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ABSTRACT

PMC is a three-dimensional Monte Carlo program written in the MAP language for the IBM 7040 computer. Output includes the scalar neutron flux integrals and reaction rates for resonant isotopes for an arbitrary energy group structure containing a minimum of approximations. The code is especially applicable to complex geometrical situations and areas where diffusion theory will not be reliable. A statistical confidence is assigned to the results. Two versions of the program are described: a "nonresonant" version in which all cross sections are supplied in the input, and a "resonant" version in which resonant cross sections are computed directly.

Simplified flow diagrams and sample problems showing the use of each version are given.

CONTENTS

FIGURES

I. INTRODUCTION

1. GENERAL PHILOSOPHY AND PURPOSE OF THE PROGRAM

There are many reactor physics problems in which diffusion theory cannot be relied upon to give satisfactory answers. For example, in regions near sources or strongly absorbing boundaries, where the neutron flux becomes highly anisotropic, high order transport calculations are necessary to give reliable estimates of the scalar flux function even though the angular distribution is of no particular interest. In general, P_n and S_n transport codes are usually limited in application to simplified geometries in one dimension. In problems where local flux distributions are highly dependent on strong absorbers in peculiar geometric arrangements, the Monte Carlo method provides a reliable calculation technique with a minimum of approximations but at a relatively high computation cost compared to other numerical methods. Usually the flexibility of a Monte Carlo calculation is greatly restricted in order to make direct comparisons with other numerical techniques because the Monte Carlo method per se is most easily adapted to three-dimensional geometries. If truly three-dimensional numerical transport codes existed, the efficiency of a Monte Carlo calculation might not then appear in such poor light. In addition to its adaptability to complicated geometries and nuclear cross sections, another advantage of the Monte Carlo technique is its intrinsic mathematical simplicity.

The purpose of the present program was to make available a three-dimensional Monte Carlo program for the calculation of the scalar neutron flux and its integrals, with associated statistical quantities, within an arbitrary energy group structure. Primary emphasis was to be placed on the ability to do so-called "cell studies" involving complicated geometries with flexible boundary conditions and sources. There is no restriction in the present program to the calculation of flux integrals and eigenvalues for complete chain reacting systems in three dimensions other than the necessary restrictions of limited in-core storage and the computing time required to reduce statistical fluctuations to within tolerable limits. Experience has shown the latter restriction to be far more limiting than the former as applied to the IBM 7040 computer for this type of problem. Water-moderated reactors of 50 to 100 thermal mean-free-paths diameter require many hours of computing to obtain an acceptable four group eigenvalue with this program even when the most elementary geometries are employed. In the other extreme, the simplest monoenergetic first flight calculation of escape probabilities for a few regions of 10 to 20 mean-free-paths width requires only five to ten minutes for accurate answers based on 5000 first flight histories.

Our original objectives for a general purpose Monte Carlo program were based on our limited experience with a number of Monte Carlo programs then in existence. Notable among them were the Reactor Burnup Code (RBU)[1] and a program written by Marius Troost[2]. A limited number of difficult transport problems were studied with RBU in 1962 at Hanford Laboratories and served to assure us that our general objectives for a three-dimensional Monte Carlo transport program were reasonable and, indeed, possible with existing computers. This experience also served to point out some of the difficulties accompanying Monte Carlo studies in general. Chief among them is the problem of statistical uncertainty inherent in the Monte Carlo method. All

bulk reactor properties computed in the Monte Carlo process result from choices of random variables and, consequently, are subject to statistical fluctuation. Any problem solution derived from these bulk parameters, whether it be neutron flux, multiplication factor, collision frequency, or lifetime, is not reliable unless a statistical confidence interval can be assigned. In this program, the scalar neutron flux was chosen to serve as the statistical basis for determining this confidence interval. Each flux tally stored by region and neutron energy group specified in the input is assigned a statistical confidence limit by the output routine based on an analysis of the trial fluxes using a Student's t-distribution. The method of terminating the particular mode of operation and the means of determining the length of trials and number required are explained in Section VII.

Since the scalar neutron flux was to be the variable by which all bulk parameters would be determined, the program was built around this computation. It was felt that the continuous collision analog flux tally employed by RBU contained inherently less variance than other models for this quantity. This model also contains other advantages which are not so apparent. Since boundary crossings and region search routines require the largest amount of computation time, this model for flux computation permits the program to obtain the most information possible within a region for each boundary crossing. In some problems this last consideration can outweigh all others.

Because the RBU cross section library format had been worked out in great detail with special consideration for a Monte Carlo program and was available on tape, the collision routine was patterned to fit this library format. For this reason much of the formalism of the collision routine is similar to that used in RBU.

Two versions of the program are presented in this report. The first is the nonresonant version for which all cross sections are supplied to the code in the input as average multigroup cross sections for homogeneous materials. The second version, called the resonant version of the program, permits the computation of resonance cross sections directly from the single-level Breit-Wigner equations in materials containing resonance isotopes. Resonance cross sections are computed at the energy of the particle, and resonance reaction rates are tallied by statistical trial along with the scalar flux. Single-level Breit-Wigner resonance parameters are supplied in the input data for each resonant isotope of a resonant material. The resonance calculation is described in Section V.

The geometry routines were modeled after the zone geometry method outlined by Marius Troost[2]. An additional feature in this program is that either cylindrical or spherical zone geometry is available, as well as the rectangular zone construction. The geometric constructions permitted for laying out regions employ what was felt to be reasonable combinations of planes intersecting with spheres and cylinders. Although the basic techniques used here would allow one to treat all conic sections, little practical use for this type of geometry could be determined.

The present programs calculate elastic scattering of fast neutrons based on isotropic scattering in the center of mass system. Subroutines to treat fast anisotropic scattering in the center of mass system and fast inelastic scattering,

using cross section data from the RBU library, will be incorporated into the program when sufficient experience and usage of the program justifies the expense of doing so.

At various points in this report, greatly simplified flow diagrams will be included to help tie the discussion together. All extraneous computations, such as indexing, have been eliminated from these diagrams.

Finally, in the last analysis, the code in its present form can be considered to be an experimental program by which the Monte Carlo process can be studied under a wide variety of conditions. With the present limitations, nuclear reactions are treated exactly over only part of the energy range of interest in reactor design. As a consequence of this, PMC is not a production code but can be used to supplement production codes in the thermal and resonance regions where difficult transport problems exist. It is the authors' opinion that the present lack of knowledge about efficient statistical estimators for bulk reactor parameters in the Monte Carlo process will be the primary limiting factor in the application of a general purpose Monte Carlo program to practical reactor problems. It is our purpose that this version of PMC will contribute to the general analysis of this basic problem.

2. OUTLINE OF MAJOR SUBROUTINES

Input Routine. Input data describing the problem are edited and prepared for use by the Monte Carlo program proper, and then stored on magnetic tape.

Initial Value Source Generator Routine. Source coordinates are generated as prescribed by the input options and stored on tape.

Storage Bank Routine. This is a routine wherein coordinates are stored or removed from the storage bank for systematic processing.

Distance to Boundary Routine. The distance to each boundary of a region is computed, and the actual boundary to be crossed is selected.

Flux Calculation and Tally Routine. The continuous collision analog flux is computed and stored in the proper location according to the current region and energy group numbers. There is provision for equilibration weighting and census termination of the particles.

Resonance Routine. Macroscopic cross sections are prepared for resonant isotopes in resonant materials within a specified energy range for use in the reaction rate tallies. Collision emission numbers are computed for the nuclear events of absorption, scattering, and fission.

Collision Routine. When a collision occurs, the possible collision events are processed according to fast or thermal conditions. Collision tallies and energy importance splitting take place.

1. Post Fission Routine: If fission has been successful, the direction and energy of the emitted fission neutrons are selected.

2. Post Elastic Routine: The type of moderation and appropriate transformations from the center of mass system to laboratory coordinates are made for the direction cosines.

3. Post Thermal Collision Routine: The thermal direction cosines are selected from the linearly anisotropic thermal distribution by acceptance-rejection technique.

Boundary Crossing Routine. Region importance splitting and attenuation of beams take place.

Zone and Region Search Routine. When a particle crosses a boundary, all the regions within a particular zone are searched until its new location is found.

Outer Boundary Reflection Routine. Beams and/or particles are wholly or partially reflected and transmitted from outer boundaries, depending on the boundary albedoes assigned in the input.

Lost Particles Routine. Logic is supplied for a number of situations whereby a particle can become lost. Provision to correct the ambiguity is included.

Output Routine and Statistical Analysis. Reaction rate and flux tallies for each of the statistical trials are processed for printing. At the conclusion, a statistical analysis of fluxes and reaction rates is made, and a confidence interval is calculated on preassigned tally boxes.

II. COORDINATES, BEAMS, AND FLUX TALLY

1. PARTICLES AND COORDINATES

Each group of particles which the program processes has a coordinate whose analog is a classical beam of neutrons. This coordinate is continuously updated at each event during the Monte Carlo process. A coordinate consists of 12 variables: N, x, y, z, α, β, γ, t, $\sqrt{E}$, g, h, m

where

N = the number of particles in the beam

x,y, z = space coordinates in the reactor reference frame

α, β, γ = direction cosines for the direction in which the beam is traveling

t = time in microseconds

$\sqrt{E}$ = square root of energy in electron volts

g = microscopic energy group in which the beam is presently located

h = region in which the beam is located

m = material in which the beam is located.

Thus, a "beam" is a coordinate of N particles traveling in the direction α, β, γ. The coordinates are random variables. As particles are followed through the reactor system, these coordinates are advanced depending on the nuclear cross sections supplied in the input or computed by the resonance routine and the outcome of decisions made by random numbers.

A block of 45 initial value coordinates created by the source generator will be read into the coordinate storage bank. The sequence of events which takes place in the Monte Carlo process begins when one of these coordinates is removed from the storage bank.

2. PARTICLE AND BEAM PROCESSING

New coordinates may be created at any time as the result of a successful collision. Depending on the nature of the collision, one or more particles will result having the x, y, z coordinates at the point of collision and traveling in the new direction, α', β', γ'. If the resultant energy of the emitted particles lies within a new energy group, the N' new particles are assigned the new group number. The entire new set of coordinates is placed in the storage bank for future processing as new beams. Should a particle be absorbed, no new coordinate set is stored. The direction of the original beam does not change when collisions and other events take place. Only the discrete particles resulting from collisions change direction. The events which take place between collisions determine how particles travel in space, where collisions will take place, and how the flux is computed and tallied.

Each particle in the beam is treated independently of the other particles and advanced across boundaries along with the beam or taken to collision independent of the beam. Each particle in the beam possesses an arbitrary neutron weight which has been assigned by region and group with the input data. Thus, the weights I_g and I_h correspond to the neutron weights assigned to the energy group g and region h, respectively. The neutron weight assigned to each particle in group g, region h will be $1/I_gI_h$. Neutron weights and splitting at boundaries will be explained in Section VII. Each particle will be advanced with the beam until it either (a) goes to collision, (b) leaks from the reactor through an external boundary, or (c) is lost crossing an internal boundary due to neutron splitting. Figure 1 gives a graphic illustration of how beams and particles are followed in the direction α, β, γ until they are terminated. Although this illustration is two dimensional, beams actually travel and cross boundaries in three-dimensional space.

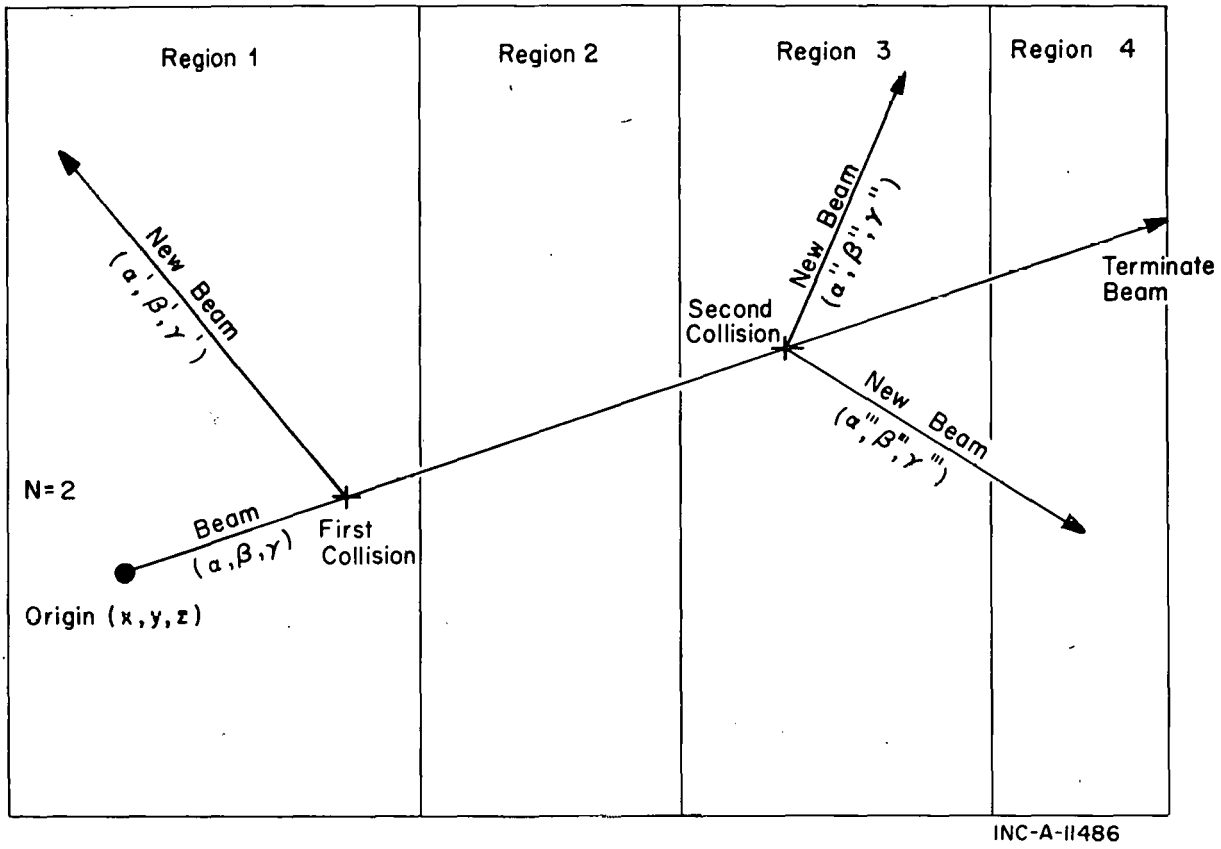


FIG. 1 ILLUSTRATION OF THE PROGRESS OF A SINGLE BEAM FROM BIRTH TO TERMINATION.

Regardless of how many particles are in the beam, the beam is followed until all particles have been processed and the beam has been attenuated to less than 1/10 of its original value.

3. FLUX TALLY AND CONTINUOUS COLLISION ANALOG

A beam originates at r_0 traveling in the direction α, β, γ with N particles, each representing $1/I_gI_h$ neutrons. The total number of neutrons in the beam is therefore:

$$I_0 = N/I_g I_h \ .$$

The flux-volume-time integral can be obtained from the beam directly by noting that the scalar flux integral along a beam out to a radius R is equivalent to an isotropic point source emitter of I_0 neutrons/sec traveling at a constant velocity, v, in a homogeneous medium of mean free path λ. The equivalent isotropic point source emitter flow through an annulus at radius r from the emission point is

$$\frac{I_0 e^{-r/\lambda}}{4\pi r^2} \quad \frac{\text{neutrons}}{\text{cm}^2 - \text{sec}} \ .$$

In the time dt, a volume $dV = r^2 d\Omega dr$ is swept out by the equivalent isotropic emitter. Since $dt = dr/v$, t is not independent of r. Therefore, the volume-time differential is

$$dVdt = r^2 d\Omega dr \ \delta(t-r/v)dt.$$

The scalar flux-volume-time integral is

$$\int_V \int_t \phi(r, t)drdt = \frac{I_0}{4\pi} \int_\Omega \int_0^R \int_{t_1}^{t_2} \frac{r^2 e^{-r/\lambda}}{r^2} \ \delta(t-r/v) \ d\Omega dr dt$$

$$= \frac{I_0}{4\pi} \int_\Omega d\Omega \int_0^R e^{-r/\lambda} dr \int_{t_1}^{t_2} \delta(t-r/v)dt = I_0\lambda(1-e^{-R/\lambda}) \ .$$

This is the scalar flux-volume-time integral arising from a point source emitter of I_0 neutrons per second over a homogeneous sphere of radius R. That the Monte Carlo beams originate at a different point each time within a region and represent each time an anisotropic source rather than an isotropic one makes no difference as long as we only wish to compute a scalar flux-volume integral. If the point source emitter above had been a beam emitted in a fixed direction, rather than isotropically, the scalar flux-volume integral would still be the same for the sphere of radius R. The continuous collision density function for the beam traveling in a direction $\vec{\Omega}$ is exactly analogous to the more familiar discrete collision track length definition of the flux for a particle traveling a distance λ_i between collisions. In the discrete collision case, the scalar flux-volume contribution is given by $I_0\lambda_i$ for the region in which it is traveling. (Note that the region can be any shape, not necessarily a sphere.) Here all I_0 neutrons went to collision at once at the end of the discrete path, λ_i. Whereas, in the analytic continuous collision model, the neutron beam undergoes continuous collisions over a distance S_b to the boundary of the region; and the scalar flux-volume integral contribution for the region is given by this model as:

$$I_0\lambda(1 - e^{-S_b/\lambda}).$$

The weight of the uncollided beam reaching the boundary is

$$I = I_o \, e^{-\frac{S_b}{\lambda}}$$

where $I_o = N/I_g I_h$, as before.

After the boundary has been crossed and the new region being entered has been determined, the beam, with new initial weight I, will now be followed across this region to the next boundary for a distance S_b', giving a flux contribution for this region of

$$\phi' = I \, \lambda' \left(1 - e^{-\frac{S_b'}{\lambda'}}\right).$$

The resultant neutron weight at the b' boundary will be

$$I' = I \, e^{-\frac{S_b'}{\lambda'}}$$

and the remaining portion of the beam will be continued on into the next region.

In the discrete track length flux tally, the average distance to collision in an infinite medium will be λ, where λ is the mean free path of the medium. For the continuous collision model the average distance to collision is given by

$$\langle r \rangle = \frac{\int_o^\infty r \, e^{-\frac{r}{\lambda}} \, dr}{\int_o^\infty e^{-\frac{r}{\lambda}} \, dr} = \lambda .$$

Both methods give the same expected value. However, in the discrete collision model the flux tally depends directly on the discrete distance to collision which is computed by means of a random number, thus producing a statistical variance. By using the analytical method for the flux tally, the variance is reduced since the flux tally is independent of the distance to collision.

The process of having collisions, advancing the particles to boundary, and beam attenuation is continued in each region through which the beam passes until all N particles have been used. The beam itself, however, continues on until the beam's neutron weight is less than one tenth of the initial value, I_0. The resultant weight is then subjected to a Russian roulette process and either terminated or doubled and continued as above.

When a beam is terminated, the latest coordinate generated from the discrete collisions suffered by the particles in the beam is picked up from the storage bank and processed as a new beam traveling in a new direction. If the bank is

empty, due to the system being subcritical or a temporarily unfavorable run of random numbers, 45 more Initial Value (IV) coordinates from the source generator are read into the bank. When one of these is picked up, the problem continues. If all the IV coordinates manufactured by the source generator have been used, the problem is finished. Other methods of ending the problem are described in Section VII.

4. COLLISIONS

Even though the flux tally is made on the basis of the analytic model with a beam of neutrons suffering continuous collisions along the path of the beam, new beams must originate as a result of a discrete collision at one or more points along the beam path since the coordinates of the collision constitute the origin of new beams. The beam consists of N particles, and it is these particles which are advanced to collision as the beam progresses through the reactor. The distance to collision is a random variable and is determined for each homogeneous region the beam enters. If the distance to collision is greater than the distance to the next boundary of the region, the particle is advanced to the boundary and the process is repeated for the next particle in the beam. If the distance to collision is less than the distance to the boundary, the particle is advanced this distance to collision and the program enters the collision routine. When the collision has been processed, including storing the daughters of the collision in the coordinate bank, the distance to collision is determined for the next particle and the process is repeated until all particles in the beam have been exhausted. The beam is then advanced to the boundary, and the particles which were advanced to the boundary from the previous region are now processed to collision or advanced to the next boundary as described above.

Each time a boundary is crossed by the beam, the program enters the zone and region search routine and the distance to boundary routine to determine the new region the beam is entering and the distance to the next boundary in that region. These routines are described in Section IV.

A simplified flow chart of the process of beam attenuation, flux tally, and advancing of particles through a region is given in Figure 2. In this figure, N refers to the number of particles which start in a new region; N" refers to the number of particles which have been advanced to the boundary without suffering a collision; I_0 is the initial weight of the beam; I is the attenuated weight of the beam at a given boundary; S_b is the distance to the next boundary in the direction α, β, γ; and S_c is the distance to collision for a given particle. In studying this flow chart, reference to Figure 1 and to the discussions in this section is necessary.

The following definitions are pertinent to Figure 2:

m = material index

g = microscopic group index

h = region index

λ_{mg} = mean free path for material m and group g

I_{min} = 10 percent of $I_0 = N/I_g I_h$

t = current particle time in microseconds (t = 0 for IV particles)

ξ = random number

V = velocity of particle

T = census time in microseconds.

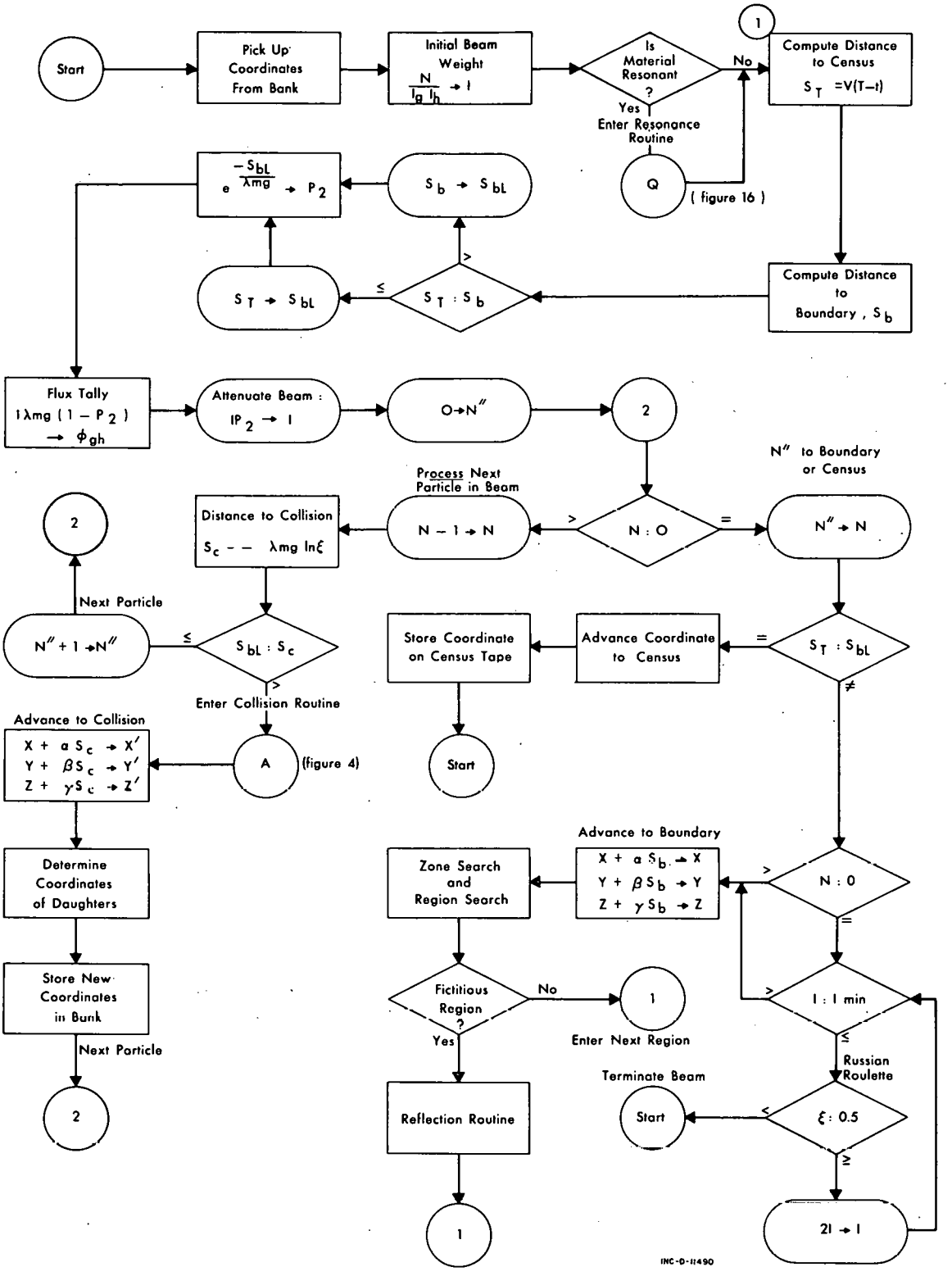


FIG. 2 SIMPLIFIED MONTE CARLO PROCEDURE.

III. NEUTRON COLLISION ROUTINES

1. DISTANCE TO COLLISION

In Section II the continuous collision analog flux tally for the beam was described. The number of neutrons per second dI which have collisions along the path of a beam I with total collision cross section Σ and traveling with the velocity v in dr at r is given by

$$-dI = I\Sigma v dt = I\Sigma dr$$

where the minus sign indicates a loss of neutrons. In this section the beam is described as possessing a dual beam-particle nature where only the discrete particles are actually advanced to collision. If particles were to go to collision continuously along the beam, the number per second dN having collisions in dr at r is

$$-dN = N\Sigma dr.$$

Upon integrating from an initial number of particles N_0 to N particles,

$$N = N_0 \, e^{-\Sigma r} .$$

The probability that a collision will occur in dr at r is given by

$$P(r)dr = -\frac{dN}{N_0} = \Sigma \, e^{-\Sigma r} dr.$$

The cumulative probability to r is given by

$$\int_0^r P(r)dr = \Sigma \int_0^r e^{-\Sigma r} dr .$$

As defined, the cumulative probability will lie between zero and one. Since it is a probability function, it can be interpreted as representing the random variable r, which is the distance to collision S_c.

$$\therefore \quad \xi = \Sigma \int_0^{S_c} e^{-\Sigma r} dr = 1 - e^{-\Sigma S_c} = 1 - e^{-S_c/\lambda}$$

where $\lambda = \frac{1}{\Sigma}$, the mean free path for the medium.

$$S_c = -\lambda \ln(1-\xi)$$

Since ξ is a random number, $1 - \xi$ will also be a random number.

$$\therefore \quad S_c = -\lambda \ln \xi$$

The random number generator used in this program is described in Reference 3. Upon request, it produces a so-called random number ξ, where $0 < \xi < 1.0$.

Because the mean free path λ changes when a boundary is crossed, the particle cannot be advanced to collision beyond a region boundary. (The collision probability distribution changes abruptly across a region interface.) Therefore, S_c for each particle is compared to S_b; and if S_c is greater than S_b, the particle is advanced to the boundary. Although it is, in principle, easy to construct the cumulative probability function for particles traversing several regions of different mean free paths, one does not know in general which region is being entered when a boundary is crossed in a general three-dimensional Monte Carlo problem. For this reason it is necessary to stop at the boundary until the new region being entered is determined. Then, a new random number defines a new distance to collision in the next region, and the same process is repeated. Since a constant plus a random number is still a random number, this procedure is equivalent to advancing to collision across several boundaries by means of the distributed probability function for several regions. The constant in this case is the constant associated with moving the particle a fixed distance, S_b, to the boundary before entering the new region.

The particle is advanced to collision in this manner, but the type of collision is yet to be determined.

2. COLLISION EMISSION NUMBERS

Given that a collision occurs at S_c, the probabilities for the various nuclear events of absorption, scattering, and fission are p_a, p_s, and p_F, respectively, where

$$p_a = \frac{\Sigma_a}{\Sigma} \ , \ p_s = \frac{\Sigma_{so}}{\Sigma} \ , \ p_F = \frac{\Sigma_F}{\Sigma} \ .$$

The average number of neutrons which will be emitted from each event is ν_a, ν_s, and ν_F, respectively. The average number of neutrons expected from each collision, due to the various events, is the sum of the neutrons from each event times the probability that the event will occur.

$$N' = 0 \cdot p_a + 1 \cdot p_s + \nu_F p_F = \text{expected number of neutrons from collision.}$$

$P_{cmg} = \nu_c P_{cmg}$ is defined as the collision emission number for the type c event in material m, group g. These collision emission numbers are used to determine the types and rates of collision in the Monte Carlo process. It is observed that $P_a = 0$, and, consequently, no time is spent processing purely absorption events since the effect of absorption is merely to reduce the other collision emission numbers proportionately.

Each time a particle advances to collision, P_{cmg} particles are expected to be emitted from each possible event. These P_{cmg} particles constitute the origin of new beams at each collision, and $N' = P_{cmg}$ is the number of particles in the new beam. The P_{cmg} are not usually integers. However, the method of advancing particles to collision described earlier requires N' to be an integer. It is therefore necessary to integerize the number of particles, P_{cmg}, emerging from each event. This is accomplished by means of the so-called Greatest Integer Function (GIF) which truncates the argument supplied to it. N' is defined as the actual integer number of particles emerging from the collision event.

$$N' = \text{GIF}\,(P_{cmg} + \xi) = \left\lfloor P_{cmg} + \xi \right\rfloor$$

where ξ is a random number between 0 and 1. (Note: Henceforth in this report, the brackets $\lfloor\ \rfloor$ will mean "Greatest Integer Function".) The fact that $\langle N' \rangle = P_{cmg}$ can easily be shown. At times the GIF will be zero depending on the random number, ξ, chosen and the value of P_{cmg}. When this happens, the event is called an unsuccessful collision, and the next P_{cmg} for this material is processed in the order they have been listed by the problem requester in the input data.

For each successful collision event, the post collision routine for that type of event is entered to compute the new energy of the emergent particles and their new direction cosines. The new coordinate is placed in the storage bank and the next P_{cmg} in order is picked up and processed. The order which has been fixed for storage of the P_{cmg} is the fission value first, followed by the elastic scatterers. Since fast inelastic scattering data have not been included in this version of the code, the post collision routine consists of the post fission routine, the post elastic scattering routine for fast neutrons, and the post elastic scattering routine for thermal neutrons (thermal scattering routine).

3. POST FISSION ROUTINE

N' particles will result from a successful fission event, as described in the last section. A cumulative fission spectrum probability table for the fissionable isotope involved must be supplied with the input data. The table consists of 21 cumulative probabilities, P_n, and their associated lower energy cutoffs, E_n, for the range $0 < P_n \leq 1.0$.

A random number, ξ, is successively compared with each P_n until

$$P_n < \xi \leq P_{n-1}$$

$$\therefore\ E_n > E \geq E_{n-1}\ .$$

Linear interpolation is then used to fix E' between E_n and E_{n-1}:

$$E' = E_{n-1} + \frac{\xi - P_{n-1}}{P_{n-1} - P_n}\ (E_{n-1} - E_n)\ .$$

The fission spectrum table number is assigned in the input for each material along with the P_{cmg} value for that material. Only one fissionable isotope is permitted in each energy group of a material.

The new direction cosines for the emergent fission particles are selected from a uniform distribution; ie, the fission particles are assumed to be emitted isotropically in the laboratory system of coordinates. The direction in which the beam is traveling is specified by the three direction cosines, α, β, γ. They are defined in terms of the polar and azimuthal angle, indicating the direction of the beam relative to the fixed reactor reference frame. This fixed reference frame is such that the whole reactor lies in the positive upper right-hand octant of a sphere.

$$\alpha = \sin\theta \cos\phi$$

$$\beta = \sin\theta \sin\phi$$

$$\gamma = \cos\theta$$

The probability of a randomly distributed point on the unit sphere lying in any annulus $d\theta$ at θ is given by

$$P(\theta)d\theta = -1/2\ d(\cos\theta) = -1/2\ d\mu$$

$$\xi = \int_{-1}^{\gamma} (-1/2)\ d\mu = -1/2\gamma + 1/2$$

$$\gamma = 1 - 2\xi\ .$$

Any one of a number of schemes can be used to find α and β by means of random numbers, but all methods are not equally efficient. If the azimuthal angle, ϕ, is selected from a uniform distribution, then $\cos\phi$ and $\sin\phi$ must be computed, and each of these takes more time than a square root calculation. The method used in this program requires only one square root calculation.

Obtain two random numbers, p and q, where $p = 2\xi_1 - 1$, $q = 2\xi_2 - 1$ ($0 < \xi_1, \xi_2 < 1.0$). If $p^2 + q^2 > 1.0$, the point $p^2 + q^2$ lies outside a circle of unit radius. In this case, reject these values and select new values of p and q to repeat the process until $p^2 + q^2 \leq 1.0$.

Then,

$$\sin\phi = \frac{2}{\sqrt{p^2 + q^2}} \quad \text{and} \quad \cos\phi = \frac{q}{\sqrt{p^2 + q^2}}$$

$$\alpha = \frac{p\sqrt{1 - \gamma^2}}{\sqrt{p^2 + q^2}} = p\sqrt{\frac{1 - \gamma^2}{p^2 + q^2}}$$

$$\beta = q\sqrt{\frac{1-\gamma^2}{p^2 + q^2}}\ , \text{ using the value for } \gamma \text{ found above.}$$

α is given the same sign as p, and β the same sign as q.

These random direction cosines are assigned to the emitted fission particles and stored along with the other coordinates in the storage bank.

4. POST ELASTIC SCATTERING ROUTINE FOR FAST NEUTRONS

In Section III-2, the elastic scattering collision emission number was defined as though only one isotope contributed all elastic scattering in the material. In actual practice, each material will be made up of a mixture of isotopes. In order to calculate the proper energy losses for each isotope involved, the scattering model must use the exact mass of that isotope in the transformation from the center of mass system of coordinates to the lab system. This exact treatment of fast scattering requires that each isotope in the mixture be treated as a separate event and be assigned its own P_{cmg}. The order in which these elastic scatterers are assigned in the input is arbitrary. The mass of a scatterer is listed with each P_{cmg} for elastic scattering. The collision routine then sequences through the list of elastic P_{cmg}'s, treating each one independently.

No provision for anisotropic scattering in the center of mass system is included in this version of the program, and all elastic scattering is treated as being isotropic in the center of mass system. For any isotope having a successful collision, a set of random direction cosines is selected as described in Section III-3. These are then the isotropic direction cosines, α'', β'', and γ'', in the center of mass for this isotope. In Section III-3 above, these random direction cosines were interpreted as being relative to the fixed reactor frame of reference. A proper treatment of elastic scattering would seem to necessitate that they now be interpreted as relative to the incident direction of the elastically scattered particle. Actually, the reference frame is arbitrary since the direction cosines are isotropically distributed, and great simplification occurs in the equations for transformation to lab coordinates if they are interpreted to be relative to the fixed reactor frame in the center of mass system. Let α', β', and γ' be the final direction cosines of the scattered particle in lab coordinates relative to the fixed frame of reference. The cosine of the angle of scattering in the lab system is given by the inner product of the vectors before and after collision:

$$\cos\theta = [\alpha', \beta', \gamma'] \cdot [\alpha, \beta, \gamma] = \alpha'\alpha + \beta'\beta + \gamma'\gamma.$$

Similarly, if the random direction cosines α'', β'', and γ'' are taken relative to the fixed frame of reference, the angle of scattering in the CM system is given by

$$\cos\Psi = [\alpha'', \beta'', \gamma''] \cdot [\alpha, \beta, \gamma] = \alpha''\alpha + \beta''\beta + \gamma''\gamma.$$

From elementary reactor theory, the transformation from the center of mass to the lab system for elastic scattering from a target at rest in the lab system is given by

$$\cos\theta = \frac{A\cos\Psi + 1}{\sqrt{A^2 + 2A\cos\Psi + 1}} \qquad (1)$$

$$\frac{E'}{E} = \frac{A^2 + 2A\cos\Psi + 1}{(A+1)^2} \qquad (2)$$

where A is the atomic mass of the isotope. This relationship is obtained by the addition of the velocity vectors from the CM to lab system as illustrated in Figure 3.

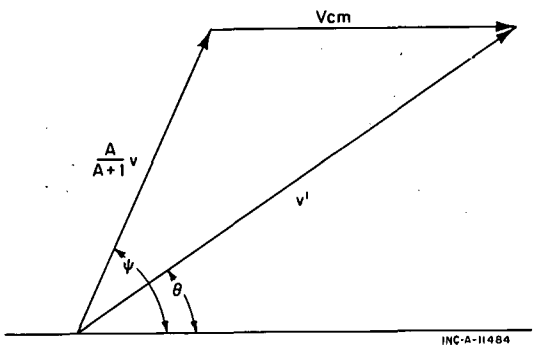


FIG. 3 CM TO LAB TRANSFORMATION.

In the figure, $\frac{A}{A+1}$ v is the velocity of the neutron relative to the center of mass, $V_{CM} = \frac{v}{A+1}$ is the velocity of the center of mass, and v' is the velocity of the neutron in the lab after collision. The direction of motion of the center of mass is (α, β, γ), since these are the direction cosines of the incident particle.

If the x, y, z components of V_{CM} and $\frac{A}{A+1}\vec{v}$ are added independently to give the x, y, z components of $\vec{v}'$, the following equations result:

$$\alpha' v' = \frac{A}{A+1} v\alpha'' + \frac{\alpha v}{A+1} = (A\alpha'' + \alpha) \frac{v}{A+1}$$

$$\beta' v' = (A\beta'' + \beta) \frac{v}{A+1}$$

$$\gamma' v' = (A\gamma'' + \gamma) \frac{v}{A+1} \quad .$$

Adding the sum of the squares of both sides of these equations gives

$$(v')^2 = C^2 \frac{v^2}{(A+1)^2}$$

where

$$C^2 = (A\alpha'' + \alpha)^2 + (A\beta'' + \beta)^2 + (A\gamma'' + \gamma)^2 \quad .$$

Substituting this expression for v' into component equations gives

$$\alpha' C = A\alpha'' + \alpha$$

$$\beta' C = A\beta'' + \beta$$

$$\gamma' C = A\gamma'' + \gamma \quad .$$

The transformation equations from the center of mass to the final laboratory direction cosines become:

$$\alpha' = \frac{A\alpha'' + \alpha}{C}$$

$$\beta' = \frac{A\beta'' + \beta}{C} \qquad (3)$$

$$\gamma' = \frac{A\gamma'' + \gamma}{C}$$

$$E' = \frac{C^2 E}{(A+1)^2} \quad .$$

Substitution of the values for $\cos\theta$ and $\cos\Psi$ in terms of these direction cosines produces Equations (1) and (2) and serves to verify the transformation.

Thus, the complete transformation is given in terms of the random direction cosines and the mass of the scatterer. If a material is made up of several heavy elements, where the heavy element scattering is relatively unimportant, then these heavy elements may be lumped together in this material and assigned an average mass. The transformation will be based on this mass. A further option is included for elastic scattering in that any element which is assigned a mass of zero in the input will be treated as a scatterer of infinite mass. The transformation equations become:

$$\alpha' = \alpha''$$

$$\beta' = \beta''$$

$$\gamma' = \gamma''$$

$$E' = E .$$

Transformation equations for scatterer whose mass is set to zero in the input.

Although anisotropic scattering coefficients are not provided for in this program, the treatment of anisotropic scattering in the CM system can be handled in terms of the above equations for elastic scattering. In this case the inner product of a set of random direction cosines α'', β'', γ'' is taken with the incident direction cosines α, β, γ.

$$\cos\Psi = \alpha''\alpha + \beta''\beta + \gamma''\gamma$$

The anisotropic CM scattering is specified in terms of a truncated set of Legendre coefficients, W_1, W_2, . . . , W_n, where the anisotropic scattering distribution function is given by

$$P(\mu)\, d\mu = (A_0 + A_1\mu + A_2\mu^2 + \ldots + A_n\mu^n)\, d\mu \quad . \qquad (4)$$

The coefficients A_0, A_1, . . . , A_n are obtained from the Legendre coefficients by means of a simple recombination of the terms in the Legendre series into a power series which is more suitable for machine computation. If $\cos\Psi$, as computed above with random direction cosines α'', β'', γ'', is substituted for μ in Equation (4), $P(\cos\Psi)$ is obtained. This value of μ is accepted if $\xi\overline{P} < P(\mu)$, where ξ is a random number and $\overline{P}$ is the maximum value of $P(\mu)$ in the range $\leq 1 \leq \mu \leq 1$; otherwise, it is rejected. If $\cos\Psi$ is rejected, a new set α'', β'', γ'' is chosen, and the process is repeated. When $\cos\Psi$ is accepted, the transformation to the lab for this isotope proceeds by way of Equations (3).

5. THERMAL SCATTERING ROUTINE

Because of the complexity of thermal inelastic scattering cross sections for atoms and molecules, the thermal inelastic scattering process is not easily adaptable to exact treatment in the Monte Carlo process. Although exact treatments of the gas model equations are possible by Monte Carlo[1], they represent a poor approximation to physical processes for molecular mixtures. Numerous thermal spectrum programs[7] exist which treat homogeneous mixtures of thermal inelastic scatterers and produce average few-group spectrum-weighted thermal cross sections. The present program is based on the use of these cross sections from a thermal spectrum code for a monoenergetic linearly anisotopic treatment of the thermal scattering process in the lab system. For each material, average Σ_{s0} and $\bar{\mu}_0$ values from the thermal spectrum code for this material mixture are specified in the input. The lab scattering probability density function for this mixture is

$$P(\mu)d\mu = (1/2 + 3/2\ \bar{\mu}_0\ \mu)\ d\mu \quad .$$

A set of random direction cosines α', β', γ' is chosen and μ is obtained from the inner product:

$$\mu = \alpha'\alpha + \beta'\beta + \gamma'\gamma \quad .$$

If $P(\mu) \geq \xi\ \bar{A} = \xi(1/2 + 3/2\ \bar{\mu}_0)$, then the random direction cosines α', β', γ' are accepted as the final lab direction cosines; if rejected, a new set α', β', γ' is chosen, and the process is repeated. Here, ξ is a random number, and $\bar{A} = 1/2 + 3/2\ \bar{\mu}_0$ is the maximum value of the function $P(\mu)$ in the interval $-1 \leq \mu \leq 1$. The energy of the thermally scattered particle remains at the same value the particle had when it initially entered the thermal group. The cutoff energy of the thermal group is arbitrary, and particles are treated as fast neutrons until they enter the thermal group. As more sophisticated thermal spectrum codes become available, higher order terms may be added to the function $P(\mu)$ so that a more exact treatment in the lab system for an anistropic thermal process can be made.

A special thermal scattering routine is being studied whereby double differential elastic scattering data will be processed in tabular form for a multigroup structure in the thermal region. In this way, inelastic scattering laws for various moderators may be treated with minimal approximation. When perfected, this version of the thermal scattering routine may be added to PMC as an alternative to the present monoenergetic treatment of thermal neutrons.

One of the merits of the monoenergetic treatment of thermal neutrons is that it is very fast compared to such models as the gas equations, which are also approximations compared to the actual physical processes involved. Since a large proportion of the total number of collisions in water-moderated systems involve thermal scattering, the thermal scattering process must not require a large amount of computation time if the program is to be practical.

If multigroup epithermal results are desired, the thermal group collisions may be shunted by specifying that the scattering P_{cmg} for the thermal group (last group) be zero for each isotope. Then, a given daughter entering the thermal group will suffer only one collision and be terminated. The thermal group cutoff

is arbitrary and is specified in the input. A truly monoenergetic problem may be run as a one-group (thermal) problem in which either fast or thermal group cross sections may be used as desired. In this case the source energy must be specified to be within this (thermal) group.

In Figure 4, a simplified flow diagram of the collision routines is presented. The following are definitions of terms used in Figure 4:

t = current particle time in microseconds

V = velocity of particle

P_{cmg} = collision emission number for this event

F_m = 1, material is fissionable; F_m = 0, material is not fissionable

i = collision event index

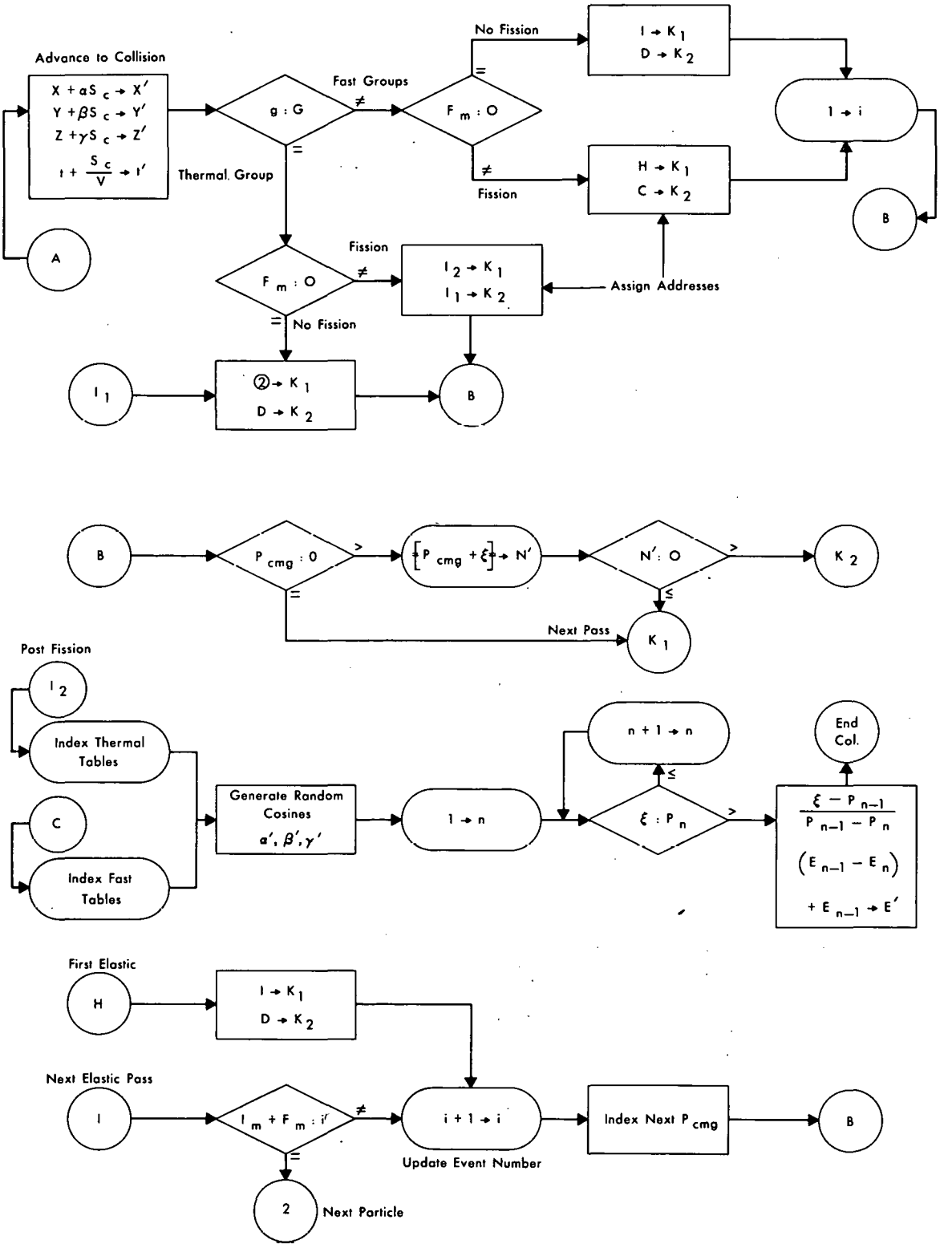
n = fission spectrum table index

I_m = total number of elastic scatterers in the material

A_{cm} = mass of scatterer (A_{cm} = 0 implies that scatterer is isotropic in lab system)

$\bar{\mu}_0$ = average thermal cosine of the polar scattering angle in the lab system

$\bar{A}$ = $1/2 + 3/2\ \bar{\mu}_0$ = maximum value of the function $P(\mu) = 1/2 + 3/2\ \bar{\mu}_0\mu$.



NOTE : K_1 and K_2 are Variable Address Switches Assigned for The Various Collision Events . (NC-D-11491)

FIG. 4A COLLISION ROUTINE.

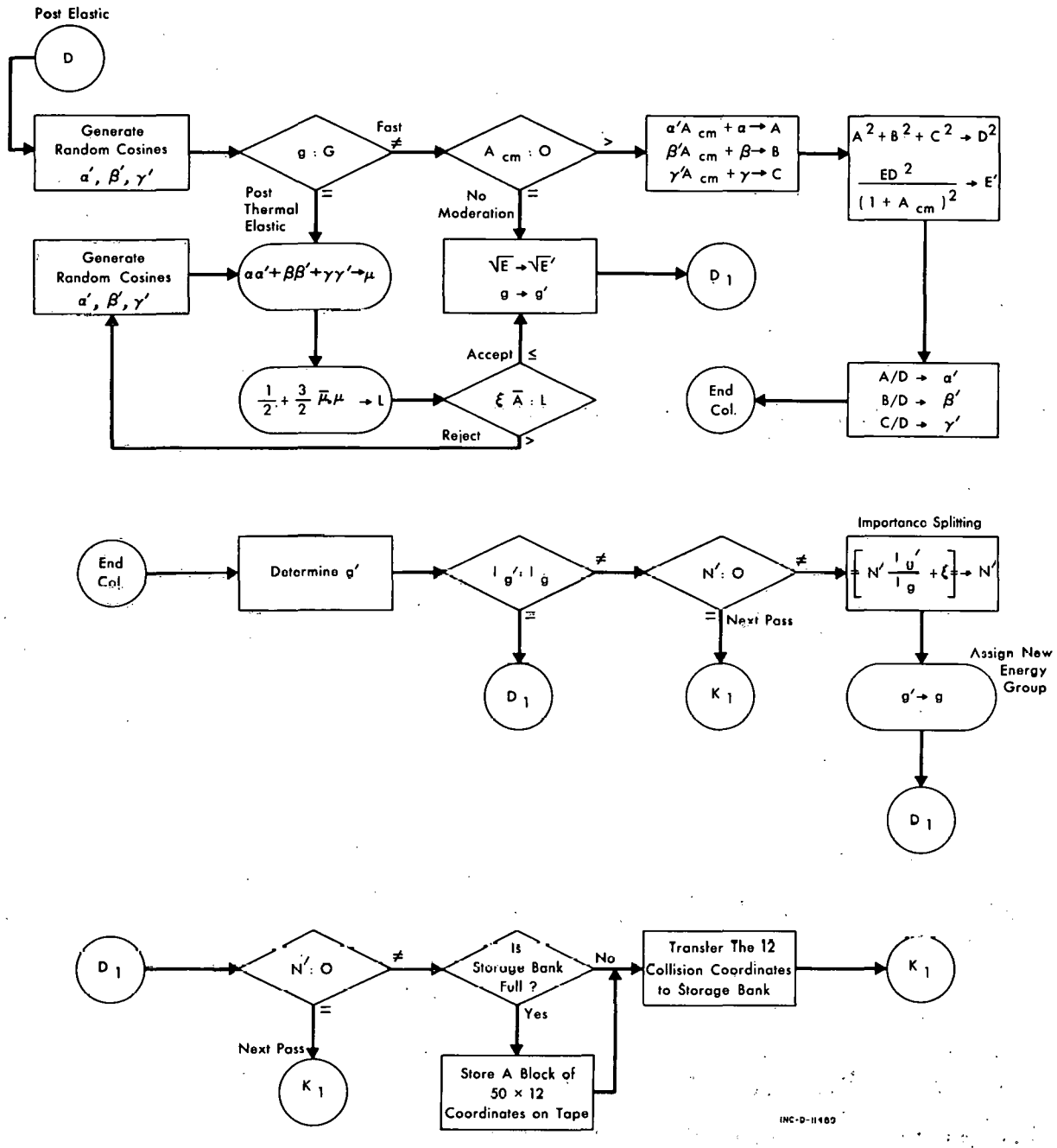


FIG. 4B COLLISION ROUTINE.

IV. GEOMETRY SPECIFICATIONS AND SEARCH ROUTINE

Each time a beam crosses an internal or an external boundary, it is necessary to find out which region has been entered. The program will locate the beam using the geometry information supplied to it in the input data. A reactor problem is described in terms of boundaries, zones, and regions. The following sections describe how regions and zones are constructed, how the search routines function, and how the distance to the next boundary in the new region is obtained. Also included is a description of external boundary reflection and transmittal and the tallies which result.

1. GEOMETRY CONSTRUCTION FOR REGIONS

The complete volume under study, including fictitious regions (described in Section IV-2), must always be placed in the first octant of the standard rectangular x, y, z coordinate system. Therefore, computed x, y, or z coordinates should never be negative. The following types of surfaces may be used to describe the region boundaries in a problem:

(1) Planes, of the general form $Ax + By + Cz = D$

(2) Points and lines
(3) Cylinders
(4) Spheres

} of the general form $(x-x_0)^2 + (y-y_0)^2 + C(z-z_0)^2 = R^2$ where C = 0 (cylinder) or 1.0 (sphere).

Planes are restricted to those lying perpendicular to the x, y, or z axis and to those which are perpendicular to the z-axis but not necessarily parallel to the x or y axis. Acceptable planes are

$Ax+By = D$

$Ax = D$

$By = D$

$Cz = D.$

Lines and points are used to indicate the center of cylinders and spheres and are described as a cylinder or sphere of radius 0.

A cylinder must lie such that its axis is parallel to the z-axis, as $(x-x_0)^2 + (y-y_0)^2 = R^2$, where x_0, y_0 is the center of the cylinder with respect to the x, y, z reference system.

Spheres are described by the standard equation:

$$(x-x_0)^2+(y-y_0)^2+(z-z_0)^2 = R^2.$$

There are four basic region types, differentiated by having 0, 1, or 2 curved boundaries. In preparing input for Input Group 8 (Section VIII-3), reference must be made to the following sections to determine the proper region type.

1.1 Region Type 1

These regions are those which have no curved boundaries. Examples are rectangular parallelepipeds and polygonal (in the x-y plane) volumes (see Figure 5). These regions are described by five to eight plane boundaries. Planes intersecting the z-axis must be parallel to the x-y plane. These planes may have any arbitrary positive z coordinate.

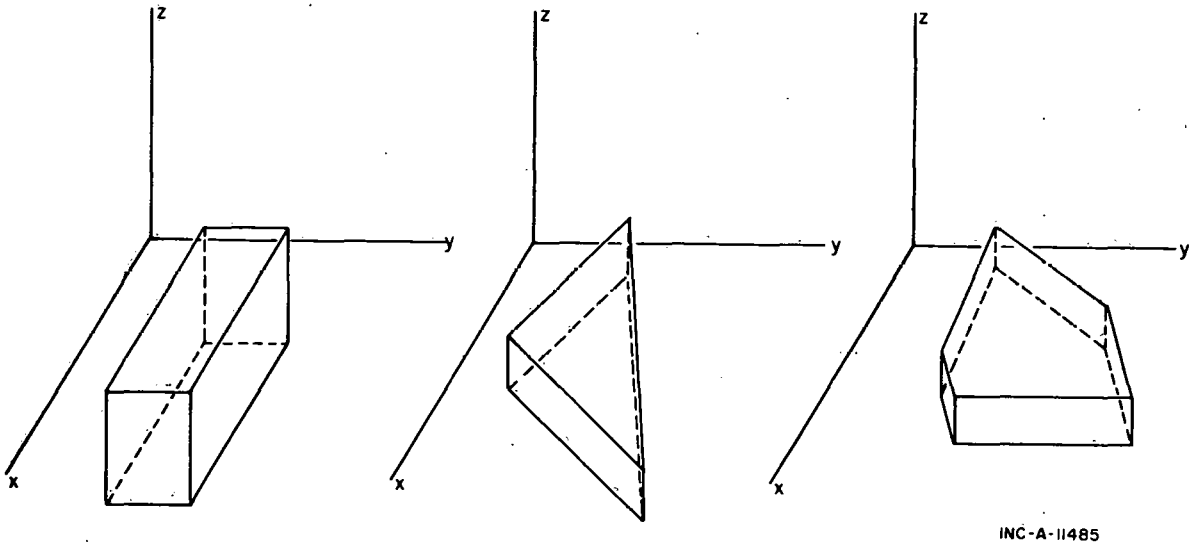


FIG. 5 REGIONS OF TYPE 1.

1.2 Region Type 2

These regions are those having one curved boundary and from five to seven plane boundaries. Planes intersecting the z-axis must be parallel to the x-y plane. The following examples (Figure 6) in the x-y plane of region type 2 (shaded areas) show some of its uses. These are not the only possible situations, but are sufficient to illustrate the possibilities. The unshaded regions may be either cylinders or spheres.

1.3 Region Type 3

These regions are those having two curved boundaries with two to six plane boundaries. This type contains a wide variety of forms, some of which are shown in Figure 7. Again, planes intersecting the z-axis must be parallel to the x-y plane. All cylinders are considered concentric. Therefore, in the case of a single cylinder, the region description must include a line cylinder of radius 0. This also holds true whenever a region lies interior to a sphere or cylinder. All of the cylinders described in the above examples must be bounded by top and bottom planes in the z dimension. These planes may be independent of those bounding any adjacent regions.

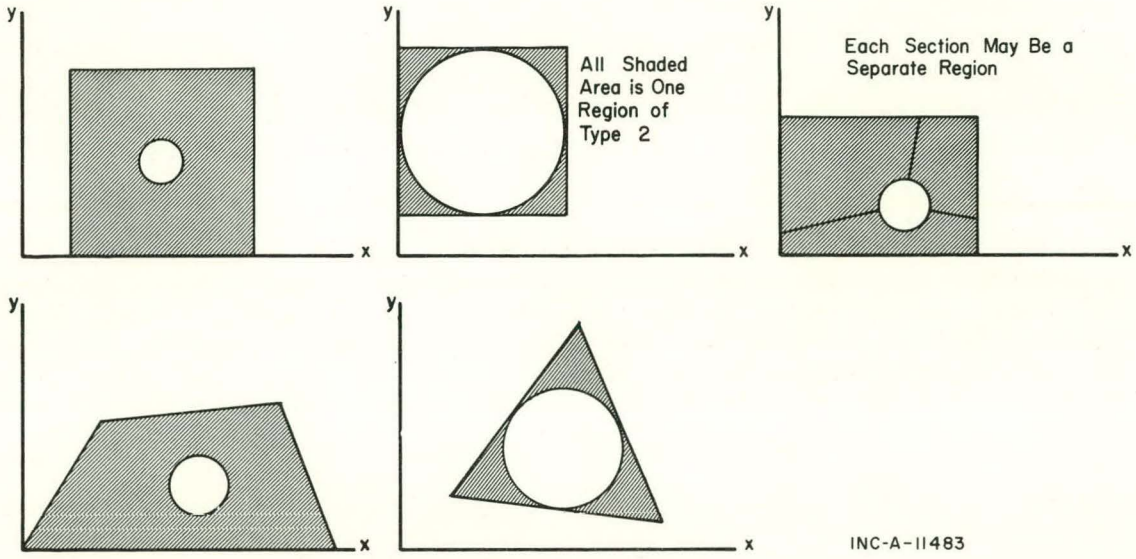


FIG. 6 REGIONS OF TYPE 2.

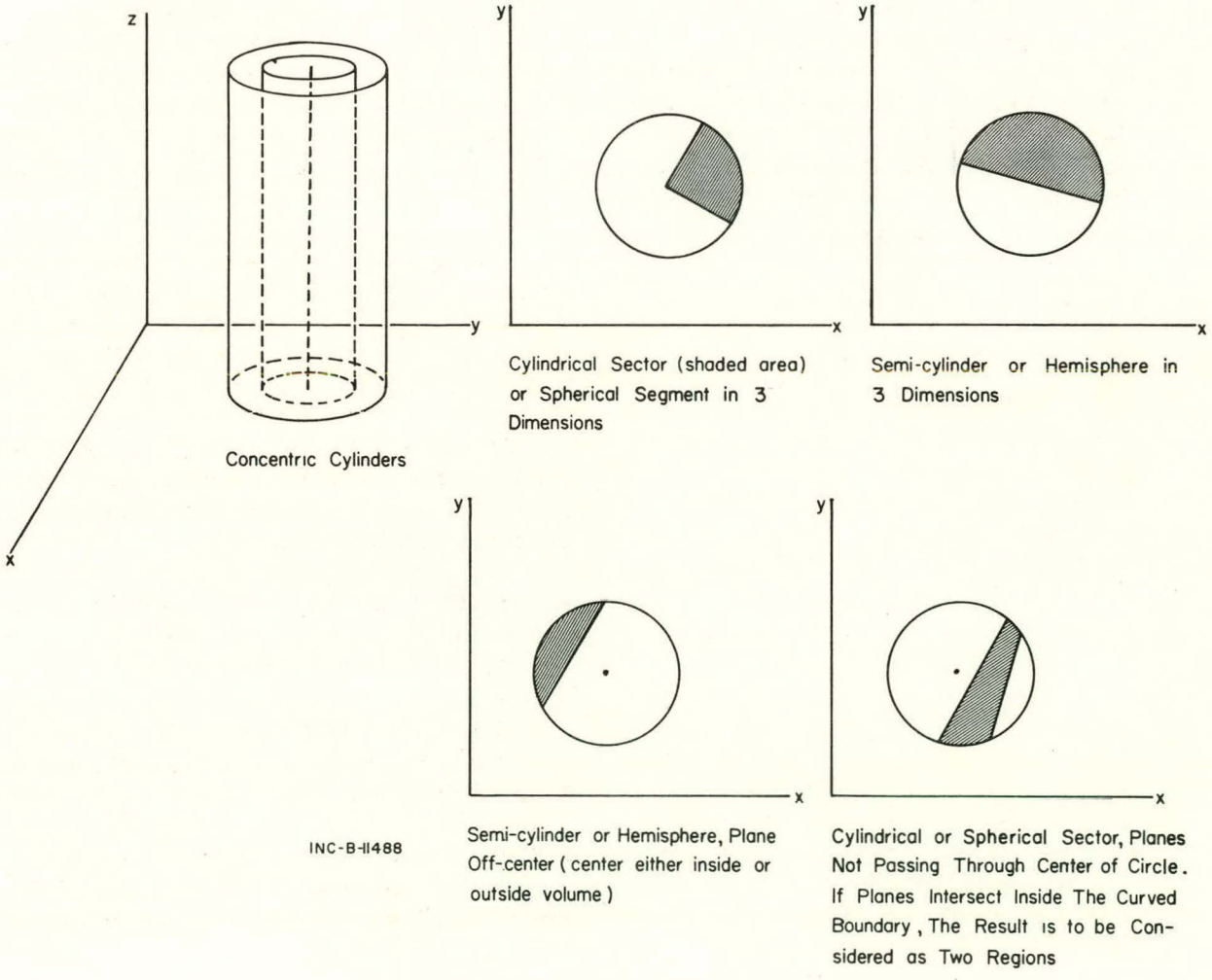


FIG. 7 REGIONS OF TYPE 3.

1.4 Region Type 4

These regions are those which have no plane boundaries. This region type is limited to concentric spheres which have two curved boundaries. All spheres are considered concentric. Therefore, for a single sphere a point must be included in the center of the sphere as part of the region description. This center point is a sphere of radius zero.

The two-boundary description of a single cylindrical or spherical volume is necessary because the distance to boundary routine has been programmed on the basis of a cylindrical or spherical annulus. Therefore, the area inside a single cylinder or sphere becomes a cylindrical or spherical annulus by including a center line or point as the second surface. The general routines will then work in this case also.

2. FICTITIOUS REGIONS

Assume that a beam has just been advanced to a boundary. In order to find out which region the beam is entering, the beam must have actually crossed the boundary of the region. A problem arises when the beam crosses an outer boundary of the reactor volume under study. At this external boundary, the beam will be reflected or passed through the boundary, depending on the value of the albedo assigned to that outer boundary. Since the beam is now physically outside the outer boundary, it does not lie in a defined region. The program must be able to resolve this ambiguity. As illustrated in Figure 8, to know which boundary was crossed is not sufficient to tell where the particle is.

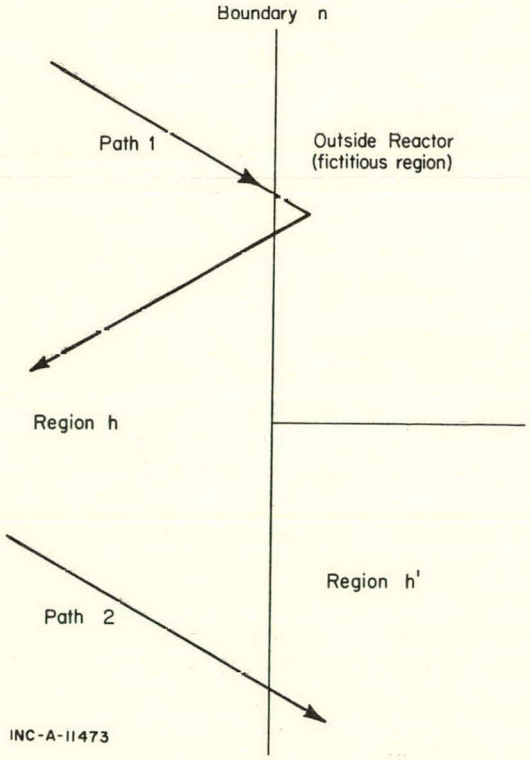


FIG. 8 USE OF FICTITIOUS REGIONS.

Along path 1, in general, the beam will be reflected after reaching the boundary since to continue to advance in the same direction would put the particle outside the reactor.

However, along path 2 we want to continue the beam in the same direction, passing on into region h', which is a region inside the reactor.

Since boundary n is crossed in both cases, there must be a method of differentiating the two situations.

To handle this problem "fictitious regions" or buffer volumes, completely surrounding the reactor cell, are used. Whenever a beam enters a fictitious region, the program can determine that a reflective boundary has been crossed and can then proceed to the reflection subroutine.

The description of a fictitious region is of the same form as an ordinary region. In the Monte Carlo input, fictitious regions follow the ordinary regions in sequence and are numbered beginning with 401, continuing consecutively. The region set number and material are both designated zero. The width of the fictitious region buffer zone is arbitrary but should be wide enough to eliminate the possibility of machine roundoff errors when the particle enters the region. A suggested width is 0.5 to 1.0 cm.

Fictitious regions are also used to tally the number of particles and neutrons which have escaped from a particular surface of the volume under study. No flux will be tallied in these regions.

As an example to show how these fictitious regions are constructed, let the reactor cell under study be a cylinder. This is surrounded with a second cylindrical region one centimeter larger in radius than the first one, and it can be called fictitious region 401. On the top and bottom of these cylinders are included two more slabs extending one centimeter above and below the planes defining the z-direction limits of the reactor cell. The volumes thus formed are fictitious regions 402 and 403. The reactor cell is now completely surrounded by fictitious regions.

3. RULES OF SIGN CONVENTION FOR REGION BOUNDARY NUMBERS

All boundaries describing the regions of the problem under study must be one of the standard forms given in Section VIII-3, Input Group 5. These boundary equations are assigned consecutive numbers by the user, independent of the regions they describe. In the region description (Input Group 8), plane boundaries are given a sign -- the sign depending upon the location of the region relative to the boundary and the fixed reference frame.

No sign and a positive (+) both indicate a positive boundary. All curved boundaries are to be assigned positive in the region description. In general, relative to the fixed reference frame, a plane boundary is to be considered positive for a region if that region lies to the right of, or above, the boundary, and negative if the region lies to the left of, or below, the boundary.

As an example, in the x-y plane, take the square region h bounded by equations numbered 1 through 4 (see Figure 9).

The signed boundary numbers describing this region are 1, -2, 3, -4.

To determine analytically the sign of a boundary for a region, substitute the coordinates of an arbitrary point which lies within the region into the expression Ax+By+Cz-D, where A, B, C, and D are the constants which determine the plane boundary under question. If the resulting number is positive, consider the boundary positive for that region; if the result is negative, assign a minus sign to the boundary for that region.

For example, take the following triangular region in the x-y plane in Figure 10:

1. x=1.0
2. x=2.0
3. y=1.0
4. y=2.0

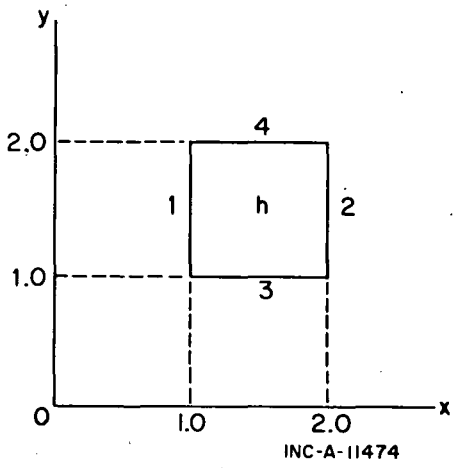
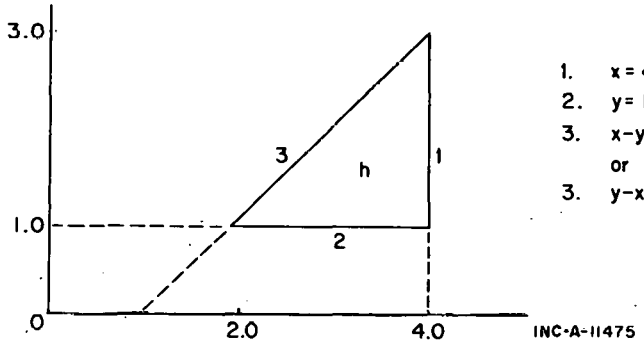


FIG. 9 BOUNDARY SIGN CONVENTION EXAMPLE 1.



1. x = 4.0
2. y = 1.0
3. x-y = 1.0
   or
3. y-x = -1.0

FIG. 10 BOUNDARY SIGN CONVENTION EXAMPLE 2.

The only boundary which is ambiguous is boundary number 3. Assume that the boundary is read in as x-y = 1. The arbitrary point (3, 1.1) is clearly within the region. Since Ax+By+Cz-D = 3.0 - 1.1 -1.0 > 0, assign boundary number 3 as positive for this region.

If, on the other hand, the boundary is read in as y-x = -1.0, then for the same point (3, 1.1), Ax+By+Cz-D = -3.0 + 1.1 + 1.0 < 0 and the minus sign must be given to the boundary number for this region.

Either equation may be used to describe the boundary, but care must be taken to see that the proper signs are specified when using this boundary to describe regions.

4. DISTANCE TO BOUNDARY ROUTINE

When a beam enters a new region traveling in the direction α, β, γ, the distance to the next boundary across the region must be computed using the general equations for plane and curved boundaries. Since it is not known at that point which boundary the beam will cross next, the distance to each boundary

in the region must be computed even though some of these distances may be negative. The true distance to boundary is the minimum positive distance. All other distances are rejected. Each region type takes a different path through the distance to boundary routine depending upon the number of curved boundaries.

4.1 Distance to a Plane Boundary

To compute the distance, S_b, to a plane boundary, $Ax+By+Cz = D$, from a point in space (x_1, y_1, z_1) traveling in the direction (α, β, γ), let p be the perpendicular distance from the point (x_1, y_1, z_1) to the plane, w, be a unit vector along p, S_b, be the distance in the direction (α, β, γ) to the plane, and v be a unit vector in the direction of S_b (see Figure 11).

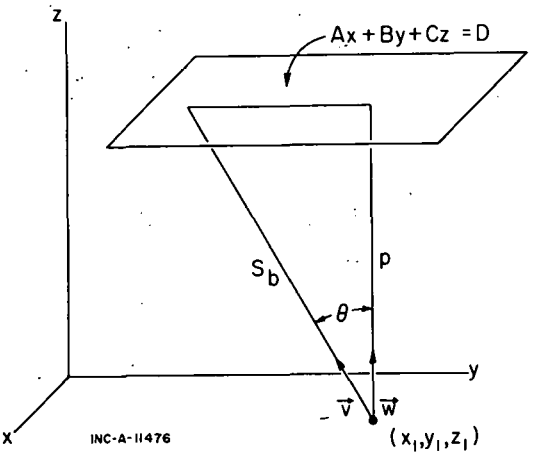


FIG. 11 DISTANCE TO A PLANE BOUNDARY.

$$\vec{w} = \alpha\vec{i} + \beta\vec{j} + \gamma\vec{k}$$

$$\vec{v} = \frac{A\vec{i} + B\vec{j} + C\vec{k}}{\sqrt{A^2 + B^2 + C^2}}$$

From elementary analytic geometry,

$$p = \frac{-(Ax_1+By_1+Cz_1-D)}{\sqrt{A^2+B^2+C^2}}$$

$$\frac{p}{S_b} = \cos\theta = \vec{w} \cdot \vec{v} = \frac{A\alpha+B\beta+C\gamma}{\sqrt{A^2+B^2+C^2}}$$

$$\therefore \quad S_b = \frac{p}{\cos\theta} = \frac{-(Ax_1+By_1+Cz_1-D)}{A\alpha+B\beta+C\gamma}$$

4.2 Distance to a Cylindrical Boundary

In Figure 12, p_1 (x_1, y_1, z_1) is the location of the particle before computing a distance to boundary, and p_2 (x_2, y_2, z_2) is the point where the path of the particle in the direction (α, β, γ) comes closest to the axis of the cylinder.

L is the distance between p_1 and p_2; L_{xy}, its projection on the x-y plane.

D is the perpendicular distance from p_2 to the axis of the cylinder.

R' is the radius of the point p_1.

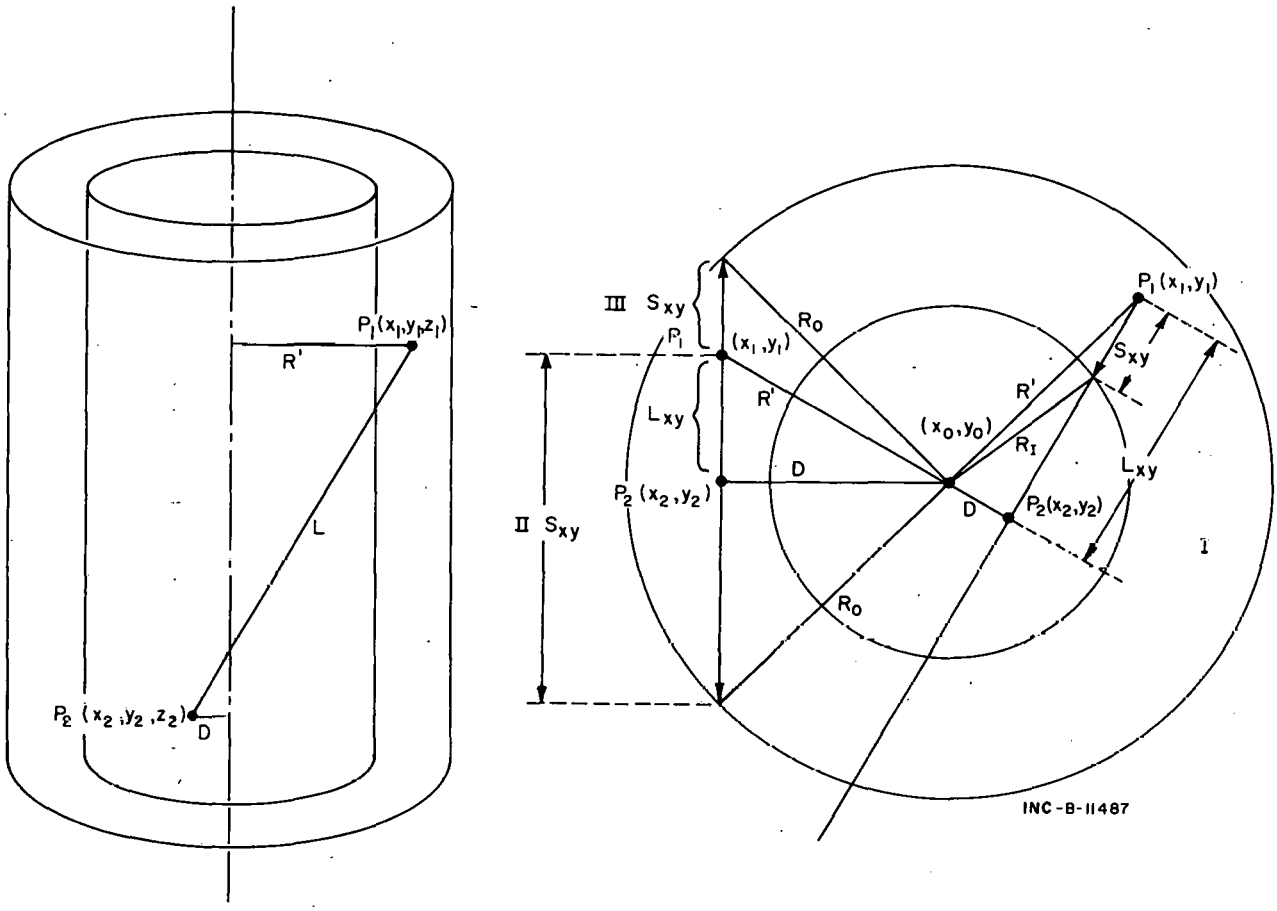


FIG. 12 DISTANCE TO A CURVED BOUNDARY.

R_o is the radius of the outer cylinder; R_I, that of the inner cylinder.

S_{xy} is the x-y projection of the three-dimensional distance S_b to a curved boundary.

Three cases for possible boundary crossings exist (Figure 12):

(1) The particle is headed inward and will cross the inner curve.

(2) The particle is headed inward, but will miss the inner curve thus crossing the outer curve (because D is larger than R_I).

(3) The particle is headed outward and will cross the outer curve.

Let

$$x' = x_1 - x_o, \quad y' = y_1 - y_o$$

$$N = \alpha^2 + \beta^2 = 1 - \gamma^2$$

$$K = \alpha x' + \beta y' \quad .$$

Then

$$\vec{R}' = x'\vec{i}+y'\vec{j}$$

$$R'^2 = x'^2+y'^2$$

$$\vec{D} = (x_2-x_o)\,\vec{i} + (y_2-y_o)\,\vec{j} \ .$$

The projection of L on the z-axis: $L_z = \gamma L$

$$L^2 = L^2_{xy} + L^2_z$$

$$L^2_{xy} = L^2-L^2_z = L^2(1-\gamma^2) = L^2N \ .$$

Similarly,

$$S^2_{xy} = S_b{}^2N \ .$$

By using the Pythagorean Theorem:

For Case (1)

$$S_{xy} = L_{xy} - \sqrt{R_I^2-D^2} \qquad S_b = L - \sqrt{\frac{R_I^2-D^2}{N}} \ .$$

For Case (2)

$$S_{xy} = L_{xy} + \sqrt{R_o^2-D^2} \qquad S_b = L + \sqrt{\frac{R_o^2-D^2}{N}} \ .$$

For Case (3)

$$S_{xy} = -L_{xy} + \sqrt{R_o^2-D^2} \qquad S_b = -L + \sqrt{\frac{R_o^2-D^2}{N}} \ .$$

Let $\vec{n}_o = \alpha\vec{i}+\beta\vec{j}+\gamma\vec{k}$ be a unit vector in the direction of particle travel.

$$x_2 = x_1+L\alpha, \quad \text{or } x_2-x_o = x_1-x_o+L\alpha = x'+L\alpha$$

$$y_2 = y_1+L\beta, \quad \text{or } y_2-y_o = y_1-y_o+L\beta = y'+L\beta$$

$$\therefore \vec{D} = (x'+L\alpha)\vec{i} + (y'+L\beta)\vec{j}$$

There is no z-axis component since D is perpendicular to the axis of the cylinder, which lies parallel to the z-axis.

Since $\vec{D} \perp \vec{n}_o$, $\vec{D} \cdot \vec{n}_o = 0$

$$\vec{D} \cdot \vec{n}_o = (x'+L\alpha)\alpha+(y'+L\beta)\beta = 0$$

$$L(\alpha^2+\beta^2)+\alpha x'+\beta y' = 0$$

$$L = -\frac{\alpha x'+\beta y'}{\alpha^2+\beta^2} = -\frac{K}{N}$$

$$\frac{\vec{R}'}{R'} \cdot \vec{n}_o = \cos(\vec{R}', \vec{n}_o)$$

$$\vec{R}' \cdot \vec{n}_o = \alpha x'+\beta y' = K .$$

If $\cos(\vec{R}', n_o) > 0$, then $\vec{n}_o$ is directed outward, away from the center. If $\cos(\vec{R}', \vec{n}_o) < 0$, then $\vec{n}_o$ is directed inward toward the center. L and -K will have the same sign, since $N > 0$. Therefore, when $L > 0$, $\vec{n}_o$ is inward, $L > 0$, $\vec{n}_o$ is outward.

In general then, the distance to boundary, S_b, is given by

$$S_b = L \pm \sqrt{\frac{R^2-D^2}{N}}$$

where $R = R_I$ and the negative sign is used if $L > 0$, and $D < R_I$ (Case 1)

$R = R_o$ and the positive sign is used if $L > 0$, and $D > R_I$ (Case 2)

$R = R_o$ and the positive sign is used if $L < 0$ (Case 3).

4.3 Distance to a Spherical Boundary

The equations for the cylindrical and spherical boundary crossings may be made the same by the following new definitions:

D is the perpendicular line drawn to the center of the sphere (x_o, y_o, z_o) from the line L.

$$z' = z_1 - z_0$$

$$R'^2 = x'^2 + y'^2 + z'^2$$

$$N = \alpha^2 + \beta^2 + \gamma^2 = 1$$

$$K = \alpha x' + \beta y' + \gamma z'$$

Since $\vec{D}$ now has a z-component,

$$\vec{D} = (x' + L\alpha)\vec{i} + (y' + L\beta)\vec{j} + (z' + L\gamma)\vec{k}$$

resulting in

$$L = -\frac{\alpha x' + \beta y' + \gamma z'}{\alpha^2 + \beta^2 + \gamma^2} = \frac{-K}{N} = -K .$$

With these definitions, the equations for the cylinder may be used to compute the distance to a spherical boundary.

5. SEARCH ROUTINES

The PMC search routine consists of two parts, the zone search and the region search. Any beam crossing a boundary is subjected to both the zone and region searches in order to locate the region being entered. Locating a region within the geometry framework is analogous to trying to locate a given county from the whole United States. On the average, half the counties in the country would have to be tested, one by one, in order to locate the county in question. The work would be made much easier and faster if the state in which the county is located were known. Then only the counties of that state need be tested. The purpose of the zone search, then, is to locate the "state" or larger subdivision of the geometry framework in which a particle is located. This greatly speeds up the region search.

There are three kinds of zone geometries: parallelepiped, cylindrical, and spherical. The user divides a Monte Carlo problem into zones using zone lines which are measured from the zone geometry center point along the x, y, and z axes (parallelepiped), radial and z axes (cylindrical), or radial axis (spherical). Each zone may contain from one to seven regions (including fictitious regions).

5.1 Parallelepiped Zones

Zones are numbered in increasing order, first in the z-direction, then in the y-direction, and finally in the x-direction. They may be compared to a pile of blocks. Block No. 1 is located nearest the zone geometry center point. Block No. 2 is just above it, No. 3 above that, and so on up the z-axis until the top of the pile is reached. The next higher numbered block will be at the bottom of the pile again, one block along the y-axis outward from block No. 1. This pile will again numerically increase along the z-axis, and so on until a wall one block thick has been built along the y-axis. Next, along the x-axis, walls parallel to the first wall are built until the zone blocks have completely covered the volume under study.

Zone lines will be read in along the x, y, z axes, respectively, as a_i, b_j, c_k (i = 1, ..., I; j = 1, ..., J; k = 1, ..., K), where I, J, and K are the total number of zone lines along the axes and a_1, b_1, c_1 are all 0. There are, then, a total of (I-1) (J-1) (K-1) zones.

Let (x_1, y_1, z_1) be the coordinate of a point. Along each axis the point will lie between two zone lines. A zone index (i, j, k) may be found from the equations:

$$a_i \leq x_1 \leq a_{i+1}$$

$$b_j \leq y_1 \leq b_{j+1}$$

$$c_k \leq z_1 \leq c_{k+1} \quad .$$

The zone number N is then given by

$$N = (J-1)(K-1)(i-1) + (K-1)(j-1) + k.$$

5.2 Cylindrical Zones

Zones are numbered in increasing order, first in the z-direction, then radially. In the block illustration, build with cylinder shaped blocks beginning with block No. 1 nearest the zone geometry center point; build upward along the z-axis until the top is reached. The next and following layers of blocks will be annuli and are built up like putting doughnuts on a stick. Each time the top of the pile is reached, start again at the bottom with a new set of doughnuts larger in radius.

Zone lines will be read in along the r and z axes, respectively, as R_i, C_k ($i = 1, \ldots, I$; $k = 1, \ldots, K$), where I and K are the total number of zone lines along the axes and R_1 and C_1 are 0. The R_i are radii with respect to the zone geometry center point. There is a total of (I-1) (K-1) zones.

Let (x_1, y_1, z_1) be the coordinate of a point and (x_o, y_o, z_o) be the zone geometry center point. The zone index (i,k) may be found from the equations

$$R_i^2 \leq (x_1-x_o)^2 + (y_1-y_o)^2 \leq R_{i+1}^2$$

$$C_k \leq z_1 \leq C_{k+1} \quad .$$

The zone number N is given by

$$N = (K-1)\ (i-1) + k \quad .$$

5.3 Spherical Zones

Zones are numbered in increasing order radially, beginning nearest the zone geometry center point. Zone lines will be read in along the r-axis, as R_i ($i = 1, \ldots, I$), where I is the total number of zone lines and $R_1 = 0$. There is a total of I-1 zones.

Let (x_1, y_1, z_1) be the coordinates of a particle and (x_o, y_o, z_o) be the zone geometry center point. The zone index, i, may be found from

$$R_i^2 \leq (x_1-x_o)^2 + (y_1-y_o)^2 + (z_1-z_o)^2 \leq R_{i+1}^2 .$$

The zone number N is just i.

To illustrate how to find the zone number of a particular zone using cylindrical zones, take two concentric cylinders cut in half (a total of four zones). Three zone lines are read in along the r-axis (I = 3) and three along the z-axis (K = 3). Zone A is the upper, inner cylinder; zone B is the lower, outer cylinder. Find zone numbers A and B (see Figure 13).

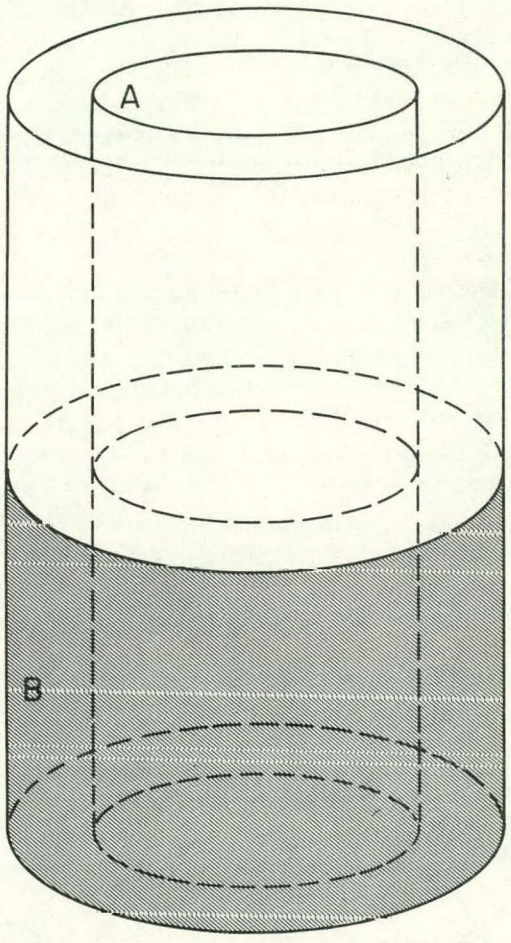


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FIG. 13 ZONE GEOMETRY EXAMPLE.

Solution:

For A: i = 1, k = 2
N = 2(1-1) + 2 = 2 -- Zone no. 2

For B: i = 2, k = 1
N = 2(2-1) + 1 = 3 -- Zone no. 3

The zone description is entirely independent of the region description and does not necessarily use the same boundaries that were used in the region description. Although in many cases, a judicious choice of zones will also allow some of the region boundaries to be zone boundaries; the two boundaries do not have to coincide. If more zone boundaries are required, they can be constructed without regard to existing region boundaries. If a complex region

construction is required, then zone boundaries may intersect region boundaries. In this case, for Input Group 13, that region is merely described as being contained in both zones. There is no restriction to the number of zones occupied by a single region, but a single zone may contain no more than seven regions. The most efficient choice of zone lines would establish only one region within each zone. The region search routine is bypassed, in this case, since once the zone is located the region is simultaneously found. This is not always possible, and if several regions are contained within a single zone, the region search routine must determine which region the particle lies in.

The region search begins as soon as the zone has been found. The region search consists of inspecting each region in the zone one by one until the region number of the location of the particle has been found. To determine if a given point is within a region, the position of the point is tested with respect to each boundary of the region. This is done in the case of plane boundaries by substituting the coordinates of the point into the equation of the boundary and comparing the sign of the number resulting from this substitution with the sign of the boundary for the particular region being tested. If any boundary is encountered where the sign assigned to it for that region does not agree with the sign of the substitution of the point into the equation, then the particle is not in that region. For curved boundaries the radius squared of the point is compared with the radius squared of the inner and outer curve for the region. In the case of a region with only one curved boundary, it is compared with the radius squared of that single curve.

Since the zone search takes less time than the region search, it is advisable, in the construction of the zone overlay, to make as many zones with one region per zone as possible or at least to create zones with only a few regions in them. Also, to save time, the largest regions in the zone should be listed first in the input (see Input Group 13). In the region search, regions are tested in the order they are read in. Within any given zone a particle has more chance of being in a large region than in a smaller one.

After the new region, h', has been found, of the N particles which reach the boundary, $\left\lfloor \frac{I_{h'}}{I_h} N + \xi \right\rfloor$ particles enter the new region, where ξ is a random number and $I_{h'}/I_h$ is the ratio of importance weights of the new and old regions (see Section VII-4).

6. EXTERNAL BOUNDARY REFLECTION

In Input Group 6 (Section VIII-3), boundary numbers which are to be external reflecting or transmitting boundaries must be indicated. Each external boundary is assigned a positive albedo (Input Group 7) by macroscopic group number. The albedo is defined as

$$\beta = \frac{\text{reflected neutron current}}{\text{total incident neutron current}} \qquad 0 \leq \beta < \infty .$$

All internal boundary crossings transmit the complete neutron beam, and the number of particles crossing the boundary are determined by the ratio of the region importance weights. For the external boundaries, the fraction of the neutron beam reflected is β, and the fraction transmitted is $1-\beta$. The same applies the the remaining particles in the beam. If N particles reach the external boundary, $\left\{\beta N + \xi\right\}$ are reflected. $N = \left\{\beta N + \xi\right\}$ are transmitted through the boundary. Those particles transmitted are printed at the end of each trial as the quantity, PLEAK. The number of neutrons those particles represent is given by $\frac{PLEAK}{I_g I_h}$ and tallied as OBSNLEAK. Both of these quantities represent actual discrete particles or neutrons leaking from a particular external boundary into a pre-assigned fictitious region and may be positive or negative depending on the value of the albedo. The continuous beam weights which cross the external boundary are given by $I(1-\beta)$ = PINLEAK, where I is the beam weight at the boundary. This last tally is the one used in computing a multiplication factor (KEFF) for the problem.

After a particle has been advanced to a boundary and it is found that the region on the other side of the boundary is fictitious, and providing the albedo of that external boundary is not zero, then reflecting direction cosines must be computed at the boundary. The equations to find these new cosines depend upon whether the external boundary is a plane or a curve.

6.1 Plane Reflecting Surface

Figure 14 illustrates in two dimensions what will occur in the program in three dimensions. The plane external boundary is read in as $Ax+By+Cz = D$ (see Input Group 5). The Monte Carlo input program will normalize this equation to $ax+by+cz = d$, where

$$a = \frac{A}{\sqrt{A^2+B^2+C^2}} \qquad b = \frac{B}{\sqrt{A^2+B^2+C^2}} \qquad c = \frac{C}{\sqrt{A^2+B^2+C^2}} \qquad d = \frac{D}{\sqrt{A^2+B^2+C^2}} .$$

Let a beam vector, $\vec{S}_1$, be traveling with direction cosines (α,β,γ) toward the plane, $ax+by+cz = d$. Let $\vec{T}$ be a unit vector normal to the plane; $\vec{P}$ be a vector parallel to a straight line drawn from the point on the plane where $\vec{T}$ is perpendicular, to the point where $\vec{S}_1$ strikes the plane; and, extend $\vec{P}$ to the point where the reflected beam vector, $\vec{S}_2$ $(\alpha', \beta', \gamma')$, will meet it.

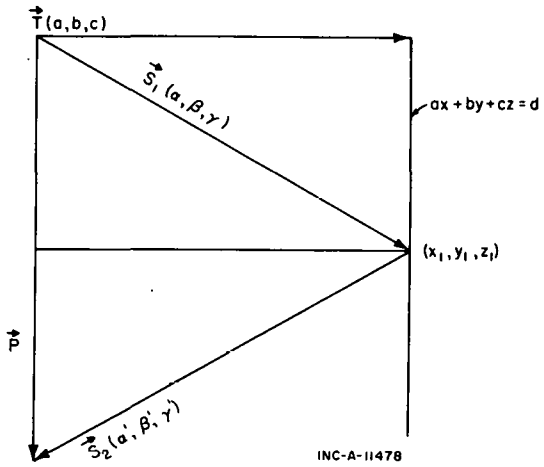


FIG. 14 REFLECTION FROM A PLANE BOUNDARY.

Then,

$$\vec{P} = \vec{S}_1 + \vec{S}_2$$

$$1/2\vec{P} = \vec{S}_1 - \vec{T} \text{ (angle of incidence = angle of reflection)}$$

$$\therefore \vec{S}_2 = \vec{S}_1 - 2\vec{T}$$

$$\text{or} \quad \vec{S}_2(\alpha',\beta',\gamma') = \vec{S}_1(\alpha,\beta,\gamma) - 2\vec{T}(a,b,c) .$$

Taking each component,

$$\alpha' \ |\vec{S}_2| = \alpha|\vec{S}_1| - 2a|\vec{T}|$$

$$\beta' \ |\vec{S}_2| = \beta|\vec{S}_1| - 2b|\vec{T}|$$

$$\gamma' \ |\vec{S}_2| = \gamma|\vec{S}_1| - 2c|\vec{T}| .$$

But, since $|\vec{S}_2| = |\vec{S}_1|$ and $|\vec{T}| = \sqrt{a^2+b^2+c^2} = 1$

$$\alpha' = \alpha - \frac{2a}{|\vec{S}_1|}$$

$$\beta' = \beta - \frac{2b}{|\vec{S}_1|}$$

$$\gamma' = \gamma - \frac{2c}{|\vec{S}_1|}$$

$$\frac{|\vec{T}|}{|\vec{S}_1|} = \cos(\vec{T}, \vec{S}_1) = \frac{\vec{T} \cdot \vec{S}_1}{|\vec{T}|\,|\vec{S}_1|} = \frac{(\alpha a+\beta b+\gamma c)\,|\vec{S}_1|}{|\vec{S}_1|} = \alpha a+\beta b+\gamma c = \frac{1}{|\vec{S}_1|}$$

$$\therefore \alpha' = \alpha - 2a(\alpha a+\beta b+\gamma c)$$

$$\beta' = \beta - 2b(\alpha a+\beta b+\gamma c) \qquad (5)$$

$$\gamma' = \gamma - 2c(\alpha a+\beta b+\gamma c) \quad .$$

6.2 Curved Reflecting Surface

For a curved external boundary, first compute the tangent plane at the point where the beam vector strikes the curve (x_1, y_1, z_1), and then proceed as in the plane boundary case.

Let $f(x,y,z) = 0$ be the equation of the curved boundary. For the particular boundaries allowed in the code, $f(x,y,z) = (x-x_o)^2 + (y-y_o)^2 + K(z-z_o)^2 - R = 0$, where (x_o, y_o, z_o) is the center of the circle, and $K = 0$ for a cylinder and $K = 1$ for a sphere.

The coefficients of the tangent plane will be:

$$A' = \frac{\partial f}{\partial x}\bigg|_{(x_1,y_1,z_1)} = 2(x_1-x_o)$$

$$B' = \frac{\partial f}{\partial y}\bigg|_{(x_1,y_1,z_1)} = 2(y_1-y_o)$$

$$C' = \frac{\partial f}{\partial z}\bigg|_{(x_1,y_1,z_1)} = 2K(z_1-z_o) \quad .$$

For the normalized tangent plane:

$$a = \frac{A'}{\sqrt{A'^2+B'^2+C'^2}} \qquad b = \frac{B'}{\sqrt{A'^2+B'^2+C12}} \qquad C = \frac{C'}{\sqrt{A'^2+B'^2+C'^2}} \quad .$$

But,

$$A'^2+B'^2+C'^2 = 4(x_1-x_o)^2 + 4(y_1-y_o)^2 + 4K^2(z_1-z_o)^2 = 4R^2$$

$$\therefore \quad a = \frac{x_1-x_o}{R}$$

$$b = \frac{y_1-y_o}{R}$$

$$c = \frac{K(z_1-z_o)}{R} \quad .$$

Using these values of a, b, and c the reflected direction cosines are computed from Equation (5).

7. REGION SETS

By the use of "region sets", the flux from several regions may be tallied in a single tally box. Any number of defined regions may be lumped together in this manner to create one flux tally box.

For reasons such as the geometry limitations of the Monte Carlo, actual physical regions of a reactor problem may have to be divided into smaller regions. This is shown in Figure 15 where the whole area outside the shaded circles is to be considered one physical region. Because of the geometry limitations, the large physical region must be divided into two smaller regions (separated by the dotted line). However, both smaller regions may be listed with the same region set number; and during the Monte Carlo run, any flux accumulated in either of the smaller regions will be tallied in the same storage location, thus effectively recreating the actual physical region.

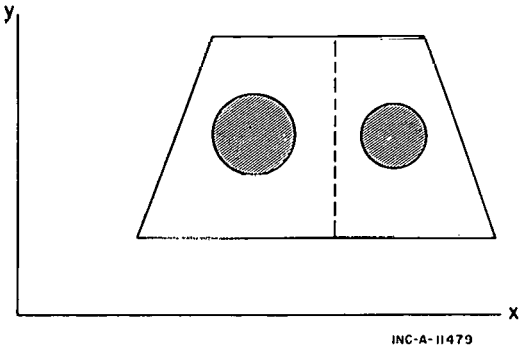


FIG. 15 ILLUSTRATION OF USE OF REGION SETS.

8. RECOMMENDED GEOMETRY CONSTRUCTION PROCEDURE

Before preparing input to the Monte Carlo program, it is suggested that the following general procedure be used when setting up the geometry for a problem. It is best to have the problem well laid out in advance:

(1) Draw two-dimensional diagrams of the problem, all three views if necessary, showing each boundary and region. All diagrams must be in the first quadrant of the x, y, z reference system.

(2) Surround the drawings by a sufficiently large buffer area for fictitious regions; draw in the fictitious regions (everything, including fictitious regions, still in the first quadrant).

(3) Establish the type of zone geometry and its center point. The center point will be at (0,0,0) for parallelepiped zones. For cylindrical zones it will be at the center of the cylindrical system and 0 along the z-axis. For spherical zones it will be the center of the spheres (located somewhere in the first octant).

(4) Number the regions consecutively; ordinary regions beginning with number 1, fictitious regions beginning with 401.

(5) Number the boundaries in any convenient order and make a list of the boundary equations, beginning with 1, and numbering them consecutively.

(6) Make a separate list of external boundary numbers.

(7) Next to each plane boundary in each region, put a + or a - sign to show the relative position of that region to the boundary.

(8) Construct the zone overlay and number the zones.

(9) Write down, by zone number, a list of regions which belong in each zone.

V. RESONANCE ROUTINE

Up to this point, no treatment of resonance cross sections has been given other than mentioning that resonance effects can be present through the cross sections which affect particles through the P_{cmg} and Σ_T. Two versions of PMC exist: one version treats nonresonant cross sections; the second version permits the calculation of resonant cross sections directly in specified energy ranges.

In previous sections the discussion was geared to the nonresonant version of the program where it was assumed that all P_{cmg}'s and λ's for each energy group and material were supplied in the input. This implies that all cross sections were calculated exterior to PMC. Therefore, each time a particle has a collision, resulting in an energy change, or enters a new material, a new set of P_{cmg}'s and λ is available by proper indexing.

In the resonant version of the program, a resolved and unresolved resonance range is specified, and those energy groups which lie either completely or partially within the resonance range have a combination of single-level Breit-Wigner resonance parameters and smooth microscopic cross sections stored. One set of resonance parameters is input for each resolved resonance in the resolved range for each resonant isotope in a material and one set is given for the unresolved range. No resonance is permitted in the thermal group, however.

Whenever a new set of coordinates is taken from the storage bank, a test is made to determine if the coordinate is in a region of a resonant material. If it is, the energy is tested for each resonant isotope in the material to see whether it lies in the resolved or unresolved range. Microscopic cross sections are also input for nonresonant isotopes in resonant materials for those groups which lie in the resonance range. For each resonant isotope, the Doppler broadened capture, fission, and scattering cross sections are computed from the Breit-Wigner single-level equations using each of the resonances lying on either side of the particle energy, E. The smooth cross sections of both resonant and nonresonant isotopes are then added to the resonant cross sections, weighted by their atom densities. The P_{cmg}'s and λ are then formed at that energy, and the particle is ready to proceed to collision. This process is repeated every time the particles and beams change energy or enter another resonant material.

The Doppler broadened capture, fission, and scattering cross sections for a resonance at energy E_0 are given for a neutron at energy, E, velocity, v, by

$$\sigma_C(E) = \frac{4\pi\hbar^2}{2m} \frac{g}{\sqrt{E_0 E}} \left(\frac{A+1}{A}\right) \frac{\Gamma_n \Gamma_\gamma}{\Gamma^2} \psi(X,\theta) + \frac{\sigma_C^*}{v} \tag{6}$$

$$\sigma_F(E) = \sigma_C(E) \frac{\Gamma_F}{\Gamma_\gamma} + \frac{\sigma_F^*}{v} \tag{7}$$

$$\sigma_S(E) = \frac{4\pi\hbar^2}{2m} \frac{g}{E_0} \left(\frac{A+1}{A}\right) \frac{\Gamma_n^2}{\Gamma^2} \psi(X,\theta) + 2g \sqrt{\frac{4\pi\hbar^2}{2m} \frac{\sigma_p}{E_0} \frac{\Gamma_n}{\Gamma}} \chi(X,\theta) + \sigma_p \tag{8}$$

where

$$\Gamma = \Gamma_n + \Gamma_\gamma + \Gamma_F$$

$$g = \frac{1}{2}\left(\frac{2J+1}{2I+1}\right) \qquad J = I \pm \frac{1}{2} \ .$$

The Doppler line functions $\psi(X,\theta)$ and $\chi(X,\theta)$ are given by

$$\psi(X,\theta) = \frac{1}{2\sqrt{\pi\theta}} \int_{-\infty}^{\infty} \frac{e^{-(X-Y)^2/4\theta}}{1+Y^2}\, dY$$

$$\chi(X,\theta) = \frac{1}{2\sqrt{\pi\theta}} \int_{-\infty}^{\infty} \frac{Y\, e^{-(X-Y)^2/4\theta}}{1+Y^2}\, dY$$

where

$$\theta = \frac{4EKT}{A\Gamma^2}$$

$$X = \frac{E-E_0}{\Gamma/2} \ .$$

K is Boltzmann's constant and the temperature T is input in degrees Kelvin. With the substitution of $u = \frac{Y-X}{2\sqrt{\theta}}$, ψ and χ may be transformed into

$$\Psi(X,\Theta) = \frac{1}{\sqrt{\pi}} \int_{-\infty}^{\infty} \frac{e^{-u^2}\, du}{1 + (2\sqrt{\Theta}\, u + X)^2}$$

$$\chi(X,\Theta) = \frac{1}{\sqrt{\pi}} \int_{-\infty}^{\infty} \frac{(2\sqrt{\Theta}\, u + X)\, e^{-u^2}\, du}{1 + (2\sqrt{\Theta}\, u + X)^2} \ .$$

These equations are now in a form adaptable to the Gaussian-Hermite quadrature formula which gives an approximation of the integral

$$\int_{-\infty}^{\infty} e^{-x^2} f(x)\, dx \qquad \text{by} \qquad \sum_{i=1}^{N} w_i f(z_i)$$

where the z_i are zeros of the Hermite polynomial of degree N, and w_i are their corresponding weights. The number of zeros, N, taken will be either 10 or 20, depending upon the size of X and Θ.

For some ranges of X and Θ the Gaussian-Hermite approximation is not sufficiently accurate and a 51 or 81 point trapezoidal integration is made. This approximation is given by

$$\Delta u \left[\frac{f_1+f_{N+1}}{2} + \sum_{n=2}^{N} f_n\right] \qquad (N = 50 \text{ or } 80)$$

where

$$f_n = \frac{e^{-u_n^2}}{1 + (2\sqrt{\Theta}\, u_n + X)^2} \quad \text{for } \psi$$

$$f_n = \frac{(2\sqrt{\Theta}\, u_n + X)\, e^{-u_n^2}}{1 + (2\sqrt{\Theta}\, u_n + X)^2} \quad \text{for } \chi .$$

The u_n are the evaluation points along the u-axis, and the $e^{-u_n^2}$ are their corresponding weights. Numerical integration takes place between u = -3.4 and u = 3.4, where the value of e^{-u^2} is approximately 10^{-5}. Δu is the distance between evaluation points along the u-axis.

In Equation (8), the interference term is summed for the two adjacent resonances before the potential scattering cross section σ_P for that isotope is added. This is true also in Equations (6) and (7) where the 1/v cross section variations, σ^*_C/v and σ^*_F/v, are not added until contributions from the two adjacent resonances are computed. The two terms σ^*_C and σ^*_F are read in as MU13 and MU14 in the resonance parameter tables immediately following the unresolved range parameters (see Input Group 28).

If E lies in the unresolved range of one or all of the resonant isotopes, the resonance cross sections can be treated within the framework of the resolved resonance routines, using Equations (6), (7), and (8) by the following method:

(1) Select the location of two resonances in the unresolved range near the energy of the particle being processed by means of a pseudo-random decision.

(2) Select values of Γ_n^o from a chi-square distribution with one degree of freedom, and values for Γ_F from a chi-square distribution with two degrees of freedom.

The position of the two resonances, E_1 and E_2, adjacent to E (the energy of the particle in the lab system) is given by

$$E_1 = \left[\frac{E}{D}\right] D$$

$$E_2 = E_1 + D$$

where D is the average level spacing for the isotope in the unresolved range. D is input with the unresolved resonance parameters in Input Group 28. E_1 will thus lie below E on the lab energy scale, depending on the value E for the neutron, in units of D. The second resonance at E_2 will be above E on the lab energy scale. This scheme fixes the location of resonances in the unresolved range and each is a distance D from the next.

The Porter-Thomas distribution of the reduced neutron width [4] relative to its mean value is a chi-square distribution with one degree of freedom. The distribution of fission widths is not as well defined. Complete treatment of fission requires a multilevel formalism and fission width distributions of many degrees of freedom. The data are, however, rather rough. In any event, the single-level formula for fission is an approximation which does not require great accuracy for fission width distributions. Since it was expected that the RBU cross section library would be used in the resonance calculations, the RBU treatment of the unresolved resonance range was adopted.

Define u and X as

$$u^2 = \frac{\Gamma_n^o}{\langle \Gamma_n \rangle}, \quad X^2 = \frac{\Gamma_F}{\langle \Gamma_F \rangle} \qquad \text{where} \quad \Gamma_n^o = \frac{\Gamma_n}{\sqrt{E_o}} .$$

The neutron width probability density function is a chi-square distribution with one degree of freedom and is given as

$$\frac{dP(u^2)}{d(u^2)}\, d(u^2) = \frac{e^{-u^2/2}}{u\sqrt{2\pi}}\, d(u^2) . \tag{9}$$

As such, the distribution in u will be normal.

$$\frac{dP(u)}{du}\, du = \sqrt{\frac{2}{\pi}}\, e^{-u^2/2}\, du \tag{10}$$

$$\therefore P(u) = \sqrt{\frac{2}{\pi}} \int_0^u e^{-u^2/2} du = \xi \tag{11}$$

P(u) lies between 0.0 and 1.0 and is set equal to a random number ξ. P(u) is interpreted as the cumulative probability.

The physics of fission requires "at least" two degrees of freedom for the distribution of fission widths. Since the single-level formalism is being used, this minimal requirement on the distribution of fission widths will be adopted.

The chi-square distribution with two degrees of freedom for the variable X^2 is

$$\frac{dP(X^2)}{d(X^2)}\, d(X^2) = \frac{e^{-X^2/2}}{2}\, d(X^2)$$

$$\therefore\ P(X^2) = \int_0^{X^2} \frac{e^{-X^2/2}}{2}\, d(X^2) = 1 - e^{-X^2/2} \quad . \tag{12}$$

If $P(X^2)$ is set equal to a random number as in Equation (11), the result is

$$= \ell n(1-\xi) - X^2/2 \quad .$$

Since $1-\xi$ is also a random number,

$$-2\ell n\xi = X^2 = \frac{\Gamma_F}{\langle \Gamma_F \rangle} \quad .$$

Neither Equation (9) nor (10) permits a closed solution like Equation (12) for the random variable u^2 or u. The following method was used by RBU to select u from the distribution in Equation (10).

$$\begin{aligned} u &= -\ell n\xi_1 \\ v &= -\ell n\xi_2 . \end{aligned} \tag{13}$$

Since u has been selected from the improper distribution function [Equation (13)], require that u be such that

$$(u-1)^2 \le 2v \quad . \tag{14}$$

If the requirement is not satisfied, reject u and v until they are acceptable. Values of u so determined will be from the proper distribution of the joint probability density function for u and v. The proof of this requires that Equation (14) be used in the joint probability density function to determine what the probability density of u will be when Equation (14) is applied. The conditional probability density of u will then be

$$\frac{du \int_0^\infty P\left[\frac{(u-1)^2}{2} \le v \,|v\right]\, dv}{\int_0^\infty \int_0^\infty P\left[\frac{(u-1)^2}{2} \le v | v\right]\, dudv} = \frac{e^{-u}du \int_{\frac{(u-1)^2}{2}}^\infty e^{-v}dv}{\int_0^\infty e^{-u}du \int_{\frac{(u-1)^2}{2}}^\infty e^{-v}dv}$$

$$= \frac{e^{-u}\, e^{-(u-1)^2/2}\, du}{\int_0^\infty e^{-u}\, e^{-(u-1)^2/2} du} = \frac{e^{-(u^2+1)/2}\, du}{\int_0^\infty e^{-(u^2+1)/2}\, du}$$

$$= \frac{e^{-1/2}\, e^{-u^2/2} du}{e^{-1/2}\sqrt{\frac{\pi}{2}}} = \sqrt{\frac{2}{\pi}}\; e^{-u^2/2} du \quad .$$

Therefore, Equation (14) yields the required probability density function for u, and the restriction $(u-1)^2/2 \leq v < \infty$ gives acceptable values of u and v.

Γ_n and Γ_F are then the following:

$$\Gamma_n = \sqrt{E_o}\, \langle \Gamma_n \rangle\; u^2$$

$$\Gamma_F = \langle \Gamma_E \rangle\; v \quad .$$

Two additional parameters, D* and ℓ*, are included in the RBU cross section library for unresolved resonance parameters. They are coefficients which give rough approximations to the variation of the level density with energy and for the presence of resonances with angular momentum greater than ℓ=0. If values other than zero for the parameters are specified in the input, then the factor

$$(1+1.9132D^*E)(1+\sqrt{1.9132E}\,\ell^*)^2 \qquad (15)$$

multiplies g in Equations (6) and (8) for the unresolved calculation only. If D* and ℓ* are zero, the equations are evaluated the same as in the resolved resonance case.

For each resonant isotope in the material, the resonant cross sections $\sigma C(E)$, $\sigma S(E)$, $\sigma F(E)$, and $\nu\sigma F(E)$ calculated from Equations (6), (7), and (8) will have added to them the smooth cross sections for that energy group. The result is multiplied by the atom density of the isotope to obtain macroscopic cross sections. These cross sections will be multiplied by the beam flux and stored as the resonant reaction rate for the isotope in the particular macroscopic group in which E lies. The resonant reaction rate for capture, nu sigma fission, and scattering accumulated throughout a statistical trial will be listed following the group fluxes for each material, isotope, and macroscopic energy group. At the conclusion of the problem, statistical confidence limits are assigned to each reaction rate total, as described in Section VII-2.

Once the macroscopic cross sections have been computed, the mean free path for this material m, group g, may be found from

$$\lambda_{mg} = \frac{1}{\Sigma_T} = (\Sigma_C + \Sigma_F + \Sigma_S)^{-1} \quad .$$

After the collision emission numbers (P_{cmg}'s) have been calculated as described in Section III-2, the program continues as in the nonresonant version until another resonant material is entered or another set of coordinates having resonant properties is picked up from the storage bank. (See Figure 16.)

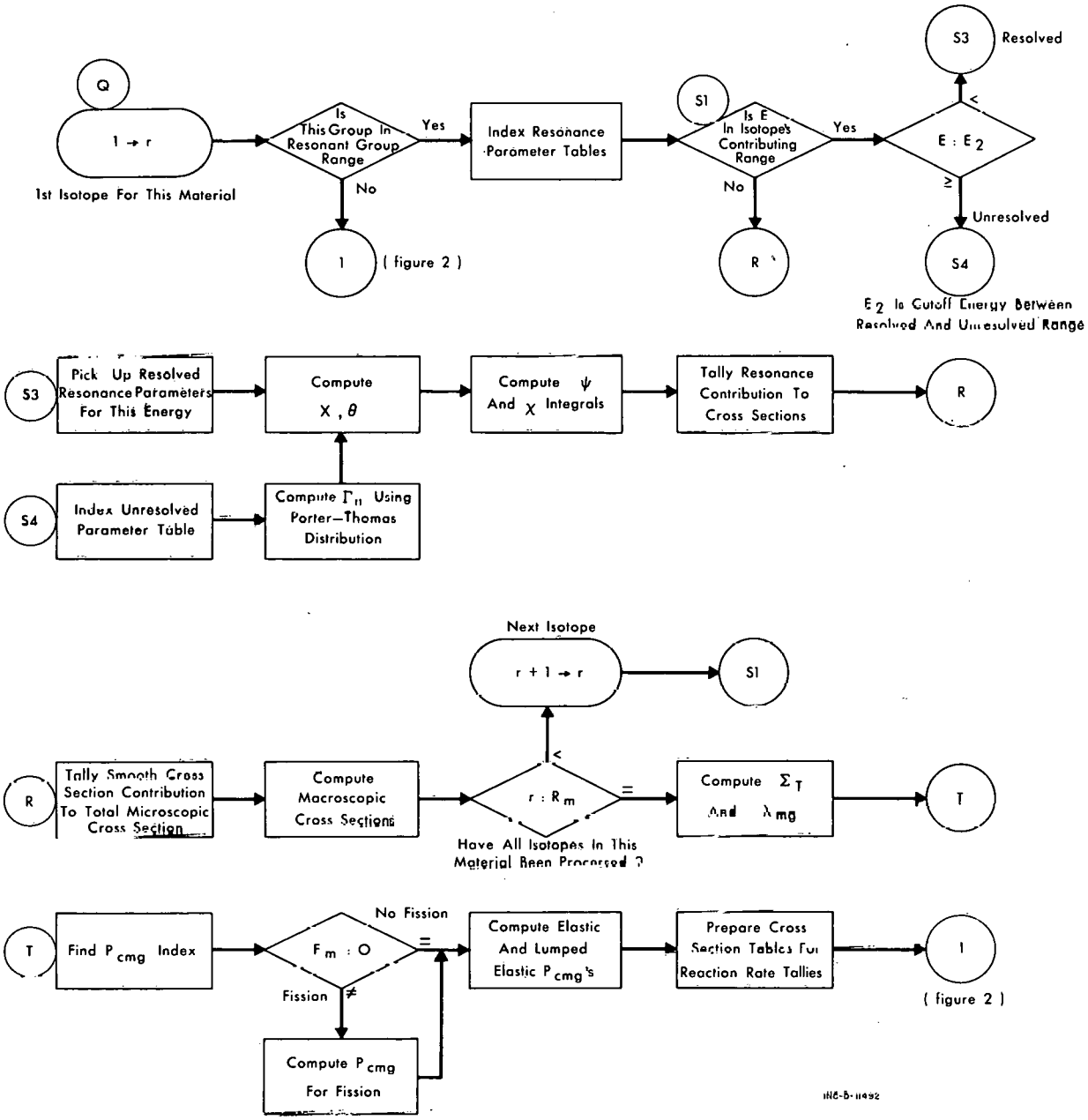


FIG. 16 OUTLINE OF RESONANCE ROUTINE.

VI. SOURCE GENERATORS

Three types of random source generators for Initial Value (IV) particles are presently available. They are an integral part of the Monte Carlo input program and include: beam source; uniform source, regionwise; and uniform isotropic point and disk source.

All types of source generators will automatically exclude those points which are generated in a fictitious region. Only those points in regions numbered through 400 will be accepted. In addition, provision is made (see Input Groups 25 and 26) to label any regions in the volume under study as "exempt" regions. Regions so labeled in the input will also have no source particles originating in them.

The source generators store the IV particles on magnetic tape unit S.SU05 in blocks of 45 particles. For this reason, in Input Group 1 the total number of IV particles is required to be a multiple of 45. Each particle consists of 12 position coordinates with the following information:

N -- number of particles in the beam which will be set to $I_g I_h$, the product of region and group importance weights

x,y,z -- reference frame coordinates

α,β,γ -- random direction cosines

PTIMF -- particle time which is entered as 0 for all source particles

ROOTE -- square root of the starting energy (read in with Input Group 25) which remains constant for all source particles for a given problem

g -- microscopic energy group

h -- region in which the source particle is located

m -- material in which the source particle is located.

The source generators use the same zone and region search routine that was described in Section IV-5 to determine the region for a source particle. If the region cannot be located after a suitable number of attempts due to improper setup of the geometry, a program stop will occur, and an error message will be printed. Some of the reasons for this type of error include: the fictitious region buffer zone may be too small; an incorrect sign may have been assigned to the plane boundaries in a region (or a region defined with the wrong boundaries); some regions may be forgotten and not included in a zone (or, in general, improper zone overlay).

The source generator input information (Input Groups 31, 32, 33) is printed after the rest of the input listing and is followed in turn by sample IV particles. Every 45th IV particle is printed on the output listing.

1. BEAM SOURCE

The beam source generates points uniformly distributed on a circle of specified radius and center located on a plane within the reactor framework. All source planes must be of the form Ax, By, or Cz = constant with the corresponding coefficients either 0.0 or ± 1.0. The neutron direction for positive A, B, or C is outward in the positive direction normal to the plane (away from a reference axis). For negative A, B, or C, the direction is inward normal to the plane (toward a reference axis). If the source plane is to lie exactly on an external boundary (on a line between real and fictitious regions), the center of the circle in the source plane should be changed by a small amount sufficient to ensure that the source plane is physically inside a real region.

2. UNIFORM SOURCE, REGIONWISE

The uniform source generator will generate points uniformly over the whole reactor volume under study or any portion of the volume as specified by a maximum and minimum point along each axis of a problem's zone geometry. These maximum and minimum points, which are read in, are measured along an axis with respect to the geometry center point and need not fall on a zone line or region boundary line. As an example, take the following annulus in two dimensions (Figure 17). Assume that the user only wants sources generated in the outer cylinder. In the input for the source generator (see Input Group 33), he should list r_2 as the maximum point along the radial axis and r_1 as the minimum point, and similar limits along the z-axis. Thus, no sources will be generated in the inner cylinder but will be uniformly distributed throughout the outer cylinder. Direction cosines will be generated randomly in an isotropic distribution (direction cosines between -1 and 1).

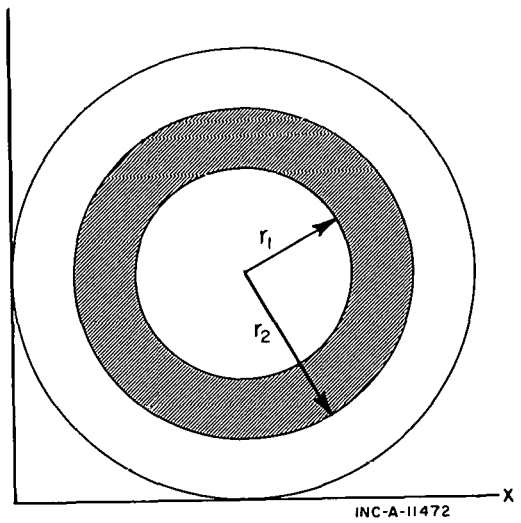


FIG. 17 SOURCE GENERATOR 2 EXAMPLE.

3. UNIFORM ISOTROPIC POINT AND DISK SOURCE

The third generator, a point and disk source generator, was programmed for use in determining the efficiency of gamma ray detectors. Uniform sources are generated over a disk of radius, r. To use this source generator the reactor problem must be oriented such that the disk is perpendicular to the z-axis, but the center of the disk may be located at any point within a real region. (Sources cannot be generated within fictitious regions). The point source generator is obtained by setting the radius of the disk to zero (RAD = 0 in Input Group 33). In either case, $\bar{x}$, $\bar{y}$, and $\bar{z}$ are the coordinates of the center of the disk or

point relative to the fixed reference frame. Provision is made to generate direction cosines such that the source beams lie within a cone whose axis is normal to the disk and which has a polar angle θ_0 relative to the z-axis, given by

$$\theta_0 = \tan^{-1} \frac{W}{D}$$

where D is the perpendicular distance from the disk to the area being irradiated by the source,

W is the radius of a circular area or one half the diagonal of a rectangular area to be irradiated.

Then, γ is given by $\gamma = \cos\theta_0 + (1-\cos\theta_0)\,\xi$, where ξ is a random number.

One may use this source without irradiating any specific area. In this case, W and D are merely parameters used in determining θ_0, the polar angle of the cone within which to generate an isotropic source. Since no sources are emitted from the backward face of the disk, the largest cone permitted is a hemisphere, where θ_0 = 90 degrees. To obtain this case, specify W as very large relative to D. Figure 18 illustrates the geometric configuration employed in using the disk or point source generator.

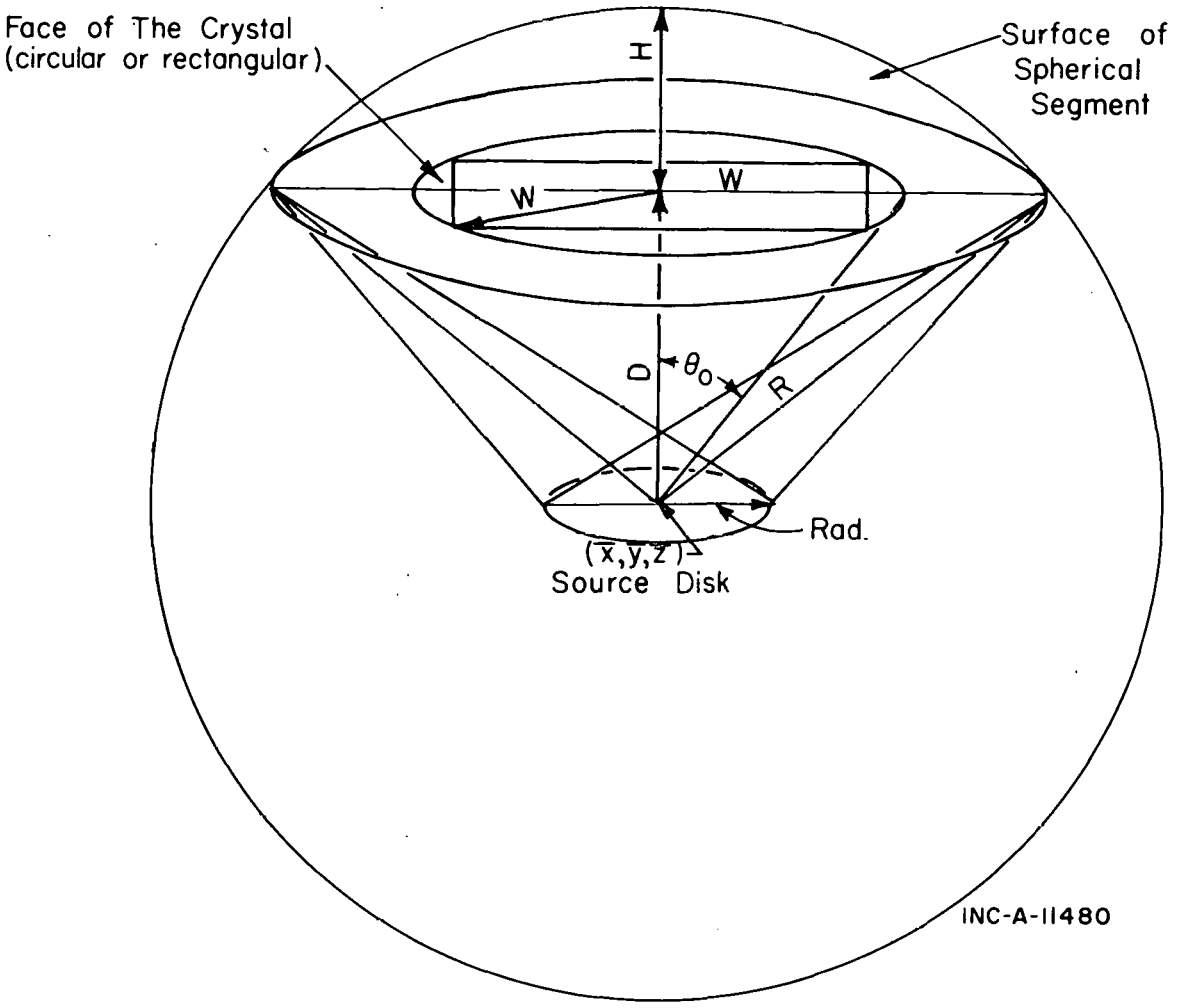


FIG. 10 SOURCE GENERATOR 3 EXAMPLE.

VII. MISCELLANEOUS INFORMATION

1. METHODS OF TERMINATING A PROBLEM

A Monte Carlo problem begins when the first 45 IV coordinates are read into the storage bank from the IV tape. The initial IV coordinate generates a succession of new coordinates depending on the results of successive collisions which occur in the system. If these daughters are used up due to leakage or unsuccessful collisions because the system is not fissionable or subcritical, the next IV coordinate will be processed in turn until all 45 have been used. Then 45 new IV coordinates will be read in and processed until all coordinates on the IV tape have been exhausted. If this occurs, the problem is terminated regardless of which method has been prescribed for termination. It is entirely possible that only one or a very few IV coordinates are necessary to generate enough daughters to continue the problem ad infinitum if the system is fissionable and the multiplication is greater than one. Because of the wide range of problems envisioned to be run with the program, and since the collision process results from random sampling, it is impossible to expect a problem requester to guess the number of sources which will terminate a problem in a given time. Also, the fluxes generated in the Monte Carlo random sampling process contain an inherent statistical fluctuation, and it is necessary in most cases to assign a confidence limit to any given result.

With these considerations in mind, three methods were chosen for terminating a problem. Two of these methods will allow the user to fix a statistical confidence limit to a result and at the same time permit him to estimate the length of running time required within narrow limits. The assignment of "CTEND" in Input Group 2 prescribes the method of termination.

Ordinarily, a statistical confidence limit is desired on the fluxes by group and region which have been generated during the entire problem. Thus, the accumulated flux in each tally box is divided into statistical blocks. Each statistical block of data is called a trial and represents an independent statistical sample of the data acquired in any given tally box. The length of the problem is determined by specifying the number of trials to be processed. At the conclusion of each trial, the flux tallied for each region set and group is written on the output tape, as well as the other individual tally boxes which record various events occurring during the running of the problem. The next trial is then begun -- until all trials are finished.

If the problem is stopped (perhaps unexpectedly) before the last trial is completed, the output will be for all trials up to and including the last completed one. The statistical analysis of fluxes will be based on this completed number of trials rather than including the incompleted trial. In this way a requester may set up an upper limit on time that a problem may be run, through the use of the $TIME card, without losing all data if the computer or its operator stops the problem before the last trial is completed. Using the merge option of the Monte Carlo output program, the problem may then be continued at a later time and coalesced with the results of the first set of trials which was completed. In this way the requester may periodically receive an evaluation of the confidence limits and continue the problem until the desired accuracy is achieved. Any number of trials may be requested on the second and succeeding runs.

1.1 CTEND = 1

If CTEND = 1, no statistical confidence limits are properly determined or even possible in this case. Collision daughters are followed in time until the progeny of any IV coordinate acquires a total flight time in microseconds equal to the input census time. At this point, that particular daughter is stored on the census tape and the immediately preceeding entry in the storage bank is processed with its progeny to census time and stored on the census tape. If the system is nonmultiplying, the progeny of the initial IV coordinate will soon be exhausted and the next IV coordinate is processed in an analogous manner. This is repeated until all IV coordinates have been processed to census time. The problem is then terminated, and all tally boxes are printed on the output listing. CTEND = 1 will not normally be used to end problems.

1.2 CTEND = 2

In general, it is very difficult to estimate the number of IV sources which will allow a problem to run a specified length of time when multiplication is present in a problem. Even in some nonmultiplying problems, it is next to impossible. If no termination by census (census time very large) is prescribed, it is possible that the problem would never terminate if left to its own devices. In the case CTEND = 2, a definite number of trials is prescribed in the input and the time in seconds for the first trial. During the first trial, in addition to the flux tallies being made for each region set and each group, a separate tally is made for the total flux accumulated in all regions and energy groups. When the first trial is completed, based on the length of time requested in the input, the next and succeeding trials are terminated when the total flux accumulated during these trials is equivalent to that for the first trial. In this way, the integral neutron flux for each trial in the problem is equal to that for every other trial even though variations exist within the individual tally boxes. Since the scalar neutron flux constitutes the basic statistical sample in this program, the statistical samples from trial to trial are of equal size in this sense. However, the running time associated with each trial varies from trial to trial. Unless the problem is very large, this running time will not vary substantially, and experience has demonstrated that it will usually lie within a plus or minus ten percent of the first trial length for most problems. When the next trial is begun, the last particle being processed when trial termination occurred is then picked up and continued in an uninterrupted manner as the first event at the beginning of the next trial.

1.3 CTEND = 3

If CTEND = 3, the length of the trials is determined by specifying the number of IV sources to be processed to termination during each trial. When this specified number of sources has been processed, the end-of-trial edits are made, and the contents of each tally box are written on the output tape for that trial for analysis at the end of the problem. If the geometry of the system is not too complex (four or five regions, ten boundaries), about 1000 collisions are processed per minute, and a little experience enables one to estimate the running time beforehand. In multigroup, nonmultipling problems, the source energy is specified to lie within the first group, and particles are followed into the thermal group where they will remain since fission is the only process by which the daughter of a thermal collision may enter a fast group. Unless a relatively

high percentage of thermal collisions result in unsuccessful collisions (absorption), it will be difficult to estimate the running time for a fixed number of IV coordinates, and CTEND = 2 is recommended for cases involving low absorbing water-moderated systems.

In either CTEND = 2 or 3, a census time may be prescribed, in which case coordinates are terminated at census and the problem is continued with the next daughter from the storage bank. In this way the trials are all processed for a definite particle flight time, but in no other way do they differ. Of course, the chains of progeny generated over the length of a trial will differ depending on the value of the census time. It is possible that with a very long census time only one parent IV coordinate will be required to generate the progeny which will survive until the end of the problem. Whereas, for a very short census time, many chains of progeny will be required since each is periodically being terminated as it reaches census. The statistics for these two cases could be quite different even though both are representative of the same physical problem.

2. STATISTICAL ANALYSIS OF FLUXES

In order to obtain a confidence limit on any bulk macroscopic parameter computed during the Monte Carlo process, it is necessary to divide the computation into a number of independent statistical samples which are accumulated during the solution run. Thus, the problem is divided into trials as explained above. For most reactor design calculations, the basic quantity from which useful macroscopic design information can be obtained is the scalar neutron flux, and this is the quantity upon which the statistical analysis is based in this program. Numerous authors have pointed out over the years how, in the Monte Carlo process as applied to neutron transport, the present condition of any progeny is dependent to some extent on the past histroy of the parent collisions which produced them. Therefore, the Monte Carlo process does not strictly satisfy the broad conjectures associated with the central limit theorem of statistics. In this program, we have made no attempt to justify these conjectures relative to the scalar flux estimator being used. Rather, we have assumed that a Student's t-distribution analysis of the trial fluxes will be indicative of the confidence limit which can be assigned to the estimator. In all the reactor test problems which have been run with the program where it was possible to obtain a solution with another method, the alternate solution method has produced results which have been within the confidence limit of the Monte Carlo result. This leads us to believe that statistics based on the Student's t-distribution may be valid for this estimator in spite of the associated uncertainties, relative to the central limit theorem.

The percent error accompanying the total trial flux for each tally box on the output listing is the relative percent error for this tally, based on a Student's t-distribution at 95 percent confidence. The variance for each tally box is given by

$$S_{gh}^2 = \frac{\sum_{i=1}^{N} \left[(\phi_{gh})_i - \bar{\phi}_{gh} \right]^2}{N-1}, \quad (g = 1, \ldots, G;\ h = 1, \ldots, HS)$$

where N is the total number of trials which have been completed, i is the trial index, and $\bar{\phi}_{gh}$ is the average flux in that tally box,

$$\bar{\phi}_{gh} = \frac{\sum_{i=1}^{N} (\phi_{gh})_i}{N} .$$

The relative percent error, per tally box, is

$$(\% \text{ error})_{gh} = \frac{t_{(N-1)} S_{gh}}{\sqrt{N}} \cdot \frac{100N}{\sum_{i=1}^{N} (\phi_{gh})_i} = \frac{100 t_{(N-1)} S_{gh}}{\sqrt{N} \bar{\phi}_{gh}}$$

where $t_{(N-1)}$ is the value of the Student's t-distribution at 95 percent confidence for N-1 degrees of freedom. To obtain the results in terms of any other percent of confidence, multiply the ratio of the new t value to the 95 percent t value times the percent error at 95 percent confidence: ie,

$$\frac{t_{(N-1)} \text{ at new confidence}}{t_{(N-1)} \text{ at } 95\%} \cdot \text{percent error at 95 percent confidence.}$$

If the merge option is used to merge two sets of trials, the statistical analysis is based on both sets of trials combined into one set.

3. EQUILIBRATION TIME

The source generators which were described in Section VI are not versatile enough to satisfy the requirements of every problem. It may be desired to allow the source to come to equilibrium in the problem before any tallies are made. An equilibration time specified in the input serves this purpose. The time coordinate of all source particles is initially set to zero. If the equilibration time is nonzero, beams and particles are processed in exactly the same manner as they would be otherwise, except that no flux tallies are made until the time coordinate of each particle has reached the equilibration time. Beyond this point all tallies are made for that particle and its progeny in a normal manner. Equilibration time must be used with great caution if problem termination is chosen with the option CTEND =2. In this case it is possible that no flux tallies would be made at all during the first trial if a long equilibration time is requested. Subsequent trials would be terminated without any tallies. In general, equilibration time cannot be used with this option if many source coordinates are required. The equilibration period may be avoided entirely by setting it equal to zero in the input data.

4. IMPORTANCE SPLITTING

In Input Groups 16 and 17 groupwise and regionwise importance weights are requested which determine the relative neutron weights of particles as they

cross region and energy boundaries. Initially, each neutron beam has a weight of one neutron. If the source coordinate appears in group g, region h, which have been assigned importance weights I_g and I_h, respectively, then the number of particles assigned to that coordinate will be I_gI_h and each particle will have a neutron weight of $1/I_gI_h$ neutrons. If a region boundary is crossed by N particles remaining in the beam where a different importance weight has been assigned in the adjacent region, then each particle crossing will be split into N' particles,

$$N' = \left[\frac{I_{h'}}{I_h} + \xi \right]$$

where ξ is a random number and $I_{h'}$ is the importance weight of the new region being entered. The neutron weight of each new particle in region h' will be $1/I_gI_{h'}$; but since the number of particles is now approximately $\frac{I_{h'}}{I_h}$, the total number of neutrons is conserved within the limits of the greatest integer function. If $I_{h'} > I_h$, the number of particles will increase, but their neutron weight will decrease. If $I_{h'} < I_h$, the number of particles will decrease, but their neutron weight will increase. In both cases the total number of neutrons crossing the boundary is approximately conserved. The same general rules apply when crossing energy boundaries. In either case the weight of the beam crossing the boundary from which the flux tally is made is not affected. But the new beams originating from the split particles which have gone to collision will receive new weights accordingly, given by $N/I_gI_{h'}$, as they are processed from the storage bank. No importance splitting occurs if all importance weights are entered as 1.0. If a large number of regions and groups are employed, all weights are set equal to 1.0 if the single number -1 is entered in Input Groups 16 and 17.

Importance splitting has the effect of reducing the variance of the flux in regions of high importance weights and a corresponding increase of variance in regions of low weights. This is because more time is spent in regions (or groups) with high weights. If strong coupling exists between regions of high and low importance weights in a given problem, the effect of splitting, even though it reduces the variance in one region, does not necessarily tend to improve the accuracy of the answer since the variance in low importance regions is increased. One benefits from this process only when low importance weights are assigned to regions of truly low physical importance in the problem. If any doubt exists in a given problem as to the assignment of weights, it is better not to use importance splitting. No general rules for this can be stated at this time, and only experience will indicate when the process is beneficial.

5. MULTIPLICATION FACTOR

The program has the capability of computing a multiplication factor. Cross sections Σ_a and $\nu\Sigma_F$ must be read in to the output program for each region set and group. The definition of the multiplication is based on the ratio of second generation neutrons to first generation neutrons, where the former are computed from the expected value of fission neutrons which would result from the scalar neutron flux computed to the end of each trial. The flux estimator is used because it has less variance associated with it compared to other

tallies made. At the end of each trial, the fission neutron reaction rate is given by

$$\sum_{gh} \phi_{gh} \nu\Sigma_{Fgh}$$

where ϕ_{gh} is the accumulated flux in the gh tally box at the end of each trial. The first generation processes are defined as all those which result in the termination of beams and amount to absorption or leakage processes. The absorption and leakage accumulated at the end of a given trial are given by

$$\sum_{gh} \phi_{gh} \Sigma a_{gh} + \sum_{j} \text{PINLEAK}_j$$

where PINLEAK_j is the tally of all partial beam weights which have been transmitted through the jth external boundary. This leakage tally will have less variance than the discrete leakage tallies which are also made. The multiplication factor is then defined as

$$K(\text{eff}) = \frac{\sum_{gh} \phi_{gh} \nu\Sigma_{Fgh}}{\sum_{gh} \phi_{gh} \Sigma a_{gh} + \sum_{j} \text{PINLEAK}_j},$$

K(eff) is printed for the total accumulated flux and leakage at the end of each trial, and the final value is that for the last trial. No confidence limits are computed for K(eff) directly but may be determined from those assigned to the fluxes. It is not generally recommended that an attempt be made to calculate multiplication factors for water-moderated thermal systems unless they are highly poisoned. If a system is a large number of mean free paths in size and at the same time permits a large proportion of scattering events at the principal fission energies, it will require a great amount of machine time to determine an eigenvalue within acceptable confidence limits. Usually the cases which are not favorable to efficient Monte Carlo solutions are those in which diffusion theory may yield acceptable answers.

VIII. MONTE CARLO INPUT AND OUTPUT PROGRAMS

PMC is actually made up of three programs. It consists of the pre-Monte Carlo Input Program, the main Monte Carlo program, and the Monte Carlo Output program. Communication between the separate programs is done by magnetic tapes. Once one program of PMC has been executed, it will never again be used during that problem run. The function of the pre-Monte Carlo Input program is to edit the user's data, to generate IV source particles, and to print the input data. From this program two data tapes are prepared for the main program. These two data tapes are the only source of input for the main program. The main Monte Carlo program, in turn, writes its results, consisting of the various tallies made in the program, on a third tape. The Monte Carlo Output program uses for input this third tape, as well as information read in from cards, in order to do the final statistical analysis and printing of the results. By eliminating the standard method of input and output to the main Monte Carlo program, a maximum of core storage is made available for use by that program.

Two versions of PMC are available: one which calculates resonance contributions to cross sections, and one which does not. The input for both versions is identical up to the point where the resonance version requires additional information, as indicated in the input specifications given in the following sections.

1. MONTE CARLO INPUT PROGRAM -- GENERAL INFORMATION

The pre-Monte Carlo Input program consists of two parts: the input edit and the source generators. The source generators have been discussed in Section VI. The input edit takes numbers from a data card, passes or rejects that data, stores the information in the core, and goes back to process the next card. When all the data cards have been read, and providing there are no program-discovered input errors, the stored information for the problem is placed on system utility tape, S.SU04, called the Monte Carlo input tape. If there are any program-discovered input errors, a program stop occurs and nothing will be written on tape. If an error is found, the card which caused the error and an error warning message will be printed out. The data following the error will continue to be checked until a total of 15 errors has been encountered, at which time the program will stop and no further data will be checked. Certain types of errors, such as not having enough data in an input group, will cause the program to assume succeeding cards have errors. When an error warning occurs, always check the first error before any others as it may be the cause of all succeeding error messages. Some of the possible errors include: exceeding the size limit of a quantity, inserting integers when the number should be floating point or vice versa, having too few or too many numbers in an input group, and leaving out an input group entirely.

Some of the numbers appearing in the printing of the Monte Carlo Input do not appear in the same form or order that they were read in. The boundary equation parameters, for instance, are normalized, radii are squared, and equations are changed into the general form

$$A(x-x_o) + B(y-y_o) + C(z-z_o) + D = 0$$

where $D > 0$ for planes and $D \leq 0$ for curves. This is the form that is required for use in the geometry routines. Radial zone lines are squared before printing.

More than one problem may be run successively by placing the second and succeeding problem input decks to the pre-Monte Carlo behind the first deck. This applies also to the input data for the Monte Carlo Output program.

The Monte Carlo input tape may be generated independent of the IV tape, through an option in Input Group 1 (number 10). Specifying 0 as the type of source generator will cause the source generator subroutine to be bypassed. If it is desired to use the same IV source tape with several Monte Carlo problems, this may be used. Of course, changes in the geometry or group structure portions may affect the source particles, so this option should be used with care. Different initial random numbers should be used in each new problem of this type. The user must also instruct the machine operators in advance to save the IV tape (S,SU05). This option should only be used if it takes a considerable amount of time to generate the IV tape for a particular problem. Since the source generator uses the zone and region search subroutines, the time required to generate a specified number of sources is proportional to the complexity of the geometry in the problem. A problem with relatively simple geometry, which can be set up on the basis of one region per zone, requires less than one minute to generate 5000 source particles. As the geometric complexity is increased, the time required increases accordingly. It is advisable to generate a new source tape each time a problem is run since it is quite likely that the time required for the computer operator to mount tapes will be more than the time required to generate a new set of sources.

2. METHOD OF INPUT

With the exception of Input Groups 0 and 3 of the pre-Monte Carlo input data and Input Group 0 of the Monte Carlo Output program input data, all input information is read in and then converted to binary form by means of the subroutine CVI (PPCo 40.0156). The following rules will be useful in preparing input to meet CVI specifications:

(1) One complete floating point or integer number constitutes a field, with the sign of the number placed in the first column of the field. A number is considered positive if no sign is given.

(2) A field is ended by any of the following: comma, blank, or asterisk. The only exception to this rule is given in 4(c). An asterisk signals the end of the number fields on a card, and the rest of the card may be used for comments. Columns 73 through 80 for Monte Carlo input will always be available for numbering or comment; therefore, no data may be punched in these columns.

(3) An integer number is indicated by a field containing a sign (if needed) followed only by numerical digits. Zeros placed in front of an integer do not affect its value.

(4) A floating point number may be indicated in three ways:

    (a) By a field with a decimal point in the desired position.

(b) By a digit in a field followed by a plus or a minus sign and a one or two digit integer exponent which is the power of 10 to which the number is raised. If no decimal point is given, the point is assumed to be in front of the first numerical digit of the field.

(c) By a digit in a field followed by an "E", a plus or a minus sign, and a one or two digit integer exponent. One blank immediately following the "E" can be used to indicate a positive exponent. As in (b), the decimal point may be given or assumed.

(5) Miscellaneous information. Any number of blanks can separate two fields. A number field may not extend to a second card. If there is not enough room before column 73 for a complete field, start the field on a new card. An integer zero (0) may be used to read in a floating point zero (0.0). The CVI subroutine will detect format errors and print out the card causing the error. Situations which are considered errors include: a blank followed by a comma, adjacent E's or multiple decimal points, adjacent commas (this will not enter a zero), a decimal point in the exponent, no digits in the exponent, and numbers too large or too small for the computer.

The sample input cards listed below will illustrate legitimate uses of the input subroutine:

(a) -231,+100,9,-15-3,+15-3,002,1+6,0 *COMMENT

(b) -231 100 9 -.00015,15E-3 2 1.E 05 0.0 *

(c) -231 100 +9 -15,0E-15 .15-3 02,100000.0 0

Cards (a), (b), and (c) are all equivalent and would read in the same numbers.

(6) Some of the input groups may require more than one card to complete the information. When this happens, end a card before column 73 with a complete field; then, continue the data on the next card, following the specifications for that input group. The following input groups may be continued in this manner: 7, 10, 11, 12, 14, 15, 16, 17, 18, 19, 23, 24, 27, 28, 29, 30, 32.

(7) It is suggested that the input deck be numbered sequentially in columns 77 through 80 or 73 through 80. Program-detected errors on a card, as well as errors determined by CVI, will cause a printing of the card on which the error was found and, eventually, will cause a program stop. Thus, numbering the cards will make it easier to locate an error. Any error found will prevent the generation of the input tapes.

3. INPUT TO THE PRE-MONTE CARLO INPUT PROGRAM

Input Group	Parameters in the Input Group (see following pages for explanations)
0.	Title Card
1.	G,GM,H,HS,F,M,ZG,S,IV,SG,MTØ
2.	CTEND,TRIALS,TRITIM/CTEND,IV/TR,TRIALS
3.	RANNØ
4.	ETIME,CTIME,XØ,YØ,ZØ
5.	Boundary Equation Parameters
6.	External Boundary Numbers
7.	Albedoes
8.	Region Descriptions
9.	TNZONE,XZL,YZL,ZZL
10.	X Zone Lines
11.	Y Zone Lines
12.	Z Zone Lines
13.	Regions per Zone
14.	Elastics, I_m
15	Fissions, F_m
16.	Group Importance Weights, I_g
17.	Region Importance Weights, I_h
18.	Mean Free Paths
19.	Microscopic Group Energy Cutoffs
20.	Macroscopic Group Energy Cutoffs
21.	P_{cmg}
22.	Thermal P_{cmg}
23.	Fission Spectrum Probability Tables
24.	Fission Spectrum Energy Tables
**	RESONANCE INPUT SECTION
25.	G_r,I,Q
26.	Resonance Materials, Atomic Masses
27.	Resonance Groups
28.	Resonance Parameter Tables
29.	Isotopes, Table Numbers, Smooth Cross Sections
30.	Resonant Material Information

** SOURCE GENERATOR INPUT SECTION

31. IV Source Generator Constants

32. IV Source Generator Exempt Regions

33. IV Source Generator Defining Parameters

Input Group 0. The first input card is a problem title card. Insert a 1 in column 1. Any alphanumeric message to describe the problem being run may be in columns 2 through 72.

Input Group 1 (Integer only) Consists of one card containing 12 integer numbers with the following information:

(1) G -- Total number of microscopic groups . . . $0 < G \leq 511$

(2) GM -- Total number of macroscopic groups . . . $0 < GM \leq 4$, and $GM \leq G$

(3) H -- Total number of real regions. . . $0 < H \leq 400$

(4) HS -- Total number of region sets (flux tally boxes) . . . $0 < HS \leq 400$, $HS \leq H$

(5) F -- Total number of fictitious regions . . . $0 < F \leq 111$

(6) M -- Total number of materials . . . $0 < M \leq 511$

(7) ZG -- Zone Geometry . . . $0 \leq ZG \leq 2$

0 indicates parallelepiped zones
1 indicates cylindrical zones
2 indicates spherical zones

(8) S -- Total number of boundary equations (surfaces) . . . $0 < S \leq 255$

(9) IV -- Total number of initial value source particles . . . $0 < IV \leq 30{,}000$

Note: IV must be a multiple of 45

(10) SG -- Type of source generator to be used for IV particles . . . $0 \leq SG \leq 3$

0 - Source generator is not to be used this time.
1 - Beam source, generating uniform points over a circular area, with direction cosines perpendicular to a plane described in Input Group 33.
2 - Uniform source over the whole volume under study, or portions of it, as described in Input Group 33, with random direction cosines.
3 - Point or disk source, angles generated in a cone.

(11) FS -- Total number of fission spectrum tables . . . $0 \leq FS \leq 10$

(12) MTO - Mono-thermal option

0 - regular thermal collision routine
1 - special thermal collision routine (not yet available)

Example: 1,1,2,2,3,1,1,8,5040,2,0,0

Input Group 2 (Integer only) Consists of one card containing 3 integer numbers with the following information:

1st number CTEND . . . The method of problem termination.

CTEND = 1, 2, or 3

1 indicates ending by census time

2 indicates a specified number of trials -- for which a length of time is given for the 1st trial

3 indicates a specified number of trials -- for which the number of IV's to be processed per trial is given.

If:	CTEND = 1	CTEND = 2	CTEND = 3
2nd number, 0		TRIALS. . .number of statistical trials to be taken	IV/TR. . .number of IV's to be processed per trial
3rd number, 0		TRITIM. . .time, in seconds for the first trial (this is clock time, not computed time)	TRIALS. . .number of statistical trials to be taken (IV/TRIALS must be equal to an integer)

Example: 2,60,120

Input Group 3 (Octal number) Consists of one card, containing one octal number, occupying the first 12 columns, which is used as the initial random number. The random number must be nonzero, unsigned (positive), and odd. No digit may be larger than 7, and the first digit must be 3 or less.

Example: 123456227651

Input Group 4 (Floating point only) Consists of one card containing 5 floating point numbers:

(1) ETIME -- Equilibration time, in microseconds. This can be zero.

(2) CTIME -- Census time, in microseconds . . . ETIME < CTIME For an infinite census time, use a very large number such as 1+8.

(3) XØ
(4) YØ } -- The geometry center point, subject to the following:
(5) ZØ

(a) If ZG = 0, then XØ = YØ = ZØ = 0

(b) If ZG = 1, then ZØ = 0, XØ and YØ ≠ 0

(c) If ZG = 2, then none of XØ, YØ, ZØ are 0

[See input group 1, item (7) for ZG]

Example: 0,1+8,3+1, 3+1,0

Input Group 5 -- Boundary Equation Parameters. Each equation is of the form: $Ax + By + Cz = D$ for planes, or $(x-x_0)^2 + (y-y_0)^2 + C(z-z_0)^2 = r^2$ for cylinders and spheres. One boundary is input per card.

All distances are in cm.

For planes list the parameters in the following order:

(1) Equation number (integer) -- these begin with 1 and increase sequentially on following cards.

(2) N - exponent of the equation (integer) (N=1 for planes)

(3) A
(4) B
(5) C
(6) D

} (floating point or 0).

The following types of planes are permissible:

$Ax = D$	ie,	$B = C = 0$
$By = D$	ie,	$A = C = 0$
$Cz = D$	ie,	$A = B = 0$
$Ax + By = D$	ie,	$C = 0$

For cylinders and spheres, list the parameters in the following order:

(1) Equation number (integer), as above

(2) N - exponent (integer) (N = 2 for cylinders and spheres)

(3) C (floating point), = 0 for cylinder, = 1.0 for sphere

(4) R - radius (not squared) (floating point or 0)

(5) x_o
(6) y_o
(7) (z_o)

} center of cylinder or sphere (floating point or 0) with respect to the (0,0,0) point of the coordinate system.

For a cylinder, the z_o term can be omitted.

Examples:

For the plane $x + 2y = 3$
1,1,1+1,2+1,0,3+1

For the cylinder $(x-3)^2 + (y-3)^2 = (2.51)^2$
2,2,0,251+1,3+1,3+1

Input Group 6 -- External (Reflecting) Boundary Numbers (Integer only) This group consists of one card of integers containing a list, in increasing order, of the boundary equation numbers (from Input Group 5.1) which form reflecting boundaries (those between fictitious and real regions).

Example: 1,17,20,21

Input Group 7 -- Albedoes (Floating point only) The albedo of a reflecting boundary is the measure of reflection of neutron beams. Albedoes may range in size from 0, for perfect transmittal, to 1.0, for perfect reflection. Albedoes greater than 1.0 indicate an external source and are acceptable.

List as follows: for each external boundary number listed in Group 6 input one albedo for each macroscopic group:

1st external boundary -- macro group 1, ... , macro group GM,

2nd external boundary -- macro group 1, ... , macro group GM,

.

.

.

nth external boundary -- macro group 1, ... , macro group GM.

The albedoes may continue on more than one card.

Example:

For the example for Input Group 6 with GM = 4

1st card - 1+1,1+1,1+1,1+1,0,0,0,0,1+1,8+0,6+0,4+0,1+1

2nd card - 5+0,1+1,5+0

or they may be given with one card for each macroscopic group as:

1st card - 1+1,1+1,1+1,1+1

2nd card - 0,0,0,0

3rd card - 1+1,8+0,6+0,4+0

4th card - 1+1,5+0,1+1,5+0

Input Group 8 -- Region Descriptions (Integer only) One region is described per card. There must be 7 to 13 numbers, as follows:

(1) Region number -- Start with 1 and continue in consecutive order on following cards, up to H. Fictitious regions must begin with number 401 and must follow the real regions, again numbered consecutively.

(2) Region set (flux tally box) -- The number of the region set in which the region flux is to be tallied. For fictitious regions, specify a 0; for any other region -- a number from 1 to HS.

(3) Region type -- The region type number as given in the list of region types described in the Monte Carlo geometry (Section IV-1).

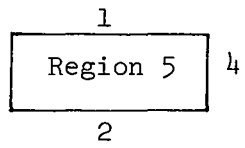
(4) Material -- The number of the material in this region, a number from 1 to M. For fictitious regions set material equal to 0.

(5) Number of boundaries for this region. Maximum is 8.

(6-13) Region boundary numbers -- a list of the signed boundary numbers (from Input Group 5) which describe this region. The signs must agree with the rules for boundary numbers given in Monte Carlo geometry (Section IV-3).

Example: For the rectangular region shown with top and bottom boundaries 5 and 6, material 2

5,5,1,2,6,-1,2,3,-4,-5,6



Input Group 9 -- (Integer only) This group consists of one card with the following four numbers on it:

(1) TNZØNE -- The total number of zones into which the problem has been divided. The maximum number is TNZØNE = 511.

(2) XZL -- The total number of zone lines along the x-axis, or in case of curved geometries (ZG = 1 or 2), the total number along the radial axis starting with the zone geometry center point.

(3) YZL -- The total number of zone lines along the y-axis. In case of curved geometry this number must be 0.

(4) ZZL -- The total number of zone lines along the z-axis. In case of spherical geometry this number must be 0.

As a check, TNZØNE must be equal to:

If: ZG = 0 (XZL-1)(YZL-1)(ZZL-1)

ZG = 1 (XZL-1)(ZZL-1)

ZG = 2 (XZL-1)

Example: 60,5,0,4

Input Group 10 -- X Zone Lines (Floating point only) A list of the distances (in cm), in increasing order, including the zone geometry center point, which form zone lines along the x or radial axis. In curved geometries, the first number must be 0, and the following numbers are radii with respect to the zone geometry center.

Example: 0,1+1,2+1,10+2

Input Group 11 -- Y Zone Lines (Floating point only) A list of the distances (in cm), in increasing order, from the zone geometry center point, which form zone lines along the y-axis in rectangular geometry. Omit this list if YZL = 0.

Input Group 12 -- Z Zone Lines (Floating point only) A list of the distances (in cm), in increasing order, from the zone geometry center point, which form zone lines along the z-axis for plane and cylindrical geometry. Omit this list if ZZL = 0.

Input Group 13 -- Regions per Zone (Integer only) One zone is listed per card, containing these numbers:

1. Zone number - Beginning with 1 and increasing consecutively on the following cards, up to TNZØNE.

2. Total number of regions in this zone.

3 to 9. A list of the regions, including fictitious regions, which are found in this zone. The maximum number of regions allowed in a zone is 7. Regions may be listed in any order, but they should be listed in order of relative size within the zone.

Example: 1,4,23,401,5,6

Input Group 14 -- Elastics (I_m) (Integer only) For each material, give the total number of elastic scatterers in that material.

Example: $I_{m_1}, I_{m_2}, \ldots, I_M$

Input Group 15 -- Fissions (F_m) (Integer only) For each material, list 0 if there is no fission in this material, or 1 if there is.

Example: $F_{m_1}, F_{m_2}, \ldots, F_M$

Input Group 16 -- Group Importance Weighting (I_g) (Floating point only) For each microscopic group, list the importance weight for that group. If all weights are to be 1.0, input the single number -1.

Example: $I_{g_1}, I_{g_2}, \ldots, I_G$

Input Group 17 -- Region Importance Weighting (I_h) (Floating point only) For each real region, list the importance weight for that region. If all weights are to be 1.0, input the single number -1.

Example: $I_{h_1}, I_{h_2}, \ldots, I_H$

Input Group 18 -- Mean Free Paths, λ_{mg} (Floating point only) List the mean free paths as follows:

List the λ for all microscopic groups, consecutively.

For each new material start a new card.

The list for any given material may extend to several cards.

Example:

Material one (1st card): $\lambda_{1g_1}, \lambda_{1g_2}, \ldots\ldots, \lambda_{1G}$

Material two (new card): $\lambda_{2g_1}, \lambda_{2g_2} \ldots\ldots, \lambda_{2G}$

.

.

.

Material M (new card): $\lambda_{Mg_1}, \lambda_{Mg_2}, \ldots\ldots, \lambda_{MG}$

Input Group 19 -- Microscopic Group Energy Cutoffs (Floating point only) For each microscopic group, g, list the lower energy cutoff for that group, in units of electron volts. The last number, for group G, will be 0.

Example: $E_{g_1}, E_{g_2}, \ldots, E_G$

Input Group 20 -- Macroscopic Group Energy Cutoffs (Floating point only) For each macroscopic group, gm, list the lower energy cutoff for that group, in units of electron volts. The last number, for group GM, will be 0.

Example: $E_{gm_1}, E_{gm_2}, \ldots, E_{GM}$

Input Group 21 -- P_{cmg} (Fast groups) For a one-group problem omit this input group. Start a new card for each new group and each new material. Mass = 0 implies no moderation for that isotope; no energy change. List the P_{cmg} numbers for the fast groups as follows:

In each material and group, list:

(1) P_F, fission collision emission number (floating point).

(2) Fission spectrum table index (integer).

Note: Omit the above if F_m for this material is 0.

(3) $P_{e\ell_1}$, elastic collision emission number; 1st elastic.

(4) $Mass_1$, ratio of isotope mass to neutron mass; 1st elastic.

$P_{e\ell_2}$; 2nd elastic

$Mass_2$; 2nd elastic

.

.

.

$P_{e\ell_{I_m}}$; I_m th elastic

$Mass_{I_m}$; I_mth elastic

} Floating point numbers

Repeat the above list for each group in a material before advancing to the next material. The list for any given m,g may extend to several cards.

If the resonance version of the program is being used, insert a 0 for each P_F or $P_{e\ell}$ which will be calculated by the program itself, ie, in every resonant group of a resonant material.

Input Group 22 -- Thermal P_{cmg} List the thermal group P_{cmg}'s as follows. Start a new card for each material:

(1) P_F, fission collision emission number (floating point).

(2) Fission spectrum table index (integer).

Note: Omit the above if F_m for this material is 0.

(3) $P_{e\ell}$, elastic collision emission number.

(4) $\bar{\mu}_o$ } As described in Section III-5

(5) $\bar{A}$ }

} Floating point numbers

No resonance is permitted in the thermal group

Input Group 23 -- Fission Spectrum Probability Tables If FS = 0, omit this input group. List the fission spectrum probability tables in order of decreasing probabilities as follows. Each table will require more than one card. On each card must be the following:

(1) Table number (integer) -- The table numbers must run sequentially, starting with 1. The number to be read in for the P_{cmg} fission (P_F) Table index refers to the table numbers described here.

(2) Card number for this table (integer). Starting at 1 for each new table, and must be numbered consecutively for each table.

(3 on) The remaining numbers on the card are the fission spectrum probability table numbers in decreasing order (floating point). There are 21 values in each table.

Any additional probability tables follow the first one in consecutive order.

Example: 1,1,1+1,9+0,8+0,7+0,.....

1,2,33-1,2-2,....,0

Input Group 24 -- Fission Spectrum Energy Table If FS = 0, omit this input group. List the fission spectrum energy, in electron volts corresponding to the probability from the table of Input Group 23 as follows:

(1) Same as Input Group 23, only insert as the number here the table number + 50 to distinguish these tables from the previous probability tables (integer).

(2) Same as Input Group 23 (integer).

(3 on) The remaining numbers on the card are the fission spectrum energy table in decreasing order (floating point). There are 21 values in each table.

Any additional energy tables follow the first one, in consecutive order.

Example: 51,1,1+6,1+5,99+3,.....

51,2,11+2,43+1,532+0,.....

RESONANCE INPUT SECTION (Omit Input Groups 25-30 if the nonresonant version is being used.)

Input Group 25 -- (Integer only) Consists of one card containing 3 integer numbers with the following information:

(1) G_r -- Total number of resonance groups, regardless which materials they are in.

(2) I -- Total number of isotopes in resonant materials; total including both resonant and nonresonant isotopes.

(3) Q -- Total number of resonance tables to be read in.

Note: If the resonant version of PMC is being used, however, the particular problem being run has no resonance; insert one card with a zero in column 1 in place of Input Groups 25-30.

Input Group 26 -- Resonance Materials, Atomic Masses Consists of one card, containing alternately: Resonant material number (in sequential order); atomic mass to neutron mass ratio, A_m, for collision lumping decision. Lumping will occur for those isotopes for which $A_i \geq A_m$ (see Section III-4).

Example: 1,2+3,2,18+3

Input Group 27 -- Resonance Groups (Integer only) A list in sequential order of microscopic group numbers in the resonant range. The largest group number will correspond to the least resonant energy. A total of G_r numbers must be read in, which may extend to several cards.

Example: 4,5,6,7

Input Group 28 -- Resonance Parameter Tables Using as many cards as are necessary, read in a resonance parameter table for each resonant isotope. These numbers correspond to the order of the parameters in the RBU library, but the definitions differ in some cases. They are all floating point numbers unless otherwise indicated.

(1) Actual resonance parameter table number (integer).

(2) Total number of parameters in the table, including both resolved and unresolved, beyond this point (integer).

(3) NE_0 (integer): Number of resolved resonances in this isotope.

(4) E_1: Upper energy cutoff in eV for the unresolved energy range.

(5) E_2: Cutoff energy in eV between the resolved and unresolved range.

(6) E_3: Lower energy cutoff in eV for the resolved range.

Numbers 7 through 12 are unresolved resonance parameters.

(7) MU1 $= < \Gamma_n^o >$: Average reduced neutron width in eV.

(8) MU2 $= < \Gamma_\gamma >$: Average capture width in eV.

(9) MU3 $= < \Gamma_F >$: Average fission width in eV.

(10) MU4 $= \nu_F$: Average number of neutrons per fission.

(11) MU5 $= D$: Average level spacing in eV for unresolved range.

(12) MU6 $= g = \frac{1}{2}\left(\frac{2J+1}{2I+1}\right)$: Statistical weight for S-wave resonances in unresolved range.

(13) MU7 $= \sigma_p$: S-wave potential scattering cross section for both the resolved and unresolved range.

(14) MU8 $= \frac{A}{A+1}$: Factor for transformation from lab to CM system, where A is the ratio of the isotope mass to neutron mass.

(15) MU9 $= 5.551 \times 10^3 A = \frac{1.9132}{4K} A$: Factor in Doppler line shape function, where K is Boltzmann's constant in eV/°K.

(16) MU10 $= 3.602 \times 10^6 \left(\frac{A+1}{A}\right) = \frac{4\pi\hbar^2}{2m} \sqrt{1.9132} \left(\frac{A+1}{A}\right)$ = Factor in resolved resonance cross section calculation [Equation (V-1)].

(17) MU11 = D^* : Linear factor to account for the variation of level density with energy [Equation (V-10)].

(18) MU12 = ℓ^* : Linear factor to account for variation of resonance cross sections in the unresolved range [Equation (V-10)].

(19) MU13 = σ_C^* : Constant for σ_C^*/v smooth capture cross section variation in the resolved and unresolved resonance range [see Equation (V-1)], in barns.

(20) MU14 = σ_F^* : Constant for a σ_F^*/v smooth fission cross section variation in the resolved and unresolved resonance range [see Equation (V-2)], in barns.

For each of the NE_0 resolved resonances in this isotope, in order of decreasing energies, list on one card:

E_0: Energy at resonance, eV.

Γ_n: Neutron width for elastic scattering, eV.

Γ_γ: Capture width, eV.

Γ_F: Fission width, eV.

g: Spin factor $= \frac{1}{2} \frac{(2J+1)}{(2I+1)}$.

Repeat Input Group 28 for all Q resonance parameter tables.

Example: See RBU Library format.

Input Group 29 -- Isotopes, Table Numbers, Smooth Cross Sections For each isotope in a resonant material, list the following:

(1) Isotope number: Actual library tape number (integer).

(2) Table number: Resonance table number to be used with this isotope (integer). If the isotope itself is not resonant, this must be set to 0.

(3) $g^{max}_{max,i}$: For all materials containing this isotope, give the group number corresponding to the maximum energy in a resonant group for all these materials (integer).

(4) $g^{min}_{min,i}$: For all materials containing this isotope, give the group number corresponding to the minimum energy in a resonant group for all of of these materials (integer).

Note: In numerical size, $g^{max}_{max,i} \leq g^{min}_{min,i}$

(5) Smooth microscopic cross sections in barns (floating point).

For each group, $g^{max}_{max,i}$ through $g^{min}_{min,i}$ (in order of increasing energy), list σ_C, σ_S, σ_F, $\nu\sigma_F$.

In part 5 start a new card with each group.

Example:

```
183,7,4,7
25+2,45+1,0,0
15+2,4+1,0,0
0,0,0,0
0,0,0,0
```

For isotope 183, groups 4 through 7 are resonant.

```
328,8,4,5
1+1,18+2,0,0
1+1,2+1,0,0
```

For isotope 328, groups 4 and 5 are resonant.

```
16,0,4,7
1-3,37+1,0,0
2-3,37+1,0,0
3-3,37+1,0,0
3-3,37+1,0,0
```

For isotope 16, groups 4 through 7 are resonant.

Input Group 30 -- Resonance Material Information For each resonant material only, read in the following in order of increasing material numbers:

(1) Resonant material number (integer).

(2) T_m : Temperature of this material in °K (floating point).

(3) g^{max}_m: Group number for the maximum energy of any group having resonance in this material (integer).

(4) g^{min}_m: Group number for the minimum energy of any group having resonance in this material (integer).

(5) R_m: Number of isotopes, both resonant and nonresonant, in this material (integer).

(6) For each of the R_m isotopes in this material list:

(a) Isotope number (integer).

(b) N^m_i: Atom density for this isotope in this material, in units of 10^{24} atoms/cm^3 (floating point).

(c) A_i: Moderating mass; the ratio of isotope mass to neutron mass (floating point).

In part 6 start on a new card with each of the R_m isotopes. Start on a new card for each additional resonant material.

Example:

2,293+3,4,7,3
183,187-2,1149+3
328,245-2,197+3
16,291-2,16+2

Resonant material 2 has 3 isotopes, some of which are resonant between energy groups 4 and 7, inclusive.

Input Group 31 -- IV Source Generator Constants This group consists of one card of two numbers as follows:

(1) Starting energy in electron volts for the IV particles (floating point)

(2) The total number of regions which are to be excluded as source regions (exempt regions) (integer)

Example: 1+8,0

Input Group 32 -- IV Source Generator Exempt Regions (Integer) If number 2 of Input Group 31 is 0, omit this input group. This group is a list of the regions which are to be excluded as source regions. Fictitious regions are automatically excluded.

Example: 9,10,1,4

Input Group 33 -- IV Source Generator Defining Parameters (Floating point)

(1) Source Generator 1

Source generator 1 generates sources uniformly from a circle of radius r, center $\bar{x},\bar{y},\bar{z}$, located on a specified plane with parameters A,B,C,D (AX+BY+CZ=D). All source planes must be of the form x, y, or z = constant with the corresponding coefficients either 0 or $\pm$ 1.0. Neutron direction for positive A,B,C is outward in the positive direction normal to the plane. Direction is inward for negative A, B, or C. If the source plane is to lie exactly on an outer boundary (on a line between real and fictitious regions), change the center of the source plane by a small amount sufficient to put the source plane physically within a real region.

For Source Generator 1, on the next card read in the following numbers (floating point):

A,B,C,$\bar{x},\bar{y},\bar{z}$,r

Example: 1+1,0,0,3+1,2+1,1+1,5+0

(2) Source Generator 2

Source Generator 2 generates points uniformly over the whole reactor or certain specified parts of it with random direction cosines.

For Source Generator 2 on the next card read in the following numbers (floating point): the max and min of the following refer to the maximum and minimum points on the axes, measured with respect to the zone geometry center point.

If Zone Geometry (ZG - Input Group 1) = 0:

$X_{max}, X_{min}, Y_{max}, Y_{min}, Z_{max}, Z_{min}$

If ZG = 1

$Z_{max}, Z_{min}, RADIUS_{max}, RADIUS_{min}$

If ZG = 2:

$RADIUS_{max}, RADIUS_{min}$

Example: (ZG = 0) 11+2,1+1,12+2,2+1,5+1,4+1

(3) Source Generator 3 (See Section VI-3)

Source Generator 3 generates sources at a point, $\bar{x},\bar{y},\bar{z}$, or from a disk with center, $\bar{x},\bar{y},\bar{z}$, and radius RAD located a distance, D, from the area which the particles are to hit. Angles are generated in the z-direction such that a cone is formed by the angles.

For Source Generator 3 on the next card read in the following six floating point numbers:

$\bar{x},\bar{y},\bar{z}$,RAD,W,D

where $\bar{x},\bar{y},\bar{z}$ is the location of the point source or the center of the disk source (this must not be located in a fictitious region).

RAD = 0 for a point source; RAD = radius for a disk source.
W = one half the length of the diagonal of the rectangular area toward which sources are directed, or the radius of the circular area if sources are directed toward that.
D = the perpendicular distance between the source and the area the source particles are to strike.

4. INPUT TO THE MONTE CARLO OUTPUT PROGRAM

Input data for the output program is a separate deck of cards from the pre-Monte Carlo Input program data deck. These data control the output editing and statistical analysis of the fluxes (and reaction rates). After Input Group 0, all input is in the standard CVI input format, as explained in Section VIII-2.

Input Group 0

Duplicate the title card for Input Group 0 of the pre-Monte Carlo input.

Input Group 1 (Integer only).

Consists of one card containing the three numbers, G,HS,F, where:

G = total number of microscopic groups

HS = total number of region sets

F = total number of fictitious regions.

These numbers must be the same as used in Input Group 1 of the pre-Monte Carlo input.

Input Group 2 (Integer only)

Consists of one card, containing the six numbers, ATH,ATG,PTH,PTG,EIGNV, MERGE, where:

ATH = total number of region sets to statistically analyze

ATG = total number of groups to statistically analyze

PTH = total number of region sets to print per trial

PTG = total number of groups to print per trial.

Note: For the four terms above, a "0" indicates none; a "-1" indicates all. After the last trial, however, the flux tallies for all region sets and all groups (regardless of value of PTH and PTG) are printed. The percent error will appear as "0" for any region set or group which is not requested to be statistically analyzed.

EIGNV: Does the user want a multiplication factor (K_{eff}) computed?

0 indicates no.
1 indicates yes.

MERGE: Are the results of two previous Monte Carlo runs to be merged together on one tape?

0 indicates no.
1 indicates yes.

Note: This option may be "1" only when the output program is run separate from the rest of the Monte Carlo program.

Input Group 3 (Integer only)

Omit this section if ATH = 0 or -1. Otherwise, list the region set numbers which are to be statistically analyzed, a total of ATH number.

Input Group 4 (Integer only)

Omit this section if ATG = 0 or -1. Otherwise, list the group numbers which are to be statistically analyzed, a total of ATG numbers.

Input Group 5 (Integer only)

Omit this section if PTH = 0 or -1. Otherwise, list the region set numbers which are to be printed each trial, a total of PTH numbers.

Input Group 6 (Integer only)

Omit this section if PTG = 0 or -1. Otherwise, list the group numbers which are to be printed each trial, a total of PTG numbers.

Any of Input Groups 3, 4, 5, and 6 may extend to several cards if necessary.

Input Group 7 (Floating point only)

Omit this section if EIGNV = 0. For EIGNV = 1 read in macroscopic cross sections Σ_a and $\nu\Sigma_F$ as follows: for each region set, read in Σ_a followed by $\nu\Sigma_F$, for all groups, ie,

Region set 1 (1st card): Σ_a, $\nu\Sigma_F$ (group 1), ..., Σ_a, $\nu\Sigma_F$ (group G)

Region set 2 (new card): Σ_a, $\nu\Sigma_F$ (group 1), ..., Σ_a, $\nu\Sigma_F$ (group G)

.

.

.

Region set HS (new card): Σ_a, $\nu\Sigma_F$ (group 1), ..., Σ_a, $\nu\Sigma_F$ (group G)

Begin a new card for each new region set. The list of cross sections for any region set may extend to several cards. However, the last number on any card must be a $\nu\Sigma_F$.

RESONANCE INPUT SECTION (Omit Input Groups 8 and 9 if the Nonresonant Version is being used.)

Input Group 8 (Integer only)

Consists of one card containing the three numbers, M,GM,IR, where:

M = total number of materials

GM = total number of macroscopic groups

IR = total number of resonant isotopes in resonant materials.

Note: If the resonant version of PMC is being used, but the particular problem being run has no resonance, insert one card with a zero in column 1 in place of Input Groups 8 and 9.

Input Group 9 (Integer only)

List the resonant isotope numbers in the same order they were read for Input Group 29 of the pre-Monte Carlo input. Do not list any of the nonresonant isotopes given in Input Group 29. A total of IR numbers are to be read; the list may extend to several cards.

5. TERMS WHICH APPEAR ON THE OUTPUT LISTINGS

The following terms, which appear on the output listing of the pre-Monte Carlo input edit and source generator, are defined:

(1) Problem title

(2) G = total number of microscopic groups
GM = total number of macroscopic groups

H = total number of nonfictitious regions in the geometry setup
HS = total number of region sets
F = total number of fictitious regions
M = total number of materials
ZG = zone geometry (0, parallelepiped; 1, cylindrical; 2, spherical)
S = total number of boundary equations
IV = total number of initial value particles (sources)
SG = type of source generator used (0, none; 1, beam; 2, uniform; 3, point or disk)
FS = total number of fission spectrum tables
MTO = monothermal option (presently = 0 only).

(3) EQ TIME = equilibration time limit, in microseconds
CENSUS TIME in microseconds

ZG XO
ZG YO
ZG ZO
} = zone geometry center point coordinates

(4) Boundary equation parameters which are normalized and put in the general form, $A(x-x_0) + B(y-y_0) + C(z-z_0) + D, =0$

NO = boundary number
N = plane (=1) or quadratic (=2) equation

A
B
C
D
x_0
y_0
z_0
} = parameters of the equation above
D(RSQ) = the negative of the radius squared, for an equation with N = 2

(5) External boundary numbers (EX.BN) and their associated albedoes, by macroscopic group

(6) Region descriptions

Region numbers, in order, with the corresponding region set, region type, material number of the region, number of boundaries, and the boundary numbers for the region. Fictitious regions are similarly listed, following the regular regions

(7) List of x zone lines (the radii will be squared for curved geometries)

(8) List of y zone lines (if there are any)

(9) List of z zone lines (if there are any)

(10) Zone contents -- zones, numbered consecutively, with the regions that are contained in each zone

(11) TNZONE = total number of zones
TNZEND = total number of zones having more than three regions in a zone
TNEBN = total number of external boundary numbers
XZL = total number of x zone lines
YZL = total number of y zone lines
ZZL = total number of z zone lines

(12) I_m - the total number of elastic scattering isotopes, per material

(13) F_m - indication of fission, per material (1, yes; 0, no)

(14) Importance weighting, listed by group

(15) Importance weighting, listed by region

(16) Mean free paths, lambda, listed by material and group (in sec^{-1})

(17) Microscopic group lower energy cutoffs (in eV)

(18) Macroscopic group lower energy cutoffs (in eV)

(19) Fast group P_{cmg}'s listed by material, and then by group. If there is no fission, the 1st, 3rd, 5th, etc, numbers are elastic collision emission numbers for each elastic scattering isotope; the alternating numbers are the corresponding masses. If there is fission for the material, the first number is the fission collision emission number, the second number is the fission spectrum table index and the following numbers are the same as in the nonfission case

(20) Thermal P_{cmg}'s listed for each material

If there is no fission, the three numbers per material are the elastic collision emission number, $\bar{\mu}_0, \bar{A}$. With fission, the first two numbers are the fission collision emission number and the fission spectrum table index followed by the same numbers as in the nonfission case

(21) A list of the fission spectrum probability tables given in order of decreasing probabilities

(22) A list of the fission spectrum energy tables given in order of decreasing energies (eV)

(23) The total number of (nonresonant) data words which were written on the input tape S.SU04. The constants are numbers such as those printed in numbers 2, 3, and 11 above

If the resonant version of the program is being used, items 24-31 will be listed

(24) GR = total number of resonant groups
IRM = total number of isotopes in resonant materials

Q = total number of resonance parameter tables given

(25) List of resonant material numbers and the corresponding lumping mass used in that material (AM)

(26) List of energy group numbers which are resonant

(27) Resonance parameter tables for both the unresolved and resolved energy ranges. The terms used are given in Input Group 28 of the pre-Monte Carlo input

(28) List of the isotope numbers used in resonant materials, their corresponding resonance parameter table (or 0 of the isotope is nonresonant), and the group number corresponding to their maximum and minimum resonant energy

(29) List of the isotope numbers used in resonant materials with their corresponding smooth microscopic cross sections σ_C, σ_S, σ_F, and $\nu\sigma_F$ for each resonant energy group

(30) List, by resonant material, of each isotope in that material: its corresponding atom density and mass to neutron mass ratio, the temperature of the material, in °K, the group number corresponding to the maximum and minimum resonant energy in this material, and RM, the number of isotopes in this material

(31) The total number of resonance data words which were written on the input tape S.SU04

(32) Source generator information

Initial energy, in eV
Number of exempt regions

(33) List of exempt regions; no sources are generated in these regions

(34) Parameters necessary to the source generator.

(a) For the beam source: in the source plane Ax + By + Cz = D, A, B, and C are printed, together with the center of the circular beam area $(\bar{x},\bar{y},\bar{z})$ and its radius -- as XBAR, YBAR, ZBAR, RAD

(b) For the uniform source: maximum and minimum point along the geometric axes limiting the range of source generation

(1) Parallelepiped zones: XMAX, XMIN, YMAX, YMIN, ZMAX, ZMIN

(2) Cylindrical zones: ZMAX, ZMIN, RADMAX, RADMIN

(3) Spherical zones: RADMAX, RADMIN

(c) For the point and disk source: XBAR, YBAR, ZBAR is the coordinate of the point source, or center of the disk

source; RAD is the radius of the disk source (=0 for a point source); W is the radius of the circular area, or one-half the length of the diagonal of the rectangular area, toward which the sources are directed. DIST is the perpendicular distance between the source and the area the source particles are to strike

(35) Sample IV source particles. Every 45th particle generated will be printed. It coordinates are: N (number of neutrons in the beam); X,Y,Z, position coordinates; ALPHA,BETA,GAMMA, direction cosines; PTIME, particle time; G, microscopic group; H, region; M, material

(36) Time required for the input edit, in seconds, and time required to generate the IV tape, also in seconds

The results of the edit of the output data from a problem run follows. If CTEND is equal to either 2 or 3, the output edit is made by trial. The contents of the flux and resonance reaction rate tally boxes are printed with the amount accumulated during the length of this trial. The contents of all other tally boxes are printed with the accumulated total for that box to the end of the trial. In addition, the last edit contains the total flux and reaction rate accumulated through the last trial with the percent of error for each box, based on the Student's t-analysis of the trial fluxes at a 95 percent confidence level. If the calculation of the multiplication factor is requested, it is listed after each trial and is based on the total flux accumulated through the end of that trial.

For each trial the following information is printed:

(1) Problem title

(2) Trial number, time in seconds for this trial to run, total flux in all region sets accumulated for this trial.

(3) Number of IV particles remaining to be processed.

(4) NPTCOL (NNTCOL), total number of particles (neutrons) which have been taken to collision through this trial.

(5) NPFCOL (NNFCOL), total number of particles (neutrons) which have been created as a result of collisions.

(6) Leakage tallies, listed by fictitious region (FREGION).PLEAK, total number of particles which have leaked through an outer boundary into the fictitious regions accumulated through this trial.
PINLEAK, continuous neutron beam weights which have leaked through an outer boundary into the fictitious regions.
OBSNLEAK, the number of neutrons represented by PLEAK which have leaked through an outer boundary into the fictitious regions.

(7) NPTHER (NNTHER), the total number of particles (neutrons) reaching the thermal group through this trial.

(8) NPTEQN (NNTEQN), the total number of particles (neutrons) which have reached equilibration through this trial. This total is 0, however, if equilibration time is input as 0.

(9) NPTCEN (NNTCEN), total number of particles (neutrons) which have reached census through this trial.

(10) PGEIW (NGEIW), total number of particles (neutrons) gained as a result of group importance weighting through this trial.

(11) PGHIW (NGHIW), total number of particles (neutrons) gained as a result of region importance weighting through this trial.

(12) PFFCOL (NFFCOL), total number of particles (neutrons) arising from fission collisions through this trial.

(13) PFUSED (NFUSED), total number of particles (neutrons) resulting from fission collisions which have been taken from the storage bank and reused through this trial.

(14) ERRBOX1, ERRBOX2, ERRBOX3, total number of errors in three categories which have resulted during the running of the problem accumulated through this trial. At any time during the running of a problem if any of the error tally boxes accumulates a number equal to ten percent of NPTCOL, a program stop will occur. In such a case, the output will not be printed, although the output tape, if saved, would include all trials completed up to the program stop.

ERRBOX1 tallies are a result of particles being lost, or trapped within a region.

ERRBOX2 tallies occur when a particle is found to lie entirely outside the problem's fictitious regions.

ERRBOX3 tallies occur when there are any of several miscellaneous errors arising from program checks throughout the code.

(15) ON TAPE, the number of blocks of 50 particles stored on the storage bank overflow tape at the end of this trial.

IVDONE, the number of IV particles processed during this trial plus the number of particles taken from the storage bank less the number of particles placed in the storage bank. This may be a negative number.

(16) If this problem has fission, the multiplication factor KEFF is given.

(17) The flux for this trial is listed by region set and group.

(18) If the resonant version is being used, the resonance reaction rate for absorption, scattering, and nu fission is listed by material, isotope, and macroscopic group.

In addition, after all trials have been completed the following is printed:

(1) Total accumulated flux over all groups, region sets, and time.
(2) Total accumulated flux for each region set and group tally box and its corresponding relative percent error from the statistical analysis.
(3) Total accumulated reaction rate in each category and its corresponding relative percent error.

The following quantities are printed on the computer console typewriter for the first and every tenth succeeding trial:

NPTC (in octal) = total number of particles taken to collision.
NPFC (in octal) = total number of particles arising from collisions.
IVP = total number of IV particles remaining to be used.
ERR = total number of type 1 errors (see 14 above).

At the conclusion of the fifth trial, a message will be typed to the computer operator instructing him to save the PMC output tape (S.SUO7) regardless of what happens if the user has indicated that he wants the tape saved. The output tape contains the flux, reaction rates, and miscellaneous tallies.

IX. REFERENCES

1. J. R. Triplett, E. T. Merrill, J. R. Burr, The RBU Reactor-Burnup Code: Formulation and Operating Procedures, HW-70049 (July 1961).

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3. L. J. Gannon and L. A. Schmittroth, Computer Generation and Testing of Random Numbers, IDO-16921 (August 1963).

4. C. E. Porter and R. G. Thomas, "Fluctuations of Nuclear Reaction Widths", Phy. Rev., 104 (October 15, 1956) pp 483-491.

5. J. Spanier, Monte Carlo Methods and Their Application to Neutron Transport Problems, WAPD-195 (July 1959).

6. H. A. Meyer (ed.), Symposium on Monte Carlo Methods, New York: John Wiley and Sons, Inc., 1956.

7. R. L. Curtis and R. A. Grimesey, INCITE -- A FORTRAN-IV Program to Generate Thermal Neutron Spectra and Multigroup Constants Using Arbitrary Scattering Kernels, IN-1062 (November 1967).

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APPENDIX A

CORE STORAGE AVAILABLE

The nonresonat version of PMC has approximately 19,000 core locations available for input data. This limit, in terms of symbols given in Sections VIII-3, -4, and -5.1, may be approximated by

(1) $2G + GM + 5H + 3F + 2M + 8S + TNZONE + XZL + YZL + ZZL + 42FS + (GM+1)\ TNEBN$

$$+ 2(G-1)\sum_{m=1}^{M} (I_m + F_m) + \sum_{m=1}^{M} (2F_m + 3) < 19{,}000.$$

In addition, the previous sum plus (HS) $G + 3F$ must be less than 24,700.

The resonant version has 18,600 core locations available. This may be expressed as restriction (1) plus

$$5M + GR + (Q+1) + 2IRM + IR + \text{size of resonance tables} + 4(IR)GR$$

$$+ 2\sum_{m=1}^{M} RM_m < 18{,}600.$$

In addition, the previous sum plus $3(IR)\ (GM)M + 3Max(RM) < 23{,}000$.

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APPENDIX B

PMC CARD DECK

The following layout of control cards and binary cards is used for a normal run of PMC program:

```
$JOB
$TIME
$IBJOB PRE-MC NOSOURCE
PRE-MONTE CARLO BINARY DECK
$ENTRY
PRE-MONTE CARLO INPUT DATA
$IBSYS
$CLOSE       S,SU05,REWIND
$TIME
$IBJOB *PMC* NOSOURCE, IOOP1
MONTE CARLO  BINARY  DECK
$ENTRY
$IBSYS
$TIME
$IBJOB MC-OUT  NOSOURCE
MONTE CARLO OUTPUT PROGRAM BINARY DECK
$ENTRY
MONTE CARLO OUTPUT PROGRAM INPUT DATA.
```

In addition, if the output tape containing the flux, reaction rate, and other tallies, is to be saved for later mergining with other output tapes, include at the end of the cards:

```
$IBSYS
$CLOSE       S,SUO7,REMOVE
```

The following layout of control cards and binary deck is used for merging two output tapes, or when the output program is run independently of the other programs:

```
$JOB
$PAUSE   (Instruct operator to mount the two tapes to be merged)
$IBJOB  MC-OUT NOSOURCE
MONTE CARLO OUTPUT PROGRAM BINARY DECK
$ENTRY
MONTE CARLO OUTPUT PROGRAM INPUT DATA
$IBSYS                          } (These cards are included only if
$CLOSE       S,SUO7,REMOVE      } the merged tape is to be saved)
```

Tapes to be merged are placed on S. SUO5 and S. SUO6 in either order. The merged results will appear on S,SU07. To merge two tapes, the MERGE option in the input data Input Group 2 must be set to 1.

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APPENDIX C

NONRESONANT SAMPLE PROBLEM

A calculation was made to obtain the thermal flux disadvantage factor for a hexagonal lattice of fuel pins. As such, only one-half the actual geometry needs to be mocked up, with perfectly reflecting albedoes used for the symmetry axes. Thus, boundary numbers 5, 6, 7, and 8 have albedoes of 1.0. Although the z-dimension is not shown, there are 4 boundaries enclosing 3 regions and 3 zones. The vertical axis fictitious regions are numbered 405 (bottom) and 406 (top). They entirely cover the rectangular area shown, both below and above. The central vertical zones contain all of the regions shown in Figure C-1.

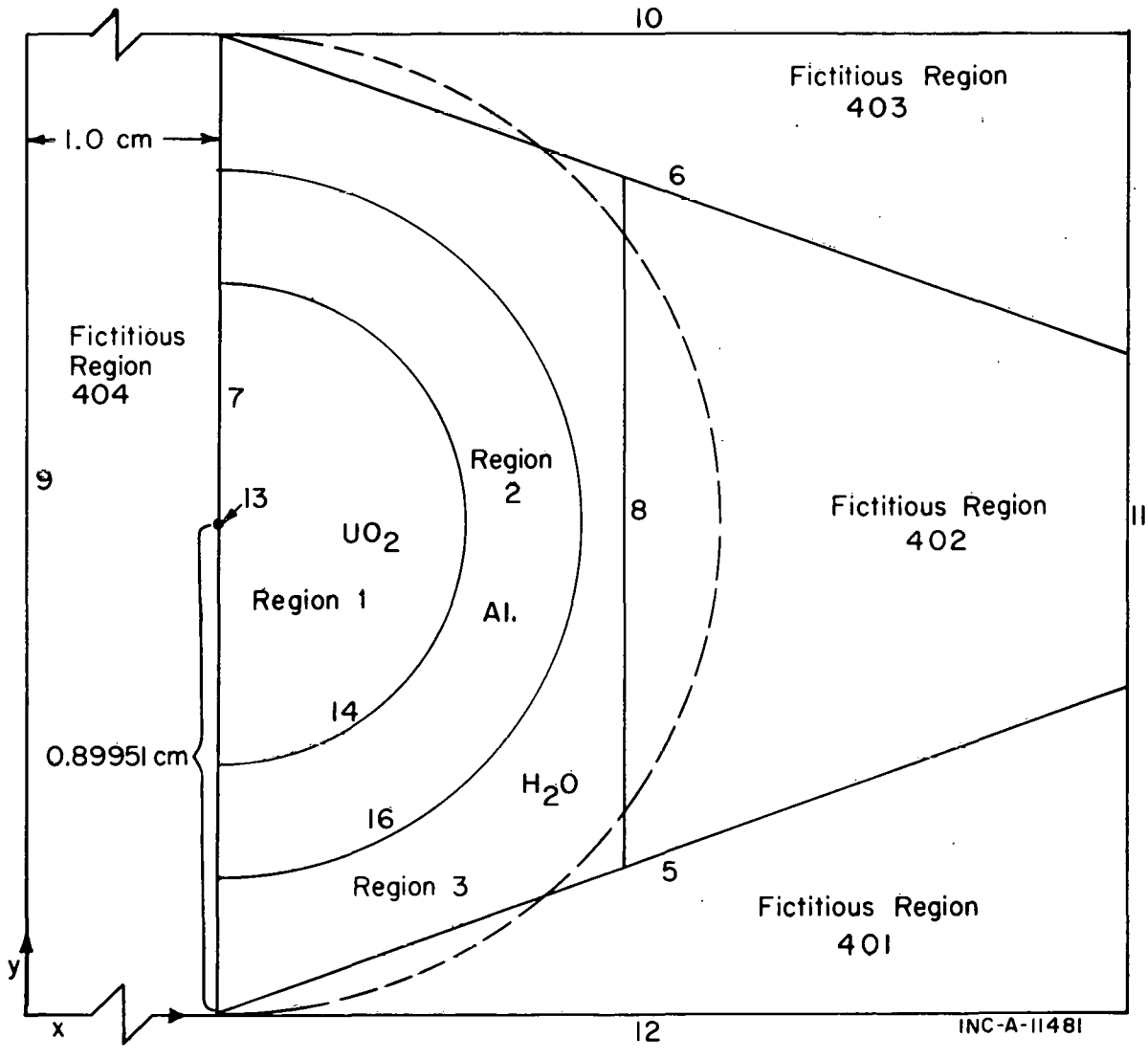


FIG. C-1 GEOMETRY DIAGRAM FOR NONRESONANT SAMPLE PROBLEM.

The problem was a one group (monoenergetic) mock-up using thermal cross sections from the TEMPEST Code for each material indicated. No fission cross sections were included in region 1 since the problem was to be monoenergetic. A homogeneous isotropic source of 45 IV coordinates per trial was generated for region 3. Eighteen trials were run for the statistical sample. Cylindrical zone geometry was employed. The region and boundary numbers are as given in the diagram in the same order they are listed on the accompanying input form.

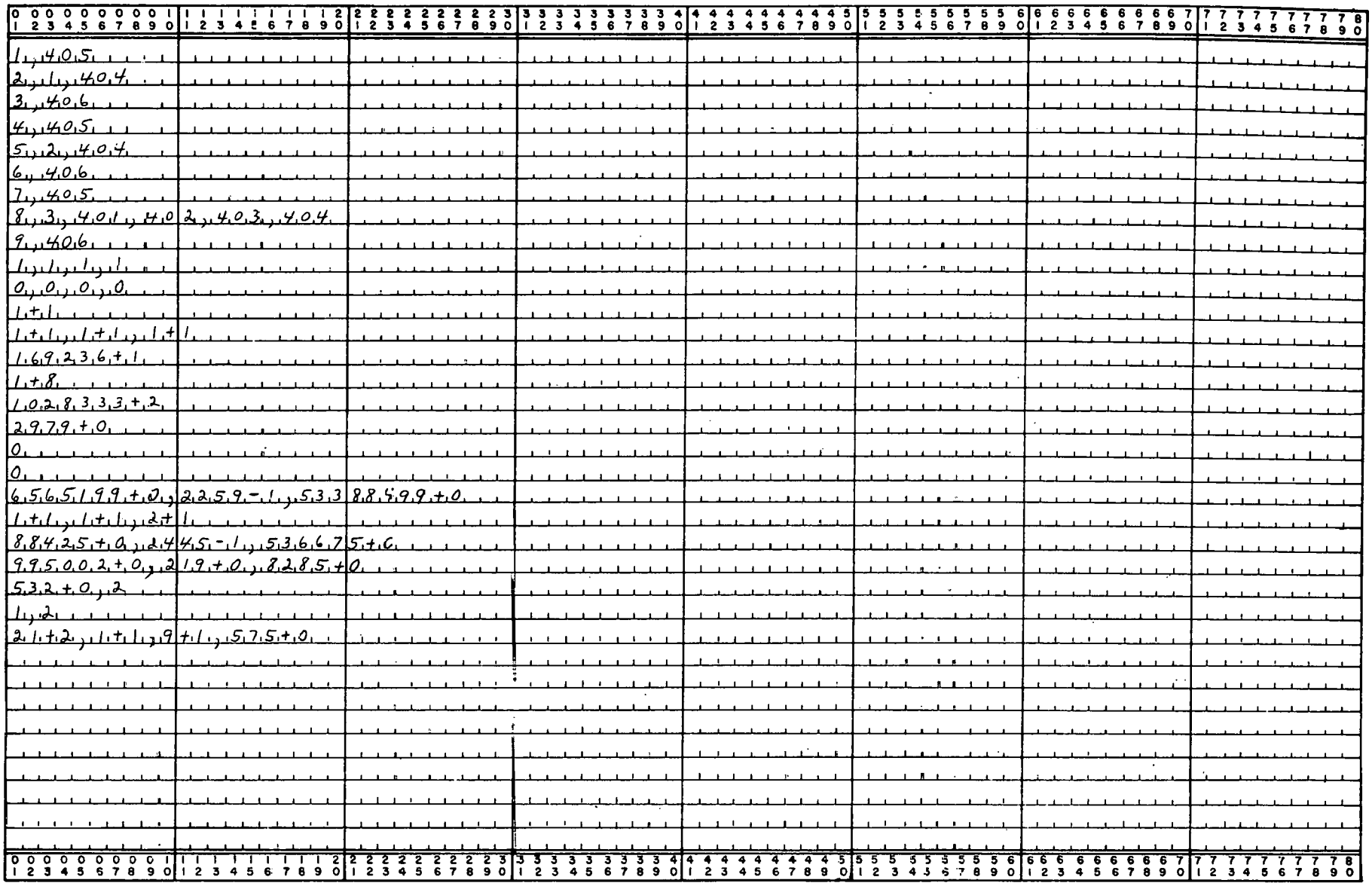
Boundary numbers 1, 2, 3, 4 are the four vertical planes from bottom to top, respectively. These same boundaries constitute vertical zone lines. The three radial region boundaries, 13, 14, 15, were also chosen to be radial zone lines. One additional radial zone line is indicated by the dashed curve in the figure. It is important to remember that the geometry center point at boundary number 13 is also both a region and zone boundary. The machine time for 18 trials was 50 minutes. Four materials were input to this problem even though only three were used. This is permissible. Sixteen boundaries were input and only 15 were actually used. This is also permissible.

In this problem, IV source particles were generated in region 3 only. The source generator will create sources uniformly in a circular annulus between the specified RADMAX and RADMIN. Since it is circular, the fictitious region to the left of the semi-cylinders must be large enough such that no sources will be generated in an undefined region, in other words, sources not in the first octant. This would cause errors in the region search routine and ultimately end in a program stop.

Other details can be inferred from the sample input form. To conserve space, the output listing shows only the first trial, last trial, and final results.

NONRESONANT SAMPLE PROBLEM -- INPUT TO PRE-MC INPUT PROGRAM

1–10	11–20	21–30	31–40	41–50	51–60	61–70	71–80
1 TRX CRIT	LATTICE 1	DIS FACTO	R NB1279 P	AGE 6FF VO	ID=AL		
1,1,3,3,6,	4,1,16,810	,2,0,0					
3,45,18							
1112223334	47						
0,1+6,1+1,	89951+0,0						
1,1,0,0,1+	1,0						
2,1,0,0,1+	1,1+1						
3,1,0,0,1+	1,21+2						
4,1,0,0,1+	1,22+2						
5,1,-57735	+0,1+1,0,-	57735+0					
6,1,57735+	0,1+1,0,23	76374+1					
7,1,1+1,0,	0,1+1						
8,1,1+1,0,	0,1779+1						
9,1,1+1,0,	0,0						
10,1,0,1+1	,0,1799024	+1					
11,1,1+1,0	,0,21+1						
12,1,0,1+1	,0,0						
13,2,0,0,1	+1,89951+0						
14,2,0,487	05+0,1+1,8	9951+0					
15,2,0,503	94+0,1+1,8	9951+0					
16,2,0,575	06+0,1+1,8	9951+0					
2,3,5,6,7,	8						
1+1,1+1,1+	1,1+1,1+1,	1+1					
1,1,3,1,5,	2,-3,7,13,	14					
2,2,3,3,5,	2,-3,7,14,	16					
3,3,2,4,7,	2,-3,5,-6,	7,-8,16					
401,0,1,0,	5,2,-3,-5,	-11,12					
402,0,1,0,	6,2,-3,5,-	6,8,-11					
403,0,1,0,	5,2,-3,6,-	10,-11					
404,0,1,0,	6,2,-3,9,-	10,12,-7					
405,0,1,0,	6,1,-2,9,-	11,12,-10					
406,0,1,0,	6,3,-4,9,-	11,12,-10					
9,4,0,4							
0,48705+0,	57506+0,11	+1					
0,1+1,21+2	,22+2						1



NONRESONANT SAMPLE PROBLEM -- INPUT TO OUTPUT PROGRAM

```
1 TRX CRIT LATTICE 1 DIS FACTOR NB1279 PAGE 6FF VOID=AL
1,3,6
-1,-1,-1,-1,0,0
```

TRX CRIT LATTICE 1 DIS FACTOR NB1279 PAGE 6FF VOID=AL

G	GM	H	HS	F	M	ZG	S	IV	SG	FS	MTO
1	1	3	3	6	4	1	16	810	2	0	0

CTEND	IV/TR	TRIALS	RANDOM NO
3	45	18	111222333447

EQ TIME	CENSUS TIME	ZG XO	ZG YO	ZG ZO
0.	0.10000000E 06	0.10000000E 01	0.89951000E 00	0.

BOUNDARY EQLATION PARAMETERS

NO	N	A	B	C	D(RSC)	XO	YO	ZO
1	1	0.	0.	0.10000000E 01	0.10000000E 01	0.	0.	0.10000000E 01
2	1	0.	0.	0.10000000E 01	0.10000000E 01	0.	0.	0.20000000E 01
3	1	0.	0.	0.10000000E 01	0.10000000E 01	0.	0.	0.22000000E 02
4	1	0.	0.	0.10000000E 01	0.10000000E 01	0.	0.	0.23000000E 02
5	1	-0.49999983E 00	0.86602531E 00	0.	0.10000000E 01	-0.10000007E 01	0.	0.
6	1	0.49999983E 00	0.86602531E 00	0.	0.10000000E 01	0.61160030E 01	0.	0.
7	1	0.10000000E 01	0.	0.	0.10000000E 01	0.20000000E 01	0.	0.
8	1	0.10000000E 01	0.	0.	0.10000000E 01	0.27790000E 01	0.	0.
9	1	0.10000000E 01	0.	0.	0.10000000E 01	0.10000000E 01	0.	0.
10	1	0.	0.10000000E 01	0.	0.10000000E 01	0.	0.27990240E 01	0.
11	1	0.10000000E 01	0.	0.	0.10000000E 01	0.31000000E 01	0.	0.
12	1	0.	0.10000000E 01	0.	0.10000000E 01	0.	0.10000000E 01	0.
13	2	0.10000000E 01	0.10000000E 01	0.	-0.	0.10000000E 01	0.89951000E 00	0.
14	2	0.10000000E 01	0.10000000E 01	0.	-0.23721770E 00	0.10000000E 01	0.89951000E 00	0.
15	2	0.10000000E 01	0.10000000E 01	0.	-0.25395552E 00	0.10000000E 01	0.89951000E 00	0.
16	2	0.10000000E 01	0.10000000E 01	0.	-0.33069400E 00	0.10000000E 01	0.89951000E 00	0.

ALBEDOES

EX,BN	MACG1	MACG2	MACG3	MACG4
2	1.000			
3	1.000			
5	1.000			
6	1.000			
7	1.000			
8	1.000			

REGION DESCRIPTIONS

REGION	REGSET	REGTYPE	MAT	NOBDYS	BOUNDARY NUMBERS--					
1	1	3	1	5	2	-3	7	13	14	
2	2	3	3	5	2	-3	7	14	16	
3	3	2	4	7	2	-3	5	-6	7	-8 16
401	0	1	0	5	2	-3	-5	-11	12	
402	0	1	0	6	2	-3	5	-6	8	-11
403	0	1	0	5	2	-3	6	-10	-11	
404	0	1	0	6	2	-3	9	-10	12	-7
405	0	1	0	6	1	-2	9	-11	12	-10
406	0	1	0	6	3	-4	9	-11	12	-10

X ZONE LINES

0. 0.23722E 00 0.35069E 00 0.12100E 01

Z ZONE LINES

0. 0.10000E 01 0.21000E 02 0.22000E 02

ZONE = HAS REGIONS

1 405

2 1 404

```
     3    406
     4    405
     5      2  404
     6    406
     7    405
     8      3  401  402  403  404
     9    406

  TNZONE  TNZEND  TNEBN    XZL     YZL     ZZL
3.5   9       1      6      4       0       4

NO OF ELASTICS PER MATERIAL-IM
   1    1    1    1

FISSION PER MATERIAL,FM =1(YES) =0(NO)
   0    0    0    0

IMPORTANCE WEIGHTING BY GROUP-IG
   1.000

IMPORTANCE WEIGHTING BY REGION-IH
   1.000    1.000    1.000

  MEAN FREE PATHS
MATERIAL  GROUP      LAMBDA
   1        1      0.16923600E 01

   2        1      0.10000000E 08

   3        1      0.10283330E 02

   4        1      0.29790000E 00


MICRO GROUP ENERGY CUTOFFS
 0.

MACRO GROUP ENERGY CUTOFFS
 0.

THERMAL PCMG, MATERIAL    1
 0.65651990E 00  0.22590000E-01  0.53388499E 00

THERMAL PCMG, MATERIAL    2
 0.10000000E 01  0.10000000E 01  0.20000000E 01

THERMAL PCMG, MATERIAL    3
 0.88424999E 00  0.24450000E-01  0.53667500E 00

THERMAL PCMG, MATERIAL    4
 0.99500199E 00  0.21900000E 00  0.82850000E 00

TOTAL NUMBER OF DATA WORDS ON TAPE     202 PLUS 36 CONSTANTS


   SOURCE GENERATOR INFORMATION

STARTING ENERGY   NO OF EXEMPT REGIONS
 0.53200000E 00            2
```

EXEMPT REGIONS
1 2

ZMAX	ZMIN	RADMAX	RADMIN
21.000	1.000	0.900	0.575

BORN STORAGE IV PARTICLES

N	X	Y	Z	ALPHA	BETA	GAMMA	PTIME	SQRT(E)	G	H	M
1	1.744	0.631	14.380	-0.131	-0.632	0.764	0.000	0.729	1	3	4
1	1.185	0.252	15.391	-0.779	0.605	-0.165	0.000	0.729	1	3	4
1	1.038	1.741	7.013	0.039	0.630	-0.776	0.000	0.729	1	3	4
1	1.662	1.261	12.714	0.977	-0.056	-0.207	0.000	0.729	1	3	4
1	1.497	0.574	10.638	0.846	0.447	0.860	0.000	0.729	1	3	4
1	1.652	0.965	11.307	0.548	-0.306	-0.086	0.000	0.729	1	3	4
1	1.560	1.423	8.887	0.196	-0.974	-0.113	0.000	0.729	1	3	4
1	1.740	0.471	15.358	-0.803	-0.536	-0.259	0.000	0.729	1	3	4
1	1.592	1.276	5.035	-0.250	-0.770	0.587	0.000	0.729	1	3	4
1	1.643	1.383	5.278	-0.100	0.599	0.795	0.000	0.729	1	3	4
1	1.046	1.647	17.156	-0.390	-0.754	0.529	0.000	0.729	1	3	4
1	1.177	1.530	1.984	0.898	0.026	0.439	0.000	0.729	1	3	4
1	1.335	1.367	3.423	-0.206	-0.949	-0.239	0.000	0.729	1	3	4
1	1.353	1.566	19.455	0.403	-0.080	0.912	0.000	0.729	1	3	4
1	1.367	0.280	3.576	-0.743	0.640	0.198	0.000	0.729	1	3	4
1	1.602	0.823	2.224	-0.736	0.311	-0.534	0.000	0.729	1	3	4
1	1.601	1.371	8.205	0.138	0.475	-0.869	0.000	0.729	1	3	4
1	1.203	0.325	6.549	0.959	0.248	0.136	0.000	0.729	1	3	4

0

TIME FOR INPUT EDIT = 4 SEC
TIME FOR IV GENERATION = 42 SEC

END-OF-DATA ENCOUNTERED ON SYSTEM INPUT FILE.

```
TRX CRIT LATTICE 1 DIS FACTOR NB1279 PAGE 6FF VOID=AL

TRIAL  1    TRIAL TIME =    114    TRIAL FLUX =     477,980

IV PART LEFT     765
NPTCOL      955    NNTCOL    0.95500000E 03
NPFCOL      912    NNFCOL    0.91200000E 03


FREGION  401    PLEAK        0    PINLEAK    0,              OBSNLEAK  =0,
FREGION  402    PLEAK        0    PINLEAK    0,              OBSNLEAK  =0,
FREGION  403    PLEAK        0    PINLEAK    0,              OBSNLEAK  =0,
FREGION  404    PLEAK        0    PINLEAK    0,              OBSNLEAK  =0,
FREGION  405    PLEAK        0    PINLEAK    0,              OBSNLEAK  =0,
FREGION  406    PLEAK        0    PINLEAK    0,              OBSNLEAK  =0,

NPTHER      912    NNTHER    0.91200000E 03
NPTEON        0    NNTEON    0.
NPTCEN        0    NNTCEN    0.
PGEIW         0    NGEIW     0.
PGHIW         0    NGHIW     0.
PFFCOL        0    NFFCOL    0.
PFUSED        0    NFUSED    0.

ERRBOX1       1    ERRBOX2      0    ERRBOX3      0
ON TAPE       0    IVDONE      45


 REGSET GROUP      TRIAL FLUX
    1     1     0.16062070E 03
    2     1     0.62262921E 02
    3     1     0.25510018E 03
```

TRX CRIT LATTICE 1 DIS FACTOR NB1279 PAGE 6FF VOID=AL

TRIAL 18 TRIAL TIME = 135 TRIAL FLUX = 568.43

IV PART LEFT 0
NPTCOL 20129 NNTCOL 0.20129000E 05
NPFCOL 19328 NNFCOL 0.19328000E 05

FREGION	401	PLEAK	0	PINLEAK	0.	OBSNLEAK	0.
FREGION	402	PLEAK	0	PINLEAK	0.	OBSNLEAK	0.
FREGION	403	PLEAK	0	PINLEAK	0.	OBSNLEAK	0.
FREGION	404	PLEAK	0	PINLEAK	0.	OBSNLEAK	0.
FREGION	405	PLEAK	0	PINLEAK	0.	OBSNLEAK	0.
FREGION	406	PLEAK	0	PINLEAK	0.	OBSNLEAK	0.

NPTHER 19328 NNTHER 0.19328000E 05
NPTEQN 0 NNTEQN 0.
NPTCEN 0 NNTCEN 0.
PGEIW 0 NGEIW 0.
PGHIW 0 NGHIW 0.
PFFCOL 0 NFFCOL 0.
PFUSED 0 NFUSED 0.

ERRBOX1 16 ERRBOX2 0 ERRBOX3 0
ON TAPE 0 IVDONE 45

REGSET	GROUP	TRIAL FLUX
1	1	0.18840741E 03
2	1	0.79133728E 02
3	1	0.30077381E 03

TRX CRIT LATTICE 1 DIS FACTOR NB1279 PAGE 6FF VOID=AL

TOTAL FLUX OVER ALL M,G = 0.10147032E 05

REGSET	GROUP	TOTAL FLUX	PER CENT ERR
1	1	0.33863907E 04	0.78397838E 01
2	1	0.14013044E 04	0.81842337E 01
3	1	0.53583404E 04	0.78421116E 01

APPENDIX D

RESONANT SAMPLE PROBLEM

This problem was chosen for its simplicity to illustrate the input and output of a resonance-type problem. The accompanying geometry diagram (Figure D-1) is self-explanatory. Cylindrical zone geometry was employed for the two cylindrical regions. Region one is a $^{238}UO_2$ pin surrounded by H_2O in region two. It was desired to obtain the average absorption cross section of U-238 in the resolved and unresolved range of region one. The macroscopic energy groups were thus set at the resolved and unresolved energy cutoffs for U-238. The lower resolved energy cutoff was chosen as 0.532 eV for convenience. There were 55 resolved resonances, and the set of unresolved parameters were as given in the input. No smooth cross sections for U-238 were input in groups four and five. Average group cross sections for 16_O and H_2O were computed by a fast spectrum code, and the appropriate P_{cmg} were then determined. The fast P_{cmg} for material one in groups four and five were input as 1.0, but were meaningless since they are in the resonance range (ie, these values will be ignored since the P_{cmg} will be recomputed from the resonance cross sections at each particle energy in groups four and five). Complete attenuation was specified in the thermal group in both regions so that no time would be wasted on thermal collisions. A total of 7200 initial value particles was generated in region 2 in a flat distribution, with an initial energy of one MeV. In this way a 1/E slowing down source was generated into the unresolved range. Forty statistical trials of 180 initial value coordinates per trial were prescribed. The running time was approximately 1.9 hours.

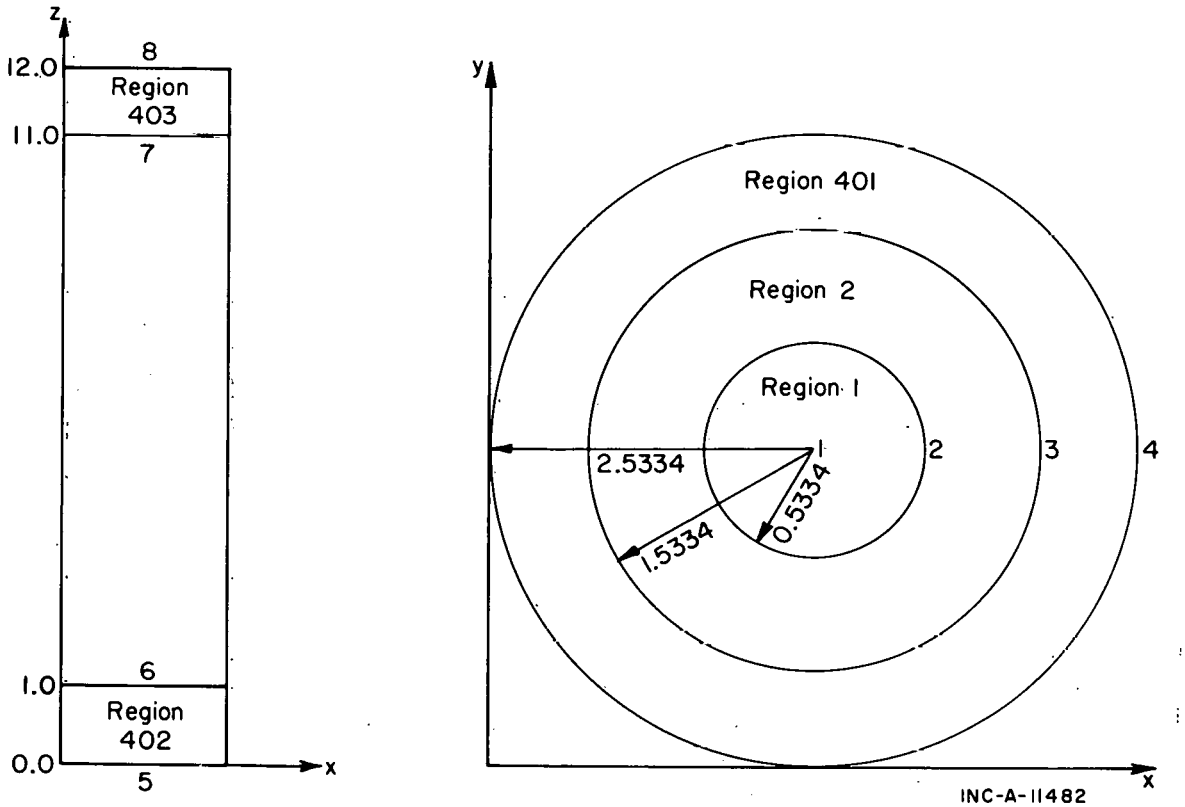
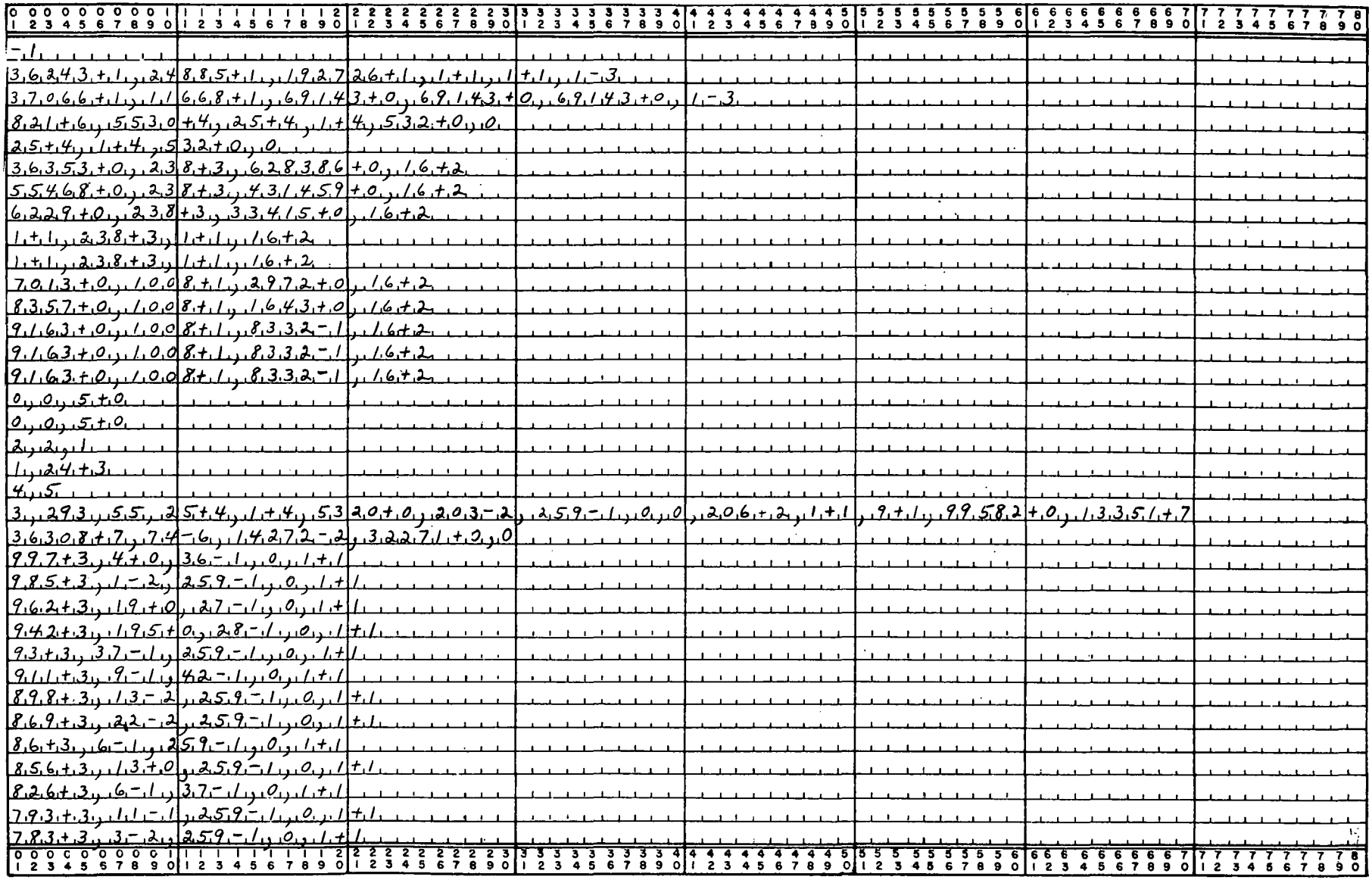
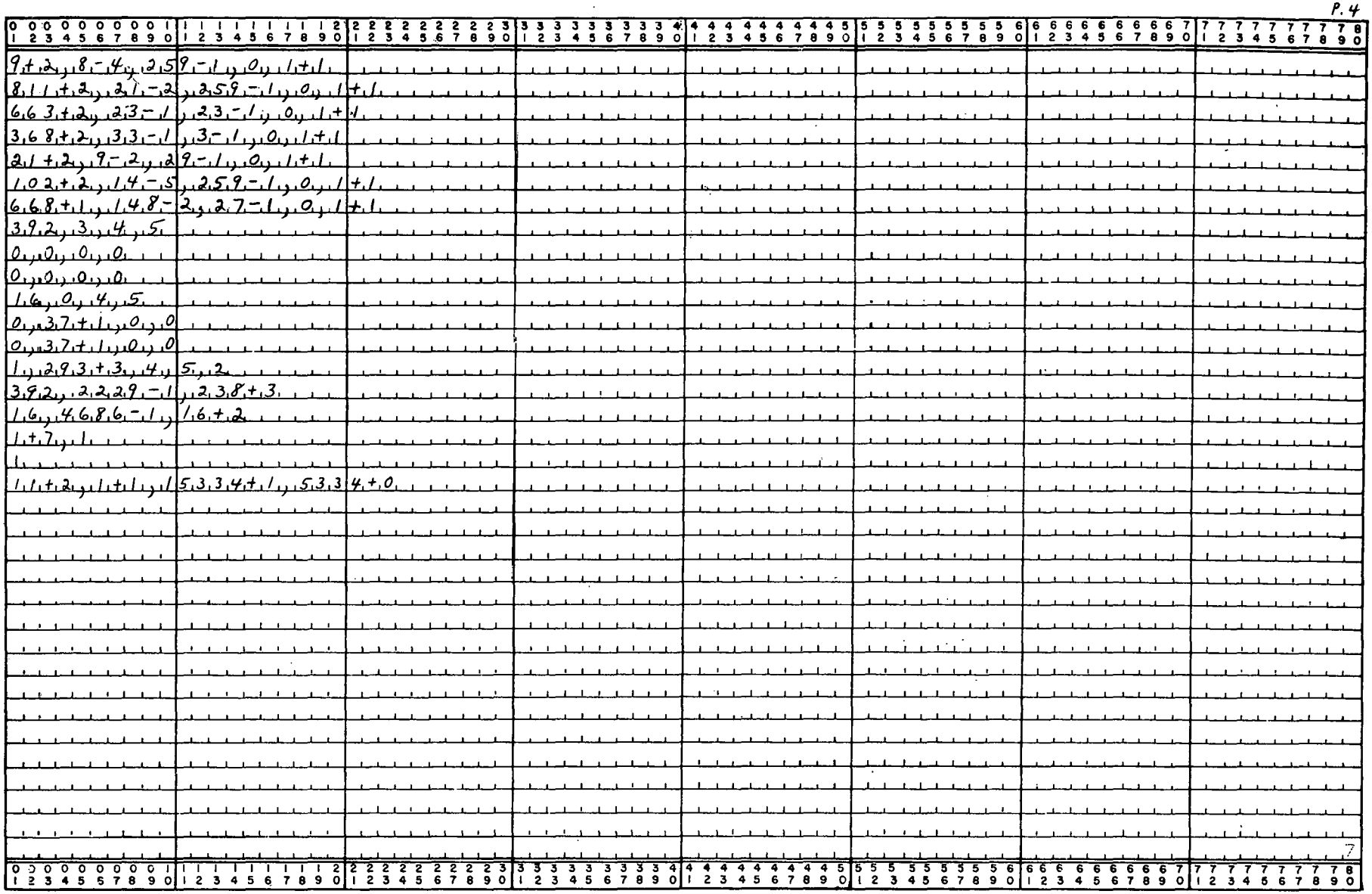


FIG. D-1 GEOMETRY DIAGRAM FOR RESONANT SAMPLE PROBLEM.

RESONANT SAMPLE PROBLEM -- INPUT TO PRE-MC INPUT PROGRAM

```
1 238 RES INT RUN WITH ZUT 6 6P 2 REG   12/27/65    2 HR RUN
6,4,2,2,3,2,1,8,7200,2,0,0
3,180,40
271346571127
0,1+8,25334+1,25334+1,0
1,2,0,0,25334+1,25334+1
2,2,0,5334+0,25334+1,25334+1
3,2,0,15334+1,25334+1,25334+1
4,2,0,25334+1,25334+1,25334+1
5,1,0,0,1+1,0
6,1,0,0,1+1,1+1
7,1,0,0,1+1,11+2
8,1,0,0,1+1,12+2
3,6,7
1+1,1+1,1+1,1+1,1+1,1+1,1+1,1+1,1+1,1+1,1+1,1+1
1,1,3,1,4,1,2,6,-7
2,2,3,2,4,2,3,6,-7
401,0,3,0,4,3,4,6,-7
402,0,3,0,4,1,4,5,-6
403,0,3,0,4,1,4,7,-8
9,4,0,4
0,5334+0,15334+1,25334+1
0,1+1,11+2,12+2
1,1,402
2,1,1
3,1,403
4,1,402
5,1,2
6,1,403
7,1,402
8,1,401
9,1,403
2,2
0,0
-1
```





RESONANT SAMPLE PROBLEM -- INPUT TO OUTPUT PROGRAM

1,2,3,8,RES,	INT,RUN,WITH,ZUT,6,	G,3,2,REG,	1,2,/2,7,/6,5,	2,HR,RUN		
6,2,3						
-1,-1,-1,-1,-1	1,0,2					
2,4,1						
3,9,2						

238 RES INT RUN WITH ZUT 6 GP 2 REQ. 12/27/65 2 HR RUN

G	GM	H	HS	F	M	ZG	S	IV	SG	FS	MTD
6	4	2	2	3	2	1	8	7200	2	0	0

CTEND	IV/TR	TRIALS	RANDOM NO
3	180	40	271346571127

EQ TIME	CENSUS TIME	ZG XO	ZG YO	ZG ZO
0.	0.10000000E 08	0.25334000E 01	0.25334000E 01	0.

BOUNDARY EQUATION PARAMETERS

NO	N	A	B	C	D(RSQ)	XO	YO	ZO
1	2	0.10000000E 01	0.10000000E 01	0.	-0.	0.25334000E 01	0.25334000E 01	0.
2	2	0.10000000E 01	0.10000000E 01	0.	-0.28451556E 00	0.25334000E 01	0.25334000E 01	0.
3	2	0.10000000E 01	0.10000000E 01	0.	-0.23513156E 01	0.25334000E 01	0.25334000E 01	0.
4	2	0.10000000E 01	0.10000000E 01	0.	-0.64181155E 01	0.25334000E 01	0.25334000E 01	0.
5	1	0.	0.	0.10000000E 01	0.10000000E 01	0.	0.	0.10000000E 01
6	1	0.	0.	0.10000000E 01	0.10000000E 01	0.	0.	0.20000000E 01
7	1	0.	0.	0.10000000E 01	0.10000000E 01	0.	0.	0.12000000E 02
8	1	0.	0.	0.10000000E 01	0.10000000E 01	0.	0.	0.13000000E 02

ALBEDOES

EX,BN	MACG1	MACG2	MACG3	MACG4
3	1.000	1.000	1.000	1.000
6	1.000	1.000	1.000	1.000
7	1.000	1.000	1.000	1.000

REGION DESCRIPTIONS

REGION	REGSET	REGTYPE	MAT	NOBDYS	BOUNDARY NUMBERS--			
1	1	3	1	4	1	2	6	-7
2	2	3	2	4	2	3	6	-7
401	0	3	0	4	3	4	6	-7
402	0	3	0	4	1	4	5	-6
403	0	3	0	4	1	4	7	-8

X ZONE LINES

0.	0.28452E 00	0.23513E 01	0.64181E 01

Z ZONE LINES

0.	0.10000E 01	0.11000E 02	0.12000E 02

ZONE = HAS REGIONS

1	402
2	1
3	403
4	402
5	2
6	403
7	402
8	401
9	403

TNZONE	TNZEND	TNEBN	XZL	YZL	ZZL
9	0	3	4	0	4

NO OF ELASTICS,PER MATERIAL=IM

2	2

FISSION PER MATERIAL=FM =1(YES) =0(NO)

```
   0    0

IMPORTANCE WEIGHTING BY GROUP=IG
ALL GROUPS HAVE WEIGHT 1.0

IMPORTANCE WEIGHTING BY REGION=IM
ALL REGIONS HAVE WEIGHT 1.0

  MEAN FREE PATHS
MATERIAL GROUP      LAMBDA
   1       1     0.36243000E 01
           2     0.24885000E 01
           3     0.19272600E 01
           4     0.10800000E 01
           5     0.10800000E 01
           6     0.99999999E-04
   2       1     0.37066000E 01
           2     0.11668000E 01
           3     0.69143000E 00
           4     0.69143000E 00
           5     0.69143000E 00
           6     0.99999999E-04

MICRO GROUP ENERGY CUTOFFS
 0.821[0000E 06  0.55300000E 04  0.25000000E 04  0.10000000E 04  0.53200000E 00  ).

MACRO GROUP ENERGY CUTOFFS
 0.25000000E 04  0.10000000E 04  0.53200000E 00  0.

FAST PCMG,BY GROUP = MATERIAL    1
 0.36353000E 00  0.23800000E 03  0.62838600E 00  0.16000000E 02
 0.55468000E 00  0.23800000E 03  0.43145900E 00  0.16000000E 02
 0.62289999E 00  0.23800000E 03  0.33415000E 00  0.16000000E 02
 0.10000000E 01  0.23800000E 03  0.10000000E 01  0.16000000E 02
 0.10000000E 01  0.23800000E 03  0.10000000E 01  0.16000000E 02

FAST PCMG,BY GROUP = MATERIAL    2
 0.70130000E 00  0.10080000E 01  0.29720000E 00  0.16000000E 02
 0.83570000E 00  0.10080000E 01  0.16430000E 00  0.16000000E 02
 0.91630000E 00  0.10080000E 01  0.83319999E-01  0.16000000E 02
 0.91630000E 00  0.10080000E 01  0.83319999E-01  0.16000000E 02
 0.91630000E 00  0.10080000E 01  0.83319999E-01  0.16000000E 02

THERMAL PCMG, MATERIAL    1
 0.              0.              0.50000000E 00

THERMAL PCMG, MATERIAL    2
 0.              0.              0.50000000E 00

TOTAL NUMBER OF DATA WORDS ON TAPE     165 PLUS 36 CONSTANTS
```

RESONANCE INPUT

GR	IRM	Q
2	2	1

RES MAT AT MASS(AM)
1 240.000

THE FOLLOWING GROUPS HAVE RESONANCE IN THIS PROBLEM
4 5

RESONANCE PARAMETER TABLE 3

UNRESOLVED PARAMETERS

NE0/MU6	E1/MU7	E2/MU8	E3/MU9	MU1/MU10	MU2/MU11	MU3/MU12	MU4/MU13	MU5/MU14
55	0.25000E 04	0.10000E 04	0.53200E 00	0.20300E-02	0.25900E-01	0.	0.	0.20600E 02
0.10000E 01	0.90000E 01	0.99582E 00	0.13351E 07	0.36308E 07	0.74000E-06	0.14272E-02	0.32271E 00	0.

RESOLVED PARAMETERS

E0	GAMMA N	GAMMA GAMMA	GAMMA FISS	G
0.99700E 03	0.40000E 00	0.36000E-01	0.	0.10000E 01
0.98500E 03	0.10000E-02	0.25900E-01	0.	0.10000E 01
0.96200E 03	0.19000E 00	0.27000E-01	0.	0.10000E 01
0.94200E 03	0.19500E 00	0.28000E-01	0.	0.10000E 01
0.93000E 03	0.37000E-01	0.25900E-01	0.	0.10000E 01
0.91100E 03	0.90000E-01	0.42000E-01	0.	0.10000E 01
0.89800E 03	0.13000E-02	0.25900E-01	0.	0.10000E 01
0.86900E 03	0.22000E-02	0.25900E-01	0.	0.10000E 01
0.86000E 03	0.60000E-01	0.25900E-01	0.	0.10000E 01
0.85600E 03	0.13000E 00	0.25900E-01	0.	0.10000E 01
0.82600E 03	0.60000E-01	0.37000E-01	0.	0.10000E 01
0.79300E 03	0.11000E-01	0.25900E-01	0.	0.10000E 01
0.78300E 03	0.30000E-02	0.25900E-01	0.	0.10000E 01
0.76700E 03	0.90000E-02	0.25900E-01	0.	0.10000E 01
0.73300E 03	0.42500E-02	0.25900E-01	0.	0.10000E 01
0.72400E 03	0.14700E-01	0.25900E-01	0.	0.10000E 01
0.71100E 03	0.17000E-01	0.50000E-01	0.	0.10000E 01
0.69600E 03	0.53000E-01	0.28000E-01	0.	0.10000E 01
0.68100E 03	0.13000E-02	0.25900E-01	0.	0.10000E 01
0.66300E 03	0.12500E 00	0.29000E-01	0.	0.10000E 01
0.63000E 03	0.90000E-02	0.25900E-01	0.	0.10000E 01
0.62200E 03	0.39000E-01	0.27000E-01	0.	0.10000E 01
0.60600E 03	0.60000E-03	0.25900E-01	0.	0.10000E 01
0.59700E 03	0.66000E-01	0.26000E-01	0.	0.10000E 01
0.58200E 03	0.42000E-01	0.26000E-01	0.	0.10000E 01
0.55800E 03	0.10000E-02	0.25900E-01	0.	0.10000E 01
0.53700E 03	0.54000E-01	0.27000E-01	0.	0.10000E 01
0.52000E 03	0.28500E-01	0.25900E-01	0.	0.10000E 01
0.49100E 03	0.10000E-02	0.25900E-01	0.	0.10000E 01
0.48000E 03	0.45000E-02	0.70000E-01	0.	0.10000E 01
0.46500E 03	0.70000E-02	0.32000E-01	0.	0.10000E 01
0.45600E 03	0.70000E-03	0.25900E-01	0.	0.10000E 01
0.43600E 03	0.14000E-01	0.28000E-01	0.	0.10000E 01
0.41200E 03	0.17000E-01	0.24000E-01	0.	0.10000E 01
0.39900E 03	0.10000E-01	0.56000E-01	0.	0.10000E 01
0.37800E 03	0.15000E-02	0.25900E-01	0.	0.10000E 01

0.34900E 03	0.45000E-01	0.23000E-01	0.	0.10000E 01
0.31250E 03	0.10000E-02	0.25900E-01	0.	0.10000E 01
0.29200E 03	0.19000E-01	0.34000E-01	0.	0.10000E 01
0.27450E 03	0.27000E-01	0.25000E-01	0.	0.10000E 01
0.26400E 03	0.23000E-03	0.25900E-01	0.	0.10000E 01
0.23800E 03	0.32000E-01	0.24000E-01	0.	0.10000E 01
0.20910E 03	0.50000E-01	0.30000E-01	0.	0.10000E 01
0.19000E 03	0.13700E 00	0.28000E-01	0.	0.10000E 01
0.16570E 03	0.34000E-02	0.25900E-01	0.	0.10000E 01
0.14590E 03	0.81000E-03	0.25900E-01	0.	0.10000E 01
0.11700E 03	0.17600E-01	0.21000E-01	0.	0.10000E 01
0.10280E 03	0.72000E-01	0.22000E-01	0.	0.10000E 01
0.90000E 02	0.80000E-04	0.25900E-01	0.	0.10000E 01
0.81100E 02	0.21000E-02	0.25900E-01	0.	0.10000E 01
0.66300E 02	0.23000E-01	0.23000E-01	0.	0.10000E 01
0.36800E 02	0.33000E-01	0.30000E-01	0.	0.10000E 01
0.21000E 02	0.90000E-02	0.29000E-01	0.	0.10000E 01
0.10200E 02	0.14000E-05	0.25900E-01	0.	0.10000E 01
0.66800E 01	0.14800E-02	0.27000E-01	0.	0.10000E 01

ISOTOPE	RES TABLE	GMAXMAX(I)	GMINMIN(I)
392	3	4	5
16	0	4	5

ODD RECORD CROSS SECTIONS

ISOTOPE	RES GROUP	SIGMA C	SIGMA S	SIGMA F	NU SIGMA F
392	4	0.	0.	C.	0.
	5	0.	0.	C.	0.
16	4	0.	0.37000E 01	C.	0.
	5	0.	0.37000E 01	C.	0.

RES MAT	ISOTOPE	TEMP	NI	GMMAX	GMMIN	AT MASS(AI)	RM
1		293.00		4	5		2
	392		0.22290E-01			0.23800E 03	
	16		0.46860E-01			0.16000E 02	

THERE ARE 340 RESONANCE DATA WORDS ON S.SUB4

SOURCE GENERATOR INFORMATION

STARTING ENERGY	NO OF EXEMPT REGIONS
0.10000000E 07	1

EXEMPT REGIONS

1

ZMAX	ZMIN	RADMAX	RADMIN
11.000	1.800	1.533	0.533

BORN STORAGE IV PARTICLES

N	X	Y	Z	ALPHA	BETA	GAMMA	PTIME	SQRT(E)	U	V	W

1	2,452	1,136	6,328	0,074	0,150	-0,986	0,000	1000,000	1	2	2
1	3,093	3,616	6,823	-0,259	-0,149	0,954	0,000	1000,000	1	2	2
1	3,208	2,002	8,770	0,924	-0,379	-0,040	0,000	1000,000	1	2	2
1	3,494	2,666	8,039	0,253	-0,394	0,884	0,000	1000,000	1	2	2
1	2,351	3,820	1,804	0,498	0,861	0,104	0,000	1000,000	1	2	2
1	1,657	1,510	4,306	-0,476	-0,170	-0,863	0,000	1000,000	1	2	2
1	2,934	3,059	5,569	-0,831	-0,472	-0,296	0,000	1000,000	1	2	2
1	2,724	3,117	3,981	-0,248	0,490	-0,836	0,000	1000,000	1	2	2
1	3,652	1,547	6,567	0,680	-0,598	-0,425	0,000	1000,000	1	2	2
1	2,837	1,880	3,564	-0,709	-0,618	-0,340	0,000	1000,000	1	2	2
1	2,020	2,186	3,899	-0,040	0,937	-0,346	0,000	1000,000	1	2	2
1	2,827	3,603	8,472	0,880	0,097	0,465	0,000	1000,000	1	2	2
1	1,159	2,622	10,758	-0,790	0,455	0,412	0,000	1000,000	1	2	2
1	1,989	3,521	10,013	0,349	-0,041	-0,936	0,000	1000,000	1	2	2
1	1,875	2,680	10,071	-0,308	0,845	0,437	0,000	1000,000	1	2	2
1	2,818	1,150	4,897	-0,236	-0,321	0,917	0,000	1000,000	1	2	2
1	3,207	3,574	5,977	0,768	0,630	0,116	0,000	1000,000	1	2	2
1	3,105	2,760	2,716	-0,193	-0,862	-0,468	0,000	1000,000	1	2	2
1	2,996	2,848	5,780	-0,249	-0,820	0,516	0,000	1000,000	1	2	2
1	3,418	2,146	2,771	0,018	0,703	-0,711	0,000	1000,000	1	2	2
1	3,949	2,548	2,465	0,306	0,824	-0,253	0,000	1000,000	1	2	2
1	3,331	2,332	5,585	0,416	-0,740	0,528	0,000	1000,000	1	2	2
1	2,139	1,101	7,883	0,632	-0,307	0,712	0,000	1000,000	1	2	2
1	2,171	1,890	8,779	-0,898	0,202	-0,390	0,000	1000,000	1	2	2
1	1,162	2,699	6,779	-0,714	-0,034	0,699	0,000	1000,000	1	2	2
1	3,564	1,814	1,834	-0,502	-0,003	-0,865	0,000	1000,000	1	2	2
1	2,125	3,605	5,835	-0,772	0,234	0,592	0,000	1000,000	1	2	2
1	2,140	3,151	1,587	0,677	-0,122	0,726	0,000	1000,000	1	2	2
1	3,445	2,752	5,680	-0,232	0,837	-0,496	0,000	1000,000	1	2	2
1	4,001	2,444	4,779	-0,072	-0,245	-0,967	0,000	1000,000	1	2	2
1	1,937	3,330	3,023	0,110	-0,782	0,613	0,000	1000,000	1	2	2
1	2,668	3,754	8,072	-0,788	-0,615	-0,012	0,000	1000,000	1	2	2
1	1,462	1,876	3,908	0,357	-0,891	-0,280	0,000	1000,000	1	2	2
1	2,314	2,039	10,148	-0,758	-0,073	-0,648	0,000	1000,000	1	2	2
1	1,416	2,412	8,068	-0,848	0,924	0,083	0,000	1000,000	1	2	2
1	3,078	1,213	8,774	-0,512	0,778	-0,364	0,000	1000,000	1	2	2
1	1,066	2,639	7,216	-0,234	0,014	-0,972	0,000	1000,000	1	2	2
1	3,378	2,165	1,302	-0,582	0,710	-0,396	0,000	1000,000	1	2	2
1	2,850	2,990	2,257	0,997	-0,037	-0,072	0,000	1000,000	1	2	2
1	1,468	1,465	5,181	0,909	-0,411	0,076	0,000	1000,000	1	2	2
1	1,839	1,465	10,432	-0,369	0,924	-0,102	0,000	1000,000	1	2	2
1	2,745	1,042	4,779	0,149	-0,365	0,919	0,000	1000,000	1	2	2
1	1,278	2,256	6,704	-0,729	-0,624	0,283	0,000	1000,000	1	2	2
1	2,867	3,817	6,261	0,286	-0,190	-0,939	0,000	1000,000	1	2	2
1	3,715	2,206	10,909	0,903	-0,371	-0,215	0,000	1000,000	1	2	2
1	2,355	1,548	4,859	0,532	0,245	-0,810	0,000	1000,000	1	2	2
1	1,869	2,592	6,757	0,600	-0,789	0,130	0,000	1000,000	1	2	2
1	2,914	1,830	8,194	0,139	0,057	-0,989	0,000	1000,000	1	2	2
1	3,287	1,493	5,793	-0,492	-0,716	-0,495	0,000	1000,000	1	2	2
1	3,217	3,250	1,671	0,599	0,126	0,791	0,000	1000,000	1	2	2
1	1,617	3,067	2,705	-0,369	0,759	-0,537	0,000	1000,000	1	2	2
1	2,162	3,307	9,357	0,443	-0,417	-0,794	0,000	1000,000	1	2	2
1	2,832	3,739	7,851	0,227	-0,132	0,965	0,000	1000,000	1	2	2
1	3,386	1,969	3,612	0,049	-0,609	0,792	0,000	1000,000	1	2	2
1	1,471	3,152	4,578	0,728	-0,201	0,656	0,000	1000,000	1	2	2
1	3,264	1,663	6,792	-0,890	0,336	0,307	0,000	1000,000	1	2	2
1	3,185	2,134	5,252	0,036	0,788	0,614	0,000	1000,000	1	2	2
1	1,765	3,140	2,999	-0,223	0,414	0,883	0,000	1000,000	1	2	2
1	3,183	1,943	10,960	0,798	-0,991	-0,123	0,000	1000,000	1	2	2
1	2,128	1,332	5,553	-0,884	-0,340	-0,320	0,000	1000,000	1	2	2

1	2,802	1,827	5,900	0,258	-0,904	-0,341	0,000	1000,000	1	2	2
1	2,404	1,735	3,786	0,823	0,437	-0,364	0,000	1000,000	1	2	2
1	1,068	2,768	7,857	-0,343	0,805	-0,485	0,000	1000,000	1	2	2
1	2,159	3,557	2,362	-0,770	0,606	0,200	0,000	1000,000	1	2	2
1	3,385	1,259	8,314	-0,505	-0,863	0,014	0,000	1000,000	1	2	2
1	1,957	2,943	4,599	-0,172	0,052	0,984	0,000	1000,000	1	2	2
1	2,308	3,318	5,226	0,916	-0,073	0,394	0,000	1000,000	1	2	2
1	3,741	2,030	6,585	-0,127	-0,304	0,944	0,000	1000,000	1	2	2
1	2,913	3,269	1,003	-0,583	-0,487	0,650	0,000	1000,000	1	2	2
1	3,508	3,596	7,733	0,575	0,689	0,442	0,000	1000,000	1	2	2
1	3,523	2,327	7,116	-0,517	0,591	-0,618	0,000	1000,000	1	2	2
1	2,714	1,033	1,381	-0,330	-0,143	-0,933	0,000	1000,000	1	2	2
1	2,484	1,561	5,134	0,582	-0,654	0,484	0,000	1000,000	1	2	2
1	1,138	2,014	10,932	0,013	0,987	0,160	0,000	1000,000	1	2	2
1	2,438	3,994	8,596	-0,961	0,010	0,275	0,000	1000,000	1	2	2
1	3,957	2,415	5,805	-0,567	0,080	-0,820	0,000	1000,000	1	2	2
1	3,088	1,323	3,199	-1,000	0,018	0,012	0,000	1000,000	1	2	2
1	1,467	1,870	9,477	-0,284	0,899	-0,334	0,000	1000,000	1	2	2
1	1,624	2,257	8,567	-0,494	-0,831	0,256	0,000	1000,000	1	2	2
1	1,528	3,404	1,276	0,541	0,477	-0,693	0,000	1000,000	1	2	2
1	3,297	3,377	9,577	-0,061	0,572	0,818	0,000	1000,000	1	2	2
1	2,014	1,509	6,353	0,234	0,160	-0,959	0,000	1000,000	1	2	2
1	2,890	1,816	9,222	0,371	0,229	-0,900	0,000	1000,000	1	2	2
1	1,523	2,996	6,968	-0,293	0,921	-0,256	0,000	1000,000	1	2	2
1	2,062	3,314	4,567	0,783	-0,597	-0,178	0,000	1000,000	1	2	2
1	2,791	3,559	1,707	0,376	0,244	0,894	0,000	1000,000	1	2	2
1	1,626	1,707	2,104	-0,838	-0,538	0,093	0,000	1000,000	1	2	2
1	1,525	2,341	4,055	0,236	0,500	-0,833	0,000	1000,000	1	2	2
1	1,733	1,548	1,993	0,459	-0,277	0,844	0,000	1000,000	1	2	2
1	3,025	3,584	7,800	-0,629	0,027	-0,777	0,000	1000,000	1	2	2
1	2,114	1,963	7,517	0,914	0,336	0,228	0,000	1000,000	1	2	2
1	3,863	2,515	7,786	0,992	-0,029	0,123	0,000	1000,000	1	2	2
1	3,249	1,813	5,692	0,163	-0,063	0,985	0,000	1000,000	1	2	2
1	3,214	3,393	8,854	-0,609	0,477	-0,634	0,000	1000,000	1	2	2
1	2,950	3,065	5,767	0,700	-0,614	-0,365	0,000	1000,000	1	2	2
1	2,482	1,993	2,750	0,881	0,434	0,190	0,000	1000,000	1	2	2
1	3,461	1,753	1,633	-0,863	0,092	-0,497	0,000	1000,000	1	2	2
1	1,268	2,753	9,836	0,956	-0,069	0,285	0,000	1000,000	1	2	2
1	3,606	1,993	10,750	0,908	-0,085	-0,411	0,000	1000,000	1	2	2
1	3,279	1,247	10,219	-0,200	-0,781	0,592	0,000	1000,000	1	2	2
1	3,447	3,663	5,355	-0,120	0,308	-0,944	0,000	1000,000	1	2	2
1	2,287	1,208	5,671	-0,876	-0,406	0,260	0,000	1000,000	1	2	2
1	1,411	2,311	5,863	-0,815	0,506	-0,281	0,000	1000,000	1	2	2
1	3,500	3,209	5,274	-0,061	0,223	0,973	0,000	1000,000	1	2	2
1	3,581	1,655	4,151	-0,835	0,361	0,415	0,000	1000,000	1	2	2
1	2,043	3,828	8,909	0,080	0,995	-0,054	0,000	1000,000	1	2	2
1	3,623	1,571	4,273	0,831	-0,533	-0,160	0,000	1000,000	1	2	2
1	2,582	1,413	1,344	0,500	-0,549	-0,670	0,000	1000,000	1	2	2
1	1,992	2,471	2,042	-0,020	-0,358	0,934	0,000	1000,000	1	2	2
1	2,216	1,837	10,008	0,383	-0,323	-0,865	0,000	1000,000	1	2	2
1	2,598	3,441	8,035	-0,342	-0,438	-0,831	0,000	1000,000	1	2	2
1	3,207	3,905	5,857	0,736	-0,335	-0,589	0,000	1000,000	1	2	2
1	2,355	3,706	1,855	0,849	-0,380	-0,368	0,000	1000,000	1	2	2
1	1,388	2,617	3,479	-0,102	-0,306	-0,946	0,000	1000,000	1	2	2
1	1,480	3,148	3,488	-0,450	0,675	0,585	0,000	1000,000	1	2	2
1	2,125	3,282	9,286	-0,448	-0,587	-0,674	0,000	1000,000	1	2	2
1	3,741	2,758	1,371	-0,811	0,546	-0,212	0,000	1000,000	1	2	2
1	3,499	2,026	10,506	0,117	-0,932	0,344	0,000	1000,000	1	2	2
1	1,788	3,770	10,306	0,001	-0,612	-0,790	0,000	1000,000	1	2	2
1	1,416	1,537	7,264	-0,588	0,462	0,664	0,000	1000,000	1	2	2

1	1,840	2,481	4,190	-0,888	0,440	-0,133	0,000	1000,000	1	2	2
1	3,753	1,839	1,125	0,972	-0,201	0,125	0,000	1000,000	1	2	2
1	3,356	1,523	6,006	-0,528	0,333	-0,781	0,000	1000,000	1	2	2
1	1,228	2,132	8,862	-0,168	0,981	0,096	0,000	1000,000	1	2	2
1	2,166	3,796	1,169	0,294	-0,313	-0,903	0,000	1000,000	1	2	2
1	1,849	1,472	8,390	-0,854	-0,502	-0,139	0,000	1000,000	1	2	2
1	2,501	1,366	3,019	0,728	-0,573	-0,376	0,000	1000,000	1	2	2
1	2,395	1,775	9,337	-0,628	0,763	-0,153	0,000	1000,000	1	2	2
1	2,906	3,317	9,044	0,461	0,115	0,880	0,000	1000,000	1	2	2
1	3,025	3,529	6,708	0,099	0,096	-0,990	0,000	1000,000	1	2	2
1	2,813	3,872	7,234	-0,912	-0,078	-0,403	0,000	1000,000	1	2	2
1	3,390	2,545	7,524	0,005	-0,178	-0,984	0,000	1000,000	1	2	2
1	3,330	3,116	4,186	0,228	0,890	-0,394	0,000	1000,000	1	2	2
1	1,960	2,972	3,484	-0,121	-0,373	-0,920	0,000	1000,000	1	2	2
1	3,215	2,588	5,212	0,957	-0,198	0,213	0,000	1000,000	1	2	2
1	1,464	2,468	1,476	-0,993	0,115	0,023	0,000	1000,000	1	2	2
1	3,291	1,716	9,054	-0,919	-0,169	0,356	0,000	1000,000	1	2	2
1	3,420	3,486	10,916	-0,545	0,657	0,522	0,000	1000,000	1	2	2
1	2,867	3,559	8,576	0,477	-0,877	0,057	0,000	1000,000	1	2	2
1	1,546	3,520	4,607	0,905	-0,251	-0,344	0,000	1000,000	1	2	2
1	3,839	3,316	6,332	-0,517	0,610	-0,600	0,000	1000,000	1	2	2
1	1,622	3,702	2,483	0,728	0,682	-0,076	0,000	1000,000	1	2	2
1	1,802	3,344	10,784	0,018	0,843	-0,537	0,000	1000,000	1	2	2
1	3,154	1,723	8,249	-0,105	0,574	-0,812	0,000	1000,000	1	2	2
1	1,560	3,151	6,566	-0,942	-0,309	0,128	0,000	1000,000	1	2	2
1	1,869	1,166	10,607	-0,709	0,446	0,546	0,000	1000,000	1	2	2
1	2,963	3,731	7,854	0,041	-0,860	0,508	0,000	1000,000	1	2	2
1	2,555	1,966	1,264	-0,757	0,552	-0,349	0,000	1000,000	1	2	2
1	3,288	2,807	8,183	0,550	-0,931	0,644	0,000	1000,000	1	2	2
1	1,023	2,791	7,069	-0,454	-0,882	-0,129	0,000	1000,000	1	2	2
1	2,787	3,919	2,622	-0,666	-0,743	-0,067	0,000	1000,000	1	2	2
1	2,186	3,258	1,549	0,490	0,016	0,872	0,000	1000,000	1	2	2
1	1,703	2,085	9,727	-0,475	0,038	-0,879	0,000	1000,000	1	2	2
1	3,704	2,521	2,994	0,313	-0,235	-0,920	0,000	1000,000	1	2	2
1	1,831	1,668	2,565	-0,592	0,350	0,734	0,000	1000,000	1	2	2
1	2,809	3,617	1,763	-0,600	0,508	0,618	0,000	1000,000	1	2	2
1	3,585	2,688	4,897	-0,728	0,531	0,433	0,000	1000,000	1	2	2
1	1,411	3,073	6,697	-0,711	-0,060	0,701	0,000	1000,000	1	2	2
1	2,936	3,200	1,669	-0,458	-0,854	0,247	0,000	1000,000	1	2	2
1	3,749	3,097	1,772	0,368	-0,008	-0,930	0,000	1000,000	1	2	2

0

TIME FOR INPUT EDIT = 10 SEC
TIME FOR IV GENERATION = 64 SEC

END-OF-DATA ENCOUNTERED ON SYSTEM INPUT FILE.

238 RES INT RUN WITH ZUT 6 GP 2 REG. 12/27/65 2 HR RUN

TRIAL 1 TRIAL TIME = 161 TRIAL FLUX = 3682.283

IV PART LEFT 7020
NPICOL 3241 NWTCOL 0.32410000E 04
NPFCOL 3062 NWFCOL 0.30620000E 04

FREGION 401 PLEAK 0 PINLEAK 0. OBSNLEAK -0.
FREGION 402 PLEAK 0 PINLEAK 0. OBSNLEAK -0.
FREGION 403 PLEAK 0 PINLEAK 0. OBSNLEAK -0.

NPTHER 0 NWTHER 0.
NPIEQN 0 NWTEQN 0.
NPICEN 0 NWTCEN 0.
PGEIW 0 NGEIW 0.
PGHIW 0 NGHIW 0.
PFFCOL 0 NFFCOL 0.
PFUSED 0 NFUSED 0.

ERRBOX1 0 ERRBOX2 0 ERRBOX3 0
ON TAPE 0 IVDONE 180

REGSET	GROUP	TRIAL FLUX
1	1	0.11201183E 03
	2	0.17075760E 03
	3	0.11751116E 02
	4	0.99130408E 01
	5	0.12875184E 03
	6	0.
2	1	0.92389351E 03
	2	0.12144903E 04
	3	0.97927412E 02
	4	0.12119760E 03
	5	0.89164713E 03
	6	0.16699991E-01

RESONANCE REACTION RATES

MATERIAL	ISOTOPE	MACRO GROUP	ABSORPTION	SCATTERING	NU FISSION
1	392	1	0.	0.	0.
		2	0.95345106E-01	0.21879220E 01	0.
		3	0.97844914E 01	0.33004125E 02	0.
		4	0.	0.	0.

2 NOT RESONANT

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238 RES INT RUN WITH ZUT 6 GP 2 REG.  12/27/65     2 HR RUN

TRIAL 40   TRIAL TIME =   167   TRIAL FLUX =   3831.012

IV PART LEFT       0
NPICOL  135295   NNTCOL   0.13529500E 06
NPFCOL  128098   NNFCOL   0.12809800E 06


FREGION  401   PLEAK       0   PINLEAK   0.              OBSNLEAK  -0.
FREGION  402   PLEAK       0   PINLEAK   0.              OBSNLEAK  -0.
FREGION  403   PLEAK       0   PINLEAK   0.              OBSNLEAK  -0.

NPTHER       0   NNTHER   0.
NPTEON       0   NNTEON   0.
NPTCEN       0   NNTCEN   0.
PGEIW        0   NGEIW    0.
PGHIW        0   NGHIW    0.
PFFCOL       0   NFFCOL   0.
PFUSED       0   NFUSED   0.

ERRBOX1     8   ERRBOX2     0   ERRBOX3     0
ON TAPE     0   IVDONE    180


 REGSEI GROUP      TRIAL FLUX
    1      1     0.11966882E 03
           2     0.16841937E 03
           3     0.15417496E 02
           4     0.18771461E 02
           5     0.14067865E 03
           6     0.
    2      1     0.91412354E 03
           2     0.13231328E 04
           3     0.10196539E 03
           4     0.10495306E 03
           5     0.92191699E 03
           6     0.14599159E-01

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                                                   RESONANCE REACTION RATES
MATERIAL   ISOTOPE  MACRO GROUP     ABSORPTION       SCATTERING        NU FISSION
    1        392          1       0.                0.                0.
                          2       0.45256329E-00    0.52411880E 01    0.
 4.10                     3       0.81194229E 01    0.33300293E 02    0.
                          4       0.                0.                0.
    2     NOT RESONANT
```

238 RES INT RUN WITH ZUT 6 GP 2 REG. 12/27/65 2 HR RUN

TOTAL FLUX OVER ALL H,G = 0.15203102E 06

REGSET	GROUP	TOTAL FLUX	PER CENT ERR
1	1	0.45680670E 04	0.34400877E 01
	2	0.69917183E 04	0.38091536E 01
	3	0.55815855E 03	0.96988446E 01
	4	0.61279987E 03	0.10393133E 02
	5	0.50919216E 04	0.55523579E 01
	6	0.21999998E-02	0.52531770E 02
2	1	0.36585230E 05	0.13187177E 01
	2	0.50046504E 05	0.31554768E 01
	3	0.42052467E 04	0.55993956E 01
	4	0.48250009E 04	0.54339151E 01
	5	0.38511310E 05	0.52200828E 01
	6	0.67398413E 00	0.61316316E 01

* *

RESONANCE REACTION RATE TOTALS

MATERIAL	ISOTOPE	MACRO GROUP	ABSORPTION	PERC ERR	SCATTERING	PERC ERR	NU FISSION	PERC ERR
1	392	1	0.	0.	0.	0.	0.	0.
		2	0.15248050E 02	18.72	0.16358618E 03	11.91	0.	0.
		3	0.26812495E 03	9.28	0.11573071E 04	5.93	0.	0.
		4	0.	0.	0.	0.	0.	0.
2	NOT RESONANT							