YES	4	YES	NO	10	P0-P3	
398	V00500	YES	4	NO	NO	10	P0-P3	
399	V00510	YES	4	NO	NO	10	P0-P3	
400	W00000	YES	4	NO	NO	10	P0-P3	comp

Table 8.2-1(9/9) Nuclide list of the Public Library (MSRACLIB_J40)

NO	NUCLIDE	FAST DATA	F-TABLE NTEMP	MCROSS LIBRARY	THERMAL F-TAB	NTEMP	THERMAL KERNEL	REMARKS
401	W01800	YES	4	NO	NO	10	P0-P3	
402	W01820	YES	4	YES	YES	10	P0-P3	
403	W01830	YES	4	YES	NO	10	P0-P3	
404	W01840	YES	4	YES	NO	10	P0-P3	
405	W01860	YES	4	YES	NO	10	P0-P3	
406	XE1240	YES	4	NO	YES	10	P0-P3	
407	XE1260	YES	4	NO	NO	1	P0-P3	
408	XE1280	YES	4	NO	NO	1	P0-P3	
409	XE1290	YES	4	NO	NO	1	P0-P3	
410	XE1300	YES	4	NO	NO	1	P0-P3	
411	XE1310	YES	4	YES	NO	10	P0-P3	
412	XE1320	YES	4	NO	NO	10	P0-P3	
413	XE1330	YES	4	NO	NO	10	P0-P3	
414	XE1340	YES	4	NO	NO	1	P0-P3	
415	XE1350	YES	4	NO	NO	10	P0-P3	
416	XE1360	YES	4	NO	NO	10	P0-P3	
417	Y00890	YES	4	NO	NO	10	P0-P3	
418	Y00900	YES	4	NO	NO	1	P0-P3	
419	Y00910	YES	4	NO	NO	1	P0-P3	
420	YB1680	YES	4	NO	YES	10	P0-P3	
421	YB1700	YES	4	NO	NO	10	P0-P3	
422	YB1710	YES	4	NO	NO	10	P0-P3	
423	YB1720	YES	4	NO	NO	10	P0-P3	
424	YB1730	YES	4	NO	NO	10	P0-P3	
425	YB1740	YES	4	NO	NO	10	P0-P3	
426	YB1760	YES	4	NO	NO	10	P0-P3	
427	ZN0640	YES	4	NO	NO	10	P0-P3	
428	ZN0650	YES	4	NO	NO	10	P0-P3	
429	ZN0660	YES	4	NO	NO	10	P0-P3	
430	ZN0670	YES	4	NO	NO	10	P0-P3	
431	ZN0680	YES	4	NO	NO	10	P0-P3	
432	ZN0700	YES	4	NO	NO	10	P0-P3	
433	ZR0000	YES	4	NO	NO	10	P0-P3	comp
434	ZR0900	YES	4	NO	NO	10	P0-P3	
435	ZR090Z	YES	4	NO	NO	8	P0-P3	
436	ZR0910	YES	4	NO	NO	10	P0-P3	
437	ZR091Z	YES	4	NO	NO	8	P0-P3	
438	ZR0920	YES	4	NO	NO	10	P0-P3	
439	ZR092Z	YES	4	NO	NO	8	P0-P3	
440	ZR0930	YES	4	NO	NO	10	P0-P3	
441	ZR093Z	YES	4	NO	NO	8	P0-P3	
442	ZR0940	YES	4	NO	NO	10	P0-P3	
443	ZR094Z	YES	4	NO	NO	8	P0-P3	
444	ZR0950	YES	4	NO	NO	10	P0-P3	
445	ZR095Z	YES	4	NO	NO	8	P0-P3	
446	ZR0960	YES	4	NO	NO	10	P0-P3	
447	ZR096Z	YES	4	NO	NO	8	P0-P3	
448	ZZ0500	NO	0	NO	NO	1	P0,P1	pseudo
449	ZZ1110	NO	0	NO	NO	1	P0,P1	pseudo

pseudo: pseudo fission products described in burn-up chain models

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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 組立単位	
	名称	記号
面積	平方メートル	m^2
体積	立方メートル	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	キログラム毎立方メートル	kg/m^3
面積密度	キログラム毎平方メートル	kg/m^2
比体積	立方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
量濃度(a)，濃度	モル毎立方メートル	mol/m^3
質量濃度	キログラム毎立方メートル	kg/m^3
輝度	カンデラ毎平方メートル	cd/m^2
屈折率(b)	（数字の） 1	1
比透磁率(b)	（数字の） 1	1

(a) 量濃度（amount concentration）は臨床化学の分野では物質濃度（substance concentration）ともよばれる。
(b) これらは無次元量あるいは次元１をもつ量であるが、そのことを表す単位記号である数字の１は通常は表記しない。

表３．固有の名称と記号で表される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(b)	rad	1 (b)	m/m
立体角	ステラジアン(b)	sr(c)	1 (b)	m^2/m^2
周波数	ヘルツ(d)	Hz		s^{-1}
力	ニュートン	N		m kg s^{-2}
圧力，応力	パスカル	Pa	N/m^2	m^{-1} kg s^{-2}
エネルギー，仕事，熱量	ジュール	J	N m	m^2 kg s^{-2}
仕事率，工率，放射束	ワット	W	J/s	m^2 kg 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 kg s^{-3} A^{-1}
静電容量	ファラド	F	C/V	m^{-2} kg^{-1} s^4 A^2
電気抵抗	オーム	Ω	V/A	m^2 kg s^{-3} A^{-2}
コンダクタンス	ジーメンス	S	A/V	m^{-2} kg^{-1} s^3 A^2
磁束	ウエーバ	Wb	Vs	m^2 kg s^{-2} A^{-1}
磁束密度	テスラ	T	Wb/m^2	kg s^{-2} A^{-1}
インダクタンス	ヘンリー	H	Wb/A	m^2 kg s^{-2} A^{-2}
セルシウス温度	セルシウス度(e)	℃		K
光束	ルーメン	lm	cd sr(c)	cd
照度	ルクス	lx	lm/m^2	m^{-2} cd
放射性核種の放射能(f)	ベクレル(d)	Bq		s^{-1}
吸収線量，比エネルギー分与，カーマ	グレイ	Gy	J/kg	m^2 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	シーベルト(g)	Sv	J/kg	m^2 s^{-2}
酸素活性	カタール	kat		s^{-1} mol

(a)SI接頭語は固有の名称と記号を持つ組立単位と組み合わせても使用できる。しかし接頭語を付した単位はもはやコヒーレントではない。
(b)ラジアンとステラジアンは数字の１に対する単位の特別な名称で、量についての情報をつたえるために使われる。実際には、使用する時には記号radおよびsrが用いられるが、習慣として組立単位としての記号である数字の１は明示されない。
(c)測光学ではステラジアンという名称と記号srを単位の表し方の中に、そのまま維持している。
(d)ヘルツは周期現象についてのみ、ベクレルは放射性核種の統計的過程についてのみ使用される。
(e)セルシウス度はケルビンの特別な名称で、セルシウス温度を表すために使用される。セルシウス度とケルビンの単位の大きさは同一である。したがって、温度差や温度間隔を表す数値はどちらの単位で表しても同じである。
(f)放射性核種の放射能（activity referred to a radionuclide）は、しばしば誤った用語で"radioactivity"と記される。
(g)単位シーベルト（PV,2002,70,205）についてはCIPM勧告2（CI-2002）を参照。

表４．単位の中に固有の名称と記号を含む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} kg s^{-1}
力のモーメント	ニュートンメートル	N m	m^2 kg s^{-2}
表面張力	ニュートン毎メートル	N/m	kg s^{-2}
角速度	ラジアン毎秒	rad/s	m m^{-1} s^{-1}=s^{-1}
角加速度	ラジアン毎秒毎秒	rad/s^2	m m^{-1} 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 s^{-3}
熱容量，エントロピー	ジュール毎ケルビン	J/K	m^2 kg s^{-2} K^{-1}
比熱容量，比エントロピー	ジュール毎キログラム毎ケルビン	J/(kg K)	m^2 s^{-2} K^{-1}
比エネルギー	ジュール毎キログラム	J/kg	m^2 s^{-2}
熱伝導率	ワット毎メートル毎ケルビン	W/(m K)	m kg s^{-3} 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} kg s^{-2}
電界の強さ	ボルト毎メートル	V/m	m kg s^{-3} A^{-1}
電荷密度	クーロン毎立方メートル	C/m^3	m^{-3} s A
表面電荷	クーロン毎平方メートル	C/m^2	m^{-2} s A
電束密度，電気変位	クーロン毎平方メートル	C/m^2	m^{-2} s A
誘電率	ファラド毎メートル	F/m	m^{-3} kg^{-1} s^4 A^2
透磁率	ヘンリー毎メートル	H/m	m kg s^{-2} A^{-2}
モルエネルギー	ジュール毎モル	J/mol	m^2 kg s^{-2} 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 kg s^{-2} K^{-1} 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} s A
吸収線量率	グレイ毎秒	Gy/s	m^2 s^{-3}
放射強度	ワット毎ステラジアン	W/sr	m^4 m^{-2} kg s^{-3}=m^2 kg s^{-3}
放射輝度	ワット毎平方メートル毎ステラジアン	$W/(m^2$ sr)	m^2 m^{-2} kg s^{-3}=kg s^{-3}
酵素活性濃度	カタール毎立方メートル	kat/m^3	m^{-3} s^{-1} mol

表５．SI 接頭語

乗数	名称	記号	乗数	名称	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６．SIに属さないが、SIと併用される単位

名称	記号	SI 単位による値
分	min	1 min=60 s
時	h	1 h =60 min=3600 s
日	d	1 d=24 h=86 400 s
度	°	1°=(π/180) rad
分	’	1’=(1/60)°=(π/10 800) rad
秒	”	1”=(1/60)’=(π/648 000) rad
ヘクタール	ha	1 ha=1 hm^2=10^4m^2
リットル	L, l	1 L=1 l=1 dm^3=10^3cm^3=$10^{-3}m^3$
トン	t	1 t=10^3 kg

表７．SIに属さないが、SIと併用される単位で、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 単位で表される数値
電子ボルト	eV	1 eV=1.602 176 53(14)×10^{-19}J
ダルトン	Da	1 Da=1.660 538 86(28)×10^{-27}kg
統一原子質量単位	u	1 u=1 Da
天文単位	ua	1 ua=1.495 978 706 91(6)×10^{11}m

表８．SIに属さないが、SIと併用されるその他の単位

名称	記号	SI 単位で表される数値
バール	bar	1 bar=0.1MPa=100 kPa=10^5Pa
水銀柱ミリメートル	mmHg	1 mmHg≈133.322Pa
オングストローム	Å	1 Å=0.1nm=100pm=10^{-10}m
海里	M	1 M=1852m
バーン	b	1 b=100fm^2=(10^{-12}cm)^2=$10^{-28}m^2$
ノット	kn	1 kn=(1852/3600)m/s
ネーパ	Np	SI単位との数値的な関係は、対数量の定義に依存。
ベル	B	
デシベル	dB	

表９．固有の名称をもつCGS組立単位

名称	記号	SI 単位で表される数値
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 =1cm^2 s^{-1}=$10^{-4}m^2$ s^{-1}
スチルブ	sb	1 sb =1cd cm^{-2}=10^4cd m^{-2}
フォト	ph	1 ph=1cd sr cm^{-2} =10^4lx
ガル	Gal	1 Gal =1cm s^{-2}=$10^{-2}ms^{-2}$
マクスウエル	Mx	1 Mx = 1G cm^2=10^{-8}Wb
ガウス	G	1 G =1Mx cm^{-2} =10^{-4}T
エルステッド(a)	Oe	1 Oe ≙ (10^3/4π)A m^{-1}

(a) ３元系のCGS単位系とSIでは直接比較できないため、等号「 ≙ 」は対応関係を示すものである。

表10．SIに属さないその他の単位の例

名称	記号	SI 単位で表される数値
キュリー	Ci	1 Ci=3.7×10^{10}Bq
レントゲン	R	1 R = 2.58×10^{-4}C/kg
ラド	rad	1 rad=1cGy=10^{-2}Gy
レム	rem	1 rem=1 cSv=10^{-2}Sv
ガンマ	γ	1 γ=1 nT=10^{-9}T
フェルミ		1 フェルミ=1 fm=10^{-15}m
メートル系カラット		1 メートル系カラット= 0.2 g = 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	1 cal=4.1858J（「15℃」カロリー），4.1868J（「IT」カロリー），4.184J（「熱化学」カロリー）
ミクロン	μ	1 μ =1μm=10^{-6}m

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