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Flexible Monte Carlo Programs

FMC-N and FMC-G

J. J. Loechler

J. E. Mac Donald

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GENERAL ELECTRIC

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FLEXIBLE MONTE CARLO PROGRAMS FMC-N AND FMC-G

J. J. Loechler
J. E. MacDonald

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ABSTRACT

Flexible Monte Carlo programs FMC-N and FMC-G (GE-ANPD Programs 516 and 515) are digital computer programs which apply Monte Carlo methods to simulate neutron and gamma ray life histories, respectively, in a source-shield configuration. As the names imply, the programs are designed for flexibility in the geometrical, material, nuclear, and source descriptions of source-shield configurations and variance reduction techniques.

The programs are designed to optimize the use of fast memory and to provide complete freedom in the dimensions of the various input quantities.

The programs are coded for an IBM-704 computer with a fast memory capacity of 32,768 storage locations and eight magnetic tape units, and for an IBM-7090 computer with a fast memory capacity of 32,768 storage locations and ten magnetic tape units on two data channels. No magnetic drum storage is necessary for either computer.

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I. INTRODUCTION

Flexible Monte Carlo programs FMC-N and FMC-G (GE-ANPD Programs 516 and 515) are digital computer programs which apply Monte Carlo methods to simulate neutron and gamma ray life histories, respectively, in a source-shield configuration. As the names imply, the programs are designed for flexibility in the geometrical, material, nuclear, and source descriptions of source-shield configurations and variance reduction techniques.

The programs are coded for an IBM-704 computer with a fast memory capacity of 32,768 storage locations and eight magnetic tape units and for an IBM-7090 computer with a fast memory capacity of 32,768 storage locations and ten magnetic tape units on two data channels. No magnetic drum storage is necessary.

Flow charts for the programs were obtained from a report by Forrest Mozer and Edward Leshan¹. Although their flow charts were followed directly for most of the coding, several changes were made to make the codes more suitable for specialized reactor and shield analysis. The more important changes were:

1. Inclusion of a modified version of the GMC source generator².
2. Elimination of the particle similarity indicator, τ .
3. Elimination of the option for using the rejection technique for sampling from the evaporation model.
4. Consistent use of energy rather than velocity.
5. Introduction of energy and weight cutoffs and a number-of-collisions cutoff following energy cutoff to replace census time.
6. Insertion of a fixed nuclear mass of 20 in FMC-N to distinguish between light and heavy nuclei. Nuclei with a mass equal to and greater than 20 are considered heavy.
7. Introduction of reflecting boundaries to simplify some geometry descriptions.
8. Modification of the history tally to count on a collision basis rather than on a region basis.
9. Modification of the absorption tally to be tallied by energy and region rather than energy alone.
10. Inclusion of the option of obtaining the energy deposition in conjunction with the absorption tally.

11. Modification of the neutron fission routine to save the fission neutron's parameters with the option of processing these neutrons by generations or saving the fission neutron's parameters on magnetic tape for later analysis.
12. Inclusion of the option of including or excluding the expectation calculations.
13. Inclusion of the option of Monte Carlo or expectation flux tallies.

Any number of regions, of a variety of geometrical shapes, are enclosed by any number of surfaces described by the general equation:

$$AX^2 + X_0X + BY^2 + Y_0Y + CZ^2 + Z_0Z - K = 0 \quad (1.1)$$

Each region is composed of a homogeneous mixture of any number of materials.

The direction, spatial, and energy parameters of source neutrons and gamma rays may be selected by a source particle generator, or they may be entered as input from cards or magnetic tape.

The neutron collision routine uses random sampling from the appropriate discrete distributions to select either an elastic-scattering event, an inelastic-scattering event, or a fission event.

Thus, the subset of the set of all possible neutron events consisting of these three events is assumed to occur with probability one. A neutron emerging from a collision is given a weight equal to the probability that it escaped absorption at that collision multiplied by the probability that it escaped absorption at all prior collisions. This is done in order to eliminate the bias introduced by the event sampling scheme of not allowing absorption to occur. Further adjustment of the weight is necessary when one or more of the optional variance reduction techniques are used. The angular distribution of scattered neutrons may be isotropic or anisotropic in the center-of-mass system at the discretion of the user.

For a fission event, the weight of the incident neutron is considered in determining the actual number of neutrons to be produced, thereby producing neutrons of weight one. The angular distribution of the fission-produced neutrons is assumed isotropic in the laboratory system. Parameters of these neutrons are stored on magnetic tape for later analysis. The weight of the incident neutron is tallied in the absorption tally, and the history is terminated.

The emission of gamma rays due to neutron absorption, inelastic scattering, and fission is optionally allowed. The weight of the incident neutron is considered in determining the actual number of gamma rays to be emitted, thereby emitting gamma rays of weight one. The angular distribution of the emitted gamma rays is assumed to be isotropic in the laboratory system. Parameters of these gamma rays are stored on magnetic tape for later analysis.

The gamma ray collision routine treats all collisions as Compton-scattering collisions, i.e., Compton scattering is assumed to occur with probability one. A gamma ray emerging from a collision is given a weight equal to the probability that it escaped absorption at that collision multiplied by the probability that it escaped absorption at all prior collisions. This is done in order to eliminate the bias introduced by assuming all collisions to be Compton scattering collisions. The absorption probabilities equal the sums of the photoelectric and pair-production probabilities.

Cumulative probability tables are used to determine the angular distribution of the Compton-scattered gamma rays. The gamma ray collision routine optionally allows the emission of 500 kev gamma rays following absorption by pair-production of the primary gamma ray. These gamma rays are then processed in their turn by the code.

Several variance reduction techniques are optionally allowed in addition to the non-optional statistical estimation technique of weighting for absorption escape at each collision. An optional statistical estimation technique may be used for scoring entrance tallies and mandatory leakage tallies. Flexibility in sampling from source spectra is achieved by using energy group-averaged constants called acceleration factors. The other options, which are forms of importance sampling, are splitting and Russian roulette depending on energy, region, and location within a region, Russian roulette on particles whose weight falls below the weight cutoff, and the exponential transformation.

Output available from these programs includes independently optional absorption or energy deposition tallies, Monte Carlo entrance and leakage tallies, expectation entrance and leakage tallies, Monte Carlo or expectation flux tallies, and history tallies of particles reaching selected regions. Monte Carlo and optional expectation leakage tallies are made for all regions external to the source-shield configuration. The absorption or energy deposition, entrance, leakage, and flux tallies are made by region and energy group. These programs also optionally print the source particle's direction cosines, spatial coordinates, and energy. The printing of the history tallies is also optional.

2. GENERAL PROGRAM DESCRIPTION

An attempt is made here to give a brief word description of the operation of programs FMC-N and FMC-G in their entirety. It is hoped that this description will serve to unify the details of these programs as given in the following sections. For the detailed workings of the programs, no word description would suffice and the reader is referred to the flow charts. This description should be helpful in following the flow of the program as given in the flow charts.

Source particles and secondary particles originate in these programs as whole neutrons or gamma rays. Statistically speaking, they are said to originate with a weight of one. Due to the various variance reduction techniques, they acquire, in general, weights different from one as their history progresses. That is, they may then represent fewer or more than a single neutron or gamma ray if their weight is less than or greater than one. In order to keep this distinction clear, weighted neutrons and gamma rays are referred to simply as particles. This definition includes the special case of weight one. The number of neutrons or gamma rays represented by a particle is then given by the particle's weight.

It is often convenient to start more than one source particle with the same source coordinates. The FMC programs treat these particles as a single statistical unit until a collision separates a particle from the unit and gives this particle a separate existence. Such a statistical unit is referred to as a cluster. Since the individual particles in a cluster have identical histories, they all have the same weight. Hence, it is meaningful to speak of the weight of a cluster. The number of neutrons or gamma rays represented by a cluster, at any point in the cluster history, is equal to the product of the number of particles in the cluster and the weight of the cluster. For convenience, the unit in the special case in which the number of particles equals one will, at times, be referred to as a cluster.

The following is a brief description of the manner in which the programs originate and calculate a particle history. Again the reader is referred to the flow charts for the actual details.

The preparation routine, for which only the source particle generator flow charts are included, is mainly an input data loading and data processing routine. Its main purpose is to determine the storage addresses of the input

and output data. Some effort has been made to optimize the storage assignments for which auxiliary tape storage is available.

The source particle generator is available for use in the preparation routine if particle parameters are not available on cards or tape. If the source particle's parameters are to be generated, they are selected by the programs according to the source generator options specified in input. The source particle's direction cosines, spatial coordinates, and energy parameters are chosen from the appropriate cumulative probability distribution tables.

If the memory capacity of the computer will not contain all the input and output for which no auxiliary tape storage is available the program stops and writes the comment "PROGRAM TOO LARGE". The user will then have to find some method of shortening these storage assignments. If the memory capacity of the computer will contain all input and output, but not the source particle's parameters, the source particle's parameters are separated into records of the maximum size, greater than one hundred particles, that will fit into memory. The excess records are then stored on magnetic tape.

As the input data are loaded into memory, the programs check the storage addresses of the blocks of data to insure that the number of entries is in agreement with the "DIMENSION AND CONTROL CONSTANTS". A check is also made to insure that the proper number of tables are present. However, interchanging of the cumulative probability distribution tables cannot be detected by the programs. Hence, the responsibility of entering these tables in their proper order is left to the user. If any inconsistencies are detected in the input data, a comment to the inconsistency and the input in error is written, and the program stops. Upon completion of the proper loading of the input data, the main program is loaded into memory, erasing the preparation routine, and control is transferred to the main program.

The main program then selects the parameters for a single source particle from memory. If none are available in memory, a record of parameters of source particles is loaded into memory from magnetic tape. If there are no records on tape, the program proceeds to the output routine. In the neutron program, the output routine prepares for the processing of the fission neutrons by generations, if they are to be processed at this time.

When the source particle's parameters have been obtained, the energy group and the region containing the particle are determined. Energy dependent quantities, acceleration factors, are then used to perform importance or adjusted

sampling from the source spectrum. The cluster is formed; and the particle's parameters are printed if desired.

The distance to the first boundary crossed in the particle's flight path is then computed as determined by the direction cosines and spatial coordinates of the particle cluster. The appropriate mean free path for the composition of the region and the energy of the particle cluster is determined. For void regions, the mean free path is taken to be infinite; the distance to the boundary is tallied as the flux, if desired; and the program proceeds to the boundary crossing. For non-void regions, the exponential attenuation to the boundary is computed for use in the various expectation tallies and the flux tally. The flux tally is then made if desired.

If the exponential transformation option is requested for the region, an adjusted mean free path, as specified in input, replaces the true mean free path in determining distances to collisions. A distance to collision is then found for each particle in the cluster. If the particle crosses the boundary before reaching its collision point, the number of particles in the cluster is depleted by one; and the number of particles entering the next region is augmented by one. If the particle has a collision in the region, the number of particles in the cluster is depleted by one, and the particle's coordinates at the collision point are stored in born storage. If the exponential transformation option was requested for the region, the appropriate weight adjustment is made before storing the weight parameter. If the "fine splitting" option was requested for the region, a component of the distance to the collision point, as specified in input, is computed and stored for later use in the collision routine.

The above process is repeated for each particle in the cluster, that is until the number of particles in the cluster is reduced to zero. If one or more particles entered the next region, a new cluster is formed for this region. If the exponential transformation was requested for the last region, an appropriate weight adjustment is made for the cluster weight. The particle's spatial coordinates at the boundary are determined. If the boundary is a reflective boundary, the particle's direction cosines are reflected.

The new region containing the particle cluster is determined. If the new region is a history tally region, the pertinent information is recorded, provided the cluster represents an actual particle. No history tallies are taken on the expectation cluster. An entrance tally is taken if desired. The

Monte Carlo tally consists of the summation of the number of neutrons or gamma rays represented by the particle cluster for the appropriate region and energy bin. The expectation tally consists of the summation of the probabilities of the neutron or gamma rays represented by the particle cluster continuing in the direction chosen and reaching the region of interest without further collisions.

If the new region is an outside region, a leakage tally is taken; and the history of the particle cluster is terminated. If the new region is not an outside region, splitting on location at region boundary crossings for the particle cluster entering the new region is done in accordance with the region importances of the two regions involved. An appropriate adjustment of the cluster weight is made in accordance with the splitting scheme. If the cluster weight is then less than the cutoff weight and the number of neutrons or gamma rays represented by the cluster is also less than the cutoff weight, a game of Russian roulette is played to decide whether or not the particle cluster is to survive. If the particle cluster survives, the weight is increased to the cutoff weight. If the cluster weight is less than the cutoff weight, but the number of neutrons or gamma rays represented by the particle cluster is greater than the cutoff weight, appropriate adjustments are made to increase the importance of each particle in the cluster so that each has a weight approximately equal to the cutoff weight. The number of particles in the cluster is then adjusted to account for the weight change.

Once the collision points for all the particles in the cluster have been found, the program follows the rest of the cluster history only in the sense of expectation quantities if the expectation tallies are desired. The expectation cluster consists of the product of the particle cluster, at the previous collision point, and the probability of this cluster continuing in the direction chosen without further collision. The expectation tallies are taken wherever they are desired along this path. When the expectation cluster falls below the cutoff weight, a game of Russian roulette is played to decide whether or not the expectation cluster is to survive. The expectation cluster history is also terminated when the cluster reaches an outside region.

When one of the terminal conditions is achieved and the cluster history terminated, the program proceeds to born storage to obtain the parameters of the various particles which have had collisions along the flight path of the cluster. When the list of particles in born storage has been exhausted, the

program proceeds to obtain the parameters of a source particle and to initiate a new history.

The manner in which FMC-N treats the collided particles is as follows. After selecting a particle from born storage, the program proceeds to calculate the macroscopic cross sections, based on the composition of the region and the incident energy of the particle, for each event. If secondary gamma rays are to be generated, the program then determines whether an absorption collision has occurred. If an absorption collision occurred, the incident neutron's weight is considered in determining the number of whole gamma rays to be generated. Direction cosines and energies are chosen for each of the gamma rays born in the neutron absorption process. The direction cosines are computed from an isotropic distribution in the laboratory system. The energies are chosen from the appropriate cumulative probability distribution table. The necessary particle information for each gamma ray is stored in the secondary gamma ray storage and later transferred to magnetic tape storage.

The probability of absorption is computed, multiplied by the proper conversion factor, and tallied for the proper region and energy absorption bin. The probability of escaping absorption is computed and compared with the small input quantity, epsilon. If the probability of escaping absorption is less than epsilon, the particle is assumed to be totally absorbed and the particle history is terminated. The program then returns to born storage for another particle. If the probability of escaping absorption is greater than epsilon, the weight of the particle is reduced by the probability of absorption.

Sampling is then done to determine the nucleus involved in the collision and whether the collision is a scattering event or a fission event. (Note that this sampling procedure is entirely independent of the sampling done in starting gamma rays born in neutron absorption.) If a scattering event is chosen, input decisions inform the program if the cosine of the polar scattering angle in the center-of-mass system is to be chosen from the appropriate cumulative probability distribution table, computed from a cosine power series, or computed by an isotropic scattering routine. The program then proceeds to the appropriate routine for the event chosen.

If a fission event were chosen, the program generates the gamma rays born in a fission event if desired. The incident neutron's weight is considered in determining the number of whole gamma rays to generate. Direction cosines and energies are chosen for each gamma ray born in the neutron fission process.

The direction cosines are computed from an isotropic distribution in the laboratory system. The energies are chosen from the appropriate cumulative probability distribution table. The necessary particle information for each gamma ray is stored in the secondary gamma ray storage and later transferred to magnetic tape storage. If the absorption tally is desired, the neutron's weight is tallied for the proper region and energy absorption bin. If the energy deposition tally is desired, the product of the neutron's weight and the fission neutron heating constant is tallied for the proper region and energy absorption bin. The incident neutron's weight is considered in determining the number of whole neutrons to generate. Direction cosines and energies are chosen for each neutron born in the neutron fission process. The direction cosines are computed from an isotropic distribution in the laboratory system. The energies are chosen from the appropriate cumulative probability distribution table. The necessary particle information for each neutron is stored in the fission neutron storage for later processing or transfer to magnetic tape storage. The program then returns to born storage for another particle.

If an inelastic scattering event on a light nucleus is chosen, the cosine of the polar scattering angle and the particle's energy in the laboratory system are computed. If the energy deposition is desired, the product of the neutron's weight and the energy loss is tallied for the proper region and energy absorption bin.

If an elastic scattering event on a heavy nucleus is chosen, the cosine of the polar scattering angle in the laboratory system is set equal to the cosine of the polar scattering angle in the center-of-mass system. The energy after scattering is taken to be equal to the energy before scattering.

If an inelastic scattering event on a light nucleus is chosen, the program generates the gamma rays born in an inelastic scattering event if desired. The incident neutron's weight is taken into consideration in determining the number of whole gamma rays to generate. Direction cosines and energies are chosen for each gamma ray born in the neutron inelastic process. The direction cosines are computed from an isotropic distribution in the laboratory system. The energies are chosen from the appropriate cumulative probability distribution table. The necessary particle information for each gamma ray is stored in the secondary gamma ray storage and later transferred to magnetic tape storage. The program then selects an excitation level energy for the event

from the appropriate cumulative probability distribution table and computes the energy of the scattered particle and the cosine of the polar scattering angle in the laboratory system. If the energy deposition is desired, the product of the neutron's weight and the energy loss is tallied for the proper region and energy absorption bin.

If an inelastic scattering event on a heavy nucleus is chosen, the program generates the gamma rays born in an inelastic scattering event, if desired. The incident neutron's weight is considered in determining the number of whole gamma rays to generate. Direction cosines and energies are chosen for each gamma ray born in the neutron inelastic process. The direction cosines are computed from an isotropic distribution in the laboratory system. The energies are chosen from the appropriate cumulative probability distribution table. The necessary particle information for each gamma ray is stored in the secondary gamma ray storage and later transferred to magnetic tape storage. The cosine of the polar scattering angle in the laboratory system is set equal to the cosine of the polar scattering angle in the center-of-mass system. The scattered energy is picked from the appropriate cumulative probability distribution table. If the energy deposition is desired, the product of the neutron's weight and the energy loss is tallied for the proper region and energy absorption bin.

The azimuthal scattering angle in the laboratory system is then chosen from a uniform distribution, and the new direction cosines in the laboratory system are computed. The direction cosines are then transformed to the fixed reference system.

Fine splitting, if desired, and splitting on energy, in accordance with the energy importances, are done and an appropriate particle cluster is determined. Suitable weight adjustments for the particle cluster are made in accordance with the splitting options. The collision routine then returns to the main program to compute the distance to the next boundary crossing and to follow the particle cluster as previously described.

The manner in which FMC-G treats the collided particles is as follows. After selecting a particle from born storage, the program proceeds to calculate the macroscopic cross sections, based on the composition of the region and the incident energy of the particle for each event. If 500 kev annihilation gamma rays are to be considered, the program determines if a pair production event has occurred. If a pair production event occurred, direction cosines for this radiation are computed from an isotropic distribution in the laboratory system.

Fine splitting, if desired, and splitting on energy, in accordance with the energy importances, are done and the new weights are computed. These parameters are then stored in post-collision born storage to be processed in their turn.

The probability of absorption is computed, multiplied by the proper conversion factor, and tallied for the proper region and energy absorption bin. The probability of escaping absorption is computed and compared with the small input quantity, epsilon. If the probability of escaping absorption is less than epsilon, the particle is assumed to be totally absorbed and the particle history is terminated. The program then returns to born storage for another particle. If the probability of escaping absorption is greater than epsilon, the weight of the particle is reduced by the probability of absorption.

The cosine of the polar scattering angle in the laboratory system is chosen from the appropriate cumulative probability distribution table, and the energy of the scattered gamma ray is computed from the Compton relationship. If the energy deposition is desired, the product of the gamma ray's weight and the energy loss is tallied for the proper region and energy absorption bin.

The azimuthal scattering angle in the laboratory system is then chosen from a uniform distribution, and the new direction cosines in the laboratory system are computed. The direction cosines are then transformed to the fixed reference system.

Fine splitting, if desired, and splitting on energy, in accordance with the energy importances, are done and an appropriate particle cluster is determined. Suitable weight adjustments for the particle cluster are made in accordance with the splitting options. The collision routine then returns to the main program to compute the distance to the next boundary crossing and to follow the particle cluster as previously described.

3. NOTATION

3.1 Subscripts

a	Absorption
b	Boundary number
c	Collision type
i	Energy group
l	Minimum value of group
m	Material number
n	Cumulative probability distribution table interval number
r	Region number
u	Maximum value of group

3.2 Superscripts

'	Prime indicates value of quantity after some event
cm	Center-of-mass system
h	Composition number
L	Laboratory system

3.3 Symbols

ξ	Pseudo random number uniformly distributed on the unit interval
$\{ \}$	Symbol signifying to use the greatest integer contained in the brackets and to drop any fractional part
$\{P_n\}$	Cumulative probability distribution sequence on n
$\{X_n\}$	Parameter sequence on n
$\{_{mci}P_n\}$	Cumulative probability distribution sequence on n, for material m, event c, and energy group, i
ϵ	A small input number, such that quantities less than this number may be equated to zero
δ	An input number larger than possible round off errors, but smaller than pertinent physical dimensions
c_2	Code indicating the method desired for generating the source particle's direction cosines
γ, α, β	The direction cosines of a particle's direction of motion
η	Function used in determining the source particle's direction cosines
c_3	Code indicating the method desired for generating the source particle's spatial coordinates
Z, X, Y	The cartesian coordinates of a particle's position

a_o, b_o, c_o	Constants for translating the axis of the source particle's spatial coordinates																
R	Radial source symmetry coordinate																
c_4	Code indicating the method desired for generating the source particle's energy																
E	Particle's energy																
E_o	Minimum allowable energy																
c_1	Code indicating the boundary type																
A, B, C, $X_o, Y_o,$ $Z_o, K_o,$	Constants describing the locus of the boundary																
r	Region number																
r_b	Value of the boundary equation evaluated at the particle's spatial coordinates																
F_2	Flag indicating whether or not r_b has been computed for this particle																
j	$\pm$ index determining the relationship of a boundary to a region																
h, e	Functions used in determining the distances to boundaries																
S_b	Distance to a boundary																
S_{bm}	Minimum positive distance to a boundary																
S_c	Distance to a collision point																
S	Distance traveled in a region since the previous collision or boundary crossing, whichever occurred later																
θ_m^h	Volume fraction of material m in composition h																
A_m	Nuclear mass of material m																
N_m	Number of nuclei per unit volume of material m																
Z_m	Number of electrons per nucleus of material m																
$\sigma_{mc}(E)$	Microscopic cross section of type c for material m and energy E																
	<table> <tr> <th>c</th><th>Cross Section Type</th></tr> <tr> <td>1</td><td>Neutron elastic scattering</td></tr> <tr> <td>2</td><td>Neutron inelastic scattering</td></tr> <tr> <td>3</td><td>Neutron fission</td></tr> <tr> <td>4</td><td>Neutron total</td></tr> <tr> <td>5</td><td>Gamma ray Compton scattering</td></tr> <tr> <td>6</td><td>Gamma ray photoelectric effect</td></tr> <tr> <td>7</td><td>Gamma ray pair production</td></tr> </table>	c	Cross Section Type	1	Neutron elastic scattering	2	Neutron inelastic scattering	3	Neutron fission	4	Neutron total	5	Gamma ray Compton scattering	6	Gamma ray photoelectric effect	7	Gamma ray pair production
c	Cross Section Type																
1	Neutron elastic scattering																
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3	Neutron fission																
4	Neutron total																
5	Gamma ray Compton scattering																
6	Gamma ray photoelectric effect																
7	Gamma ray pair production																
$\Sigma_{mc}^h(E)$	Macroscopic cross section of type c at energy E for material m in composition h																
$\Sigma_a^h(E)$	Total macroscopic absorption cross section at energy E in composition h																
$\Sigma_s^h(E)$	Combined macroscopic cross section for fission, elastic, and inelastic scattering at energy E in composition h																

E_{il}	Minimum energy of energy group i
E_{iu}	Maximum energy of energy group i
E_{mc}	Minimum energy for anisotropic scattering of type c in material m
${}^{mc}_{a1}(E)$, ${}^{mc}_{a2}(E)$	Coefficients of a cosine power series describing the neutron angular distribution after a scattering collision of type c in material m at energy E
M	Function used in determining the polar scattering angles by the cosine power series rejection technique
μ^{cm}	Cosine of the polar scattering angle in the center-of-mass system
μ^L	Cosine of the polar scattering angle in the laboratory system
E_{mx}	Excitation energy level value
B_{mx}	Function evaluated in a light inelastic collision
ν_{mi}	Average number of neutrons produced per fission in material m by a neutron in energy group i
N_{mci}	Average number of gamma rays emitted as a result of a collision of type c in material m by a neutron in energy group i
N_n	Actual number of neutrons or gamma rays emitted as a result of a neutron interaction
ϕ	Azimuthal scattering angle
N_{mc}	The number of particles in a cluster
W	The weight of each particle in the cluster
W_0	The cutoff weight
N	The number of neutrons or gamma rays that N_{mc} particles represent $N = WN_{mc}$
D_{ir}	Conversion factor for energy group i in region r
A_i	Acceleration factor for energy group i
I_i	Importance of energy group i
I_r	Importance of region r
Δ_1	Flag indicating fine splitting, exponential transformation, or neither of these options
d	Component of S_c used in the fine splitting option
c_{ir}	Fine splitting input constants for energy group i in region r
ΔW^s	Function computed in the fine splitting option
$\lambda^h(E)$	Computed mean free path in composition h at energy E
λ^r	Input mean free path for region r to be used in the exponential transformation
λ^r	Exponential transformation mean free path whose value lies between $\lambda^h(E)$ and λ^r
k	Function computed in the exponential transformation
N_t	Tally of the number of source (whole) particles run

E_t	Tally of the total energy of the source particles
N_z	A particle coordinate indicating the number of collisions the particle has undergone
H_1	The region numbers of the source region and the regions in which the first three collisions occurred
H_2	The region numbers in which the last four collisions occurred
P	The probability that the particle will continue to the region boundary in the direction chosen without a collision
D	Input code indicating the method desired for exponential transformation or fine splitting
H_m	Fission neutron heating constant for material m
C_c	Function computed in the light collision processes

3.4 Flow Chart Codes

Subroutine

Connector

Decision

Computation

Random number selection

4. PHYSICAL DESCRIPTION

The physical description of the source-shield configuration consists of related descriptions of the geometry, material, and nuclear properties of the materials.

4.1 Geometry

The source-shield configuration's geometrical description is accomplished by describing the boundaries of regions by the general equation:

$$AX^2 + X_0X + BY^2 + Y_0Y + CZ^2 + Z_0Z - K = 0. \quad (4.1)$$

Most desired geometrical shapes may be approximated by the proper specification of the parameters, A, B, C, X_0 , Y_0 , Z_0 , and K. Each region must have as many boundaries as necessary to completely enclose the region. Once a boundary has been described, it may be used as the boundary or parts of the boundaries for any number of regions. Outer boundaries for outside regions need not be described.

In an effort to simplify the geometrical input and reduce the computer running time, five special cases of the above equation are included. A code number, c_1 , is used to specify which equation is to be used and which parameters are required. The code number has the following meanings:

$$c_1 = 1 \quad AX^2 + X_0X + BY^2 + Y_0Y + CZ^2 + Z_0Z - K = 0 \quad (4.1)$$

$$c_1 = 2 \quad A(X-X_0)^2 + B(Y-Y_0)^2 + C(Z-Z_0)^2 - K = 0 \quad (4.2)$$

$$c_1 = 3 \quad (X-X_0)^2 + (Y-Y_0)^2 - K = 0 \quad (4.3)$$

$$c_1 = 4 \quad X - K = 0 \quad (4.4)$$

$$c_1 = 5 \quad Y - K = 0 \quad (4.5)$$

$$c_1 = 6 \quad Z - K = 0 \quad (4.6)$$

where A, B, C, X_0 , Y_0 , Z_0 , and K are the input parameters. Only the parameters necessary to define the selected equation may be entered as input. The coordinates of a point, Z, X, and Y, are the particle's spatial coordinates and are supplied by the program.

Equations 4.1, 4.2, 4.4, and 4.5 may be used to define reflecting plane surfaces by attaching a negative sign to the appropriate value of c_1 . Reflecting planes must contain or be parallel to the Z-axis. Attempted use of equations 4.1 and 4.2 without eliminating the Z terms or use of equations 4.3 and 4.6 to describe reflecting planes will result in program stops.

The simplest possible equation should be used to define a surface. Equations 4.1 and 4.2 may be used to define multiple surfaces; however, this should be done with extreme caution as they can give rise to what will be called "reflected regions". A reflected region is a region which is not described in input but exists because the conditions required for a particle to be in a region are satisfied for some region described elsewhere in the configuration. Figure 1 gives an example of this situation.

FIGURE 1

The above figure is rotated about the Z-axis. Region 1 will form a cone, which is the described region; and region 1A will form a cone, which is the reflected region. Any particles in the cone formed by region 1A will satisfy the conditions that are required for a particle to be in region 1.

Various regions may be described in several ways, and the regions for which the tallies are desired will dictate the method of describing the region. The example in Figure 2, in which the letters, A, B, C, and D, refer to surfaces and the numbers, 1, 2, 3, 4, 5, 6, and 7, refer to region numbers, shows the mid-plane section of a right circular cylinder centered in an enclosing sphere. Surface A is the spherical surface; surface B is the cylindrical surface; and surfaces C and D are plane surfaces intersecting the cylinder.

Figure 2

If tallies are desired in each of the regions as drawn,

region 1 should be bounded by surfaces B, C, and D;

region 2 should be bounded by surfaces A, B, C, and D;

region 3 should be bounded by surfaces A, B, and D;

region 4 should be bounded by surfaces A, B, and D;

region 5 should be bounded by surfaces A, B, and C; and

region 6 should be bounded by surfaces A, B, and C.

In all cases, region 7 is an outside region and only its inner surface, surface A, needs be described.

If regions 2, 3, and 5 are to be considered as one region for tallying purposes,

region 1 should be bounded by surfaces B, C, and D;

region (2,3,5) should be bounded by surfaces A and B;

region 4 should be bounded by surfaces A, B, and D; and

region 6 should be bounded by surfaces A, B, and C.

If regions 3 and 4 and regions 5 and 6 are to be considered as single regions for tallying purposes,

region 1 should be bounded by surfaces B, C, and D;

region 2 should be bounded by surfaces A, B, C, and D;

region (3,4) should be bounded by surfaces A and D; and

region (5,6) should be bounded by surfaces A and C.

It can readily be seen that several other combinations are possible by allowing the surfaces to exist or not exist between the regions, and an effort to describe all these combinations would be of no benefit.

In the above region descriptions, surfaces C and D were plane surfaces defined by $c_1=4$, $c_1=5$, or $c_1=6$. For simplicity, $c_1=6$; then the boundary equations become $Z=C$ and $Z=D$. If it is desired to consider regions 3 and 5 and regions 4 and 6 as single regions, the reflected regions may be used by choosing $c_1=1$ and setting $A=0$, $B=0$, $X_0=0$, and $Y_0=0$; C , Z_0 , and K are then chosen such that the roots of the resulting degenerate quadratic $CZ^2 + Z_0Z - K = 0$ are the Z -coordinates of surfaces C and D. Then the conditions that a particle be out of regions 1 and 2 and in regions 3 and 5 or regions 4 and 6 are identical. Therefore, regions 5 and 6 may be omitted from the descriptions as they are already defined as regions 3 and 4, respectively.

Regions 2, 3, 4, 5, and 6 cannot be combined into a single region because of the concavity caused by the intersection of surfaces C and D with surface B. Failure to describe any of the regions in the configuration will cause the comment "NO REGION FOR PARTICLE" with the parameters N_{MC} , γ , α , β , Z , X , Y , and E to be printed. The history of this particle will then be terminated.

4.2 Material

Each region is composed of a homogeneous mixture of materials (isotopes, elements, compounds, or mixtures) which are described individually by input nuclear data. This material description is accomplished by associating the desired composition with each region by means of a code number in the description for each region. Equating this code number to zero indicates a void region for which no nuclear data are entered. A composition must be specified for an outside region; however, any composition may be specified since the material properties are not used by the program. Compositions are expressed as volume fractions, θ_m^h , which are entered as input for each material contained in the configuration. The subscript m and the superscript h indicate material and composition, respectively.

4.3 Nuclear

Each material is composed of elementary units which can be involved in collision processes and will be called "nuclei". Although use of this term is correct only for isotopes, it will be used to designate units with average nuclear properties of the actual nuclei constituting the material. Quantities necessary to describe a material for FMC-N are the nuclear mass, A_m , and the number of nuclei per unit volume, N_m . Quantities necessary to describe a material for FMC-G are the number of nuclei per unit volume, N_m , and the number of electrons per nucleus, Z_m .

Energy groups, which are defined by the energy bounds, may be any size desired. The number of groups is also arbitrary; but once these quantities are set, they must remain constant for all materials. Output data are tallied for the same energy groups.

Microscopic cross sections required for FMC-N are (1) the elastic scattering cross section, σ_{m1} , (2) the inelastic scattering cross section, σ_{m2} , (3) the fission cross section, σ_{m3} , and (4) the total neutron cross section, σ_{m4} . Microscopic cross sections required for FMC-G are (1) the Compton scattering cross section, σ_{m5} , (2) the cross section for the photoelectric effect, σ_{m6} , and (3) the cross section for pair production, σ_{m7} .

Microscopic cross sections must be entered as straight line segments across each energy group, but they need not be continuous at the energy bounds. Therefore, there must be two entries for each material at each energy bound, except in the Compton scattering cross section which is not a function of material. The first entry, at each energy bound, is the second value of the cross section for the lower energy group, and the second entry is the first value of the cross section for the higher energy group. This is not true, of course, for the minimum and maximum energy bounds, where only one entry is needed for each. A linear interpolation on energy within an energy group is used to determine the actual cross section used for each particle. The initial entry for each material is an average value of the cross section to be used whenever the particle's energy falls below the cutoff energy. This value must be entered for the proper spacing of input whether or not a number of collisions cutoff is applied.

Neutron angular distributions after elastic and inelastic scattering may be anisotropic in the center-of-mass system if the incident neutron energy equals or exceeds minimum energies specified for each material and scattering event. The angular distributions are isotropic in the center-of-mass system for lower energies. The angular distributions, for anisotropic scattering, may be determined from cumulative probability distribution tables or by evaluation of a second degree cosine power series. If the cosine power series are used, one set of coefficients is required for each energy bound and each material for each scattering event. If cumulative probability distribution tables are used, a table is required for each energy group and each material for each scattering event in which the anisotropic scattering can occur.

The energy of a neutron after inelastic scattering from a light nucleus ($A_m < 20$) is computed using excitation levels determined from nuclear excitation probability tables. The energy after scattering from a heavy nucleus is determined from neutron spectrum cumulative probability distribution tables. In each case, a table is required for each material and energy group in which the event can occur.

Fission neutron energies are also chosen from cumulative probability distribution tables for each material and each energy group. One value of the average number of neutrons emitted per fission, ν_{mi} , is entered for each material and each energy group. The energies of gamma rays emitted in neutron absorption, fission, and inelastic scattering reactions are also determined from cumulative probability distribution tables. Separate tables are required for each material, neutron energy group, and type of event in which the reaction can occur.

The angular distribution of gamma rays after Compton scattering is determined from cumulative probability distribution tables, which are entered for each energy group only. The scattered energy is then computed from the Compton relationship.

Conversion factors must be entered in both programs for each energy group and each interior region for converting the absorption tallies to other desired units.

5. SOURCE DESCRIPTION

The direction, spatial, and energy coordinates of source neutrons and gamma rays may be entered as input from magnetic tape, from cards, or selected by the source particle generator, which is included in the programs. Fission neutrons and secondary gamma rays stored on magnetic tape from previous runs may be processed as source particles in subsequent runs.

5.1 Source Particle Generator

The source particle generator is a routine for initiating the seven parameters required to define each particle. These parameters are the direction cosines, γ , α , and β , the respective spatial coordinates, Z , X , and Y , and the energy E . The routine is available for use with either the neutron or the gamma ray program. If the source particle parameters are available from some other source, this routine is by-passed.

A source distribution may be approximated by the source particle generator by the use of input cumulative probability distribution tables to determine the parameters of each particle. It is assumed that the description of the source distribution can be separated into angular, spatial, and energy distributions. Additional separability assumptions are made in the approximation of the angular and spatial distributions in some of the options described on the following pages.

Each cumulative probability distribution table consists of two sets of data: (1) the values of the cumulative probabilities, and (2) the values of the corresponding random variables. $\{P_n\}$ is defined as the cumulative probability distribution sequence, and $\{X_n\}$ is the corresponding random variable sequence. The cumulative probability distribution sequence must start with one and end with zero. The cumulative probability distributions and their corresponding random variables are entered consecutively in pairs in each probability table. The probability in each pair immediately precedes the value of its corresponding random variable:

$$P_0 = 1, X_0, P_1, X_1, P_2, X_2, \dots, P_{N-1}, X_{N-1}, P_N = 0, X_N \quad (5.1)$$

Selections from the cumulative probability distribution tables are made in the following manner. A random number, ξ_n , is selected, and the proper interval is chosen by:

$$P_n < \xi_n \leq P_{n-1} \quad (5.2)$$

A second random number, ξ_{n+1} , is then selected, and the random variable is computed as:

$$X = X_n + \xi_{n+1} (X_{n-1} - X_n) \quad (5.3)$$

The number of cumulative probability distribution tables required by the source particle generator varies according to the methods used in selecting the parameters. Control words determine which method is to be used and the number of tables required. These tables are entered consecutively with the required number of tables for the direction cosines entered first; the required number of tables for the spatial coordinates are entered next; and the energy distribution table is entered last.

Equating any or all of the control words to zero allows the user to enter the concerned parameters as card input immediately following any cumulative probability distribution tables which may be required. The ordering of this input is γ , α , β , Z , X , Y , and E for each particle. Any parameters which are computed must be eliminated from this sequence.

5.1.1 Direction Cosines

A control word, c_2 , is used to select, from the following options, the desired method for selection of particle direction cosines:

$c_2=1$ One cumulative probability distribution table is required to provide the distribution of γ :

$$\gamma = \gamma_n + \xi_n (\gamma_{n-1} - \gamma_n) \quad (5.4)$$

Random numbers are selected until

$$\eta = (2\xi_{n+1} - 1)^2 + (2\xi_{n+2} - 1)^2 \leq 1; \text{ then} \quad (5.5)$$

$$\alpha = (2\xi_{n+1} - 1) [(1-\gamma^2)/\eta]^{\frac{1}{2}} \quad (5.6)$$

$$\beta = (2\xi_{n+2} - 1) [(1-\gamma^2)/\eta]^{\frac{1}{2}} \quad (5.7)$$

$c_2=2$ Two cumulative probability distribution tables are required. The first table provides the distribution of γ :

$$\gamma = \gamma_n + \xi_n (\gamma_{n-1} - \gamma_n) \quad (5.8)$$

The second table provides the distribution of α :

$$\alpha = \alpha_n + \xi_{n+1} (\alpha_{n-1} - \alpha_n) \quad (5.9)$$

α is selected until

$$\gamma^2 + \alpha^2 \leq 1; \text{ then} \quad (5.10)$$

$$\beta = \pm[1 - \gamma^2 - \alpha^2]^{\frac{1}{2}} \quad (5.11)$$

$c_2=3$ No cumulative probability distribution tables are required. γ , α , and β are chosen at random from an isotropic distribution.

$$\gamma = 2\xi_n - 1 \quad (5.12)$$

Random numbers are selected until

$$\eta = (2\xi_{n+1} - 1)^2 + (2\xi_{n+2} - 1)^2 \leq 1; \text{ then} \quad (5.13)$$

$$\alpha = (2\xi_{n+1} - 1) [(1-\gamma^2)/\eta]^{\frac{1}{2}} \quad (5.14)$$

$$\beta = (2\xi_{n+2} - 1) [(1-\gamma^2)/\eta]^{\frac{1}{2}} \quad (5.15)$$

5.1.2 Spatial Coordinates

A control word, c_3 , is used to select, from the following options, the desired method for selection of particle spatial coordinates:

$c_3=1$ Three cumulative probability distribution tables are required. The first table provides the distribution of Z:

$$Z = Z_n + \xi_n (Z_{n-1} - Z_n) \quad (5.16)$$

The second table provides the distribution of X:

$$X = X_n + \xi_{n+1} (X_{n-1} - X_n) \quad (5.17)$$

The third table provides the distribution of Y:

$$Y = Y_n + \xi_{n+2} (Y_{n-1} - Y_n) \quad (5.18)$$

This option provides a method of approximating a source with arbitrary separable spatial distributions over Z, X, and Y.

$c_3=2$ Two cumulative probability distribution tables are required. The first table provides the distribution of Z:

$$Z = Z_n + \xi_n (Z_{n-1} - Z_n) \quad (5.19)$$

The second table provides the distribution of $R = [X^2 + Y^2]^{\frac{1}{2}}$. X and Y are chosen at random from a distribution which is uniform per unit area in a ring of radius R_{n-1} to R_n . Random numbers are selected until:

$$\frac{(R_n)^2}{(R_{n-1})^2} < \xi_{n+1}^2 + \xi_{n+2}^2 \leq 1; \text{ then} \quad (5.20)$$

$$X = \pm \xi_{n+1} (R_{n-1}) + a_o \quad (5.21)$$

$$Y = \pm \xi_{n+2} (R_{n-1}) + b_o \quad (5.22)$$

This option provides a method of approximating a source possessing cylindrical symmetry.

$c_3=3$ One cumulative probability distribution table is required to provide the distribution of $R = [X^2 + Y^2 + Z^2]^{\frac{1}{2}}$. X, Y, and Z are chosen at random from a distribution which is uniform per unit volume. Random numbers are selected until:

$$\frac{(R_n)^2}{(R_{n-1})^2} < \xi_n^2 + \xi_{n+1}^2 + \xi_{n+2}^2 \leq 1; \text{ then} \quad (5.23)$$

$$Z = \pm \xi_n (R_{n-1}) + c_o \quad (5.24)$$

$$X = \pm \xi_{n+1} (R_{n-1}) + a_o \quad (5.25)$$

$$Y = \pm \xi_{n+2} (R_{n-1}) + b_o \quad (5.26)$$

This option provides a method of approximating a source possessing spherical symmetry.

The $\pm$ sign for any of the quantities is chosen at random by the selection of a random bit. The constants a_o , b_o , and c_o are provided as a means of translating the axis of any or all of the space variables, where applicable, such that off-axis sources may be described.

5.1.3 Energy

A control word, c_4 , is used to determine the method of selecting particle energies:

$c_4=1$ One cumulative probability distribution table is required to provide the energy distribution of the particles:

$$E = E_n + \xi_n (E_{n-1} - E_n) \quad (5.27)$$

6. PARTICLE HISTORY

The Monte Carlo method is a statistical technique used in analysis of physical and mathematical processes that are probabilistic in nature or can be so interpreted.

The transport of neutrons and gamma rays in a reactor and shield is an example of a physical process, probabilistic in nature, which can be conveniently handled by the methods of Monte Carlo. Average properties of neutrons and gamma rays such as absorption, flux, entrance, leakage, and history tallies are determined in this approach by tracing individual histories chosen from the appropriate probability distributions and averaging the scores from these histories for the desired quantities for a sufficiently large number of histories.

Thus, it is necessary to determine regions containing particles, distances to region boundaries, distances to collisions, types of collisions, changes in particle direction and energy after a scattering collision, location of birth of secondary particles, energy and direction at birth of secondary particles, particle weights, and conditions for termination of each particle history.

6.1 Geometry Calculation

The geometry calculations consist of determining the regions containing the particles and computing the distances to boundary crossings and collisions.

6.1.1 Region Containing Particle

Immediately following the boundary number, in the region description data, is the number of the region that the particle will most likely enter if the particle crosses this boundary. The program checks to see if the particle is in the region. If the particle is not in the region, the program checks all the remaining regions, in monotonically increasing order until the region containing the particle is found or the region search returns to the first region searched. If the particle cannot be found in any region, the comment "NO REGION FOR PARTICLE" with the parameters N_{MC} , γ , α , β , Z , X , Y , and E is printed. The history of this particle is then terminated. The programs perform a cyclic search on the regions, that is, after the last listed region has been excluded, they return to the first listed region and continue from there. For source particles, where no regions are specified, the programs

start the search with region one. Thus, for maximum efficiency, the source regions should be the first regions listed. When it is possible to enter several regions on crossing a particular boundary, these regions should have consecutive region numbers, and the region with the smallest region number should be entered as the most likely region to be entered.

In determining the region containing the particle, each boundary is treated separately. Therefore, a particle will be in a specified region only if the particle is in that region with respect to all the boundaries of the region. The region in which the particle is located is determined by the following procedure. Using the input parameters and the particle's spatial coordinates, Z, X, and Y, a quantity, r_b , is determined:

$$c_1 = 1 \quad r_b = AX^2 + X_0X + BY^2 + Y_0Y + CZ^2 + Z_0Z - K \quad (6.1)$$

$$c_1 = 2 \quad r_b = A(X-X_0)^2 + B(Y-Y_0)^2 + C(Z-Z_0)^2 - K \quad (6.2)$$

$$c_1 = 3 \quad r_b = (X-X_0)^2 + (Y-Y_0)^2 - K \quad (6.3)$$

$$c_1 = 4 \quad r_b = X - K \quad (6.4)$$

$$c_1 = 5 \quad r_b = Y - K \quad (6.5)$$

$$c_1 = 6 \quad r_b = Z - K \quad (6.6)$$

It can easily be seen that this quantity, r_b , evaluated at the particle's spatial coordinates, Z, X, and Y, is negative if the particle's position is on one side of the boundary. It is zero if the particle's position is on the boundary, and it is positive if the particle's position is on the other side of the boundary. Therefore, the quantity, r_b , determines on which side of the boundary the particle is located. This condition would be sufficient for determining the region containing the particle provided all the boundaries were entered in a particular sequence and the number of boundaries a region may have is limited. Rather than impose this restriction on the user, the concept of "inner" and "outer" boundaries is used. An inner boundary is defined as a boundary for which the quantity, r_b , is positive in relation to the particle's position. An outer boundary is then a boundary for which the quantity, r_b , is negative in relation to the particle's position. Since the same boundary may be an outer boundary for one region and an inner boundary for another region, a sign index ($\pm$), j, is attached to each boundary of each region to specify

the inner or outer relation. The sign is chosen such that the product, jr_b , for the particle's position, Z , X , and Y , is negative if the particle is in the region with respect to the boundary from which r_b is determined.

For a particle to be in a region, the product, jr_b , must be less than or equal to zero for all the boundaries of the region. The region is excluded from containing the particle whenever the product, jr_b , is greater than zero for any of the boundaries of the region.

The following example is given to clarify the assignment of these indices. Four cylindrical regions are formed by rotating the rectangles shown in Figure 3 about the Z -axis. The description of these regions and surfaces is accomplished as follows:

FIGURE 3

Surface A	$c_1 = 6$	$K = -4$		
Surface B	$c_1 = 6$	$K = -2$		
Surface C	$c_1 = 6$	$K = 2$		
Surface D	$c_1 = 6$	$K = 4$		
Surface E	$c_1 = 3$	$X_0 = 0$	$Y_0 = 0$	$K = 4$
Surface F	$c_1 = 3$	$X_0 = 0$	$Y_0 = 0$	$K = 16$

Boundaries for region 1 are B, C, and E;

boundaries for region 2 are A, B, and F;

boundaries for region 3 are B, C, E, and F; and

boundaries for region 4 are C, D, and F.

For region 1, boundary B $j = -$

boundary C $j = +$

boundary E $j = +$

For region 2, boundary A $j = -$

boundary B $j = +$

boundary F $j = +$

For region 3, boundary B	j = -
boundary C	j = +
boundary E	j = -
boundary F	j = +
For region 4, boundary C	j = -
boundary D	j = +
boundary F	j = +

Care must be exercised in describing these regions to prevent the actual region from being reflected into some other part of the configuration. These reflected regions may or may not cause trouble, depending on whether or not the reflection is intentional. If the reflection is unintentional, the trouble can be avoided by numbering the regions so that the program will test the true region, in trying to locate the region containing the particle, before testing the region which is being reflected. Another method of eliminating a reflected region is by the insertion of planes between the true region and the reflected region and naming this plane as a boundary of the region.

6.1.2 Distance to Boundary

The straight line distances from a particle's position in a region to all boundaries of the region are computed using the particle's direction cosines. The boundary which gives the minimum positive distances gives the actual boundary crossed and the distance to the boundary.

The distance to a boundary, S_b , is computed from the quadratic equation

$$h S_b^2 + 2 e S_b + r_b = 0 \quad (6.7)$$

which is derived from equations defining the intersecting straight line and the surface. The index, j , determines the proper root to use:

$$S_b = \frac{-e + j \sqrt{e^2 - h r_b}}{h} \quad (6.8)$$

Two special cases are considered. If the boundary is a plane surface and only one solution exists,

$$S_b = \frac{-r_b}{2e} \quad (6.9)$$

If two equal solutions exist,

$$S_b = \frac{-e}{h} \quad (6.10)$$

In the above equations, the quantity, r_b , has the meaning previously defined, and h and e depend upon the boundary type as follows:

$$c_1 = 1 \quad e = (2AX+X_0) \frac{\alpha}{2} + (2BY+Y_0) \frac{\beta}{2} + (2CZ+Z_0) \frac{\gamma}{2} \quad (6.11)$$

$$h = A\alpha^2 + B\beta^2 + C\gamma^2 \quad (6.12)$$

$$c_1 = 2 \quad e = A\alpha(X-X_0) + B\beta(Y-Y_0) + C\gamma(Z-Z_0) \quad (6.13)$$

$$h = A\alpha^2 + B\beta^2 + C\gamma^2 \quad (6.14)$$

$$c_1 = 3 \quad e = \alpha(X-X_0) + \beta(Y-Y_0) \quad (6.15)$$

$$h = 1-\gamma^2 \quad (6.16)$$

$$c_1 = 4 \quad e = \frac{\alpha}{2} \quad (6.17)$$

$$h = 0 \quad (6.18)$$

$$c_1 = 5 \quad e = \frac{\beta}{2} \quad (6.19)$$

$$h = 0 \quad (6.20)$$

$$c_1 = 6 \quad e = \frac{\gamma}{2} \quad (6.21)$$

$$h = 0 \quad (6.22)$$

In the above equations, X , Y , and Z are the particle's spatial coordinates; α , β , and γ are the particle's direction cosines; and the **remaining** quantities are the input boundary parameters.

6.1.3 Distance to Collision

Once the region containing the particle and the distance to the boundary have been determined, the mean free path for the energy and region is computed by forming the reciprocal of the sum of the total macroscopic cross sections for all the materials in the region:

$$\lambda^h(E) = \left[\sum_{m=1}^M \sigma_m^h N_m \sigma_{m4}(E) \right]^{-1} \quad (6.23)$$

The distance to the point of collision is then computed as:

$$S_c = -\lambda^h(E) \ln \xi \quad (6.24)$$

6.1.4 Reflective Boundary

In addition to being changed by collision processes, a particle's direction can be changed by encounter with a plane reflective boundary. This event leads to reflection of the particle in accordance with the usual reflection law, i.e., the angle between the normal to the plane at the point of encounter and the direction of the reflected particle equals the angle between the normal to the plane at the point of encounter and the incident direction of the particle as follows:

$$\alpha' = \left[\frac{\alpha(Y_0^2 - X_0^2) - 2\beta X_0 Y_0}{(X_0^2 + Y_0^2) \sqrt{\alpha^2 + \beta^2}} \right] \sqrt{1 - \gamma^2} \quad (6.25)$$

$$\beta' = \left[\frac{\beta(X_0^2 - Y_0^2) - 2\alpha X_0 Y_0}{(X_0^2 + Y_0^2) \sqrt{\alpha^2 + \beta^2}} \right] \sqrt{1 - \gamma^2} \quad (6.26)$$

$$\gamma' = \gamma$$

where γ , α , and β are the particle's direction cosines and X_0 and Y_0 are either input parameters or one or zero depending upon the boundary type:

$c_1 = 1, 2$	X_0 and Y_0 are input parameters
$c_1 = 4$	$X_0 = 1, Y_0 = 0$
$c_1 = 5$	$X_0 = 0, Y_0 = 1$
$c_1 = 3, 6$	Illegal boundary types

6.2 Collision Processes

The collision routine selects one of the possible events, other than non-fission absorption, that can occur at a collision and processes the particle according to the event chosen.

6.2.1 Neutron

The microscopic neutron cross sections, σ_{mc} , for elastic scattering, $c=1$, inelastic scattering, $c=2$, fission, $c=3$, and the total neutron cross section, $c=4$, must be entered as input for all the materials in the source-shield configuration. These cross sections must be entered as discontinuous functions

of energy, i.e., values must be entered for both the lower and upper energy bounds for each energy group. They will be subscripted il for the lower bound and iu for the upper bound of energy group i .

At each collision, the macroscopic cross sections for all types of events and for all materials in the region are computed. The values of the microscopic cross sections are obtained by linear interpolation:

$$\sigma_{mc}(E) = \sigma_{mc}(E_{il}) + \frac{(E - E_{il})}{(E_{i+1} - E_{il})} \left[\sigma_{mc}(E_{iu}) - \sigma_{mc}(E_{il}) \right] \quad (6.28)$$

unless the incident energy is less than or equal to the cutoff energy, when

$$\sigma_{mc}(E) = \sigma_{mc}(E_0) \quad (6.29)$$

The macroscopic cross sections are computed as the product of the microscopic cross sections, the volume fractions of the material in the region, and the number of nuclei per unit volume of the materials for each material in the region:

$$\Sigma_{mc}^h(E) = \theta_m^h N_m \sigma_{mc}(E) \quad (6.30)$$

The absorption cross section for each material is the difference between the total cross section and the sum of the elastic scattering, inelastic scattering, and the fission cross sections:

$$\Sigma_{ma}^h(E) = \Sigma_{m4}^h(E) - \Sigma_{m1}^h(E) - \Sigma_{m2}^h(E) - \Sigma_{m3}^h(E) \quad (6.31)$$

The total absorption cross section is the sum of the absorption cross section for all the materials in the region:

$$\Sigma_a^h(E) = \sum_{m=1}^M \Sigma_{ma}^h(E) \quad (6.32)$$

The mean free path is the reciprocal of the sum of the total cross sections of the materials in the region:

$$\lambda^h(E) = \left[\sum_{m=1}^M \Sigma_{m4}^h(E) \right]^{-1} \quad (6.33)$$

A particle emerging from a collision is given a weight equal to the probability that it escaped absorption at that point multiplied by the probability that it escaped absorption at all prior collisions. This eliminates the bias introduced by the event sampling scheme. The weight of the particle after collision becomes:

$$W = W \left[1 - \lambda^h(E) \Sigma_a^h(E) \right] \quad (6.34)$$

unless $[1 - \lambda^h(E) \Sigma_a^h(E)]$ is less than the input quantity, ϵ , at which the particle is assumed totally absorbed, and the particle history is terminated.

The combined macroscopic cross section for elastic scattering, inelastic scattering, and fission in the region becomes:

$$\Sigma_s^h(E) = \sum_{m=1}^M \sum_{c=1}^3 \Sigma_{mc}^h(E) = \frac{1 - \lambda^h(E) \Sigma_a^h(E)}{\lambda^h(E)} \quad (6.35)$$

Selection of the collision event in a particular material is made in the following manner. A random number is selected, and the difference is formed between the random number and the cross section for an individual event divided by the combined macroscopic cross section for the region

$$\xi_n = \xi_n - \Sigma_{mc}^h(E) / \Sigma_s^h(E) \quad (6.36)$$

varying the macroscopic cross section first on m ($m=1, 2, 3, \dots, M$) and then on c ($c=1, 2, 3$) until ξ_n becomes negative. The values of m and c that accomplish this determine the material causing the event and the type of event, respectively.

If the event is a scattering event, the neutron's incident energy is compared with an input minimum energy, E_{mc} , which is a function of material and scattering event. If the incident energy is less than or equal to this energy, the scattering is assumed isotropic in the center-of-mass system; and the cosine of the polar scattering angle is:

$$\mu^{cm} = 2\xi - 1 \quad (6.37)$$

If the incident energy is greater than this energy, the scattering is assumed anisotropic in the center-of-mass system. The angular dependence is then optionally determined from input cumulative probability distribution

tables or by evaluating a second-degree cosine power series. If cumulative probability distribution tables are used for the scattering probabilities, they must be used consistently throughout the calculation. Likewise, if a second-degree cosine power series is used for the scattering probabilities, it must be used consistently throughout the calculation. The cumulative probability distribution tables and the cosine power series may not be intermixed.

If cumulative probability distribution tables are used, the proper table for the energy group, material, and scattering event is selected. A random number, ξ_1 , is selected to locate the proper interval within the table. A second random variable, ξ_2 , is chosen, and the value of the random variable is determined by linear interpolation:

$$\mu^{cm} = \mu_n^{cm} + \xi_2 (\mu_{n-1}^{cm} - \mu_n^{cm}) \quad (6.38)$$

If the cosine power series is used and the incident neutron energy, E , is greater than the cutoff energy, E_0 , the values of the random variables are determined by linear interpolation:

$$mc_{a1}(E) = mc_{a1}(E_i) + \frac{(E - E_i)}{(E_{i+1} - E_i)} [mc_{a1}(E_{i+1}) - mc_{a1}(E_i)] \quad (6.39)$$

$$mc_{a2}(E) = mc_{a2}(E_i) + \frac{(E - E_i)}{(E_{i+1} - E_i)} [mc_{a2}(E_{i+1}) - mc_{a2}(E_i)] \quad (6.40)$$

If the incident neutron energy, E , is less than or equal to the cutoff energy, E_0 , then:

$$mc_{a1}(E) = mc_{a1}(E_0) \quad (6.41)$$

$$mc_{a2}(E) = mc_{a2}(E_0) \quad (6.42)$$

A rejection technique is then used to determine the cosine of the polar scattering angle in the center-of-mass system. A random number, ξ_1 , is selected, and the cosine of the polar scattering angle, μ^{cm} , is tentatively set to:

$$\mu^{cm} = 2\xi_1 - 1 \quad (6.43)$$

A second random number, ξ_2 , is then selected, and a test is made to determine whether

$$M\xi_2 \leq 1 + mc_{a1}(E) \mu^{cm} + mc_{a2}(E) (\mu^{cm})^2 \quad (6.44)$$

or

$$M\xi_2 > 1 + mc_{a1}(E) \mu^{cm} + mc_{a2}(E) (\mu^{cm})^2 \quad (6.45)$$

where

$$M = 1 - \frac{[mc_{a1}(E)]^2}{4mc_{a2}(E)} \quad (6.46)$$

if

$$mc_{a2}(E) < 0 \quad \text{and} \quad |mc_{a1}(E)| < |2mc_{a2}(E)| ,$$

and

$$M = 1 + |mc_{a1}(E)| + mc_{a2}(E) \quad (6.47)$$

for all other conditions.

If equation (6.44) holds, the value of μ^{cm} is retained. If equation (6.45) holds, the value of μ^{cm} is rejected and the process is repeated.

If the nuclear mass of the scattering nucleus is less than or equal to 20, the cosine of the polar scattering angle in the center-of-mass system is converted to the corresponding angle in the laboratory system. If the nuclear mass of the scattering nucleus is greater than 20, it is assumed infinite and the cosine of the polar scattering angle in the laboratory system is set equal to the cosine of the polar scattering angle in the center-of-mass system.

For a heavy elastic collision, the scattered energy remains unchanged:

$$E' = E \quad (6.48)$$

For a heavy inelastic collision, the scattered energy is determined from cumulative probability distribution tables which are entered for each material and each energy group for the inelastically scattered neutrons. A random number, ξ_1 , is selected to locate the proper interval within the table. A second random number, ξ_2 , is chosen, and the value of the random variable is determined by linear interpolation:

$$E' = E_n + \xi_2 (E_{n-1} - E_n) \quad (6.49)$$

If the energy deposition is desired rather than the absorption tally, the product of the neutron's weight and the energy loss is tallied for the proper region and energy absorption bin:

$$H = W(E-E') \quad (6.50)$$

For a light elastic collision, the cosine of the polar scattering angle in the center-of-mass system is converted to the cosine of the polar scattering angle in the laboratory system:

$$\mu^L = \frac{1 + A_m \mu^{cm}}{(1 + A_m^2 + 2A_m \mu^{cm})^{\frac{1}{2}}} \quad (6.51)$$

The requirements of the conservation of energy and momentum and a knowledge of the cosine of the polar scattering angle in the center-of-mass system are applied in determining the scattered energy:

$$E' = \frac{E(1 + A_m^2 + 2A_m \mu^{cm})}{(1 + A_m)^2} \quad (6.52)$$

If the energy deposition is desired rather than the absorption tally, the product of the neutron's weight and the energy loss is tallied for the proper region and energy absorption bin:

$$H = W(E-E') \quad (6.53)$$

For a light inelastic collision, the excitation energy must be considered. This is accomplished by including an additional term in the equation for conservation of energy to account for the change in the internal energy of the nucleus:

$$B_{mx} = \left[1 - \frac{(A_m + 1)}{A_m} \frac{E_{mx}}{E} \right]^{\frac{1}{2}} \quad (6.54)$$

where E_{mx} is the excitation energy level value.

The conversion of the cosine of the polar scattering angle in the center-of-mass system to the cosine of the polar scattering angle in the laboratory system is then accomplished in the usual manner:

$$\mu^L = \frac{1 + A_m B_{mx} \mu^{cm}}{(1 + A_m^2 B_{mx}^2 + 2A_m B_{mx} \mu^{cm})^{\frac{1}{2}}} \quad (6.55)$$

The scattered energy then becomes:

$$E' = \frac{E(1 + A_m^2 B_{mx}^2 + 2A_m B_{mx} \mu^{cm})}{(1 + A_m)^2} \quad (6.56)$$

If the energy deposition is desired rather than the absorption tally, the product of the neutron's weight and the energy loss is tallied for the proper region and energy absorption bin:

$$H = WEA_m \left[\frac{1 + B_{mx}^2 - 2B_{mx} \mu^{cm}}{(1 + A_m)^2} \right] \quad (6.57)$$

The azimuthal scattering angle, ϕ , is then chosen from a uniform distribution on the closed interval $[-\pi, \pi]$:

$$\phi = \pi(2\xi - 1) \quad (6.58)$$

The direction cosines in the laboratory system are computed from the polar and azimuthal angles in this system. The direction cosines in the reference system fixed in the source-shield configuration are then computed as:

$$\alpha' = \frac{\alpha \gamma \cos \phi [1 - (\mu^L)^2]^{\frac{1}{2}} - \beta \sin \phi [1 - (\mu^L)^2]^{\frac{1}{2}}}{[1 - \gamma^2]^{\frac{1}{2}}} + \alpha \mu^L \quad (6.59)$$

$$\beta' = \frac{\beta \gamma \cos \phi [1 - (\mu^L)^2]^{\frac{1}{2}} + \alpha \sin \phi [1 - (\mu^L)^2]^{\frac{1}{2}}}{[1 - \gamma^2]^{\frac{1}{2}}} + \beta \mu^L \quad (6.60)$$

$$\gamma' = -\cos \phi [1 - (\mu^L)^2]^{\frac{1}{2}} [1 - \gamma^2]^{\frac{1}{2}} + \gamma \mu^L \quad (6.61)$$

if $(1 - \gamma^2) > \epsilon$; or

$$\alpha' = \cos \phi [1 - (\mu^L)^2]^{\frac{1}{2}} \quad (6.62)$$

$$\beta' = \sin \phi [1 - (\mu^L)^2]^{\frac{1}{2}} \quad (6.63)$$

$$\gamma' = \mu^L$$

if $(1 - \gamma^2) \leq \epsilon$.

In the above equations:

$$\sin \phi = -[1 - \cos^2 \phi]^{\frac{1}{2}} \quad (6.65)$$

if $\phi \leq 0$; or

$$\sin \varphi = [1 - \cos^2 \varphi]^{\frac{1}{2}} \quad (6.66)$$

if $\varphi > 0$.

In these equations, α , β , and γ are the direction cosines of the particle in the fixed reference system before scattering.

For a neutron fission event, the incident neutron's weight must be taken into consideration in determining the number of whole neutrons to be emitted:

$$N_n = \left[Wv_{mi} + \xi \right] \quad (6.67)$$

The energy of each fission neutron is chosen from the appropriate cumulative probability distribution table which is entered for each material and each neutron energy group. A random number, ξ_1 , is selected to locate the proper interval within the table. A second random number, ξ_2 , is chosen, and the value of the random variable is determined by linear interpolation:

$$E = E_n + \xi_2 (E_{n-1} - E_n) \quad (6.68)$$

The direction cosines of the fission neutrons are computed from an isotropic distribution in the laboratory system:

$$\gamma = 2\xi_1 - 1 \quad (6.69)$$

Random numbers are selected until

$$\eta = \xi_n^2 + \xi_{n+1}^2 \leq 1; \text{ then} \quad (6.70)$$

$$\alpha = \pm \xi_n [(1 - \gamma^2)/\eta]^{\frac{1}{2}} \quad (6.71)$$

$$\beta = \pm \xi_{n+1} [(1 - \gamma^2)/\eta]^{\frac{1}{2}} \quad (6.72)$$

The $\pm$ sign is chosen at random by the selection of a random bit.

The necessary particle information, the seven parameters required to define each particle, for each fission neutron is stored in the fission neutron storage for later processing or transfer to magnetic tape storage. The weight of the incident neutron is tallied for the proper region and energy absorption bin unless the energy deposition is desired. If the energy deposition is desired, the product of the incident neutron's weight and the fission heating constant, H_m , is tallied for the proper region and energy absorption bin:

$$H = WH_m \quad (6.73)$$

The history of the incident neutron is then terminated, and the program returns to obtain parameters of another particle from born storage.

Gamma rays may be emitted in neutron absorption, inelastic scattering, and fission events. If gamma rays are to be emitted for these events, the appropriate number of gamma rays to be emitted, N_{mci} , for the material, collision type, and incident energy group is selected. The incident neutron's weight must be taken into consideration in determining the number of whole gamma rays to be emitted:

$$N_n = \{WN_{mci} + \xi\} \quad (6.74)$$

The energy of each gamma ray is chosen from the appropriate cumulative probability distribution table which is entered for each material and each neutron energy group. A random number, ξ_1 , is selected to locate the proper interval within the table. A second random number, ξ_2 , is chosen, and the value of the random variable is determined by linear interpolation:

$$E_\gamma = E_n + \xi_2 (E_{n-1} - E_n) \quad (6.75)$$

The direction cosines of the emitted gamma rays are computed from an isotropic distribution in the laboratory system in the same manner as the fission-produced neutrons. The necessary particle information, the seven parameters required to define each particle, for each gamma ray is stored in the secondary gamma ray storage for later transfer to magnetic tape storage.

6.2.2 Gamma Ray

The microscopic gamma ray cross sections, σ_{mc} , for the photoelectric effect, $c=6$, and pair production, $c=7$, must be entered as input for all the materials in the source-shield configuration. The microscopic gamma ray cross section for Compton scattering, $c=5$, is entered only once. These cross sections must be entered as discontinuous functions of energy, i.e., values must be entered for both the lower and upper energy bounds for each energy group. They will be subscripted il for the lower bound and iu for the upper bound of energy group i .

At each collision, the macroscopic cross sections for all types of events and for all materials in the region are computed. The values of the microscopic cross sections are determined by linear interpolation:

$$\sigma_{mc}(E) = \sigma_{mc}(E_{i\ell}) + \frac{(E - E_i)}{(E_{i+1} - E_i)} \left[\sigma_{mc}(E_{iu}) - \sigma_{mc}(E_{i\ell}) \right] \quad (6.76)$$

unless the incident energy is less than or equal to the cutoff energy, when

$$\sigma_{mc}(E) = \sigma_{mc}(E_0) \quad (6.77)$$

The macroscopic cross sections for the photoelectric effect and pair production are computed as the product of the microscopic cross sections per nucleus, the volume fraction of the material in the region, and the number of nuclei per unit volume of the materials for each material in the region:

$$\Sigma_{mc}^h(E) = \theta_m^h N_m \sigma_{mc}(E) \quad (6.78)$$

The macroscopic cross section for Compton scattering is computed as the product of the microscopic cross section per electron, the volume fraction of the material in the region, the number of nuclei per unit volume of the material, and the number of electrons per nucleus of the material for each material in the region:

$$\Sigma_{m5}^h(E) = \theta_m^h N_m Z_m \sigma_5(E) \quad (6.79)$$

The total absorption cross section is the sum of the macroscopic cross sections for the photoelectric effect and pair production:

$$\Sigma_a^h(E) = \sum_{m=1}^M \sum_{c=6}^7 \Sigma_{mc}^h(E) \quad (6.80)$$

The mean free path is the reciprocal of the sum of the total absorption cross section and the Compton scattering cross section:

$$\lambda^h(E) = \left[\Sigma_a^h(E) + \sum_{m=1}^M \Sigma_{m5}^h(E) \right]^{-1} \quad (6.81)$$

A particle emerging from a collision is given a weight equal to the probability that it escaped absorption at that point multiplied by the probability that it escaped absorption at all prior collisions. This eliminates the bias introduced by the event sampling scheme. The weight of the particle after collision becomes:

$$W = W[1 - \lambda^h(E) \Sigma_a^h(E)] \quad (6.82)$$

unless $[1 - \lambda^h(E) \Sigma_a^h(E)]$ is less than the input quantity, ϵ , at which the particle is assumed totally absorbed, and the particle history is terminated.

The angular dependence of the scattered gamma ray is determined from the appropriate cumulative probability distribution table which is entered as input for each energy group. A random number, ξ_1 , is selected to locate the proper interval within the table. A second random number, ξ_2 , is chosen, and the value of the random variable is determined by linear interpolation:

$$\mu^L = \mu_n^L + \xi_2 (\mu_{n-1}^L - \mu_n^L) \quad (6.83)$$

The scattered energy of the gamma ray is computed from the Compton relationship:

$$E' = \frac{E}{1 + \frac{E}{.511} (1 - \mu^L)} \quad (6.84)$$

where the units of E are Mev.

If the energy deposition is desired rather than the absorption tally, the product of the gamma ray's weight and the energy loss is tallied for the proper region and energy absorption bin:

$$H = W(E - E') \quad (6.85)$$

A pair of 500 kev gamma rays may be emitted in the pair production event. If 500 kev gamma rays are to be emitted, one gamma ray is emitted with an assigned energy of 500 kev. The weight of the incident gamma ray is doubled and assigned to this gamma ray to account for the pair. The direction cosines of this gamma ray are computed from an isotropic distribution in the laboratory system. The parameters of this gamma ray are then stored in born storage to be processed in its turn.

7. OPTIONAL VARIANCE REDUCTION TECHNIQUES

Certain variance reduction techniques are allowed in the program in addition to the non-optional statistical estimation technique of weighting for absorption escape at each collision. All of these options are forms of importance sampling except for a statistical estimation technique used for scoring expectation entrance, leakage, and flux tallies.

7.1 Acceleration Factors

In addition to the variance reduction techniques described above, there is an optional method in the program which provides advantages in sampling from the source spectrum. This method involves the use of energy group constants called acceleration factors. The acceleration factors offer a method of readjusting the initial size of each particle cluster, N_{mc} , by an appropriate energy group constant, A_i .

The method of systematic sampling may be applied to reshape the source spectrum after having picked from a uniform distribution or some other incorrect distribution. By specifying the acceleration factor as a positive quantity, the adjustment is:

$$N'_{mc} = \{N_{mc} A_i + \xi\} \quad (7.1)$$

$$W' = 1 \quad (7.2)$$

The method of importance sampling may be applied to obtain more particles in the energy groups of interest without altering the source spectrum. By specifying the acceleration factor as a negative quantity, the adjustment is:

$$N'_{mc} = \{ |N_{mc} A_i| + \xi \} \quad (7.3)$$

$$W' = 1/N'_{mc} \quad (7.4)$$

7.2 Splitting and Russian Roulette

Splitting may be used on location at region boundary crossings, on change in energy at collision points, and on change in location, direction, and energy at collision points.

Splitting results in a change in the number of particles being followed. Consequently, the particle weights must be correctly adjusted to eliminate the bias introduced by this scheme. This can be accomplished by requiring that the total number of neutrons or gamma rays involved in the process is conserved:

$$W'N'_{mc} = WN_{mc} \quad (7.5)$$

Thus, the new weight is

$$W' = \frac{WN_{mc}}{N'_{mc}} \quad (7.6)$$

Splitting on location at region boundary crossings is accomplished by means of input region importances, I_r . The number of particles in the cluster after each boundary crossing is:

$$N'_{mc} = \left[N_{mc} \frac{I'_r}{I_r} + \xi \right] \quad (7.7)$$

If the new region is one of less importance, then $I'_r/I_r < 1$; and the integer part of $\left[N_{mc} \frac{I'_r}{I_r} + \xi \right]$ can be zero. Hence, although the new weight to be assigned to each particle is

$$W' = \frac{WN_{mc}}{N'_{mc}}, \quad (7.8)$$

W' is approximated by

$$W' = W \frac{I_r}{I'_r} \quad (7.9)$$

since

$$\frac{N'_{mc}}{N_{mc}} \approx \frac{I_r}{I'_r} \quad (7.10)$$

Note that if N_{mc} is 1 and $I'_r/I_r < 1$, the above procedure can be interpreted as a game of chance in which the particle survives I'_r/I_r of the time with a weight of I_r/I'_r and is killed the rest of the time. This process is called Russian roulette.

If the new region is one of greater importance, then $I'_r/I_r > 1$; and the above process results in the number of particles in the cluster being increased. The weight of the cluster becomes

$$W' = \frac{WN_{mc}}{N_{mc}} \quad (7.11)$$

Splitting on change in energy at collision points is accomplished by means of input energy group importances, I_i . The number of particles in a cluster after each collision is

$$N'_{mc} = \left[N_{mc} \frac{I'_i}{I_i} + \xi \right] \quad (7.12)$$

Each particle in a cluster has a separate entity after its collision, i.e., a separate collision point is picked for each particle. Therefore, at each collision point, N_{mc} equals one and the above equation becomes

$$N'_{mc} = \left[\frac{I'_i}{I_i} + \xi \right] \quad (7.13)$$

If the new energy group is one of less importance, then $I'_i/I_i < 1$; and the integer part of $\left[\frac{I'_i}{I_i} + \xi \right]$ can be zero. Hence, although the new weight to be assigned to each particle is

$$W' = \frac{WN_{mc}}{N'_{mc}}, \quad (7.14)$$

W' is approximated by

$$W' = \frac{WI_i}{I'_i} \quad (7.15)$$

since

$$\frac{N'_{mc}}{N_{mc}} \approx \frac{I'_i}{I_i} \quad (7.16)$$

This is again a case of Russian roulette.

If the new energy group is one of greater importance, then $I'_i/I_i > 1$; and the process results in the number of particles in the cluster being increased. The weight of the cluster becomes

$$W' = \frac{W}{N'_{mc}} \quad (7.17)$$

Splitting on change in location, direction, and energy at collision points is called "fine splitting". As the first step in fine splitting, a weight change is computed as the exponential of the product of a computed distance component, d , and an input quantity, c_{ir} .

$$\Delta W_s = \exp - (c_{ir} d) \quad (7.18)$$

The input quantities, c_{ir} , are entered for each energy group and each region in which fine splitting is desired. Components, d , of the distance traversed between the collision point and the previous collision or boundary-crossing point, whichever occurred later, may be computed along any of the three axes of the rectangular coordinate system or the radius of either the cylindrical or spherical coordinate systems. Any of the following equations may be used for computing these components in a region by proper assignment of the code number in the fourth word in the description of the region:

Code No.	Equation	
D = 0	$d = 0$ (Perform no fine splitting)	(7.19)
D = 8	$d = X_n - X$	(7.20)
D = 9	$d = X - X_n$	(7.21)
D = 10	$d = Y_n - Y$	(7.22)
D = 11	$d = Y - Y_n$	(7.23)
D = 12	$d = Z_n - Z$	(7.24)
D = 13	$d = Z - Z_n$	(7.25)
D = 14	$d = [(X_n - X_0)^2 + (Y_n - Y_0)^2]^{\frac{1}{2}} - [(X - X_0)^2 + (Y - Y_0)^2]^{\frac{1}{2}}$	(7.26)
D = 15	$d = [(X - X_0)^2 + (Y - Y_0)^2]^{\frac{1}{2}} - [(X_n - X_0)^2 + (Y_n - Y_0)^2]^{\frac{1}{2}}$	(7.27)
D = 16	$d = [(X_n - X_0)^2 + (Y_n - Y_0)^2 + (Z_n - Z_0)^2]^{\frac{1}{2}} - [(X - X_0)^2 + (Y - Y_0)^2 + (Z - Z_0)^2]^{\frac{1}{2}}$	(7.28)
D = 17	$d = [(X - X_0)^2 + (Y - Y_0)^2 + (Z - Z_0)^2]^{\frac{1}{2}} - [(X_n - X_0)^2 + (Y_n - Y_0)^2 + (Z_n - Z_0)^2]^{\frac{1}{2}}$	(7.29)

In the above equations, no subscript refers to the particle's spatial coordinates at the last collision or boundary crossing point, whichever occurred later. The subscript, n , refers to the particle's spatial coordinates at its present collision point, and the subscript, 0 , refers to the axis of translation of the boundary equation under consideration.

Consideration of the conservation of particles at a collision leads to the following equation for the number of particles resulting from fine splitting:

$$N'_{mc} = \left[\frac{1}{\Delta W_s} + \xi \right] \quad (7.30)$$

Since the distance component, d , and the region and energy group constant, c_{ir} , may be positive or negative, the number of particles resulting may be increased or decreased.

The programs actually combine the sets of equations for both methods of splitting at collision points as follows:

$$N'_{mc} = \left[\frac{I'_i}{I_i} \frac{1}{\Delta W_s} + \xi \right] \quad (7.31)$$

It can be seen that if

$$\frac{I'_i}{I_i} \frac{1}{\Delta W_s} < 1, \quad (7.32)$$

a game of Russian roulette is played with the particle. If the particle survives, its weight is

$$W' = \frac{I'_i}{I_i} \frac{W}{\Delta W_s} \quad (7.33)$$

Whereas, if

$$\frac{I'_i}{I_i} \frac{1}{\Delta W_s} > 1, \quad (7.34)$$

one or more particles result, and the weight becomes

$$W' = \frac{W}{N'_{mc}} \quad (7.35)$$

It can easily be verified that these equations reduce to the appropriate equations for splitting on change in energy by substituting $\Delta W_s = 1$. Furthermore, it can easily be verified that these equations reduce to the appropriate equations for fine splitting by equating the energy importances.

Russian roulette is also played with particles whose weight has fallen below the cutoff weight, W_0 . These particles are eliminated the fraction $(1 - W/W_0)$ of the time and retained with increased weight, W_0 , the remainder of the time.

7.3 Exponential Transformation

The exponential transformation is a method of increasing the probability of success of a particle by varying the mean free path length as a function of the particle's direction in a region. The mean free path length of the particle may be varied in any of the six directions, $\pm\alpha$, $\pm\beta$, $\pm\gamma$, as desired. An input code number in the fourth word in the description of the region determines the direction which the mean free path length is to be varied. It is also possible to use the exponential transformation in a manner that is independent of direction by setting the code number equal to 1.

<u>Code Number</u>	<u>Equation</u>	
D = 0	$k = 0$ (perform no exponential transformation)	(7.36)
D = 1	$k = 1$	(7.37)
D = 2	$k = \alpha$	(7.38)
D = 3	$k = -\alpha$	(7.39)
D = 4	$k = \beta$	(7.40)
D = 5	$k = -\beta$	(7.41)
D = 6	$k = \gamma$	(7.42)
D = 7	$k = -\gamma$	(7.43)

Then, provided $k > 0$, the mean free path length of the particle in the region is converted from $\lambda^h(E)$ to

$$\lambda^r(E) = \left[\frac{k}{\lambda^r} + \frac{1-k}{\lambda^h(E)} \right]^{-1} \quad (7.44)$$

If $k < 0$, the particle is already proceeding in the direction of interest and the exponential transformation will be ignored.

If the exponential transformation is performed in the region and it is found that a collision has occurred in the region, the bias introduced by the exponential transformation is removed from the weight of the particle by:

$$W' = W \frac{\lambda^r(E)}{\lambda^h(E)} \exp - S_c \left(\frac{1}{\lambda^h(E)} - \frac{1}{\lambda^r(E)} \right) \quad (7.45)$$

If the exponential transformation is performed in the region and it is found that a boundary crossing has occurred, the bias introduced by the exponential transformation is removed from the weight of the particle by:

$$W' = W \exp - S_b \left(\frac{1}{\lambda^h(E)} - \frac{1}{\lambda^r(E)} \right) \quad (7.46)$$

8. OUTPUT

Output available from these programs include absorption or energy deposition tallies, Monte Carlo entrance and leakage tallies, expectation entrance and leakage tallies, Monte Carlo or expectation flux tallies, and Monte Carlo history tallies of particles reaching selected regions. Leakage tallies are made for all regions external to the source-shield configuration. The absorption, energy deposition, entrance, leakage, and flux tallies are made for each energy group in each region in which the tallies are desired.

The absorption tally for each region and energy is formed by summing over all collisions in the region the product of the weight of the particle just prior to the collision and the probability of absorption at the collision for each particle in the energy group. In order to make the absorption tally more useful, this product for each collision is multiplied by a region and energy dependent conversion factor. If the energy deposition is desired in the source-shield configuration, the appropriate energy loss from the type of collision is calculated and summed into these tallies. These tallies are made for all regions internal to the source-shield configuration, but the output for the region may be suppressed by equating the first conversion factor for the region to zero.

The expectation tally for a selected region and energy group is formed as follows:

1. The weight of a particle or cluster in the energy group after each collision or after leaving the source is multiplied by the probability of the particle or cluster continuing in the direction chosen and entering the selected region without further collision.
2. These products are then summed over all particle or cluster histories.

A Monte Carlo entrance tally for a selected region and energy group is formed by summing over all clusters the product of the weight of each particle cluster entering the selected region in the energy group and the number of neutrons or gamma rays in each cluster. Monte Carlo and expectation entrance tallies for outside regions are called leakage tallies and are mandatory. Monte Carlo and expectation entrance tallies are formed for each region in which these tallies are desired. The expectation tallies may be omitted by specifying that the expectation computations are to be omitted.

Flux tallies for selected regions, including void regions, and energy groups may be Monte Carlo or expectation tallies. However, both cannot be formed in a single problem. Actually the flux tally is a quantity proportional to the flux; the flux can be computed by dividing the flux tally by the region volume.

The Monte Carlo flux tally for a region and energy group is formed by summing the product of the number of neutrons or gamma rays represented by a particle cluster and the average track length of the particle cluster in the region, for all particles in the energy group which enter the region. Forming the Monte Carlo flux tally does involve the computation of expected values to a certain degree. Hence, the tally is not determined in an entirely straight-forward manner. Nevertheless, it is called a Monte Carlo flux tally because the path of the particle, as determined by random sampling, actually crosses the region.

The expectation flux tally for a region and energy group is formed by summing over all particle cluster histories the contributions of a particle cluster in the energy group to the region expectation entrance tally multiplied by the average track length of the particle cluster in the region had it entered the region. Of course, a valid expectation flux tally cannot be expected if the expectation tally is not computed.

The history tally for a region contains the following information about Monte Carlo particles entering the region:

1. Spatial coordinates, Z, X, and Y, of the entry point for each particle.
2. Direction cosines, γ , α , and β , at the entry point for each particle.
3. Energy at the entry point for each particle.
4. Number of collisions undergone by each particle in reaching the region.
5. Number of the region in which each particle originated.
6. Number of the regions in which the first three collisions occurred for each particle.
7. Numbers of the regions for each particle in which the last four collisions occurred before reaching the history tally region.
8. Number of neutrons or gamma rays that each particle represents that reaches the region.

The history tallies are always stored on magnetic tape and are included in the output if desired. For the convenience of any history tally tape integration programs, the order and format of the history tally are:

1. The number of collisions the particle has undergone. A fixed point integer appearing in the address portion of the register.
2. The source region number and the region numbers in which the first three collisions occurred. Four fixed point integers packed into one register with nine bits allotted to each integer. Read from right to left.
3. The region numbers in which the last four collisions occurred. Four fixed point integers packed into one register with nine bits allotted to each integer. Read from right to left.
4. The number of the region in which this history tally was taken. A fixed point integer appearing in the address portion of the register.
5. The number of neutrons or gamma rays that this particle represents, WN_{mc} . A floating point number.
6. The direction cosine, γ . A floating point number.
7. The direction cosine, α . A floating point number.
8. The direction cosine, β . A floating point number.
9. The spatial coordinate, Z . A floating point number.
10. The spatial coordinate, X . A floating point number.
11. The spatial coordinate, Y . A floating point number.
12. The energy, E . A floating point number.

The number of particles in each record, the number of particles in the last record, and the number of records on the history tally tape are included in the output of the main program to relay this information to the user in case it may be desired. An end-of-file mark appears only after the last record. All information is stored consecutively in the records. There are no record identification words at the start of the record. The IBM-704 version has a logical checksum as the last word on each record.

A list of the source particles may be included in the output. If these are desired, the order of the parameters is as follows:

WN_{mc} , γ , α , β , Z , X , Y , and E

WN_{mc} is included to obtain the cluster after acceleration.

9. PROGRAM INSTRUCTIONS

9.1 Input

A data input card contains a total of 80 columns. Column 1 is reserved for control purposes; columns 2-72 are available for input data; and columns 73-80 may be used only for card identification purposes.

Input data may consist of one of two formats: floating point numbers or fixed point integers. The control punch in column 1 indicates which format is to be used for the data on the card. A blank in column 1 indicates the continuation of the previous control punch.

The acceptable control punches are:

- 3 Floating point number format.
- 4 Fixed point integer format. All fixed point integers must have a scaling factor immediately following the control punch. Like the control punch, the scaling factor is continued until changed. For this program all scaling factors are 35.
- 8 End-of-record condition if data follow this control punch. The format of the previous control punch is continued for the data on this card.
- 9 Comment card which is to be ignored by the loading routine.
- = The 3-8 punch which may be used to signify that the data are to be in the floating point format and an end-of-record condition is desired.
- The 4-8 punch which may be used to signify that the data are to be in the fixed point integer format and an end-of-record condition is desired.

Each word of input data and each scaling factor must be terminated by a comma or a blank space. Each word of input data may be preceded by as many blank spaces as desired.

The following general rules should be followed in writing input data:

1. Indicate blank spaces between words of data by encircling the number of blanks desired, e.g., write ACCELERATION FACTORS as ACCELERATION ① FACTORS.
2. Floating point numbers are restricted to eight digits, excluding exponents, for each word of data. Decimal points, plus signs, minus signs, and slashes are not counted as digits.
3. Plus signs may be omitted.

4. Commas indicate the end of an input word and may not be used to indicate thousands, millions, etc., i.e., write 25,604 as 25604.

To eliminate the need of placing restrictions on the lengths of the various tables, input data are stored in consecutive locations immediately following the main program. This feature makes mandatory the preservation of the input order and the insertion of the proper number of values as given in this section of the report. The format of each block of data is specified to inform the user of the necessary control punch. The end-of-record or end-of-file conditions are also specified where they are required. The end-of-record conditions are attached to particular data tables, and the end-of-file conditions are attached to groups of data tables. If the entire data table is to be omitted, as specified in the "DIMENSION AND CONTROL CONSTANTS", the end-of-record condition is also omitted. If the data table is omitted because the control constants immediately preceding the table are zero, the end-of-record condition is required and is obtained by placing an 8 punch in column 1 of the last control card. All end-of-file conditions, except when the source particle's parameters are on magnetic tape, are required. Failure to comply to these conditions will result in error returns and program stops.

To save the laborious writing of zeros to fill in a table which is not used by the program, the symbol, N=, may be used where N is the number of zeros to be inserted. This symbol causes the loading routine to space across N locations before loading the next piece of data. Say, for example, that the fission cross sections are not needed. Let the number of energy groups, NE, be 19 and the number of materials, NM, be 4. Then the fission cross section table requires:

$$NM(2NE+1) = 4(2 \times 19 + 1) = 156$$

words of memory. One of the following two methods may be used to space across this table:

9 (1) FISSION (1) CROSS (1) SECTIONS

3155=,

80₁

or

9 (1) FISSION (1) CROSS (1) SECTIONS

=155=,0,

Note that one zero is supplied on the last card to effect the end-of-record condition that is required for this particular table. If the following were used

9(1)FISSION(1)CROSS(1)SECTIONS
=156=,

the end-of-file condition would result as there are no data on this card.

Blocks of data, which may be omitted, are governed by control constants. The one exception to this rule is the fission cross section table which must be entered even though there are no fissionable materials in the configuration.

Cumulative probability distributions and their corresponding random variables are entered consecutively in pairs in each probability table. The cumulative probability distribution sequence must start with one and end with zero. The probability in each pair immediately precedes the value of its corresponding random variable:

$$P_0 = 1, X_0, P_1, X_1, P_2, X_2, \dots, P_{N-1}, X_{N-1}, P_N = 0, X_N$$

All cumulative probability distribution tables are immediately preceded by a table of control constants, which indicate the optional number of pairs of entries in the tables. Any table that is not needed may be eliminated by equating the appropriate control constant to zero.

9.1.1 FMC-N

DIMENSION AND CONTROL CONSTANTS

Fixed Point Integers

1. Number of energy groups
2. Number of materials
3. Number of compositions
4. Number of boundaries
5. Number of regions
6. Number of outside regions
7. Number of entrance tally regions
8. Number of flux tally regions

Negative sign - tally expectation flux

Positive sign - tally Monte Carlo flux

9. Number of history tally regions
10. Number of regions in which the exponential transformation is used

11. Number of regions in which fine splitting is used
12. Total number of source particles to be started
13. Number of particles to be started in a cluster for each source particle, N_{mc}
14. Number of collisions to be considered after the particle's energy has fallen below the cutoff energy, E_0
15. Number of iterations to be performed on the random number sequence before obtaining the first random number to be used by the program.
16. Variable random number. If program number is satisfactory, enter 1=. See Appendix I.
17. Random number multiplier. If program number is satisfactory, enter 1=. See Appendix I.
18. The maximum number of particles that may appear in a cluster, considering all types of splitting, accelerating, and N_{mc} . (It may be wise to include several extra particles to account for particles surviving Russian roulette.)
19. Number of words required to describe all boundaries
20. Number of words required to describe all regions
21. Number of words required to describe all cumulative probability distribution tables excluding the source particle generator data.
22. 0 - Pick the cosine of the center-of-mass scattering angle from cumulative probability distribution tables.
1 - Compute the cosine of the center-of-mass scattering angle from a second-degree cosine power series.
- 23: 0 - Do not generate secondary gamma rays
1 - Generate secondary gamma rays
24. 0 - Do not print source particle parameters
1 - Print source particle parameters
25. 0 - Do not print history tallies
1 - Print history tallies
26. 0 - Compute absorption tallies
1 - Compute energy deposition tallies
27. 0 - No fissionable material is present in the configuration
1 - Fissionable material is present in the configuration
28. 0 - Do not compute expectation tallies
1 - Compute expectation tallies
29. +N - Number of fission neutron generations to process before storing the remaining fission neutrons on magnetic tape. Print the accumulated output for each fission neutron generation.
-N - Number of fission neutron generations to process before storing the remaining fission neutrons on magnetic tape. Print the output only at the completion of the run.

- 30. 0 - Select the source particle parameters from the appropriate cumulative probability distribution tables.
- 1 - The source particle parameters are available in card input.
- N - The number of records of source particle parameters on magnetic tape.
- 31. Number of particles, represented by the seven parameters, in each record on magnetic tape.
- 32. Number of particles, represented by the seven parameters, in the last record on magnetic tape.

Floating Point Numbers

- 33. A small number, δ , used in the geometry calculations to avoid roundoff errors.
- 34. A small number, ϵ , used to equate smaller quantities to zero.
- 35. Cutoff weight, W_0
- 36. Cutoff energy, E_0

An end-of-record condition is required.

SOURCE PARTICLE GENERATOR INPUT

If the source particle parameters are available on magnetic tape, omit all source particle generator input. No end-of-record or end-of-file conditions are permitted.

If the source particle parameters are available as card input, enter the seven parameters for each particle on a single card. The parameters must be in the floating point number format and in the following order:

γ , α , β , Z, X, Y, and E

An end-of-file condition is required.

If the source particle parameters are to be selected from the appropriate cumulative probability distribution tables, the following input must be supplied:

Fixed Point Integers

- 1. Number of particles, represented by the seven parameters, to be generated for this source region.
- 2. The method desired for the generation of the source particle's direction cosines.
- 3. The method desired for the generation of the source particle's spatial coordinates.
- 4. The method desired for the generation of the source particle's energy.

5. The number of combined cumulative probability distribution and random variable entries in the first table for the particle's direction cosines. Enter 0 if not used.
6. The number of combined cumulative probability distribution and random variable entries in the second table for the particle's direction cosines. Enter 0 if not used.
7. The number of combined cumulative probability distribution and random variable entries in the first table for the particle's spatial coordinates. Enter 0 if not used.
8. The number of combined cumulative probability distribution and random variable entries in the second table for the particle's spatial coordinates. Enter 0 if not used.
9. The number of combined cumulative probability distribution and random variable entries in the third table for the particle's spatial coordinates. Enter 0 if not used.
10. The number of combined cumulative probability distribution and random variable entries in the table for the particle's energy. Enter 0 if not used.

Floating Point Numbers

11. Translation of the source region with respect to the X-axis, a_0 .
12. Translation of the source region with respect to the Y-axis, b_0 .
13. Translation of the source region with respect to the Z-axis, c_0 .
14. Start of the combined cumulative probability distribution and random variable tables. All tables are entered consecutively.

The required number of cumulative probability distribution tables for the direction cosines are entered first; the required number of cumulative probability distribution tables for the spatial coordinates are entered next; and the energy distribution table is entered last.

If any of the source particle's parameters are supplied as input, they immediately follow the above data. All seven parameters may be entered through this routine if enough accuracy cannot be retained entering the parameters of each particle on a single card. Again the ordering of these parameters is γ , α , β , Z , X , Y , and E . Any of the parameters which are selected by the routine must be omitted from this sequence, and no space may be allotted to them. If any of the parameters are entered in this manner, they must be entered for each source particle in the source region.

An end-of-record condition is required.

All of the above source particle generator input must be repeated for each source region desired.

An end-of-file condition is required.

ENERGY GROUP BOUNDS

Floating Point Numbers

Energies which define the bounds of the energy groups are entered consecutively by starting with the lowest energy. One value more than the number of energy groups must be entered. The energy groups may be equal or unequal, as desired, but they must be constant for all data for the problem.

ENERGY IMPORTANCE CONSTANTS, I_i

Floating Point Numbers

The energy importance constants are entered for each energy group in order of increasing energy. The first energy importance constant is for energy group zero and is to be used when the particle's energy falls below the cutoff energy. This constant is required whether or not a number-of-collisions cutoff is applied. Therefore, there must be one more energy importance constant than there are energy groups.

ACCELERATION FACTORS, A_i

Floating Point Numbers

The acceleration factors are entered for each energy group in order of increasing energy. The first acceleration factor is for energy group zero and is to be used when the source particle's energy is less than the cutoff energy. This constant is required whether or not a number-of-collisions cutoff is applied. Therefore, there must be one more acceleration factor than there are energy groups.

An end-of-file condition is required.

REGION BOUNDARY DESCRIPTION

Floating Point Numbers

1. The number of the boundary position in the boundary table.
2. The code number, c_1 , for the type of boundary.
3. The coefficients needed to describe the locus of the boundary.
(See below for coefficients and ordering.)

The above quantities are entered consecutively for all the boundaries in the configuration.

An end-of-record condition is required.

The following list defines the boundary code number, c_1 , and specifies the coefficients and the ordering of the coefficients necessary to describe the boundaries.

$$c_1=1 \quad AX^2 + X_0X + BY^2 + Y_0Y + CZ^2 + Z_0Z - K = 0$$

Input: $X_0, Y_0, Z_0, A, B, C, K$

$$c_1=2 \quad A(X-X_0)^2 + B(Y-Y_0)^2 + C(Z-Z_0)^2 - K = 0$$

Input: $X_0, Y_0, Z_0, A, B, C, K$

$$c_1=3 \quad (X-X_0)^2 + (Y-Y_0)^2 - K = 0$$

Input: X_0, Y_0, K

$$c_1=4 \quad X - K = 0$$

Input: K

$$c_1=5 \quad Y - K = 0$$

Input: K

$$c_1=6 \quad Z - K = 0$$

Input: K

REGION DESCRIPTION CODES

Fixed Point Integers

1. The number of the region position in the region table. Outside regions must be listed last. A negative sign indicates the region is a history tally region.
2. The number of boundaries required to completely enclose the region. A negative sign indicates the region is a flux tally region.
3. The number of the composition of the region. If the composition of the region is void, enter zero. A negative sign indicates the region is an outside region and a leakage tally is taken.
4. A code number 0-17 to indicate the type of exponential transformation or type of fine splitting to be performed in the region. The code number definitions are:

<u>Code Number</u>	<u>Definition</u>
$D = 0$	No exponential transformation or fine splitting is to be performed in this region.
$D = 1$	$k = 1$

<u>Code Number</u>	<u>Definition</u>
D = 2	$k = \alpha$
D = 3	$k = -\alpha$
D = 4	$k = \beta$
D = 5	$k = -\beta$
D = 6	$k = \gamma$
D = 7	$k = -\gamma$
D = 8	$d = X_n - X$
D = 9	$d = X - X_n$
D = 10	$d = Y_n - Y$
D = 11	$d = Y - Y_n$
D = 12	$d = Z_n - Z$
D = 13	$d = Z - Z_n$
D = 14	$d = [(X_n - X_0)^2 + (Y_n - Y_0)^2] - [X - X_0]^2 + (Y - Y_0)^2]$
D = 15	$d = [(X - X_0)^2 + (Y - Y_0)^2] - [(X_n - X_0)^2 + (Y_n - Y_0)^2]$
D = 16	$d = [(X_n - X_0)^2 + (Y_n - Y_0)^2 + (Z_n - Z_0)^2] - [(X - X_0)^2 + (Y - Y_0)^2 + (Z - Z_0)^2]$
D = 17	$d = [(X - X_0)^2 + (Y - Y_0)^2 + (Z - Z_0)^2] - [(X_n - X_0)^2 + (Y_n - Y_0)^2 + (Z_n - Z_0)^2]$

A negative sign indicates the region is an entrance tally region.

5. The number of the boundary position of this region in the boundary table. The sign indicates the value of j for this boundary of this region.
6. The number of the region that the particle will most likely enter if it passes through this boundary.
7. Steps 5 and 6 must be repeated for each boundary required to enclose the region.
8. Steps 1 through 7 must be repeated for each region in the configuration. Outer boundaries for outside regions may be omitted.

An end-of-record condition is required.

REGION IMPORTANCE CONSTANTS, I_r

Floating Point Number

The region importance constants are entered for each inside region in order of region numbering. No region importance constants may be entered for outside regions.

The following table must be omitted if the exponential transformation is not used in the configuration.

REGION MEAN FREE PATH LENGTHS, λ^r

Floating Point Numbers

The region mean free paths are entered for each region in which the exponential transformation is used in order of the region numbering. No region mean free paths may be entered for outside regions.

An end-of-file condition is required.

MATERIAL VOLUME FRACTIONS, θ_m^h

Floating Point Numbers

The material volume fractions are entered first for each material and then for each composition.

NUCLEI PER UNIT VOLUME, N_m

Floating Point Numbers

The number of nuclei per unit volume are entered for each material in sequence.

NUCLEAR MASS, A_m

Floating Point Numbers

The nuclear masses are entered for each material in sequence.

MINIMUM ENERGY FOR ANISOTROPIC SCATTERING, E_{mc}

Floating Point Numbers

The minimum energies for anisotropic neutron scattering are entered first for each material for elastic scattering and then for each material for inelastic scattering.

An end-of-file condition is required.

MICROSCOPIC ELASTIC SCATTERING CROSS SECTIONS, $\sigma_{ml}(E)$

Floating Point Numbers

The microscopic elastic scattering cross sections are entered first for each material and then for each energy bound in order of increasing energy. The first microscopic cross section for each material is the average

microscopic cross section for energy group zero and is to be used when the particle's energy falls below the cutoff energy. These microscopic cross sections for each material are required whether or not a number-of-collisions cutoff is applied.

The microscopic cross sections from the cutoff energy to the maximum energy must be entered for both the lower and upper energy bounds of each energy group to form straight line segments across each energy group. For each energy bound other than the first and last, there must be two entries for each event. This allows discontinuities to exist at the energy bounds.

The microscopic cross sections table will appear as:

$$\begin{aligned}
 &\sigma_{1c}(E_0), \sigma_{2c}(E_0), \sigma_{3c}(E_0), \dots, \sigma_{mc}(E_0), \\
 &\sigma_{1c}(E_{1l}), \sigma_{2c}(E_{1l}), \sigma_{3c}(E_{1l}), \dots, \sigma_{mc}(E_{1l}), \\
 &\sigma_{1c}(E_{1u}), \sigma_{2c}(E_{1u}), \sigma_{3c}(E_{1u}), \dots, \sigma_{mc}(E_{1u}), \\
 &\sigma_{1c}(E_{2l}), \sigma_{2c}(E_{2l}), \sigma_{3c}(E_{2l}), \dots, \sigma_{mc}(E_{2l}), \\
 &\sigma_{1c}(E_{2u}), \sigma_{2c}(E_{2u}), \sigma_{3c}(E_{2u}), \dots, \sigma_{mc}(E_{2u}), \\
 &\vdots \\
 &\sigma_{1c}(E_{nl}), \sigma_{2c}(E_{nl}), \sigma_{3c}(E_{nl}), \dots, \sigma_{mc}(E_{nl}), \\
 &\sigma_{1c}(E_{nu}), \sigma_{2c}(E_{nu}), \sigma_{3c}(E_{nu}), \dots, \sigma_{mc}(E_{nu}),
 \end{aligned}$$

An end-of-record condition is required.

MICROSCOPIC INELASTIC SCATTERING CROSS SECTIONS, $\sigma_{m2}(E)$

Floating Point Numbers

The microscopic inelastic scattering cross sections are entered in the same manner as the microscopic elastic scattering cross sections.

An end-of-record condition is required.

MICROSCOPIC FISSION CROSS SECTIONS, $\sigma_{m3}(E)$

Floating Point Numbers

The microscopic fission cross sections are entered in the same manner as the microscopic elastic scattering cross sections.

An end-of-record condition is required.

MICROSCOPIC TOTAL CROSS SECTIONS, $\sigma_{m4}(E)$

Floating Point Numbers

The microscopic total cross sections are entered in the same manner as the microscopic elastic scattering cross sections.

An end-of-record condition is required.

If the angular distributions of neutrons, after elastic and inelastic scattering, are to be determined from cumulative neutron angular distribution probability tables, the following four tables must be omitted.

COEFFICIENT $mc_{a1}(E)$ FOR ELASTIC SCATTERING

Floating Point Numbers

The values of the coefficients, $mc_{a1}(E)$, of the second-degree cosine power series for computing the angular distributions of neutrons elastically scattered are entered first for each material and then for each energy bound in order of increasing energy. A coefficient for each material must be entered for each energy bound to form jointed straight-line segments across each energy group. The initial entry for each material is used when the particle's energy falls below the cutoff energy.

The coefficient table will appear as:

$$\begin{aligned} &1c_{a1}(E_0), 2c_{a1}(E_0), 3c_{a1}(E_0), \dots, mc_{a1}(E_0), \\ &1c_{a1}(E_1), 2c_{a1}(E_1), 3c_{a1}(E_1), \dots, mc_{a1}(E_1), \\ &1c_{a1}(E_2), 2c_{a1}(E_2), 3c_{a1}(E_2), \dots, mc_{a1}(E_2), \\ &\vdots \\ &1c_{a1}(E_n), 2c_{a1}(E_n), 3c_{a1}(E_n), \dots, mc_{a1}(E_n), \end{aligned}$$

An end-of-record condition is required.

COEFFICIENT $mc_{a2}(E)$ FOR ELASTIC SCATTERING

Floating Point Numbers

The values of the coefficients, $mc_{a2}(E)$, of the second-degree cosine power series for computing the angular distributions of neutrons elastically scattered are entered in the same manner as the value of the coefficients, $mc_{a1}(E)$, for elastic scattering.

An end-of-record condition is required.

COEFFICIENT $mc_{a1}(E)$ FOR INELASTIC SCATTERING

Floating Point Numbers

The values of the coefficients, $mc_{a1}(E)$ of the second-degree cosine power series for computing the angular distributions of neutrons inelastically scattered are entered in the same manner as the values of the coefficients, $mc_{a1}(E)$, for elastic scattering.

An end-of-record condition is required.

COEFFICIENT $mc_{a2}(E)$ FOR INELASTIC SCATTERING

Floating Point Numbers

The values of the coefficients, $mc_{a2}(E)$, of the second-degree cosine power series for computing the angular distributions of neutrons inelastically scattered are entered in the same manner as the values of the coefficients, $mc_{a1}(E)$, for elastic scattering.

An end-of-record condition is required.

If the angular distributions of neutrons, after elastic and inelastic scattering, are to be computed from a second-degree cosine power series, the following four tables must be omitted.

ELASTICALLY-SCATTERED NEUTRON ANGULAR DISTRIBUTION PROBABILITY TABLE CONTROL CONSTANTS

Fixed Point Integer

The elastically-scattered neutron angular distribution probability table control constants have the number of combined cumulative probability and random variable entries in the probability tables which follow immediately for each material and energy group. The control constants are entered first for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

ELASTICALLY-SCATTERED NEUTRON ANGULAR DISTRIBUTION PROBABILITY TABLES

Floating Point Numbers

The combined cumulative neutron angular distribution probabilities and their corresponding random variables are entered first for each material and then for each energy group in order of increasing energy. The first

value for each material is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

INELASTICALLY-SCATTERED NEUTRON ANGULAR DISTRIBUTION PROBABILITY TABLE CONTROL CONSTANTS

Fixed Point Integers

The inelastically-scattered neutron angular distribution probability table control constants give the number of combined cumulative probability and random variable entries in the probability tables which follow immediately for each material and energy group. The control constants are entered first for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

INELASTICALLY-SCATTERED NEUTRON ANGULAR DISTRIBUTION PROBABILITY TABLES

Floating Point Numbers

The combined cumulative neutron angular distribution probabilities and their corresponding random variables are entered first for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

INELASTICALLY-SCATTERED NEUTRON SPECTRUM PROBABILITY TABLE CONTROL CONSTANTS

Fixed Point Integers

The inelastically-scattered neutron spectrum probability table control constants give the number of combined cumulative probability and random variable entries in the probability tables which follow immediately for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

INELASTICALLY-SCATTERED NEUTRON SPECTRUM PROBABILITY TABLES

Floating Point Numbers

If $A_m \geq 20$, the combined cumulative neutron spectrum probabilities and their corresponding random variables are entered. If $A_m < 20$, the combined

cumulative nuclear excitation probabilities and their corresponding random variables are entered. The tables are intermixed consistent with the ordering of the materials. They are entered first for each material and then for each energy group in order of increasing energy. The first table for each material is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

CONVERSION FACTORS, D_{ir}

Floating Point Numbers

The conversion factors for converting absorption tallies to any desired quantities are entered first for each inside region and then for each energy group in order of increasing energy. The first conversion factor for each inside region is for energy group zero, to be used when the particle's energy falls below the cutoff energy. These conversion factors are required whether or not a number-of-collisions cutoff is applied. Equating this conversion factor to zero will suppress the absorption tally printout for this region.

An end-of-record condition is required.

If there are no fine splitting regions in the configuration, the following table must be omitted.

FINE SPLITTING CONSTANTS, c_{ir}

Floating Point Numbers

The fine splitting constants are entered first for each fine splitting region and then for each energy group in order of increasing energy. The first fine splitting constant for each fine splitting region is for energy group zero, to be used when the particle's energy falls below the cutoff energy. These fine splitting constants are required whether or not a number-of-collisions cutoff is applied.

An end-of-record condition is required.

If there is no fissionable material in the configuration, the following four tables must be omitted.

If there is fissionable material in the configuration and the absorption tally is desired, the first table must be omitted.

FISSION ENERGY DEPOSITION CONSTANTS, H_m

Floating Point Numbers

A fission energy deposition constant is entered for each material in sequence.

NEUTRONS EMITTED PER FISSION, ν_{mi}

Floating Point Numbers

The number of neutrons emitted per fission are entered first for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

FISSION NEUTRON SPECTRUM PROBABILITY TABLE CONTROL CONSTANTS

Fixed Point Integers

The fission neutron spectrum probability table control constants give the number of combined cumulative probability and random variable entries in the probability tables which follow immediately for each material and energy group. The control constants are entered first for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

FISSION NEUTRON SPECTRUM PROBABILITY TABLES

Floating Point Numbers

The combined cumulative fission neutron spectrum probabilities and their corresponding random variables are entered first for each material and then for each energy group in order of increasing energy. The first table for each material is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

If no secondary gamma rays are to be generated, the following nine tables must be omitted. If secondary gamma rays are to be generated, all nine tables must be included.

GAMMA RAYS EMITTED PER NEUTRON INELASTIC SCATTER, N_{m2i}

Floating Point Numbers

The number of gamma rays emitted per neutron inelastic scatter are entered first for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

GAMMA RAYS EMITTED PER NEUTRON FISSION, N_{m3i}

Floating Point Numbers

The number of gamma rays emitted per neutron fission are entered first for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

GAMMA RAYS EMITTED PER NEUTRON ABSORPTION

Floating Point Numbers

The number of gamma rays emitted per neutron absorption are entered first for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

NEUTRON INELASTIC SCATTERING GAMMA RAY SPECTRUM PROBABILITY TABLE CONTROL CONSTANTS

Fixed Point Integers

The control constants in this table give the number of combined cumulative probability tables for determining the spectra of gamma rays which result from inelastic scattering of neutrons. The control constants are entered first for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

NEUTRON INELASTIC SCATTERING GAMMA RAY SPECTRUM PROBABILITY TABLES

Floating Point Numbers

The combined cumulative probabilities for determining the spectra of gamma rays, which result from inelastic scattering of neutrons, and their corresponding random variables are entered first for each material and then for each energy group in order of increasing energy. The first table for each material is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

NEUTRON FISSION GAMMA RAY SPECTRUM PROBABILITY TABLE CONTROL CONSTANTS

Fixed Point Integers

The control constants in this table give the number of combined cumulative probability and random variable entries in the following probability tables for determining the spectra of gamma rays which result from neutron fission. The control constants are entered first for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

NEUTRON FISSION GAMMA RAY SPECTRUM PROBABILITY TABLES

Floating Point Numbers

The combined cumulative probabilities for determining the spectra of gamma rays, which result from neutron fission, and their corresponding random variables are entered first for each material and then for each energy group in order of increasing energy. The first table for each material is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

NEUTRON ABSORPTION GAMMA RAY SPECTRUM PROBABILITY TABLE CONTROL CONSTANTS

Fixed Point Integers

The control constants in this table give the number of combined cumulative probability and random variable entries in the following probability tables for determining the spectra of gamma rays which result from neutron absorption. The control constants are entered first for each material and

then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

NEUTRON ABSORPTION GAMMA RAY SPECTRUM PROBABILITY TABLES

Floating Point Numbers

The combined cumulative probabilities for determining the spectra of gamma rays which result from neutron absorption. The control constants are entered first for each material and then for each energy group in order of increasing energy. The first value for each material is used when the particle's energy falls below the cutoff energy.

NEUTRON ABSORPTION GAMMA RAY SPECTRUM PROBABILITY TABLES

Floating Point Numbers

The combined cumulative probabilities for determining the spectra of gamma rays, which result from neutron absorption, and their corresponding random variables are entered first for each material and then for each energy group in order of increasing energy. The first table for each material is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

An end-of-file condition is required.

9.1.2 FMC-G

DIMENSION AND CONTROL CONSTANTS

Fixed Point Integers

1. Number of energy groups
2. Number of materials
3. Number of compositions
4. Number of boundaries
5. Number of regions
6. Number of outside regions
7. Number of entrance tally regions
8. Number of flux tally regions
 - Negative sign - tally expectation flux
 - Positive sign - tally Monte Carlo flux
9. Number of history tally regions

10. Number of regions in which the exponential transformation is used.
11. Number of regions in which fine splitting is used.
12. Total number of source particles to be started.
13. Number of particles to be started in a cluster for each source particle, N_{mc} .
14. Number of collisions to be considered after the particle's energy has fallen below the cutoff energy, E_0 .
15. Number of iterations to be performed on the random number sequence before obtaining the first random number to be used by the program.
16. Variable random number. If program number is satisfactory, enter 1=. See Appendix I.
17. Random number multiplier. If program number is satisfactory, enter 1=. See Appendix I.
18. The maximum number of particles that may appear in a cluster, considering all types of splitting, accelerating, and N_{mc} . (It may be wise to include several extra particles to account for particles surviving Russian roulette.)
19. Number of words required to describe all boundaries.
20. Number of words required to describe all regions.
21. Number of words required to describe all cumulative probability distribution tables, excluding the source particle generator data.
22. 0 - Do not generate 500 kev gamma rays resulting from pair production
1 - Generate 500 kev gamma rays resulting from pair production
23. 0 - Do not print source particles
1 - Print source particles
24. 0 - Do not print history tallies
1 - Print history tallies
25. 0 - Compute absorption tallies
1 - Compute energy deposition tallies
26. 0 - Do not compute expectation tallies
1 - Compute expectation tallies
27. 0 - Select the source particle parameters from the appropriate cumulative probability distribution tables.
-1 - The source particle parameters are available on card input.
N - The number of records of source particle parameters on magnetic tape.
28. Number of particles, represented by the seven parameters, in each record on magnetic tape.
29. Number of particles, represented by the seven parameters, in the last record on magnetic tape.

Floating Point Numbers

30. A small number, δ , used in the geometry calculations to avoid round-off errors.
31. A small number, ϵ , used to equate smaller quantities to zero.

32. Cutoff weight, W_0

33. Cutoff energy, E_0

An end-of-record condition is required.

SOURCE PARTICLE GENERATOR INPUT

If the source particle parameters are available on magnetic tape, omit all source particle generator input. No end-of-record or end-of-file conditions are permitted.

If the source particle parameters are available as card input, enter the seven parameters for each particle on a single card. The parameters must be in the floating point number format and in the following order:

γ , α , β , Z , X , Y , and E

An end-of-file condition is required.

If the source particle parameters are to be selected from the appropriate cumulative probability distribution tables, the source particle generator input is entered in the same manner as for the neutron problem.

An end-of-record condition is required.

All the source particle generator input must be repeated for each source region desired.

An end-of-file condition is required.

ENERGY GROUP BOUNDS

Floating Point Numbers

The energies which define the bounds of the energy groups are entered consecutively starting with the lowest energy. One value more than the number of energy groups must be entered. The energy groups may be equal or unequal, as desired, but they must be constant for all data for the problem.

ENERGY IMPORTANCE CONSTANTS, I_i

Floating Point Numbers

The energy importance constants are entered for each energy group in order of increasing energy. The first energy importance constant is for energy group zero and is to be used when the particle's energy falls below the cutoff energy. This constant is required whether or not a number-of-collisions cutoff is applied. Therefore, there must be one more energy importance constant than there are energy groups.

ACCELERATION FACTORS, A_i

Floating Point Numbers

The acceleration factors are entered for each energy group in order of increasing energy. The first acceleration factor is for energy group zero and is to be used when the source particle's energy is less than the cutoff energy. This constant is required whether or not a number-of-collisions cutoff is applied. Therefore, there must be one more acceleration factor than there are energy groups.

An end-of-file condition is required.

REGION BOUNDARY DESCRIPTION

Floating Point Numbers

The region boundary description data are entered in the same manner as for the neutron problem.

An end-of-record condition is required.

REGION DESCRIPTION CODES

Fixed Point Integers

The region description codes are entered in the same manner as for the neutron problem.

An end-of-record condition is required.

REGION IMPORTANCE CONSTANTS, I_r

Floating Point Numbers

The region importance constants are entered for each inside region in order of region numbering. No region importance constants may be entered for outside regions.

The following table must be omitted if the exponential transformation is not used in the configuration.

REGION MEAN FREE PATH LENGTHS, λ^r

Floating Point Numbers

The region mean free paths are entered for each region in which the exponential transformation is used in order of the region numbering. No

region mean free paths may be entered for outside regions.

An end-of-file condition is required.

MATERIAL VOLUME FRACTIONS, θ_m^h

Floating Point Numbers

The material volume fractions are entered first for each material and then for each composition.

NUCLEI PER UNIT VOLUME, N_m

Floating Point Numbers

The number of nuclei per unit volume are entered for each material in sequence.

NUMBER OF ELECTRONS PER NUCLEUS, Z_m

Floating Point Numbers

The number of electrons per nucleus are entered for each material in sequence.

An end-of-file condition is required.

MICROSCOPIC COMPTON SCATTERING CROSS SECTIONS, $\sigma_5(E)$

Floating Point Numbers

The microscopic Compton scattering cross sections are entered for each energy bound in order of increasing energy. The first microscopic cross section is the average microscopic cross section for energy group zero and is to be used when the particle's energy falls below the cutoff energy. This microscopic cross section is required whether or not a number-of-collisions cutoff is applied.

The microscopic cross sections from the cutoff energy to the maximum energy must be entered for both the lower and upper energy bounds of each energy group to form straight-line segments across each energy group. There must be two entries for each energy bound other than the first and the last energy bounds. This allows discontinuities to exist at the energy bounds.

The microscopic Compton scattering cross section table will appear as:

$\sigma_5(E_0), \sigma_5(E_{1l}), \sigma_5(E_{1u}), \sigma_5(E_{2l}), \sigma_5(E_{2u}), \dots, \sigma_5(E_{nl}), \sigma_5(E_{nu})$

An end-of-record condition is required.

MICROSCOPIC PHOTOELECTRIC EFFECT CROSS SECTIONS, $\sigma_{m6}(E)$

Floating Point Numbers

The microscopic photoelectric effect cross sections are entered in the same manner as the microscopic elastic scattering cross sections in the neutron problem.

An end-of-record condition is required.

MICROSCOPIC PAIR PRODUCTION CROSS SECTIONS, $\sigma_{m7}(E)$

Floating Point Numbers

The microscopic pair production cross sections are entered in the same manner as the microscopic elastic scattering cross sections in the neutron problem.

An end-of-record condition is required.

ANGULAR DISTRIBUTION PROBABILITY TABLE CONTROL CONSTANTS

Fixed Point Integers

The angular distribution probability table control constants give the number of combined cumulative probability and random variable entries in the probability tables which follow immediately for each energy group. The control constants are entered for each energy group in order of increasing energy. The first value is used when the particle's energy falls below the cutoff energy.

ANGULAR DISTRIBUTION PROBABILITY TABLES

Floating Point Numbers

The combined cumulative gamma ray angular distribution probabilities and their corresponding random variables are entered for each energy group in order of increasing energy. The first table is used when the particle's energy falls below the cutoff energy.

An end-of-record condition is required.

CONVERSION FACTORS, D_{1r}

Floating Point Numbers

The conversion factors for converting absorption tallies to any desired quantities are entered first for each inside region and then for each energy

group in order of increasing energy. The first conversion factor is for energy group zero and is to be used when the particle's energy falls below the cutoff energy. These conversion factors are required whether or not a number-of-collisions cutoff is applied. Equating this conversion factor to zero will suppress the absorption tally printout for this region.

An end-of-record condition is required.

If there are no fine splitting regions in the configuration, the following table must be omitted.

FINE SPLITTING CONSTANTS, c_{ir}

Floating Point Numbers

The fine splitting constants are entered first for each fine splitting region and then for each energy group in order of increasing energy. The first fine splitting constant for each fine splitting region is for energy group zero and is to be used when the particle's energy falls below the cutoff energy. These fine splitting constants are required whether or not a number-of-collisions cutoff is applied.

An end-of-record condition is required.

An end-of-file condition is required.

9.2 Operation

9.2.1 Deck Makeup

1. Preparation routine binary deck
2. Input data
3. Main program binary deck

9.2.2 Procedure

Sense switch convention

No. 1	Down - Program on cards loaded through the card reader
	Up - Program on magnetic tape
No. 2	Down - Input on cards loaded through the card reader
	Up - Input on magnetic tape
No. 3	Down - Output on "on-line" printer
	Up - Output on magnetic tape

Depressing sense switch five during the operation of the program will cause an "on-line" printout of the number of particles in born storage and the number of source particles left to be processed. The program will pause and wait for appropriate action to be taken. Depressing the start button will cause the program to continue in its normal sequence of events. A transfer to the output routine will cause the program to print the present tallies and prepare magnetic tapes for the appropriate quantities. The source particles remaining will be lost.

Tape units for the IBM-704 program:

- No. 1 System and package tape - user package PK1
- No. 2 Input tape - program and input decks
- No. 3 Output tape
- No. 4 Not used by the program - Production library tape
- No. 5 Source particle overflow tape
- No. 6 Born storage overflow tape
- No. 7 Input source particle tape - neutron fission particle overflow tape
- No. 8 Not used - Nuclear data tape
- No. 9 History tally tape
- No. 10 Secondary gamma ray tape

Tape units for the IBM-7090 program using data channels A and B. The "on-line" printer and card reader are connected to data channel A.

- A-1 System tape
- A-2 Input tape - program and input decks
- A-3 Output tape
- A-4 Input source particle tape - neutron fission particle overflow tape
- A-5 Not used - Nuclear data tape
- A-6 Born storage overflow tape
- B-1 Secondary gamma ray tape
- B-2 Not used
- B-3 History tally tape
- B-4 Not used - punch tape
- B-5 Source particle overflow tape
- B-6 Library and package tape - use package PK1

The programs are coded so that no limitations are placed on the sizes of the various input quantities. However, if the input data and output storage requirements, for which no magnetic tape unit is reserved, exceeds the 32,768 core memory, the program will stop and print the comment "PROGRAM TOO LARGE". The user will then have to condense his problem in some manner to run his problem.

10. SAMPLE PROBLEM

The sample problem describes a fictitious configuration and composition. The purpose of the problem is to exemplify input description.

Figure 4 illustrates the source-shield configuration. The rectangles shown are revolved about the Z-axis forming a cylindrical configuration. The Roman numerals are the region numbers and the Arabic numerals are the boundary numbers. Boundaries 1, 2, 3, 4, 5, 6, 7, and 8 form planes perpendicular to the Z-axis. Boundaries 9, 10, 11, and 12 form cylinders with the Z-axis as the origin. Boundaries 1, 8, and 12 form the outer surface of the configuration. Regions XI, XII, and XIII are outside regions which are required for leakage tallies and termination of particles escaping the system.

Region I contains fifty per cent of material two and fifty per cent of material three. Regions II, III, and VIII contain one hundred per cent of material one. Region IV contains one hundred per cent of material three. Regions V and IX contain one hundred per cent of material four. Regions VI, VII, and X contain one hundred per cent of material two. The composition of the outside regions XI, XII, and XIII is never used, but is required for the spacing of the input in the region description codes.

The source particle's spatial coordinates were picked from a point source located at the origin of the configuration. The angular distribution of the source particles was assumed isotropic in the laboratory system, and the source spectrum was uniformly distributed between five and fifteen Mev.

Secondary gamma rays, which were generated throughout the configuration in the neutron problem, were used as source particles for the gamma ray problem.

No exponential transformation or splitting were used in either problem.

Figure 4

10.1 Input

1 2

73

88

9 DIMENSION AND CONTROL CONSTANTS

435 19 4 5 12 13 3 9 10 1 0 0 500 2 3 1
 1= 1= 2 50 130 1200 0 1 0 1 0 0 1 0 0 0 0
 -1/-4 1/-4 .1 .1

DCC1
 DCC2
 DCC3

SOURCE GENERATOR DATA

435 500 3 1 1 0 0 2 2 2 2
 30 0 0
 1 0 0 0
 1 0 0 0
 1 0 0 0
 81 15 0 5

SGD1
 SGD2
 SGD3
 SGD4
 SGD5
 SGD6
 EOF

9 ENERGY BOUNDS

30.1000 0.1302 0.1695 0.2206 0.2872 0.3738 0.4866
 0.6335 0.8246 1.0734 1.3974 1.8190 2.3680 3.0825
 4.0127 5.2236 6.7998 8.8517 11.523 15

EB01
 EB02
 EB03

9 ENERGY IMPORTANCES

31 1 1 1 1 1 1 1
 1 1 1 1 1 1 1
 1 1 1 1

EI01
 EI02
 EI03

9 ACCELERATION FACTORS

31 1 1 1 1 1 1 1
 1 1 1 1 1 1 1
 1 1 1 1

AF01
 AF02
 AF03
 EOF

9 BOUNDARY DESCRIPTION

301 6 -5
 02 6 -4
 03 6 -3
 04 6 -2
 05 6 +2
 06 6 +3
 07 6 +4
 08 6 +6
 09 3 +0 +0 04
 10 3 +0 +0 09
 11 3 +0 +0 16
 812 3 +0 +0 25

BD01
 BD02
 BD03
 BD04
 BD05
 BD06
 BD07
 BD08
 BD09
 BD10
 BD11
 BD12

9 REGION DESCRIPTION

435 +01 -3 +5 +0 -04 08 +05 02 +09 03
 02 -3 +1 -0 -05 01 +06 05 +09 03
 03 -4 +1 -0 -03 09 +06 05 -09 01 +10 04
 04 -4 +3 -0 -02 10 +06 05 -10 03 +11 07
 05 -3 +4 -0 -06 02 +07 06 +11 07
 06 -3 +2 -0 -07 05 +08 13 +11 07
 07 -4 +2 -0 -01 11 +08 13 -11 04 +12 12
 08 -3 +1 -0 -03 09 +04 01 +09 03
 09 -3 +4 -0 -02 10 +03 08 +10 04
 10 -3 +2 -0 -01 11 +02 09 +11 07
 11 +1 -1 +0 +01 02
 12 +1 -1 +0 +12 07
 8 -13 +1 -1 +0 -08 06

RD01
 RD02
 RD03
 RD04
 RD05
 RD06
 RD07
 RD08
 RD09
 RD10
 RD11
 RD12
 RD13

9 REGION IMPORTANCES

31 1 1 1 1 1 1 1
 1 1

RI01
 RI02

FOF

MC01
MC02
MC03
MC04
MC05

NPV1

AMN1

ASL1
EOF

EXE00
EXE01
EXE02
EXE03
EXE04
EXE05
EXE06
EXE07
EXE08
EXE09
EXE10
EXE11
EXE12
EXE13
EXE14
EXE15
EXE16
EXE17
EXE18
EXE19

IXE00-03
IXE06
IXE07
IXE08
IXE09
IXE10
IXE11
IXE12
IXE13
IXE14
IXE15
IXE16
IXE17
IXE18
IXE19

FXE00-19

TXE00

8
9 MATERIAL COMPOSITION TABLE. BE,C,FE,PB.

31 0 0 0
0 0 0
0 1 0
0 0 0 1
0 .5 .5 0

9 NUCLEI PER UNIT VOLUME. BE,C,FE,PB.

3.1202 .1128 .0849 .0330

9 ATOMIC MASS NUMBERS. BE,C,FE,PB.

39.013 12.01 55.85 207.21

9-ANISOTROPIC SCATTERING LIMITS. ELASTIC - INELASTIC.

31.0734 1.0734 1.0734 1.0734 15.1 15.1 15.1 15.1

8
9 ELASTIC CROSS SECTIONS. BE,C,FE,PB.

35.4000 4.5500 4.0000 11.000
5.4000 4.5500 4.0000 11.000 5.3095 4.4444 4.1006 10.336
5.3095 4.4444 4.1006 10.323 5.1916 4.3069 4.2315 9.5105
5.2289 4.3069 4.2315 9.5238 4.9263 4.1466 4.4020 8.5352
4.7547 4.1466 4.4020 8.5352 4.3607 4.0257 4.6239 7.2482
4.4722 4.0257 4.6239 6.0867 3.9413 3.7795 4.9127 7.9562
3.7900 3.8127 5.3622 6.6012 3.5921 3.2730 3.0879 4.4211
2.8067 3.2928 2.7479 6.3251 5.8379 3.3228 2.6195 4.7444
5.0863 3.2487 2.6333 6.4472 3.3654 2.9023 2.5086 5.3688
3.2487 2.6333 6.4472 3.3654 2.9023 2.5086 5.3688
3.2956 2.8389 2.0979 5.2559 2.3668 2.0060 1.8786 4.8228
2.2927 2.1800 1.8941 4.7583 1.5203 1.7519 1.8552 4.6317
1.3137 2.8676 1.8606 4.6079 1.0747 1.5362 1.8381 4.6804
1.5042 1.4728 1.8232 4.7896 2.0380 2.4714 1.8974 5.5494
1.4642 1.9503 1.9055 5.6346 0.4153 2.4349 1.9874 5.7361
0.5296 2.7801 1.9712 5.7820 0.8060 0.8856 2.2543 4.8719
0.6267 1.1057 2.2955 4.8703 0.7930 1.1013 2.2660 3.6883
0.9632 1.0979 2.2593 3.5877 0.8826 1.2864 1.9490 2.6734
0.7726 0.7182 1.9361 2.6518 0.8420 0.8016 1.5769 2.4003
80.8749 0.9746 1.6815 2.5406 1.0042 0.4515 1.0191 3.0357

9 INELASTIC CROSS SECTIONS. BE,C,FE,PB.

34.44
0.0000 0.0000 0.0000 0.0034 0.0000 0.0000 0.0000 0.0102
0.0000 0.0000 0.0000 0.0209 0.0000 0.0000 0.0000 0.1048
0.0000 0.0000 0.0097 0.1048 0.0000 0.0000 0.0291 0.2141
0.0000 0.0000 0.0223 0.2141 0.0000 0.0000 0.3236 0.3563
0.0000 0.0000 0.3749 0.3563 0.0000 0.0000 0.6193 0.5414
0.0000 0.0000 0.6050 0.5414 0.0000 0.0000 0.7906 0.7823
0.0000 0.0000 0.7852 0.7923 0.0000 0.0000 0.9986 1.0661
0.0456 0.0000 1.0135 1.0049 0.3165 0.0000 1.1878 1.2393
0.2971 0.0000 1.1859 1.2925 0.4441 0.0000 1.4029 1.8936
0.4453 0.0000 1.4024 1.9514 0.6317 0.0822 1.3929 2.3434
0.6528 0.0825 1.3891 2.2715 0.6887 0.1982 1.3563 2.3965
0.6537 0.1982 1.3562 2.3900 0.5764 0.3489 1.3142 2.5871
0.5878 0.3488 1.3142 2.6217 0.5497 0.5448 1.2597 2.6895
80.5497 0.5448 1.2224 2.6527 0.5000 0.8000 1.1166 2.5000

9 FISSION CROSS SECTIONS. BE,C,FE,PB.

*155= 0

9 TOTAL CROSS SECTIONS. BE,C,FE,PB.

35 000 4.5500 4.0000 11.000

5.4000	4.5500	4.0000	11.000	5.3095	4.4444	4.1006	10.336
5.3095	4.4444	4.1006	10.323	5.1916	4.3069	4.2315	9.5105
5.2289	4.3007	4.2315	9.5238	4.9263	4.1466	4.4020	8.5352
4.7537	4.1588	4.4020	8.5352	4.3607	4.0257	4.6239	7.2482
4.4722	4.0357	4.6239	6.0867	3.9413	3.7795	4.9127	7.9512
3.7900	3.8127	5.3622	6.0812	3.2921	3.2730	5.0879	4.4313
2.8067	3.2938	2.7479	6.3460	5.8379	3.3228	2.6195	4.8497
5.0863	3.2487	2.6333	6.5520	3.3654	2.9023	2.5377	5.5829
3.6641	2.7757	2.5212	6.0856	3.5782	2.8310	2.4781	6.1740
3.2956	2.8389	2.4728	5.6122	2.3668	2.0060	2.4979	5.3642
2.2922	2.1800	2.4991	5.2997	1.6533	1.7519	2.6458	5.4140
1.5136	2.6676	2.6458	5.4002	1.9195	1.5362	2.8367	5.7465
2.4136	1.4728	2.8367	5.7945	3.3688	2.4714	3.0852	6.7887
2.7613	1.9503	3.0914	6.9271	1.8597	2.4349	3.3903	7.6297
1.9593	2.2801	3.3736	7.7334	1.8538	0.9678	3.6472	7.2153
1.7683	1.1882	3.6846	7.1418	1.8917	1.2995	3.6352	6.0848
2.0269	1.2961	3.6290	5.9777	1.7664	1.6352	3.3114	5.2605
1.6678	1.0670	3.2985	5.2735	1.5656	1.3464	2.9299	5.0898
81.5985	1.4194	2.9713	5.1933	1.5042	1.2515	2.3676	5.5357

TXE01
TXE02
TXE03
TXE04
TXE05
TXE06
TXE07
TXE08
TXE09
TXE10
TXE11
TXE12
TXE13
TXE14
TXE15
TXE16
TXE17
TXE18
TXE19

9 ELASTIC SCATTERING CONTROL CONSTANTS

435	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	0	0	0	0	7	7	7	7	7	7	7	7	7	7	7	7	7
	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7

ESCC1
ESCC2
ESCC3
ESCC4
ESCC5

9 ELASTIC SCATTERING PROBABILITY TABLES

31	1	.83	.87	.563	.50	.5	.00	.438	-.50	.171	-.87	0	-1
1	1	.96	.97	.630	.64	.5	.09	.463	-.42	.090	-.94	0	-1
1	1	.96	.97	.630	.64	.5	.09	.463	-.42	.090	-.94	0	-1
1	1	.83	.87	.563	.50	.5	.00	.438	-.50	.171	-.87	0	-1
1	1	.83	.87	.563	.50	.5	.00	.438	-.50	.171	-.87	0	-1
1	1	.96	.97	.630	.64	.5	.09	.463	-.42	.090	-.94	0	-1
1	1	.96	.97	.630	.64	.5	.09	.463	-.42	.090	-.94	0	-1
1	1	.83	.87	.563	.50	.5	.00	.438	-.50	.171	-.87	0	-1
1	1	.83	.87	.563	.50	.5	.00	.438	-.50	.171	-.87	0	-1
1	1	.96	.97	.630	.64	.5	.09	.463	-.42	.090	-.94	0	-1
1	1	.96	.97	.630	.64	.5	.09	.463	-.42	.090	-.94	0	-1
1	1	.83	.87	.563	.50	.5	.00	.438	-.50	.171	-.87	0	-1
1	1	.83	.87	.563	.50	.5	.00	.438	-.50	.171	-.87	0	-1
1	1	.96	.97	.630	.64	.5	.09	.463	-.42	.090	-.94	0	-1
1	1	.96	.97	.630	.64	.5	.09	.463	-.42	.090	-.94	0	-1
1	1	.83	.87	.563	.50	.5	.00	.438	-.50	.171	-.87	0	-1
1	1	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1
1	1	.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
1	1	.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
1	1	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1
1	1	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1
1	1	.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
1	1	.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
1	1	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1
1	1	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1
1	1	.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
1	1	.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
1	1	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1
1	1	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1

ESM1E10
ESM2E1
ESM3E10
ESM4E10
FSM1F1
ESM2E11
ESM3E11
ESM4E1
ESM1E12
ESM2E12
ESM3E1
ESM4E12
ESM1E13
ESM2E1
ESM3E13
ESM4E13
FSM1F1
ESM2E14
ESM3E14
ESM4E1
ESM1E15
ESM2E15
ESM3E1
ESM4E15
ESM1E16
ESM2E1
FSM3E16
ESM4E16
FSM1F1

1	1	.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
1	1	.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
1	1	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1
1	7	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1
1		.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
1	1	.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
1	1	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1
1	1	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1
1	1	.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
1	1	.93	.97	.560	.64	.5	.09	.498	-.42	.140	-.94	0	-1
81	1	.75	.87	.515	.50	.5	.00	.485	-.50	.250	-.87	0	-1

ESM2E1
ESM3E17
ESM4E17
ESM1E1
ESM2E18
ESM3E18
ESM4E1
ESM1E19
ESM2E19
ESM3E1
ESM4E19

9. INELASTIC SCATTERING CONTROL CONSTANTS

435 .75 0

ISCC1

9 INELASTIC NEUTRON SPECTRUM CONTROL CONSTANTS

435 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0

INSC1

0 0 0 0 0 0 5 5 0 0 5 5 0 0 5 5

INSC2

0 0 5 5 0 0 5 5 0 0 5 5 0 0 5 5

INSC3

2 2 5 5 2 2 5 5 3 3 5 5

INSC4

4 4 5 5 5 5 5 5 6 6 5 5

INSC5

9 INELASTIC NEUTRON SPECTRUM PROBABILITY TABLES

31 .030 .75 .024 .50 .015 .25 .01 0 0

NSM3E06

1 .030 .75 .024 .50 .015 .25 .01 0 0

NSM4E0

1 .030 .75 .024 .50 .015 .25 .01 0 0

NSM3E07

1 .030 .75 .024 .50 .015 .25 .01 0 0

NSM4E07

1 .040 .75 .031 .50 .023 .25 .01 0 0

NSM3E0

1 .040 .75 .031 .50 .023 .25 .01 0 0

NSM4E08

1 .046 .75 .035 .50 .024 .25 .01 0 0

NSM3E09

1 .046 .75 .035 .50 .024 .25 .01 0 0

NSM4E0

1 .080 .75 .060 .50 .040 .25 .02 0 0

NSM3E10

1 .080 .75 .060 .50 .040 .25 .02 0 0

NSM4E10

1 .080 .75 .060 .50 .040 .25 .02 0 0

NSM3E1

1 .080 .75 .060 .50 .040 .25 .02 0 0

NSM4E11

1 .130 .75 .105 .50 .060 .25 .03 0 0

NSM3E12

1 .130 .75 .105 .50 .060 .25 .03 0 0

NSM4E1

1 1.20 0 0

NSM1E13

1 1.20 0 0

NSM2E13

1 .190 .75 .150 .50 .105 .25 .06 0 0

NSM3E1

1 .190 .75 .150 .50 .105 .25 .06 0 0

NSM4E13

1 2.40 0 0

NSM1E14

1 2.40 0 0

NSM2E1

1 .270 .75 .220 .50 .170 .25 .10 0 0

NSM3E14

1 .270 .75 .220 .50 .170 .25 .10 0 0

NSM4E14

1 2.40 0 0

NSM1E1

1 2.40 0 0

NSM2E15

1 .380 .75 .310 .50 .230 .25 .14 0 0

NSM3E15

1 .380 .75 .310 .50 .230 .25 .14 0 0

NSM4E1

1 4.80 .50 2.40 0 0

NSM1E16

1 4.80 .50 2.40 0 0

NSM2E16

1 .490 .75 .410 .50 .310 .25 .21 0 0

NSM3E1

1 .490 .75 .410 .50 .310 .25 .21 0 0

NSM4E16

1 6.79 .67 4.80 .33 2.40 0 0

NSM1E17

1 6.79 .67 4.80 .33 2.40 0 0

NSM2E1

1 .690 .75 .520 .50 .410 .25 .28 0 0

NSM3E17

1 .690 .75 .520 .50 .410 .25 .28 0 0

NSM4E17

1 7.90 .75 6.79 .50 4.80 .25 2.4 0 0

NSM1E1


```

1 0.6 0 .1 1 0.6 0 .1
1 0.8 0 .1 1 0.8 0 .1
1 1.0 0 .1 1 1.0 0 .1
1 1.2 0 .1 1 1.2 0 .1
1 1.6 0 .1 1 1.6 0 .1
1 2.0 0 .1 1 2.0 0 .1
1 2.0 0 .1 1 3.0 0 .1
1 3.0 0 .1 1 3.0 0 .1
1 4.0 0 .1 1 4.0 0 .1
1 4.0 0 .1 1 4.0 0 .1
1 5.2 0 .1 1 5.2 0 .1
1 5.2 0 .1 1 5.2 0 .1
1 6.6 0 .1 1 6.6 0 .1
1 6.6 0 .1 1 6.6 0 .1
1 8.8 0 .1 1 8.8 0 .1
1 8.8 0 .1 1 8.8 0 .1
1 9.8 0 .1 1 9.8 0 .1
81 9.8 0 .1 1 9.8 0 .1

```

9 FISSION GAMMA RAY SPECTRUM CONTROL CONSTANTS

```

435 75= 0

```

9 ABSORPTION GAMMA RAY SPECTRUM CONTROL CONSTANTS

```

435 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
0 0 0 0 0 0 0 0 0 0 0 0 0 2 0
0 0 2 0 0 0 2 0 2 0 2 0 2 0 2 0
2 0 2 0 2 0 2 0 2 0 2 0 2 0 2 0
2 0 2 0 2 0 2 0 2 0 2 0

```

9 ABSORPTION GAMMA RAY SPECTRUM PROBABILITY TABLES

```

1 8.0 0 .1
1 8.0 0 .1 1 8.0 0 .1
1 6.8 0 .1 1 8.0 0 .1
1 6.8 0 .1 1 8.0 0 .1
1 6.8 0 .1 1 8.0 0 .1
1 6.8 0 .1 1 8.0 0 .1
1 6.8 0 .1 1 8.0 0 .1
1 6.8 0 .1 1 8.0 0 .1
1 6.8 0 .1 1 8.0 0 .1
1 6.8 0 .1 1 8.0 0 .1
81 6.8 0 .1 1 8.0 0 .1
8

```

```

IM3,4E 0
IM3,4E09
IM3,4E10
IM3,4E
IM3,4E12
IM1,3E13
IM41F1
IM3,4E14
IM1,2E15
IM3,4E
IM1,2E16
IM3,4E16
IM1,2E
IM3,4E17
IM1,2E18
IM3,4E
IM1,2E19
IM3,4E19

```

```

FGE01-19

```

```

AGSC1
AGSC2
AGSC3
AGSC4
AGSC5

```

```

AM3E08
AM3E0910
AM1,3E11
AM1,3E
AM1,3E13
AM1,3E14
AM1,3E
AM1,3E16
AM1,3E17
AM1,3E
AM1,3E19
EOF

```

9 DIMENSION AND CONTROL CONSTANTS

435 16 4 5 12 13 3 9 10 1 0 0 886 1 3 1
 19329291415 1= 1 50 130 260 0 0 1 0 1 2 500 386
 =1/-4 1/-4 .1 .1

DCC01

DCC02

DCC03

9 ENERGY BOUNDS

3.1 .15 .2 .3 .4 .5 .6 .8 1
 1.5 2 3 4 5 6 8 10

EB01

EB02

9 ENERGY IMPORTANCES

31 1 1 1 1 1 1 1 1
 1 1 1 1 1 1 1 1

EI01

EI02

9 ACCELERATION FACTORS

31 1 1 1 1 1 1 1 1
 1 1 1 1 1 1 1 1

AF01

AF02

EOF

8

9 BOUNDARY DESCRIPTION

301 6 -5
 02 6 -4
 03 6 -3
 04 6 -2
 05 6 +2
 06 6 +3
 07 6 +4
 08 6 +6
 09 3 +0 +0 04
 10 3 +0 +0 09
 11 3 +0 +0 16
 812 3 +0 +0 25

BD01

BD02

BD03

BD04

BD05

BD06

BD07

BD08

BD09

BD10

BD11

BD12

9 REGION DESCRIPTION

435 +01 -3 +5 +0 -04 08 +05 02 +09 03
 02 -3 +1 -0 -05 01 +06 05 +09 03
 03 -4 +1 -0 -03 09 +06 05 -09 01 +10 04
 04 -4 +3 -0 -02 10 +06 05 -10 03 +11 07
 05 -3 +4 -0 -06 02 +07 06 +11 07
 06 -3 +2 -0 -07 05 +08 13 +11 07
 07 -4 +2 -0 -01 11 +08 13 -11 04 +12 12
 08 -3 +1 -0 -03 09 +04 01 +09 03
 09 -3 +4 -0 -02 10 +03 08 +10 04
 10 -3 +2 -0 -01 11 +02 09 +11 07
 11 +1 -1 +0 +01 02
 12 +1 -1 +0 -12 07
 8 -13 +1 -1 +0 -08 06

RD01

RD02

RD03

RD04

RD05

RD06

RD07

RD08

RD09

RD10

RD11

RD12

RD13

9 REGION IMPORTANCES

31 1 1 1 1 1 1 1 1
 8

RI01

EOF

9 VOLUME FRACTIONS

31 0 0 0
 0 1 0 0
 0 0 1 0
 0 0 0 1
 0 .5 .5 0

VF01

VF02

VF03

VF04

VF05

9 NUCLEI PER UNIT VOLUME. BE,C,FE,PB.

3.1202 .1128 .0849 .0330

NUV01

9 ATOMIC NUMBERS. BE,C,FE,PB.

8

AN01
EOF

9 COMPTONS SCATTERING.

3.5856	.4930	.4435	.4435	.4065	.4065	.3535	
.3535	.3170	.3170	.2890	.2890	.2700	.2700	.2350
.50	.2110	.2110	.1715	.1715	.1465	.1465	.1150
.1150	.0960	.0960	.0830	.0830	.0730	.0730	.0600
8.0600	.0510						

CSE00-03
CSE04 07
CSE08-11
CSE12-15
CSE16

9 PHOTOELECTRIC EFFECT. BE, C, FE, PB.

3.0208	.1730	148.0	1634				
.0000	.0170	19.10	1780	.0000	.0040	5.400	596.0
.0000	.0040	5.400	596.0	.0000	.0000	2.230	275.0
.0000	.0000	2.230	275.0	.0000	.0000	.6600	93.40
.0000	.0000	.6600	93.40	.0000	.0000	.2900	45.70
.0000	.0000	.2900	45.70	.0000	.0000	.1600	26.10
.0000	.0000	.1600	26.10	.0000	.0000	.1000	17.30
.0000	.0000	.1000	17.30	.0000	.0000	.0500	9.500
.0000	.0000	.0500	9.500	.0000	.0000	.0300	6.200
.0000	.0000	.0300	6.200	.0000	.0000	.0000	3.000
.0000	.0000	.0000	3.000	.0000	.0000	.0000	2.000
.0000	.0000	.0000	2.000	.0000	.0000	.0000	1.100
.0000	.0000	.0000	1.100	.0000	.0000	.0000	.8000
.0000	.0000	.0000	.8000	.0000	.0000	.0000	.6000
.0000	.0000	.0000	.6000	.0000	.0000	.0000	.4900
.0000	.0000	.0000	.4900	.0000	.0000	.0000	.3500
8.0000	.0000	.0000	.3500	.0000	.0000	.0000	.2800

PEE00
PEE01
PEE02
PEE03
PEE04
PEE05
PEE06
PEE07
PEE08
PEE09
PEE10
PEE11
PEE12
PEE13
PEE14
PEE15
PEE16

9 PAIR PRODUCTION. BE,C,FE,PB.

384*	.0000	.0000	.0000	.0000	.5/-4	.7/-4	.0003	.0010
	.5/-4	.7/-4	.0003	.0010	.0002	.0003	.0010	.0040
	.0002	.0003	.0010	.0040	.0004	.0007	.0030	.0090
	.0004	.0007	.0030	.0090	.0008	.0010	.0050	.0200
	.0008	.0010	.0050	.0200	.0020	.0020	.0110	.0300
8.	.0020	.0020	.0110	.0300	.0030	.0040	.0160	.0500

PPE00-10
PPE11
PPE12
PPE13
PPE14
PPE15
PPE16

9 ANGULAR DISTRIBUTION CONTROL CONSTANTS

435 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8

ADCC1

9 ANGULAR DISTRIBUTION PROBABILITY TABLES

31.000	-1.00	.9639	-.906	.8706	-.643	.7479	-.259
.5646	.2590	.3498	.5430	.1097	.9060	.0000	1.000
1.000	-1.00	.9680	-.906	.8836	-.643	.7682	-.259
.5909	.2590	.3763	.6430	.1217	.9060	.0000	1.000
1.000	-1.00	.9709	-.906	.8926	-.643	.7829	-.259
.6103	.2590	.3973	.6430	.1321	.9060	.0000	1.000
1.000	-1.00	.9731	-.906	.8995	-.643	.7936	-.259
.6249	.2590	.4143	.6430	.1413	.9060	.0000	1.000
1.000	-1.00	.9755	-.906	.9071	-.643	.8059	-.259
.6417	.2590	.4350	.6430	.1537	.9060	.0000	1.000
1.000	-1.00	.9777	-.906	.9146	-.643	.8182	-.259
.6585	.2590	.4567	.6430	.1682	.9060	.0000	1.000
1.000	-1.00	.9804	-.906	.9234	-.643	.8333	-.259
.6795	.2590	.4848	.6430	.1898	.9060	.0000	1.000
1.000	-1.00	.9827	-.906	.9315	-.643	.8479	-.259
.7006	.2590	.5133	.6430	.2153	.9060	.0000	1.000

ADE01
ADE01B
ADE02A
ADE02
ADE03A
ADE03B
ADE04
ADE04B
ADE05A
ADE05
ADE06A
ADE06B
ADE07
ADE07B
ADE08A
ADE08'

1.000	-1.00	.9842	-.906	.9370	-.643	.8582	-.259
.7162	.2590	.5345	.6430	.2365	.9060	.0000	1.000
1.000	-1.00	.9852	-.906	.9404	-.643	.8645	-.259
.7313	.2590	.5578	.6430	.2578	.9060	.0000	1.000
1.000	-1.00	.9862	-.906	.9442	-.643	.8724	-.259
.7389	.2590	.5656	.6430	.271	.9060	.0000	1.000
1.000	-1.00	.9869	-.906	.9468	-.643	.8776	-.259
.7476	.2590	.5777	.6430	.2854	.9060	.0000	1.000
1.000	-1.00	.9874	-.906	.9489	-.643	.8820	-.259
.7552	.2590	.5883	.6430	.2984	.9060	.0000	1.000
1.000	-1.00	.9879	-.906	.9507	-.643	.8858	-.259
.7619	.2590	.5978	.6430	.3102	.9060	.0000	1.000
1.000	-1.00	.9883	-.906	.9523	-.643	.8892	-.259
.7678	.2590	.6063	.6430	.3211	.9060	.0000	1.000
1.000	-1.00	.9886	-.906	.9537	-.643	.8912	-.259
8.7729	.2590	.6138	.6430	.3300	.9060	.0000	1.000

9 ABSORPTION CONVERSION FACTORS

31	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
1	1	1	1	1	1	1	1	1	1
81	1	1	1	1	1	1	1	1	1

8

ADE09A
ADE09
ADE10A
ADE10B
ADE11
ADE11B
ADE12A
ADE12
ADE13A
ADE13B
ADE14
ADE14B
ADE15A
ADE15
ADE16A
ADE16B

ACFE00
ACFE01
ACFE0
ACFE03
ACFE04
ACFE0
ACFE06
ACFE07
ACFE0
ACFE09
ACFE10
ACFE1
ACFE12
ACFE13
ACFE1
ACFE15
ACFE16
EOF

10.2 Output

(These runs were obtained
on the IBM-704)

11925

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-----
11925      JJ LOECHLER      FMC NEUTRON      00516 UNIT 4313 P CC
++      TAPES 5      6      7      9      10
++      FOR PROGRAM STOP OR TIME EXCEEDED TRANSFER TO 11074
++ ON TIME, FORCE PM SAVE      10
ID. 1, BG. 4725 , CH. 53259      DATE 01/18/61.      TAPES 8, PUN 0, CDR 0
***** START TIME 10.12      ***** STOP TIME *****
++ STARTING TIME AFTER MACHINE SETUP STOP      10.13
-----

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-----NUMBER OF RECORDS ON SECONDARY GAMMA TAPE-----2-----
NUMBER OF PARTICLES IN EACH RECORD 500
-----NUMBER OF PARTICLES IN LAST RECORD-----386-----

-----NUMBER OF RECORDS ON HISTORY TALLY TAPE-----1-----
NUMBER OF PARTICLES IN EACH RECORD 1294
-----NUMBER OF PARTICLES IN LAST RECORD-----106-----

-----NUMBER OF COLLISION CUTOFFS AFTER ENERGY CUTOFF 2.15575,01-----
NUMBER OF WEIGHT CUTOFFS 0.00000,00
-----LAST RANDOM NUMBER-----19329291415-----

-----TOTAL NUMBER OF SOURCE NEUTRONS 1.00000,03-----
-----TOTAL SOURCE ENERGY 9.93588,03-----

----- REGION ---1-----		
ENERGY	ABSORPTION	MONTE CARLO FLUX
0.00000,00	0.00000,00	5.26584,01
1.00000-01	0.00000,00	1.16161,01
1.30200-01	0.00000,00	1.72275,01
1.69500-01	0.00000,00	4.38397,01
2.20600-01	0.00000,00	4.97675,01
2.87200-01	0.00000,00	4.86060,01
3.73800-01	0.00000,00	8.90172,01
4.86600-01	2.41486-03	5.30535,01
6.33500-01	0.00000,00	5.84816,01
8.24600-01	2.34981-09	5.18100,01
1.07340,00	0.00000,00	2.13927,01
1.39740,00	0.00000,00	1.24318,01
1.81900,00	0.00000,00	1.85320,01
2.36800,00	6.18094-09	1.20908,01
3.08250,00	1.30920-08	4.86736,01
4.01270,00	5.67662-08	1.39891,02
5.22360,00	1.08546-01	4.49236,02
6.79980,00	6.27278-01	5.07664,02
8.85170,00	1.84072,00	6.28361,02
1.15230,01	4.31010,00	7.20416,02
*****	*****	*****
TOTALS	6.88906,00	3.03477,03

----- REGION 2 -----				
ENERGY	ABSORPTION	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	1.47329,01	1.01716,01	1.08890,01
1.00000-01	0.00000,00	3.92126,00	1.99493,00	1.49810,00
1.30200-01	0.00000,00	3.51823,00	2.99205,00	3.17170,00
1.69500-01	0.00000,00	3.58908,00	4.72495,00	3.74853,00
2.20600-01	0.00000,00	3.73054,00	5.32450,00	4.03732,00
2.87200-01	0.00000,00	5.85696,00	2.74292,00	4.39550,00
3.73800-01	0.00000,00	5.93101,00	5.21083,00	5.76880,00
4.86600-01	0.00000,00	6.66863,00	3.31938,00	4.14204,00
6.33500-01	0.00000,00	4.29245,00	3.88505,00	4.04975,00
8.24600-01	0.00000,00	5.88294,00	4.68180,00	4.06413,00
1.07340,00	0.00000,00	2.66095,00	9.50894-01	1.21102,00
1.39740,00	6.30809-02	4.04262,00	2.74914,00	2.40767,00
1.81900,00	0.00000,00	0.00000,00	0.00000,00	0.00000,00
2.36800,00	0.00000,00	1.53461,00	1.51719,00	2.47670,00
3.08250,00	4.92278-01	4.73357,00	1.94407,00	2.27802,00
4.01270,00	1.03349,00	1.30443,01	9.34635,00	8.09308,00
5.22360,00	1.30864,00	3.07072,01	3.24925,01	3.44774,01
6.79980,00	1.44007,00	2.99780,01	1.20942,01	3.41825,01
8.85170,00	1.16859,00	3.64564,01	4.26130,01	4.24963,01
1.15230,01	4.00834-01	3.43660,01	3.68300,01	3.84151,01
*****	*****	*****	*****	*****
TOTALS	5.90697,00	2.15648,02	2.05585,02	2.11803,02

----- REGION 3 -----				
ENERGY	ABSORPTION	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	4.46869,01	2.73414,01	2.77418,01
1.00000-01	0.00000,00	2.30268,01	1.47648,01	1.19237,01
1.30200-01	0.00000,00	1.96679,01	9.58219,00	1.07886,01
1.69500-01	0.00000,00	3.90837,01	2.76968,01	2.51640,01
2.20600-01	0.00000,00	3.96599,01	1.82408,01	2.18559,01
2.87200-01	0.00000,00	6.20955,01	3.67495,01	3.61078,01
3.73800-01	0.00000,00	6.27646,01	4.42567,01	4.31119,01
4.86600-01	0.00000,00	3.69206,01	2.16999,01	2.52521,01
6.33500-01	0.00000,00	3.90221,01	2.85774,01	2.72831,01
8.24600-01	0.00000,00	2.23090,01	1.78297,01	1.54353,01
1.07340,00	0.00000,00	2.09768,01	1.37427,01	1.24321,01
1.39740,00	1.18766-01	9.77967,00	3.09378,00	3.89713,00
1.81900,00	9.78971-01	1.66467,01	7.39198,00	7.43170,00
2.36800,00	2.31872,00	1.62058,01	7.30621,00	6.09581,00
3.08250,00	4.11579,00	3.52121,01	2.05855,01	1.87405,01
4.01270,00	1.92898,00	6.12859,01	4.78422,01	4.71551,01
5.22360,00	1.08016,01	1.74672,02	1.47516,02	1.41263,02
6.79980,00	8.56917,00	1.70181,02	1.52157,02	1.55660,02
8.85170,00	6.95179,00	2.18105,02	2.04722,02	2.03590,02
1.15230,01	1.91420,00	2.36639,02	2.54900,02	2.18992,02
*****	*****	*****	*****	*****
TOTALS	3.76980,01	1.34894,03	1.09300,03	1.05992,03

REGION 4				
ENERGY	ABSORPTION	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	6.04657,01	2.87660,01	2.61714,01
1.00000-01	0.00000,00	3.47504,01	1.49784,01	1.54798,01
1.30200-01	0.00000,00	2.01009,01	1.31736,01	1.59649,01
1.69500-01	0.00000,00	5.44952,01	2.83755,01	2.13623,01
2.20600-01	0.00000,00	5.36381,01	2.63792,01	2.60856,01
2.87200-01	0.00000,00	6.14571,01	3.00139,01	3.37838,01
3.73800-01	0.00000,00	7.98358,01	2.99350,01	3.10120,01
4.86600-01	0.00000,00	3.98620,01	2.09512,01	1.96966,01
6.33500-01	0.00000,00	3.53807,01	1.86614,01	1.79767,01
8.24600-01	2.94496-09	3.33239,01	1.64477,01	1.53454,01
1.07340,00	0.00000,00	2.77279,01	1.46047,01	1.51544,01
1.39740,00	9.61157-09	1.00501,01	4.00052,00	5.42041,00
1.81900,00	0.00000,00	8.81538,00	8.62389,00	9.92609,00
2.36800,00	1.28436-08	1.82268,01	1.24409,01	1.14274,01
3.08250,00	1.66625-08	2.67776,01	2.10642,01	1.98035,01
4.01270,00	2.25324-08	5.52715,01	4.55211,01	3.93869,01
5.22360,00	7.97459-02	1.33958,02	1.16985,02	1.18930,02
6.79980,00	2.50731-01	1.14410,02	1.15018,02	1.22280,02
8.85170,00	1.27039,00	1.61785,02	1.53701,02	1.51411,02
1.15230,01	1.80706,00	1.81032,02	1.83181,02	1.70314,02
*****	*****	*****	*****	*****
TOTALS	3.40793,00	1.21136,03	9.02822,02	8.86932,02

REGION 5				
ENERGY	ABSORPTION	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	1.50714,01	8.34782,00	8.37274,00
1.00000-01	0.00000,00	1.06224,01	4.14807,00	3.61835,00
1.30200-01	0.00000,00	1.30509,01	7.13386,00	7.26151,00
1.69500-01	0.00000,00	1.33512,01	8.68268,00	7.79571,00
2.20600-01	0.00000,00	1.10761,01	6.24789,00	7.19490,00
2.87200-01	0.00000,00	1.93305,01	1.28758,01	8.84706,00
3.73800-01	0.00000,00	1.61883,01	7.16465,00	8.27667,00
4.86600-01	2.42137-08	1.56557,01	7.47396,00	5.89240,00
6.33500-01	1.17315-08	1.15242,01	4.83011,00	7.78155,00
8.24600-01	0.00000,00	5.05440,00	4.80196,00	3.72941,00
1.07340,00	1.25735-09	8.86493,00	5.57684,00	5.07107,00
1.39740,00	0.00000,00	1.56781,00	1.42489,00	2.61401,00
1.81900,00	5.64466-09	3.48844,00	2.69828,00	2.80876,00
2.36800,00	0.00000,00	3.72025,00	4.08992,00	4.80909,00
3.08250,00	1.70465-08	1.17355,01	4.71903,00	6.17251,00
4.01270,00	1.11433-08	1.87638,01	1.85464,01	1.58234,01
5.22360,00	5.68707-08	4.24754,01	3.44484,01	3.40285,01
6.79980,00	0.00000,00	3.19183,01	2.95864,01	2.97026,01
8.85170,00	6.43694-08	5.24954,01	4.46220,01	4.19541,01
1.15230,01	0.00000,00	4.52178,01	4.55414,01	4.47632,01
*****	*****	*****	*****	*****
TOTALS	1.92277-07	3.51173,02	2.62960,02	2.56718,02

REGION 6				
ENERGY	ABSORPTION	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	1.68889,01	5.94992,00	8.16502,00
1.00000-01	0.00000,00	8.38726,00	2.60410,00	1.59092,00
1.30200-01	0.00000,00	8.52719,00	4.36676,00	4.27771,00
1.69500-01	0.00000,00	1.51292,01	3.74907,00	3.95289,00
2.20600-01	0.00000,00	1.83559,01	3.88860,00	4.60504,00
2.87200-01	0.00000,00	1.25877,01	6.93116,00	3.50169,00
3.73800-01	0.00000,00	1.02724,01	3.93936,00	6.00718,00
4.86600-01	1.15969-03	2.32265,01	9.53348,00	7.97956,00
6.33500-01	0.00000,00	1.40457,01	3.34232,00	5.31366,00
8.24600-01	0.00000,00	6.99811,00	3.90743,00	2.73752,00
1.07340,00	0.00000,00	5.78850,00	4.18194,00	2.61110,00
1.39740,00	0.00000,00	9.77582,00	9.46269-01	1.38223,00
1.81900,00	0.00000,00	1.12011,01	2.15318,00	2.15595,00
2.36800,00	0.00000,00	1.03403,01	2.37584,00	3.04288,00
3.08250,00	0.00000,00	1.55710,01	2.97547,00	4.62457,00
4.01270,00	8.37903-09	2.37222,01	9.72669,00	6.78396,00
5.22360,00	8.44771-10	4.33722,01	2.09148,01	2.00864,01
6.79980,00	0.00000,00	4.80595,01	2.72318,01	2.09935,01
8.85170,00	0.00000,00	4.11005,01	2.37326,01	2.71807,01
1.15230,01	2.54768-08	4.95717,01	2.86622,01	2.79064,01
*****	*****	*****	*****	*****
TOTALS	1.15972-03	3.92922,02	1.71113,02	1.64899,02

 REGION 7

ENERGY	ABSORPTION	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	5.02688,01	2.81215,01	2.87236,01
1.00000-01	0.00000,00	2.99722,01	1.14663,01	1.01593,01
1.30200-01	0.00000,00	3.37913,01	1.09802,01	1.22912,01
1.69500-01	0.00000,00	4.67937,01	3.38878,01	2.39525,01
2.20600-01	0.00000,00	4.12702,01	2.20897,01	2.59205,01
2.87200-01	0.00000,00	4.51695,01	3.01597,01	3.03289,01
3.73800-01	0.00000,00	4.90571,01	3.64217,01	4.11043,01
4.86600-01	2.76131-03	3.58640,01	2.53394,01	2.82341,01
6.33500-01	0.00000,00	3.16041,01	1.98173,01	2.00786,01
8.24600-01	0.00000,00	3.13221,01	2.25185,01	1.86788,01
1.07340,00	0.00000,00	2.46543,01	1.65447,01	1.63830,01
1.39740,00	0.00000,00	1.10511,01	4.33147,00	7.78408,00
1.81900,00	0.00000,00	2.19588,01	1.21607,01	1.27748,01
2.36800,00	0.00000,00	2.03109,01	1.05889,01	9.71359,00
3.08250,00	0.00000,00	3.44306,01	1.87403,01	1.78160,01
4.01270,00	5.83133-08	7.72377,01	3.19746,01	3.14761,01
5.22360,00	1.69971-09	1.31842,02	9.70107,01	1.01124,02
6.79980,00	0.00000,00	1.24758,02	1.03153,02	9.85924,01
8.85170,00	6.55342-09	1.40899,02	1.24315,02	1.35044,02
1.15230,01	1.65674-07	1.71387,02	1.66173,02	1.44614,02
*****	*****	*****	*****	*****
TOTALS	2.76154-03	1.15364,03	8.25794,02	8.14761,02

REGION 8				
ENERGY	ABSORPTION	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	4.49049,00	2.74674,00	3.09748,00
1.00000-01	0.00000,00	2.29728,00	1.45521,00	1.65209,00
1.30200-01	0.00000,00	1.63479,00	7.31668-01	1.18253,00
1.69500-01	0.00000,00	8.54626,00	5.65697,00	3.48742,00
2.20600-01	0.00000,00	8.54942,00	4.87900,00	3.61074,00
2.87200-01	0.00000,00	1.14224,01	6.85812,00	7.76783,00
3.73800-01	0.00000,00	1.16662,01	7.91479,00	1.03143,01
4.86600-01	0.00000,00	3.38519,00	2.94495,00	4.02545,00
6.33500-01	0.00000,00	4.85366,00	5.51149,00	3.53007,00
8.24600-01	0.00000,00	1.94401,00	1.45267,00	1.73943,00
1.07340,00	0.00000,00	1.58580,00	9.33236-01	1.40563,00
1.39740,00	4.37696-02	1.24972,00	7.70748-01	9.83258-01
1.81900,00	0.00000,00	1.70613,00	2.41091,00	2.07480,00
2.36800,00	1.32153-01	5.98016-01	1.35295,00	1.06866,00
3.08250,00	2.80321-01	5.77183,00	2.77261,00	4.63198,00
4.01270,00	1.33100,00	1.10885,01	8.46646,00	1.15333,01
5.22360,00	4.52586-01	2.53826,01	2.80915,01	2.32310,01
6.79980,00	1.31949,00	2.15247,01	2.25933,01	2.05216,01
8.85170,00	1.09829,00	3.25795,01	3.08678,01	3.05981,01
1.15230,01	4.86257-01	3.17611,01	3.66332,01	3.22287,01
*****	*****	*****	*****	*****
TOTALS	5.14387,00	1.92038,02	1.75044,02	1.68684,02

----- REGION 9 -----				
ENERGY	ABSORPTION	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	5.96616,00	4.65100,00	4.45238,00
1.00000-01	0.00000,00	4.34295,00	2.43001,00	2.14539,00
1.30200-01	0.00000,00	3.57492,00	1.80899,00	1.97274,00
1.69500-01	0.00000,00	5.05137,00	4.67132,00	4.01586,00
2.20600-01	0.00000,00	1.76127,01	8.51972,00	9.37742,00
2.87200-01	0.00000,00	1.25779,01	2.92446,00	5.93840,00
3.73800-01	0.00000,00	1.04131,01	9.84829,00	8.66834,00
4.86600-01	2.83557-09	1.93233,00	9.75960-01	3.00702,00
6.33500-01	4.34662-09	5.70351,00	4.87138,00	3.90311,00
8.24600-01	0.00000,00	1.69077,00	1.45324,00	1.17531,00
1.07340,00	0.00000,00	3.99734,00	1.51887,00	1.54518,00
1.39740,00	0.00000,00	2.46940-01	2.55134-01	3.54033-01
1.81900,00	5.72280-09	7.29501,00	5.19730,00	3.62574,00
2.36800,00	0.00000,00	1.58849,00	7.65659-01	1.18892,00
3.08250,00	0.00000,00	6.25145,00	4.82614,00	5.71101,00
4.01270,00	0.00000,00	6.63629,00	6.28760,00	9.86701,00
5.22360,00	1.27472-08	2.99913,01	2.81767,01	2.42404,01
6.72980,00	4.05979-08	2.41533,01	1.62861,01	1.67420,01
8.85170,00	5.44840-09	2.33891,01	2.25738,01	2.43414,01
1.15230,01	0.00000,00	2.39156,01	2.65857,01	2.66322,01
*****	*****	*****	*****	*****
TOTALS	7.16985-08	1.96331,02	1.54827,02	1.58911,02

----- REGION 10 -----				
ENERGY	ABSORPTION	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	9.58957,00	5.66117,00	7.82130,00
1.00000-01	0.00000,00	2.65017,00	2.51008,00	2.50050,00
1.30200-01	0.00000,00	1.24458,00	9.88866-01	1.38042,00
1.69500-01	0.00000,00	6.92367,00	3.52156,00	1.87322,00
2.20600-01	0.00000,00	1.17744,01	8.60803,00	9.41246,00
2.87200-01	0.00000,00	1.07077,01	3.97384,00	4.86521,00
3.73800-01	0.00000,00	8.50896,00	7.86777,00	6.10613,00
4.86600-01	0.00000,00	1.50922,00	9.75960-01	3.49806,00
6.33500-01	0.00000,00	3.21519,00	3.87887,00	3.70273,00
8.24600-01	0.00000,00	2.55546,00	2.37727,00	2.02673,00
1.07340,00	0.00000,00	2.42233,00	9.33236-01	7.19302-01
1.39740,00	0.00000,00	3.46342,00	2.55134-01	9.49920-01
1.81900,00	0.00000,00	9.82168,00	4.47619,00	3.70387,00
2.36800,00	0.00000,00	2.26306,00	1.80743,00	1.90196,00
3.08250,00	0.00000,00	5.34995,00	4.34034,00	4.50951,00
4.01270,00	3.55219-08	8.80971,00	4.76663,00	6.74959,00
5.22360,00	4.97518-09	2.67281,01	2.26002,01	1.89152,01
6.79980,00	0.00000,00	1.39479,01	1.33173,01	1.66654,01
8.85170,00	0.00000,00	1.59131,01	1.74890,01	2.12439,01
1.15230,01	0.00000,00	2.33618,01	2.46617,01	2.14755,01
*****	*****	*****	*****	*****
TOTALS	4.04970-08	1.70760,02	1.35071,02	1.40021,02

----- REGION 11 -----		
ENERGY	MONTE CARLO LEAKAGE	EXPECTATION LEAKAGE
0.00000,00	5.66331,00	4.56304,00
1.00000-01	1.53529,00	1.59706,00
1.30200-01	9.88866-01	9.18600-01
1.69500-01	2.70143,00	1.14976,00
2.20600-01	7.63316,00	6.28244,00
2.87200-01	1.98308,00	3.97140,00
3.73800-01	4.94607,00	4.13702,00
4.86600-01	1.96002,00	3.05338,00
6.33500-01	1.94605,00	2.04074,00
8.24600-01	3.34963,00	1.62036,00
1.07340,00	1.00000-00	2.01795,00
1.39740,00	2.55134-01	9.39891-01
1.81900,00	5.47619,00	5.99313,00
2.36800,00	0.00000,00	2.13776-01
3.08250,00	3.56723,00	4.25089,00
4.01270,00	8.38493,00	9.80256,00
5.22360,00	2.26591,01	1.92293,01
6.79980,00	1.10317,01	1.35205,01
8.85170,00	1.66326,01	1.77969,01
1.15230,01	1.96879,01	1.87354,01
*****	*****	*****
TOTALS	1.21402,02	1.19854,02

----- REGION 12 -----		
ENERGY	MONTE CARLO LEAKAGE	EXPECTATION LEAKAGE
0.00000,00	2.27173,01	2.16835,01
1.00000-01	1.19731,01	9.07862,00
1.30200-01	1.15845,01	1.50348,01
1.69500-01	2.21307,01	1.68411,01
2.20600-01	1.69483,01	2.63803,01
2.87200-01	2.46572,01	1.89501,01
3.73800-01	2.96634,01	2.98175,01
4.86600-01	1.47176,01	1.87012,01
6.33500-01	1.78506,01	1.46291,01
8.24600-01	1.71986,01	1.33979,01
1.07340,00	1.23409,01	1.25066,01
1.39740,00	3.28254,00	7.85697,00
1.81900,00	1.26237,01	1.09240,01
2.36800,00	9.19172,00	7.15898,00
3.08250,00	1.90531,01	1.99245,01
4.01270,00	3.84608,01	3.62723,01
5.22360,00	8.90985,01	7.57365,01
6.79980,00	8.89644,01	8.87468,01
8.85170,00	1.04627,02	1.13812,02
1.15230,01	1.26705,02	1.22275,02
*****	*****	*****
TOTALS	6.93789,02	6.99728,02

----- REGION 13 -----		
ENERGY	MONTE CARLO LEAKAGE	EXPECTATION LEAKAGE
0.00000,00	2.79920,00	3.40960,00
1.00000-01	0.00000,00	1.14993,00
1.30200-01	2.96951,00	1.23653,00
1.69500-01	9.83723-01	1.27950,00
2.20600-01	0.00000,00	1.17837,00
2.87200-01	2.84788,00	2.24456,00
3.73800-01	9.89095-01	1.67951,00
4.86600-01	8.23411,00	4.93679,00
6.33500-01	1.95578,00	3.27824,00
8.24600-01	1.96256,00	1.27963,00
1.07340,00	2.94578,00	1.72474,00
1.39740,00	2.90464,00	2.22622,00
1.81900,00	1.35941,00	1.32920,00
2.36800,00	4.93588,00	4.37568,00
3.08250,00	5.02711-01	1.23881,00
4.01270,00	5.33871,00	3.95829,00
5.22360,00	1.42398,01	1.48364,01
6.79980,00	1.47134,01	1.56351,01
8.85170,00	1.77175,01	1.62492,01
1.15230,01	1.68024,01	1.79692,01
*****	*****	*****
TOTALS	1.04202,02	1.01216,02

HISTORY TALLIES

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
GAMMA = 9.184-01 ALPHA = -3.886-01 BETA = 7.444-02 Z = 6.000,00 X = -2.539,00 Y = 4.863-01 ENERGY = 1.443,01

REGION 13 NEUTRONS 9.963-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 1
GAMMA = 8.823-01 ALPHA = -3.238-01 BETA = -3.416-01 Z = 6.000,00 X = -2.131,00 Y = -3.254,00 ENERGY = 7.163,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 5
GAMMA = 9.367-01 ALPHA = -9.452-02 BETA = -3.372-01 Z = 6.000,00 X = -1.524,00 Y = 1.093,00 ENERGY = 5.014-01

REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 6
GAMMA = 9.446-01 ALPHA = -1.865-02 BETA = 3.277-01 Z = 6.000,00 X = -3.241,00 Y = 2.345,00 ENERGY = 1.914,00

REGION 13 NEUTRONS 9.846-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 4
GAMMA = 9.996-01 ALPHA = -4.340-03 BETA = 2.958-02 Z = 6.000,00 X = 3.234,00 Y = 1.381,00 ENERGY = 8.993,00

REGION 13 NEUTRONS 3.594-01 COLLISIONS 4 COLLISIONS OCCURED IN REGIONS 1 1 1 3 3
GAMMA = 7.361-01 ALPHA = -3.486-01 BETA = -5.802-01 Z = 6.000,00 X = 1.061-01 Y = -4.179,00 ENERGY = 2.246,00

REGION 13 NEUTRONS 8.145-01 COLLISIONS 3 COLLISIONS OCCURED IN REGIONS 1 1 1 2
GAMMA = 9.373-01 ALPHA = 1.405-01 BETA = 3.191-01 Z = 6.000,00 X = 9.341-01 Y = 2.857,00 ENERGY = 2.637,00

REGION 13 NEUTRONS 9.625-01 COLLISIONS 3 COLLISIONS OCCURED IN REGIONS 1 1 1 6
GAMMA = 2.464-01 ALPHA = -2.389-01 BETA = -9.393-01 Z = 6.000,00 X = -3.823,00 Y = -6.085-01 ENERGY = 7.318-01

REGION 13 NEUTRONS 7.769-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 2 6
GAMMA = 5.717-01 ALPHA = -8.171-01 BETA = -7.469-02 Z = 6.000,00 X = -3.333,00 Y = 9.482-01 ENERGY = 6.273,00

REGION 13 NEUTRONS 9.933-01 COLLISIONS 6 COLLISIONS OCCURED IN REGIONS 1 1 1 2 7 4 6
GAMMA = 5.385-01 ALPHA = 7.042-01 BETA = 4.628-01 Z = 6.000,00 X = 2.859,00 Y = -9.113-01 ENERGY = 6.925-01

REGION 13 NEUTRONS 9.889-01 COLLISIONS 4 COLLISIONS OCCURED IN REGIONS 1 1 1 5 5
GAMMA = 9.174-01 ALPHA = -3.963-01 BETA = -3.756-02 Z = 6.000,00 X = -2.916,00 Y = 3.184,00 ENERGY = 5.745-01

REGION 13 NEUTRONS 9.652-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 1
GAMMA = 9.825-01 ALPHA = -8.004-02 BETA = 1.683-01 Z = 6.000,00 X = -5.135-01 Y = 1.032,00 ENERGY = 1.339,01

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
GAMMA = 9.893-01 ALPHA = -1.460-01 BETA = -3.210-03 Z = 6.000,00 X = -8.853-01 Y = -1.947-02 ENERGY = 1.317,01

REGION 13 NEUTRONS 9.412-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 2
GAMMA = 7.106-01 ALPHA = -3.798-02 BETA = 7.026-01 Z = 6.000,00 X = -5.097-01 Y = 3.940,00 ENERGY = 1.234,01

REGION 13 NEUTRONS 5.027-01 COLLISIONS 9 COLLISIONS OCCURED IN REGIONS 1 1 6 6 2 5 5 6
GAMMA = 8.336-01 ALPHA = 5.107-02 BETA = 5.500-01 Z = 6.000,00 X = -1.143-01 Y = 3.691,00 ENERGY = 3.471,00

REGION 13 NEUTRONS 9.823-01 COLLISIONS 3 COLLISIONS OCCURED IN REGIONS 1 4 6 6
GAMMA = 2.583-01 ALPHA = -2.939-01 BETA = 9.203-01 Z = 6.000,00 X = 6.787-01 Y = 1.845,00 ENERGY = 5.748-01

REGION 13 NEUTRONS 8.801-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 4 3
GAMMA = 7.661-01 ALPHA = -4.477-01 BETA = -4.612-01 Z = 6.000,00 X = -1.449,00 Y = -4.596,00 ENERGY = 8.355,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS 4 COLLISIONS OCCURED IN REGIONS 1 1 1 6 6
 GAMMA = 7.843-01 ALPHA = 5.954-01 BETA = 1.744-01 Z = 6.000,00 X = -3.054-01 Y = 2.814,00 ENERGY = 2.880,00

REGION 13 NEUTRONS 7.424-01 COLLISIONS 8 COLLISIONS OCCURED IN REGIONS 1 5 3 1 2 2 2 5
 GAMMA = 7.988-01 ALPHA = -3.844-01 BETA = -4.628-01 Z = 6.000,00 X = -1.862,00 Y = -3.264,00 ENERGY = 1.000-01

REGION 13 NEUTRONS 9.838-01 COLLISIONS 5 COLLISIONS OCCURED IN REGIONS 1 1 1 1 7 7
 GAMMA = 9.218-01 ALPHA = 2.102-01 BETA = -3.258-01 Z = 6.000,00 X = -1.268,00 Y = 1.965,00 ENERGY = 5.186-01

REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 6
 GAMMA = 7.211-01 ALPHA = 5.095-01 BETA = -4.695-01 Z = 6.000,00 X = 1.185,00 Y = 7.278-02 ENERGY = 6.684,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 6
 GAMMA = 9.574-01 ALPHA = 2.706-01 BETA = -1.013-01 Z = 6.000,00 X = 1.068,00 Y = 2.232-01 ENERGY = 6.919,00

REGION 13 NEUTRONS 9.820-01 COLLISIONS 3 COLLISIONS OCCURED IN REGIONS 1 1 3 7
 GAMMA = 8.607-01 ALPHA = -5.083-01 BETA = -2.863-02 Z = 6.000,00 X = -1.937,00 Y = 4.298,00 ENERGY = 1.175,00

REGION 13 NEUTRONS 9.983-01 COLLISIONS 11 COLLISIONS OCCURED IN REGIONS 1 4 3 4 3 7 3 2
 GAMMA = 9.391-01 ALPHA = -1.943-01 BETA = 2.836-01 Z = 6.000,00 X = -1.288,00 Y = 5.809-01 ENERGY = 1.520-01

REGION 13 NEUTRONS 8.979-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 8
 GAMMA = 9.790-01 ALPHA = 1.936-01 BETA = -6.419-02 Z = 6.000,00 X = 4.550-01 Y = -4.512-01 ENERGY = 7.604,00

REGION 13 NEUTRONS 9.626-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 1
 GAMMA = 9.148-01 ALPHA = 3.961-01 BETA = -7.897-02 Z = 6.000,00 X = 2.613,00 Y = -5.981-01 ENERGY = 1.018,00

REGION 13 NEUTRONS 9.855-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 1
 GAMMA = 8.687-01 ALPHA = 4.929-01 BETA = -4.856-02 Z = 6.000,00 X = 3.397,00 Y = -3.572-01 ENERGY = 7.979,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 6
 GAMMA = 7.630-01 ALPHA = -1.630-01 BETA = -6.255-01 Z = 6.000,00 X = -1.332,00 Y = -1.837,00 ENERGY = 7.830,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 6
 GAMMA = 7.014-01 ALPHA = -1.233-02 BETA = -7.127-01 Z = 6.000,00 X = -9.756-01 Y = -2.123,00 ENERGY = 7.717,00

REGION 13 NEUTRONS 7.415-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 2
 GAMMA = 9.495-01 ALPHA = -1.666-01 BETA = 2.659-01 Z = 6.000,00 X = -1.827,00 Y = 8.222-01 ENERGY = 4.935,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 1
 GAMMA = 9.294-01 ALPHA = -3.632-01 BETA = -6.515-02 Z = 6.000,00 X = -2.358,00 Y = -4.266-01 ENERGY = 5.073,00

REGION 13 NEUTRONS 8.959-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 2
 GAMMA = 8.317-01 ALPHA = -5.075-01 BETA = -2.252-01 Z = 6.000,00 X = -2.708,00 Y = -1.791,00 ENERGY = 1.156,01

REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 7
 GAMMA = 4.158-01 ALPHA = -1.490-01 BETA = 8.972-01 Z = 6.000,00 X = -3.145-01 Y = 6.252-01 ENERGY = 4.386,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 8.600-01 ALPHA = -5.044-02 BETA = 5.077-01 Z = 6.000,00 X = -3.519-01 Y = 3.542,00 ENERGY = 1.388,01

REGION 13 NEUTRONS 8.012-01 COLLISIONS 3 COLLISIONS OCCURED IN REGIONS 1 1 6 2
 GAMMA = 9.311-01 ALPHA = -3.303-01 BETA = 1.546-01 Z = 6.000,00 X = -1.698,00 Y = 2.091,00 ENERGY = 5.951,00

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REGION 13 NEUTRONS 9.999-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 7 6
GAMMA = 6.725-01 ALPHA = -3.714-01 BETA = -6.401-01 Z = 6.000,00 X = 2.131,00 Y = 1.986,00 ENERGY = 5.434-01
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REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 5
GAMMA = 6.440-01 ALPHA = 5.473-01 BETA = -5.346-01 Z = 6.000,00 X = 3.152,00 Y = -1.580,00 ENERGY = 1.494,01
-----
REGION 13 NEUTRONS 9.911-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 1 1
GAMMA = 8.754-01 ALPHA = 3.709-01 BETA = 3.101-01 Z = 6.000,00 X = 2.842,00 Y = 3.155,00 ENERGY = 5.567,00
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REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
GAMMA = 9.185-01 ALPHA = 4.318-02 BETA = -3.929-01 Z = 6.000,00 X = 2.821-01 Y = -2.567,00 ENERGY = 1.415,01
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REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 6
GAMMA = 4.196-01 ALPHA = -2.092-01 BETA = 8.833-01 Z = 6.000,00 X = -3.536-01 Y = 3.837-01 ENERGY = 1.567,00
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REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
GAMMA = 8.394-01 ALPHA = 3.347-01 BETA = -4.282-01 Z = 6.000,00 X = 2.392,00 Y = -3.061,00 ENERGY = 1.265,01
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REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 6
GAMMA = 9.622-01 ALPHA = -2.639-01 BETA = 6.794-02 Z = 6.000,00 X = 1.422,00 Y = -2.224,00 ENERGY = 1.201,01
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REGION 13 NEUTRONS 2.000,00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
GAMMA = 7.726-01 ALPHA = 1.189-02 BETA = -2.321-01 Z = 6.000,00 X = 7.335-02 Y = -1.432,00 ENERGY = 1.005,01
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REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
GAMMA = 9.831-01 ALPHA = -1.446-01 BETA = 1.123-01 Z = 6.000,00 X = -8.828-01 Y = 6.852-01 ENERGY = 9.147,00
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REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
GAMMA = 9.880-01 ALPHA = 8.510-02 BETA = 1.288-01 Z = 6.000,00 X = 5.168-01 Y = 7.825-01 ENERGY = 1.326,01
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REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
GAMMA = 8.830-01 ALPHA = -1.066-01 BETA = 4.571-01 Z = 6.000,00 X = -7.243-01 Y = 3.106,00 ENERGY = 8.298,00
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REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
GAMMA = 7.940-01 ALPHA = 2.694-01 BETA = 5.450-01 Z = 6.000,00 X = 2.036,00 Y = 4.119,00 ENERGY = 1.001,01
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REGION 13 NEUTRONS 8.245-01 COLLISIONS 13 COLLISIONS OCCURED IN REGIONS 1 3 7 4 5 6 6 6
GAMMA = 3.422-01 ALPHA = -4.343-01 BETA = 8.332-01 Z = 6.000,00 X = -6.611-01 Y = 3.077,00 ENERGY = 1.000-01
-----
REGION 13 NEUTRONS 1.000-00 COLLISIONS 3 COLLISIONS OCCURED IN REGIONS 1 7 7 7
GAMMA = 9.843-01 ALPHA = 1.417-01 BETA = -1.055-01 Z = 6.000,00 X = -2.238,00 Y = 8.567-01 ENERGY = 1.218,00
-----
REGION 13 NEUTRONS 9.837-01 COLLISIONS 9 COLLISIONS OCCURED IN REGIONS 1 1 1 1 3 7 4 6
GAMMA = 2.676-01 ALPHA = -7.413-03 BETA = -9.635-01 Z = 6.000,00 X = -1.004,00 Y = -2.759,00 ENERGY = 1.707-01
-----
REGION 13 NEUTRONS 9.474-01 COLLISIONS 3 COLLISIONS OCCURED IN REGIONS 1 1 1 3
GAMMA = 8.749-01 ALPHA = -1.275-01 BETA = -4.671-01 Z = 6.000,00 X = -1.886,00 Y = -2.268,00 ENERGY = 5.021-01
-----
REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 5
GAMMA = 7.600-01 ALPHA = -5.559-01 BETA = 3.367-01 Z = 6.000,00 X = -5.464-01 Y = 1.686,00 ENERGY = 9.967-01
-----
REGION 13 NEUTRONS 7.537-01 COLLISIONS 8 COLLISIONS OCCURED IN REGIONS 1 7 3 3 4 3 4 5
GAMMA = 6.967-01 ALPHA = -6.361-01 BETA = 3.317-01 Z = 6.000,00 X = -9.160-01 Y = -1.667,00 ENERGY = 1.000-01
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REGION 13 NEUTRONS 9.535-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 J
 GAMMA = 9.651-01 ALPHA = -1.086-01 BETA = 2.382-01 Z = 6.000,00 X = -6.328-01 Y = 1.404,00 ENERGY = 7.366,00

REGION 13 NEUTRONS 5.162-01 COLLISIONS 6 COLLISIONS OCCURED IN REGIONS 1 3 3 3 6 6 6
 GAMMA = 2.556-01 ALPHA = 9.273-01 BETA = 2.735-01 Z = 6.000,00 X = 3.339,00 Y = 1.195,00 ENERGY = 5.132-01

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 8.522-01 ALPHA = 5.199-01 BETA = 5.928-02 Z = 6.000,00 X = 3.661,00 Y = 4.174-01 ENERGY = 9.071,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 5
 GAMMA = 7.487-01 ALPHA = 3.225-01 BETA = -5.792-01 Z = 6.000,00 X = 3.146,00 Y = -2.001,00 ENERGY = 9.071,00

REGION 13 NEUTRONS 9.899-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 1 1
 GAMMA = 8.517-01 ALPHA = 4.477-01 BETA = -2.724-01 Z = 6.000,00 X = 3.238,00 Y = -1.988,00 ENERGY = 6.064,00

REGION 13 NEUTRONS 9.637-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 1
 GAMMA = 8.294-01 ALPHA = -2.821-01 BETA = -4.823-01 Z = 6.000,00 X = -1.935,00 Y = -4.268,00 ENERGY = 1.108,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 9.957-01 ALPHA = -8.876-03 BETA = 9.235-02 Z = 6.000,00 X = -5.349-02 Y = 5.565-01 ENERGY = 1.034,01

REGION 13 NEUTRONS 8.532-01 COLLISIONS 3 COLLISIONS OCCURED IN REGIONS 1 2 1 6
 GAMMA = 9.949-01 ALPHA = -9.740-02 BETA = -2.439-02 Z = 6.000,00 X = 2.213,00 Y = -1.023,00 ENERGY = 4.683,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 7
 GAMMA = 9.598-01 ALPHA = -7.200-02 BETA = 2.712-01 Z = 6.000,00 X = 2.891,00 Y = -4.248-01 ENERGY = 9.290,00

REGION 13 NEUTRONS 8.012-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 3 4
 GAMMA = 8.927-01 ALPHA = 2.363-01 BETA = 3.838-01 Z = 6.000,00 X = -1.976,00 Y = 2.503,00 ENERGY = 4.103,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 6
 GAMMA = 5.812-01 ALPHA = 6.214-01 BETA = -5.255-01 Z = 6.000,00 X = -2.467,00 Y = -1.913,00 ENERGY = 8.354,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 7.925-01 ALPHA = 3.122-01 BETA = 5.239-01 Z = 6.000,00 X = 2.363,00 Y = 3.966,00 ENERGY = 1.048,01

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 7.838-01 ALPHA = -4.674-01 BETA = -4.089-01 Z = 6.000,00 X = -3.578,00 Y = -3.130,00 ENERGY = 8.388,00

REGION 13 NEUTRONS 2.000,00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 7.721-01 ALPHA = 6.000-01 BETA = -2.096-01 Z = 6.000,00 X = 4.663,00 Y = -1.629,00 ENERGY = 1.071,01

REGION 13 NEUTRONS 9.891-01 COLLISIONS 5 COLLISIONS OCCURED IN REGIONS 1 1 2 2 6 6
 GAMMA = 8.459-01 ALPHA = -5.272-01 BETA = -8.100-02 Z = 6.000,00 X = -1.608,00 Y = 1.366,00 ENERGY = 4.528-01

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 8.486-01 ALPHA = -5.203-01 BETA = -9.540-02 Z = 6.000,00 X = -3.679,00 Y = -6.745-01 ENERGY = 7.556,00

REGION 13 NEUTRONS 7.685-01 COLLISIONS 5 COLLISIONS OCCURED IN REGIONS 1 1 3 6 6 6
 GAMMA = 7.305-01 ALPHA = 5.266-01 BETA = -4.348-01 Z = 6.000,00 X = -1.349-01 Y = -8.237-01 ENERGY = 2.436,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 7 6
 GAMMA = 3.987-01 ALPHA = 9.121-01 BETA = -9.535-02 Z = 6.000,00 X = -6.587-01 Y = 1.330-01 ENERGY = 8.396,00

REGION 13	NEUTRONS 2.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 1						
GAMMA =	8.176-01	ALPHA = -4.039-01	BETA = -4.103-01	Z = 6.000,00	X = -2.964,00	Y = -3.011,00	ENERGY = 1.231,01		
REGION 13	NEUTRONS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 1						
GAMMA =	8.492-01	ALPHA = -5.277-01	BETA = 1.629-02	Z = 6.000,00	X = -3.729,00	Y = 1.151-01	ENERGY = 6.686,00		
REGION 13	NEUTRONS 1.000-00	COLLISIONS 2	COLLISIONS OCCURED IN REGIONS 1 4 4						
GAMMA =	9.572-01	ALPHA = 2.764-01	BETA = -8.550-02	Z = 6.000,00	X = 7.872-01	Y = 2.760,00	ENERGY = 3.146-01		
REGION 13	NEUTRONS 9.996-01	COLLISIONS 4	COLLISIONS OCCURED IN REGIONS 1 5 5 7 7						
GAMMA =	8.048-01	ALPHA = -5.053-01	BETA = -3.114-01	Z = 6.000,00	X = -4.590-01	Y = 3.882,00	ENERGY = 5.084-01		
REGION 13	NEUTRONS 2.000,00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 1						
GAMMA =	9.238-01	ALPHA = -2.241-01	BETA = -3.105-01	Z = 6.000,00	X = -1.456,00	Y = -2.017,00	ENERGY = 5.736,00		
REGION 13	NEUTRONS 1.000-00	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS 1 6						
GAMMA =	8.533-01	ALPHA = 3.677-01	BETA = 3.697-01	Z = 6.000,00	X = 1.426-01	Y = 3.396,00	ENERGY = 9.332,00		
REGION 13	NEUTRONS 2.000,00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 1						
GAMMA =	8.400-01	ALPHA = -1.085-03	BETA = 5.425-01	Z = 6.000,00	X = -7.749-03	Y = 3.875,00	ENERGY = 5.930,00		
REGION 13	NEUTRONS 8.136-01	COLLISIONS 2	COLLISIONS OCCURED IN REGIONS 1 3 6						
GAMMA =	7.374-01	ALPHA = 4.387-01	BETA = 5.136-01	Z = 6.000,00	X = 4.541,00	Y = 1.514,00	ENERGY = 2.555,00		
REGION 13	NEUTRONS 9.984-01	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS 1 1						
GAMMA =	9.994-01	ALPHA = -9.417-03	BETA = -3.385-02	Z = 6.000,00	X = 7.805-01	Y = -8.461-01	ENERGY = 6.243,00		
REGION 13	NEUTRONS 9.463-01	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS 1 1						
GAMMA =	8.936-01	ALPHA = 1.626-01	BETA = 4.183-01	Z = 6.000,00	X = 1.926,00	Y = 1.556,00	ENERGY = 1.665,00		
REGION 13	NEUTRONS 9.584-01	COLLISIONS 9	COLLISIONS OCCURED IN REGIONS 1 1 4 1 4 1 6 6						
GAMMA =	3.210-01	ALPHA = 1.273-02	BETA = 9.470-01	Z = 6.000,00	X = -1.894,00	Y = 4.050,00	ENERGY = 1.816,00		
REGION 13	NEUTRONS 9.997-01	COLLISIONS 3	COLLISIONS OCCURED IN REGIONS 1 4 3 7						
GAMMA =	9.760-01	ALPHA = -6.737-02	BETA = -2.069-01	Z = 6.000,00	X = 2.806,00	Y = -3.261,00	ENERGY = 3.222-01		
REGION 13	NEUTRONS 8.964-01	COLLISIONS 3	COLLISIONS OCCURED IN REGIONS 1 1 1 6						
GAMMA =	7.428-01	ALPHA = 6.665-01	BETA = -6.285-02	Z = 6.000,00	X = 1.354,00	Y = 1.938,00	ENERGY = 9.503,00		
REGION 13	NEUTRONS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 1						
GAMMA =	7.866-01	ALPHA = 5.611-01	BETA = -2.576-01	Z = 6.000,00	X = 4.280,00	Y = -1.965,00	ENERGY = 9.325,00		
REGION 13	NEUTRONS 9.718-01	COLLISIONS 2	COLLISIONS OCCURED IN REGIONS 1 3 6						
GAMMA =	9.524-01	ALPHA = -1.093-01	BETA = -2.845-01	Z = 6.000,00	X = 2.405,00	Y = -2.186,00	ENERGY = 6.293,00		
REGION 13	NEUTRONS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 1						
GAMMA =	9.494-01	ALPHA = 2.987-01	BETA = 9.713-02	Z = 6.000,00	X = 1.888,00	Y = 6.139-01	ENERGY = 1.389,01		
REGION 13	NEUTRONS 9.712-01	COLLISIONS 12	COLLISIONS OCCURED IN REGIONS 1 1 1 9 5 6 6 6						
GAMMA =	4.353-01	ALPHA = -4.938-01	BETA = -7.528-01	Z = 6.000,00	X = -2.767,00	Y = 1.125-01	ENERGY = 1.668-01		
REGION 13	NEUTRONS 1.000-00	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS 1 5						
GAMMA =	8.500-01	ALPHA = -2.429-01	BETA = -4.674-01	Z = 6.000,00	X = -1.222,00	Y = -5.406-01	ENERGY = 1.487-01		

REGION 13 NEUTRONS 9.427-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 4 7
 GAMMA = 7.805-01 ALPHA = 5.199-01 BETA = -3.473-01 Z = 6.000,00 X = -1.531-01 Y = 6.039-01 ENERGY = 4.818,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 9.812-01 ALPHA = -1.879-01 BETA = 4.320-02 Z = 6.000,00 X = -1.149,00 Y = 2.641-01 ENERGY = 1.007,01

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 9.695-01 ALPHA = 2.153-01 BETA = -1.173-01 Z = 6.000,00 X = 1.332,00 Y = -7.258-01 ENERGY = 1.179,01

REGION 13 NEUTRONS 7.915-01 COLLISIONS 5 COLLISIONS OCCURED IN REGIONS 1 1 1 2 5 7
 GAMMA = 9.510-01 ALPHA = 2.935-02 BETA = 3.077-01 Z = 6.000,00 X = 3.888,00 Y = 2.835,00 ENERGY = 2.878,00

REGION 13 NEUTRONS 8.123-01 COLLISIONS 4 COLLISIONS OCCURED IN REGIONS 1 1 3 1 6
 GAMMA = 8.170-01 ALPHA = 4.977-01 BETA = 2.913-01 Z = 6.000,00 X = 1.905,00 Y = 2.369,00 ENERGY = 5.951,00

REGION 13 NEUTRONS 4.786-01 COLLISIONS 9 COLLISIONS OCCURED IN REGIONS 1 3 3 6 6 6 6 5
 GAMMA = 8.219-01 ALPHA = 5.695-01 BETA = 1.516-02 Z = 6.000,00 X = 1.888,00 Y = -3.331,00 ENERGY = 1.000-01

REGION 13 NEUTRONS 9.500-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 1
 GAMMA = 9.411-01 ALPHA = 2.589-01 BETA = -2.175-01 Z = 6.000,00 X = 1.491,00 Y = -7.989-01 ENERGY = 6.717,00

REGION 13 NEUTRONS 7.477-01 COLLISIONS 6 COLLISIONS OCCURED IN REGIONS 1 1 1 1 1 1 2
 GAMMA = 7.837-01 ALPHA = -5.910-01 BETA = -1.911-01 Z = 6.000,00 X = -4.010,00 Y = 1.031-01 ENERGY = 2.373,00

REGION 13 NEUTRONS 8.481-01 COLLISIONS 7 COLLISIONS OCCURED IN REGIONS 1 1 1 3 4 7 2 6
 GAMMA = 9.822-01 ALPHA = -1.875-01 BETA = 8.965-03 Z = 6.000,00 X = -1.508,00 Y = -1.848,00 ENERGY = 3.648-01

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 7.992-01 ALPHA = -6.003-01 BETA = 3.175-02 Z = 6.000,00 X = -4.507,00 Y = 2.384-01 ENERGY = 1.369,01

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 7.965-01 ALPHA = 7.322-02 BETA = -6.002-01 Z = 6.000,00 X = 5.516-01 Y = -4.522,00 ENERGY = 7.328,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 7.903-01 ALPHA = 4.682-01 BETA = 3.953-01 Z = 6.000,00 X = 3.554,00 Y = 3.002,00 ENERGY = 8.745,00

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 9.629-01 ALPHA = 2.249-01 BETA = 1.490-01 Z = 6.000,00 X = 1.401,00 Y = 9.282-01 ENERGY = 1.443,01

REGION 13 NEUTRONS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 8.534-01 ALPHA = 1.520-02 BETA = -5.210-01 Z = 6.000,00 X = 1.069-01 Y = -3.663,00 ENERGY = 1.011,01

REGION 13 NEUTRONS 8.364-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 1 2
 GAMMA = 8.667-01 ALPHA = 4.287-01 BETA = -2.552-01 Z = 6.000,00 X = 2.700,00 Y = -2.383,00 ENERGY = 1.007,01

REGION 13 NEUTRONS 9.482-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 1
 GAMMA = 8.037-01 ALPHA = 5.906-01 BETA = -7.185-02 Z = 6.000,00 X = 4.866,00 Y = -6.142-01 ENERGY = 6.245,00

REGION 13 NEUTRONS 8.160-01 COLLISIONS 3 COLLISIONS OCCURED IN REGIONS 1 1 3 4
 GAMMA = 8.438-01 ALPHA = -1.076-01 BETA = 5.258-01 Z = 6.000,00 X = 2.808,00 Y = 6.878-01 ENERGY = 5.103-01

STOP TIME BEFORE MACHINE RESTORED TO STANDARD SET-UP 10.28
--
***** STOP TIME 10.29 *****

12031

12031 J. J. LOCHLER FMC GAMMA RAY 00515 UNIT 4313 P CC
 ++ TAPES 5 6 7(0067) 9
 ++ FOR PROGRAM STOP OR TIME EXCEEDED TRANSFER TO 10210
 ++ ON TIME, FORCE PM SAVE 7
 ID. 1, BG, 4725, CH, 53259 DATE 01/19/61, TAPES 7, PUN 0, CDR C
 ***** START TIME 05.82 ***** STOP TIME *****
 ++ STARTING TIME AFTER MACHINE SETUP STOP 05.86

NUMBER OF RECORDS ON HISTORY TALLY TAPE	1
NUMBER OF PARTICLES IN EACH RECORD	1582
NUMBER OF PARTICLES IN LAST RECORD	74

NUMBER OF COLLISION CUTOFFS AFTER ENERGY CUTOFF	2.06906-01
NUMBER OF WEIGHT CUTOFFS	3.04283,00
LAST RANDOM NUMBER	16664092607

TOTAL NUMBER OF GAMMA RAYS	8.86000,02
TOTAL SOURCE ENERGY	3.35051,03

REGION 1

ENERGY	ABSORPTION	MONTE CARLO FLUX
0.00000,00	7.91904,00	1.25117,00
1.00000-01	1.41015,01	2.72368,01
1.50000-01	4.22390,00	2.79955,01
2.00000-01	3.37515,00	5.68001,01
3.00000-01	8.12930-01	3.91097,01
4.00000-01	3.17166-01	2.78800,01
5.00000-01	6.90277-02	1.94648,01
6.00000-01	1.03298-01	2.44900,01
8.00000-01	8.52255-02	3.82323,01
1.00000-02	3.07757-02	7.99190,01
1.50000,00	0.00000,00	7.03745,01
2.00000,00	1.48374-03	1.51809,02
3.00000,00	3.44352-03	9.96955,01
4.00000,00	8.89492-03	8.39889,01
5.00000,00	1.81995-02	9.69800,01
6.00000,00	5.70646-02	1.28382,02
8.00000,00	2.87857-02	5.71169,01
*****	*****	*****
TOTALS	3.11559,01	1.03073,03

REGION 2

ENERGY	ABSCPTION	MONTI CARLO FLUX	MONTI CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	1.83576-01	0.00000,00	2.68798-C1
1.00000-01	0.00000,00	2.51002,00	2.73075,00	1.59834,00
1.50000-01	0.00000,00	4.26563,00	0.00000,00	2.41037-C1
2.00000-01	0.00000,00	5.62463,00	4.80850,00	4.18406,00
3.00000-01	0.00000,00	2.81991,00	1.25181,00	2.19593,00
4.00000-01	0.00000,00	1.35911,00	0.00000,00	2.87692-C1
5.00000-01	0.00000,00	2.16365,00	2.00000,00	1.30607,00
6.00000-01	0.00000,00	6.56632,00	2.99509,00	3.97635,00
8.00000-01	0.00000,00	5.38467,00	2.21434,00	3.33671,00
1.00000-00	0.00000,00	9.78447,00	4.99890,00	4.50412,00
1.50000,00	0.00000,00	6.85281,00	3.29964,00	3.92102,00
2.00000,00	4.12882-05	1.81531,01	1.49990,01	1.52428,01
3.00000,00	5.11309-04	1.81502,01	1.19983,01	1.10531,01
4.00000,00	1.71175-03	1.33764,01	1.00000,01	8.59920,00
5.00000,00	0.00000,00	9.74805,00	8.92705,00	9.34925,00
6.00000,00	6.47342-03	1.76200,01	1.10000,01	1.14075,01
8.00000,00	0.00000,00	7.38591,00	6.00000,00	5.97466,00
*****	*****	*****	*****	*****
TOTALS	8.73777-03	1.32649,02	8.93234,01	8.74466,01

REGION 3

ENERGY	ABSORPTION	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	1.54285-02	5.06680,00	1.69839,00	1.46282,00
1.00000-01	0.00000,00	2.97503,01	9.77714,00	9.87294,00
1.50000-01	0.00000,00	4.25247,01	2.25231,01	1.91135,01
2.00000-01	0.00000,00	6.55587,01	2.96660,01	2.84737,01
3.00000-01	0.00000,00	2.02245,01	1.67518,01	1.55926,01
4.00000-01	0.00000,00	3.08853,01	1.81074,01	1.99119,01
5.00000-01	0.00000,00	1.53949,01	1.09540,01	1.00090,01
6.00000-01	0.00000,00	3.75764,01	1.78636,01	1.87891,01
8.00000-01	0.00000,00	3.43688,01	2.18898,01	2.20825,01
1.00000-00	0.00000,00	7.74795,01	4.38169,01	4.22700,01
1.50000,00	0.00000,00	7.29080,01	3.68898,01	3.70981,01
2.00000,00	1.56934-04	1.11841,02	5.99748,01	6.32620,01
3.00000,00	2.51942-03	1.06986,02	5.29273,01	4.77676,01
4.00000,00	1.73651-03	8.88797,01	5.29885,01	5.23465,01
5.00000,00	8.04311-03	6.93448,01	3.69235,01	3.61216,01
6.00000,00	3.17602-02	7.76037,01	5.40000,01	5.69934,01
8.00000,00	0.00000,00	4.42344,01	2.90000,01	2.77699,01
*****	*****	*****	*****	*****
TOTALS	5.96447-02	7.33222,02	5.15814,02	5.08944,02

REGION 4

ENERGY	ABSORPTION	Monte Carlo FLUX	Monte Carlo Entrance	Expectation Entrance
0.00000,00	1.08183,01	8.68502,01	1.82223,00	2.09314,00
1.00000-01	2.26924,01	2.48142,01	1.04673,01	1.08738,01
1.50000-01	1.33748,01	3.97145,01	2.26252,01	2.11457,01
2.00000-01	6.90605,00	5.45101,01	2.76733,01	2.48120,01
3.00000-01	1.14471,00	4.26449,01	1.08755,01	1.00131,01
4.00000-01	6.95001-01	3.16095,01	1.28031,01	1.21700,01
5.00000-01	3.73721-01	3.79026,01	1.29615,01	9.77150,00
6.00000-01	2.85841-01	5.07140,01	2.28976,01	2.27871,01
8.00000-01	1.79681-01	5.27104,01	1.99758,01	1.96278,01
1.00000-00	1.37954-01	7.50165,01	4.99800,01	4.83794,01
1.50000,00	0.00000,00	6.57125,01	4.37648,01	4.33969,01
2.00000,00	1.26519-03	1.05038,02	6.19746,01	6.33015,01
3.00000,00	4.78549-03	8.65677,01	5.59849,01	5.29837,01
4.00000,00	1.30465-02	8.25750,01	5.59008,01	5.39738,01
5.00000,00	3.29006-02	6.29749,01	3.99235,01	4.00421,01
6.00000,00	3.58462-02	7.60680,01	5.10000,01	5.83888,01
8.00000,00	5.77626-02	3.63729,01	2.60000,01	2.47237,01
*****	*****	*****	*****	*****
TOTALS	5.67541,01	9.16022,02	5.26630,02	5.18484,02

REGION 5

ENERGY	ABSORPTION	MCNTE CARLC FLUY	MCNTE CARLC ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	0.00000,00	0.00000,00	2.76442-C1
1.00000-01	3.81871,00	9.41102-C2	3.33841,00	2.38676,00
1.50000-01	5.50212,00	4.79357-C1	2.45049,00	3.10971,00
2.00000-01	1.75901,01	3.07901,00	9.89494,00	9.69476,00
3.00000-01	2.63437,00	2.14142,00	1.97390,00	3.93279,00
4.00000-01	2.58859,00	3.53444,00	2.90654,00	3.76950,00
5.00000-01	4.02405,00	7.18526,00	6.28385,00	4.68920,00
6.00000-01	3.48603,00	8.97370,00	6.98199,00	6.31165,00
8.00000-01	2.51806,00	9.89822,00	7.97758,00	7.78866,00
1.00000-00	1.93445,00	1.46041,01	1.19987,01	1.36953,01
1.50000,00	1.40666,00	1.48123,01	1.19293,01	1.01685,01
2.00000,00	1.14017,00	2.76170,01	2.29981,01	2.47885,01
3.00000,00	8.72344-01	2.78275,01	1.79854,01	2.17732,01
4.00000,00	4.37502-01	1.88570,01	1.30000,01	1.18703,01
5.00000,00	4.02961-01	1.63277,01	1.49259,01	1.53665,01
6.00000,00	5.35216-01	3.13985,01	2.20000,01	2.35659,01
8.00000,00	7.29512-C2	1.54789,01	1.20000,01	1.02225,01
*****	*****	*****	*****	*****
TOTALS	5.09643,01	2.02310,02	1.69485,02	1.73411,02

REGION 6

ENERGY	ABSORPTION	MCNTE CARLO FLUX	MCNTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	0.00000,00	0.00000,00	0.00000,00	0.00000,00
1.00000-01	5.06483-03	2.94934,00	0.00000,00	9.09832-25
1.50000-01	1.08918-03	8.75985,00	3.00940,00	2.59816,00
2.00000-01	0.00000,00	1.98024,00	1.36903-01	6.01629-01
3.00000-01	0.00000,00	7.23514,00	3.40864,00	2.66786,00
4.00000-01	0.00000,00	5.13589,00	8.87541-01	1.22892,00
5.00000-01	0.00000,00	4.29744,00	2.20548,00	1.28772,00
6.00000-01	0.00000,00	1.17154,01	4.89317,00	4.87049,00
8.00000-01	0.00000,00	6.77977,00	2.28140,00	2.43457,00
1.00000-00	0.00000,00	2.26656,01	7.69611,00	5.29006,00
1.50000,00	0.00000,00	1.20126,01	4.20291,00	5.16343,00
2.00000,00	1.28819-04	3.01661,01	1.65757,01	1.51346,01
3.00000,00	2.97806-04	2.16794,01	1.18165,01	1.37185,01
4.00000,00	0.00000,00	2.04499,01	7.92034,00	6.40715,00
5.00000,00	5.45331-03	1.81219,01	8.00000,00	9.30236,00
6.00000,00	1.73968-02	3.36821,01	1.50000,01	1.47906,01
8.00000,00	0.00000,00	2.81431,01	9.00000,00	6.21539,00
*****	*****	*****	*****	*****
TOTALS	2.94307-02	2.35814,02	7.84340,01	9.17114,01

REGION 7

ENERGY	Absorption	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	4.58345-02	2.71291,00	1.32833-01	1.58762-C1
1.00000-01	2.20953-02	1.60703,01	7.15544,00	7.56780,00
1.50000-01	3.50207-03	1.78502,01	2.45748,00	1.16820,01
2.00000-01	0.00000,00	3.98776,01	1.32704,01	1.73655,01
3.00000-01	0.00000,00	3.27000,01	2.06615,01	2.10282,01
4.00000-01	0.00000,00	2.52222,01	1.49136,01	1.60417,01
5.00000-01	0.00000,00	3.25978,01	2.43709,01	1.62866,01
6.00000-01	0.00000,00	5.86604,01	3.60295,01	3.19025,01
8.00000-01	0.00000,00	4.45116,01	2.65413,01	2.66247,01
1.00000-00	0.00000,00	7.99205,01	4.56544,01	4.85275,01
1.50000,00	0.00000,00	5.75880,01	4.56670,01	4.41328,01
2.00000,00	3.94563-04	1.12756,02	7.65359,01	7.57167,01
3.00000,00	1.50231-03	8.30442,01	6.27920,01	6.11478,01
4.00000,00	4.22631-03	7.88576,01	5.98825,01	5.94298,01
5.00000,00	3.58143-03	5.38807,01	3.29124,01	4.04146,01
6.00000,00	1.98186-02	9.24989,01	6.40000,01	6.46140,01
8.00000,00	0.00000,00	3.93325,01	2.89920,01	2.88656,01
*****	*****	*****	*****	*****
TOTALS	1.00961-01	8.83088,02	5.68983,02	5.72207,02

REGION 8

ENERGY	ABSORPTION	Monte Carlo FLUX	Monte Carlo Entrance	Expectation Entrance
0.00000,00	0.00000,00	7.80311-C2	1.00000-01	1.66856-C1
1.00000-01	0.00000,00	1.25114,00	1.02890,00	1.61392,00
1.50000-01	0.00000,00	3.17444,00	2.19648,00	3.22211,00
2.00000-01	0.00000,00	6.69428,00	4.46639,00	4.26514,00
3.00000-01	0.00000,00	3.62622,00	3.27490,00	3.37376,00
4.00000-01	0.00000,00	8.95388-C1	1.00000-00	8.92926-C1
5.00000-01	0.00000,00	2.28663,00	2.29844,00	2.14056,00
6.00000-01	0.00000,00	8.78744-C1	0.00000,00	7.07384-C1
8.00000-01	0.00000,00	4.96603,00	3.99959,00	4.62086,00
1.00000-00	0.00000,00	1.05181,01	8.99816,00	6.37797,00
1.50000,00	0.00000,00	3.64779,00	3.00000,00	5.69633,00
2.00000,00	2.10705-C4	1.90639,01	1.19982,01	9.27336,00
3.00000,00	0.00000,00	4.77386,00	4.00000,00	5.76267,00
4.00000,00	6.10161-C4	1.43163,01	0.00000,00	8.87944,00
5.00000,00	0.00000,00	1.52623,01	1.10000,01	7.64398,00
6.00000,00	0.00000,00	8.14313,00	7.00000,00	6.02225,00
8.00000,00	0.00000,00	1.00311,01	0.00000,00	6.91436,00
*****	*****	*****	*****	*****
TOTALS	8.20866-C4	1.09827,02	8.17611,01	8.02939,01

REGION 9

ENERGY	ABSORPTION	MONTE CARLO FLUX	MONTE CARLO ENTRANCE	EXPECTATION ENTRANCE
0.00000,00	1.74333,00	3.22979-C2	1.69456,00	1.28587,00
1.00000-01	6.01341,00	1.62813-01	5.36060,00	4.17859,00
1.50000-01	3.73223,00	2.22026-01	2.29351,00	4.89835,00
2.00000-01	8.68876,00	1.60516,00	4.78855,00	6.28618,00
3.00000-01	6.57526,00	2.73612,00	2.99939,00	3.77811,00
4.00000-01	3.99603,00	3.40970,00	3.95503,00	2.95979,00
5.00000-01	2.50360,00	3.94987,00	5.98339,00	5.09576,00
6.00000-01	1.61686,00	3.21454,00	1.99508,00	2.93370,00
8.00000-01	7.68278-01	3.83922,00	3.99959,00	4.14053,00
1.00000-00	1.18422,00	8.70985,00	1.18319,01	9.88935,00
1.50000,00	5.15400-01	4.48234,00	4.00000,00	6.26240,00
2.00000,00	7.43342-01	1.49703,01	1.19974,01	1.06456,01
3.00000,00	1.94875-01	9.61446,00	7.00000,00	7.96426,00
4.00000,00	2.63155-01	1.67933,01	1.30000,01	1.28347,01
5.00000,00	2.42440-01	1.37282,01	1.00000,01	8.37024,00
6.00000,00	3.82330-01	1.19445,01	1.20000,01	1.02104,01
8.00000,00	0.00000,00	8.27328,00	8.00000,00	7.45998,00
*****	*****	*****	*****	*****
TOTALS	3.91635,01	1.07688,02	1.10899,02	1.09194,02

REGION 10

ENERGY	DESCRIPTION	MONTE CARLO FLUX	MONTE CARLO ENRANCE	EXPECTATION ENRANCE
0.00000,00	0.00000,00	0.00000,00	0.00000,00	6.32070-28
1.00000-01	0.00000,00	1.55659,00	9.03196-01	5.54693-01
1.50000-01	6.99190-09	1.38182,00	0.00000,00	6.98979-03
2.00000-01	0.00000,00	5.27090,00	0.00000,00	1.44479-01
3.00000-01	0.00000,00	9.93265,00	9.66438-01	7.86319-01
4.00000-01	0.00000,00	1.98532,00	0.00000,00	6.41682-01
5.00000-01	0.00000,00	3.08737,00	3.98835,00	2.28211,00
6.00000-01	0.00000,00	6.20698,00	4.07348,00	4.38397,00
8.00000-01	0.00000,00	6.58542,00	4.58182,00	5.54758,00
1.00000-00	0.00000,00	1.06403,01	6.83258,00	6.51769,00
1.50000,00	0.00000,00	4.41173,00	4.73453,00	5.36243,00
2.00000,00	1.36494-04	1.17398,01	9.82208,00	1.01211,01
3.00000,00	4.77464-04	1.27169,01	9.82561,00	9.65324,00
4.00000,00	0.00000,00	1.28366,01	1.00000,01	9.39654,00
5.00000,00	0.00000,00	1.38747,01	7.92554,00	7.76058,00
6.00000,00	2.33978-03	5.88773,00	4.00000,00	6.47463,00
8.00000,00	0.00000,00	1.07617,01	9.00000,00	7.09110,00
*****	*****	*****	*****	*****
TOTALS	3.65293-03	1.18879,02	7.66536,01	7.67251,01

REGION 11

ENERGY	Monte Carlo LEAKAGE	EXPECTATION LEAKAGE
0.00000,00	0.00000,00	0.00000,00
1.00000-01	9.03196-C1	4.54104-01
1.50000-01	9.27891-C1	1.61746,00
2.00000-01	9.03335-C1	1.00679,00
3.00000-01	9.66438-C1	1.62521,00
4.00000-01	0.00000,00	1.07888,00
5.00000-01	3.98835,00	1.28897,00
6.00000-01	3.07365,00	3.86507,00
8.00000-01	5.57877,00	4.06718,00
1.00000-00	9.83258,00	7.81588,00
1.50000,00	2.85846,00	4.35664,00
2.00000,00	6.91871,00	8.96383,00
3.00000,00	7.92458,00	8.90044,00
4.00000,00	1.29977,C1	1.11340,01
5.00000,00	7.92554,00	7.34118,00
6.00000,00	6.00000,00	8.82947,00
8.00000,00	9.00000,00	6.72727,00
*****	*****	*****
TOTALS	7.98692,C1	7.96764,01

REGION 12

ENERGY	MCNTE CARLC LEAKAGE	EXPECTATION LEAKAGE
0.00000,00	4.63824,C1	5.36908-C1
1.00000-01	9.38728,00	9.69197,00
1.50000-01	1.24539,C1	1.38778,C1
2.00000-C1	1.43686,C1	1.67003,C1
3.00000-C1	2.44464,C1	2.24213,C1
4.00000-C1	1.52631,C1	1.60774,C1
5.00000-C1	2.18522,C1	1.84305,C1
6.00000-01	3.21941,C1	3.11637,C1
8.00000-01	2.84655,C1	2.73620,C1
1.00000-00	4.47460,C1	5.23990,01
1.50000,00	4.06661,C1	4.10702,C1
2.00000,00	7.27064,C1	7.23467,C1
3.00000,00	6.18028,C1	5.94517,C1
4.00000,00	5.58916,C1	5.73015,01
5.00000,00	3.29194,C1	3.92310,C1
6.00000,00	6.30000,C1	6.40716,01
8.00000,00	2.69920,C1	2.60753,C1
*****	*****	*****
TOTALS	5.57619,C2	5.68209,C2

REGION 13

ENERGY	MONTE CARLO LEAKAGE	EXPECTATION LEAKAGE
0.00000,00	0.00000,00	0.00000,00
1.00000-01	0.00000,00	0.00000,00
1.50000-01	6.70224-01	1.33768,00
2.00000-01	1.99047,00	1.33676,00
3.00000-01	2.18927,00	2.14867,00
4.00000-01	3.94702-01	8.32028-01
5.00000-01	5.35259-01	3.78921-01
6.00000-01	2.45755,00	3.82359,00
8.00000-01	2.98140,00	1.31016,00
1.00000-00	5.90486,00	4.82680,00
1.50000,00	3.00000,00	4.48695,00
2.00000,00	8.74875,00	1.01255,01
3.00000,00	6.89934,00	7.65545,00
4.00000,00	8.00000,00	5.35435,00
5.00000,00	7.99455,00	9.66444,00
6.00000,00	9.00000,00	1.13834,01
8.00000,00	1.00000,01	7.32529,00
*****	*****	*****
TOTALS	7.07664,01	7.19920,01

HISTORY TALLIES

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REGION 13	GAMMAS 9.186-01	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS	5	5		
	GAMMA = 5.745-01	ALPHA = 5.538-01	BETA = -6.026-01	Z = 6.000,00	X = 1.976,00	Y = -8.833-01	ENERGY = 2.835,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	6			
	GAMMA = 8.143-01	ALPHA = 4.825-01	BETA = -3.225-01	Z = 6.000,00	X = -3.176,00	Y = 2.266,00	ENERGY = 2.046,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	6			
	GAMMA = 9.586-01	ALPHA = 3.702-02	BETA = 2.822-01	Z = 6.000,00	X = -2.213,00	Y = 2.170,00	ENERGY = 4.028,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	3			
	GAMMA = 8.975-01	ALPHA = 2.744-01	BETA = 3.451-01	Z = 6.000,00	X = 4.458,00	Y = 1.877,00	ENERGY = 5.167,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	3			
	GAMMA = 7.362-01	ALPHA = -6.693-01	BETA = 1.001-01	Z = 6.000,00	X = -3.353,00	Y = -9.428-02	ENERGY = 4.652,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	2			
	GAMMA = 9.937-01	ALPHA = 4.156-02	BETA = -9.375-02	Z = 6.000,00	X = -6.764-01	Y = -1.122,00	ENERGY = 5.803,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	2			
	GAMMA = 8.323-01	ALPHA = -1.411-01	BETA = 5.360-01	Z = 6.000,00	X = -1.063,00	Y = 1.532,00	ENERGY = 2.537,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	1			
	GAMMA = 8.091-01	ALPHA = 3.025-01	BETA = 5.002-01	Z = 6.000,00	X = 2.259,00	Y = 3.640,00	ENERGY = 1.167,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	4			
	GAMMA = 9.323-01	ALPHA = 3.607-01	BETA = 2.629-02	Z = 6.000,00	X = 2.574-01	Y = -1.725,00	ENERGY = 8.976,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	1			
	GAMMA = 9.034-01	ALPHA = 4.205-01	BETA = 8.391-02	Z = 6.000,00	X = 1.254,00	Y = 7.697-01	ENERGY = 4.621,00
REGION 13	GAMMAS 9.999-01	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS	3	7		
	GAMMA = 6.531-01	ALPHA = -2.844-01	BETA = 7.018-01	Z = 6.000,00	X = 2.870-01	Y = -3.839,00	ENERGY = 3.679-01
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	2			
	GAMMA = 9.824-01	ALPHA = -1.472-01	BETA = 1.150-01	Z = 6.000,00	X = -1.770-02	Y = 7.318-01	ENERGY = 2.757,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	5			
	GAMMA = 5.966-01	ALPHA = 6.866-01	BETA = -4.155-01	Z = 6.000,00	X = 6.678-01	Y = -9.941-01	ENERGY = 9.747,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	3			
	GAMMA = 9.000-01	ALPHA = -4.132-01	BETA = -1.386-01	Z = 6.000,00	X = -9.835-01	Y = -4.746-01	ENERGY = 3.093,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	2			
	GAMMA = 8.242-01	ALPHA = -4.036-01	BETA = 3.972-01	Z = 6.000,00	X = -3.300-01	Y = 1.607,00	ENERGY = 3.534,00
REGION 13	GAMMAS 1.000-00	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS	1	6		
	GAMMA = 6.665-01	ALPHA = 3.432-01	BETA = -6.617-01	Z = 6.000,00	X = 2.413,00	Y = -4.115-01	ENERGY = 1.213,00
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS	3			
	GAMMA = 9.484-01	ALPHA = 1.565-01	BETA = -2.758-01	Z = 6.000,00	X = 2.354,00	Y = 1.032,00	ENERGY = 2.947,00

REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 1					
GAMMA = 8.780-01 ALPHA = 4.737-01 BETA = 6.895-02 Z = 6.000,00 X = 3.950,00 Y = -1.002,00 ENERGY = 6.363,00								
REGION 13	GAMMAS 9.977-01	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS 3 6					
GAMMA = 8.966-01 ALPHA = -3.013-01 BETA = -3.244-01 Z = 6.000,00 X = 1.165,00 Y = -1.432,00 ENERGY = 5.613,00								
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 3					
GAMMA = 7.700-01 ALPHA = -4.952-01 BETA = -4.023-01 Z = 6.000,00 X = -1.377,00 Y = 1.146-01 ENERGY = 4.190,00								
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 1					
GAMMA = 9.176-01 ALPHA = 3.068-01 BETA = -2.528-01 Z = 6.000,00 X = 1.284,00 Y = -2.279,00 ENERGY = 5.313,00								
REGION 13	GAMMAS 9.058-01	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS 1 5					
GAMMA = 9.494-01 ALPHA = 6.941-02 BETA = 3.063-01 Z = 6.000,00 X = 1.125,00 Y = 3.018,00 ENERGY = 3.601,00								
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 4					
GAMMA = 8.383-01 ALPHA = 3.464-01 BETA = 4.211-01 Z = 6.000,00 X = 4.197,00 Y = -3.344-01 ENERGY = 1.586,00								
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 3					
GAMMA = 5.288-01 ALPHA = 5.995-01 BETA = -6.008-01 Z = 6.000,00 X = 3.023,00 Y = -3.616,00 ENERGY = 6.309,00								
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 1					
GAMMA = 9.072-01 ALPHA = 2.809-01 BETA = 3.133-01 Z = 6.000,00 X = 1.094,00 Y = 2.765,00 ENERGY = 5.105,00								
REGION 13	GAMMAS 9.922-01	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS 4 4					
GAMMA = 7.373-01 ALPHA = -5.598-01 BETA = 3.782-01 Z = 6.000,00 X = -2.491,00 Y = 4.551-01 ENERGY = 8.887-01								
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 1					
GAMMA = 9.253-01 ALPHA = 3.150-01 BETA = -2.113-01 Z = 6.000,00 X = 2.054,00 Y = -1.329,00 ENERGY = 3.191,00								
REGION 13	GAMMAS 9.891-01	COLLISIONS 3	COLLISIONS OCCURED IN REGIONS 3 4 4 4					
GAMMA = 8.382-01 ALPHA = -5.064-01 BETA = 2.026-01 Z = 6.000,00 X = -4.764,00 Y = -6.528-01 ENERGY = 8.673-01								
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 6					
GAMMA = 6.733-01 ALPHA = 6.574-01 BETA = 3.383-01 Z = 6.000,00 X = -1.568-01 Y = -2.876,00 ENERGY = 1.611,00								
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 6					
GAMMA = 3.650-01 ALPHA = 5.762-01 BETA = -7.313-01 Z = 6.000,00 X = 2.066,00 Y = -4.402,00 ENERGY = 8.956,00								
REGION 13	GAMMAS 9.935-01	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS 1 2					
GAMMA = 7.077-01 ALPHA = 5.252-01 BETA = -4.726-01 Z = 6.000,00 X = 3.193,00 Y = -3.392,00 ENERGY = 3.429,00								
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 1					
GAMMA = 8.320-01 ALPHA = 5.238-01 BETA = -1.829-01 Z = 6.000,00 X = 3.096,00 Y = -1.202,00 ENERGY = 6.320,00								
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 7					
GAMMA = 9.441-01 ALPHA = 2.930-01 BETA = -1.509-01 Z = 6.000,00 X = -9.591-02 Y = 3.359,00 ENERGY = 6.135,00								
REGION 13	GAMMAS 9.994-01	COLLISIONS 1	COLLISIONS OCCURED IN REGIONS 1 2					
GAMMA = 8.106-01 ALPHA = -4.318-01 BETA = -3.955-01 Z = 6.000,00 X = -7.857-01 Y = -3.015,00 ENERGY = 2.597,00								
REGION 13	GAMMAS 1.000-00	COLLISIONS -0	COLLISIONS OCCURED IN REGIONS 3					
GAMMA = 9.795-01 ALPHA = 2.004-01 BETA = -1.821-02 Z = 6.000,00 X = 3.341,00 Y = 1.192,00 ENERGY = 6.548,00								

REGION 13 GAMMAS 9.215-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 5
 GAMMA = 6.961-01 ALPHA = 7.163-01 BETA = -4.844-02 Z = 6.000,00 X = 2.042,00 Y = -7.695-01 ENERGY = 7.678-01

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 3
 GAMMA = 8.829-01 ALPHA = -3.313-02 BETA = -4.684-01 Z = 6.000,00 X = 2.041,00 Y = -4.446,00 ENERGY = 6.763,00

REGION 13 GAMMAS 9.999-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 4 4 4
 GAMMA = 9.999-01 ALPHA = -7.552-04 BETA = 3.780-02 Z = 6.000,00 X = -9.427-01 Y = 3.648,00 ENERGY = 1.380,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 4
 GAMMA = 7.956-01 ALPHA = 1.446-01 BETA = -5.883-01 Z = 6.000,00 X = -2.006,00 Y = -4.363,00 ENERGY = 4.542,00

REGION 13 GAMMAS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 6
 GAMMA = 7.187-01 ALPHA = -3.048-01 BETA = -6.249-01 Z = 6.000,00 X = -8.352-01 Y = -1.986,00 ENERGY = 1.013,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 9.327-01 ALPHA = -1.208-01 BETA = 3.398-01 Z = 6.000,00 X = -2.497,00 Y = 1.394,00 ENERGY = 8.776,00

REGION 13 GAMMAS 3.947-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 3 5 5
 GAMMA = 8.238-01 ALPHA = 2.561-01 BETA = 5.058-01 Z = 6.000,00 X = 2.079,00 Y = -7.256-01 ENERGY = 4.092-01

REGION 13 GAMMAS 9.968-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 4 6
 GAMMA = 8.643-01 ALPHA = -4.532-01 BETA = -2.181-01 Z = 6.000,00 X = 2.239,00 Y = -1.042,00 ENERGY = 5.379,00

REGION 13 GAMMAS 9.996-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 3 1
 GAMMA = 8.697-01 ALPHA = -1.524-01 BETA = -4.694-01 Z = 6.000,00 X = -1.883,00 Y = -2.886,00 ENERGY = 2.832,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 4
 GAMMA = 7.960-01 ALPHA = -5.836-01 BETA = -1.607-01 Z = 6.000,00 X = -1.934-01 Y = 8.274-02 ENERGY = 8.480,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 9
 GAMMA = 2.357-01 ALPHA = 5.338-03 BETA = -5.491-01 Z = 6.000,00 X = 2.716-03 Y = -3.593,00 ENERGY = 5.610,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 7
 GAMMA = 9.743-01 ALPHA = 2.187-01 BETA = 5.315-02 Z = 6.000,00 X = 4.762,00 Y = 7.252-01 ENERGY = 9.591,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 3
 GAMMA = 9.112-01 ALPHA = -3.871-02 BETA = 4.102-01 Z = 6.000,00 X = 1.673,00 Y = 2.396,00 ENERGY = 4.588,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 9.297-01 ALPHA = 2.059-01 BETA = 3.053-01 Z = 6.000,00 X = 8.508-01 Y = 1.753,00 ENERGY = 6.590,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 8.362-01 ALPHA = -5.252-01 BETA = 1.575-01 Z = 6.000,00 X = -4.539,00 Y = 9.784-01 ENERGY = 4.354,00

REGION 13 GAMMAS 9.921-01 COLLISIONS 3 COLLISIONS OCCURED IN REGIONS 4 4 4 7
 GAMMA = 9.979-01 ALPHA = -4.163-02 BETA = 4.892-02 Z = 6.000,00 X = -3.780,00 Y = 2.434,00 ENERGY = 2.476-01

REGION 13 GAMMAS 5.353-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 5 5 5
 GAMMA = 4.894-01 ALPHA = 2.054-01 BETA = -8.475-01 Z = 6.000,00 X = 3.490,00 Y = -2.174,00 ENERGY = 5.408-01

REGION 13 GAMMAS 9.050-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 8 1 5
 GAMMA = 5.759-01 ALPHA = -7.934-01 BETA = -1.972-01 Z = 6.000,00 X = -3.806,00 Y = -9.545-01 ENERGY = 1.129,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 5
 GAMMA = 4.441-01 ALPHA = 8.225-01 BETA = -3.555-01 Z = 6.000,00 X = 4.221,00 Y = 2.362-01 ENERGY = 9.002-01

REGION 13 GAMMAS 8.311-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 5 5
 GAMMA = 9.454-01 ALPHA = 3.002-01 BETA = 1.272-01 Z = 6.000,00 X = 3.022,00 Y = 3.053,00 ENERGY = 2.279,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 9.262-01 ALPHA = -1.155-01 BETA = 3.589-01 Z = 6.000,00 X = 5.553-01 Y = 1.342,00 ENERGY = 3.450,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 6
 GAMMA = 8.167-01 ALPHA = 1.294-01 BETA = 5.623-01 Z = 6.000,00 X = 1.279,00 Y = 2.017,00 ENERGY = 8.082,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 9.657-01 ALPHA = -2.438-01 BETA = -8.934-02 Z = 6.000,00 X = -8.138-01 Y = 1.221,00 ENERGY = 8.443,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 9.293-01 ALPHA = -1.370-01 BETA = -3.430-01 Z = 6.000,00 X = -2.784-01 Y = -1.887-01 ENERGY = 9.672,00

REGION 13 GAMMAS 3.156-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 1 5
 GAMMA = 8.254-01 ALPHA = -3.101-01 BETA = 4.718-01 Z = 6.000,00 X = -4.702,00 Y = 1.235,00 ENERGY = 3.401-01

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 3
 GAMMA = 9.993-01 ALPHA = 3.116-02 BETA = 1.971-02 Z = 6.000,00 X = 2.906-01 Y = 2.519,00 ENERGY = 1.429,00

REGION 13 GAMMAS 5.412-01 COLLISIONS 5 COLLISIONS OCCURED IN REGIONS 7 3 5 5 5 5
 GAMMA = 9.651-01 ALPHA = 2.615-01 BETA = -1.679-02 Z = 6.000,00 X = 7.448-01 Y = 1.083,00 ENERGY = 7.589-01

REGION 13 GAMMAS 9.948-01 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 1 6
 GAMMA = 5.337-01 ALPHA = -8.288-01 BETA = -1.680-01 Z = 6.000,00 X = 2.071-01 Y = 1.444,00 ENERGY = 7.399-01

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 8.940-01 ALPHA = 1.905-01 BETA = 4.055-01 Z = 6.000,00 X = 2.549,00 Y = 2.055,00 ENERGY = 5.076,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 8.817-01 ALPHA = 4.171-01 BETA = -2.203-01 Z = 6.000,00 X = 3.418,00 Y = -1.974,00 ENERGY = 6.297,00

REGION 13 GAMMAS 6.702-01 COLLISIONS 5 COLLISIONS OCCURED IN REGIONS 3 5 5 6 6 6
 GAMMA = 5.225-01 ALPHA = -7.887-01 BETA = 1.966-01 Z = 6.000,00 X = 2.404,00 Y = 1.570,00 ENERGY = 1.960-01

REGION 13 GAMMAS 1.000-00 COLLISIONS 1 COLLISIONS OCCURED IN REGIONS 2 6
 GAMMA = 8.646-01 ALPHA = -3.916-01 BETA = 3.150-01 Z = 6.000,00 X = -2.430,00 Y = 1.436,00 ENERGY = 1.540,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 7
 GAMMA = 7.293-01 ALPHA = 1.584-01 BETA = -6.656-01 Z = 6.000,00 X = -5.264-01 Y = -7.031-01 ENERGY = 8.245,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 6
 GAMMA = 6.785-01 ALPHA = 2.797-01 BETA = -6.793-01 Z = 6.000,00 X = -1.853,00 Y = 5.792-02 ENERGY = 6.206,00

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 1
 GAMMA = 8.165-01 ALPHA = 5.419-01 BETA = -1.992-01 Z = 6.000,00 X = 4.476,00 Y = -1.289,00 ENERGY = 4.245,00

REGION 13 GAMMAS 9.983-01 COLLISIONS 3 COLLISIONS OCCURED IN REGIONS 3 6 6 6
 GAMMA = 6.914-01 ALPHA = -4.415-01 BETA = 5.719-01 Z = 6.000,00 X = -3.918,00 Y = 1.798,00 ENERGY = 2.261-01

 REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 3
 GAMMA = 2.231-01 ALPHA = 2.328-01 BETA = 3.061-01 Z = 6.000,00 X = -5.659-01 Y = 2.043,00 ENERGY = 2.190,00

REGION 13 GAMMAS 8.737-01 COLLISIONS 2 COLLISIONS OCCURED IN REGIONS 1 5 6
 GAMMA = 2.662-01 ALPHA = 3.395-01 BETA = -9.021-01 Z = 6.000,00 X = -7.086-01 Y = -2.427,00 ENERGY = 3.861-01

REGION 13 GAMMAS 1.000-00 COLLISIONS -0 COLLISIONS OCCURED IN REGIONS 5
 GAMMA = 2.700-01 ALPHA = 2.302-01 BETA = 7.788-02 Z = 6.000,00 X = 3.545,00 Y = 2.809-01 ENERGY = 3.495,00

----- STOP TIME BEECRE MACHINE RESICRED TO STANDARD SET-UP ----- 05.97 -----
* THIS PROGRAM PROCESSED ALL AVAILABLE CARDS *
----- ***** STOP TIME 05.98 ----- *****

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1. Mozer, Forrest, Lockheed Missile Systems Division, Palo Alto, California, and Leshan, Edward J., American Standard Atomic Energy Division, Mountain View, California, "A General Monte Carlo Code for Shielding Calculations", LNP-NR-47, Vol. II, Proceedings 5th Semi-Annual ANP Shielding Information Meeting, May 14-15, 1958.
2. Leshan, Edward J., "A General Purpose Monte Carlo Program for the IBM-704, Part I, Method", American Radiation and Standard Sanitary Corporation, Atomic Energy Division, Redwood City, California, September, 1956.

APPENDIX I

Random Numbers

The pseudo random numbers used by FMC are obtained by multiplying the previous random number by a constant and normalizing the least significant half of the product to a floating point number between 0 and 1. The first random number is 22444379091, and the multiplying constant is 30786013581. If these numbers are not satisfactory, they may be changed as shown in the input instructions. If numbers are entered in input, they must be odd.

All possible odd numbers in the range of mod 2^{35} are generated by the method.

APPENDIX II

Rejection Technique for Selection of Polar Scattering Angle

The probability density distribution for the cosine of the polar scattering angle in the center-of-mass system is approximated by

$$f(\mu) = 1 + a_1\mu + a_2\mu^2 \quad (1)$$

The rejection technique used to select a value of μ from this distribution consists of three steps:

- (1) Select a random number, ξ_1 , and form

$$\mu_1 = 2 \xi_1 - 1. \quad (2)$$

This procedure results in a selection of μ from a uniform distribution on the interval

$$-1 \leq \mu \leq 1.$$

- (2) Determine the maximum value, M , that $f(\mu)$ can have in the interval

$$-1 \leq \mu \leq 1.$$

- (3) Form $f(\mu_1)/M$, select a second random number, ξ_2 , and compare

$$\xi_2: \frac{f(\mu_1)}{M}$$

If $\xi_2 \leq f(\mu_1)/M$, the value μ_1 is retained. If $\xi_2 > f(\mu_1)/M$, the value μ_1 is rejected; and steps 1, 2, and 3 are repeated.

This process results in μ being selected from $f(\mu)$.

The maximum value, M , that $f(\mu)$ can have in the interval, $-1 \leq \mu \leq 1$, is determined in the following way. The first derivative of $f(\mu)$ is

$$f'(\mu) = a_1 + 2a_2\mu. \quad (3)$$

The second derivative is

$$f''(\mu) = 2a_2. \quad (4)$$

If $a_2 < 0$, $f(\mu)$ has a maximum in the closed interval, $-1 \leq \mu \leq 1$; and the maximum occurs at

$$\mu_M = -a_1/2a_2. \quad (5)$$

If $a_2 < 0$ and $|a_1| < |2a_2|$, $f(\mu)$ has a maximum in the open interval $-1 < \mu < 1$. In this event,

$$M = f(\mu_M) = 1 - \frac{a_1^2}{4a_2} . \quad (6)$$

If $a_2 < 0$, but the condition $|a_1| < |2a_2|$ does not hold, the maximum of $f(\mu)$ occurs at $\mu_M = 1$ or $\mu_M = -1$. If the maximum occurs at $\mu_M = -1$, then $a_1 < 0$ or the maximum could not occur at $\mu_M = -1$. Similarly, if the maximum occurs at $\mu = 1$, $a_1 > 0$ so that, in either event,

$$M = 1 + |a_1| + a_2 . \quad (7)$$

If $a_2 > 0$, $f(\mu)$ has a minimum in the interval $-1 \leq \mu \leq 1$. There can be only one minimum; hence, the curve must increase on both sides of the minimum. Therefore, $f(\mu)$ must have its largest value at either $\mu_M = 1$ or $\mu_M = -1$. By the above reasoning, M is again given by

$$M = 1 + |a_1| + a_2 . \quad (8)$$

If $a_2 = 0$, $f(\mu)$ has a point of inflection in the interval $-1 \leq \mu \leq 1$. In this event, the maximum value of $f(\mu)$ must again occur at either $\mu_M = 1$ or $\mu_M = -1$, and, as before,

$$M = 1 + |a_1| + a_2 . \quad (7)$$

APPENDIX IIIMechanics of Scattering

An expression for the energy of a neutron after elastic or inelastic scattering is derived in this appendix.

The following definitions are needed:

$v_1 \equiv$ initial velocity of the neutron in the laboratory system,

$v_2 \equiv$ velocity of the neutron in the laboratory system after scattering,

$v_a \equiv$ velocity of the neutron in the center-of-mass system after scattering,

$v_b \equiv$ velocity of the scatterer in the center-of-mass system after scattering,

$A \equiv$ mass of the scatterer,

$E_x \equiv$ change in interval energy of the scatterer (excitation energy),

$\mu^{\text{cm}} \equiv$ cosine of the polar scattering angle in the center-of-mass system,

$v_m \equiv$ velocity of the center of mass of the neutron-scatterer system.

The kinetic energy in the center of mass system before collision is

$$\frac{1}{2} \left(\frac{A v_1}{A+1} \right)^2 + \frac{1}{2} A \left(\frac{v_1}{A+1} \right)^2. \quad (1)$$

The kinetic energy in the center-of-mass system after collision is

$$\frac{1}{2} (v_a)^2 + \frac{1}{2} A (v_b)^2. \quad (2)$$

The equation expressing conservation of energy in the center of mass system is

$$\frac{1}{2} \left(\frac{A v_1}{A+1} \right)^2 + \frac{1}{2} A \left(\frac{v_1}{A+1} \right)^2 = \frac{1}{2} v_a^2 + \frac{1}{2} A v_b^2 + E_x. \quad (3)$$

The equation expressing conservation of momentum in the center of mass system is

$$v_a = A v_b. \quad (4)$$

Equations 3 and 4 can be solved simultaneously to give

$$\frac{v_a^2}{2} = \frac{A^2}{(A+1)^2} \frac{v_1^2}{2} - \frac{A}{(1+A)} E_x \quad (5)$$

It can be shown that

$$v_2^2 = v_m^2 + v_a^2 + 2 v_m v_a \mu^{cm} \quad (6)$$

Substituting equation 5 and $v_m = v_1/(A+1)$ in equation 6 gives

$$\frac{v_2^2}{2} = \frac{v_1^2}{2(A+1)^2} + \frac{A^2 v_1^2}{2(A+1)^2} - \frac{A}{(1+A)} E_x + \frac{v_1}{(A+1)} \frac{A^2 v_1^2}{(A+1)^2} - 2 \frac{A}{(A+1)} E_x \mu^{cm} \quad (7)$$

The energy of the neutron in the laboratory system after scattering is

$$E' = \frac{1}{2} v_2^2. \quad (8)$$

The energy of the neutron in the laboratory system before scattering is

$$E = \frac{1}{2} v_1^2. \quad (9)$$

Substituting equations 8 and 9 into equation 7 give

$$E' = \frac{E}{(1+A)^2} \left[1 + A^2 \left(1 - \frac{(1+A)}{A} \frac{E_x}{E} \right) + 2A \sqrt{1 - \frac{(A+1)}{A} \frac{E_x}{E}} \mu^{cm} \right]. \quad (10)$$

Define

$$B = \left[1 - \frac{(A+1)}{A} \frac{E_x}{E} \right]^{\frac{1}{2}} \quad (11)$$

Substituting equation 11 into equation 10 gives

$$E' = \frac{E(1 + A^2 B^2 + 2 AB \mu^{cm})}{(1 + A)^2}. \quad (12)$$

If E_x is zero, B is equal to one; and equation 12 reduces to

$$E' = \frac{E(1 + A^2 + 2A \mu^{cm})}{(1 + A)^2}. \quad (13)$$

APPENDIX IV

Transformation of Direction Cosines

The method of transforming direction cosines from the center-of-mass system to the basic reference system fixed in the source-shield assembly is described in this appendix. The equations relating the direction cosines of the particle in the fixed reference system before scattering to the direction cosines of the particle in the fixed system after scattering are derived.

The accompanying sketches illustrate the three coordinate systems of interest. Coordinate system (a) represents the fixed reference system. The direction of the particle prior to scattering is shown in this system by a solid line. The direction cosines in this system for the direction of the incident particle are denoted by α , β , γ .

Coordinate system (b) represents the center-of-mass coordinate system. The Z-axis in this system is parallel to the direction of the incident particle and directed in the same sense. The direction of the particle after scattering is shown in this system by a dash-dot line. All quantities in this system are denoted by a double prime.

Coordinate system (c) represents the laboratory coordinate system. The Z-axis in this system is parallel to the Z-axis of the center-of-mass coordinate system and directed in the same sense; hence, it is parallel to the direction of the incident particle and directed in the same way. The direction of the particle after scattering is shown in this system by a dash-dot line. All quantities in this system are denoted by a triple prime.

Coordinate system (d) represents the fixed reference system. The direction of the particle after scattering is shown in this system by a solid line. The direction cosines in this system for the direction of the scattered particle are denoted by α' , β' , γ' .

The cosine of the polar scattering angle, $\mu^{cm} = \cos \theta^{cm}$, is chosen in the center-of-mass system as explained previously. The quantity μ^{cm} is then transformed to the laboratory coordinate system:

$$\mu^{cm} \xrightarrow{T} \mu^L, \quad \mu^L = \cos \theta^L$$

The azimuthal scattering angle, ϕ , is chosen in the laboratory system. The distribution of the azimuthal scattering angle is uniform in the laboratory system, since it is uniform in the center-of-mass system; and the transformation from the center-of-mass system to the laboratory system is along the direction of the Z-axis of both systems.

It is convenient to define spherical coordinates for the directions of the incident and scattered particle:

$\Lambda \equiv$ polar angle coordinate of incident particle in the fixed system,

$\Gamma \equiv$ azimuthal angle coordinate of incident particle in the fixed system,

(a) - Fixed Reference System

(b) - Center-of-Mass Coordinate System

(c) - Laboratory Coordinate System

(d) - Fixed Reference System

$\psi \equiv$ polar angle coordinate of scattered particle in the fixed system,

$\eta \equiv$ azimuthal angle coordinate of scattered particle in the fixed system.

It follows from these definitions and the definition of direction cosines that:

$$\begin{aligned}\alpha &= \sin \Lambda \cos \Gamma \\ \beta &= \sin \Lambda \sin \Gamma \\ \gamma &= \cos \Lambda\end{aligned}\quad (1)$$

$$\begin{aligned}\alpha' &= \sin \psi \cos \eta \\ \beta' &= \sin \psi \sin \eta \\ \gamma' &= \cos \psi\end{aligned}\quad (2)$$

The equations for transforming the spherical coordinate in the laboratory system, (θ^L, φ) , to the spherical coordinate in the fixed system, (ψ, η) , for the scattered particle are

$$\cos \psi = \cos \Lambda \cos \theta^L - \sin \Lambda \sin \theta^L \cos \varphi \quad (3)$$

$$\sin \psi \cos \eta = -\sin \Gamma \sin \theta^L \sin \varphi + \cos \Lambda \cos \Gamma \sin \theta^L \cos \varphi + \sin \Lambda \cos \Gamma \cos \theta^L$$

$$\sin \psi \sin \eta = \cos \Gamma \sin \theta^L \sin \varphi + \cos \Lambda \sin \Gamma \sin \theta^L \cos \varphi + \sin \Lambda \sin \Gamma \cos \theta^L$$

Substitution of equations 1 and 2 into equation 3 gives

$$\begin{aligned}\alpha' &= \frac{-\beta \sin \varphi \sqrt{1 - (\mu^L)^2}}{\sqrt{1 - \gamma^2}} + \frac{\gamma \alpha \sqrt{1 - (\mu^L)^2} \cos \varphi}{\sqrt{1 - \gamma^2}} + \alpha \mu^L \\ \beta' &= \frac{\alpha \sin \varphi \sqrt{1 - (\mu^L)^2}}{\sqrt{1 - \gamma^2}} + \frac{\gamma \beta \sqrt{1 - (\mu^L)^2} \cos \varphi}{\sqrt{1 - \gamma^2}} + \beta \mu^L \\ \gamma' &= \gamma \mu^L - \sqrt{1 - \gamma^2} \sqrt{1 - (\mu^L)^2} \cos \varphi\end{aligned}\quad (4)$$

In the event that the direction of the incident particle is coincident with or nearly coincident with the direction of the Z-axis of the fixed reference system, then γ is approximately equal to one and

$$(1 - \gamma^2) \leq \epsilon; \quad \epsilon \ll 1. \quad (5)$$

When this condition holds, the direction cosines of the scattered particle in the fixed reference system, α' , β' , γ' , are approximately equal to the direction cosines of the scattered particle in the laboratory reference system:

$$\begin{aligned}\alpha' &= \sin \theta^L \cos \varphi = (1 - (\mu^L)^2)^{\frac{1}{2}} \cos \varphi \\ \beta' &= \sin \theta^L \sin \varphi = (1 - (\mu^L)^2)^{\frac{1}{2}} \sin \varphi \\ \gamma' &= \cos \theta^L = \mu^L\end{aligned}\quad (6)$$

When φ is replaced by $\pi(2\xi-1)$ in equations 4 and 6, equations 6.51, 6.52, 6.53, 6.54, 6.55, and 6.56 result.

APPENDIX V

Acceleration Factors

A general expression for the calculation of acceleration factors is derived in this appendix. Use of this expression is illustrated by application to special cases.

A set of source parameters is defined as a unique set of values for a source particle's location, energy, and direction. Such a set of parameters may be chosen by the source particle generator portion of the program or specified by card input as described previously.

It is convenient to make the following additional definitions:

- N = number of independent sets of source parameters,
- j = all source parameters in a set of source parameters except the source energy parameter,
- K = the source energy parameter,
- $N_{mc}^{j,K}$ = the number of source particles with the same source parameters. This quantity is identical to the number of particle's initially present in a cluster, N_{mc} , previously defined, but with dependence on the source parameters explicitly written.
- $N_{mc}^{j,K'}$ = the number of source particles with the same source parameters after adjustment of $N_{mc}^{j,K}$ by the acceleration factor, A_i , of the energy group i in which the energy parameter lies:

$$N_{mc}^{j,K'} = A_i N_{mc}^{j,K} \quad (1)$$

In order that $N_{mc}^{j,K'}$ be integer, it is actually determined by a truncation process which utilizes a uniformly distributed random number. It is more precise for the determination of acceleration factors, however, to relate $N_{mc}^{j,K}$ and $N_{mc}^{j,K'}$ by means of equation 1.

- N_i = number of source particles in energy group i before use of acceleration factor A_i ,
- N_i' = number of source particles in energy group i after use of acceleration factor A_i ,
- M = total number of source particles before use of acceleration factors,
- M' = total number of source particles after use of acceleration factors,
- $f(E)$ = probability density distribution from which source particle energy parameters are initially chosen.

$S^*(E)$ = probability density distribution from which source particle energy parameters are chosen through use of the acceleration factors.

$S(E)$ = correct source energy spectrum probability density distribution.

r_1 = probability that the energy of a source particle lies in energy group 1 as determined by $f(E)$.

p_1^* = probability that the energy of a source particle lies in energy group 1 as determined by $S^*(E)$.

p_1 = probability that the energy of a source particle lies in energy group 1 as determined by $S(E)$.

It follows from the definition of r_1 , p_1^* , p_1 , and $f(E)$, $S^*(E)$, $S(E)$, that

$$r_1 = \int_{E_{1l}}^{E_{1u}} f(E') dE', \quad p_1^* = \int_{E_{1l}}^{E_{1u}} S^*(E') dE', \quad p_1 = \int_{E_{1l}}^{E_{1u}} S(E') dE' \quad (2)$$

The following relationships are an immediate consequence of the definitions:

$$N_1 = \sum_j \sum_K N_{mc}^{j,K} Q(K, \Delta E_1) = M r_1 \quad (3)$$

$$N_1' = \sum_j \sum_K N_{mc}^{j,K'} Q(K', \Delta E_1) = M' p_1^* \quad (4)$$

$$M = \sum_1 N_1 = \sum_j \sum_K N_{mc}^{j,K'} \quad (5)$$

$$M' = \sum_1 N_1' = \sum_j \sum_K N_{mc}^{j,K'} \quad (6)$$

$$N_1' = A_1 N_1 \quad (7)$$

where $Q(K, \Delta E_1) = 1$ if the interval $\Delta E_1 = E_{1u} - E_{1l}$ contains K ,
 $Q(K', \Delta E_1) = 0$ otherwise

the subsequent development assumes that output from the FMC programs are to be normalized to M' .

The following condition must hold if bias in the energy sampling scheme is to be avoided.

$$N_1' W_1 = M' p_1 \quad (8)$$

where W_1 is the weight assigned to each particle immediately after use of the acceleration factors. The expression for W_1 is obtained from equations (8) and (4)

$$W_i = \frac{p_i}{p_i^*} \quad (9)$$

this is the general expression for the weight.

The general expression for the acceleration factor, A_i , is obtained from equations (3), (7), (8), and (9)

$$A_i = \frac{M'}{M} \frac{p_i^*}{r_i} \quad (10)$$

When the input values are such that $A_i > 0$ for all i no importance sampling is done on the source energy spectrum, so that

$$S^*(E) = S(E)$$

and therefore,

$$p_i^* = p_i$$

The expressions for W_i and A_i reduce to

$$W_i = 1, \quad (11)$$

$$A_i = \frac{M' p_i}{M r_i}, \quad (12)$$

respectively.

Now consider several special cases when $A_i > 0$.

Case 1

The source generator is used to select all parameters for source particles by random sampling from $S(E)$. Then, if sufficient sampling is done,

$$r_i = p_i \quad (13)$$

$$\text{and } A_i = \frac{M'}{M} \quad (14)$$

This result is reasonable because the source particles were chosen by the source generator from the correct distribution $S(E)$; hence, since the A_i 's are not used to do importance sampling when $A_i > 0$, the A_i 's must all be equal in order that the shape of the source particle energy distribution be unchanged after application of the acceleration factors.

Case 2

All source parameters except the energy parameter are selected by the source generator. Each set of source parameters is, by means of card input, assigned once to each of the desired energy group by giving the energy

parameter K a value equal to the average energy of each group. The quantity $N_{mc}^{j,K}$ is taken to be one for all sets of source parameters. Then

$$r_i = \frac{N}{M} \quad (15)$$

Equations (15) and (12) give

$$A_i = \frac{M' p_i}{N} \quad (16)$$

It is evident from equation (16) that for Case 2 the acceleration factors are proportional to the probability that the energy of a source particle lies in the corresponding energy group. This result is reasonable because the source particle energy distribution was specified to be a uniform distribution; hence, the source spectrum shape must be given entirely by the values of the acceleration coefficients. This can only be true if equation (16) holds.

When the input values of A_i are such that $A_i < 0$ the acceleration factors are used for purposes of importance sampling. In the present version of the program it is assumed that for this case

$$M = M' \quad (17)$$

$$f(E) = S(E) \quad (18)$$

It follows from equation (18) that, for this case,

$$r_i = p_i$$

Then the expressions for W_i and A_i are

$$W_i = \frac{p_i}{p_i^*}$$

$$A_i = \frac{p_i^*}{p_i}$$

so that

$$W_i = \frac{1}{A_i}$$

The present version of the program incorrectly computes W_i for this case using

$$W_i = \frac{1}{N_{mc}^{j,K}}$$

APPENDIX VI

Average Track Length for Flux Determination

The calculation for determining the average track length in a region for a particle or particle cluster traveling in the region with a given direction, energy, and starting point is described in this appendix. The expressions for Monte Carlo and expectation flux tallies are derived for the general case and for the special case of a region void of material.

The initial point of a track length in a region is on a boundary of the region or at a scattering point within the region. Let (X_0, Y_0, Z_0) represent such an initial point within or on the boundary of a region R . Let $\vec{r}$ represent a line originating at (X_0, Y_0, Z_0) and directed along the path of the particle. Let S represent the distance from (X_0, Y_0, Z_0) to the first boundary crossing measured along $\vec{r}$. The contribution to the average track length, t_1^R , in the region R due to collision between (X_0, Y_0, Z_0) and S is

$$t_1^R = \int_0^S r \Sigma_t^h(E) e^{-\Sigma_t^h(E)r} dr \quad (1)$$

where h = composition of region R ,

$\Sigma_t^h(E)$ = total macroscopic cross section for particle of energy E in composition h .

Equation 1 can be integrated to give

$$t_1^R = -Se^{-\Sigma_t^h(E)S} - \frac{e^{-\Sigma_t^h(E)S}}{\Sigma_t^h(E)} + \frac{1}{\Sigma_t^h(E)} \quad (2)$$

The probability, $p(r > S)$, that the particle will have a collision outside region R is

$$p(r > S) = 1 - \int_0^S \Sigma_t^h(E) e^{-\Sigma_t^h(E)r} dr \quad (3)$$

Equation 3 can be integrated to give

$$p(r > S) = e^{-\Sigma_t^h(E)S} \quad (4)$$

For all collisions outside region R , the value of the track length in region R is S . Hence, the contribution to the average track length in region R , t_2^R , due to collisions outside region R is

$$t_2^R = Se^{-\Sigma_t^h(E)S} \quad (5)$$

The average track length in region R is the sum of these contributions:

$$t^R = t_1^R + t_2^R = \frac{1}{\Sigma_t^h(E)} (1 - e^{-\Sigma_t^h(E)S}) \quad (6)$$

This can also be written

$$t^R = \lambda^h(E) (1 - e^{-S/\lambda^h(E)}) \quad (7)$$

The average track length for the number of neutrons or gamma rays represented by a particle or cluster, T^R , which enters a region is

$$T^R = W N_{mc} \lambda^h(E) (1 - e^{-S/\lambda^h(E)}) \quad (8)$$

where W = weight of particles in the cluster,
 N_{mc} = number of particles in the cluster.

The Monte Carlo flux for energy group i in a region R, ϕ_i^R , is proportional to the sum of the average track length of all neutrons or gamma rays traveling in the region whose energy lies in energy group i :

$$\phi_i^R \propto \sum T^R \quad (9)$$

where the sum is over all track lengths of neutrons or gamma rays in region R with energy lying in energy group i .

The quantity on the right hand side of this proportionality is computed by the programs. The Monte Carlo flux for energy group i in region R, normalized to unit source strength, can be obtained from the machine result by using

$$\phi_i^R = \frac{1}{MV} \sum T^R \quad (10)$$

where M = total number of source particles,
 V = volume of the region.

When $\lambda^h(E)$ becomes infinite, equation 8 gives an indeterminate result for T^R . The correct equation for this special case can be obtained by using the polynomial series expansion for $e^{-S/\lambda^h(E)}$:

$$e^{-S/\lambda^h(E)} = 1 - \frac{S}{\lambda^h(E)} + \frac{S^2}{(\lambda^h(E))^2 2!} - \frac{S^3}{(\lambda^h(E))^3 3!} + \dots \quad (11)$$

and substituting this in equation 8. The result is

$$T^R = W N_{mc} S. \quad (12)$$

Equation 12 is used by the programs to determine ϕ_i^R for void regions.

The expectation flux tally for a region and energy group is computed in the following manner:

The product of the weight of a particle cluster in the energy group and the number of particles in the cluster is formed immediately after a collision or after leaving the source. This quantity is denoted by the symbol π^i . The weight of a particle cluster in the energy group and the number of particles in the cluster immediately after a collision or after leaving the source are denoted in this appendix by the symbols W^i and N_{mc}^i , respectively. Then

$$\pi^i = W^i N_{mc}^i \quad (13)$$

The quantity π^i is multiplied by the probability, P , that the particle proceeds along the chosen direction without collision from the collision or source location to the region boundary. This quantity is denoted by π . Then

$$\pi = W^i N_{mc}^i P \quad (14)$$

The average expectation track length in a region R , e_{TR}^R , for the number of neutrons or gamma rays represented by a particle or cluster, is obtained by multiplying π by the average track length of the particle or cluster in the region had it entered the region

$$e_{TR}^R = \pi \lambda^h(E) (1 - e^{-S/\lambda^h(E)}) \quad (15)$$

The expectation flux for energy group i in a region R , $e_{\phi_1^R}^R$, is proportional to the sum of the average expectation track lengths of all neutrons or gamma rays represented by particles or clusters that have a non-zero probability of entering the region directly after leaving the source or after collision and whose energy lies in energy group i

$$e_{\phi_1^R}^R \propto \sum e_{TR}^R \quad (16)$$

where the sum is over all track lengths of neutrons or gamma rays with non-zero probability of reaching region R with energy lying in energy group i .

The quantity on the right hand side of this proportionality is computed by the programs. The expectation flux for energy group i in region R , normalized to unit source strength, can be obtained from the machine result by using

$$e_{\phi_1^R}^R = \frac{1}{MV} \sum e_{TR}^R \quad (17)$$

where M = total number of source particles,
 V = volume of the region.

The expression for the expectation flux for a void region is obtained in the same way as in the case of the Monte Carlo flux. This expression is

$$e_{TR}^R = \pi S. \quad (18)$$

APPENDIX VII
FLOW CHARTS

AIRCRAFT NUCLEAR PROPULSION DEPARTMENT

GENERAL ELECTRIC