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j.r.beeler
j.w.ryan
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Monte Carlo
Research Series

monte carlo
program for cylindrical
geometry: part one
physics and method

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ABSTRACT

This report describes a Monte Carlo program for the IBM 704. It is intended to be primarily useful in the computation of intensive reactor parameters. The program is suited to cylindrical geometry wherein parallel tubular regions are arbitrarily located in a matrix of concentric circular annular regions. Normally the annular regions correspond to moderator and/or reflector regions and the interjected tubular regions to fuel elements.

The program requires a 32,768 word core memory and four tape units. Package PK-1 from the library of General Electric packages is also required.

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LIST OF SYMBOLS

- m_i denotes unique area in x - y plane, either a cylindrical tube or an annular ring.
- m_n denotes new m value.
- m_t denotes last cylindrical tube number.
- m_{max} denotes last (outermost) annular ring number.
- n_1 denotes the lower longitudinal division of the reactor (bottom cap).
- n_2 denotes the center longitudinal division of the reactor (center region).
- n_3 denotes the upper longitudinal division of the reactor (top cap).
- x_0 denotes the initial x coordinate of neutron in cm.
- y_0 denotes the initial y coordinate of neutron in cm.
- z_0 denotes the initial z coordinate of neutron in cm.
- x_1 denotes x coordinate of intersection of closest annular circle or tube with path line of neutron in cm.
- y_1 denotes y coordinate of intersection of closest annular circle or tube with path line of neutron in cm.
- z_1 denotes the z coordinate of intersection of closest annular circle or tube with path line of neutron in cm.
- x denotes present x coordinate of neutron in cm.
- y denotes present y coordinate of neutron in cm.
- z denotes present z coordinate of neutron in cm.
- β denotes fission yield.
- ϵ_g denotes minimum deviation in k_n between generations before halting program.
- δ denotes multiplication factor to increase generation.
- k_n denotes generation multiplication factor $\frac{\text{Fission points in old table}}{\text{Fission points in new table}}$
- N denotes number of fission positions in old table
- U denotes lethargy of neutron.
- θ denotes the angle of travel of the neutron with respect to the z axis in radians. (Varies from 0 to π radians.) (polar angle fixed coordinate system)

LIST OF SYMBOLS (cont'd)

- ϕ denotes the angle of travel of the neutron in the x - y plane with respect to the positive x axis in radians. (Varies from 0 to 2π radians.) (Azimuth angle fixed coordinate system).
- λ denotes the jump length selected for the neutron in cm.
- α denotes the record number of table No. 1 for desired n - m region.
 $\alpha = (n - 1) m_{\max} + m$
- γ denotes length along path line in x - y plane to change of interface in cm.
- $\gamma(m)$ denotes length along path line in x - y plane to new annular circle in cm.
- γ^* denotes length along path line in x - y plane to intersection with cylindrical tube in cm.
- p denotes perpendicular distance between path line and center of cylindrical tube or annular circle in cm.
- d denotes distance from intersection of p and path line to outside of cylindrical tube or annular circle in cm.
- DP denotes direction of path of neutron in an annular circle.
- V_m denotes dot product of unit vector along path line, and line between center of cylindrical tube and the neutron take-off point.
- R denotes radius of the (x_1, y_1) point from the center of all annular circles in cm.
- Δ denotes the distance the neutron travels in the x - y plane when executing a jump
- S denotes the birth-to-capture distance of a neutron in cm.
- h_i denotes the x coordinate of the axis of the i^{th} tube circle or annular ring in cm.
- k_i denotes the y coordinate of the axis of the i^{th} tube circle or annular ring in cm.
- a_i denotes the radius of the i^{th} tube circle or annular ring in the x - y plane in cm.
- σ_s denotes microscopic cross-section for scattering in barns.
- σ_a denotes microscopic cross-section for absorption in barns.
- σ_f denotes microscopic cross-section for fission in barns.

1.0 Introduction

A Monte Carlo Calculation (MCC) for neutron transport problems can often be described as a group of seven semi-independent computing subroutines which are connected with each other through two memory banks. These seven subroutines are:

- 1) Source Subroutine
- 2) Cross Section Subroutine
- 3) Free Path Subroutine
- 4) Tracing Subroutine
- 5) Reaction Type Subroutine
- 6) Scattering Subroutine
- 7) Information Tally or Output Subroutine

The function of each sub-routine is represented, in outline form, in Table I. A brief description of each subroutine follows below. The function of each routine is:

- 1) Source: The source subroutine determines the space coordinates of the fission birth point for each neutron, and the reference number of the particular material region containing the fission point. In addition, it determines the energy, direction, and statistical weight of the neutron at fission.
- 2) Cross Section: Given a region number and an energy the cross section subroutine computes the total cross section and any other particular reaction cross sections asked for by another subroutine.
- 3) Free Path: The free path subroutine computes a total or scattering free path whenever it is given the appropriate cross sections.
- 4) Tracing: The tracing subroutine is the most complex of the seven listed here. Given a region number, free path, direction and initial position it computes the coordinates of the free path termination point, taking possible material region changes into account. It also computes the track length of the neutron in each region traversed.
- 5) Reaction Type: Given the particular reaction cross sections, from the cross section subroutine, for a given region and neutron energy, the reaction type subroutine selects an appropriate neutron-nucleus reaction to occur at the free path terminal point.
- 6) Scattering: The scattering subroutine selects the scattering angles and computes the exit energy of the neutron after scattering.
- 7) Information Tally: The information tally routine is peculiar to a specific problem type. It adds up neutron production in a criticality calculation, for example, or track length in a flux problem or number of neutrons slowing down below energy E in a slowing down calculation. It also computes changes in statistical weight.

TABLE I

GENERALIZED MONTE CARLO SUBROUTINE SCHEDULE1.0 Source Subroutine

- 1.1 Initial Position
  - 1.11 Initial Region Number
  - 1.12 Initial Space Coordinates
- 1.2 Initial Energy (or Lethargy)
- 1.3 Initial Orientation (direction cosines)
- 1.4 Initial Weight

2.0 Cross Section Subroutine

- 2.1 Total Cross Section
- 2.2 Particular Cross Sections
  - 2.21 Scattering Cross Section
    - 2.211 Elastic Scattering
    - 2.212 Inelastic Scattering
  - 2.22 Fission Cross Section
  - 2.23 Absorption Cross Section

3.0 Free Path Subroutine

- 3.1 Total Free Path
- 3.2 Scattering Free Path
- 3.3 Absorption Free Path

4.0 Tracing Subroutine

- 4.1 Boundary Intersection Point
  - 4.11 Revised Initial Space Coordinates
  - 4.12 Revised Region Number
  - 4.13 Revised Free Path

TABLE I (cont'd)

4.2 Free Path Termination Point

4.21 Termination Point Space Coordinates

4.22 Termination Point Region Number

4.3 Initial Region Track Length

4.31 Intermediate Region Track Lengths

4.32 Terminal Region Track Length

5.0 Reaction Type Subroutine

5.1 Scattering

5.11 Elastic Scattering

5.111 Light Nucleus Target

5.112 Heavy Nucleus Target

5.12 Inelastic Scattering

5.2 Absorption

5.21 Non-fission Absorption

5.22 Fission Absorption

5.3 Element Selector

6.0 Scattering Computation Subroutine

6.1 Exit Direction Cosines

6.11 CM Scattering Exit Direction Cosines

6.12 Local LAB Scattering Exit Direction Cosines

6.13 Fixed LAB Scattering Exit Direction Cosines

6.2 Energy Change

6.3 Exit Energy

7.0 Information Tally Subroutine

7.1 Flux Tally Per Region

TABLE I (cont'd)

- 7.2 Criticality Tally (Fission Production)
- 7.3 Power Tally
- 7.4 Fission Point Selection
- 7.5 Leakage Tally
- 7.6 Weight Computation

8.0 Long Memory Bank

- 8.1 Region Number - Material Composition Cross Reference
- 8.11 Region Number - Number of elements Cross Reference
- 8.12 Region Number - Element Numbers
- 8.13 Region Number - Number Densities of Elements
- 8.2 Element Number - Cross Sections Cross Reference
- 8.3 Region Number - Geometrical Dimension Cross Reference
- 8.4 Region Number - Nearest Neighbor Region Number Cross Reference

9.0 Short Memory Bank

- 9.1 Initial Space Coordinates (Source)
- 9.2 Initial Region Number (Source)
- 9.3 Initial Energy (Source)
- 9.4 Current Revised Initial Coordinates
- 9.5 Current Energy
- 9.6 Current Revised Region Number
- 9.7 Current Weight
- 9.8 Current Revised Free Path
- 9.9 Fixed LAB System Direction Cosines
- 9.10 Initial Scattering Position Coordinates
- 9.11 Terminal Scattering Position Coordinates
- 9.12 Terminal Reaction Position Coordinates

Figure 1: Example of a Geometry Appropriate to the Basic 3-D Monte Carlo Program

The short term memory remembers current information pertaining to the history of the particular neutron whose history is being traced on the computing machine. It contains, for example, the neutron energy, direction, weight, region number, and starting point. The long term memory contains the dimensions of all the material regions making up the reactor, the chemical composition of each material region and the reaction cross sections for each chemical element in the reactor.

The particular program described in this report does not contain an information tally subroutine. Only the first six of the seven subroutines listed above for a general Monte Carlo computation are included in what will be called the Basic 3-D Program. The time lapse from the start of coding to the end of revision and checkout for the first six subroutines of the Basic Program was about 17 months. By this time, the original program design was obsolete. Therefore, for economic reasons, work was frozen at the sixth subroutine. The Basic Program is, in essence, a history generator.

An information subroutine must be written and attached to the Basic Program for each particular type of application. Two sorting routines have been developed and will be reported in subsequent papers.

The Basic Program is obsolete for two reasons, one having to do with geometrical restrictions and the other with machine time.

1) Geometrical Restrictions: The particular geometry for which the tracing subroutine was designed is not general enough for many problems of interest to a reactor analysis group. Specifically, the program can not cope with a non-uniform fuel loading in all tubular fuel regions but is restricted to either.

- a) a reactor comprised of nested annular sleeves or
- b) a reactor in which only the central fuel tube is non-uniformly loaded.

2) Machine Time Restrictions: Some of the sampling methods are too slow by present standards. The fission lethargy sampling and the lethargy change calculation at scattering are done awkwardly. Subsequent experience has shown that the cross section computation can be done more quickly and a random number routine has recently been developed which is three times as fast as the one used in the Basic Program.

2.0 General Description of the Basic 3-D Monte Carlo Program

2.1 This program is the consequence of an assignment to determine a method for computing the heterogeneity correction in General Electric, ANPD multi-group reactor programs. The heterogeneity correction, per se, is an adjustment which must be made on the diffusion coefficient and migration length, computed on a simple density change basis, in order that a homogeneous reactor computation model can be made appropriate to a heterogeneous reactor.

2.2 This program uses a straightforward analogue method. An arbitrary neutron source distribution in space and lethargy can be used to begin a calculation. The two sources most easy to use are either the fission lethargy distribution or a fixed lethargy, coupled with an arbitrary spatial distribution. Scattering calculations are performed assuming isotropic scattering in the center-of-mass coordinate system. The exact lethargy change is computed for each individual scattering event; no type of "average" lethargy change is used. The microscopic cross-sections are computed by linear interpolation from the point values given in XDC 57-7-83. These cross sections were computed according to a method due to J. R. Terrall, Theoretical Physics Unit and W. B. Henderson, Reactor Analysis Unit.

2.3 The program nucleus is a group of three open-end subroutines: (1) a three-dimensional spatial tracing subroutine, (2) a macroscopic cross section subroutine and (3) a scattering (slowing down) subroutine. These three subroutines remain essentially unchanged from one particular problem to another. The lethargy distribution of the source is usually one of the two mentioned in (1.2).

The spatial distribution of the source is constructed by listing fission point locations on a tape. An auxiliary systematic sampling program can be used to generate these points. The positional output, interpreted by the program as the positions of generated source points, is written in cartesian coordinates on an output tape. Several output subroutine designs are available to sort the positional output. For example, the positions of neutrons may be desired in spherical coordinates or cylindrical coordinates rather than cartesian coordinates. Having selected a coordinate system an assortment of histogram routines can be used to view the data from several aspects.

2.4 The Basic Program has been combined with the necessary particular auxiliaries in two instances to give two other programs:

- 1) Slowing Down Program and Leakage Program
- 2) Diffusion Coefficient Program

2.5 The Basic Program simulates a reactor in the following way: A table of original fission (source) points is given, and an average of 2.58 neutrons are followed from birth to termination from each point in this table. One neutron is traced at a time. If fission takes place at a collision point, this point is noted on a new table, and the program proceeds to trace the next neutron from the old table. If the neutron is absorbed (non-fission capture), the region in which the capture occurs is noted, and a count of all the captures for each region is kept. The program then proceeds to trace the next neutron from the old table. Should scattering take place, the lethargy is modified, and a new direction of travel and a new jump length are selected. The neutron is followed until a non-scattering result occurs. The possibility that the neutron may escape is accounted for. The program keeps track of the number of escapes from various regions.

2.6 When all of the fission points are finished, the new table, containing the coordinates of each point at which fission occurred, is converted to the old table status, and is employed as the fission point source table for the succeeding (second) generation. The fission-capture points generated during the second generation are used as the source points for the third generation and so forth for an indefinite number of generations.

2.7 There are two ways in which the program may be stopped

- 1) One can stop the program when the original fission point table is exhausted.
- 2) The program can be stopped after an arbitrary number of generations has been completed.

3.0 Physical Description of the Reactor

3.1 The reactor geometry appropriate to the Basic Program is a nest of right circular annular regions. Fifty right circular cylindrical tubes may be positioned into the annular nest with their axes mutually parallel and parallel with the axis of the nest. See Fig. 1, p. 11. The annular nest is further divided into three longitudinal regions by the planes $z = c_1$ and $z = c_2$. The plane ends of the nest are $z = 0$ and $z = c_3$. A particular region is specified by a couple (m, n) . The interposed cylindrical tubes are denoted by $m = 1, 2, \dots, t$. $m = t$ denotes the last tube. The annular rings of nest are denoted by integers $m \geq t + 1$. The longitudinal region between the planes $z = 0$ and $z = c_1$ is specified by $n = 1$, that between $z = c_1$ and $z = c_2$ by $n = 2$, and that between $z = c_2$ and $z = c_3$ by $n = 3$. Each tube and annular region may therefore possess three subregions, each with a different homogeneous material composition. A particular subregion is uniquely specified by the numbers (m, n) .

If $m \leq t$, (m, n) refers to the subregion inside the m -th tube region and inside the n -th longitudinal region. If $m = t + 1$, (m, n) denotes the subregion within the innermost annular ring, and in the n -th longitudinal region, excluding any tube subregions. If $m > t + 1$, (m, n) refers to the subregion between the m -th and $(m - 1)$ th annular rings and in the n -th longitudinal region, excluding any interposed tube subregions. Tube numbers take precedence over annular region numbers. Any point within a particular tube will be associated with a tube number $m \leq t$ rather than the index of the annular region which, geometrically, also contains the point.

3.2 A limit of 50 regions (total number of tubes plus annular rings) was set because the running time would be prohibitive for a larger limit. Initial studies will thus employ only a few regions.

4.0 Permanent Input Tables

4.1 Three permanent input tables are employed. Table #1 contains the materials present in each volume defined by a particular couple $(m - n)$. (Refer to Section 3.0 for the definition of indicies m and n). Table #2 contains the cross section characteristics for every material used in the reactor. Table #3 contains the location of the axes of all tube circles and annular rings (in the $x - y$ plane), and the radius of each.

4.2 Table #1

4.21 Table #1 is composed of several records. Each record pertains to a unique $m-n$ volume. This data is read into core storage beginning at cell 5000. Each $m-n$ volume may contain a maximum of ten materials. A record contains numbers identifying the materials present in a particular $m-n$ volume and the corresponding atomic densities of these materials. The first word of each record contains the identifying $m-n$ region "packed" in one "word" in fixed point notation. (n is stored in the decrement field

and m in the address field). The second word contains, in fixed point, the number of materials in this volume. The remaining words go in pairs with the identifying material number first, followed by the corresponding atomic density. Every record is 22 words in length; however, not all words are employed. The "used" words may vary from four to twenty-two, depending upon the number of materials in the volume concerned. (The first two words plus ten possible pairs of words for the materials). The identifying material numbers are a code to denote specific materials; they do not correspond to mass or atomic number. Materials may be placed on each record in any arbitrary manner, and need not be arranged in a particular order. The format of each record for Table #1 is as follows:

Word No.

1	n and m packed in one cell $\left\{ \begin{array}{l} n \text{ in decrement} \\ m \text{ in address} \end{array} \right\}$ in fixed point
2	Number of materials in this volume in fixed point
3	μ_1 (First material in volume in fixed point)
4	η_1 (Atomic density of μ_1 in floating point)
5	μ_2 (Second material, [if there are two or more], in fixed point)
6	η_2 (Atomic density of μ_2 in floating point)
.	.
.	.
.	.
.	.
21	μ_{10} (Tenth material, if there are ten , in fixed point)
22	η_{10} (Atomic density of μ_{10} in floating point)

4.22 By forming these records in a definite order, a method can be employed whereby the desired record may be determined without reading and checking

every record. If α denotes the desired record, then α may be computed by the formula

$$\alpha = (n - 1) m_{\max} + m$$

4.23 Given n and m the information concerning the materials in the associated volume may be obtained from the proper record from Table #1. A check is included to insure that the table is formed correctly. (The given m and n may be compared with word number 1 of the record).

Therefore, Table #1 is formed in the following fashion:

<u>Record No.</u>	<u>n</u>	<u>Volume</u>
1	1	1
2	1	2
3	1	3
...	...	...
$m_{\max}$	1	last m
$m_{\max} + 1$	2	1
$m_{\max} + 2$	2	2
...	...	...
$2m_{\max}$	2	last m
$2m_{\max} + 1$	3	1
...	...	...
$3m_{\max}$	3	last m

4.3 Table #2

4.31 For every material used in the reactor, microscopic cross-section information is needed for various lethargy levels. This data is stored in Table #2 in core memory starting at cell 6700. Each record contains the above mentioned data for one material only. Therefore, the number of records in the table will equal the number of materials used in the reactor. The first word of each record is the identifying material number in fixed point notation. The second word is the mass number, $A(\mu)$, in floating point. The following words are divided into groups of four where the first of the four denotes the lethargy level, the second, the microscopic cross-section for scattering at the above lethargy level, the third, the microscopic cross section for absorption at this lethargy level, the fourth, the microscopic cross section for fission at this lethargy level. There are twenty lethargy levels noted on each record, thus each and every record contains 82 words. The format of each record for Table #2 is as follows:

Word No.

1	μ_1 (Identifying material number in fixed point)
2	$A(\mu_1)$ (Mass number of μ_1 in floating point)
3	U_1 (First lethargy level)
4	σ_{s1} (Microscopic cross section for scattering for U_1)
5	σ_{a1} (Microscopic cross section for absorption for U_1)
6	σ_{f1} (Microscopic cross section for fission for U_1)
7	U_2 (Second lethargy level)
8	σ_{s2} (Microscopic cross section for scattering for U_2)
9	σ_{a2} (Microscopic cross section for absorption for U_2)
10	σ_{f2} (Microscopic cross section for fission for U_2)
11	.

Word No.

12

.
.
.
.
..
.
.
.
.

79

 U_{20} (Twentieth lethargy level)

80

 σ_{s20} (Microscopic cross section for scattering for U_{20})

81

 σ_{a20} (Microscopic cross section for absorption for U_{20})

82

 σ_{f20} (Microscopic cross section for fission for U_{20})

4.32 The lethargy levels must be inserted in ascending order, U_1 being the lowest lethargy level and U_{20} being the highest lethargy level.

4.33 These records are also placed in a definite order in Table #2. The fissionable material must be noted with a material identification number 1. Other material identifying numbers will run from 2 through the last one (there must be no skips in numbering).. Thus, if the last record contains material number 14, there will be 14 records in Table #2. In this manner, the desired material data can be readily obtained without reading every record. By comparing the given material identification number with the first word of the selected records, one can check that the table was formed properly. Table #2 is formed as follows:

Record No.Material Identification No.

1

 $\mu_1 = 1$ (Fissionable material)

2

 $\mu_2 = 2$

3

 $\mu_3 = 3$.
.
..
.
.

last

 $\mu_{\text{last}} = \text{last}$

4.4 Table #3

4.41 Table #3 is read into core storage as input data beginning at cell 7800. This table contains information about the location and size of the tube circles and annular rings. There are groups of words in this table, each pertaining to a unique tube or ring. The first group concerns m_1 ; the second, m_2 ; the third, m_3 ; etc. Each group contains three words (h, k, a) where:

h_i denotes the x coordinate of the axis of the i^{th} tube circle or annular ring.

k_i denotes the y coordinate of the axis of the i^{th} tube circle or annular ring.

a_i denotes the radius of the i^{th} tube circle or annular ring in the x - y plane.

The format of input table #3 is as follows:

<u>Cell No.</u>	
7800	h_1
7801	k_1
7802	a_1
7803	h_2
7804	k_2
7805	a_2
•	•
•	•
$7800 + 3_i - 3$	h_i
$7800 + 3_i - 2$	k_i
$7800 + 3_i - 1$	a_i

4.42 Since the program is limited to 50 regions, the total number of tube circles plus annular rings must be less than, or equal to, 50.

5.0 Computing Tables

5.1 Another table employed in the 3-D Monte Carlo Problem is the fission point table. This table contains the following information:

- 1) Point number;
- 2) Position in reactor (x, y, and z, coordinates);
- 3) The m and n subscripts denoting in which volume of the reactor this point is located.

5.2 The table of initial fission points is given as part of the input data and is located on Tape #5. Each point is described on one record, thus the number of records equals the number of fission points. Each and every record contains five words in the following format:

Word No.

1	Point number on table in floating point.
2	x coordinate in floating point.
3	y coordinate in floating point.
4	z coordinate in floating point.
5	n-m (subscripts denoting unique volume - n in decrement, m in address in fixed point).

5.3 Each point (record) in the table is acted upon sequentially, and from each point, either two or three neutrons are emitted. If the outcome of any neutron emitted should be a fission capture, a "new" table is formed with exactly the same format as above. Initially, this new table is formed on Tape #6. The table on which the program is operating, is designated as the "old" table, and the one which the program is forming, is called the "new" table. Both tables are called computing tables. When the "old" table is

exhausted, the "new" table is changed to the old table status, and another "new" table of fission points is formed, erasing the original "old" table. Thus, two tapes (#5 and #6) are employed as computing tables. When the program is completed, "see section on "Method of Halting Program")", the "new" computing table, is printed out as part of the results.

6.0 Procedure for Making Tape

6.1 An auxiliary 704 program is employed to make on tape the Computing Table necessary for the 3-D Monte Carlo problem. This tape is made prior to the running of the problem. The computing table is composed of a number of fission positions where each position is described by a different record on the tape. This table is formed on Tape #5 when employed in the 3-D problem. Outlined below is the format of the data cards for each record.

Card No.

1	D2204, 35, (n-m),
2	,2200, Pt. No., xo, yo, zo,

Each record is one fission point and succeeding records have the same format as above. However, if the (n-m) volume is the same as the previous fission point, the type 1 data card may be omitted for this record. The detailed program for making this initial computing table is listed on page .

A summary of the procedure for making the tape is as follows:

- 1) Set tape to be formed on a Tape Unit #5.
- 2) Place data cards behind program deck, and using Package PK-1, read information into computer.
  - a) Transition card should read 03720₈ (Octal Number)

7.0 Machine Procedure

7.1 The table of original fission positions is formed on Tape #5 first. (See "Procedure for Making Tape" section). PK-1 is employed in this step, and DECK #1 is used.

Sense Switches

#1 down
#2 down
All others up

Tapes

#1 - PK-1
#3 - Output tape
#5 - BLANK for computing Table

LOAD: Tape

Crt. used: No.

Tapes to be saved: #5

Stops: (03724)₈ Data card error
(03730)₈ Tape error

Final Stop: (03730)₈

7.2 For running the 3-D program, Tape #5 formed above must be employed. The procedure is listed below.

Sense Switches

#1 down
#2 down
All others up

Tapes

#1 - PK-1
#3 - Output tape (for results)
#5 - Computing table (Tape formed above)
#6 - Blank (For alternating computing table)

LOAD: Tape

Crt. used: No.

Tapes to be saved: #5 and #6

Stops: (See list of stops) Post mortems will be taken automatically at all error points.

Final Stop: (00057)₈

Tapes to be listed: #3

8.0 Methods of Halting Program

8.1 Following the flow chart of Block #1, page 46, one can see the conditions necessary for the program to terminate. The generation multiplication factor k_n is compared with the previous k_n , (k_{n-1}), and a test is employed to see if the absolute value of the difference is less than a given arbitrary constant. If the difference is less, an "end counter" is stepped by one. If not, the end counter is set to zero. When this "end counter" reaches four, the program prints out the results and halts. In other words, when the ratio of the number of generated to initial fission points is within a given tolerance for four consecutive generations, the program is halted. This is the present method of terminating the program where this arbitrary constant ϵ_g is read as input data.

8.2 Alternate methods of stopping the program may be employed with minor modifications in the coding. The termination constant (4 for above) may be changed by inserting an octal correction card as follows:

Column	37	41	43	45	47	51		Modification #1

05543	SGA	xxxxx

where xxxxx is the octal address of any fixed point constant desired. By changing location (06164)_g (octal) as follows:

Columns	37	41	43	45	47	51		Modification #2

06164	HTJ	06166

the program will stop after tracing each neutron, and may be continued by pressing the start button. By controlling the arbitrary constant ϵ_g and employing the above modifications, one can trace any desired number of neutrons or any desired number of generations.

- 8.3 Example #1: Suppose one wishes to trace the path of three neutrons through the reactor. Employ Modification #2 and hit the start button when the computer stops. When computer stops again, hit start button once more, thus tracing the third neutron, and after the next stop, manually start at $(05632)_8$ to print final results.
- 8.4 Example #2: Suppose one wishes to trace through only one table (one generation). By choosing an ϵ_g larger than 2^{15} times the original number of fission points, and by applying modification #1 above, with the address of a fixed point one $(03736)_8$, one can guarantee that the conditions for halting the program will be satisfied after the initial computing table is completely traced.

9.0 List of Stops

<u>INDEX C</u>	<u>OCTAL ADDRESS</u>	<u>REASON AND PROCEDURE</u>
73037	04741	Data card error. Correct and rerun.
72275	05503	Generation factor (k_n) is zero; no fission points were generated.
72233	05545	Final stop (program is complete).
71614	06164	Modification #1 is being employed and one neutron has just been traced. Hit start button if more tracings are desired or manually transfer to 05632 to obtain print out of results.
71555	06223	Error return from square root subroutine (Block 2-1b. Get post mortem and analyze case)
71523	06255	{Error return from Exponential Subroutine} {Error return from Square Root Subroutine} {Error return from Exponential Subroutine} {Error return from Exponential Subroutine} {Block 2-7. Get post mortem & analyze case.}
71513	06265	
71510	06270	
71504	06274	
71467	06311	Error return from Ln Subroutine (Block 2-8; compute lethargy). Get post mortem and analyze case.
71453	06325	Error return from Ln Subroutine (Block 2-11; compute jump length). Get post mortem and analyze case.
71355	06423	Wrong record was selected from table 1. n-m region does <u>not</u> equal the n-m region selected. Get post mortem and check the format of table 1.
71304	06474	Error return from Exponential Subroutine (Block 2A-28; compute σ_a). Get post mortem and analyze case.
71211	06567	Error return from Sine Subroutine (Block 3-1) Get post mortem and analyze case.
71205	06573	Error return from Cosine Subroutine (Block 3-2) Get post mortem and analyze case.
71136	06642	Error return from Square Root Subroutine (Block 3-7; compute d). Get post mortem and analyze case.
71101	06677	Error return from Square Root Subroutine (Block 3-11; compute γ). Get post mortem and analyze case.
71070	06710	Error return from Square Root Subroutine (Block 3-12; compute R). Get post mortem and analyze case.

<u>INDEX C</u>	<u>OCTAL ADDRESS</u>	<u>REASON AND PROCEDURE</u>
70756	07022	Error return from Square Root Subroutine (Block 3-23; compute d). Get post mortem and analyze case.
70643	07135	Error return from square Root Subroutine (Block 3-38; compute γ_m). Get post mortem and analyze case.
70540	07240	Error return from Square Root Subroutine (Block 3-17; compute d). Get post mortem and analyze case.
70424	07354	Error return from Square Root Subroutine (Block 3-55; compute γ^*). Get post mortem and analyze case.
70353 70347	07425 07431	{Error return from Sine Subroutine } {Block 3A-1. Get Error return from Cosine Subroutine } {post mortem & analyze case. }
67717	10061	Error return from Exponential Subroutine (Block 4-21; compute σ_a). Get post mortem and analyze case.
67340	10440	Error return from Square Root Subroutine (Block 4B-14; compute $\mathcal{D}$). Get post mortem and analyze case.
67275	10503	Wrong record was selected from table 2. M_i does not equal M selected. Get post mortem and check the format of table 2.
67264	10514	Error return from Sine Subroutine (Block 5-10; rejection technique). Get post mortem and analyze.
67242 67226	10536 10552	{Error return from Cosine Subroutine } {Block 5-12. Get Error return from L_n Subroutine } {post mortem and analyze case. }
67207 67200 67173	10571 10600 10605	{Error return from Cosine Subroutine } {Block 5-14. Get Error return from Sq. Root Subroutine } {post mortem and Error return from Cosine Subroutine } {analyze case. }
67161	10617	Error return from Square Root Subroutine (Block 5-15; compute Sin θ lab). Get post mortem and analyze case.
67152	10626	Error return from Cosine Subroutine (Block 5-17; compute Cos ϕ lab). Get post mortem and analyze case.
67145	10633	Error return from Sine Subroutine Subroutine (Block 5-18; compute Sin ϕ lab). Get post mortem and analyze case.
67125	10653	Error return from Square Root Subroutine (Block 5-20; compute Sin θ_n). Get post mortem and analyze case.
66761	11017	Tape check error. Get post mortem and analyze case.

NOTE: Automatic post mortems are taken at all anticipated stops with the 2's complement of the stop address appearing in index register C.

PROGRAM PRINT OUT

After each generation key information is printed on the on-line printer.

KN	ENDCR	UIB
X.XX	X	XX

KN denotes the generation factor.

ENDCR denotes the end counter.

UIB denotes the number of new fission positions.

All other information will normally be printed on tape #3. (Forced on-line printing may be accomplished by having sense switch #3 down).

The following pages represent the print out of the program results. After every generation, the print out of the results, (excluding the input data), is obtained.

3D MONTE CARLO REACTOR SIMULATOR PROGRAM

TEST XXXX

DATE XXX.XX,XXXX

INPUT DATA

INITIAL RANDOM NUMBER X.XXXXXX,XX

INITIAL NUMBER OF FISSION POSITIONS XX

TOP OF BOTTOM CAP (C1) XX

TOP OF MIDDLE REGION (C2) XX

TOP OF REACTOR (C3) XX

LETHARGY CUTOFF LEVEL X.XXXXXXX,XX

LAST M TUBE NUMBER XX

TOTAL NUMBER OF M REGIONS XX

TABLE NO. 1

N	M	NO. OF MAT.	MAT. ID. NO.	LOADING
X	X	X	X	X.XXXXXXX,XX
			X	X.XXXXXXX,XX
			X	X.XXXXXXX,XX
			X	X.XXXXXXX,XX

N	M	NO. OF MAT.	MAT. ID. NO.	LOADING
X	X	X	X	X.XXXXXXX,XX
			X	X.XXXXXXX,XX
			X	X.XXXXXXX,XX
			X	X.XXXXXXX,XX
			X	X.XXXXXXX,XX
			X	X.XXXXXXX,XX

N	M	NO. OF MAT.	MAT. ID. NO.	LOADING
X	X	X	X	X.XXXXXXX,XX
			X	X.XXXXXXX,XX

TABLE NO. 3

M REGION	H(X)	K(Y)	A(RADIUS)
1	X.XXXX	X.XXXX	XX.XXXX
2	X.XXXX	X.XXXX	XX.XXXX
3	X.XXXX	X.XXXX	XX.XXXX
4	X.XXXX	X.XXXX	XX.XXXX
5	X.XXXX	X.XXXX	XX.XXXX
.	.	.	.
.	.	.	.
.	.	.	.

COMPUTING TABLE PRINTOUT

POS. NO.	X	Y	Z	N	M
1	X.XXXX	X.XXXX	X.XXXX	X	X
2	X.XXXX	X.XXXX	X.XXXX	X	X
3	X.XXXX	X.XXXX	X.XXXX	X	X
.	.	.	.	.	.
.	.	.	.	.	.
.	.	.	.	.	.

RESULTS

AVERAGE BIRTH-TO-CAPTURE DISTANCE X.XXXXXXX,XX

AVERAGE OF SQUARE OF BIRTH-TO-CAPTURE DISTANCE X.XXXXXXX,XX

ESCAPE INFORMATION

TOTAL ESCAPES OUT WALL OF BOTTOM CAP	XX
TOTAL ESCAPES OUT WALL OF MIDDLE REGION	XX
TOTAL ESCAPES OUT WALL OF TOP CAP	XX
TOTAL ESCAPES OUT TOP OF REACTOR	XX
TOTAL ESCAPES OUT BOTTOM OF REACTOR	XX

TUBE ESCAPES

M REGION	TOP FACE	BOTTOM FACE
1	XX	XX
2	XX	XX
..	.	.
.	.	.
.	.	.

CAPTURE INFORMATION

M REGION	TUBE CAPTURE	M-CIRCLE CAPTURE
1	XX	
2	XX	
3	.	
.	.	
.	.	
.	.	
X		XX
X		XX
X		XX
.		.
.		.

COMPUTING TABLE PRINTOUT

POS NO.	X	Y	Z	N	M
1	X.XXXX	X.XXXX	X.XXXX	X	X
2	X.XXXX	X.XXXX	X.XXXX	X	X
3	X.XXXX	X.XXXX	X.XXXX	X	X
.	.	.	.	.	.
.	.	.	.	.	.
.	.	.	.	.	.

10.0 Deck Make-Up

The 3-D Monte Carlo Reactor Simulator Program employs the General Electric Package PK-1. One input table must be on tape, and this is formed prior to running the 3-D program. The make-up of both decks are listed below.

Deck #1 (Make Computing Table on Tape #5)

Program Deck to make Computing Table on Tape

Package Select Card to select PK-1

CHIC Checkout Identification Card

Transition card (03720)₈

Data (Information concerning original fission positions)

End of file card

Deck #2

Post Mortem Control Card

Program Deck (3-D Monte Carlo Reactor Simulator program)

CHIC Checkout Identification Card

Transition card (04737)₈

Data (Input information to the program)

11.0 Data Sheets

See following pages.

3D MONTE CARLO REACTOR SIMULATOR

DATA SHEET

ORIGINAL FISSION TABLE DATA

D2204,	35	(n-m)			
.2200,	Pt.No	x_o	y_o	z_o	
D2204,	35	(n-m)			
.2200,	Pt.No	x_o	y_o	z_o	
D2204,	35	(n-m)			
.2200,	Pt.No	x_o	y_o	z_o	
D2204,	35	(n-m)			
.2200,	Pt.No	x_o	y_o	z_o	
D2204,	35	(n-m)			
.2200,	Pt.No	x_o	y_o	z_o	
D2204,	35	(n-m)			
.2200,	Pt.No	x_o	y_o	z_o	
D2204,	35	(n-m)			
.2200,	Pt.No	x_o	y_o	z_o	

Pt. No. denotes point number and should run sequentially (1,2,3, ...).

x_o denotes initial x coordinate of neutron in cm.

y_o denotes initial y coordinate of neutron in cm.

z_o denotes initial z coordinate of neutron in cm.

(n-m) denotes in which unique volume of the reactor this point is located.

For $n = 1$, $(n-m) = 262144 + m$

For $n = 2$, $(n-m) = 524288 + m$

For $n = 3$, $(n-m) = 786432 + m$

3D MONTE CARLO REACTOR SIMULATOR

 DATA SHEET
 TABLE #1 - REACTOR COMPOSITION

D	xxxx	,	35	,	(n-m)	,	NOMATV	,	μ_1	,
C	xxxx+3	,	η_1	,						
D	xxxx+4	,	35	,	μ_2	,				
C	xxxx+5	,	η_2	,						
D	xxxx+6	,	35	,	μ_3	,				
C	xxxx+7	,	η_3	,						
D	xxxx+8	,	35	,	μ_4	,				
C	xxxx+9	,	η_4	,						
D	xxxx+10	,	35	,	μ_5	,				
C	xxxx+11	,	η_5	,						
D	xxxx+12	,	35	,	μ_6	,				
C	xxxx+13	,	η_6	,						

For n = 1, (n-m) = 262144 + m
 For n = 2, (n-m) = 524288 + m
 For n = 3, (n-m) = 786432 + m

D	xxxx+14	,	35	,	μ_7	,
C	xxxx+15	,	η_7	,		
D	xxxx+16	,	35	,	μ_8	,
C	xxxx+17	,	η_8	,		
D	xxxx+18	,	35	,	μ_9	,
C	xxxx+19	,	η_9	,		
D	xxxx+20	,	35	,	μ_{10}	,
C	xxxx+21	,	η_{10}	,		

(n-m) denotes unique volume of the reactor.

NOMATV denotes number of materials in this volume (10 max).

μ_i denotes identifying material number.

η_i denotes corresponding atomic density in units of 10^{24} per cm^3 .

xxxx = 5000 + (Record No. - 1) 22

Note: Each sheet pertains to a unique n-m volume and represents one record of information. There must be no records skipped in forming this table.

3D MONTE CARLO REACTOR SIMULATOR

DATA SHEET

TABLE #3 - REACTOR GEOMETRY INFORMATION

C	7800	,	h_1	,	k_1	,	a_1	,	h_2	,	k_2	,	a_2	,	h_3	,
	k_3	,	a_3	,	h_4	,	k_4	,	a_4	,	h_5	,	k_5	,	a_5	,
	h_6	,	k_6	,	a_6	,	...	,	...	,		,		,		,
		,		,		,		,		,		,		,		,
		,		,		,		,		,		,		,		,
		,		,		,		,		,		,		,		,
		,		,		,		,		,		,		,		,
		,	h_i	,	k_i	,	a_i	,		,		,		,		,
		,		,		,		,		,		,		,		,
		,		,		,		,		,		,		,		,
		,		,		,		,		,		,		,		,
		,		,		,		,		,		,		,		,

h_i denotes the x coordinate of the axis of the i^{th} tube circle or annular ring.

k_i denotes the y coordinate of the axis of the i^{th} tube circle or annular ring.

a_i denotes the radius of the i^{th} tube circle or annular ring in the x-y plane.

UNCLASSIFIED

UNCLASSIFIED

3D MONTE CARLO REACTOR SIMULATOR

DATA SHEET

INPUT CONSTANTS

D	2000	,	35	,	m_t	,	m_{max}	,	NOMAT	,

m_t denotes last cylindrical tube number.

m_{max} denotes last (outtermost) annular ring number.

NOMAT denotes number of materials in the reactor.

C	2003	,	NOFP	,	ϵ_g	,	δ	,	C_1	,	C_2	,
	C_3	,	1st R.N.	,	U_T	,						

NOFP denotes number of fission points on original computing table.

ϵ_g denotes minimum deviation in k_n between generations before halting program.

δ denotes multiplication factor to increase generation.

C_1 denotes dividing plane between bottom cap and center region of the reactor in cm.

C_2 denotes dividing plane between center region and top cap of the reactor in cm.

C_3 denotes top plane of the reactor in cm.

1st R.N. denotes number employed to start the random number subroutine.

U_T denotes lethargy cut off level.

B	2323	,	2	,	TEST ID. NO.

The test identification number may be a maximum of 12 characters (numbers or letters).

2326	,	2	,	DATE

This card is to be last data card; it contains the date, described by a maximum of 12 characters.

12.0 Flow Charts

See following pages.

BASIC FLOW DIAGRAM

GENERATION PROCESS

BLOCK 1

Equation 1

$$K_1 = 1.361 e^{(-10R_3)} \sinh \sqrt{20R_3}$$

Equation 2

$$I = -\left(\frac{1}{\sum T_i} \log_e [1-R_5]\right)$$

ANGLES, JUMP LENGTH AND LETHARGY

BLOCK 2

TRACE PATH OF NEUTRON
BLOCK 3

TRACE PATH OF NEURON
(CONTINUED) BLOCK 3

TRACE PATH OF NEUTRON
(CONTINUED) BLOCK 3

14
69
0

2
07
14

ESCAPE ROUTINE

BLOCK 3a

UNCLASSIFIED

ESCAPE ROUTINE

BLOCK 3a (Continued)

S.C.P. ROUTINE
SCATTER-CAPTURE-FISSION

SCATTER

DETERMINE OUTCOME OF NEUTRON
BLOCK 4

CAPTURE

ΔU SUBROUTINE

DETERMINE OUTCOME OF NEUTRON

BLOCK 4a

CORRECT JUMP LENGTH ROUTINE A

CORRECT JUMP LENGTH ROUTINE B

CORRECT JUMP LENGTH ROUTINE C

DETERMINE OUTCOME OF NEUTRON
BLOCK 4a (Continued)

DETERMINE OUTCOME OF NEUTRON

BLOCK 4b

BIRTH TO CAPTURE ROUTINE

DETERMINE OUTCOME OF NEUTRON

BLOCK 4b (Continued)

EQUATIONS

$$1. \Delta U = -\ell_n \left[\frac{1}{2} \left(\{1+F(\mu)\} + \{1-F(\mu)\} \cos \pi R_{10} \right) \right]$$

$$2. \cos \theta_{LAB} = \left[\frac{A(\mu) \cos R_{10}\pi + 1}{(A(\mu)^2 + 2 A(\mu) \cos R_{10}\pi + 1)^{1/2}} \right]$$

$$3. \sin \theta_{LAB} = \sqrt{1 - \cos^2 \theta_{LAB}}$$

$$4. \cos \phi_{LAB} = \cos 2\pi R_{12}$$

$$5. \sin \phi_{LAB} = \sin 2\pi R_{12}$$

$$6. \cos \theta_m = \cos \theta \cos \theta_{LAB} - \sin \theta \sin \theta_{LAB} \sin \phi_{LAB}$$

$$7. \sin \theta_m = \sqrt{1 - \cos^2 \theta_m}$$

$$8. \cos \phi_m = \frac{\sin \phi \sin \theta_{LAB} \cos \phi_{LAB} + \cos \theta \cos \phi \sin \theta_{LAB} \sin \phi_{LAB} + \sin \theta \cos \phi \cos \theta_{LAB}}{\sin \theta_m}$$

$$9. \sin \phi_m = \frac{-\cos \phi \sin \theta_{LAB} \cos \phi_{LAB} + \cos \theta \sin \phi \sin \theta_{LAB} \sin \phi_{LAB} + \sin \theta \sin \phi \cos \theta_{LAB}}{\sin \theta_m}$$

SCATTER ANGLES AND LETHARGY CHANGE

BLOCK 5

120-058

* MAKE COMPUTING TABLE ON TAPE

BEGIN REW 5	REWIND TAPE 5
READ1 SXJ DIP,4	READ RECORD
ZER	READ UNTIL END OF RECORD
UNJ WRITE	END OF FILE RETURN
HTJ	ERROR RETURN
SXJ TRW,4	WRITE BINARY RECORD
ZER 2200,1	START AT LOC 2200
ZER 5,0,5	USE TAPE 5, 5 WORDS
HTJ	ERROR RETURN
UNJ READ1	READ NEXT RECORD
WRITE WEF 5	WRITE END OF FILE
HTJ	STOP
DIP EQU 33	(DATA LOAD PROGRAM)

* TAPE READ WRITE SUBROUTINE
 * CALLING SEQUENCE
 *   SXJ TRW,4
 *   ZER INITIAL ADDRESS,B,X
 *   ZER TAPE NUMBER,C,N
 *   ERROR RETURN
 *   NORMAL RETURN FOR WRITE -- END OF RECORD RETURN FOR READ
 *   END OF FILE RETURN FOR READ -- NOT USED FOR WRITE, MAY BE OMITTED
 * STORAGE REQUIRED -- 74 WORDS PLUS 4 WORDS OF FREE STORAGE
 * B-0 FOR READ, 1 FOR WRITE
 * C-0 ON WRITE
 * C-0 ON READ IF END OF RECORD SKIP IS TO BE USED
 * C-1 ON READ IF EITHER THE NUMBER OF WORDS OR THE FINAL ADDRESS ARE USE
 * X IS THE FINAL ADDRESS IF SPECIFIED
 * N IS THE NUMBER OF WORDS IF SPECIFIED AND MUST BE 0 IF NOT SPECIFIED
 * THE LAST WORD OF THE UNIT RECORD IS THE COMPLEMENT OF THE CHECK SUM

TRW	SRT	READ-WRITE SUBROUTINE