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MORSE-DD

A MONTE CARLO CODE USING MULTI-
GROUP DOUBLE DIFFERENTIAL FORM
CROSS SECTIONS

July 1984

Masayuki NAKAGAWA and Takamasa MORI

日本原子力研究所
Japan Atomic Energy Research Institute

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MORSE-DD, A Monte Carlo Code Using Multi-group
Double Differential Form Cross Sections

Masayuki NAKAGAWA and Takamasa MORI

Department of Reactor Engineering
Tokai Research Establishment, JAERI

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A modified version of the Monte Carlo code MORSE-CG has been developed. The present version is useful to solve neutron transport problems in materials with highly anisotropic neutron scattering cross section as seen on neutronics calculations in fusion reactors. Main parts of modifications are the cross section and analysis routines, and an addition of the external source routines.

In order to treat accurately the anisotropy of neutron scattering, the present code uses the multi-group double-differential form cross sections instead of the conventional Legendre expansion method. As a result, the energy-angle correlation can be accurately taken into account in a frame of multi-group approximation. The angular distribution after a collision can be continuously determined by sampling from the probability table. The difficulties of negative flux and ray effect encountered in the conventional P_1 method are eliminated in the present method.

In the analysis module, the point, surface cross, track length and collision estimators are implemented. This report contains the description of modification, input instruction, job control instruction, sample problem and auxiliary programs.

Keywords : Monte Carlo Method, MORSE-CG, Neutron Transport Code,
Cross Section Library, Double Differential Cross Section,
Input Data Format, Fixed Source, Tally

M O R S E - D D

多群二重微分型断面積ライブラリーを用いたモンテカルロコード

日本原子力研究所東海研究所原子炉工学部

中川 正幸・森 貴正

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モンテカルロコードM O R S E - C Gの改良版を作成した。本コードは特に、核融合炉のニュートロニクス計算で見られるような強い非等方散乱断面積を持つ物質中の中性子輸送計算を行うために開発した。主な修正は、断面積関係、評価ルーチン及び外部ソースルーチンの追加である。

中性子の非等方散乱を精度良く取扱うため、従来のルジャンドル展開法に代って、多群二重微分型断面積を用いる。その結果、エネルギーと散乱角の相関が多群近似の範囲で正確に考慮されている。散乱後の角度分布は、確率表から連続的にサンプルして決定する。この方法では、これまでの P_f 展開法では生じた負の中性子束や、ray effect は起らない。評価ルーチンでは、point, surface cross, track length, collision の各評価法が使用できる。

本報告には、修正した部分、入力形式、ジョブ制御言語、例題の入出力及び、補助ルーチンに関して記述されている。

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1. INTRODUCTION

The MORSE code¹⁾ has been extensively used in various fields of neutron and gamma transport calculations. Many modified versions have been developed to add some useful functions or to extend an availability. This document describes a recently developed version of the MORSE-CG code¹⁾. Main parts of modification in the present version are the modules concerning cross section treatment and external source routines. This version has been developed to be used mainly in the fusion neutronics calculations. In the original MORSE code, multigroup cross sections are used and an anisotropy of neutron scattering is presented by using the Legendre expansion (P_1) method. As encountered in fusion reactor shielding and blanket neutronics, the anisotropy of neutron scattering angular distribution plays an important role at the higher energy region. The conventional P_1 expansion method sometimes significantly mispredicts the neutron transport phenomena in the materials with highly anisotropic scattering cross sections. Moreover, conventional multi-group method can not accurately take account of the energy-angle correlated kinematics. In the MORSE code, the scattering angle is allowed only for discrete values. As a result, users sometimes encounter difficulties such as a negative flux and a ray effect.

At the present MORSE-DD code, the group to group scattering matrix is not presented by the Legendre expansion coefficients but directly given for many angular meshes such as a form of double differential cross sections (DDX). As a result, the difficulties mentioned above are not encountered and, furthermore users can expect more reliable results²⁾. In order to apply such multi-group cross sections, the subroutines handling cross sections and determining a flight direction after a collision have been modified as described

in chapter 2.

Various external source geometries have been installed in the code. Users can choose a single source cell and also do any combinations of them. In the analysis routines, the track length, surface cross and collision estimators are implemented other than point estimator. Such routines will be easily adopted by users.

The input instruction is given in Chapter 3. Since the most parts of input data are the same as those in the MORSE-CG code, details of such data are not described in this report. Users should refer the original report of MORSE-CG. The job control languages described in Chapter 4 are shown for FACOM/M380 machine which is compatible with IBM 360/370 series machines. The present version is written in FORTRAN 77 and can be easily compiled in FORTRAN H. In Chapter 6, input description of some auxiliary codes are given. These codes may be useful to process the output file from the MORSE run.

2. NEW FEATURES OF MORSE-DD

2.1 Cross Section

1) Multi-group double-differential form cross section

The collision source term in the transport equation is written as

$$\begin{aligned} q(r, \Omega, E) = & \int dE' \int d\Omega' \psi(r, \Omega', E') \sum_k N_k \{ \sigma_{el}^k (\Omega' \rightarrow \Omega, E' \rightarrow E) \\ & + \sum_i \sigma_{in,i}^k (\Omega' \rightarrow \Omega, E' \rightarrow E) + \sum_m m \sigma_{n,mn}^k (\Omega' \rightarrow \Omega, E' \rightarrow E) \\ & + \sum_x \sigma_{n,n'x}^k (\Omega' \rightarrow \Omega, E' \rightarrow E) \}, \end{aligned} \quad (1)$$

where σ^k denotes the microscopic differential cross section of nuclide k , and the subscript el stands for elastic scattering, in,i discrete and continuum level inelastic scattering, n,mn neutron multiplying reaction emitting m neutrons, $n, n'x$ neutron and charged particle emission reaction. We define the production cross section by

$$\sigma_{pr}(E') = \sigma_{el}(E') + \sum_i \sigma_{in,i}(E') + \sum_m m \sigma_{n,mn}(E') + \sum_x \sigma_{n,n'x}(E') \quad (2)$$

and

$$R_x(E') = \sigma_x(E') / \sigma_{pr}(E'). \quad (3)$$

Substituting Eqs.(2) and (3), Eq.(1) is rewritten as

$$\begin{aligned} q(r, \Omega, E) = & \int dE' \int d\Omega' \psi(r, \Omega', E') \sum_k N_k \sigma_{pr}^k(E') \sum_x R_x^k(E') \cdot \\ & P_x^k(\mu_o; E' \rightarrow E), \end{aligned} \quad (4)$$

where

$$\mu_o = \Omega \cdot \Omega'.$$

The energy-angle distribution function P_x is represented by using the Legendre polynomials in the conventional MORSE code. At the present code, we use a direct presentation of the distribution in energy and angle. In multi-group formulation, P_x is written as $P_x^k(\mu_o^j, g' \rightarrow g)$ which means a transfer probability from group g' to group g in angular interval μ_o^j . This matrix can be calculated basing on kinematics. These multi-group cross sections are produced by using the PROF-DD³⁾ code. The summation of reaction type x is reduced to the double differential form cross section (DDX)

$$I(\mu_o^j, g' \rightarrow g) = \sum_x R_x^{g'} P_x(\mu_o^j, g' \rightarrow g), \quad (5)$$

which includes a multiplication of neutrons emitted by (n, mn) reactions.

In MORSE-DD, the cross sections are read from a standard binary cross section tape as the same as the original MORSE code. The format of present cross section library is slightly different from the ANISN format.

The scattering cross section and the P_o component scattering matrix are replaced by the production cross section and the scattering angular interval, respectively. The macroscopic cross sections are calculated by mixing the microscopic cross sections with use of atomic number densities. The macroscopic scattering probability is calculated by

$$\bar{I}(\mu_j; g' \rightarrow g) = \sum_i N_i \sigma_{pr,g}^i I_i(\mu_j; g' \rightarrow g) / \sum_i N_i \sigma_{pr,g}^i, \quad (6)$$

and \bar{I} is renormalized as

$$\sum_j \bar{I}(\mu_j ; g' \rightarrow g) = 1. \quad (7)$$

As a result, the scattering matrix is calculated by

$$\sum_s (g' \rightarrow g) = \sum_j \sum_{pr}^{g'} \bar{I}(\mu_j ; g' \rightarrow g). \quad (8)$$

Since the scattering probability is given by the table as described above, the method to determine a flight direction after a collision is modified. In the original MORSE code, the flight direction is determined among the discrete directions which are obtained by using a generalized Gaussian quadrature. However, such procedure is not necessary if we use the present probability table. At the first, the secondary energy is determined by sampling from the probability table for energy distribution, $P(g' \rightarrow g)$

$$P(g' \rightarrow g) = \sum_s (g' \rightarrow g) / \sum_g \sum_s (g' \rightarrow g), \quad (9)$$

and then the scattering angle is determined by sampling from the cumulative probability table for angular distribution by using a random number in the range $0 < R < 1$. The cumulative probability is given by

$$P(\mu) = \int_{-1}^{\mu} d\mu_j \bar{I}(\mu_j ; g' \rightarrow g), \quad (10)$$

$$P(-1) = 0, \quad P(1) = 1.$$

This probability is presented by piece wise linear functions, hence the cosine of scattering angle μ can be continuously obtained. That is, a value of μ is uniquely determined corresponding to a random

number. Accordingly, the ray effect is not appeared, and also the negative flux or the negative weight problem is eliminated since the probability $P(\mu)$ is non negative in the range $-1 \leq \mu \leq 1$.⁴⁾

2) Cross Section Library

The multi-group cross sections have been produced from ENDF/B4⁵⁾ and JENDL3-PR1⁶⁾. The number of angular intervals is twenty in the range $-1 \leq \mu \leq 1$ ($\Delta\mu = 0.1$) and the number of energy group is 125. Note that the last group is assumed to be a thermal group and the cross section of 125 th group should be input by users if necessary. The slowing down cross section of 124 th group to 125 th group is given in the library. The group structure shown in Table 2.1 has been chosen to be appropriate for fusion reactor neutronics calculations. The identification number of nuclide is shown in Appendix.

The cross sections are read from unit 99 and written on unit IXTAPE and 96 after processed in the form which is directly used in the code. If users hope to use repeatedly these processed cross sections in another problems or in restarting the previous run, the processed data on units IXTAPE and 96 should be saved in order to reduce a computation time. The detail of format is described in Chapter 4.

2.2 External Source

The source routines are supplied by users in the MORSE code. The basic information to specify is geometry of source cell, direction of flight and energy spectrum. The types of sources installed in the present code are described below. If the source region consists of some unit cells, the relative intensity of each source cell, f_i should be input, and normalized as $\sum_i f_i = 1$. The coordinate of source particle

Table 2.1 Group structure of 125 group double differential form cross section library

GROUP	ENERGY RANGE	MULTIGROUP STRUCTURE		ENERGY RANGE	LETHARGY RANGE
		GROUP	LETHARGY RANGE		
1	1.6231E+07 - 1.6487E+07	-0.500 -	-0.484	37	7.3161E+06 - 7.7879E+06
2	1.5980E+07 - 1.6231E+07	-0.484 -	-0.469	38	6.8728E+06 - 7.3161E+06
3	1.5732E+07 - 1.5980E+07	-0.469 -	-0.453	39	6.4564E+06 - 6.8728E+06
4	1.5488E+07 - 1.5732E+07	-0.453 -	-0.437	40	6.0652E+06 - 6.4564E+06
5	1.5248E+07 - 1.5488E+07	-0.437 -	-0.422	41	5.6978E+06 - 6.0652E+06
6	1.5012E+07 - 1.5248E+07	-0.422 -	-0.406	42	5.3525E+06 - 5.6978E+06
7	1.4779E+07 - 1.5012E+07	-0.406 -	-0.391	43	5.0282E+06 - 5.3525E+06
8	1.4550E+07 - 1.4779E+07	-0.391 -	-0.375	44	4.7236E+06 - 5.0282E+06
9	1.4324E+07 - 1.4550E+07	-0.375 -	-0.359	45	4.4374E+06 - 4.7236E+06
10	1.4102E+07 - 1.4324E+07	-0.359 -	-0.344	46	4.1686E+06 - 4.4374E+06
11	1.3883E+07 - 1.4102E+07	-0.344 -	-0.328	47	3.9160E+06 - 4.1686E+06
12	1.3668E+07 - 1.3883E+07	-0.328 -	-0.312	48	3.6787E+06 - 3.9160E+06
13	1.3456E+07 - 1.3668E+07	-0.312 -	-0.297	49	3.4559E+06 - 3.6787E+06
14	1.3248E+07 - 1.3456E+07	-0.297 -	-0.281	50	3.2465E+06 - 3.4559E+06
15	1.3042E+07 - 1.3248E+07	-0.281 -	-0.266	51	3.0498E+06 - 3.2465E+06
16	1.2840E+07 - 1.3042E+07	-0.266 -	-0.250	52	2.8650E+06 - 3.0498E+06
17	1.2641E+07 - 1.2840E+07	-0.250 -	-0.234	53	2.6914E+06 - 2.8650E+06
18	1.2445E+07 - 1.2641E+07	-0.234 -	-0.219	54	2.5284E+06 - 2.6914E+06
19	1.2252E+07 - 1.2445E+07	-0.219 -	-0.203	55	2.3752E+06 - 2.5284E+06
20	1.2062E+07 - 1.2252E+07	-0.203 -	-0.187	56	2.2313E+06 - 2.3752E+06
21	1.1875E+07 - 1.2062E+07	-0.187 -	-0.172	57	2.0961E+06 - 2.2313E+06
22	1.1691E+07 - 1.1875E+07	-0.172 -	-0.156	58	1.9691E+06 - 2.0961E+06
23	1.1510E+07 - 1.1691E+07	-0.156 -	-0.141	59	1.8498E+06 - 1.9691E+06
24	1.1331E+07 - 1.1510E+07	-0.141 -	-0.125	60	1.7377E+06 - 1.8498E+06
25	1.1156E+07 - 1.1331E+07	-0.125 -	-0.109	61	1.5335E+06 - 1.7377E+06
26	1.0983E+07 - 1.1156E+07	-0.1156E+07 -0.109 -	-0.094	62	1.3533E+06 - 1.5335E+06
27	1.0812E+07 - 1.0983E+07	-0.094 -	-0.078	63	1.1943E+06 - 1.3533E+06
28	1.0645E+07 - 1.0812E+07	-0.078 -	-0.063	64	1.0540E+06 - 1.1943E+06
29	1.0480E+07 - 1.0645E+07	-0.063 -	-0.047	65	9.3013E+05 - 1.0540E+06
30	1.0317E+07 - 1.0480E+07	-0.047 -	-0.031	66	8.2084E+05 - 9.3013E+05
31	1.0157E+07 - 1.0317E+07	-0.031 -	-0.016	67	7.2438E+05 - 8.2084E+05
32	9.9999E+06 - 1.0157E+07	-0.016 -	-0.000	68	6.3927E+05 - 7.2438E+05
33	9.3940E+06 - 9.9999E+06	-0.000 -	0.063	69	5.6415E+05 - 6.3927E+05
34	8.8249E+06 - 9.3940E+06	0.063 -	0.125	70	4.9786E+05 - 5.6415E+05
35	8.2902E+06 - 8.8249E+06	0.125 -	0.188	71	4.3936E+05 - 4.9786E+05
36	7.7879E+06 - 8.2902E+06	0.188 -	0.250	72	3.8774E+05 - 4.3936E+05

Table 2.1 (Cont'd.)

GROUP	ENERGY RANGE	MULTIGROUP STRUCTURE GROUP	LETHARGY RANGE	ENERGY RANGE		LETHARGY RANGE
				100	7.1016E+03 - 9.1186E+03	
73	3.4217E+05 - 3.8774E+05	3.250 - 3.375	100	5.5307E+03 - 7.1016E+03	7.250 - 7.500	
74	3.0197E+05 - 3.4217E+05	3.375 - 3.500	101	5.5307E+03 - 7.1016E+03	7.250 - 7.500	
75	2.6649E+05 - 3.0197E+05	3.500 - 3.625	102	4.3073E+03 - 5.5307E+03	7.500 - 7.750	
76	2.3517E+05 - 2.6649E+05	3.625 - 3.750	103	3.3546E+03 - 4.3073E+03	7.750 - 8.000	
77	2.0754E+05 - 2.3517E+05	3.750 - 3.875	104	2.6125E+03 - 3.3546E+03	8.000 - 8.250	
78	1.8315E+05 - 2.0754E+05	3.875 - 4.000	105	2.0346E+03 - 2.6125E+03	8.250 - 8.500	
79	1.6163E+05 - 1.8315E+05	4.000 - 4.125	106	1.5846E+03 - 2.0346E+03	8.500 - 8.750	
80	1.4264E+05 - 1.6163E+05	4.125 - 4.250	107	1.2341E+03 - 1.5846E+03	8.750 - 9.000	
81	1.2588E+05 - 1.4264E+05	4.250 - 4.375	108	9.6110E+02 - 1.2341E+03	9.000 - 9.250	
82	1.1109E+05 - 1.2588E+05	4.375 - 4.500	109	5.8293E+02 - 9.6110E+02	9.250 - 9.500	
83	9.8035E+04 - 1.1109E+05	4.500 - 4.625	110	3.5357E+02 - 5.8293E+02	9.750 - 10.250	
84	8.6515E+04 - 9.8035E+04	4.625 - 4.750	111	2.1445E+02 - 3.5357E+02	10.250 - 10.750	
85	7.6349E+04 - 8.6515E+04	4.750 - 4.875	112	1.3007E+02 - 2.1445E+02	10.750 - 11.250	
86	6.7378E+04 - 7.6349E+04	4.875 - 5.000	113	7.8891E+01 - 1.3007E+02	11.250 - 11.750	
87	5.9461E+04 - 6.7378E+04	5.000 - 5.125	114	4.7850E+01 - 7.8891E+01	11.750 - 12.250	
88	5.2474E+04 - 5.9461E+04	5.125 - 5.250	115	2.9023E+01 - 4.7850E+01	12.250 - 12.750	
89	4.6308E+04 - 5.2474E+04	5.250 - 5.375	116	1.7603E+01 - 2.9023E+01	12.750 - 13.250	
90	4.0867E+04 - 4.6308E+04	5.375 - 5.500	117	1.0677E+01 - 1.7603E+01	13.250 - 13.750	
91	3.6065E+04 - 4.0867E+04	5.500 - 5.625	118	6.4758E+00 - 1.0677E+01	13.750 - 14.250	
92	3.1827E+04 - 3.6065E+04	5.625 - 5.750	119	3.9278E+00 - 6.4758E+00	14.250 - 14.750	
93	2.8087E+04 - 3.1827E+04	5.750 - 5.875	120	2.3823E+00 - 3.9278E+00	14.750 - 15.250	
94	2.4787E+04 - 2.8087E+04	5.875 - 6.000	121	1.4449E+00 - 2.3823E+00	15.250 - 15.750	
95	2.1874E+04 - 2.4787E+04	6.000 - 6.125	122	8.7640E-01 - 1.4449E+00	15.750 - 16.250	
96	1.9304E+04 - 2.1874E+04	6.125 - 6.250	123	5.3156E-01 - 8.7640E-01	16.250 - 16.750	
97	1.5034E+04 - 1.9304E+04	6.250 - 6.500	124	3.2241E-01 - 5.3156E-01	16.750 - 17.250	
98	1.1709E+04 - 1.5034E+04	6.500 - 6.750	125	1.0010E-05 - 3.2241E-01	17.250 - 27.630	
99	9.1186E+03 - 1.1709E+04	6.750 - 7.000				

is denoted by (X_s, Y_s, Z_s) , and random numbers are by R_1, R_2 and R_3 which are in the range $0 < R < 1$.

(1) Geometry of Source Cell and Distribution of Source Particle Density

1. Point

$$X_s = X_0, \quad Y_s = Y_0, \quad Z_s = Z_0$$

2. Rectangular Parallelepiped with Uniform Distribution

Each side should be parallel to coordinate axes. The averaged particle density is constant in $XL \leq x \leq XR$, $YL \leq y \leq YR$, $ZL \leq z \leq ZR$,

$$X_s = R_1 \cdot (XR - XL) + XL,$$

$$Y_s = R_2 \cdot (YR - YL) + YL,$$

$$Z_s = R_3 \cdot (ZR - ZL) + ZL.$$

3. Rectangular Parallelepiped with Cosine Distribution

The averaged particle density is the cosine distribution along x , y , and z direction.

$$X_s = \frac{1}{\pi} \cdot (XR - XL) \cdot \{\sin^{-1}(2R_1 - 1) + \frac{\pi}{2}\} + XL,$$

$$Y_s = \frac{1}{\pi} \cdot (YR - YL) \cdot \{\sin^{-1}(2R_1 - 1) + \frac{\pi}{2}\} + YL,$$

$$Z_s = \frac{1}{\pi} \cdot (ZR - ZL) \cdot \{\sin^{-1}(2R_1 - 1) + \frac{\pi}{2}\} + ZL.$$

4. Cylinder with Uniform Distribution

The averaged particle density is constant in $0 \leq r \leq XR$, $ZL \leq z \leq ZR$.

The axis of cylinder should be parallel to z axis.

$$Y_s = XR \cdot \sqrt{R_1} \cdot \cos\theta + X_0,$$

$$Y_s = XR \cdot \sqrt{R_1} \cdot \sin\theta + Y_0,$$

$$Z_s = (ZR - ZL) \cdot R_3 + ZL,$$

where $\theta = 2\pi \cdot R_2$

5. Cylinder with Cosine Distribution along z Axis

The averaged particle density is constant in (x,y) plane and cosine distribution along z direction.

$$X_s = XR \cdot \sqrt{R_1} \cdot \cos\theta + X_0 ,$$

$$Y_s = XR \cdot \sqrt{R_1} \cdot \sin\theta + Y_0 ,$$

$$Z_s = \frac{1}{\pi} \cdot (ZR - ZL) \cdot \{\sin^{-1}(2R_2 - 1) + \frac{\pi}{2}\} + ZL .$$

6. Sphere and Spherical Shell with Uniform Distribution

The averaged particle density is uniform in $0 \leq r \leq XR$ (sphere) or $XL \leq r \leq XR$ (spherical shell).

$$\left. \begin{array}{l} X_s = XR \cdot R_1^{1/3} \cdot R_2 + X_0 , \\ Y_s = XR \cdot R_1^{1/3} \cdot \sqrt{1-R_2^2} \cdot \cos\{(2R_3 - 1)\pi\} + Y_0 \\ Z_s = XR \cdot R_1^{1/3} \cdot \sqrt{1-R_2^2} \cdot \sin\{(2R_3 - 1)\pi\} + Z_0 \end{array} \right\} \text{sphere}$$

$$\left. \begin{array}{l} X_s = \{(XR^3 - XL^3) R_1 + XL^3\}^{1/3} \cdot R_2 + X_0 \\ Y_s = \{(XR^3 - XL^3) R_1 + XL^3\}^{1/3} \cdot \sqrt{1-R_2^2} \cdot \cos\theta\{(2R_3 - 1)\pi\} + Y_0 \\ Z_s = \{(XR^3 - XL^3) R_1 + XL^3\}^{1/3} \cdot \sqrt{1-R_2^2} \cdot \sin\theta\{(2R_3 - 1)\pi\} + Z_0 \end{array} \right\} \text{spherical cell}$$

7. Circular Plane with Uniform Distribution

The averaged particle density is uniform on a plane of $0 \leq r \leq XR$.

The plane should be perpendicular to z axis.

$$X_s = XL \cdot \sqrt{R_1} \cdot \cos\theta + X_0 ,$$

$$Y_s = XL \cdot \sqrt{R_1} \cdot \sin\theta + Y_0 ,$$

$$z_s = z_0 ,$$

where $\theta = 2\pi \cdot R_2$.

8. Cylindrical Shell with Uniform Distribution

The averaged particle density is uniform in $XL \leq r \leq R$, $ZL \leq Z \leq ZR$.

The axis of cylinder should be parallel to z axis.

$$\begin{aligned} x_s &= \{(XR^2 - XL^2)R_1 + XL^2\}^{1/2} \cdot \cos\theta + X_0, \\ y_s &= \{(XR^2 - XL^2)R_1 + XL^2\}^{1/2} \cdot \sin\theta + Y_0, \\ z_s &= (ZR - ZL) \cdot R_2 + ZL, \end{aligned}$$

where $\theta = 2\pi \cdot R_3$

(2) Flight Direction of Source Particle

Users can choose a probability distribution of flight direction among the following options. The direction cosines to x, y, z coordinates, U_s , V_s and W_s are determined by using random numbers.

1. Mono-directional Distribution

$$\begin{aligned} U_s &= UINP , \\ V_s &= VINP , \\ W_s &= WINP . \end{aligned}$$

2. Isotropic Distribution

$$\begin{aligned} U_s &= 2 \cdot R_1 - 1 , \\ \phi &= 2\pi \cdot R_2 , \\ V_s &= \sqrt{1-U_s^2} \cdot \cos \phi , \\ W_s &= \sqrt{1-U_s^2} \cdot \sin \phi . \end{aligned}$$

3. Cosine Distribution against z Direction

$$W_s = \sqrt{R_1} , \quad \text{for NEMIS} = 3$$

$$w_s = -\sqrt{R_1} , \quad \text{for NEMIS} = -3$$

$$\phi = 2\pi \cdot R_x ,$$

$$U_s = \sqrt{1-w_s^2} \cdot \cos \phi ,$$

$$V_s = \sqrt{1-w_s^2} \cdot \sin \phi .$$

4. Arbitrary Distribution

The probability distributions in polar and azimuthal angles, ANGDIS(I,J) and FAIDIS(J), respectively should be given by the user in input data as a function of angle and energy. After determining the energy group I1, the interval of polar angle D (J1) is sampled from the cumulative probability function as

$$\sum_{J=1}^{J1} \text{ANGDIS}(I1, J) > R_1 .$$

By using the random number R_1 ,

$$w_s = \frac{\sum_{J=1}^{J1} \text{ANGDIS}(I1, J) - R_1}{\text{ANGDIS}(I1, J1)} \cdot \{ \text{DEG}(J1) - \text{DEG}(J1+1) \}$$

$$+ \text{DEG} (J1+1) .$$

Similarly, the interval of azimuthal angle is sampled, and U_s and V_s are determined as follows;

$$\phi = \frac{\sum_{K=1}^{K1} \text{FAIDIS}(K) - R_2}{\text{FAIDIS}(K1)} \cdot \{ \text{FAI}(K1) - \text{FAI}(K1+1) \} ,$$

$$+ \text{FAI}(K1+1)$$

$$U_s = \sqrt{1-w_s^2} \cdot \cos \phi ,$$

$$V_s = \sqrt{1-w_s^2} \cdot \sin \phi .$$

5. Cosine Distribution against Radial Direction

At first, the flight direction U'_s , V'_s and W'_s are determined against z axis, and rotated to the radial direction as

$$\Omega(U_s, V_s, W_s) = \Pi \cdot \Omega'(U'_s, V'_s, W'_s),$$

where Π is the matrix to transfer angles.

(3) Energy Distribution of Source Particle

1. Arbitrary Stepwise Distribution

The probability distribution should be given by the user for energy bins, which is read in by subroutine SORIN.

2. Watt's Fission Spectrutm

$$f(E) \propto e^{-E/a} \sinh(\sqrt{bE}),$$

where a and b are constants.

3. Arbitrary Distribution

The probability distribution should be given by the user for energy points. The source energy is determined by linearly interpolating in log-log or in linear-linear between the input energy points.

4. Mono-Energy Distribution

The source particle has a constant energy.

2.3 Tallies

MORSE-DD provides five standard tallies, but these basic tallies can be easily modified by the user. All tallies are normalized to be per one starting particle. They are also a function of energy and angle. The tally data are edited on a user file, so the user can process them into interesting quantities (e.g., reaction rate) by using auxiliary programs.

(1) Flux at a Point

The flux at a detector consists of the contribution to a detector directly from the source and the probability of a contribution to a detector every time a particle has a collision, which is written as follow;

$$\begin{aligned}\phi(r_d, \Omega) = & \int dr' S(P', \Omega) \frac{1}{|r' - r_d|^2} \exp(-\beta(r', r_d)) \\ & + \int dr' \int d\Omega' \sum_s P(\Omega' \rightarrow \Omega) \delta(\Omega, \frac{r' - r_d}{|r' - r_d|}) \phi(r', \Omega') \cdot \\ & \frac{1}{|r' - r_d|^2} \exp(-\beta(r', r_d)),\end{aligned}$$

where

$$\beta(r', r_d) = \int_0^{|r' - r_d|} \sum_t (r' - R) \frac{(r' - r_d)}{|r' - r_d|} dR.$$

MORSE-DD calculates the flux basing on the following formula,

$$\begin{aligned}\phi(r_d, \Omega) = & \sum_i w_i p_i \frac{1}{|r_i - r_d|^2} \exp(-\beta(r_i, r_d)) \\ & + \sum_j w_j \frac{\sum_s (r_i)}{\sum_t (r_i)} p(\mu) \frac{1}{|r_j - r_d|^2} \exp(-\beta(r_j, r_d)),\end{aligned}$$

where

w_i = particle weight born at the source,
 w_j = particle weight before a collision,
 P_i = probability of particle being born into the solid angle $d\Omega$.
 $P(\mu)$ = probability that a scattering angle will be μ .

The first term is called as a uncollided flux and the second one as a last flight or next event estimator. As seen from the above expression, this scheme has an $1/R^2$ singularity (R is the distance between collision and detector) for a detector in a scattering medium which makes the theoretical variance of the estimator infinite. The probability P_i is determined from the angular distribution of the source particle defined in the source routines. The probability $P(\mu)$ is calculated by using the scattering probability matrix Eq.(6) as follows;

$$P(\mu) = \frac{1}{2\pi} \frac{I(\mu_k, g' \rightarrow g)}{\Delta \mu_k},$$

where $g' \rightarrow g$ = energy transfer from group g' to group g ,
 $\Delta \mu_k$ = width of k th angular interval.

It is recommended that the point estimator should be used in regions of little scattering (e.g., air) or in voids because a large-weighted particle interacting arbitrarily close to the detector can severely perturb both the detector result and its variance due to the $1/R^2$ singularity. To cope with this difficulty, the code can apply the bounded sphere approximation in the neighbouring region of the detector.

If we can assume that the scattering is isotropic and uniform in a spherical region surrounding the detector point, the $1/R^2$ singularity can be removed by scoring $\exp(-\sum r)/4\pi c^2$ instead of $\exp(-\sum_t r)/4\pi r^2$

within this sphere. In the code, C is approximately given by the formula,

$$c^2 = a^2 \left[\frac{2}{\lambda^2} + \left(1 + \frac{2}{\lambda}\right) / (1 - e^{-\lambda}) \right],$$

where a = radius of sphere,

$$\lambda = \sum_t a_t.$$

For a typical problem, the value of a is on the order of $0.1 \sim 0.5$ in a mean free path unit.

When a point estimator is used, the computation time rapidly increases with increasing the number of detectors. To reduce the computation time, the code can use the probabilistic sampling technique outside an imaginary sphere of a specified radius R_{max} , if the real collision occurs outside the sphere, this event is scored with the probability $(R_{max}/R)^2$. For the scored event, the weight of particle is increased by a factor of $(R_{max}/R)^2$. This technique is effective for a large system.⁷⁾

(2) Flux and Current across a Surface

The current between angles μ_1 and μ_2 over a surface S is given by

$$\psi = \int_S ds \int_{\mu_1}^{\mu_2} d\mu \int_0^{2\pi} d\phi \Phi(r, \mu, \phi) \mu,$$

where

μ = cosine of polar angle between normal direction of surface and particle flight direction

ϕ = azimuthal angle

The angular flux integrated over S is given by

$$\Phi(s, \mu_i) = \int_S ds \int_{\mu_{i-1}}^{\mu_i} d\mu \int_0^{2\pi} d\phi \{ \mu \Phi(r, \mu, \phi) \} \frac{1}{\mu}$$

The flux in the cosine bin μ_i is obtained by scoring the particle j crossing S as

$$\bar{\phi}(\mu_i) = \sum_j w_j / \mu_j / S ,$$

and the current tally is given by

$$\bar{\psi}(\mu_i) = \sum_j w_j / S .$$

Note that the current tally is not the net current of nuclear reactor theory, but equal to the number of particles crossing S .

(3) Track Length Estimate

The flux is calculated by summing the track length travelled in the specified region as follows;

$$\phi = \sum_i w_i \cdot l_i / V ,$$

where l_i = track length of particle i

w_i = weight of particle i

V = volume of the specified region

The track length estimator is very reliable for both the detector result and its variance.

(4) Collision Estimate

The collision estimator scores $W \cdot \frac{1}{\sum_t}$ at each collision event that occurs within the specified region V , that is, the flux is calculated by

$$\phi = \sum_i w_i \cdot \frac{1}{\sum_{t,i}} / V .$$

The collision estimator tends to suffer statistically in optically

thin regions since few collisions occur there. This statistical problem is usually improved by using the track length estimator. The surface crossing, track length and collision estimators all tend to suffer from increasing statistical errors as the volume of the detector region becomes arbitrarily small. The next event (point detector) estimator is a candidate for such problems. The user should be careful in choosing effective estimators for his problem.

3. INPUT INSTRUCTIONS

3.1 Random Walk Input Instructions

The input read by subroutine INPUT1 is as follows:

CARD A (20A4)

Title card.

(Any character other than a blank or alphabetic in column one will terminate the job.)

CARD B (10I5, F5.0,2I5)

- NSTRT - number of particles per batch.
- NMØST - maximum number of particles allowed for in the bank(s); may equal NSTRT if no splitting, fission, and secondary generation.
- NITS - number of batches.
- NQUIT - number of sets of NITS batches to be run without calling subroutine INPUT.
- NGPQTN - number of neutron groups being analyzed.
- NGPQTG - number of gamma-ray groups being analyzed. When DDX cross section is used, NGPQTG should be equal to 0.
- NMGP - number of primary particle groups for which cross sections are stored; should be same as NGP (or the same as NGG when NGP = 0) on Card XB read by subroutine XSEC.
- NMTG - total number of groups for which cross sections are stored; should be same as NGP+NGG as read on Card XB read by subroutine XSEC.
- NCØLTP - set greater than zero if a collision tape is desired; the collision tape is written by the user routine BANKR.
- IADJM - set greater than zero for an adjoint problem.
- AXTIM - maximum clock time in minutes allowed for the problem to be on the computer (c.p.u. time); e.g., 4.5 entered here allows 4 and 1/2 minutes.
- MEDIA - number of cross-section media; should agree with NMED on Card XB read by subroutine XSEC.
- MEDALB - albedo scattering medium is absolute value of MEDALB; if MEDALB = 0, no albedo information to be read in MEDALB < 0, albedo only problem - no cross sections are to be read, MEDALB > 0, coupled albedo and transport problem.

CARD C (4I5, 5E10.5)

ISØUR - source energy group if > 0 ,
 if $ISØUR < 0$ or if $ISØUR = 0$ and NGPFS $\neq 0$, SØRIN
 is called for input of Cards E1 and E2.

NGPFS - number of groups for which the source spectrum is to
 be defined. If $ISØUR < 0$, $NGPFS \geq 2$.

ISBIAS - no source energy biasing if set equal to zero; otherwise
 the source energy is to be biased, and Cards E2
 are required.

NØTUSD - an unused variable.

WTSTRT - weight assigned to each source particle.

EBØTN - lower energy limit of lowest neutron group (eV)
 (group NMGP).

ENØTG - lower energy limit of lowest gamma-ray group (eV)
 (group NMTGO).

TCUT - age in sec at which particles are retired; if $TCUT = 0$,
 no time kill is performed.

VELTH - velocity of group NMGP when $NGPQT M > 0$; i.e., thermal-
 neutron velocity (cm/sec).

CARD D (7E10.4)

XSTRRT	}	coordinates for source particles.
YSTRRT		
ZSTRRT		
AGSTRRT	- starting age for source particles.	
UINP	}	source particle direction cosines if all are zero, isotropic directions are chosen.
VINP		
WINP		

Source data on Cards C and D will be overridden by any changes in subroutine SØINP. Subroutine SØINP is called from subroutine SØURCE.

CARDS E1 (7E10.4) (Omit if $ISØUR$ on Card C > 0 or if $ISØUR = NGPFS = 0$)

NGPFS values of FS, where FS equals the unnormalized fraction of source particles in each group.

CARDS E2 (7E10.4) (Omit if $ISØUR > 0$ or if $ISØUR \leq 0$ and $ISBIAS = 0$)

If $ISBIAS > 0$, NGPFS values of BFS, the relative importance of a source in group I, are required.

CARDS F (7E10.4)

NMTG values of ENER, the energies (in eV) at the upper edge of the energy group boundaries.

NOTE: The lower energies of groups NMGP and NMTG were read on Card C.

CARD G (215,5X,36I1,5X,13I1) (Omit if NCOLTP on Card B ≤ 0)

NHISTR - logical tape number for the first collision tape.

NHISMX - the highest logical number that a collision tape may be assigned.

NBIND(J), J=1, 36 - an index to indicate the collision parameters to be written on tape.

NCOLLS(J), J=1, 13 - an index to indicate the types of collisions to be put on tape.

(See tables 3.2 for information concerning NBLND.)

CARD H (Z12), on IBM-360; (020) on CDC-6600

RAND0M - starting random number.

In restart run, this value is replaced by the value read from the restart tape.

CARD I (7I5)

NSPLT - index indicating that splitting is allowed if > 0 .

NKILL - index indicating that Russian roulette is allowed if > 0 .

NPAST - index indicating that exponential transform is invoked if > 0 (subroutine DIREC required).

NOLEAK - index indicating that non-leakage is invoked if > 0 .

IEBIAS - index indicating that energy biasing is allowed if > 0 .

MXREG - number of regions described by geometry input (will be set to one if ≤ 0).

MAXGP - group number of last group for which Russian roulette or splitting or exponential transform is to be performed. For adjoint, set = NMTG or overstoring results.

CARD J (6I5, 4E10.5) (Omit if NSPLT + NKILL + NPAST = 0)

NGP1 } from energy group NGP1 to energy group NGP2, inclusive, in
 NDG } steps of NDG and from region NRG1 to NRG2, inclusive, in
 NGP2 } steps of NDRG, the following weight standards and path-
 NRG1 } stretching parameters are assigned. If NGP1 = 0, groups 1
 NDRG } to MAXGP will be used; if NRG1 = 0, regions 1 to MXREG
 NRG2 } will be used (both in steps of one). Usually NDG = 1 and
 NDRG = 1.

WTHIHI - weight above which splitting will occur.

WTLOW1 - weight below which Russian roulette is played.

Table 3.1 Variables That May Be Written on Tape (NBIND)

J	Variable*	J	Variable
1	NCØLL	19	WTBC
2	NAME	20	ETAUSD
3	IG	21	ETA
4	U	22	AGE
5	V	23	ØLDAGE
6	W	24	NREG
7	X	25	NMED
8	Y	26	NAMEX
9	Z	27	WATEF
10	WATE	28	BLZNT
11	IGØ	29	BLZØN
12	UØLD	30	VEL(IG)
13	VØLD	31	VEL(IGØ)
14	WØLD	32	TSIG
15	XØLD	33	PNAB
16	YØLD	34	NXTRA
17	ZØLD	35	EXTRAI
18	ØLDWT	36	EXTRA2

WTAVE1 - weight given those particles surviving Russian roulette.
 PATH - path-length stretching parameters for use in exponential transform (usually $0 \leq \text{PATH} < 1$).

End Cards J with negative value of NGPl (ex., -1 in columns 4 and 5).

CARDS K (7E10.4) (Omit if IEBIAS on Card I ≤ 0).

((EPRØB(IG,NREG), IG = 1, NMTG), NREG = 1, MXREG)

Values of the relative energy importance of particles leaving a collision in region NREG. Input for each region must start on a new card.

CARD L (4I5)

NSØUR - set ≤ 0 for a fixed source problem; otherwise the source is from fissions generated in a previous batch.
 MFISTR - index for fission problem, if ≤ 0 no fissions are allowed.
 NKCALC - the number of the first batch to be included in the estimate of k; if ≤ 0 no estimate of k is made.
 NØRMF - the weight standards and fission weights are unchanged if ≤ 0 ; otherwise fission weights will be multiplied, at the end of each batch, by the latest estimate of k and the weight standards are multiplied by the ratio of fission weights produced in previous batch to the average starting weight for the previous batch. For time-dependent decaying systems, NØRMF should be > 0 .

CARDS M (7E10.4) (Omit if MFISTP on Card L ≤ 0)

(FWLØ(I), I = 1, MXREG) values of the weight to be assigned to fission neutrons.

CARDS N (7E10.4) (Omit if MFISTP on Card L ≤ 0)

(FSE(IG,IMED), IG = 1, NMTG), IMED = 1, MEDIA) the fraction of fission-induced source particles in group IG and medium IMED.

NOTE: Input for each medium must start on a new card.

CARDS O (7E10.5) (Omit if NGPQTN = 0 or NGPQTG = 0, i.e., include if coupled neutron-gamma-ray problem)

((GWLØ(IG,NREG) IG = 1, NMGP or NMTG - NMGP), NREG = 1, MXREG) - values of the probability of generating a gamma ray. NMGP groups are read for each region in a forward problem and NMTG-NMGP for an adjoint. Input for each region must start on a new card.

3.2 Combinatorial Geometry Input Instructions

The combinatorial geometry input data is read by the JØMIN subroutine, except for the region volumes VNØR(I), which are read by the GTVLIN subroutine whenever IVØPT = 3. For clarity of terminology, the terms "regions" and "media" have essentially the same meaning as in the Ø5R Geometry Package, but are constructed in a different manner. The term "zone" is the same as the "region" as defined in the original combinatorial geometry package. The term "body" has the same meaning as in the original combinatorial geometry package.

CARD CGA (2I5,10X,10A6)

- IVØPT - option which defines the method by which region volumes are determined; if
 - IVØPT = 0, volumes set equal to 1,
 - IVØPT = 1, concentric sphere volumes are calculated,
 - IVØPT = 2, slab volumes (1-dim.) are calculated,
 - IVØPT = 3, volumes are input by card.
- IDBG - if IDBG > 0, subroutine PR is called to print results of combinatorial geometry calculations during execution. Use only for debugging.
- JTY - alphanumeric title for geometry input (columns 21-80).

CARDS CGB (2X,A3,1X,I4,6D10.3)

One set of CGB cards is required for each body and for the END card (see Table 3.2). Leave columns 1-6 blank on all continuation cards.

- ITYPE - specifies body type or END to terminate reading of body data (for example BØX, RPP, ARB, etc.). Leave blank for continuation cards.
- IALP - body number assigned by user (all input body numbers must form a sequence set beginning at 1). If left blank, numbers are assigned sequentially. Either assign all or none of the numbers. Leave blank for continuation cards.
- FPD(I) - real data required for the given body as shown in Table 3.2. This data must be in cm.

CARDS CGC (2X,A3,I5,9(A2,I5))

Input zone specification cards. One set of cards required for each input zone, with input zone numbers being assinged sequentially.

- IALP - IALP must be a nonblank for the first card of each set of cards defining an input zone. If IALP is blank, this card is treated as a continuation of the previous zone card.
 IALP - END denotes the end of zone description.
- NAZ - total number of zones that can be entered upon leaving any of the bodies defined for this input region (some zones may be counted more than once). Leave blank for continuation cards for a given zone. (If NAZ \leq 0 on the first card of the zone card set, then it is set to 5). This is used to allocate blank common.
 Alternate IIBIAS(I) and JTY(I) for all bodies defining this input zone.
- IIBIAS(I) - specify the "OR" operator if required for the JTY(I) body.
- JTY(I) - body number with the (+) or (-) sign as required for the zone description.

CARDS CGD (14I5)

- MRIZ(I) - MRIZ(I) is the region number in which the "Ith" input zone is contained ($I = 1$, to the number of input zones). Region numbers must be sequentially defined from 1.

CARDS CGE (14I5)

- MMIZ(I) - MMIZ(I) is the medium number in which the "Ith" input zone is contained ($I = 1$, to the number of input zones). Medium numbers must be sequentially defined from 1.

CARDS CGF (7D10.5) (Omit if IVOPT \neq 3)

- VNOR(I) - volume of the "Ith" region ($I = 1$ to MXREG, the number of regions).

Table 3.2 Input Required on CGB Cards for Each Body Type

Card Columns Body Type	I TYPE 3-5	I ALP	Real Data 11-20	Defining 21-30	Particular Body 31-40	51-50	61-70	Number of Cards Needed
Box	B@X	I ALP is assigned by the user or by the code if left blank.	Vx H2x Xmin	Vy H2y Xmax	Vz H2z Ymin	H1x H3x Ymax	H1y H3y Zmin	H1z H3z Zmax
Right Parallel- piped	RPP							1 of 2 2 of 2 1
Sphere	SPH		Vx	Vy	Vz	R	-	-
Right Circular Cylinder	RCC		Vx R	Vy -	Vz -	Hx -	Hy -	Hx -
Right Elliptic Cylinder	REC		Vx R1x	Vy R1y	Vz R1z	Hx R2x	Hy R2y	Hx R2z
Ellipsoid	ELL		V1x L	V1y -	V1z -	V2x -	V2y -	V2z -
Truncated Right Cone	TRC		Vx L1	Vy L2	Vz -	Hx -	Hy -	Hx -
Right Angle Wedge	WED or RAW		Vx H2x	Vy H2y	Vz H2z	H1x H3x	H1y H3y	H1z H3z
								1 of 2 2 of 2

Table 3.2 (Cont'd.)

Card Columns Body Type	ITYPE	IALP	11-20	Real Data 21-30	Defining 31-40	Particular Body 41-50	51-60	61-70	Number of Cards Needed
Arbitrary	ARB		V1x	V1y	V1z	V2x	V2y	V2z	1 of 5
Polyhedron			V3x	V3y	V3z	V4x	V4y	V4z	2 of 5
			V5x	V5y	V5z	V6x	V6y	V6z	3 of 5
			V7x	V7y	V7z	V8x	V8y	V8z	4 of 5
									5 of 5
									Face Descriptions (see note below)

NOTE: Card 5 of the arbitrary polyhedron input contains a four-digit number for each of the six faces of an ARB body. The format is 6D10.3, beginning in column 11.

3.3 MØRSEC - Cross-Section Module Input Instructions

CARD XA (20A4)

Title card for cross sections. This title is also written on tape if a processed tape is written; therefore, it is suggested that the title be definitive.

CARD XB (13I5)

- NGP - the number of primary groups for which there are cross sections to be stored. Should be same as NMGP input in MØRSE.
- NDS - number of primary downscatters for NGP (usually NGP).
- NGG - number of secondary groups for which there are cross sections to be stored.
- NDSG - number of secondary downscatters for NGG (usually NGG).
- INGP - total number of groups for which cross sections are to be input.
- ITBL - table length, i.e., the number of cross sections for each group (usually equal to number of downscatters + number of upscatters + 3).
- ISGG - location of within-group scattering cross sections (usually equal to number of upscatters + 4).
- NMED - number of media for which cross sections are to be stored - should be same as MEDIA input in MØRSE.
- NELEM - number of elements for which cross sections are to be read.
- NMIX - number of mixing operations (elements times density operations) to be performed (must be ≥ 1).
- NCØEF - number of coefficients for each element, including P_0 . Set 1 if DDX mode is used.
- NSCT - number of discrete angles (usually NCOEF/2_{integral} when IDDX = 0). number of angular interval (when IDDX = 1).
- ISTAT - flag to store Legendre coefficients if greater than zero.
- IDDX - 0/1 = P_ℓ expansion mode/DDX mode.

CARD XC (11I5)

- IRDSG[†] - switch to print the cross sections as they are read if > 0 .

- ISTR† - switch to print cross sections as they are stored if
 > 0. No output for cross section if ISTR < 0 and
 IDDX = 1.
- IFMU† - switch to print intermediate results of μ 's calculation
 if > 0.
- IMOM† - switch to print moments of angular distribution if > 0.
- IPRINT† - switch to print angles and probabilities if > 0.
- IPUNT† - switch to print results of bad Legendre coefficients
 if > 0.
- IDTF† - switch to signal that input format is DTF-IV format
 if > 0; otherwise, ANISN format is assumed.
- IXTAPE - logical tape unit if binary cross section (ANISN-type),
 set equal to 0 if cross sections are from cards.
 If DDX = 1, DDX library is read from Unit 99 and the
 processed ANISN-type cross sections are written on unit
 IXTAPE. If negative, the processed cross sections and
 other necessary data from previous run will be read.
 In this case (IXTAPE < 0), no cross sections from cards
 and no mixing cards may be input when IDDX = 0, while
 all cross section data (element ID and mixing card etc.)
 are required when IDDX = 1. The absolute value of IXTAPE
 is the logical unit for the processed ANISN-type corss
 sections. The additional processed DDX library is read
 from unit 96.
- JXTAPE - logical tape unit of a processed cross section tape to
 be written (IDDX = 0). This processed tape will contain
 the title card, the variables from common LØCSIG and the
 pertinent cross sections from blank common. JXTAPE is
 dummy input for DDX mode. In the case of DDX mode, logical
 units of processed cross sections to be written are units
 96 and IXTAPE.
- IØ6RT - logical tape unit of a point cross-section tape in 06R
 format.
- IGOPT - last group (MØRSE multigroup structure) for which the
 06R point cross sections are to be used (\leq NMGP).

† Switches are ignored if IXTAPE < 0.

CARD XC' (7E10.5) (Omit if IDDX ≠ 1)

Boundaries of angular interval. NSCT data is necessary in descending order. The last data should be equal to -1.

CARD XD (14I5) (Omit if IXTAPE ≤ 0 and IDDX = 0)

Element identifiers for cross section tape. If negative and IDDX = 1, the cross sections for last group (NMGP group) are read from the following cards. IF element identifiers are in the same order as elements on Tape, the efficiency of the code is increased due to fewer tape rewinds.

CARDS XE (I5, 5E10.3) (Omit if element identifiers on Card XD > 0)

NELEM (see Card XB) cards are required.

MAT	- element identifier
SIGTO	- total cross section of NMGP group
SIGPR	- production cross section of NMGP group
SIGC	- absorption cross section of NMGP group
SIGF	- fission cross section of NMGP group
SIGFE	- $\nu\sigma_f$ of NMGP group

CARDS XF (2I5,E10.5) (Omit if IXTAPE < 0 and IDDX = 0)

NMIX (see Card XB) cards are required.

KM	- medium number.
KE	- element number occurring in medium KM (negative value indicates last mixing operation for that medium). Failure to have a negative value causes code not to generate angular probabilities for that media (LEGEND and ANGLE no caled).
RHØ	- density of element KE in medium KM.

CARD XG (Omit if IXTAPE ≠ 0)

If cross sections are in free-form, a card with in columns 2 and 3 must precede the actual data.

ANISN format if IDTF ≤ 0; otherwise, DTF-IV format. Cross sections for INGP groups with a table length ITBL for NELEM elements each with NCØEF coefficients.

CARD XH (I5) (Omit if IO6RT ≤ 0)

NXPM	- number of point cross-section sets per medium found on an 06R tape.
	= 1, total cross section only,
	= 2, total + scattering cross section,
	= 3, total, scattering, and $\nu*fission$ cross section.

NOTE: Cross sections and cross-section input data may be checked independently of MORSE utilizing XCHEKR. The input to XCHEKR consists of the cross-section cards XA through XG preceded by a card as follows:

Format (4I5)

- IADJM - set greater than zero for an adjoint problem.
- MEDIA - number of cross-section media; should equal NMED on Card XB.
- NMGP - number of primary particle energy groups for which cross sections are to be stored; should equal NGP on Card XB.
- NMTG - total number of energy groups for which cross sections are to be stored. Should be equal to INGP on Card XB.

3.4 Analysis Input Instructions

The following data are read from cards by SCORIN:

CARD AA (20A4)

Title information - will be immediately output.

CARD BB (13I5,I2,I3)

- ND - number of detectors (set = 1 if ≤ 0).
- NNE - number of primary particle (neutron) energy bins to be used (must be \leq NE).
- NE - total number of energy bins (set = 0 if ≤ 1).
- NT - number of time bins for each detector (may be negative, in which case $|NT|$ values are to be read and used for every detector) (set = 0 if $|NT| \leq 1$).
- NA - number of angle bins (set = 0 if ≤ 1).
- NRESP - number of energy-dependent response functions to be used (set = 1 if ≤ 0).
- NEX - number of extra arrays of size NMTG to be set aside (useful, for example, as a place to store an array of group-to-group transfer probabilities for estimator routines).
- NEXND - number of extra arrays of size ND to be set aside (useful, for example, as a place to store detector-dependent counters).
- NTAPE - indicator of restart option if NTAPE > 0 .
The result of previous run is read from unit NTAPE,
and the data for next run is written on unit NTAPE + 1.
- NNBAT - dummy input. Set 0.
- NRUN2 - 0/1 = first run/restart run.
- IOUTE - unit of output flux.
0/1 = per unit lethargy/per unit energy.
- IEST - indicator of flux estimator
- 0 - point estimator
- 1 - surface cross estimator (sphere of which center is placed at the origin)
- 2 - surface cross estimator (circular plane perpendicular to Z-axis)
- 3 - track length estimator
- 4 - collision estimator
- 1 - arbitrary combination of above estimators

For the estimators except for point one, the detector regions or detector surfaces should be defined by the geometry input.

INCF - flag to calculate uncollided flux if INCF = 1.
 This flag is effective only for the point estimator.

IOUTF - Brief output indicator if IOUTF ≠ 0.
 Two-digit integer (IOUTF1 × 10 + IOUTF2).

IOUTF1 : indicator for the results of edit.

- 3 - Only integrated response.
- 2 - response + energy dependent total flux
- 1 - response + energy dependent total flux + angular flux
- 0 - all

IOUTF2 : indicator for random walk information.

- 4 - No output
- 3 - (regionwise) real collision, albedo collision, fission
- 2 - 3 + Splitting
- 1 - 2 + Russian roulette
- 0 - all

CARD BB1 (5I5) (Omit if IEST ≠ -1)

IE1 - number of point estimator
 IE2 - number of surface cross estimator (sphere)
 IE3 - number of surface cross estimator (circular plane)
^{*}IE4 - number of track length estimator
^{*}IE5 - number of collision estimator.

* Should be equal to the region numbers defined by the geometry input.

CARD CC (5E10.4) ND cards are required if IEST = 0,1,2, and (IE1 + IE2 + IE3) cards are if IEST = -1.

X,Y,Z, RMAX, RMIN - location of point or surface cross estimators.

For the point estimator,

(X,Y,Z) : detector location
^{*}RMAX : radius (cm) of sphere. The bounded sphere approximation is applied within the sphere.
^{*}RMIN : scoring is performed with the probability $(RMIN/R)^2$ outside the sphere with radius RMIN.

For the surface cross estimator (sphere),

the detector is the surface of sphere with radius R^{**} and with the center at the origin.

$$R^2 = X^2 + Y^2 + Z^2,$$

For the surface cross estimator (circular plane), the detector surface is the circular plane perpendicular to Z axis of radius X with the center at $(0,0,Z)^{**}$. RMAX and RMIN are dummy for the surface cross estimator.

* The same values of RMAX, RMIN, which are defined on the last card, are commonly used for all the point estimators.

** Should input in increasing order of Z value.

Note that the distance between the above points and the XSTART, YSTART, ZSTART values and the initial age, AGSTART, will be used to define the lower limit of the first time bin.

CARD DD (20A4)

Title or units for total responses for all detectors. Will be used in columns 54 through 133 of the title for the print of these arrays.

CARD EE (20A4)

Title or units for each total response for all detectors.

CARD FF (7E10.4)

Response function values. NMTG values will be read in each set of FF cards. Input order is from energy group 1 to NMTG (order of decreasing energy).

NOTE: Cards EE and FF are read in the following order:

EE, FF1, . . . FFN, EE, FF1, . . . FFN, etc. NRESP sets of EE, FF cards will be read.

CARD GG (20A4) (Omit if NE ≤ 1)

Units for energy-dependent fluence for all detectors.

CARDS HH (14I5) (Omit if NE ≤ 1)

Energy group numbers defining lower limit of energy bins (in order of increasing group number). The NNE (if > 0) energy must equal NGPQTN; the NE entry must be set to NMGP + NGPQTG for a combined problem, or else NGPQTG or NGPQTN.

CARD II (20A4) (Omit if |NT| ≤ 1)

Units for time-dependent total responses for all detectors.

CARD JJ (20A4) (Omit if |NT| ≤ 1 or NE ≤ 1)

Units for time and energy-dependent fluence for all detectors.

CARDS KK (7E10.4) (Omit if $|NT| \leq 1$)

NT values of upper limits of time bins for each detector (in order of increasing time and detector number). The values for each detector must start on a new card. $|NT|$ values only are read if NT is negative. They are then used for every detector.

CARD LL (20A4) (Omit if NA ≤ 1)

Units for angle- and energy-dependent fluence for all detectors.

CARD MM (7E10.4) (Omit if NA ≤ 1)

NA values of upper limits of angle bins (actually cosine bins; the NA_{th} value must equal one).

Following the input for the SAMBØ analysis module, input cards for user-written routines INSCØR, SØURCE, and ENDRUN.

3.5 External Source Module Input Instruction.

The input for external source module is read by subroutine S\$INP called from subroutine S\$OURCE.

CARD S1 (I5)

IS : Number of source IS ≠ 0

CARD S2 (7F10.4)

S(I), I = 1, IS : relative intensity of i'th source

CARD S3 (10X,3I5,2E10.3,I5)

NSOURCE - Geometry and distribution of source.

1 point source

2 rectangular parallelepiped source, uniform distribution

3 rectangular parallelepiped source, cosine distribution

4 cylindrical source, uniform distribution

5 cylindrical source, cosine distribution

6 sphere or spherical shell source, uniform distribution

7 circular plane source, uniform distribution

8 cylindrical shell source, uniform distribution

NEMIS - Angular distribution of source particles

1 monodirectional

2 isotropic

3 cosine distribution against Z-direction

> 0 positive direction only

< 0 negative direction only

4 arbitrary emission

5 cosine distribution against radial direction

positive: outward, negative: inward

(effective for NSOURCE = 6,7)

NSPEC - source energy spectrum

0 arbitrary spectrum read by S\$URIN. (IS\$UR < 0)

1 Watt's fission spectrum

-2	arbitrary pointwise spectrum.	S(u) log-log interpolation
+2		S(E) linear-linear interpolation

3 mono-energy

ETOP	dummy input
EBOT	

JDIREC transformation of directional cosine.

1 normal

2 Y ← X, Z ← Y, X ← Z

3 $Y \leftarrow Z, Z \leftarrow X, X \leftarrow Y$
 4 arbitrary rotation of direction
 5 arbitrary rotation of direction and position

CARD S4 (10X,3F10.4)

XO,YO,ZO Origin of source

CARD S5 (10X,6F10.4)

XL,XR,YL,YR,ZL,ZR geometrical input of source: See Table 3.3

Table 3.3 Geometry Input for Source

NSOURCE	XO	YO	ZO	XL	XR	YL	YR	ZL	ZR
1	XO	YO	ZO	—	—	—	—	—	—
2	—	—	—	XL	XR	YL	YR	ZL	ZR
3	—	—	—	XL	XR	YL	YR	ZL	ZR
4	XO	YO	—	—	XR	—	—	ZL	ZR
5	XO	YO	—	—	XR	—	—	ZL	ZR
6	XO	YO	ZO	XR	XR	—	—	—	—
7	XO	YO	ZO	XL	—	—	—	—	—
8	XO	YO	—	XL	XR	—	—	ZL	ZR
9	XO	YO	ZO	—	XR	—	—	—	—

— means dummy input.

CARD S6 (10X,3F10.3) (enter if NEMIS = 1)

UINP,VINP,WINP: direction cosine of mono-direction

$$UINP^2 + VINP^2 + WINP^2 = 1$$

CARD S7 (15) (enter if $|NSPEC| \geq 2$)IMAX1 No. of energy points for which the energy spectrum is given. IMAX1 should be > 2 .CARD S8 (2E10.5) (IMAX1 cards are necessary if $|NSPEC| \geq 2$)

(E(I), FE(I)) source spectrum.

Unit fo FE(I) is per unit lethargy or per unit energy when NSPEC on CARD S3 is -2 or 2, respectively.

CARD S9 (3I5) (Omit if NEMIS $\neq 4$)

IMAX number of energy boundaries.

JMAX number of polar angle mesh.

KMAX number of azimuthal angle mesh.

CARD S10 (7E10.4) (Omit if NEMIS ≠ 4)

ENG(I), I = 1, IMAX upper energy boundaries for the I'th angular distribution is used.

CARD S11 (7E10.4) (Omit if NEMIS ≠ 4)

DEG(I), I = 1, JMAX Cosine bins of polar angles, for which angular distribution is defined.

CARD S12* (7E10.4) (Omit if NEMIS ≠ 4)

(ANGDIS(I,J), J = 1, JMAX-1) Probability with which the source particle within I'th energy interval and J'th polar angle bin is generated. The probability is normalized such as

$$\sum_{J=1}^{JMAX-1} ANGDIS(I,J) = 1.0$$

(IMAX-1) sets of CARD S12 are required.

CARD S13 (7E10.4) (Omit if NEMIS ≠ 4 or KAMX = 0)

FAI(I), I = 1, KMAX azimuthal angle bins for which azimuthal probability is given.

CARD S14* (7E10.4) (Omit if NEMIS ≠ 4 or KMAX = 0)

FAIDIS(I), I = KMAX-1 azimuthal probability.

The probability is normalized such as

$$\sum_{J=1}^{KMAX-1} FAIDIS(I) = 1.0$$

CARD S15 (3E10.4) (Omit if JDIREC ≤ 3)

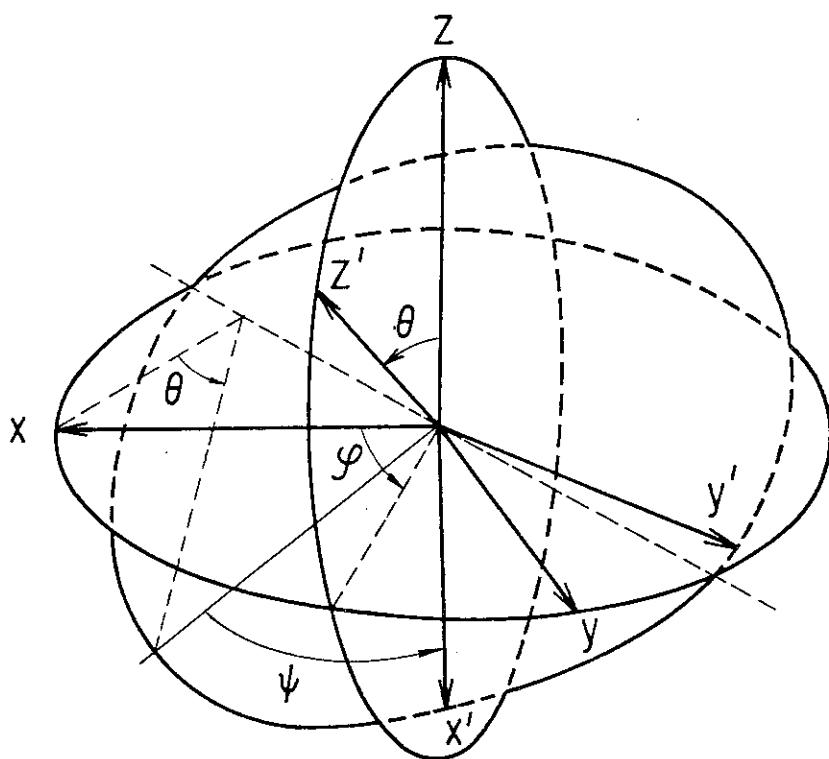
TH1, FA1, PSA11 data for the rotation of source.

see Fig. 3.1.

CARDS S3 ~ S15 are repeatedly input for IS sources (see CARD S1).

* The angular distribution function P of I'th energy interval is defined by

$$P_I(\mu, \varphi) = \begin{cases} \frac{ANGDIS(I,J)*FAIDIS(K)}{(FAI(K) - FAI(K+1))*(DEG(I) - DEG(I+1))} \\ \text{for } DEG(I+1) < \mu \leq DEG(I), FAI(K+1) < \varphi \leq FAI(K), \\ 0 \quad \text{for otherwise} \end{cases}$$



z - direction \rightarrow z' - direction
 x - direction \rightarrow x' - direction
 y - direction \rightarrow y' - direction

Fig. 3.1 Rotation of coordinates
 The angles (θ, ϕ, ψ) correspond to (TH1, FAL, PSAII) in
 card S15

4. Job Control Instructions

4.1 Job Control Statements

A sample of job control statements for FACOM-M-series computer is shown in Fig. 4.1.

Executive load module of MORSE-DD has been prepared in J3803.

XXMORSE.LOAD(D). The size of blank common in this module is 70k words, and a user should replace Main routine by user's one having an appropriate size of blank common.

4.2 Binary File Format

I/O files required by MORSE-DD are summarized in Table 4.1. Format of double differential type cross section library and details of output and their formats for each output file in DDX mode are as follows.

i) Double differential type cross section library (unit 99)

```

  1, Element
    MAXI, IDUMMY, IDUMMY, MATNO, (TITLE(I), I = 1,12)
    (MAXSD(I), I = 1, MAXI)
    ( $\sigma_{pr}^g$ ,  $\sigma_f^g$ ,  $\sigma_a^g$ ,  $v\sigma_f^g$ ,  $\sigma_t^g$ , g = 1, MAXI)
  1, MAXI
    ((I ( $\mu_j$ , g → g'), j = 1, MAXMU), g' = 1, MAXSD(g))

```

ii) Processed cross sections (unit 96)

[1, Media
 [1, Group
 [$\Sigma_{pr}^g, \Sigma_f^g, \Sigma_c^g, v\Sigma_t^g, \Sigma_t^g, (\Sigma_s(g \rightarrow g'), g' = 1, NGP),$
 [$((P_{\mu_k}(g \rightarrow g'))^*, k = 1, NSCT), g' = 1, NGP).$
 * $P_{\mu_k}(g \rightarrow g') = - \int_1^{\mu_k} \bar{I}(\mu, g \rightarrow g').$

iii) Group transfer matrixes (unit IXTAPE)

[1, Element
 [MATNO, MATNO, K, MATNO, (TITLE(I), I = 1, 12).
 [$((\sigma(I,J), I = 1, IHT), J = 1, NGP).$

$\sigma(I,J)$ is microscopic cross section with the same format as P_0

component of those in P_1 mode. In DDX mode, the data from

IXTAPE are used only to obtain the maximum slowing down group.

iv) Tallies (unit 95)

$(E(g), g = 1, NGP + 1)$: energy boundaries.

[1, Detector,
 [$((F(g), \Delta F(g)), g = 1, NGP)$. : total flux and its fractional standard deviation (1σ).

$(A(m), m = 1, NA + 1)$: boundaries of angle bins.

[1, Detector,
 [$((FA(m,g), \Delta FA(m,g)), m=1, NA, g=1, NGP)$: angular flux and its fractional standard deviation.

[1, Detector,
 [$((UF(g), \Delta UF(g)), g = 1, NGP)$: non-collided total flux and its fractional standard deviation.

[1, Detector,
 [((UAF(m,g), ΔUAF(m,g)), m=1,NA), g=1,NGP) : non-collided angular
 flux and its fractional standard deviation.

(V) Restart dump (unit NTAPE + 1)

RANDOM : next random number.
 NBATS, NSTRT : number of batches previously run and
 number of particles per batch.
 LL,LLL,(A(I),I=LL,LLL) : data in blank common, LL = LOCUD + 1,
 LLL = LMAX. They are arrays of the sums
 over all batches and the sums of
 squared batch estimates.

```
//JCLG JOB
// EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER XXXXXXXX, [REDACTED],0431.100
T.4 C.7 W.2 P.0 I.5
OPTP MSGCLASS=R,[REDACTED]
// EXEC FORT77,SO='J3803.MORSEUP',Q='FORT',
// A='ELM(MAIN90)'
// EXEC LKEDIT77,LM='J3803.XXMORSE',CNTL=NO
INCLUDE OLDLM(D)
ENTRY MAIN
NAME TEMPNAME
// EXEC GO
/*
//**-- UNIT 99 :      DDX LIBRARY
//FT99F001 DD DSN=J3803.DDXLIB1.DATA,DISP=SHR,LABEL=(,,,IN)
/*
//**-- UNIT 98 :      SCRATCH FILE FOR PROCESSING DDX LIBRARY
// EXPAND DISK,DDN=FT98F001,SPC='100,40'
/*
//**-- UNIT 96 :      OUTPUT OF PROCESSED DDXLIBRARY (MACRO)
// EXPAND DISK,DDN=FT96F001,SPC='50,40'
/*
//**-- UNIT 95 :      OUTPUT OF RESULT FOR PLOTTING ETC.
//FT95F001 DD DSN=J3803.PLOTDATA.DATA,DISP=(NEW,CATLG,DELETE),
// SPACE=(TRK,(1,1),RLSE),DCB=(RECFM=VBS,LRECL=6208,BLKSIZE=6212),
// UNIT=TSSWK
/*
//**-- UNIT IXTAPE :  OUTPUT OF PROCESSED ANISN-TYPE CROSS SECTIONS FROM
//                  DDX LIBRARY
// EXPAND DISK,DDN=FT20F001,SPC='30,30'
/*
//**-- UNIT NTAPE :  INPUT OF RESTART INFORMATION FROM PREVIOUS RUN
// EXPAND DISKTO,DDN=FT08F001,DSN=J3803.RES1
/*
//**-- UNIT NTAPE+1:  OUTPUT OF RESTART INFORMATION FOR NEXT RUN
//FT09F001 DD DSN=J3803.RESTART.DATA,DISP=(NEW,CATLG),
// SPACE=(TRK,(1,1),RLSE),DCB=(RECFM=VBS,LRECL=6208,BLKSIZE=6212),
// UNIT=TSSWK
/*
//**-- UNIT 5 :      CARD INPUT FOR MORSE-DDX
//SYSIN DD DSN=J3803.MORSECG.DATA(DDXSLAB),DISP=SHR
++
//
```

Fig.4.1 Job control statements

Table 4.1 MORSE-DD file requirements

Logical unit	Contents	Remarks	I/O*
99	Binary DDX library	This file is used as input unit from DDX library.	I
98	Scratch file in DDX mode	Scratch file used as a temporary storage in DDX mode	O,I
96	Processed cross section in DDX mode	Scratch file used in DDX mode. Processed data of cross section ($\Sigma_x, \Sigma_s \cdot P(g \rightarrow g)$, $P(\mu_j, g \rightarrow g)$) are written on this file.	O,I O,I
95	Binary output of tally data	This file may be used as input file for auxiliary programs	O
IXTAPE	Binary P_ℓ library. Scratch file in DDX mode.	This file is used as input unit from conventional P_ℓ library. In DDX mode, group transfer matrixes with the same format as P_ℓ library are written on this file.	I (in P_ℓ) O,I (in DDX)
JXTAPE	Processed cross section in P_ℓ mode	Processed cross sections are written on this file	O
NTAPE	Restart information	In a restart run, a restart dump information of a previous run is read from this file	I
NTAPE + 1	Restart dump	If a restart dump is required, a restart dump information is overwritten on this file every 5 batches.	O
6	Output of final result	Final result of MORSE calculation	O
5	Card-image input for MORSE calculation	Card-image input for MORSE-DDX	I

* I and O mean read-only and write-only files, respectively.

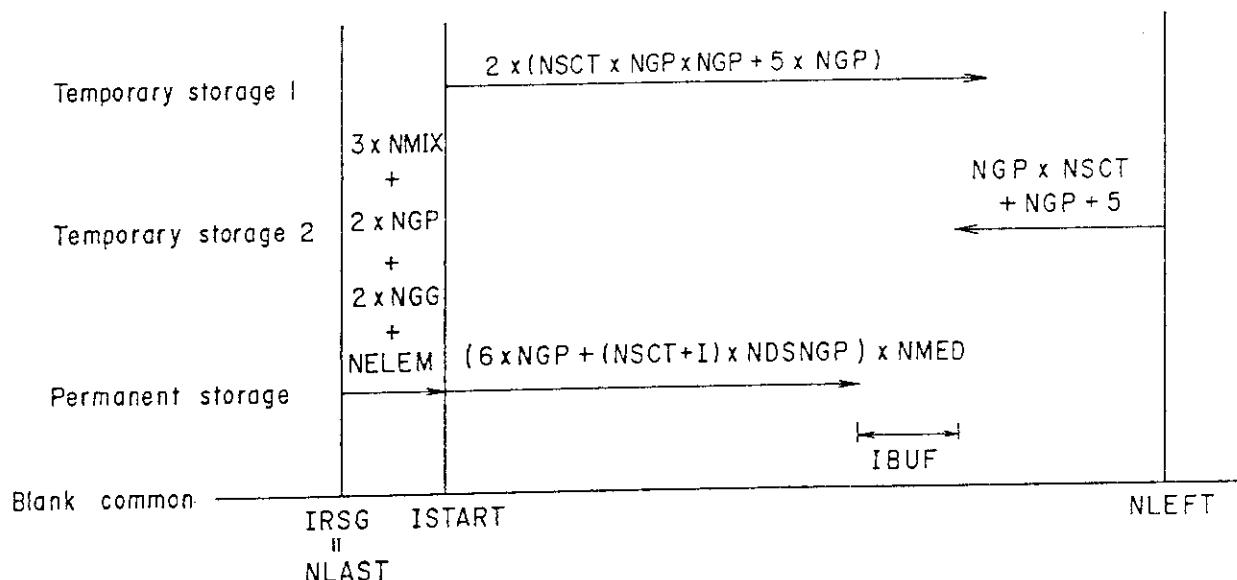
4.3 Core Requirement

The core storage requirement in bytes by MORSE-DD is the order of $570,000 + 4 \times (\text{blank common size in words})$. The size of the blank common used in cross section processing is different between the P_l and the DDX modes.

The location in the blank common used for cross section processing in DDX mode is shown in Fig. 4.2. The blank common size required in this case is given by

$$\text{Memory} = 3 \times \text{NMIX} + 2 \times (\text{NGP} + \text{NGG}) + \text{NELEM}$$

$$+ \text{Maximum} \left\{ \begin{array}{l} 2 \times (\text{NSCT} \times \text{NGP} \times \text{NGP} + 5 \times \text{NGP}) \\ (6 \times \text{NGP} + (\text{NSCT} + 1) \times \text{NDSNGP}) \times \text{NMED} \\ + \text{NGP} \times \text{NSCT} + \text{NGP} + 5 \end{array} \right.$$



NLEFT : Blank common size

NDSNGP : Maximum size of group-to-group transfer matrixes

Fig.4.2 Location of blank common used for cross section processing

5. SAMPLE INPUT AND OUTPUT

The sample MORSE-DD input data is shown in Fig. 5.1. This problem calculates neutron flux distributions in a cylindrical Li_2O pile with radius of 35 cm and height of 80 cm. The whole cylinder is divided into 17 regions in which the neutron flux is calculated by using the track length estimator. The source is positioned at $x = 0$, $y = 0$, $z = 0$, of which angular distribution is monodirectional, that is, parallel to z axis. The composition of material is shown in Table 5.1. The cross sections of thermal group (125 th group) are input by cards.

Table 5.1 Composition of Material

Nuclide	MAT number	Atomic number density (10^{24} atom/cm 3)
${}^6\text{Li}$	1271	4.2729 - 3
${}^7\text{Li}$	1272	5.3422 - 2
O	1276	2.8848 - 2
Fe	1192	1.0785 - 3
Cr	1191	2.9922 - 4
Ni	1190	1.3086 - 4
Mn	1197	2.3921 - 5

The brief output is shown in Fig.5.2.

L120 CYLINDER OF 80 CM LENGTH, DDX FROM ENDF/B4

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	T	U	V	W	X	Y	Z
10000010000	40	1	125	0	125	0	0	1.5	1	0	0.	2200.														
-1	46	0	1.0	1.	E-5	0.	0.	0.	0.	0.	0.															
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.															
3.8510E-2	5.294E-1	2.7990E-1	2.9150E-2	5.4310E-3	2.1840E-3	1.3590E-3																				
1.2330E-3	1.1070E-3	1.1360E-3	1.0480E-3	9.8790E-4	1.0280E-3	1.1470E-3																				
1.2460E-3	1.3170E-3	1.5090E-3	1.6360E-3	1.7890E-3	1.9300E-3	2.0850E-3																				
2.1680E-3	2.1960E-3	2.3010E-3	2.5050E-3	4.6300E-3	7.6100E-3	4.9590E-3																				
3.4100E-3	2.8090E-3	2.8310E-3	2.8950E-3	2.9860E-3	2.9870E-3	3.0690E-3																				
3.0560E-3	5.9950E-3	5.8960E-3	5.8260E-3	5.6770E-3	5.1130E-3	4.5220E-3																				
3.9070E-3	3.3710E-3	2.5390E-3	1.6560E-3	1.5390E-3	1.5730E+07	1.5490E+07	1.5250E+07	1.5010E+07	1.4770E+07	1.4430E+07	1.4100E+07	1.3880E+07	1.3670E+07	1.3460E+07	F-01											
1.4780E+07	1.4550E+07	1.4320E+07	1.4080E+07	1.3840E+07	1.3610E+07	1.3380E+07	1.3150E+07	1.2920E+07	1.2690E+07	1.2450E+07	1.2220E+07	1.2000E+07	1.1780E+07	1.1550E+07	F-02											
1.3250E+07	1.3040E+07	1.2840E+07	1.2640E+07	1.2440E+07	1.2240E+07	1.2040E+07	1.1840E+07	1.1640E+07	1.1440E+07	1.1240E+07	1.1040E+07	1.0840E+07	1.0640E+07	1.0440E+07	F-03											
1.1880E+07	1.1690E+07	1.1510E+07	1.1330E+07	1.1150E+07	1.0960E+07	1.0760E+07	1.0560E+07	1.0360E+07	1.0160E+07	9.9000E+07	9.3940E+06	8.8250E+06	8.3530E+06	7.8830E+06	F-04											
1.0650E+07	1.0480E+07	1.0320E+07	1.0160E+07	1.0000E+07	9.9000E+07	9.3940E+06	8.8250E+06	8.3530E+06	8.0000E+06	6.4560E+06	6.0650E+06	5.6980E+06	5.3160E+06	4.9310E+06	F-05											
8.2900E+06	7.7880E+06	7.3160E+06	6.8730E+06	6.4370E+06	4.4370E+06	4.1690E+06	3.9160E+06	3.6460E+06	3.3750E+06	3.0930E+06	2.8290E+06	2.5280E+06	2.3750E+06	2.1870E+06	F-06											
5.3530E+06	5.0280E+06	4.7240E+06	4.4370E+06	4.1690E+06	3.0500E+06	2.8650E+06	2.6910E+06	2.5280E+06	2.3750E+06	2.1870E+06	1.7300E+06	1.5340E+06	1.3330E+06	1.1330E+06	F-07											
3.4560E+06	3.2470E+06	3.0500E+06	2.8650E+06	2.6910E+06	2.4940E+06	2.4790E+04	2.4790E+04	2.4790E+04	2.4790E+04	2.1870E+04	1.9300E+04	1.5030E+04	1.3030E+04	1.1030E+04	F-08											
2.2310E+06	2.0960E+06	1.9690E+06	1.8500E+06	1.7300E+06	1.6100E+06	1.5000E+06	1.4000E+06	1.3000E+06	1.2000E+06	1.1000E+06	1.0000E+06	9.0000E+05	8.0000E+05	7.0000E+05	6.3930E+05	F-09										
1.1940E+06	1.0540E+06	9.3010E+05	8.2080E+05	7.2080E+05	6.2080E+05	5.2080E+05	4.2080E+05	3.2080E+05	2.2080E+05	1.2080E+05	1.1080E+05	1.0080E+05	9.0080E+05	8.0080E+05	F-10											
4.9790E+05	4.3940E+05	3.8770E+05	3.4220E+05	3.0200E+05	2.6450E+05	2.2590E+05	1.8590E+05	1.4590E+05	1.0590E+05	1.0110E+05	9.8040E+04	9.0040E+04	8.0040E+04	7.0040E+04	F-11											
2.0750E+05	1.8320E+05	1.6160E+05	1.4260E+05	1.2590E+05	1.0590E+05	8.6310E+04	7.6310E+04	6.6310E+04	5.6310E+04	4.6310E+04	4.0870E+04	3.40870E+04	2.80870E+04	2.20870E+04	F-12											
8.6520E+04	7.6350E+04	6.7350E+04	6.7350E+04	6.7350E+04	F-13																					
3.6070E+04	3.1830E+04	2.8090E+04	2.8090E+04	2.8090E+04	F-14																					
1.1710E+04	9.1190E+03	7.1020E+03	5.5310E+03	4.3070E+03	3.3070E+03	2.3070E+03	1.3070E+03	3.3550E+03	3.3550E+03	3.6130E+03	3.9280E+03	4.2440E+03	4.6420E+03	5.0420E+03	F-15											
2.0350E+03	1.5850E+03	1.2340E+03	9.6110E+02	5.8290E+02	5.8290E+02	5.8290E+02	F-16																			
1.3010E+02	7.8890E+01	4.7780E+01	2.9020E+01	1.7600E+01	1.7600E+01	1.0668E+01	6.4766E+00	6.4766E+00	6.4766E+00	6.4766E+00	6.4766E+00	6.4766E+00	6.4766E+00	6.4766E+00	F-17											
3.9280E+00	2.3820E+00	1.4450E+00	8.7640E-01	5.3160E-01	3.2240E-01	3.2240E-01	3.2240E-01	H-01																		
0.	1.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	I-01											
125	1	125	1	1	18	0.	0.	0.25	0.	1.0	0.0	0.	0.	0.	J-01											
-1	0	0	0	0	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	J-02											
3	0	0	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	L-01											
RCC	1	0.0	0.0	0.0	0.0	10.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	CGA-01											
RCC	2	0.0	0.0	0.0	0.0	10.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	CGB-02											
RCC	3	0.0	0.0	0.0	0.0	20.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	CGB-03											
RCC	4	0.0	0.0	0.0	0.0	30.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	CGB-04											
RCC	5	0.0	0.0	0.0	0.0	40.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	CGB-05											
RCC	6	0.0	0.0	0.0	0.0	50.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	CGB-06											
RCC	7	0.0	0.0	0.0	0.0	60.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	CGB-07											
RCC	8	0.0	0.0	0.0	0.0	70.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	CGB-08											
RCC	9	0.0	0.0	0.0	0.0	80.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	CGB-09											
RCC	10	0.0	0.0	0.0	0.0	90.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	CGB-10											
RCC	11	0.0	0.0	0.0	0.0	100.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	CGB-11											
															CGB-12											
															CGB-13											
															CGB-14											
															CGB-15											
															CGB-16											
															CGB-17											
															CGB-18											
															CGB-19											
															CGB-20											
															CGB-21											
															CGB-22											
															CGB-23											
															CGB-01											
															CGB-02											
															CGB-03											
															CGB-04											

Fig.5.1-1 Sample input data for MORSE-DD

Fig. 5.1-2

Fig. 5.1-3

L120 CYLINDER OF 80 CM LENGTH, DDX FROM ENDF/B4
THIS CASE WAS BEGUN ON TUESDAY, JUNE 12, 1984

NSTRT	NMOSIT	NITS	NQUIT	NPQTIN	NPQTG	NMGPF	NCOLTP	IADJM	MAXIM	MEDIA	MEDALB
	10000	4.0	1	125	0	125	0	0	1.50	1	0
ISOUR	NGPFS	ISBIAS		WTSTART	1.0000E+00	EBOTN	0.0	EBOTG	0.0	TCUT	VELTH
	-1	4.6	0								
				XSTRAT	YSTRAT	ZSTRAT	AGSTRAT	UINP	VINP	WINP	WINP
				0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
				SOURCE DATA	UNNORMALIZED	NORMALIZED	FRACTION				
GROUP				3.8510E-02	5.2940E-01	2.7990E-01	2.9150E-02	5.4310E-03	2.1840E-03	1.3590E-03	1.2350E-03
1				0.038508	0.529371	0.279885	0.029148	0.005431	0.002184	0.001359	0.001235
2											
3											
4											
5											
6											
7											
8											
9											
10											
11											
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37											
38											
39											
40											
41											
42											
43											
44											
45											
46											
TOTAL				1.0001E+00							

Fig.5.2 Sample output of MORSE-DD

```

121      2.3820E+00      1.9131E+06
122      1.4450E+00      1.4900E+06
123      8.7640E-01      1.1604E+06
124      5.3160E-01      9.0372E+05
125      3.2240E-01      2.2000E+03

INITIAL RANDOM NUMBER = 001234567123

NSPLT= 0      NKILL= 1      NPAST= 0      NOLEAK= 0      IEBIAS= 0      MXREG= 18      MAXGP= 125
WEIGHT STANDARDS FOR SPLITTING AND RUSSIAN ROULETTE AND PATHLENGTH STRETCHING PARAMETERS
NGP1      NDG      NGR2      NRG1      NRG      NRG2      WTHI1      WTLOW1      WTAVE1      XNU
125      1       125      1       1       1       18      0.0      2.5000E-01      1.0000E+00      0.0
NSOUR= 0      MFISTP= 0      NKCALC= 0      NORMF= 0

```

Fig.5.2 (Continued)

L120 SLAB

IVOPT = 3

IDBG = 0

			BODY DATA	
RCC	1	0.0	0.0	0.1000000D+02 0.0
RCC	2	0.0	0.0	0.1000000D+02 0.0
RCC	3	0.0	0.0	0.2000000D+02 0.0
RCC	4	0.0	0.0	0.3000000D+02 0.0
RCC	5	0.0	0.0	0.4000000D+02 0.0
RCC	6	0.0	0.0	0.5000000D+02 0.0
RCC	7	0.0	0.0	0.6000000D+02 0.0
RCC	8	0.0	0.0	0.7000000D+02 0.0
RCC	9	0.0	0.0	0.8000000D+02 0.0
RCC	10	0.0	0.0	-0.1000000D+02 0.0
RCC	11	0.0	0.0	-0.2000000D+02 0.0
END	12	0.0	0.0	0.0 0.0
NUMBER OF BODIES	11			
LENGTH OF FPD-ARRAY	107			

			INPUT ZONE DATA	
1	0	1	0	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
2	0	1	2	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
3	0	1	3	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
4	0	1	4	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
5	0	1	5	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
6	0	1	6	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
7	0	1	7	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
8	0	1	8	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
9	0	-1	9	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
10	0	-1	2	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
11	0	-1	3	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
12	0	-1	4	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
13	0	-1	5	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
14	0	-1	6	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
15	0	-1	7	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
16	0	-1	8	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
17	0	-1	9	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
18	0	-2	-3	-4 -5 -6 -7 -8 -9 -10 -11 -12 -13 -14 -15 -16 -17 -18 -19
END	0	0	0	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
NUMBER OF INPUT ZONES	18			
NUMBER OF CODE ZONES	18			
LENGTH OF INTEGER ARRAY	448			

CODE ZONE INPUT ZONE ZONE DATA LOC. NO. OF BODIES REGION NO. MEDIA NO. Fig. 5.2 (Continued)

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74	75	76	77	78	79	80	81	82	83	84	85	86	87	88	89	90	91	92	93	94	95	96	97	98	99	100	101	102	103	104	105	106	107	108	109	110	111	112	113	114	115	116	117	118	119	120	121	122	123	124	125	126	127	128	129	130	131	132	133	134	135	136	137	138	139	140	141	142	143	144	145	146	147	148	149	150	151	152	153	154	155	156	157	158	159	160	161	162	163	164	165	166	167	168	169	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	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	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	268	269	270	271	272	273	274	275	276	277	278	279	280	281	282	283	284	285	286	287	288	289	290	291	292	293	294	295	296	297	298	299	300	301	302	303	304	305	306	307	308	309	310	311	312	313	314	315	316	317	318	319	320	321	322	323	324	325	326	327	328	329	330	331	332	333	334	335	336	337	338	339	340	341	342	343	344	345	346	347	348	349	350	351	352	353	354	355	356	357	358	359	360	361	362	363	364	365	366	367	368	369	370	371	372	373	374	375	376	377	378	379	380	381	382	383	384	385	386	387	388	389	390	391	392	393	394	395	396	397	398	399	400	401	402	403	404	405	406	407	408	409	410	411	412	413	414	415	416	417	418	419	420	421	422	423	424	425	426	427	428	429	430	431	432	433	434	435	436	437	438	439	440	441	442	443	444	445	446	447	448	449	450	451	452	453	454	455	456	457	458	459	460	461	462	463	464	465	466	467	468	469	470	471	472	473	474	475	476	477	478	479	480	481	482	483	484	485	486	487	488	489	490	491	492	493	494	495	496	497	498	499	500	501	502	503	504	505	506	507	508	509	510	511	512	513	514	515	516	517	518	519	520	521	522	523	524	525	526	527	528	529	530	531	532	533	534	535	536	537	538	539	540	541	542	543	544	545	546	547	548	549	550	551	552	553	554	555	556	557	558	559	560	561	562	563	564	565	566	567	568	569	570	571	572	573	574	575	576	577	578	579	580	581	582	583	584	585	586	587	588	589	590	591	592	593	594	595	596	597	598	599	600	601	602	603	604	605	606	607	608	609	610	611	612	613	614	615	616	617	618	619	620	621	622	623	624	625	626	627	628	629	630	631	632	633	634	635	636	637	638	639	640	641	642	643	644	645	646	647	648	649	650	651	652	653	654	655	656	657	658	659	660	661	662	663	664	665	666	667	668	669	670	671	672	673	674	675	676	677	678	679	680	681	682	683	684	685	686	687	688	689	690	691	692	693	694	695	696	697	698	699	700	701	702	703	704	705	706	707	708	709	710	711	712	713	714	715	716	717	718	719	720	721	722	723	724	725	726	727	728	729	730	731	732	733	734	735	736	737	738	739	740	741	742	743	744	745	746	747	748	749	750	751	752	753	754	755	756	757	758	759	760	761	762	763	764	765	766	767	768	769	770	771	772	773	774	775	776	777	778	779	780	781	782	783	784	785	786	787	788	789	790	791	792	793	794	795	796	797	798	799	800	801	802	803	804	805	806	807	808	809	8010	8011	8012	8013	8014	8015	8016	8017	8018	8019	8020	8021	8022	8023	8024	8025	8026	8027	8028	8029	8030	8031	8032	8033	8034	8035	8036	8037	8038	8039	8040	8041	8042	8043	8044	8045	8046	8047	8048	8049	8050	8051	8052	8053	8054	8055	8056	8057	8058	8059	8060	8061	8062	8063	8064	8065	8066	8067	8068	8069	8070	8071	8072	8073	8074	8075	8076	8077	8078	8079	8080	8081	8082	8083	8084	8085	8086	8087	8088	8089	8090	8091	8092	8093	8094	8095	8096	8097	8098	8099	80100	80101	80102	80103	80104	80105	80106	80107	80108	80109	80110	80111	80112	80113	80114	80115	80116	80117	80118	80119	80120	80121	80122	80123	80124	80125	80126	80127	80128	80129	80130	80131	80132	80133	80134	80135	80136	80137	80138	80139	80140	80141	80142	80143	80144	80145	80146	80147	80148	80149	80150	80151	80152	80153	80154	80155	80156	80157	80158	80159	80160	80161	80162	80163	80164	80165	80166	80167	80168	80169	80170	80171	80172	80173	80174	80175	80176	80177	80178	80179	80180	80181	80182	80183	80184	80185	80186	80187	80188	80189	80190	80191	80192	80193	80194	80195	80196	80197	80198	80199	80200	80201	80202	80203	80204	80205	80206	80207	80208	80209	80210	80211	80212	80213	80214	80215	80216	80217	80218	80219	80220	80221	80222	80223	80224	80225	80226	80227	80228	80229	80230	80231	80232	80233	80234	80235	80236	80237	80238	80239	80240	80241	80242	80243	80244	80245	80246	80247	80248	80249	80250	80251	80252	80253	80254	80255	80256	80257	80258	80259	80260	80261	80262	80263	80264	80265	80266	80267	80268	80269	80270	80271	80272	80273	80274	80275	80276	80277	80278	80279	80280	80281	80282	80283	80284	80285	80286	80287	80288	80289	80290	80291	80292	80293	80294	80295	80296	80297	80298	80299	80300	80301	80302	80303	80304	80305	80306	80307	80308	80309	80310	80311	80312	80313	80314	80315	80316	80317	80318	80319	80320	80321	80322	80323	80324	80325	80326	80327	80328	80329	80330	80331	80332	80333	80334	80335	80336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OPTION 3 WAS USED IN CALCULATING VOLUMES, FOR 18 REGIONS,
0-SET VOLUMES = 1, 1-CENTRIC SPHERES, 2-SLABS, 3-INPUT VOLUMES.

VOLUMES (CM**) USED IN COLLISIONS DENSITY AND TRACK LENGTH ESTIMATORS.									
REG	1	2	3	4	5	6	7	8	9
VOLUME	$3.1420+03$	$3.1420+03$	$3.1420+03$	$3.1420+03$	$3.1420+03$	$3.1420+03$	$3.1420+03$	$3.1420+03$	$3.1420+04$
REG	11	12	13	14	15	16	17	18	
VOLUME	$3.534D+04$	$1.0000D+00$							
NGEOM=	54913,								
NGLAST=	55792								

Fig.5.2 (Continued)

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XSEC OF L16,L17,016,SUS,ENDF/B4,DDX20

NUMBER OF PRIMARY GROUPS (NGP) 125
NUMBER OF PRIMARY DOWNSCATTERS (NDS) 125
NUMBER OF SECONDARY GROUPS (NGG) 0
NUMBER OF SECONDARY DOWNSCATTERS (NDSG) 0
NUMBER OF PRIM+SEC GROUPS (INGP) 125
TABLE LENGTH (ITBL) 128
LOC OF WITHIN GROUP (SIG GG) (ISGG)
NUMBER OF MEDIA (NMED) 1
NUMBER OF INPUT ELEMENTS (NELEM) 7
NUMBER OF MIXING ENTRIES (NMIX) 7
NUMBER OF COEFFICIENTS (NCOEF) 1
NUMBER OF ANGLES (NSCT) 20
RESTORE COEFF (ISTAT) 0
ADJOINT SWITCH (FROM MORSE) 0
DOUBLE DIFF. C.S. SWITCH 1

INPUT/OUTPUT OPTIONS
IRDG (AS READ) 0
ISTR (AS STORE) -1
IFMU (MUS) 0
IMOM (MOMENTS) 0
IPRIN (ANGLES,PROB) 0
IPUN (IMPOSSIBLE COEF) 0
CARD FORMAT (IDTF) 0
INPUT TAPE (IXTAP) 20
MORSEC TAPE (JXTAPE) 0
O6R TAPE (IO6RT) 0

ANGLE STRUCTURE
 0.900E+00 0.800E+00 0.700E+00 0.600E+00 0.500E+00 0.400E+00 0.300E+00 0.200E+00 0.100E+00 0.0
 -0.100E+00 -0.200E+00 -0.300E+00 -0.400E+00 -0.500E+00 -0.600E+00 -0.700E+00 -0.800E+00 -0.900E+00 -0.100E+00

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Fig.5.2 (Continued)

ELEMENTS FROM LIBRARY TAPE
IDENTIFIERS-1271-1272-1276-1192-1190-1191-1197

SUB RESTOR MATNO 1269

SUB RESTOR MATNO 1271

CARD INPUT--MATNO,SIGTO SIGPR,SIGC,SIGF,SIGFF = 1271 0.0

0.0 0.0 0.0 0.0

DDX LIBRARY INFORMATION
1RY GROUPS NUMBER 125
2RY - 125
ANGLES - 20

*** ELEMENT 1 ID= 1271 H G C
SUB RESTOR MATNO 1272 SIGPR,SIGC,SIGF,SIGFF = 1272 3.7207E+01 3.16047E+01 0.0
CARD INPUT--MATNO,SIGTO

0.0

DDX LIBRARY INFORMATION
1RY GROUPS NUMBER 125
2RY - 125
ANGLES - 20

*** ELEMENT 2 ID= 1272 H G JC
SUB RESTOR MATNO 1274
SUB RESTOR MATNO 1289
SUB RESTOR MATNO 1275
SUB RESTOR MATNO 1276
CARD INPUT--MATNO,SIGTO SIGPR,SIGC,SIGF,SIGFF = 1276 0.0

0.0 0.0 0.0 0.0

DDX LIBRARY INFORMATION
1RY GROUPS NUMBER 125
2RY - 125
ANGLES - 20

*** ELEMENT 3 ID= 1276 H G iC
SUB RESTOR MATNO 1156
SUB RESTOR MATNO 1280
SUB RESTOR MATNO 1193
SUB RESTOR MATNO 1194
SUB RESTOR MATNO 1150
SUB RESTOR MATNO 1195
SUB RESTOR MATNO 1191
SUB RESTOR MATNO 1192
CARD INPUT--MATNO,SIGTO SIGPR,SIGC,SIGF,SIGFF = 1192 1.6684E+01 1.4726E+00 1.9578E+00 0.0

DDX LIBRARY INFORMATION
1RY GROUPS NUMBER 125
2RY - 125
ANGLES - 20

*** ELEMENT 4 ID= 1192 H G-JC
SUB RESTOR MATNO 1190 SIGPR,SIGC,SIGF,SIGFF = 1190 0.0

0.0 0.0 0.0 0.0

DDX LIBRARY INFORMATION

Fig.5.2 (Continued)

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1RY GROUPS NUMBER      125
2RY      -              -
ANGLES     -              20

*** ELEMENT    5   ID=  1190   H   G-1C
SUB RESTOR MATNO  1197
SUB RESTOR MATNO  1295
SUB RESTOR MATNO  1287
SUB RESTOR MATNO  1288
SUB RESTOR MATNO  1269
SUB RESTOR MATNO  1271
SUB RESTOR MATNO  1272
SUB RESTOR MATNO  1274
SUB RESTOR MATNO  1289
SUB RESTOR MATNO  1275
SUB RESTOR MATNO  1276
SUB RESTOR MATNO  1156
SUB RESTOR MATNO  1280
SUB RESTOR MATNO  1193
SUB RESTOR MATNO  1194
SUB RESTOR MATNO  1150
SUB RESTOR MATNO  1195
SUB RESTOR MATNO  1191
CARD INPUT--MATNO,SIGTO SIGPR,SIGC,SIGF,SIGFF = 1191  0.0
                                                0.0   0.0   0.0

DDX LIBRARY INFORMATION
1RY GROUPS NUMBER      125
2RY      -              -
ANGLES     -              20

*** ELEMENT    6   ID=  1191   H   G- C
SUB RESTOR MATNO  1192
SUB RESTOR MATNO  1190
SUB RESTOR MATNO  1197
CARD INPUT--MATNO,SIGTO SIGPR,SIGC,SIGF,SIGFF = 1197  0.0
                                                0.0   0.0   0.0

DDX LIBRARY INFORMATION
1RY GROUPS NUMBER      125
2RY      -              -
ANGLES     -              20

*** ELEMENT    7   ID=  1197   H   G-aC
STORAGE ALLOCATIONS
CROSS SECTIONS START AT 55793
LAST LOCATION USED (PERM) 163144
TEMP LOCATIONS USED      877370 TO 880000
EXCESS STORAGE (TEMP)    714226
MAX. LOCATION (NLEFT)    880000

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Fig.5.2 (Continued)

MIXING TABLE

MEDIA	1	CONTAINS ELEMENT	1	WITH DENSITY	4.2729E-03
MEDIA	1	CONTAINS ELEMENT	2	WITH DENSITY	5.3422E-02
MEDIA	1	CONTAINS ELEMENT	3	WITH DENSITY	2.8848E-02
MEDIA	1	CONTAINS ELEMENT	4	WITH DENSITY	1.0785E-03
MEDIA	1	CONTAINS ELEMENT	5	WITH DENSITY	1.3086E-04
MEDIA	1	CONTAINS ELEMENT	6	WITH DENSITY	2.9922E-04
MEDIA	1	CONTAINS ELEMENT	-7	WITH DENSITY	2.3921E-05
			BANKS START AT	163145	
			LAST LOCATION USED	283144	

Fig.5.2 (Continued)

TRACK DETECTOR

ND= 17, NNE=125, NE=125, NT= 0, NA= 0, NRESP= 1, NX= 5, NXHD= 2
 NTAPE= 8, NNBAT= 0, NRUN2= 0

*** FOLLOWING ESTIMATOR IS USED ***

UNIT OF FLUX (PER UNIT LETHARGY)

NEXT TO DET. 17 <----> TRACK LENGTH ESTIMATOR

DET X Y Z RAD

GROUP	RESP(1)
1	1.0000E+00
2	1.0000E+00
3	1.0000E+00
4	1.0000E+00
5	1.0000E+00
6	1.0000E+00
7	1.0000E+00
8	1.0000E+00
9	1.0000E+00
10	1.0000E+00
11	1.0000E+00
12	1.0000E+00
13	1.0000E+00
14	1.0000E+00
15	1.0000E+00
16	1.0000E+00
17	1.0000E+00
18	1.0000E+00
19	1.0000E+00
20	1.0000E+00
21	1.0000E+00
22	1.0000E+00
23	1.0000E+00
24	1.0000E+00
25	1.0000E+00
26	1.0000E+00
27	1.0000E+00
28	1.0000E+00
29	1.0000E+00
30	1.0000E+00
31	1.0000E+00
32	1.0000E+00
33	1.0000E+00
34	1.0000E+00
35	1.0000E+00
36	1.0000E+00
37	1.0000E+00
38	1.0000E+00
39	1.0000E+00
40	1.0000E+00
41	1.0000E+00
42	1.0000E+00
43	1.0000E+00
44	1.0000E+00
45	1.0000E+00
46	1.0000E+00
47	1.0000E+00
48	1.0000E+00
49	1.0000E+00
50	1.0000E+00
51	1.0000E+00

Fig.5.2 (Continued)

TIME REQUIRED FOR INPUT WAS 5 SECONDS.
YOU ARE USING THE DEFAULT VERSION OF STRUN WHICH DOES NOTHING.

***START BATCH 1

SOURCE DATA

```
*****  
*** SOURCE INPUT DATA ***  
*****
```

*** SOURCE NO. 1 INTENSITY = 1.00000E+00

*** POINT SOURCE ***

XO,YO,ZO,X1,X2,Y1,Y2,Z1,Z2 = 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0

MONODIRECTIONAL EMISSION

UO,VO,W0 0.0 0.0

ENERGY SPECTRUM OF SOURCE NEUTRON

ENERGY	FLUX	FLUX(G)	DFLUX
0.1555E+08	0.2261E+01	0.5621E+02	-2
0.1500E+08	0.1680E+02	-0.2431E+02	0.2586697E+00
0.1448E+08	0.7056E+01	-0.8183E+02	0.6595361E+00
0.1397E+08	0.3794E+00	-0.12666E+02	0.7411225E+00
0.1348E+08	0.2417E+00	-0.21211E+02	0.7520046E+00
0.1301E+08	0.1133E+00	-0.1247E+02	0.7580578E+00
0.1248E+08	0.6757E-01	-0.1320E+02	0.7617241E+00
0.1166E+08	0.2768E-01	0.3133E+01	0.7647465E+00
0.1062E+08	0.3716E-01	0.12735E+00	0.7718557E+00
0.9519E+07	0.3768E-01	-0.1532E+01	0.7760501E+00
0.8439E+07	0.3209E-01	0.1977E+01	0.7803946E+00
0.7485E+07	0.4067E-01	-0.1053E+01	0.7849612E+00
0.6642E+07	0.3586E-01	0.1722E+01	0.7897801E+00
0.5886E+07	0.4418E-01	-0.3719E+00	0.7950214E+00
0.5213E+07	0.4223E-01	0.2477E+01	0.8010026E+00
0.4617E+07	0.5704E-01	0.1517E+01	0.8098265E+00
0.4018E+07	0.7043E-01	-0.9965E+00	0.82222137E+00
0.3312E+07	0.5808E-01	0.1022E+01	0.84774730E+00
0.2511E+07	0.8394E-01	-0.6280E-01	0.8929933E+00
0.1331E+07	0.8108E-01	-0.2655E+00	0.9448886E+00
0.6600E+06	0.6731E-01	-0.1195E+01	0.9848936E+00
0.2340E+06	0.1949E-01	-0.1392E+01	0.996744E+00
0.6100E+05	0.3002E-02	-0.7831E+00	0.1000000E+01
0.5400E+04	0.4496E-03	0.0	0.1000000E+01
WTAVE	IAVE	WAVE	YAVE
1.000E+04	19.69	0.0	0.0
		1.0000	0.0

NUMBER OF COLLISIONS OF TYPE NCOLL

SOURCE SPLIT(D)	FISHN	GAMGEN	REALCOLL	ALBEDO	BDRYX	ESCAPE	E-CUT	TIMEKILL	R R KILL	R SURV	GAMLOST
10000	0	0	0	197754	0	87476	9611	0	0	389	0

TIME REQUIRED FOR THE PRECEDING BATCH WAS 37 SECONDS.

***START BATCH 2

SOURCE DATA

WTAVE

AGEAVE

AGEAVE

RANDOM=EEE021E3A1E?	TIMEKILL	R R KILL	R SURV	GAMLOST
	0	0	389	0

Fig.5.2 (Continued)

```

1.000E+04   19.51   0.0   0.0   1.0000   0.0   0.0   0.0   0.0   0.0   0.0   0.0

NUMBER OF COLLISIONS OF TYPE NCOLL
SOURCE SPLIT(D)   FISHN   GAMGEN   REALCOLL   ALBEDO   BDRYX   *E-CUT TIMEKILL R R KILL R R SURV GAMLOST
          0         0         0        197229      0       87361      0           0           399      0           0           0

TIME REQUIRED FOR THE PRECEDING BATCH WAS 36 SECONDS.

***START BATCH 3
RANDOM=D94FDD73D787

SOURCE DATA
WAVEAVE  IAVE    UAVE    VAVE    WAVE    XAVE    YAVE    ZAVE    AGEAVE
1.000E+04 19.45   0.0     0.0     1.0000   0.0     0.0     0.0     0.0     0.0

NUMBER OF COLLISIONS OF TYPE NCOLL
SOURCE SPLIT(D)   FISHN   GAMGEN   REALCOLL   ALBEDO   BDRYX   E-CUT TIMEKILL R R KILL R R SURV GAMLOST
          0         0         0        195568      0       86411      0           0           369      0           0           0

TIME REQUIRED FOR THE PRECEDING BATCH WAS 36 SECONDS.

RUN TERMINATED BY EXECUTION TIME LIMIT
0 RUNS OF 40 BATCHES AND 1 RUN OF 3 BATCHES COMPLETED.

** NEXT RANDOM NUMBER IS CF79EF8B5727
RESTART INFORMATION HAS WRITTEN INTO FILE UNIT 9 AFTER 3 BATCHES IS COMPLETED

```

Fig.5.2 (Continued)

THIS CASE WAS RUN ON TUESDAY, JUNE 12, 1984

INTEGRATED FLUX

DETECTOR	RESPONSES(DETECTOR)		INTEGRATED FLUX TOTAL
	FSD UNCOLL	FSD RESPONSE UNCOLL	
1	0.0	0.0	4.8649E-03
2	0.0	0.0	3.0642E-03
3	0.0	0.0	1.5331E-03
4	0.0	0.0	6.9913E-04
5	0.0	0.0	3.1623E-04
6	0.0	0.0	1.4841E-04
7	0.0	0.0	6.7098E-05
8	0.0	0.0	2.5875E-05
9	0.0	0.0	2.3822E-04
10	0.0	0.0	3.3114E-04
11	0.0	0.0	2.7570E-04
12	0.0	0.0	1.8200E-04
13	0.0	0.0	1.0451E-04
14	0.0	0.0	5.6287E-05
15	0.0	0.0	2.8186E-05
16	0.0	0.0	1.0988E-05
17	0.0	0.0	1.0000E+01

Fig.5.2 (Continued)

DETECTOR NO.	FLUENCE(ENERGY, DETECTOR) /LETHARGY									
	1	2	3	4	5	6	7	8	9	10
ENERGIES 1.649E+07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
1.623E+07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
1.598E+07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
1.573E+07	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
1.549E+07	1.242E-03	3.962E-04	1.201E-04	2.650E-05	0.0	0.0	0.0	0.0	0.0	0.0
1.525E+07	0.140	0.212	0.308	0.317	0.0	0.0	0.0	0.0	0.0	0.0
1.501E+07	8.711E-03	2.822E-03	9.104E-04	2.787E-04	8.759E-05	3.580E-05	2.744E-05	1.359E-05	0.0	7.840E-07
1.478E+07	0.014	0.041	0.052	0.041	0.096	0.379	0.423	1.000	0.0	0.587
1.455E+07	2.135E-02	7.402E-03	2.488E-03	8.147E-04	2.454E-04	5.722E-05	8.288E-06	8.812E-06	3.002E-07	7.538E-07
1.432E+07	0.008	0.003	0.063	0.049	0.155	0.221	0.656	0.804	1.000	0.121
1.410E+07	2.264E-02	1.021E-02	3.613E-03	1.020E-03	3.147E-04	1.350E-04	4.466E-05	1.483E-05	2.475E-07	9.381E-06
1.388E+07	0.018	0.038	0.040	0.107	0.169	0.106	0.393	0.613	0.921	0.312
1.367E+07	1.470E-02	7.816E-03	3.144E-03	1.045E-03	2.914E-04	6.267E-05	1.461E-05	6.2336E-06	2.981E-06	2.822E-05
1.346E+07	7.155E-03	5.274E-03	2.335E-03	8.791E-04	2.699E-04	7.305E-05	2.996E-05	1.024E-05	6.136E-06	4.313E-05
1.324E+07	0.017	0.008	0.027	0.038	0.056	0.355	0.332	0.111	0.231	0.122
1.304E+07	4.205E-03	3.706E-03	1.681E-03	6.693E-04	2.670E-04	7.831E-05	2.511E-05	6.218E-06	8.617E-06	6.098E-05
1.284E+07	0.027	0.012	0.097	0.164	0.157	0.157	0.343	0.564	0.552	0.291
1.264E+07	3.669E-03	2.989E-03	1.365E-03	4.638E-04	1.481E-04	5.560E-05	6.049E-06	0.0	1.374E-05	7.419E-05
1.243E+07	0.019	0.050	0.008	0.038	0.156	0.423	0.258	0.0	0.191	0.130
1.222E+07	2.710E-03	2.245E-03	1.055E-03	3.638E-04	1.660E-04	7.395E-05	2.446E-05	1.321E-05	2.544E-05	7.114E-05
1.201E+07	0.027	0.003	0.061	0.044	0.066	0.361	0.349	0.213	0.184	0.126
1.180E+07	2.436E-03	2.088E-03	9.988E-04	2.673E-04	6.809E-05	3.318E-05	6.169E-06	1.2336E-06	3.190E-05	6.674E-05
1.159E+07	0.014	0.078	0.019	0.015	0.203	0.601	0.295	1.000	0.092	0.012
1.138E+07	1.864E-03	1.5322E-03	6.546E-04	2.709E-04	9.347E-05	5.202E-05	7.059E-06	5.761E-06	2.608E-05	5.508E-05
1.117E+07	0.035	0.067	0.074	0.163	0.270	0.372	1.000	1.000	0.191	0.038
1.096E+07	1.729E-03	1.310E-03	5.734E-04	2.690E-04	1.137E-04	4.169E-05	1.566E-05	1.654E-05	3.730E-05	6.318E-05
1.075E+07	0.032	0.025	0.030	0.135	0.273	0.386	0.743	0.608	0.045	0.059
1.054E+07	1.461E-03	1.063E-03	4.917E-04	1.9322E-04	7.067E-05	1.324E-05	1.733E-05	0.0	3.973E-05	7.218E-05
1.033E+07	0.047	0.042	0.037	0.260	0.157	0.325	0.571	0.0	0.110	0.008

Fig.5.2 (Continued)

NEUTRON DEATHS	
KILLED BY RUSSIAN ROULETTE	NUMBER 1157
ESCAPED	WEIGHT 4.41579E-08
REACHED ENERGY CUTOFF	28843 2.19127E+04
REACHED TIME CUTOFF	0 0.0
	0 0.0
NUMBER OF SCATTERINGS	
MEDIUM	NUMBER
1	590451
TOTAL	590451

Fig.5.2 (Continued)

REAL SCATTERING COUNTERS

ENERGY GROUP	REGION 1		REGION 2		REGION 3		REGION 4		REGION 5		REGION 6	
	NUMBER	WEIGHT										
1	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
2	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
3	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
4	222	2.22E+02	82	8.20E+01	23	2.30E+01	11	1.10E+01	0	0.0	0	0.0
5	1687	1.69E+03	560	5.59E+02	166	1.66E+02	52	5.19E+01	13	1.30E+01	1	9.95E-01
6	4264	4.26E+03	1388	1.39E+03	487	4.86E+02	152	1.51E+02	50	4.98E+01	14	1.39E+01
7	5684	5.68E+03	2141	2.13E+03	718	7.14E+02	216	2.15E+02	53	5.27E+01	21	2.08E+01
8	4395	4.39E+03	2048	2.04E+03	743	7.37E+02	239	2.37E+02	62	6.13E+01	24	2.37E+01
9	2964	2.95E+03	1559	1.54E+03	603	5.96E+02	205	2.02E+02	55	5.42E+01	17	1.67E+01
10	1423	1.41E+03	1040	1.03E+03	452	4.44E+02	188	1.84E+02	57	5.57E+01	17	1.65E+01
11	815	8.05E+02	752	7.38E+02	323	3.16E+02	127	1.24E+02	39	3.77E+01	14	1.36E+01
12	742	7.32E+02	582	5.70E+02	254	2.48E+02	83	8.03E+01	32	3.10E+01	10	9.65E+00
13	575	5.66E+02	468	4.57E+02	224	2.17E+02	70	6.75E+01	27	2.60E+01	9	8.59E+00
14	504	4.95E+02	402	3.92E+02	173	1.68E+02	64	6.18E+01	18	1.72E+01	5	4.76E+00
15	388	3.81E+02	341	3.31E+02	158	1.53E+02	58	5.56E+01	20	1.90E+01	15	1.44E+01
16	313	3.07E+02	273	2.65E+02	131	1.26E+02	55	3.79E+01	19	1.79E+01	4	3.83E+00
17	282	2.76E+02	238	2.30E+02	105	1.01E+02	41	3.93E+01	19	1.80E+01	3	2.78E+00
18	274	2.68E+02	212	2.05E+02	100	9.57E+01	28	2.67E+01	17	1.60E+01	8	7.43E+00
19	231	2.25E+02	166	1.60E+02	84	8.02E+01	26	2.49E+01	23	2.12E+01	10	9.38E+00
20	224	2.18E+02	154	1.48E+02	92	8.77E+01	32	3.03E+01	13	2.23E+01	8	7.49E+00
21	210	2.05E+02	145	1.39E+02	69	6.55E+01	26	2.47E+01	16	1.51E+01	13	1.21E+01
22	213	2.06E+02	141	1.35E+02	52	4.98E+01	31	2.90E+01	17	6.35E+00	13	2.66E+00
23	178	1.72E+02	113	1.08E+02	55	5.22E+01	27	2.54E+01	5	4.63E+00	3	2.65E+00
24	171	1.66E+02	108	1.04E+02	59	5.52E+01	17	1.59E+01	7	6.62E+00	5	4.18E+00
25	187	1.80E+02	98	9.31E+01	60	5.59E+01	24	2.28E+01	10	9.18E+00	2	1.73E+00
26	151	1.46E+02	105	1.00E+02	56	5.24E+01	29	2.70E+01	8	7.20E+00	3	1.77E+00
27	174	1.69E+02	95	9.14E+01	37	3.49E+01	16	1.51E+01	8	7.45E+00	3	2.89E+00
28	159	1.55E+02	92	8.86E+01	38	3.60E+01	14	1.29E+01	5	4.73E+00	3	2.89E+00
29	189	1.84E+02	79	7.60E+01	34	3.23E+01	16	1.59E+01	7	6.55E+00	5	4.61E+00
30	148	1.44E+02	76	7.27E+01	41	3.89E+01	16	1.48E+01	7	6.55E+00	4	3.79E+00
31	124	1.20E+02	98	9.31E+01	51	4.75E+01	18	1.67E+01	13	1.22E+01	3	2.63E+00
32	156	1.51E+02	111	1.06E+02	42	3.96E+01	21	1.97E+01	4	3.79E+00	2	1.85E+00
33	547	5.27E+02	355	3.38E+02	166	1.56E+02	80	7.41E+01	32	2.89E+01	13	1.16E+01
34	557	5.36E+02	344	3.26E+02	172	1.62E+02	75	6.97E+01	23	2.09E+01	9	7.92E+00
35	542	5.22E+02	394	3.77E+02	159	1.49E+02	71	6.61E+01	25	2.28E+01	13	1.17E+01
36	575	5.57E+02	387	3.68E+02	157	1.49E+02	77	7.20E+01	26	2.41E+01	15	4.38E+00
37	565	5.46E+02	382	3.64E+02	178	1.68E+02	93	8.59E+01	24	2.18E+01	5	4.52E+00
38	515	4.99E+02	323	3.08E+02	157	1.47E+02	71	6.56E+01	22	2.00E+01	6	5.51E+00
39	508	4.89E+02	316	2.99E+02	168	1.57E+02	71	6.57E+01	30	2.75E+01	8	7.27E+00
40	506	4.87E+02	357	3.38E+02	162	1.51E+02	78	5.35E+01	24	2.13E+01	7	6.65E+00
41	478	4.60E+02	356	3.37E+02	156	1.46E+02	53	4.89E+01	31	2.84E+01	13	1.17E+01
42	684	4.66E+02	338	3.20E+02	150	1.41E+02	76	7.01E+01	19	1.75E+01	13	1.11E+01
43	548	5.27E+02	331	3.15E+02	162	1.52E+02	71	6.59E+01	27	2.43E+01	10	9.14E+00
44	578	5.56E+02	376	3.56E+02	142	1.32E+02	81	7.52E+01	26	2.50E+01	9	7.80E+00
45	597	5.76E+02	369	3.51E+02	150	1.39E+02	76	6.91E+01	24	2.20E+01	7	6.08E+00
46	614	5.92E+02	371	3.51E+02	176	1.64E+02	76	7.01E+01	21	2.17E+01	13	1.15E+01
47	665	6.42E+02	381	3.61E+02	184	1.71E+02	79	7.33E+01	25	2.24E+01	7	6.46E+00
48	732	7.08E+02	445	4.22E+02	175	1.64E+02	79	7.28E+01	26	2.43E+01	15	1.33E+01
49	680	6.58E+02	428	4.05E+02	192	1.80E+02	89	8.26E+01	38	3.40E+01	5	4.59E+00
50	821	7.96E+02	511	4.84E+02	188	1.77E+02	97	9.05E+01	47	4.19E+01	11	9.35E+00
51	721	6.97E+02	494	4.68E+02	231	2.17E+02	82	7.57E+01	32	2.90E+01	14	1.21E+01
52	725	6.99E+02	452	4.31E+02	218	2.02E+02	82	7.50E+01	25	2.29E+01	11	9.80E+00
53	654	6.34E+02	448	4.27E+02	213	2.00E+02	92	8.46E+01	42	3.86E+01	12	1.04E+01
54	721	6.97E+02	497	4.71E+02	251	2.35E+02	99	9.11E+01	34	3.08E+01	12	1.10E+01
55	655	6.34E+02	454	4.33E+02	237	2.21E+02	83	7.73E+01	33	3.09E+01	14	1.22E+01
56	661	5.43E+02	409	3.89E+02	191	1.78E+02	81	7.53E+01	25	2.29E+01	18	1.62E+01
57	605	5.87E+02	422	4.03E+02	191	1.80E+02	82	7.45E+01	27	2.46E+01	15	1.39E+01
58	627	6.09E+02	399	3.81E+02	221	2.09E+02	73	6.72E+01	28	2.56E+01	18	1.64E+01

Fig. 5.2 (Continued)

6. AUXILIARY PROGRAMS

6.1 REACT

REACT code calculates the reaction rate by using the output from MORSE-DD (unit 95) and the prepared reaction cross section library, and plots the spatial distribution of the calculated reaction rates. The ID numbers of reaction cross sections prepared are shown in Appendix.

The input for the REACT code is as follows:

CARD A (*)

ND number of detectors

NE absolute value of NE is the number of energy groups .

If NE is negative, cross section of /NE/'th group is read from a card.

NRES number of reactions to be calculated.

IPLOT 0/1 = not plotted/plotted

IN logical unit of flux input.

IFLUX -1/0/1 = output flux from ANISN-DD/MORSE/ANISN

ILETH unit of flux

0/1 = $\phi(E)/\phi(u)$

IND number of sets of detectors

RNORM normalization factor

ICON -1/0/1 = cumurative/no effect/groupwise

CARD B (*) Omit if IND = 0

(IMX(I), I+1, IND)

IMX(IND) must be ND. If IMX(I) is negative, the reaction rate calculated for the detectors (IMX(I-1) + 1 ~ IMX(I)) are not plotted.

* free format

CARD C

Title for the input flux.

CARD D (*)

(R(I), I=1, ND) x-coordinates for the detector I.

Cards E, E' are repeated by NRES sets.

CARD E (2A4)

ID number for reaction cross section in a PDS file.

CARD E' (*) Enter if NE < 0

cross section of |NE| 'th group.

If IPLOT on CARD A is zero, return to CARD A. The following input is used to plot the results and to obtain reaction rates of any positions by the interpolation.

CARD F (10A4)

main title for the figure.

CARD G (*)

XWIDTH length of X-axis.

UWIDTH length of Y-axis.

Cards H and I are repeated for all the cases to be plotted which are specified by the positive IMX on CARD B.

CARD H (*)

NEXP number of spatial points at which the reaction rates are required

IUNIT dummy input

NOPT interpolation method

- 2 linear-linear.
- 1 linear-log.
- 1 linear-log. Akima's interpolation method is used.
- 2 linear-linear. Akima's interpolation method is used.

CARD I (*)

(RR(I), I=1, NEXP) x-coordinates of points at which the reaction rates are required.

Return to CARD A.

A sample input and job control statements are shown in Fig. 6.1.

6.2 Supplimentary Subroutines for MORSE-DD

Auxiliary subroutine package MORSEUP is prepared for several different-types of calculation.

Module BDRYXC

Current through the sphere is estimated by the surface cross estimator. Card input for BDRYXC is the same as that for spherical surface cross estimator of flux in MORSE-DD.

Module BDRZC

BDRZC is the current estimator for the circular plane perpendicular to z-direction. Input is the same as that for the flux estimator.

Module DIRECX

Path-length stretching is carried out on the x-component of path.*

Module DIRECZ

Path-length stretching is carried out on the z-component of path. *

- * In the standard MORESE-DD, path-length stretching is performed on the radial component of path.

Module COMPRESS

The present version of MORSE-DD needs a large core memory to analyze a complex problem. If users do not need any information about random walk simulation such as numbers of real collision, energy cut off and Russian roulette kill, they can save effectively core memory by using module COMPRES. The core memory size reduced by using module COMPRESS is about $(11 \times \text{MXREG} \times \text{MAXGP} + 8 \times \text{NMTG} \times \text{MEDIA})$.

```

//JCLG JOB
//JCLG EXEC JCLG
//SYSIN DD DATA,DLM='++'
// JUSER XXXXXXXX,MORI.TAKAMAS,0431.100,FIBJOB
  C.4 T.1 W.0 I.3 GRP
  OOPTP PASSWORD=XX,MSGCLASS=R
// EXEC LMG0,LM='J3803.XXMORSE',PNM=REACT
// EXPAND GRNLP,SYSOUT=M
//RESPONSE DD DSN=J3803.PROFDDX2.PDS124G.DATA,DISP=SHR
//FT10F001 DD DSN=J3803.PLTSLAB.DATA(DDX2),DISP=SHR
//FT11F001 DD DSN=J3803.PLTSLAB.DATA(DDXLIB1),DISP=SHR
//SYSIN DD *
   17 -125      1      0     10      0      1      3  0.053422      1
   8      16     -17
DDX-OLD
 5. 7*10. 5. 7*10. 200.
L17091
0.
   17 -125      1      1     11      0      1      3  0.053422      1
   8      16     -17
DDX-NEW
 5. 7*10. 5. 7*10. 200.
L17091
0.
REACTION RATE IN L120 SLAB
150 200
0 0 0
0 0 0
0 0 0
0 0 0
++
//
```

Fig.6.1a Job control statements and sample input data for REACT

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Fig.6.1b Sample output of REACT

		FLUX FROM DDX-NEW															
ND	NE	NRES	IPLT	IN	ILFLUX	ILETH	IND	RNOLM	17	-125	1	1	11	0	1	3	5.3422E-02
SAME	DETECTOR TO DET.NO.	OF	----,	IF NEGATIVE,	NOT	PLOTTED											
8	16	-17															
DISTANCE																	
0.50000E+01	0.15000E+02	0.25000E+02	0.35000E+02	0.45000E+02	0.55000E+02	0.65000E+02	0.75000E+02	0.50000E+01	0.15000E+02	0.50000E+02	0.65000E+02	0.75000E+02	0.50000E+02	0.65000E+02	0.75000E+02	0.50000E+01	0.15000E+02
0.25000E+02	0.35000E+02	0.45000E+02	0.55000E+02	0.65000E+02	0.75000E+02	0.20000E+03											
REACTION CROSS SECTION -- L17091																	
RESCX	NE																
0.0																	
2.9810E-01	3.0030E-01	3.0246E-01	3.0464E-01	3.0674E-01	3.0880E-01	3.1263E-01	3.1824E-01	3.2389E-01	3.2953E-01								
3.3505E-01	3.3938E-01	3.4367E-01	3.4786E-01	3.5201E-01	3.5655E-01	3.6161E-01	3.6702E-01	3.7310E-01	3.7913E-01								
3.8052E-01	3.8463E-01	3.8869E-01	3.9231E-01	3.9549E-01	3.9868E-01	4.0197E-01	4.0530E-01	4.0854E-01	4.1072E-01								
4.1200E-01	4.1331E-01	4.1542E-01	4.1835E-01	4.2107E-01	4.2381E-01	4.285E-01	4.3168E-01	4.3559E-01	4.3959E-01								
3.7033E-01	3.2616E-01	2.4619E-01	1.5423E-01	8.8714E-02	4.6806E-02	2.3410E-02	1.4920E-02	1.0168E-02	5.7344E-03								
2.7520E-03	1.6998E-03	2.6672E-04	0.0	0.0	0.0	0.0	0.0	0.0	0.0								
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0								
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0								
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0								
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0								
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0								
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0								
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0								
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0								
ND(R)	L17091																
1	4.5665E-05	5.8890E-07	1.3610E-01														
	1.49E-03	4.63E-03	1.56E-03														
2	2.4568E-05	8.0387E-07	0.0														
	2.17E-03	3.97E-03	0.0														
3	1.0022E-05	6.8333E-07	0.0														
	3.31E-03	4.34E-03	0.0														
4	3.6725E-06	4.2414E-07	0.0														
	5.32E-03	5.43E-03	0.0														
5	1.2889E-06	2.2233E-07	0.0														
	9.09E-03	7.30E-03	0.0														
6	4.4183E-07	1.0498E-07	0.0														
	1.55E-02	1.04E-02	0.0														
7	1.5511E-07	4.5778E-08	0.0														
	2.58E-02	1.56E-02	0.0														
8	5.1999E-08	1.8611E-08	0.0														
	4.33E-02	2.40E-02	0.0														

REACTION RATE IN L120 SLAB

XWIDYWD
0.15000E+03 0.20000E+03

JMAX1= 4

Fig.6.1b (Continued)

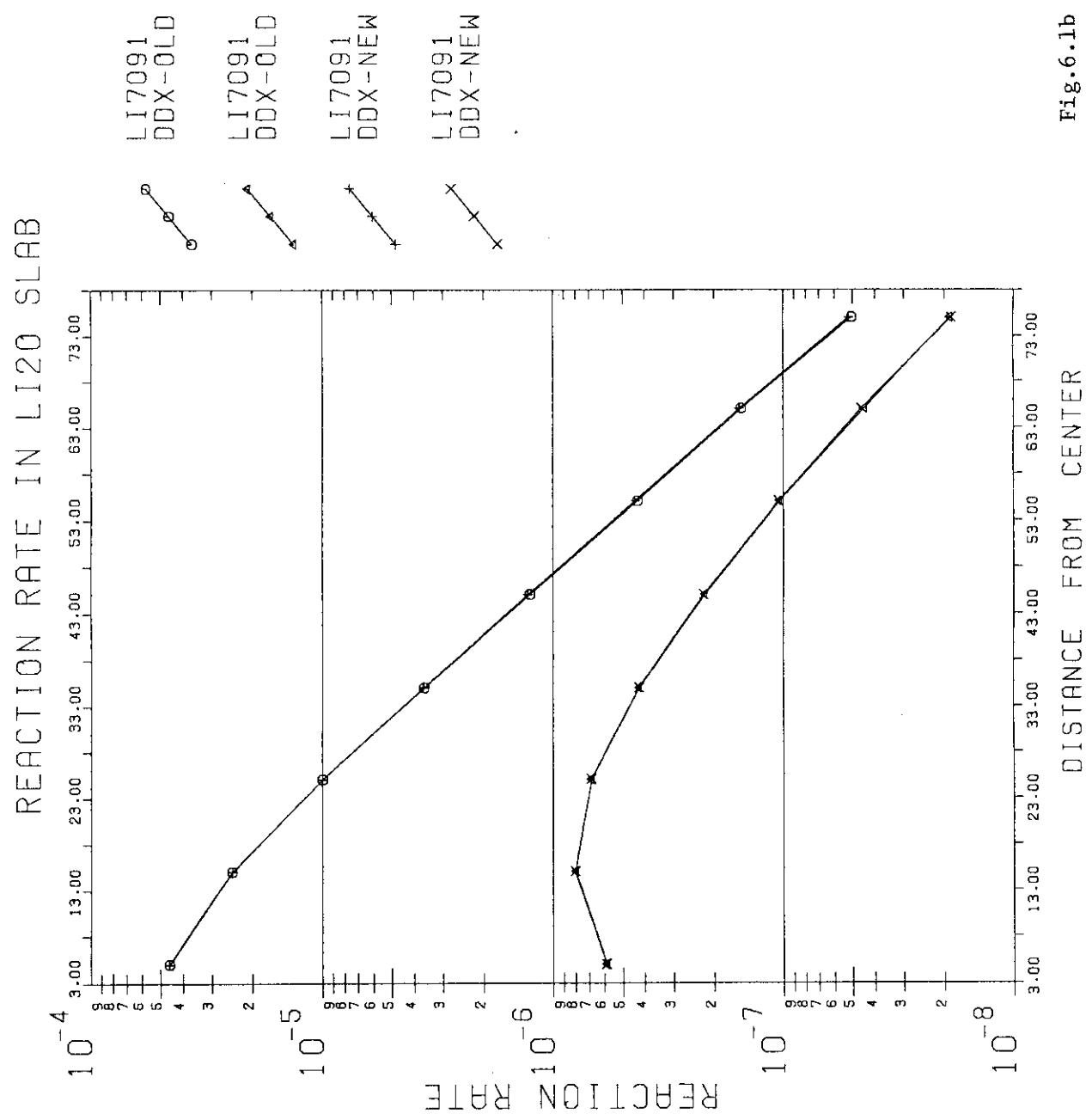


Fig.6.1b (Continued)

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- 2) Nakagawa, M., Mori, T. and Ishiguro, Y., Proc. Sixth Int. Conf. on Radiation Shielding, Vol.I, 171-179, Tokyo (1983)
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- 4) Brockmann, H., Nucl. Sci. Eng. 77, 377 (1981)
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APPENDIX TABLE OF MATERIAL NUMBERS IN CROSS SECTION LIBRARY

Table A.1 Identification number of elements
in 124-group DDXLIB1 library

Element	ID number		Element	ID number	
	ENDF/B4	JENDL-3PR1		ENDF/B4	JENDL-3PR1
H	1269		¹⁴ Si	1194	
⁶ Li	1271	306	K	1150	
⁷ Li	1272	307	Ca	1195	
¹² C	1274	612	Cr	1191	2400
⁹ Be	1289		Fe	1192	2600
¹⁴ N	1275		Ni	1190	2800
¹⁶ O	1276	816	⁵⁵ Mn	1197	
²³ Na	1156		Cu	1295	
Mg	1280		Mo	1287	
²⁷ Al	1193		Pb	1288	

Table A.2 Identification of reaction cross section
in 124-group response library

Reaction	ID name	Data library	MAT	MT
$^6\text{Li}(\text{n},\alpha)\text{T}$	LI6107	ENDF/B-IV	1271	107
	LIJ107	JENDL-3PR1	306	107
$^7\text{Li}(\text{n},\text{n}'\alpha)\text{T}$	LI7091	ENDF/B-IV	1272	91
	LIJ205	JENDL-3PR1	307	205
$^{27}\text{Al}(\text{n},\alpha)^{24}\text{Na}$	Al7107	ENDF/B-IV	6193	107
	Al7107B5	ENDF/B-V	6313	107
$^{58}\text{Ni}(\text{n},2\text{n})^{57}\text{Ni}$	NI8016	ENDF/B-IV	6419	16
	NI8016B5	ENDF/B-V	6433	16
$^{58}\text{Ni}(\text{n},\text{p})^{58}\text{Co}$	NI8103	ENDF/B-IV	6419	103
	NI8103B5	ENDF/B-V	6433	103
$^{197}\text{Au}(\text{n},2\text{n})^{196}\text{Au}$	AU7016	ENDF/B-IV	1283	16
	AU7016B5	ENDF/B-V	6379	16
$^{197}\text{Au}(\text{n},\gamma)$	AU7102	ENDF/B-IV	1283	102
	AU7102B5	ENDF/B-V	6379	102
$^{115}\text{In}(\text{n},\text{n}')^{115m}\text{In}$	IN5051B5	ENDF/B-V	6437	51
$^{115}\text{In}(\text{n},\gamma)$	IN5102B5	ENDF/B-V	6437	102
$^{235}\text{U}(\text{n},\text{f})$	U25018	ENDF/B-IV	1261	18
$^{238}\text{U}(\text{n},\text{f})$	U28018	ENDF/B-IV	1262	18
$^{238}\text{U}(\text{n},\gamma)$	U28102	ENDF/B-IV	1262	102
$^{232}\text{Th}(\text{n},\text{f})$	TH2018	ENDF/B-IV	1296	18
$^{237}\text{Np}(\text{n},\text{f})$	NP7018	ENDF/B-IV	1263	18
$^{239}\text{Pu}(\text{n},\text{f})$	PU9018	ENDF/B-IV	1264	18
$^{239}\text{Pu}(\text{n},\gamma)$	PU9102	ENDF/B-IV	1264	102