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SIMCRI : A SIMPLE COMPUTER CODE FOR CALCULATING NUCLEAR CRITICALITY PARAMETERS

March 1986

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This is a user's manual for a simple criticality calculation code SIMCRI. The code has been developed to facilitate criticality calculation on a single unit of nuclear fuel. SIMCRI makes an extensive survey with a little computing time. Cross section library MGCL for SIMCRI is the same one for the Monte Carlo criticality code KENO-IV; it is, therefore, easy to compare the results of the two codes. SIMCRI solves eigenvalue problems and fixed source problems based on the one space point B_1 equation. The results include infinite and effective multiplication factor, critical buckling, migration area, diffusion coefficient and so on. SIMCRI is comprised in the criticality safety evaluation code system JACS.

Keywords: SIMCRI, Computer Code, Criticality Safety, JACS, MGCL, KENO-IV

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臨界因子簡易計算コードSIMCRI

日本原子力研究所東海研究所燃料安全工学部
中丸　昇一*・菅原　伸彦**・内藤　俶孝
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本報告書は臨界因子計算コードSIMCRIの使用手引である。このコードは，核燃料物質の単一ユニットに関する臨界計算を簡便に行なうことを目的として開発した。SIMCRIコードは広汎なパラメータ・サーベイを迅速に実行する。また，モンテカルロ法臨界計算コードKENO-Ⅳと同一の断面積ライブラリを用いるため，両コードの結果を容易に比較できる。SIMCRIコードは一点近似のB_1方程式により固有値問題および固定中性子源問題を解き，中性子の無限増倍係数・実効増倍係数，臨界バックリング，中性子移動面積，拡散定数，等を算出する。SIMCRIコードは臨界安全性評価コード・システムJACSに収納されている。

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1. Introduction

For the criticality analysis of nuclear fuel facilities, Japan Atomic Energy Research Institute has developed a criticality safety evaluation code system JACS[1),2)]. One of the basic codes in JACS is a Monte Carlo criticality code KENO-IV[3)]. The Monte Carlo code affords detailed analysis, however, it consumes a lot of computing time, and it is inappropriate for quick surveys.

We have therefore developed the SIMCRI code to calculate quickly and effectively for the single unit system where the fuel is homogeneous and the geometry is rather simple. SIMCRI solves eigenvalue problems and fixed source problems based on the one space point B_1 neutron transport equation. SIMCRI uses the cross section library which is applicable to KENO-IV without any conversion. It is therefore easy to evaluate the results of SIMCRI by KENO-IV, or to use SIMCRI in the preparatory calculation for KENO-IV. The cross section library is generated by the MAIL code from the multigroup cross section library MGCL. MAIL and MGCL are included in JACS.

Chapter 2 is the computer code description with the information about data files, subroutines and error messages. Chapter 3 describes numerical methods employed. Chapter 4 explaines about input data. For a sample input and output, one can refer to chapter 5. The appendix presents a brief manual of the MAIL code which makes an effective macroscopic cross section libary for SIMCRI.

2. Code Description

(1) Machine Available

FACOM - M380.

(2) Operating Systems

OS-IV/F4 (FACOM).

(3) Machine Requirements

The FACOM version requires 870 KB.

(4) Programming Languages

FORTRAN 77

(JIS C 6201-1982, ISO 1539-1980, ANSI X3.9-1978).

(5) Restrictions on the Problem

The minimum and maximum number of energy groups are 3 and 140 respectively.

(6) Typical Running Time

To obtain a k_{eff} of a homogeneous UO_2-H_2O system with 137 energy groups, it takes about one second by the FACOM M-380.

(7) Related Programs

The MAIL code. It generates effective mecroscopic cross sections.

(8) Other Programing or Operating Information

Subroutine SPECTR and subroutine CYLREF uses utilities built in the JAERI computing system.

(9) Data Files

Input and output files are shown in Figure 1 and Table 1. GDFILE is a graphic file. If there is no graphic output, the definition of GDFILE can be omitted, that is, GDFILE may be left unallocated or unopened. The name GDFILE is used in the JAERI computing system, and may be different elsewhere. A file of unit 16 is used to store summaries of the results. Even if the unit 16 is not used or defined, execution proceeds without an error.

(10) Subroutines and Common Blocks

Tree structure of the subroutine is shown in Figure 2. Table 2 gives the subroutine summary. Figure 3 shows a sample JCL to make an overlay structure of subroutines. For common blocks, please refer to Table 3.

(11) Error Messages

Figure 4 shows a general format of a message. The meaning of each message is fully explained in its body.

Table 1 Attributes of the data file.

Logical Unit No.	File Organization	Record Format	Record Length [byte]	Disposition	Usage
5	PS	FB	80	SHR	input cards
6	PS	FBA	137	OLD	output lists
41	PS	VBS	—— *)	SHR	cross sections
16	PS	FB	120	OLD	summary records

*) Required length is varied with a number of energy groups.
It is convenient to choose some maximum record length permitted.

Table 2 Alphabetical subroutine summary.

Name	Function
BCALC	To calculate a buckling corresponding to a specified K_{eff}.
BJINIT	To set initial values of a buckling, neutron flux and current.
(Block data)	To set values for initial neutron flux, fixed source spectrum and constants of reflectors
COLLAP	To collapse reaction rates, etc. to two or three energy groups.
CRITQ	To calculate various quantities.
CYLEPS	To obtain a calculational error about reflector savings of a cylinder, employing a Bessel function package.
CYLREF	To calculate reflector savings of a cylinder by the CYLEPS routine.
INPUT	To control subroutines for data reading.
KCALC	To calculate a K_{eff} corresponding to a specified buckling.
KEFF	To obtain a k_{eff}, neutron flux and current.
KINF	To obtain a k_{∞} and neutron flux of on infinite core.
MACRO	To process macroscopic cross sections prior to the calculation on the B_1 equation.
MAILIN	To read cross section library prepared by the MAIL code.
MAIN	To manage execution.
OUTPUT	To print out results to files of unit 6 and 16.
RECVR	To retrieve summary records.
SIMIN	To read and check input cards.

Table 2 (continued)

Name	Function
SPECTR	To plot graphs using vector-scan-type plotting utilities.
SPHREF	To calculate reflector savings of a sphere.
TEST 1	To print out input cards.
TEST 2	To edit input parameters in each case.
TEST 3	To list comments for all regions or nuclides in a cross section library.
TEST 4	To print out a neutron balance table on an infinite core.
TEST 5	To print out a neutron balance table on a finite core.

Table 3 Common blocks.

Name	Type	Size
SCOM1	Character	148 B
SCOM2	Number	42 KB

Note: No blank common used. All numbers are assigned as single precision.

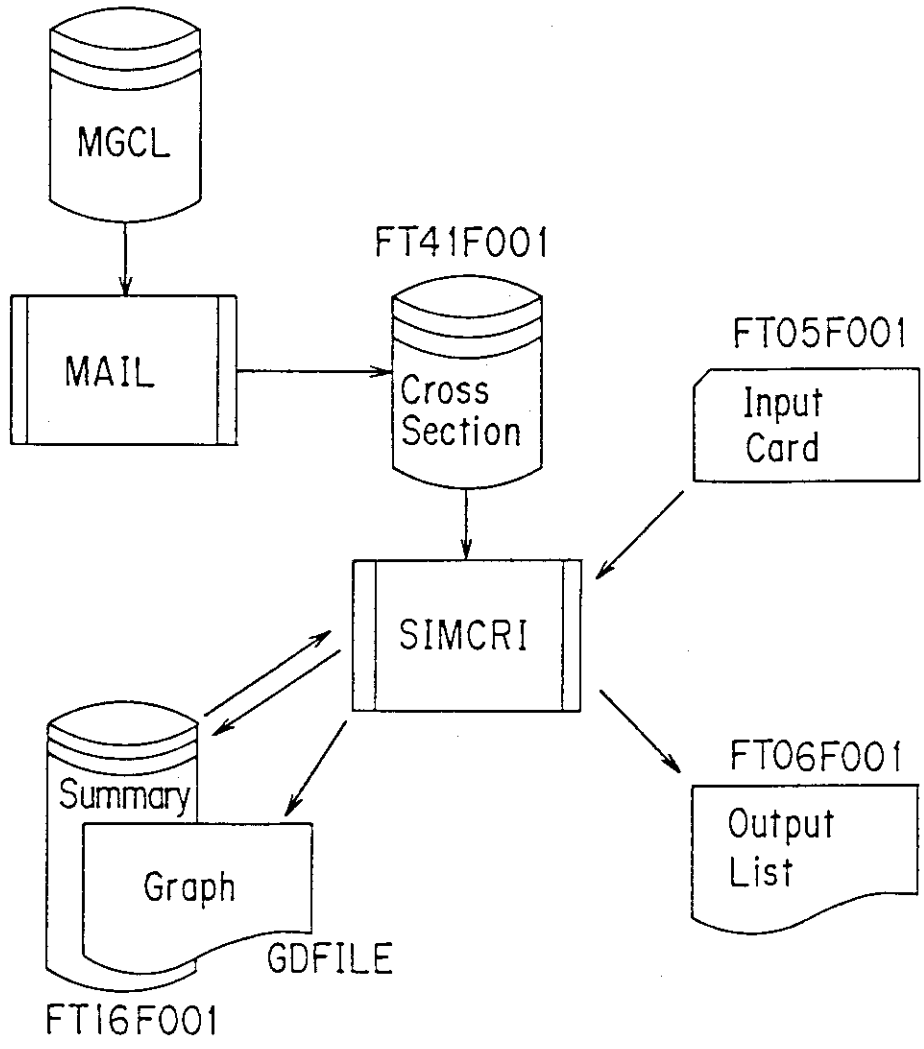

Fig. 1 Data files for SIMCRI.
MGCL is a multigroup constants library.
MAIL is a code that generates effective macroscopic cross sections.

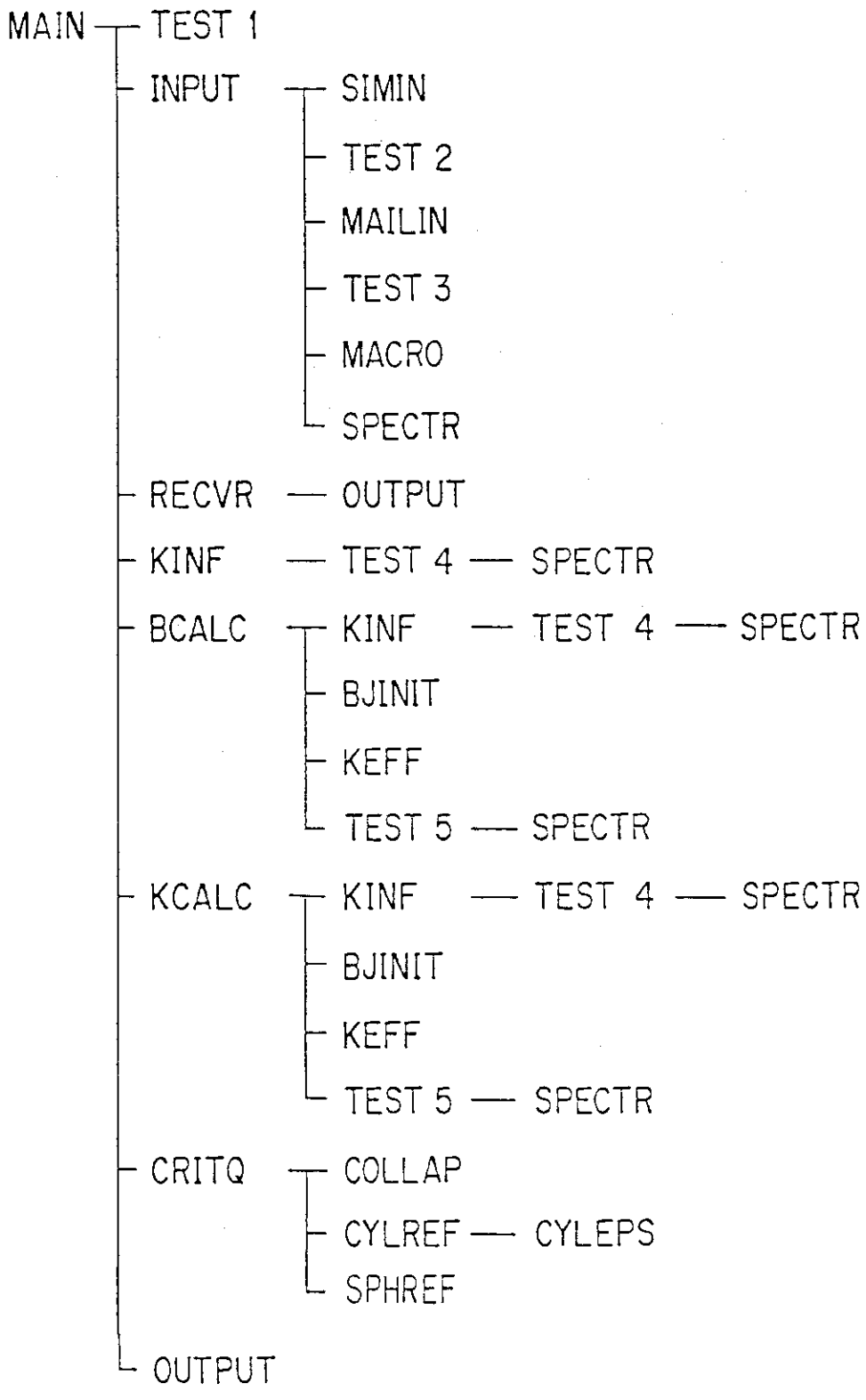

Fig. 2 Tree structure of SIMCRI.

```
//JCLG JOB
//JCLG EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER 03327611,SH.NAKAMARU,0943.01
   T.0 W.0 I.2 C.1 OPN
  OPTP PASSWORD=S,NOTIFY=J7611
//COMPILE EXEC FORT77,SO='J7611.SIMCRI',A='ELM(*),OPT(3),ALC'
//*****************************************************************
//*  SYS9.GGS.LOAD  - GRAPHIC LIBRARY  FOR ALL DEVICES             *
//*  SYS9.PNL.LOAD  - GRAPHIC LIBRARY  FOR LASER PRINTER           *
//*  SYS9.SSL.LOAD  - SCIENTIFIC SUBROUTINE LIBRARY NO.1           *
//*  SYS9.SSL2.LOAD - SCIENTIFIC SUBROUTINE LIBRARY NO.2           *
//*****************************************************************
//LINK        EXEC PGM=JQAL,REGION=1024K,COND=(4,LT),PARM='OVLY,XCAL'
//SUBSYS        DD SUBSYS=(VPCS,'SIZE=(OOOOOK,OOM)')
//SYSLIB        DD   DSN=SYS9.GGS.LOAD,DISP=SHR
//               DD   DSN=SYS9.PNL.LOAD,DISP=SHR
//               DD   DSN=SYS9.SSL.LOAD,DISP=SHR
//               DD   DSN=SYS9.SSL2.LOAD,DISP=SHR
//               DD   DSN=SYS1.FORTLIB,DISP=SHR
//SYSPRINT      DD   SYSOUT=*
//SYSTERM       DD   SYSOUT=*
//SYSUT1        DD   UNIT=VIO,SPACE=(TRK,(30,10))
//SYSLMOD       DD   DSN=J7611.SIMCRI.LOAD,DISP=OLD
//SYSLIN        DD   DSN=&&OBJ,DISP=(OLD,DELETE)
//               DD   *
 INSERT  MAIN
 OVERLAY NODE1
  INSERT  BCALC
  OVERLAY NODE1
  INSERT  KCALC
 OVERLAY NODE2(REGION)
  INSERT  TEST4
  OVERLAY NODE2
  INSERT  CYLEPS
 OVERLAY NODE3(REGION)
  INSERT  TEST1
  OVERLAY NODE3
  INSERT  INPUT
  OVERLAY NODE3
  INSERT  RECVR
  OVERLAY NODE3
  INSERT  KINF
  OVERLAY NODE3
  INSERT  BJINIT
  OVERLAY NODE3
  INSERT  KEFF
  OVERLAY NODE3
  INSERT  TEST5
  OVERLAY NODE3
  INSERT  CRITQ
```

Fig. 3 Sample JCL to make an overlay structured load module.

(to be continued)

```
 OVERLAY NODE4(REGION)
  INSERT  SIMIN
  OVERLAY NODE4
  INSERT  TEST2
  OVERLAY NODE4
  INSERT  MAILIN
  OVERLAY NODE4
  INSERT  TEST3
  OVERLAY NODE4
  INSERT  MACRO
  OVERLAY NODE4
  INSERT  OUTPUT
  OVERLAY NODE4
  INSERT  SPECTR
  OVERLAY NODE4
  INSERT  COLLAP
  OVERLAY NODE4
  INSERT  SPHREF
  OVERLAY NODE4
  INSERT  CYLEPS
 ENTRY MAIN
 NAME  OVERLAY(R)
/*
++
//
```

Fig. 3 (continued)

* aaa * bbbbbb : (body of the message)

aaa = MSG A simple message indicating no error.

= WRN A warning. The execution continues but some unusual treatment may be taken.

= ERR A severe error has been occurred. and the execution will stop.

bbbbbb = A name of the subroutine which issued the message.

Fig. 4 General format of a message.

3. Method of Solution

3.1 Basic Equation

The basic equation is the one space point B_1 neutron transport equation:

$$J(E) + \Sigma_t(E)\psi(E) = \int_0^\infty dE' \Sigma_s^{(0)}(E' \to E)\psi(E') + S(E) , \quad (1)$$

$$-B^2\psi(E) + 3\gamma(E)\Sigma_t(E)J(E) = \int_0^\infty dE' \Sigma_s^{(1)}(E' \to E)J(E') , \quad (2)$$

where

$\psi(E)$: neutron flux,
$J(E)$: neutron current,
$\Sigma_t(E)$: total cross cection,
$\Sigma_s^{(0)}(E' \to E)$: P_0 component of scattering cross section,
$\Sigma_s^{(1)}(E' \to E)$: P_1 component of scattreing cross section,
$S(E)$: neutron source,
B^2 : buckling,
E : energy,

$$\gamma(E) = \frac{a^2(E)}{3} \cdot \frac{\tan^{-1}a(E)}{a(E) - \tan^{-1}a(E)} , \quad (3)$$

$$a(E) = \frac{B}{\Sigma_t(E)} . \quad (4)$$

The cross sections are all macroscopic. The neutron source S(E) is

$$S(E) = \frac{\chi(E)}{K_{eff}} \int_0^\infty dE' \nu\Sigma_f(E')\psi(E') + \frac{S_0(E)}{1 - K_{eff}} , \quad (5)$$

where

$\chi(E)$: fission spectrum, $\int_0^\infty dE\chi(E) = 1$,
ν : number of neutrons emmitted per fission,
Σ_f : fission cross section,
K_{eff} : effective multiplication factor,
$S_0(E)$: fixed neutron source.

If the fixed neutron source $S_0(E)$ equals zero, the basic equations (1) and (2) define an eigenvalue problem. The eigenvalue is K_{eff} or B^2. When a K_{eff} is specified, the code calculates a corresponding B^2, and vice versa. If $S_0(E)$ is not zero, the code solves a fixed source problem with a specified B^2 or K_{eff}, where the multiplication factor K_{eff} must be less than 1.0.

The type of the problem which may be well solved by the B_1 equations (1) and (2) are those in which the fuel is homogeneous and the core dimension is large enough compared with the neutron mean free path, and the effect of the higher order anisotropic scattering is insignificant.

3.2 Neutron Flux and Current

In order to obtain the neutron flux ψ and the neutron current J, the equations (1) and (2) are solved by the multigroup power iteration method with successive over relaxation (SOR).

If there is no fixed source, ψ and J are normalized so that the equation (5) takes the form

$$S(E) = \chi(E) \ . \tag{6}$$

Then the equations (1) and (2) are solved with a buckling given by the user. The calculational flow is shown in Figure 5. In the iteration for the thermal neutron flux or thermal neutron current, a relaxation or acceleration factor is constant. On the contrary, in the outer iteration, the acceleration factor is altered every iteration, such as

$$\psi_{i+1}(E) = \psi_i(E) + \alpha_i(\psi'_{i+1}(E) - \psi_i(E)) \ ,$$

where

$\psi_{i+1}(E)$: $\psi(E)$ to be used in the (i+1)-th iferation,
$\psi'_{i+1}(E)$: $\psi(E)$ resulted from the i-th iteration,
$\psi_i(E)$: $\psi(E)$ used in the i-th iteration,
α_i : acceleration factor in the i-th iteration,

$$\alpha_i = \min\left(\alpha_0, \frac{-1}{2}\left(1 + \frac{B^2 - c_i}{B^2 + c_i}\right)\right) ,$$

$$c_i^{-1} = 3\,[\Sigma_t(E_0) - \Sigma_s^{(0)}(E_0 \to E_0')] \times [\gamma_i(E_0)\Sigma_t(E_0) - \Sigma_s^{(1)}(E_0 \to E_0')] ,$$

α_0 : maximum value of α_i, specified by the user,

E_0 : the highest energy group

γ_i : γ in the i-th iteration, γ is defined by the equation (3).

An algorithm for getting the neutron flux of an infinite core is similar to the above one, except that the calculations for the neutron current and outer iteration are omitted.

When a K_{eff} is specified, the equations (1) and (2) are solved repeatedly with different bucklings until the resulted K_{eff} becomes the specified one. The error in the buckling guess, ΔB^2, is estimated as follows.

$$B^2 = \frac{K_\infty / K_{eff} - 1}{M^2} ,$$

$$\Delta B^2 = \frac{K_\infty}{M^2} \Delta\left(\frac{1}{K_{eff}}\right) ,$$

where M^2 is the migration area.

In case of a fixed source problem, the code uses the equation (5) instead of (6). The fixed neutron source $S_0(E)$ may be explicitly given by the input cards, however, SIMCRI contains three types of fission spectrum for the convenience:

(a) LWR-type fission spectrum,

(b) ^{235}U fission spectrum,

(c) ^{239}Pu fission spectrum.

The three spectra are illustrated in Figure 6, 7 and 8. And Table 4 gives the spectra multiplied by the energy width of each energy group. Their energy structure is shown in Table A-3.

Table 4 Fission spectra in SIMCRI, multiplied by the energy width.

Group No.	LWR-type	^{235}U	^{239}Pu
1	5.0000E-03	4.9367E-05	8.0060E-05
2	7.2000E-03	1.5945E-04	2.4238E-04
3	1.1300E-02	4.3978E-04	6.3129E-04
4	1.6700E-02	1.0548E-03	1.4393E-03
5	2.3000E-02	2.2355E-03	2.9167E-03
6	3.3000E-02	4.2473E-03	5.3261E-03
7	5.0000E-02	7.3257E-03	8.8699E-03
8	7.4000E-02	1.1600E-02	1.3616E-02
9	9.7000E-02	1.7030E-02	1.9450E-02
10	1.1500E-01	2.3387E-02	2.6072E-02
11	1.3500E-01	3.0276E-02	3.3038E-02
12	1.6500E-01	3.7202E-02	3.9836E-02
13	1.9000E-01	4.3655E-02	4.5972E-02
14	2.1000E-01	4.9184E-02	5.1037E-02
15	2.3500E-01	5.3457E-02	5.4753E-02
16	2.6000E-01	5.6286E-02	5.6992E-02
17	2.5000E-01	5.7626E-02	5.7760E-02
18	2.3800E-01	5.7557E-02	5.7175E-02
19	2.2100E-01	5.6244E-02	5.5431E-02
20	2.1000E-01	5.3912E-02	5.2762E-02
21	1.9500E-01	5.0803E-02	4.9414E-02
22	1.8000E-01	4.7160E-02	4.5621E-02
23	1.6800E-01	4.3200E-02	4.1590E-02
24	1.5600E-01	3.9112E-02	3.7495E-02
25	1.4400E-01	3.5046E-02	3.3472E-02
26	1.2500E-01	3.1118E-02	2.9622E-02
27	1.1000E-01	2.7408E-02	2.6015E-02
28	9.4000E-02	2.3970E-02	2.2693E-02
29	8.1000E-02	2.0831E-02	1.9677E-02
30	7.1000E-02	1.8004E-02	1.6972E-02
31	5.9000E-02	1.5484E-02	1.4571E-02
32	5.2000E-02	1.3260E-02	1.2458E-02
33	4.3000E-02	1.1312E-02	1.0613E-02
34	3.6000E-02	9.6179E-03	9.0129E-03
35	3.0000E-02	8.1532E-03	7.6322E-03
36	2.5500E-02	6.8935E-03	6.4469E-03
37	2.2500E-02	5.8150E-03	5.4337E-03
38	1.8800E-02	4.8952E-03	4.5708E-03
39	1.5900E-02	4.1135E-03	3.8384E-03
40	1.3200E-02	3.4511E-03	3.2185E-03
41	1.1400E-02	2.8914E-03	2.6951E-03
42	1.0000E-02	2.4194E-03	2.2542E-03
43	8.1000E-03	2.0223E-03	1.8834E-03
44	6.8000E-03	1.6887E-03	1.5722E-03
45	5.6000E-03	1.4090E-03	1.3113E-03
46	4.7000E-03	1.1747E-03	1.0930E-03
47	3.8000E-03	9.7868E-04	9.1042E-04
48	3.3000E-03	8.1493E-04	7.5792E-04
49	2.7500E-03	6.7822E-04	6.3066E-04
50	2.2000E-03	5.6419E-04	5.2454E-04

(to be continued)

Table 4 (continued)

Group No.	LWR-type	^{235}U	^{239}Pu
51	1.8500E-03	4.6914E-04	4.3611E-04
52	1.5500E-03	3.8997E-04	3.6246E-04
53	1.3500E-03	3.2406E-04	3.0117E-04
54	1.0500E-03	2.6921E-04	2.5017E-04
55	8.6000E-04	2.2359E-04	2.0776E-04
56	7.1000E-04	1.8566E-04	1.7250E-04
57	6.0000E-04	2.8208E-04	2.6206E-04
58	4.2000E-04	1.9431E-04	1.8050E-04
59	2.8000E-04	1.3378E-04	1.2426E-04
60	2.0000E-04	9.2068E-05	8.5512E-05
61	1.3500E-04	6.3344E-05	5.8830E-05
62	9.5000E-05	4.3571E-05	4.0464E-05
63	6.4000E-05	2.9964E-05	2.7827E-05
64	4.3000E-05	2.0604E-05	1.9134E-05
65	3.0000E-05	1.4166E-05	1.3155E-05
66	0.0	9.7386E-06	9.0434E-06
67	0.0	6.6945E-06	6.2165E-06
68	0.0	4.6016E-06	4.2730E-06
69	0.0	3.1628E-06	2.9369E-06
70	0.0	2.1737E-06	2.0185E-06
71	0.0	1.4939E-06	1.3872E-06
72	0.0	1.0266E-06	9.5326E-07
73	0.0	7.0545E-07	6.5503E-07
74	0.0	4.8472E-07	4.5007E-07
75	0.0	3.3302E-07	3.0921E-07
76	0.0	2.2877E-07	2.1240E-07
77	0.0	1.5713E-07	1.4589E-07
78	0.0	1.0790E-07	1.0018E-07
79	0.0	7.4081E-08	6.8773E-08
80	0.0	5.0845E-08	4.7198E-08
81	0.0	3.4882E-08	3.2378E-08
82	0.0	2.3919E-08	2.2199E-08
83	0.0	1.6390E-08	1.5209E-08
84	0.0	1.1222E-08	1.0411E-08
85	0.0	7.6746E-09	7.1183E-09
86	0.0	5.2410E-09	4.8596E-09
87	0.0	3.5725E-09	3.3110E-09
88	0.0	2.4291E-09	2.2501E-09
89	0.0	1.6464E-09	1.5239E-09
90	0.0	1.1112E-09	1.0275E-09
91	0.0	7.4570E-10	6.8862E-10
92	0.0	4.9663E-10	4.5779E-10
93	0.0	1.8093E-10	1.6647E-10
94	0.0	1.4639E-10	1.3449E-10
95	0.0	1.1797E-10	1.0819E-10
96	0.0	9.4621E-11	8.6590E-11
97	0.0	7.5455E-11	6.8869E-11
98	0.0	5.9746E-11	5.4355E-11
99	0.0	4.6893E-11	4.2488E-11
100	0.0	3.6397E-11	3.2806E-11

(to be continued)

Table 4 (continued)

Group No.	LWR-type	^{235}U	^{239}Pu
101	0.0	2.7346E-11	2.4926E-11
102	0.0	2.0898E-11	1.8531E-11
103	0.0	1.5271E-11	1.3358E-11
104	0.0	1.0730E-11	9.1912E-12
105	0.0	7.0823E-12	5.8509E-12
106	0.0	4.1682E-12	3.1888E-12
107	0.0	0.0	0.0
108	0.0	0.0	0.0
109	0.0	0.0	0.0
110	0.0	0.0	0.0
111	0.0	0.0	0.0
112	0.0	0.0	0.0
113	0.0	0.0	0.0
114	0.0	0.0	0.0
115	0.0	0.0	0.0
116	0.0	0.0	0.0
117	0.0	0.0	0.0
118	0.0	0.0	0.0
119	0.0	0.0	0.0
120	0.0	0.0	0.0
121	0.0	0.0	0.0
122	0.0	0.0	0.0
123	0.0	0.0	0.0
124	0.0	0.0	0.0
125	0.0	0.0	0.0
126	0.0	0.0	0.0
127	0.0	0.0	0.0
128	0.0	0.0	0.0
129	0.0	0.0	0.0
130	0.0	0.0	0.0
131	0.0	0.0	0.0
132	0.0	0.0	0.0
133	0.0	0.0	0.0
134	0.0	0.0	0.0
135	0.0	0.0	0.0
136	0.0	0.0	0.0
137	0.0	0.0	0.0

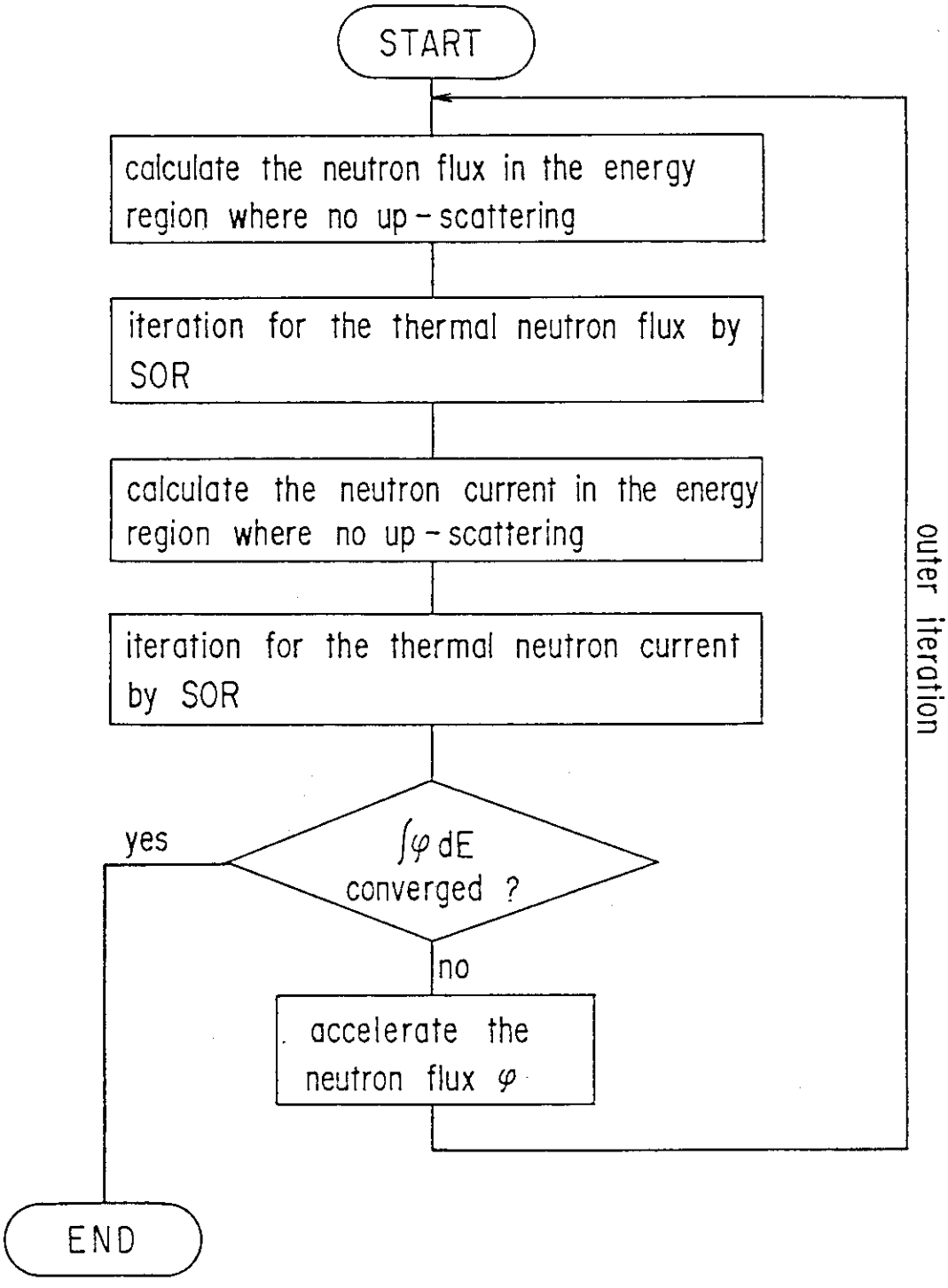

Fig. 5 Schematic flow of calculation with a given buckling.

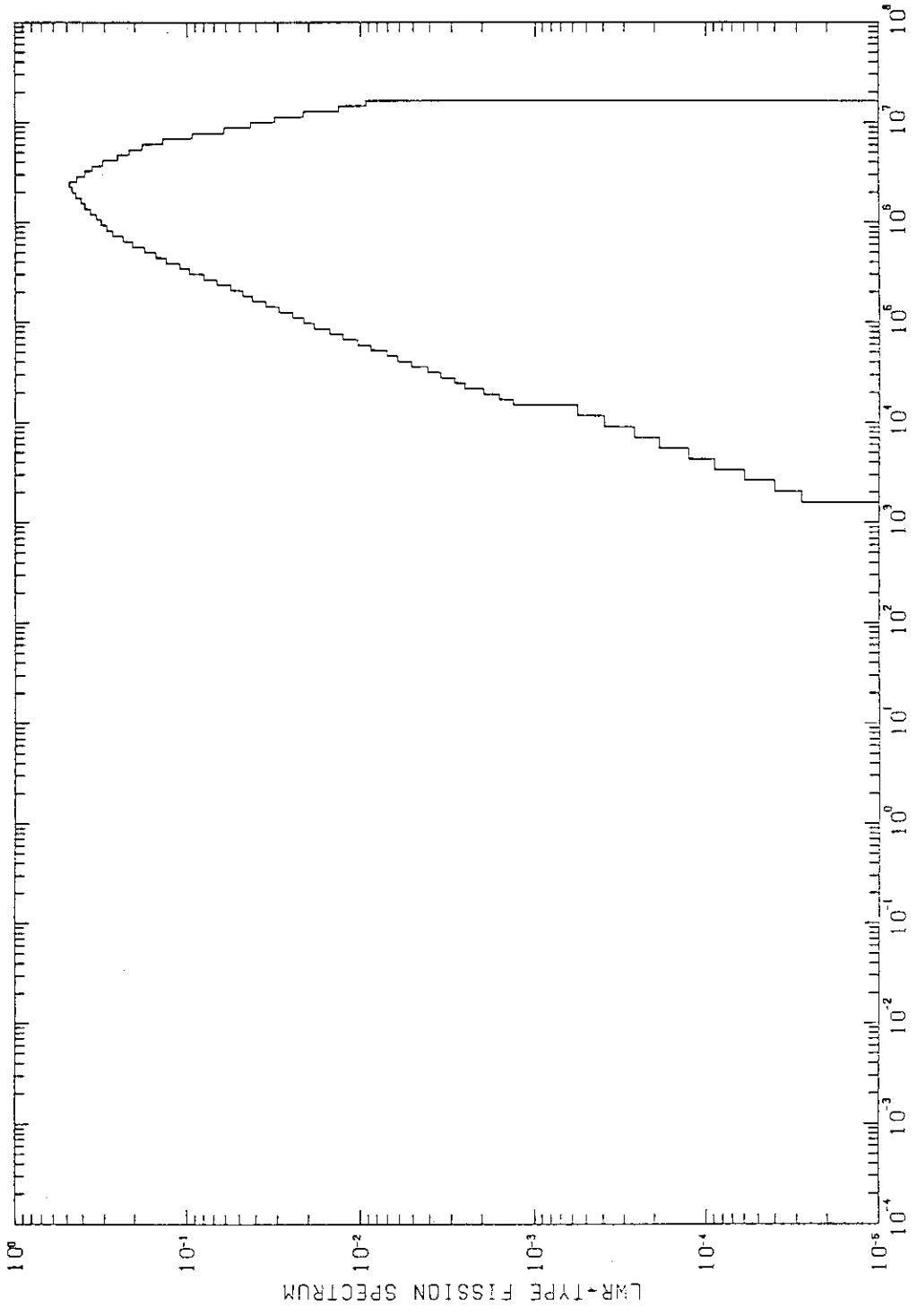

Fig. 6 LWR-type fission spectrum in SIMCRI.

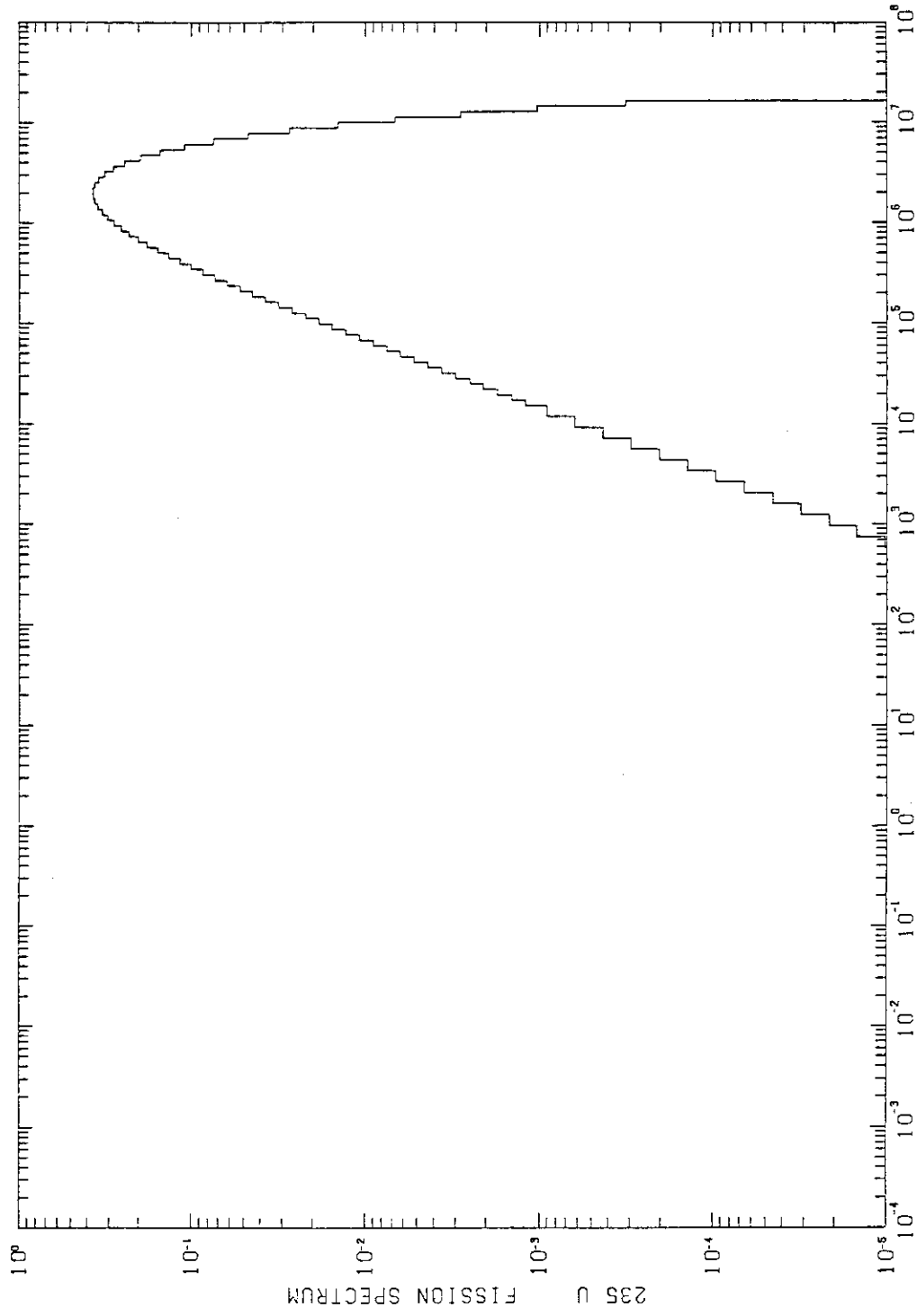

Fig. 7 ^{235}U fission spectrum in SIMCRI.

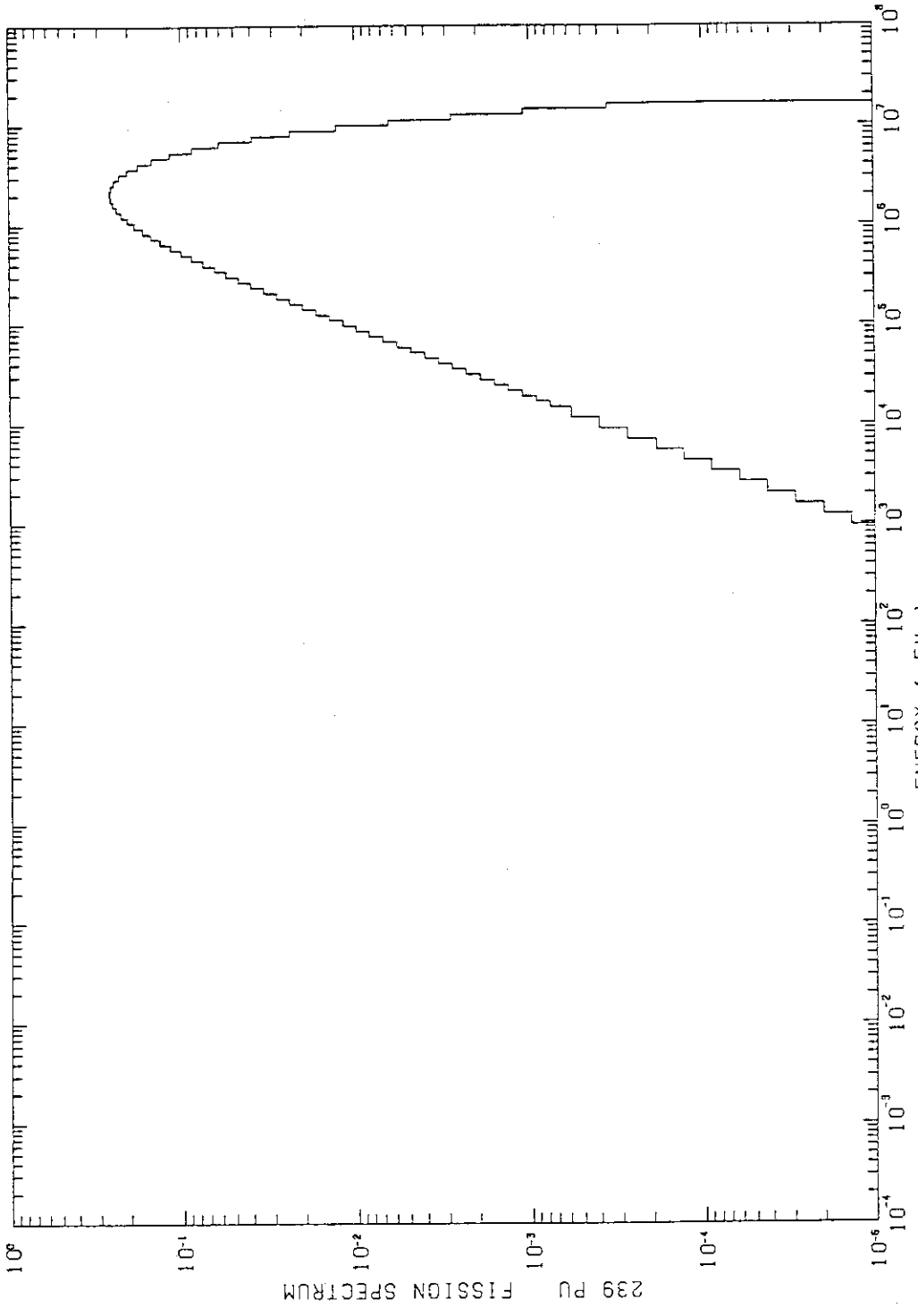

Fig. 8 ^{239}Pu fission spectrum in SIMCRI.

3.3 Criticality Parameters

The integration over the energy is, in programming implementation, interpreted as a summation over the energy groups.

(1) Infinite Multiplication Factor K_∞

$$K_\infty = \int_0^\infty dE\ \nu\Sigma_f(E)\psi_\infty(E) \ , \tag{7}$$

where $\psi_\infty(E)$ is the neutron flux of an infinite core.

(2) Effective Multiplication Factor K_{eff}

$$K_{eff} = \int_0^\infty dE\ \nu\Sigma_f(E)\psi(E) \ , \tag{8}$$

where $\psi(E)$ is the neutron flux of a finite core.

(3) Buckling B^2

When a K_{eff} is given SIMCRI calculates buckling repeatedly. For more details, see Section 3.2.

(4) Diffusion Coefficient D

$$D = \frac{\int_0^\infty dE\ J(E)}{B^2 \int_0^\infty dE\ \psi(E)} \ . \tag{9}$$

(5) Diffusion Length L and Neutron Age τ

$$L^2 = \frac{\int_{thermal} dE\ J(E)}{\int_{thermal} dE\ \Sigma_a(E)\psi(E)} \ , \tag{10}$$

$$\tau = \frac{1}{B^2} \cdot \frac{\int_{fast} dE\ J(E)}{\int_{fast} dE\ \Sigma_a(E)\psi(E) + \int_{fast} dE \int_{thermal} dE'\ \Sigma_s^{(0)}(E\to E')\psi(E)} \tag{11}$$

The cutoff energy may be specified by the input card, and its default value is 1.45 eV.

(6) Migration Area M^2

By the following three definitions, three M^2's are calculated.

$$M_1^2 = \frac{K_\infty/K_{eff} - 1}{B^2} \quad , \tag{12}$$

$$M_2^2 = \frac{\int_0^\infty dE\ J(E)}{B^2 \int_0^\infty dE\ \Sigma_a(E)\psi(E)}$$

$$= \frac{D\int_0^\infty dE\ \psi(E)}{\int_0^\infty dE\ \Sigma_a(E)\psi(E)}$$

$$= \frac{K_{\infty,eff}/K_{eff} - 1}{B^2} \quad , \tag{13}$$

$$M_3^2 = L^2 + \tau \quad , \tag{14}$$

where $K_{\infty,eff}$ denotes the ratio (fission/absorption) of a finite core:

$$K_{\infty,eff} = \frac{\int_0^\infty dE\ \nu\Sigma_f(E)\psi(E)}{\int_0^\infty dE\ \Sigma_a(E)\psi(E)} \quad , \tag{15}$$

(7) Bare Dimensions

Corresponding to the buckling, dimensions of bare cores are calculated. As for geometry, slab, cylinder and sphere are considered. The spacial distribution of the neutron flux is supposed to be in the fundamental mode of the Helmholtz equation, $\nabla^2\psi + B^2\psi = 0$. And the extrapolation length λ is assumed as

$$\lambda = 0.71\ \lambda_{tr}$$
$$= 0.71 \times 3D \quad , \tag{16}$$

coefficient defined by the equation (9). Then the bare dimensions are obtained by the following relations.

slab thickness : $\frac{\pi}{B} - 2\lambda$, (17)

cylinder radius: $\frac{z_0}{B} - \lambda$, (18)

$$\text{sphere radius} \quad : \quad \frac{\pi}{B} - \lambda \ , \tag{19}$$

where z_0 (= 2.40483) is the minimum zero point of the Bessel function J_0.

(8) Reflector Sarings

Reflector savings are calculated with one energy group equation:

$\nabla^2\phi + B^2\phi = 0$ in the core,

$\nabla^2\phi - \kappa^2\phi' = 0$ in the reflector,

$\phi = \phi'$ on the boundary,

$D\nabla n\phi = D'\nabla n\phi'$ on the boundary,

where

ϕ : neutron flux in the core,

ϕ' : neutron flux in the reflector,

B : buckling,

κ : inverse of the diffusion length in the reflector,

D : diffusion coefficient of the core,

D' : diffusion coefficient of the reflector,

∇n : gradient normal to the boundary.

The code contains typical values of κ and D', and they are choosen by an input parameter. The values of κ and D' are given in Table 5. The values are prepared through solving fixed source problems by SIMCRI.

Providing that the reflector is infinite, following relations hold:

$$\text{cylinder} \quad : \quad \frac{D'}{D} = \frac{B}{\kappa} \cdot \frac{J_1(B\,R')}{J_0(B\,R')} \cdot \frac{K_0(\kappa R')}{K_1(\kappa R')} \ , \tag{20}$$

$$\text{slab} \quad : \quad \frac{D'}{D} = \frac{B}{\kappa} \cdot \tan B\,T' \ , \tag{21}$$

$$\text{sphere} \quad : \quad \frac{D'}{D} = \frac{1 - B\,R'\cot B\,R'}{1 + \kappa R'} \ , \tag{22}$$

where

R' : radius of a reflected cylinder or sphere,

T' : thickness of a reflected slab,

J_0, J_1 : Bessel functions,

K_0, K_1 : modified Bessel functions.

Table 5 Constants used in the calculation for reflector savings.

Reflector ID	Reflector (density, g/cm^3)		Diffusion Coeff. D' [cm]	Inverse Diffusion Length κ [cm^{-1}]
1	Water	(0.9982)	0.476	0.3
2	Air	(0.0012)	1869.0	4.27×10^{-5}
3	Plexiglas	(1.182)	0.4703	0.158
4	Polyethylene	(0.930)	0.4201	0.197
5	SUS-304	(7.91)	0.9807	0.0510
6	SUS-316	(7.91)	0.9731	0.0532
7	Aluminum	(2.699)	4.244	0.0116
8	Lead	(11.340)	1.065	0.0165
9	Cadmium	(8.640)	1.314	0.0689
10	Zircaloy-4	(6.533)	1.209	0.0326

According to these relations (20), (21) and (22), reflected dimensions R' and T' are obtained. The secant method is used to solve the equation (20).

Finaly, reflector savings are calculated as

$$\text{cylinder} : R - R' , \tag{23}$$

$$\text{slab} : \frac{1}{2}(T - T') , \tag{24}$$

$$\text{sphere} : R - R' , \tag{25}$$

where R and T are the bare dimensions.

(9) Values Collapsed to Three Energy Groups

$$\text{Leakage} : \int dE\ J(E) , \tag{26}$$

$$\text{Absorption} : \frac{\int dE\ \Sigma_a(E)\psi(E)}{\int dE\ \psi(E)} , \tag{27}$$

$$\text{Fission} : \frac{\int dE\ \nu\Sigma_f(E)\psi(E)}{\int dE\ \psi(E)} , \tag{28}$$

$$\text{Scattering} : \frac{\int dE \int_0^\infty dE' \Sigma_s^{(0)}(E\to E')\psi(E)}{\int dE\ \psi(E)} , \tag{29}$$

$$P_1\text{-Scattering} : \frac{\int dE \int_0^\infty dE' \Sigma_s^{(1)}(E\to E')J(E)}{\int dE\ J(E)} , \tag{30}$$

$$\text{Fission Spectrum} : \int dE\ x(E) , \tag{31}$$

$$\text{Flux} : \int dE\ \psi(E) . \tag{32}$$

All the $\int dE$ stand for the four cases:

fast region : $\int_{E_2}^{E_{max}} dE$,

resonant region : $\int_{E_1}^{E_2} dE$,

thermal region : $\int_{E_{min}}^{E_1} dE$,

one group : $\int_{E_{min}}^{E_{max}} dE$.

The energy boundaries E_1 and E_2 may be specified by the input card, and their default valus are 0.62 eV and 7.1 keV respectively.

4. Input Data Instructions

4.1 Cross Section

The cross section sets of different materials are placed sequentially in cross section library. Each cross section set is identified by a sequential number corresponding to its order of location in the library. The number is called "region number" or "nuclide ID". If the library is prepared by the MAIL code, the region number is equal to the order of material description in the input cards for MAIL.

The cross section set has the same format as that for the KENO-IV cade, and Figure 9 shows the FORTRAN statements to read the library. A brief manual of the MAIL code is given in the appendix.

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thermal region : $\int_{E_{min}}^{E_1} dE$,

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Statement 1

```
READ(41,END=9020) NRGN,IDUM, IGM, IDUM, IDUM, IDUM
```

NRGN : Number of cross section sets
IGM : Number of energy groups
IDUM : Dummy

Statement 2

```
READ(41,END=9040) (MTITL(1), I=1, 18), INNO
&        ,(SIGA(I),SIGNF(I),SIGT(I),CHI(I),I=1, IGM)
&        ,(SIGSO(L,K), L=1,IGM), K=1,IGM)
&        ,(SIGS1(L,K), L=1,IGM), K=1,IGM)
```

MTITL : Comment for the cross section set
INNO : Region number (Nuclide ID)
SIGA : Σ_a
SIGNF : $\nu\Sigma_f$
SIGT : Σ_t
CHI : χ
SIGSO : $\Sigma_s^{(0)}$
SIGS1 : $\Sigma_s^{(1)}$

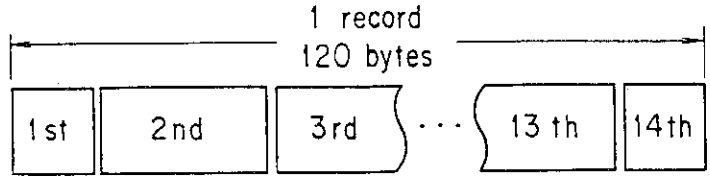
Fig. 9 FORTRAN statements to read cross sections.

The statement 1 is used once in the first. Then the statement 2 is repeated until INNO becomes a desired region number.

4.2 Summary Record

SIMCRI outputs principal results on a magnetic disk or tape. Let it be called Summary Record. The summary records are stored on a PS (Physical Sequential) file and identified by thier order in the file. The contents of each record are shown in Figure 10. To save memory, a summary record is written in internal or binary expression.

The code reads the binary records, changes its expression to the external one and prints them out on papers. By specifing two integers, say m and n, the code retrieves summary records from the m-th to the n-th in that PS file. Whenever a file is allocated or connected to the unit 16, SIMCRI always outputs a summary record to the file. If the user wants no summary output, do not define or allocate the unit 16; it causes no error. The unit 16 is also used to retrieve summary records.



field	length (byte)	contents
1	2	'??' (delimiter)
2	72	STITL
3	4	NOREFL
4	4	k_∞
5	4	$k_{\infty, eff}$
6	4	k_{eff}
7	4	B^2
8	4	τ
9	4	L^2
10	4	D
11	4	δ_{cyl}
12	4	δ_{slb}
13	4	δ_{sph}
14	2	'%%' (delimiter)

Fig. 10 Format of a summary record. The meanings of the contents are given in Sections 3.3 and 4.3.

4.3 Card Input

The card has 80 columns. Columns from 73 to 80 are not used. Blank means zero. There are five types in the input card as shown in Figure 11. Card 1 and Card 2 are indispensable for each case. Different cases may be written one after another in a deck of cards (successive problem) without a delimitting card. In case of a successive problem, the results of each case are passed to the next case as initial guess values.

Format of each parameter is given in a parentheses as (A72) in the following explanation. More detailed information will be given in the next Section 4.4.

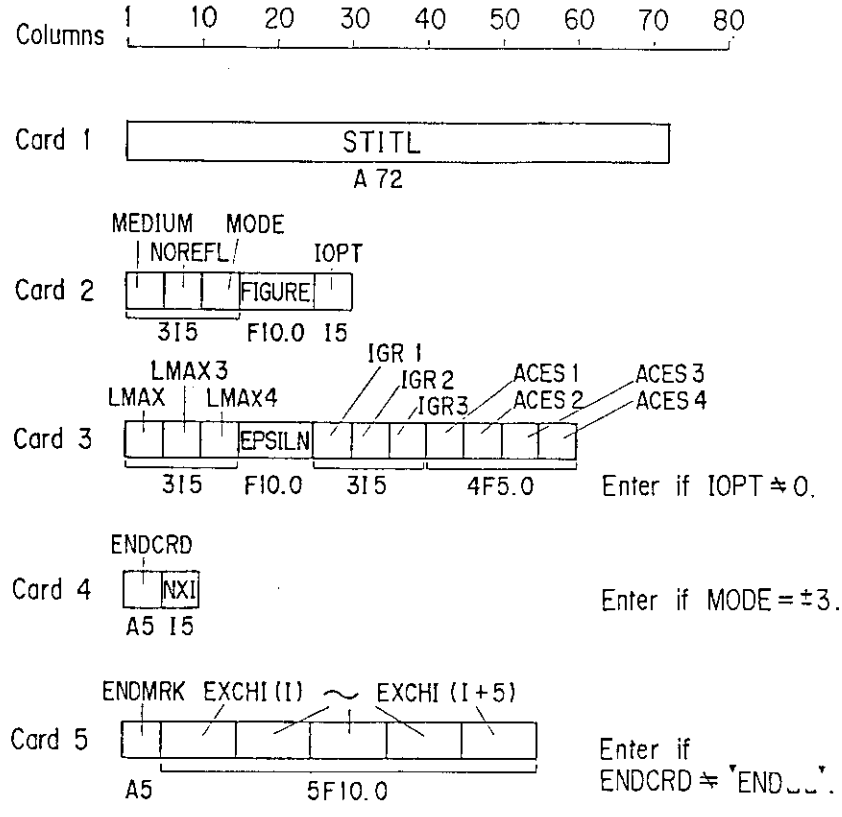


Fig. 11 Layout of input cards.

Card 1 Title Card

parameter		meaning
1	STITL (A72)	Title of a case.

Card 2 Control Card

parameter		meaning
1	MEDIUM (I5)	Region number or nuclide ID of cross section data. If MODE = 9 (3rd parameter of Card 2), MEDIUM is the first record number of summary records to be retrieved.
2	NOREFL (I5)	Reflector ID, given in Table 5. If MODE = 9 (3rd parameter of Card 2), NOREFL is the last record number of summary records to be retrieved.

NOTE: In the calculation for the neutron flux, no reflector is considered. The flux is of a bare core.

3	MODE (I5)	Execution mode.
	=0	To calculate on an infinite core only.
	=1,-1*)	To solve an eigenvalue problem with the k_{eff} specified by FIGURE (4th parameter of Card 2).
	=2,-2*)	To solvean eigenvalue problem with the B^2 specified by FIGURE (4th parameter of Card 2).
	=3,-3*)	To solve a fixed source problem with the k_{eff} specified by FIGURE (4th parameter of Card 2).
	=4,-4*)	To solve a fixed source problem with B^2 specified by FIGURE (4th parameter of Card 2).
	=9	To retrieve summary records.

*) If MODE is negative, the code puts out additional lists and graphs to check the calculation. When MODE is positive, no graph is plotted.

4	FIGURE (F10.0)	k_{eff} or B^2 (cm^{-1}), according to MODE (3rd parameter of Card 2).
5	IOPT (I5)	Parameter for the use of the optional Card 3.
	= 0	No Card 3 to be used.
	≠ 0	Card 3 to be used.

Card 3 Optional Card. Enter only if IOPT ≠ 0 (5th parameter of Card 2).

parameter		meaning
1	LMAX (I5)	Maximum number of iterations used in general.
2	LMAX3 (I5)	Maximum number of iterations in the calculation for the thermal neutron current.
3	LMAX4 (I5)	Maximum number of iterations in the calculation for the thermal neutron flux.
4	EPSILN (F10.5)	Convergence criteria.
5	IGR1 (I5)	Cutoff energy in the calculation for L^2 and τ. The lowest group number of the fast energy region in two energy groups.
6	IGR2 (I5)	Higher energy boundary. The lowest group number in the fast energy region in three energy groups.
7	IGR3 (I5)	Lower energy boundary. The lowest group number in the resonant region, used with IGR2 (previous parameter).
8	ACES1 (F5.0)	Acceleration factor in the iteration for the thermal neutron flux of an infinite core.
9	ACES2 (F5.0)	The minimum value of the acceleration factor in the outer iteration.
10	ACES3 (F5.0)	Acceleration factor in the iteration for the thermal neutron current.

11	ACES4 (F5.0)	Acceleration factor in the interation for the thermal neutron flux of a finite core.

Card 4 Fixed source type. Enter only if MODE = ±3 (3rd parameter of Card 2).

parameter		meaning
1	ENDCRD (A5)	Parameter for the use of additional spectrum data given by Card 5.
	='END '	No Card 5 to be used.
	≠'END '	Card 5 to be used.
2	NXI (I5)	ID number of spectrum prepared in the code.
	=-1	No prepared spectrum used. The fission spectrum is constructed by Card 5 only.
	= 0	LWR-type fission spectrum.
	= 1	^{235}U fission spectrum.
	= 2	^{239}Pu fission spectrum.

Card 5 Additional fixed source spectrum. Enter only if ENDCRD ≠ 'END ' (1st parameter of Card 4)

parameter		meaning
1	ENDMRK (A5)	Card 5 is repeated until ENDMRK is set as 'END '.
2	EXCHI (5F10.0)	Additional spectrum.

In a series of Card 5, multigroup values of EXCHI are placed from left to right and from up to down according to the descending order of energy levels. The leftest value in the first Card 5 is EXCHI in the highest energy region. The series of Card 5 must be ended or truncated by the card where ENDMRK = 'END '. When truncated, the rest of EXCHI are taken as zero's. Finally, the fixed source is constructed as a normalized sum of prepared spectrum specified by NXI and additional spectrum EXCHI.

4.4 Detailed Information

(1) STITL (1st parameter of Card 1)

As shown in Figure 10, the title is included in a summary record. So it will be very convienient to set some rules on STITL when processing a lot of summary records with the aid of a computer.

(2) MEDIUM (1st parameter of Card 2)

If the number is so large that there is no corresponding cross section sets, a list of comments for each cross section set in the library is printed out.

(3) FIGURE (4th parameter of Card 2)

Default values are:

$k_{eff} = 1.0$,

$B_c^2 = 0.01\ cm^{-1}$.

These values are allocated when FIGURE is less than or equal to zero or blank.

(4) Card 3

In case of a successive problems, and if Card 3 is used in one of the cases, all the following cases uses the same values for parameters of Card 3 until the next Card 3 appears.

Default values are:

LMAX = 1000,

LMAX3 = 30,

LMAX4 = 20,

EPSILN = 1.0×10^{-5},

IGR1 = 12 (IGM = 26),
95 (IGM = 137),
IGM/2 (else),

IGR2 = 6 (IGM = 26),
60 (IGM = 137),
IGM/3 (else),

IGR3 = 14 (IGM = 26),
101 (IGM = 137),
IGM/3*2(else),

where IGM is the number of energy groups.

ACES1 = 0.5 ,
ACES2 = -0.5 ,
SCES3 = 0.2 ,
ACES4 = 0.5 .

If -9.0 is specified for ACES1 ~ ACES4, it is taken as 0.0 and no acceleration is performed in the iteration.

5. Sample Problem

The code is made to obtain a k_{eff} when the buckling is 1.0×10^{-2} cm^{-2} and the fuel is

$Pu(NO_3)_4$ aqueous solution ,

Pu	50	gPu/ℓ ,
$^{240}Pu/^{239}Pu$	10	wt % ,
$^{241}Pu, ^{242}Pu$	0	wt % .

Input data with job control statenents are shown in Figure 12, and its output are listed in Figure 13. In this sample, MODE is negative and four check lists and four graphs are obtained additionaly. There is no file of the unit number 16, and no summary record is output.

```
//JCLG JOB
//JCLG EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER 03327611,SH.NAKAMARU,0943.01
   T.0 W.0 I.0 C.1 OPN GRP
  OPTP PASSWORD=S,NOTIFY=J7611
//SAMPLE  EXEC  PGM=TEMPNAME
//STEPLIB   DD  DSN=J7611.SIMCRI.LOAD,DISP=SHR
//FT06F001  DD  SYSOUT=*
//*T16F001  DD  DSN=J7611.SIMCRI.SUMMARY.DATA,DISP=OLD
//FT41F001  DD  DSN=J7611.KENOLIB.DATA,DISP=SHR
//FT05F001  DD  *
 SAMPLE PROBLEM: PU(NO3)2  50.0(G-PU/L), 240/239=10(WT%), 26-GR.
    1     1    -2       0.01
/*
//GDFILE  DD  SYSOUT=H
//MPTMST  DD  DSN=SYS1.KPATNLIB,DISP=SHR
++
//
```

Fig. 12 Sample input with JCL statements.

J7611. SIMCRI. LOAD is the load module,
J7611. KENOLIB. DATA is made by MAIL
with a JCL shown in Fig. A-1.

```
INPUT DATA IMAGE LIST

CARD
SEQ.----5----0----5----0----5----0----5----0----5----0----5----0----5----0----5----0----5----0----5----0
  1 SAMPLE PROBLEM: PU(NO3)2  50.0(G-PU/L), 240/239=10(WT%), 26-GR.
  2    1    1   -2     0.01
 ----5----0----5----0----5----0----5----0----5----0----5----0----5----0----5----0----5----0----5----0
```

Fig. 13　Sample cutput. (to be continued)

```
*MSG*    TEST2  : INTERPRETED INPUT DATA

TITLE              :  SAMPLE PROBLEM: PU(NO3)2  50.0(G-PU/L), 240/239=10(WT%), 26-GR.

REGION     NUMBER  :      1
REFLECTOR NUMBER   :      1
EXECUTION MODE     :     -2

MAX OF ITERATIONS FOR FLUX (OUTER)   :    1000
     - FOR THERMAL NEUTRON CURRENT   :      30
        - FOR THERMAL NEUTRON FLUX   :      20
CONVERGENCE CRITERION                :    1.00E-05
ACC. FACTOR FOR INFINITE-CORE-FLUX   :    0.50
    - FOR FINITE-CORE-FLUX (OUTER)   :   -0.50
     - FOR THERMAL NEUTRON CURRENT   :    0.20
        - FOR THERMAL NEUTRON FLUX   :    0.50

K-EFFECTIVE           ( MODE 1/3 )   :    1.0000
BUCKLING              ( MODE 2/4 )   :    1.0000E-02
FIXED SOURCE   TYPE   ( MODE 3/4 )   :    0

GIVEN SOURCE SPECTRUM                :-
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0
 0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0        0.0

FOR THE BOUNDARIES OF 2/3-GROUP COLLAPSING , PLEASE LOOK AT THE FINAL OUTPUT .
```

Fig. 13 (additional list, to be continued)

```
*MSG*   TEST3  : LIST OF TITLES IN THE CROSS SECTION LIBRARY .

  NO. : TITLE  (  26 ENERGY GROUPS ,   3 REGIONS )
    1 : *(U+PU)NO3* PU=50.0(G-PU/L) 240/239=10(WT%) 26-GR.
    2 : TITLE CARD FOR 2-ND REGION
    3 : 3-RD REGION

*MSG*   SPECTR  : FISSION SPECTRUM HAS BEEN PLOTTED .

      SAMPLE PROBLEM: PU(NO3)2  50.0(G-PU/L), 240/239=10(WT%), 26-GR.

*MSG*   TEST4  : TERMS OF A B1/P1-EQUATION ( INFINITE CORE )

MAX OF ITERATIONS= 1000  , CONVERGENCE CRITERION= 1.000E-05
ITERATION :  FLUX=   35  , CONVERGENCE :  FLUX  = 9.827E-06

(1): SIGMA-T * FLUX   (2): SIGMA-SO * FLUX   (3): SOURCE   <4>: (1)-(2)-(3)

 GR.      (1)          (2)          (3)          <4>
   1   2.0E-01      1.0E-01      1.0E-01  : -1.2E-07 :
   2   1.5E+00      9.5E-01      5.9E-01  : -2.4E-07 :
   3   2.2E+00      1.9E+00      2.8E-01  :  0.0     :
   4   1.4E+00      1.3E+00      2.8E-02  : -1.2E-06 :
   5   1.3E+00      1.3E+00      4.6E-03  :  3.2E-07 :
   6   1.1E+00      1.1E+00      3.5E-04  :  2.7E-07 :
   7   2.4E+00      2.4E+00      8.6E-05  :  5.2E-07 :
   8   1.6E+00      1.6E+00      0.0      : -9.5E-07 :
   9   1.6E+00      1.6E+00      0.0      :  0.0     :
  10   1.6E+00      1.6E+00      0.0      : -9.5E-07 :
  11   1.3E+00      1.3E+00      0.0      :  0.0     :
  12   4.2E-01      4.2E-01      0.0      :  1.8E-07 :
  13   4.2E-01      4.2E-01      0.0      :  2.4E-07 :
  14   2.9E-01      2.9E-01      0.0      :  1.8E-07 :
  15   6.2E-01      6.2E-01      0.0      :  4.8E-07 :
  16   2.4E-01      2.4E-01      0.0      :  3.0E-07 :
  17   2.7E-01      2.7E-01      0.0      :  2.4E-07 :
  18   3.3E-01      3.3E-01      0.0      :  1.2E-07 :
  19   4.9E-01      4.9E-01      0.0      : -9.5E-07 :
  20   9.3E-01      9.3E-01      0.0      : -4.9E-06 :
  21   1.9E+00      1.9E+00      0.0      : -9.5E-06 :
  22   3.3E+00      3.3E+00      0.0      : -1.5E-05 :
  23   4.5E+00      4.5E+00      0.0      : -1.3E-05 :
  24   4.2E+00      4.2E+00      0.0      : -7.6E-06 :
  25   2.0E+00      2.0E+00      0.0      : -1.9E-06 :
  26   2.4E-01      2.4E-01      0.0      :  6.0E-08 :

SUM :   3.6E+01      3.5E+01      1.0E+00  : -5.4E-05 :

K-INFINITIVE= 1.565E+00

*MSG*   SPECTR  : SPECTRUM OF NEUTRON FLUX IN INFINITE CORE HAS BEEN PLOTTED .
```

Fig. 13 (additional list, to be continued)

SAMPLE PROBLEM: PU(NO3)2 50.0(G-PU/L), 240/239=10(WT%), 26-GR.

MSG TEST5 : TERMS OF TWO B1-EQUATIONS (FINITE CORE)

	FLUX(TOTAL)	CURRENT(THERMAL)	FLUX(THERMAL)	BUCKLING SEARCH
ITERATIONS				
MAX FOR EACH :	1000	30	20	1000
CUMULATIVE SUM :	13	19	151	0
CONVERGENCE				
CRITERION :	1.000E-05	1.000E-05	1.000E-05	1.000E-05
RESULTS :	3.001E-06	7.011E-08	7.543E-06	1.959E-01

(1): CURRENT (2): SIGMA-T * FLUX (3): SIGMA-SO * FLUX (4): SOURCE
<5>: (1)+(2)-(3)-(4)
(6): - B*B * FLUX (7): 3 * GAMMA * SIGMA-T * CURRENT (8): 3 * SIGMA-S1 * CURRENT
<9>: (6)+(7)-(8)

GR.	(1)	(2)	(3)	(4)	<5>	(6)	(7)	(8)	<9>
1	3.7E-02	1.3E-01	6.5E-02	1.0E-01	-1.2E-07	-1.1E-02	1.5E-02	3.9E-03	7.5E-09
2	9.5E-02	1.3E+00	7.6E-01	5.9E-01	2.0E-06	-5.0E-02	7.5E-02	2.5E-02	-6.7E-08
3	4.2E-02	1.8E+00	1.6E+00	2.8E-01	9.5E-07	-3.3E-02	6.8E-02	3.4E-02	-2.2E-08
4	1.2E-02	1.1E+00	1.1E+00	2.8E-02	-1.2E-06	-1.2E-02	3.3E-02	2.2E-02	-2.2E-08
5	7.1E-03	1.1E+00	1.1E+00	4.6E-03	-6.3E-07	-8.6E-03	2.7E-02	1.8E-02	-2.2E-08
6	4.5E-03	8.6E-01	8.6E-01	3.5E-04	-2.7E-07	-6.2E-03	1.9E-02	1.3E-02	-3.7E-09
7	8.4E-03	1.9E+00	1.9E+00	8.6E-05	-1.4E-06	-1.3E-02	3.7E-02	2.4E-02	-2.2E-08
8	5.2E-03	1.3E+00	1.3E+00	0.0	0.0	-8.6E-03	2.3E-02	1.4E-02	-1.9E-08
9	5.0E-03	1.2E+00	1.3E+00	0.0	0.0	-8.4E-03	2.2E-02	1.4E-02	0.0
10	4.8E-03	1.2E+00	1.2E+00	0.0	-9.5E-07	-8.2E-03	2.1E-02	1.3E-02	-3.7E-09
11	4.0E-03	1.0E+00	1.0E+00	0.0	-2.9E-06	-6.8E-03	1.7E-02	1.1E-02	0.0
12	1.3E-03	3.2E-01	3.2E-01	0.0	-2.4E-07	-2.2E-03	5.7E-03	3.5E-03	1.4E-09
13	9.5E-04	3.2E-01	3.3E-01	0.0	-6.0E-08	-1.8E-03	5.1E-03	3.3E-03	-1.9E-09
14	7.5E-04	2.2E-01	2.2E-01	0.0	-1.2E-07	-1.4E-03	3.6E-03	2.2E-03	1.4E-09
15	1.4E-03	4.8E-01	4.8E-01	0.0	1.8E-07	-2.8E-03	7.1E-03	4.2E-03	7.5E-09
16	4.1E-04	1.9E-01	1.9E-01	0.0	-6.0E-08	-9.6E-04	2.4E-03	1.4E-03	7.0E-10
17	3.1E-04	2.1E-01	2.1E-01	0.0	0.0	-8.8E-04	2.2E-03	1.4E-03	-9.3E-10
18	3.6E-04	2.5E-01	2.5E-01	0.0	6.0E-08	-1.1E-03	2.4E-03	1.3E-03	1.6E-09
19	5.0E-04	3.7E-01	3.8E-01	0.0	1.0E-06	-1.7E-03	3.2E-03	1.5E-03	2.1E-09
20	7.7E-04	7.1E-01	7.1E-01	0.0	3.0E-06	-3.1E-03	5.2E-03	2.1E-03	1.4E-09
21	1.3E-03	1.5E+00	1.5E+00	0.0	5.7E-06	-6.0E-03	9.1E-03	3.1E-03	3.7E-09
22	1.8E-03	2.5E+00	2.5E+00	0.0	1.0E-05	-9.8E-03	1.4E-02	4.0E-03	3.7E-09
23	1.8E-03	3.4E+00	3.4E+00	0.0	6.7E-06	-1.2E-02	1.5E-02	3.7E-03	-1.6E-09
24	8.9E-04	3.2E+00	3.2E+00	0.0	6.7E-06	-8.5E-03	1.0E-02	1.5E-03	-2.3E-09
25	2.2E-04	1.5E+00	1.5E+00	0.0	9.5E-07	-3.1E-03	3.2E-03	1.8E-04	-6.7E-10
26	1.2E-05	1.8E-01	1.8E-01	0.0	0.0	-2.6E-04	2.5E-04	-8.2E-06	-1.0E-10
SUM :	2.4E-01	2.8E+01	2.7E+01	1.0E+00	3.0E-05	-2.2E-01	4.5E-01	2.2E-01	-1.6E-07

K-EFFECTIVE= 1.196E+00 , BUCKLING= 1.000E-02

MSG SPECTR : SPECTRUM OF NEUTRON FLUX IN FINITE CORE HAS BEEN PLOTTED .

MSG SPECTR : SPECTRUM OF LEAKAGE , NET CURRENT , HAS BEEN PLOTTED .

Fig. 13 (additional list, to be continued)

```
(1)    TITLE              :  SAMPLE PROBLEM: PU(NO3)2  50.0(G-PU/L), 240/239=10(WT%), 26-GR.

       CORE               :     1 , *(U+PU)NO3* PU=50.0(G-PU/L) 240/239=10(WT%) 26-GR.
       REFLECTOR          :     1 , LIGHT WATER  0.9982(G/CM**3) , 20 C
       EXECUTION MODE     :    -2 , CALCULATE K-EFFECTIVE WITH GIVEN BUCKLING                    ( TEST MODE )
       BUCKLING           :   1.000E-02

(2)                          MAX/CRITERION     FLUX(INFINITE)     FLUX(FINITE)     B**2 SEARCH
       ITERATIONS         :        1000                35                13               0
       CONVERGENCE        :    1.0E-05              9.8E-06           3.0E-06

(3)    INFINITE CORE

       K-INFINITIVE       :       1.565

            LEAKAGE  ABSORPTION     FISSION  SCATTERING  P1-SCATTER.  SPECTRUM-CHI      FLUX     ENERGY GR.
  1     0.0          1.25E-02    1.06E-02    7.66E+00    4.21E+00     1.00E+00     1.52E+01  (    1 -    6 )
  2     0.0          9.98E-02    5.69E-02    9.47E+00    5.68E+00     8.63E-05     6.45E+00  (    7 -   14 )
  3     0.0          8.85E-01    1.50E+00    1.82E+01    4.47E+00     0.0          6.57E+00  (   15 -   26 )
SUM     0.0          9.97E-01    1.57E+00    3.53E+01    1.44E+01     1.00E+00     2.82E+01

(4)    FINITE CORE

       K-EFFECTIVE        :       1.196
          BUCKLING        :   1.000E-02

                                                                                          ENERGY GR.
       FISSION/ABSORP.    :       1.570          THERMAL NEUTRON AGE  :    2.899E+01  (    1 -   12 )
       DIFFUSION COEFF.   :   1.068E+00          DIFFUSION LENGTH **2 :    1.570E+00  (   13 -   26 )

       MIGRATION AREA     :   3.090E+01  ( ( KINF/KEFF - 1 )/B**2 )
       MIGRATION AREA     :   3.106E+01  (   LEAK./ABSORP./B**2   )
       MIGRATION AREA     :   3.056E+01  (       TAU + L**2       )                  ( UNIT OF LENGTH : CM )

            LEAKAGE  ABSORPTION     FISSION  SCATTERING  P1-SCATTER.  SPECTRUM-CHI      FLUX     ENERGY GR.
  1     1.97E-01     8.95E-03    8.32E-03    6.21E+00    3.41E+00     1.00E+00     1.21E+01  (    1 -    6 )
  2     3.03E-02     7.74E-02    4.44E-02    7.43E+00    4.45E+00     8.63E-05     5.06E+00  (    7 -   14 )
  3     9.66E-03     6.75E-01    1.14E+00    1.39E+01    3.41E+00     0.0          5.01E+00  (   15 -   26 )
SUM     2.37E-01     7.62E-01    1.20E+00    2.75E+01    1.13E+01     1.00E+00     2.21E+01

(5)    DIMENSIONS         :     BARE   REFLECTED     SAVINGS        - BY ONE ENERGY GR. EQUATION ( UNIT : CM )
     CYLINDER DIAMETER          43.5         34.5         4.5
        SLAB THICKNESS          26.9         18.6         4.1
      SPHERE DIAMETER           58.3         48.1         5.1

*MSG*    MAIN  : THERE WERE     8 MESSAGES .
```

Fig. 13 (to be continued)

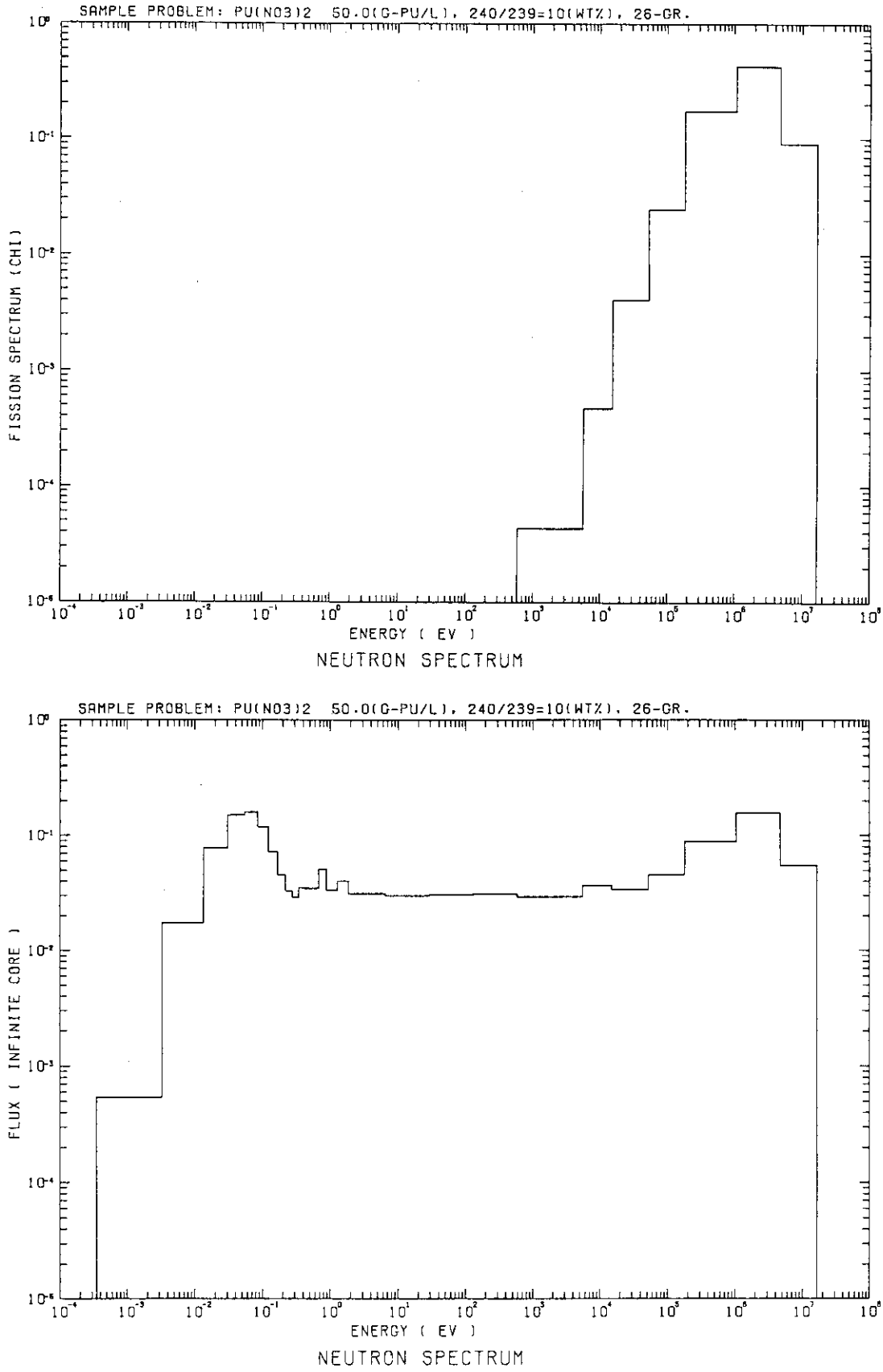
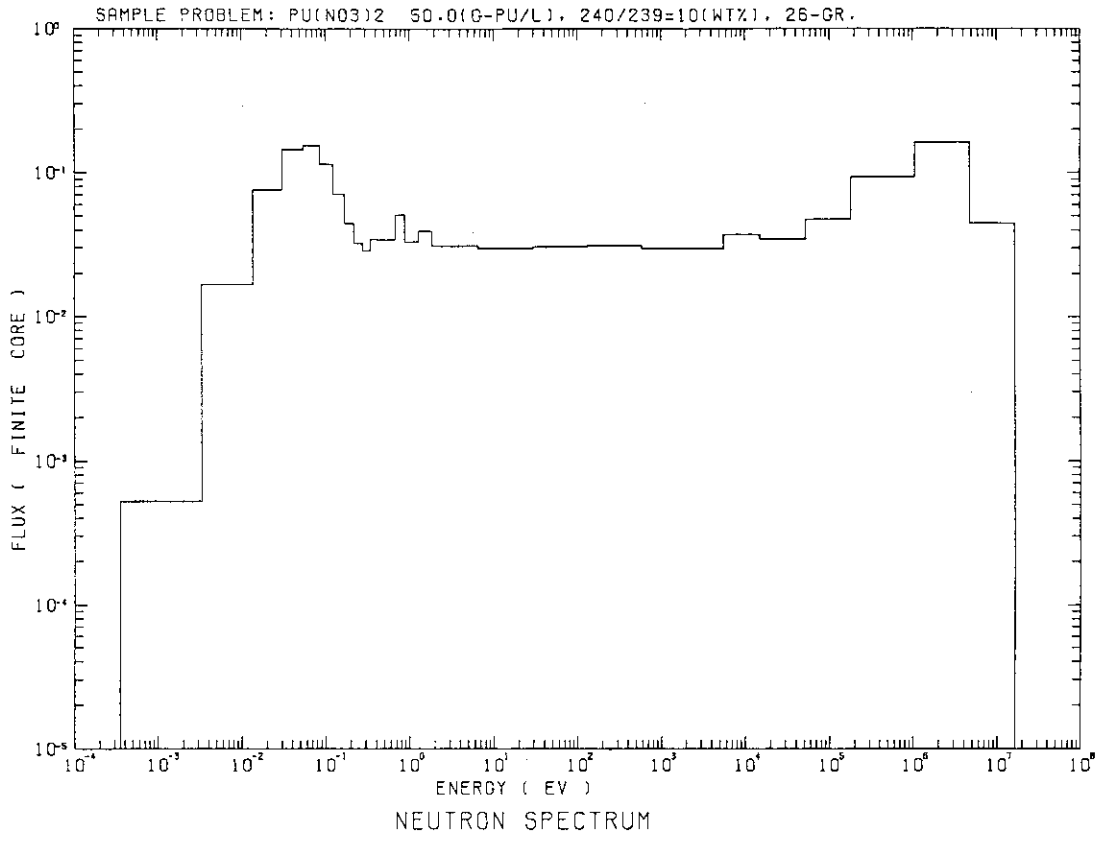


Fig. 13 (additional list, to be continued)



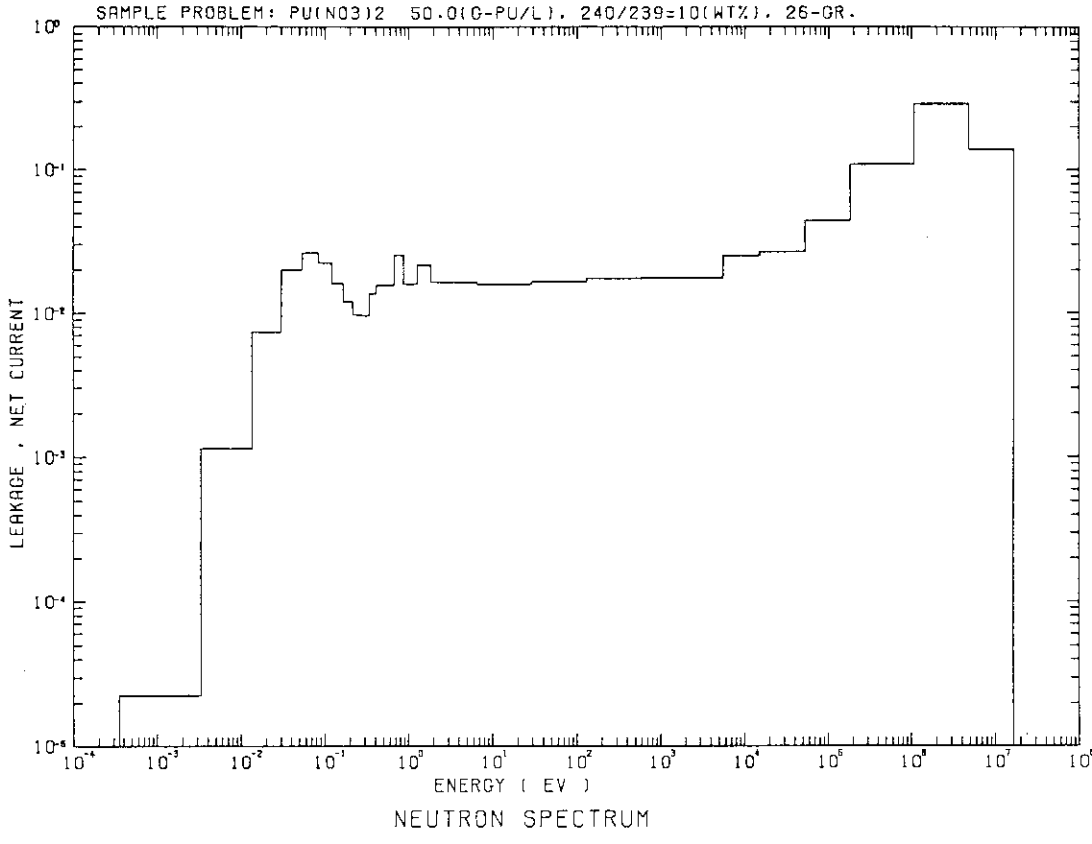


Fig. 13 (additional list)

Acknowledgement

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References

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3. L.M. PETRIE, N.F. CROSS, "KENO-IV An Improved Monte Carlo Criticality Program", ORNL-4938 (1975).

Appendix Brief Manual of the MAIL Code

Card 1 Format (4i5)

MAXREG	- Number of regions
IGM	- Number of groups (137,109,67,26)
IOPT	- Calculation option
	0 - only σ_{eff} calculation
	1 - ANiSN library
	2 - KENO library
	3 - ANiSN library and KENO library
IPRINT	- ANISN library or KENO library print option
	0 - no print
	1 - print

The following input data (from card 2 to card)7) are iterated from region 1 to MAXREG.

Card 2 Format (12A4)

Title card.

Card 3 Format (4I5, F10.0)

NMAX	- Number of nuclides
IREG	- Dancoff correction option
	0 - no correction
	1 - Dancoff correction for slab geometry
	2 - Dancoff correction for cylindrical geometry of rectangular lattice
	-2 - Dancoff correction for cylindrical geometry of hexagonal lattice
IPRINT	- Print option
	0 - σ^j_{eff}, f-table and g-table print
	1 - only σ^j_{eff} print
	2 - only f-table and g-table print
	3 - no print
NSTOP	- Number of iteration count limits. (10*)
EPSL	- Maximum relative change for σ_o calculation (0.0001)

* Default values are shown in parentheses. These are replaced by non-zero input data.

Card 4 Format (6(5X,I7))

(JNAME(i), i=1,NMAX)

Specify nuclide id. number from nuclide 1 to NMAX. Nuclide id. number is set as follows.

$^{235}_{92}U$

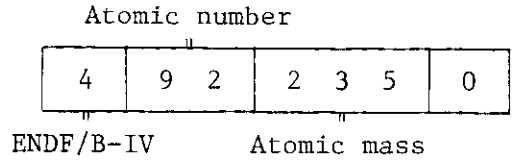

Table A-1 is a list of nuclide IDs of the 300 K MGCL

Card 5 Format (6F12.0)

(DEN(i), i=1, NMAX)

Specify atomic number density from nuclide 1 to NMAX.

Card 6 required only if IREG = 1

Card 6 Format (3F12.0)

THFUEL	- Thickness of the fuel
THMODL	- Thickness of the moderator
CR)SS1	- Macroscopic total cross-section for the moderator region at epi-thermal energy region

Card 7 requires only if IREG = 2 or IREG = -2

Card 7 Format (4F12.0)

PICHI	- Length of the fuel pitch
PELETR	- Radius of the fuel pellet
FUELR	- Radius of the fuel
CROSS1	- Macroscopic total cross-section for the moderator region at epi-thermal energy region

Fig. A-1 shows a sample input. Table A-2 describes the data files to be used. MGCL is available in 137 or 26 energy groups, and their energy structure is given in Table A-3.

Table A-1 Nuclide ID (300 K MGCL)

4003020	ZR-NAT	4010010	H- 1	4010020	D- 2	4020030	HE- 3
4020040	HE- 4	4040090	BE- 9	4050100	B- 10	4050110	B- 11
4060120	C- 12	4070140	N- 14	4080160	O- 16	4090000	F-NAT
4110230	NA- 23	4120000	MG-NAT	4130270	AL- 27	4140000	SI-NAT
4160000	S-NAT	4170000	CL-NAT	4190000	K-NAT	4200000	CA-NAT
4220000	TI-NAT	4240000	CR-NAT	4250550	MN- 55	4260000	FE-NAT
4270590	CO- 59	4280000	NI-NAT	4290000	CU-NAT	4360800	KR- 80
4400900	ZR- 90	4420000	MO-NAT	4430990	TC- 99	4441020	RU-102
4451030	RH-103	4451050	RH-105	4471070	AG-107	4471090	AG-109
4480000	CD-NAT	4481130	CD-113	4491130	IN-113	4491150	IN-115
4531350	I-135	4541310	XE-131	4541350	XE-135	4551330	CS-133
4551340	CS-134	4581440	CE-144	4601430	ND-143	4601440	ND-144
4601450	ND-145	4601460	ND-146	4611470	PM-147	4611480	PM-148
4611481	PM-148M	4611490	PM-149	4621480	SM-148	4621490	SM-149
4621500	SM-150	4621510	SM-151	4621520	SM-152	4621530	SM-153
4631530	EU-153	4631540	EU-154	4631550	EU-155	4640000	GD-NAT
4641550	GD-155	4641560	GD-156	4641570	GD-157	4641580	GD-158
4661610	DY-161	4661620	DY-162	4661630	DY-163	4661640	DY-164
4791970	AU-197	4820000	PB-NAT	4902320	TH-232	4912330	PA-233
4922330	U-233	4922340	U-234	4922350	U-235	4922360	U-236
4922380	U-238	4932370	NP-237	4942380	PU-238	4942390	PU-239
4942400	PU-240	4942410	PU-241	4942420	PU-242	4952410	AM-241
4952430	AM-243	4962440	CM-244	9990000	FP		

NOTE: These nuclide ID numbers are for the MAIL code, and have no relations with MEDIUM (1st parameter of input Card 2 for SIMCRI).

Table A-2 Data Files for MAIL

Logical Unit No.	File Organization	Record Format	Disposition	Usage
1	PS	VBS	SHR	MGCL
2	PS	VBS	NEW	(temporary)
3	PS	VBS	OLD	KENO library
4	PS	VBS	OLD	ANISN library
5	PS	FB	SHR	input cards
6	PS	FBA	OLD	output list
31	PS	VBS	SHR	SMF*)

*) Scattering Matrix File

Table A-3 Energy structure of 137/26 groups.

Group No. 137	Group No. 26	Upper Boundary [eV]
1		1.6487E+07
2		1.4550E+07
3		1.2840E+07
4		1.1331E+07
5	1	1.0000E+07
6		8.8250E+06
7		7.7880E+06
8		6.8729E+06
9		6.0653E+06
10		5.3526E+06
11		4.7237E+06
12		4.1686E+06
13		3.6788E+06
14		3.2465E+06
15		2.8650E+06
16	2	2.5284E+06
17		2.2313E+06
18		1.9691E+06
19		1.7377E+06
20		1.5335E+06
21		1.3534E+06
22		1.1943E+06
23		1.0540E+06
24		9.3014E+05
25		8.2085E+05
26		7.2440E+05
27		6.3928E+05
28		5.6416E+05
29	3	4.9787E+05
30		4.3937E+05
31		3.8774E+05
32		3.4218E+05
33		3.0187E+05
34		2.6649E+05
35		2.3518E+05
36		2.0754E+05
37		1.8316E+05
38		1.6163E+05
39		1.4264E+05
40		1.2588E+05
41	4	1.1109E+05
42		9.8037E+04
43		8.6517E+04
44		7.6351E+04
45		6.7379E+04
46		5.9462E+04
47		5.2475E+04
48		4.6309E+04
49		4.0868E+04
50		3.6066E+04
51	5	3.1828E+04
52		2.8088E+04
53		2.4788E+04
54		2.1875E+04
55		1.9305E+04
56		1.7036E+04
57		1.5034E+04
58	6	1.1709E+04
59		9.1188E+03
60		7.1017E+03
61		5.5308E+03
62		4.3074E+03
63		3.3546E+03
64		2.6126E+03
65	7	2.0347E+03
66		1.5846E+03
67		1.2341E+03
68		9.6112E+02
69		7.4852E+02
70		5.8295E+02
71		4.5400E+02
72	8	3.5358E+02
73		2.7536E+02
74		2.1445E+02
75		1.6702E+02
76		1.3007E+02
77		1.0130E+02
78	9	7.8893E+01
79		6.1442E+01
80		4.7851E+01
81		3.7267E+01
82		2.9023E+01
83		2.2603E+01
84	10	1.7603E+01
85		1.3710E+01
86		1.0677E+01
87		8.3153E+00

(to be continued)

Table A-3 (continued)

Group No.		Upper Boundary
137	26	[eV]
88		6.4760E+00
89		5.0435E+00
90	11	3.9279E+00
91		3.0590E+00
92		2.3824E+00
93		1.8554E+00
94	12	1.6374E+00
95		1.4450E+00
96		1.2752E+00
97	13	1.1254E+00
98		9.9312E-01
99		8.7642E-01
100	14	7.7344E-01
101		6.8256E-01
102		6.0236E-01
103	15	5.3158E-01
104		4.6912E-01
105		4.1399E-01
106	16	3.8925E-01
107		3.6528E-01
108		3.4206E-01
109	17	3.1961E-01
110		2.9792E-01
111		2.7699E-01
112	18	2.5683E-01
113		2.3742E-01
114		2.1878E-01
115	19	2.0090E-01
116		1.8378E-01
117		1.6743E-01
118	20	1.5183E-01
119		1.3700E-01
120		1.2293E-01
121	21	1.0962E-01
122		9.7078E-02
123		8.5295E-02
124	22	7.4274E-02
125		6.4015E-02
126		5.4518E-02
127	23	4.5783E-02
128		3.7811E-02
129		3.0600E-02
130	24	2.4152E-02
131		1.8465E-02
132		1.3541E-02
133	25	9.3792E-03
134		5.9793E-03
135		3.3414E-03
136	26	1.4657E-03
137		3.5209E-04
END		3.3066E-05

```
//JCLG JOB
//JCLG EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER 03327611,SH.NAKAMARU,0943.01
   T.1 C.2 I.3 OPN
  OPTP PASSWORD=S,NOTIFY=J7611
//MAIL    EXEC  PGM=TEMPNAME
//STEPLIB   DD  DSN=J3069.MAIL.LOAD,DISP=SHR
//FT31F001  DD  DSN=J1446.SMF26302.DATA,DISP=SHR,LABEL=(,,,IN)
//FT01F001  DD  DSN=J1446.CL26300.DATA,DISP=SHR,LABEL=(,,,IN)
//*T31F001  DD  DSN=J3375.SM137300.DATA,DISP=SHR,LABEL=(,,,IN)
//*T01F001  DD  DSN=J3375.CL137300.DATA,DISP=SHR,LABEL=(,,,IN)
//FT02F001  DD  DISP=NEW,UNIT=TSSWK,SPACE=(TRK,(10,10)),
//         DCB=(RECFM=VBS,LRECL=19064,BLKSIZE=19068)
//FT03F001  DD  DSN=J7611.KENOLIB.DATA,DISP=(NEW,CATLG,DELETE),
//         UNIT=D0950,SPACE=(TRK,(10,10),RLSE),
//         DCB=(RECFM=VBS,LRECL=19064,BLKSIZE=19068)
//*T04F001  DD  DSN=JXXXX.ANILIB.DATA,DISP=(NEW,CATLG,DELETE),
//*        UNIT=TSSWK,SPACE=(TRK,(10,10),RLSE),
//*        DCB=(RECFM=VBS,LRECL=19064,BLKSIZE=19068)
//FT06F001  DD  SYSOUT=*
//FT05F001  DD  *
    3   26    2
*(U+PU)NO3* PU=50.0(G-PU/L) 240/239=10(WT%) 26-GR.
    5    0    3
     4010010     4070140     4080160     4942390     4942400
 6.52902E-02 5.03635E-04 3.41560E-02 1.14506E-04 1.14029E-05
TITLE CARD FOR 2-ND REGION
    3    0    3
     4080160     4942390     4942400
 8.88530E-03 4.22140E-03 2.21252E-04
3-RD REGION
    2    0    3
     4010010     4080160
  6.6735E-02  3.3368E-02
/*
++
//
```

Fig. A-1 Sample input of the MAIL code.