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BMC-I: THE BATTELLE MONTE CARLO CODE

D. H. Thomsen
T. M. Traver

June 1970

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BMC-I: The Battelle Monte Carlo Code

by

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ABSTRACT

The Battelle Monte Carlo code (BMC) is a general purpose neutronic Monte Carlo code written in Fortran-V for use on the UNIVAC-1108 computer. The code was developed primarily for use in solving some of the more complex problems encountered in predicting the neutronic characteristics of thermal reactors. It is flexible as to the size and type of problem to be solved and in the cross section and geometry detail which may be used. The neutron fluxes and reaction rates in the energy range from 0 to 10 MeV are calculated in one, two or three space dimensions. Statistical confidence limits are assigned to the results. The cross sections utilized in the BMC code are processed from data in the ENDF/B format.

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BMC-I: The Battelle Monte Carlo Code

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I. INTRODUCTIONA. PHILOSOPHY AND HISTORY OF THE DEVELOPMENT

There are many neutronics problems encountered in reactor design studies, shielding, dosimetry studies, etc., which can not be adequately solved using the presently developed diffusion and/or transport theory methods. Moreover, experimental solutions to certain problems are either not feasible or practical. The Monte Carlo method is a powerful tool for solving these classes of problems because it provides a reliable calculational technique with a minimum of approximations.

The history of the development of the BMC code starts with the RBU code⁽¹⁾. While the RBU code was developed for performing accurate calculation of the burnup history of a reactor, it became evident that the inherent potential of the RBU code was in the Monte Carlo portion. Subsequent development concentrated on the Monte Carlo portion of the RBU code which became known as the RBU Monte Carlo Code⁽²⁾. The advent of the UNIVAC-1107 and 1108 computers at the Hanford Project required a decision on code conversion, because the RBU Monte Carlo Code was written in machine language for use on the IBM-7090 machine. It was decided to convert the code to FORTRAN for use on the UNIVAC machine and to incorporate some desired improvements. The conversion was completed and the resulting code renamed the BATTELLE MONTE CARLO CODE (BMC).

B. GENERAL FEATURES OF THE BMC CODE

The Battelle Monte Carlo Code (BMC) is a general purpose Monte Carlo code and is primarily designed for use in calculating nuclear parameters associated with thermal reactors.

The neutron flux, isotopic reaction rates, group-averaged cross sections, and leakages are calculated in three-space dimensions and over the energy range from 0 to 10 MeV. The reaction types considered are fission, capture, inelastic and $n-2n$ scatter, elastic scattering with isotropic or anisotropic angular distributions, and thermal elastic scattering using the ideal gas scattering model. The energy distribution of the neutrons are continuous; however, the cross sections are averaged over up to 2000 microscopic energy groups. There are special provisions so that resonance cross sections can be calculated by the code for each neutron energy using the Doppler-broadened Breit-Weigner single-level formulae.

The isotopic material cross sections are processed from the ENDF/B format⁽³⁾. The data can be provided either from the Battelle Master Library, from the ENDF/B Library, or from any source which can be put into the ENDF/B format.

The BMC code geometry routines can handle any region which can be enclosed by a set of boundaries of the general form

$$A(x-x_0)^2 + B(y-y_0)^2 + C(z-z_0)^2 - K = 0.$$

Provisions are included for seven special forms of the above equations including planes, cylinders and spheres. Also, special region geometry

routines may be used; for example, one for a rectangular lattice of clad fuel rods has already been coded.

Statistics for the values calculated are obtained by making a series of calculations on equal sized sets of neutron histories called "batches" and averaging the results from each batch. The initial space-velocity-angle coordinates for each neutron of a batch are either picked from a random source distribution or from the fission particles produced by the previous batch.

Neutron absorption is accounted for by reducing the weight of the neutrons at each collision. When the weight is reduced sufficiently the neutrons are terminated by Russian Roulette. A special method of

termination is available for thermal neutrons; that is, only part of the neutrons entering the thermal energy range are followed. The results are tallied into a second tally block. The results of the second tally set are then multiplied by the correct ratio and added to the complete tally set. This method allows a control on the relative amount of time spent calculating in the thermal range to that spent in the epithermal range.

The results calculated by the Monte Carlo routine are collapsed from microscopic to macroscopic energy group detail by the output routine. The values obtained, along with their standard deviations, are the region-energy fluxes, the isotopic reaction rates and flux-averaged cross sections, cell reaction rates, and leakages. The fluxes can also be plotted along with their error bars.

The BMC code is coded in FORTRAN V (95%) and SLEUTH (5%) for use on the UNIVAC 1108 computer system. There are two cross section processing support codes, BMCLIB and LIBR, which are written in FORTRAN V.

THE BMC CODE PACKAGE

The BMC code package is made up of two portions: a portion which prepares the cross-section and reaction data library, and a Monte Carlo portion as depicted in Figure 1. The BMCLIB code takes isotopic or material cross sections and reaction data and prepares a cumulative cross section library tape (CCT). The data can be taken from cards or from a library tape which is written in the ENDF/B format. The CCT library tape can be saved between problems. New isotopes can be added to an old CCT tape. Minor changes to the CCT tape can be made using the LIBR code.

The BMC code is made up of three segments or overlays comprised of the input routines, the Monte Carlo routines and the output routines. The input routines use card input and the CCT library tape to prepare the data needed by the Monte Carlo and output routines. The Monte Carlo routines are then loaded and the Monte Carlo calculation performed. The output routines are then used to process the print, and/or plot the Monte Carlo output. Statistics are also calculated for the various output values.

Section II of the report describes the theories and techniques which are used in the Monte Carlo calculation. In Section III, some of the major Monte Carlo routines are described. The output routines, the problem input, and a special application of the BMC code for first-flight collision probability calculations are discussed in the following sections. The last two sections deal with the BMCLIB code and the LIBR code.

II. THE BMC MONTE CARLO ROUTINES

A flow diagram is shown in Figure 2 which describes the sequence of events in following the histories of a "batch" of neutrons. This is the path followed if importance weighting is not being used. The BMC code uses a beam type instead of a particle type Monte Carlo technique. It starts each tract with a beam. Carried along with the beam is one or more neutrons. The beam is used for tallying the flux, leakage, and resonance reaction rates. The neutrons are used for producing collisions. The beam strength is reduced by the negative exponential of the mean free path traveled between each boundary. A beam is followed until all of the neutrons have either collided or leaked. It can then be terminated either by leakage or by Russian Roulette.

A neutron beam is obtained by subroutine GETN. The neutron is then located and the distance to the next boundary is calculated by subroutine LÖCPRT which also supplies the material and tally region numbers. Subroutine MFPTH obtains the mean free path for this material. If a resonance calculation is needed, the resonance routines are called. The distance-to-collision is compared to the number of mean free paths to the boundary. If a collision occurs, subroutine COLLN is called to perform the collision mechanics. COLLN checks to see if a fission occurred and stores the fission neutrons in the fission bank. It uses the probability of nonabsorption to determine if a new scattered neutron will be produced. It then

FIGURE 1
THE BMC CODE PACKAGE

FIGURE 2

GENERAL FLOW DIAGRAM OF THE MONTE CARLO CODE
FOR NEUTRONS. IMPORTANCE WEIGHTING NOT BEING USED,

determines the scattering nuclide and scattering event. A new direction and velocity are calculated and the scattered neutrons are stored in born storage.

Subroutine EXILY is called to tally the flux, leakage, and resonance tallies. The beam strength is adjusted for the distance traveled in the region and tested to determine if it should be followed into the next region. The beam is then advanced to the next boundary where it crosses into the next region or is reflected. Subroutine LQCPRT is called again and the process is repeated. Note that each beam is followed until it is terminated by leakage or by Russian Roulette.

The flow of the calculation is altered slightly if region importance weighting is being used. The main difference is that the calculation-of-distance to collision is made after the mean free path is obtained and must be made again in each region.

A. TREATMENT OF NEUTRON ABSORPTION

Neutron absorption is accounted for by reducing the weight of a neutron by the absorption probability at each collision. The reduction is done by multiplying the weight before the collision by the non-absorption probability (Pna) for the collision. This process can be written as $W_t' = W_t \times P_{na}$. The nonabsorption probability can be defined as the total scattering neutron production divided by the total cross section or

$$P_{na} = \frac{\sum_s + \sum_{in} + 2\sum_{n-2n}}{\sum_T}$$

The extra neutron from n,2n scattering is treated as negative absorption.

It is seen that no matter how many collisions the neutron suffers, the weight will never become zero. There has to be some way of killing the neutrons with low weight while conserving the total neutron weight.

Terminating the Neutron History

Several obvious ways of terminating a neutron history are: let it escape, let it slow down past a lower cut-off energy, or to kill it when the weight is reduced below a certain minimum. The second method works fine for calculations of nonthermal parameters. The last method works for some problems but it does not conserve the total neutron absorption. The BMC code can use any of the above three methods to kill a neutron or it can use two other methods which are designed to statistically conserve the absorption. The two other methods are Russian Roulette and 'weight ratioing'.

Russian Roulette. Russian Roulette, as the name implies, uses chance to determine if a neutron survives. Given a minimum weight (Wt_{min}) and a survival weight (Wt_s), Russian Roulette is performed by picking a random number (ξ) between 0 and 1 and if the ratio of the neutron weight to the survival weight is greater than the random number, then the neutron is given the survival weight. Russian Roulette is only played if the weight has been reduced below the minimum weight. This can be written as

$$\text{if } Wt < Wt_{min} \text{ and if } \xi \leq \frac{Wt}{Wt_s} \text{ then set } Wt = Wt_s$$

but if $\xi > \frac{Wt}{Wt_s}$ then set $Wt = 0$ and kill the neutron. On the average with a large number of samples, the weight will be preserved by this method.

The RBU Monte Carlo code and some other codes do not use the weight reduction method to account for absorption; instead they sample a random number against the nonabsorption probability at each collision. This is the same as setting $W_{t_{min}}$ and W_{t_s} both equal to unity.

Weight Ratioing. For very thermal systems, it may take a very large number of collisions to kill neutrons by using Russian Roulette. This is because the weight reduction at each collision is very small (≈ 1). Often this results in obtaining answers concerning thermal parameters which are statistically very accurate while the accuracy of the epithermal parameters is very poor. A method called "weight ratioing" was devised to allow control of the time spent in the thermal compared to the epithermal calculations.

Weight ratioing uses two flux tally sets: one tally set for neutrons slowing down and one for neutrons which have been slowed down past an entry energy (E_{th}). The neutrons reaching the thermal tally range (second tally set) can have their energy increased above E_{th} but the results are still tallied into the thermal tally set.

The way weight ratioing works is to tally the results of a neutron history into tally set 1 until it is slowed down past the energy E_{th} . Russian Roulette is then played with each neutron so that R_{th} neutrons entering the thermal tally set are rejected for each one entering the thermal tally range. The ones which survive are followed and the results tallied into a second tally set. The final tallies are the sums of the values from the first tally set plus the values from the second tally set times a weighting. This ratio is the weight entering the thermal tally range divided by the weight lost by neutrons followed in the thermal tally

range or approximately $1/(1-R_{th})$. This method statistically conserves the neutron absorption and allows a way of controlling the statistics obtained for epi-thermal parameters relative to those obtained for thermal parameters.

This method has not been thoroughly tested to determine the correct values of the different parameters involved or when it should or should not be used. However, it has been observed that a value for E_{th} slightly higher than 6 kT works fairly well. Figure 3 is a graph which is useful in trying to decide the values to use for R_{th} and for Russian Roulette. It shows the number of collisions needed on the average to produce one absorption using Russian Roulette as a function of the average nonabsorption probability.

FIGURE 3

B. THE NEUTRON FLUX AND THE NEUTRON BEAM

The neutron flux at energy E integrated over volume and energy is the total neutron track length in the volume, or

$$\phi_{if} = \int_{V_f} \int_{E_i} \phi(E, V) dE dV = \sum_{n=1}^N \ell_n Wt_n$$

where V_f is the volume of region f, ℓ_n is the track length and Wt_n is the weight of the n th neutron in region f and energy group i. Note that the reaction rate for the k th event is $R_{i,f,k} = \Sigma_{i,f,k} \phi_{if}$ where $\Sigma_{i,f,k}$ is the macroscopic cross section.

Instead of tallying the track length of individual neutrons the BMC code tallies the estimated track length for a beam of neutrons going in the same direction as the individual neutron. The flux contribution to a region and energy group can be written as

$$\phi_{if} = \sum_{n=1}^N \lambda_{if} \pi_n (1 - e^{-S_{bn}/\lambda_{if}}) \text{ where } \lambda_{if} \text{ is the mean free path}$$

in the i th energy group of region f, S_{bn} is the distance to the outside boundary of the region for the n th beam, and π_n is the beam strength at the beginning of the n th flight of the beam. The beam strength is set equal to the weight of the neutron at the start of the neutron path. It is then diminished by the factor $e^{-S_{bn}/\lambda_{if}}$

at each boundary crossing. Note that the neutron is followed until it has a collision or leaks from the cell while the beam is followed even after the collision. If the neutron leaks then the beam is also terminated; otherwise the beam is terminated by using Russian Roulette.

C. LEAKAGE ESTIMATE

The leakage tallied by the beam crossing a boundary is just the beam strength, π , at the boundary. The leakage is tallied as a function of the larger macro-energy groups.

D. IMPORTANCE WEIGHTING

The BMC code has provisions for zone importance weighting. Zone importance weighting provides a method for spending more time in regions (zones) of greater importance while reducing the time spent in regions of low importance. When importance weighting is used the number of neutrons in the beam and the weight of each neutron is modified when the beam passes between regions of different importance.

If region 1 has importance ZWT_1 and region 2 has importance ZWT_2 then as a neutron beam of PN_1 neutrons and weight Wt_1 passes to region 2 the new weight and number of neutrons become

$$Wt_2 = \frac{ZWT_1}{ZWT_2} \cdot Wt_1$$

and

$$PN_2 = \left[\frac{ZWT_2}{ZWT_1} \cdot PN_1 + \xi \right]$$

where ξ is a random number and $\left[\right]$ means greatest integer.

E. NEUTRON BATCHES

A BMC calculation is comprised of a set of calculations called batches. Each batch contains the same number of neutron histories. The neutron histories can start as neutrons born at fission or as source neutrons put into the system using any desired source distribution. The results from the batches are then averaged to obtain the final calculated values.

F. FISSION DESCENDANT OR DIRECT SOURCE PROBLEMS

The BMC code can be used in two problem modes. One mode, the fission descendant problem, allows fission neutrons to be born as a result of collisions. These fission progeny are stored and used as starting neutrons for the next batch. The other mode, called the direct source problem, obtains all starting neutrons from the source routine. In this case, there is no need to save the fission neutrons, so the fission reaction is not sampled.

G. THE NEUTRON POSITION-DIRECTION-VELOCITY PARAMETERS

Associated with each neutron is a set of parameters which describe its position, direction, and velocity. The position is described by the position vector x, y, z where the units are in centimeters, and an index I_h which tells which geometrical region the neutron is in. The direction is defined by the three normalized direction cosines α , β , and γ . The velocity is defined by the velocity, V and an index, I_v , the microscopic energy-velocity group which the velocity V falls into. The velocity units are centimeters per micro second.

The neutron beam also has several parameters associated with it. They are the weight, W_t , the number of particles, N_p , and the beam strength (π , BPI).

H. ENERGY GROUP STRUCTURES

The energy range studied by the BMC code is from 0 to 10 ~~MV~~ or any part of this energy range. The neutron energy-velocity distribution is continuous in the BMC calculation; however, the energy dependent input is divided into energy groups. There are three different energy group structures. They are called micro-groups, macro-groups, and broad groups.

1. Micro-Groups

The micro-group is the smallest energy group structure. The cross sections and reaction probabilities are group-averaged over each micro-group. The code uses the micro-averaged values for any velocity which falls within the velocity limits of the micro-group. The fluxes are also tallied for each micro-group.

The code can use up to 2000 micro-groups for very limited problems. Most calculations with the BMC code and the older RBU code have been made using 190 energy groups. There are 56 groups below 1 ev.; above 1 eV, the groups are spaced equally in lethargy with a 1/8 lethargy width.

2. Macro-Groups

The macro-groups are the energy groups used for output. Their boundaries must coincide with those of the micro-groups but they may contain one or more micro-groups. The macro-group structure is usually picked to coincide with a group structure used by typical multi-group reactor cell codes. The leakage and resonance reaction rate tallies are made by macro-group.

3. Broad-Groups

The broad-groups are energy bands over which the Monte Carlo calculation is made. If there is more than one broad group, the cross sections from the first broad group are read in and all the neutrons for the batch are slowed down below the minimum energy of the broad group. The cross sections for the next broad group are read in and the process repeated. So far, no more than one broad group has been used; however, the capability for using more than one group is present.

If the number of broad groups is specified in the input as zero, the code will use one broad group covering the entire energy range described on the BMC cross section tape (CCT). If only a portion of the energy range described by the CCT is desired, then one broad group can be described which covers the desired energy span and the cross sections for only that range will be selected from the CCT.

CODE CHECK TALLIES

A set of values are tallied as the Monte Carlo calculation is being made which serve as code check tallies. They are a set of values which are not needed as output, but serve as a means of checking the performance and characteristics of the Monte Carlo code. The code check tallies are obtained directly from the neutron histories and are not results obtained using the flux estimator. As a note of caution, the answers obtained using the code check tallies are usually not as accurate as those obtained using the flux estimators.

The code check tallies include such things as the number of initial fission or source neutrons, the number and weight of neutrons to and from collision, the number of neutrons leaking from the system, the loss of beam strength and weight by application of Russian Roulette, and the number and weight of neutrons entering, followed in, and leaving the thermal tally range, etc.

The code check tallies are written out at the completion of each Monte Carlo batch. Average values with associated statistical errors are also written by the output routines.

III. THE MAJOR BMC MONTE CARLO ROUTINES

This section will describe the functions of each of the major routines and sections of the BMC Monte Carlo Code. The order described in Figure 1 will be followed. First the random number routines will be discussed briefly.

A. The BMC Random Number Routines

The random number routines used in the BMC code are those used in the IBM-7090 version of the OSR Monte Carlo code.⁽⁴⁾ They were converted to SLEUTH for use on the Univac-1108 computer. **They** have also been modified⁽⁵⁾ to make use of some of the features of the Univac interger arithmetic. Table I gives the names of the random number routines along with their distribution.

TABLE IRANDOM NUMBER ROUTINES USED IN THE BMC CODE

<u>Fortran Calling Statement</u>	<u>Random Number Distribution</u>
X = RNF(IDUMMY)	Uniform distribution on the interval $0 \leq x \leq 1.$
X = SRNF(IDUMMY)	Uniform distribution on the interval $-1 \leq x \leq 1.$
X = RNL0G(IDUMMY)	Exponential distribution $P(x)dx = e^{-x}dx \quad 0 \leq x \leq \infty.$
CALL AZIRN(SIN, COS)	The sine and cosine of ϕ where ϕ is uniformly distributed on the interval $0 \leq \phi < 2\pi$. A random azimuthial angle.
CALL POLRN(SIN, COS)	The sin and cosine of θ where $\cos\theta$ is uniformly distributed on the interval $-1 \leq \cos\theta \leq 1$. A random polar angle.
CALL RNCOS(ALPHA, BETA, GAMMA)	An isotropic unit vector with direction cosines ALPHA, BETA, and GAMMA.
E=RNMAX(T)	Maxwellian energy distribution around temperature T. $P(E)dE = \left(\frac{4}{\pi T^3} \right)^{1/2} E^{1/2} e^{-\frac{E}{T}} dE$
R = FISRN(IDUMMY)	Neutron velocity squared from the Watt fission spectrum. $P(R)dR = c e^{-R/T} \sinh 2 \sqrt{E_f R} dR$ Where $T = 0.965 \times 1.91322 \times 10^6$ and $E_f = 0.533 \times 1.91322 \times 10^6.$

B. GETN - ROUTINE TO GENERATE, SAVE, AND RETRIEVE NEUTRONS

The GETN subroutine is used to obtain source or fission neutrons or to retrieve collision neutrons. Other entries to GETN are used to store neutrons from fission or scattering and to store neutrons which have been slowed below the minimum energy of the broad-group.

1. Storage Blocks used by GETN

GETN uses two storage blocks: born storage and the fission bank. Born storage is used to save the neutrons born from collision or slowed down while the fission bank is used to store the neutrons from fission. The fission bank is used only for fission descendent problems. The neutron parameters which are saved in born storage are $x, y, z, \alpha, \beta, \gamma, V, I_h$, and W_t . Only the spatial parameters x, y, z , and I_h are saved in the fission bank. When a neutron is taken from the fission bank the other parameters are obtained by GETN using an isotropic distribution for the direction cosines, and a random fission distribution for the velocity. The fission distribution is presently that for uranium-235; however, other fission distributions could easily be used.

When GETN is called, born storage is checked first. If it is empty then the fission bank is checked or the source routine is called. After a neutron has been picked the velocity index I_v is obtained by subroutine IFIND and the number of particles is set to one. If the broad group and/or batch is finished, the return from GETN is via the call list.

2. The Source Routine

The source routine is used to pick the initial parameters for the neutron histories. If the calculation is a fission descendant problem, the source routine will only be used for the first batch and only the spatial parameters will be picked. For a direct source problem, all the parameters are generated by the source routine. The user can write a source routine for a particular problem or use the one built into the BMC code.

The source routine in the BMC code is very versatile. It allows the selection of the neutron spatial distribution using combinations of point, equal volume, $\cosine\left(\frac{\pi}{2} \frac{r}{R}\right)$, $J_0(2.405 r/R)$, and $(\sin\pi r)/\pi r$ distributions. The velocities can be picked using combinations of point, fission, Maxwellian, and/or slowing down distributions. The angular distribution is either isotropic or mono-directional.

If the slowing down distribution is selected, a rejection technique is used to select the velocity. Let $a = \left(\frac{A-1}{A+1}\right)^2$ and V_c be the upper cutoff velocity which the neutrons are being slowed past. Now select two random number ξ_1 and ξ_2 . Calculate $V = V_c \left[\sqrt{a} + (1-\sqrt{a}) \xi_1 \right]$ and accept V if $\xi_2 \leq \frac{1}{1-a} \left[\frac{V}{V_c} - a \frac{V_c}{V} \right]$; otherwise reject V and repeat the two steps.

C. CALCULATION OF DISTANCE TO COLLISION

The distance to collision in units of mean free paths is obtained by sampling from the distribution $P(\lambda)d\lambda = e^{-\lambda}d\lambda$.

D. LOCPRT - The BMC Geometry Routine

The geometry routine, LOCPRT, is used to locate which region a neutron-beam is in and to calculate the distance to the next boundary. A control interger (IEFN) is used to guide the calculation. If IBFN = 1 the neutron is a source, fission, or collision neutron. If IBFN = 2 the neutron-beam has just crossed the boundary into another region. If IBFN = 3 the neutron-beam has just been reflected back into the same region. LOCPRT also contains the boundary reflection routines for reflecting a neutron-beam either isotropically or with a mirror image reflection.

1. Region Definition

Regions are identified by an index I_h . For each region one specifies the number of boundaries (N_{b_h}), an index (L_{sp_h}) telling if this is a special region, the number of the material (M_h) contained in the region, and the tally region (f_h) that this region is part of.

For each of the N_{b_h} boundaries one specifies four quantities. They include the index of the boundary (I_{b_h}), a word (IGD) which contains two bits of information, and two indexes (IQ_1 and IQ_2) that tell which region to look in for the neutron if it leaves region I_h across boundary i_{b_h} . Region IQ_1 is first checked and if the neutron is not in that region a complete region search is made starting in region IQ_2 .

The two bits of information in IGD are G_b and D_b . G_b is a sign that tells which side of the boundary the neutron is on to be inside of the region. D_b gives the number of the leakage tally set to use if a neutron crosses this boundary. The way that G_b and D_b are combined in IGD is

illustrated by the equations:

$$G_b = \text{sign of IGD}, D_b = |IGD| - 1.$$

If the value of $D_b > 50$ then the neutron beam strength crossing the boundary will be tallied but the beam will not be treated as if it were leaking from the system.

The regions must be combined in a way so that a neutron can always be located. Also the regions should not overlap each other. It is not necessary to define all of space if the cell is surrounded by reflecting or leakage boundaries and all initial neutrons are inside the cell.

Special Regions

The LOCPRT routine allows the use of special regions. Special regions are regions containing geometrical designs which are to be treated by a separate geometry routine. One special geometry routine for a rectangular lattice of clad fuel rods has already been written.

LOCPRT first finds the distance to the outer boundary of the special region. The special routine is then called to calculate the distance to the nearest boundary of the special regions.

3. Boundary Definitions

There are eight types of boundaries which can presently be used by the LOCPRT routine. The type of each boundary is specified by an index I_d . Another index, IOM_b is used to determine if the neutron will cross boundary I_b or will be reflected from the boundary.

Each boundary defines a boundary function r . When $r = 0$ the neutron is on the boundary. The sign of r when the neutron is not on the boundary tells which side of the boundary the neutron is on. A point is said to be inside of a region if the product of $G_b \cdot r$ is negative for all the boundaries in the region.

Table II gives the boundary index, boundary function and description of the eight boundaries now included in BMC. It would also be easy to add other type boundaries to the code.

TABLE II;
BMC CODE BOUNDARY FUNCTIONS

<u>I_d</u>	<u>Boundary Function</u>	<u>Description</u>
1	$r \equiv x - x_0$	Plane at $x = x_0$
2	$r \equiv y - y_0$	Plane at $y = y_0$
3	$r \equiv z - z_0$	Plane at $z = z_0$
4	$r \equiv y - ax - b$	Plane on $y = ax + b$
5	$r \equiv x^2 + y^2 - R^2$	Cylinder of radius R centered on z axis
6	$r \equiv (x-x_0)^2 + (y-y_0)^2 - R^2$	Cylinder of radius R centered at x_0, y_0
7	$r \equiv x^2 + y^2 + z^2 - R^2$	Sphere of radius R centered at the origin
8	$r \equiv A(x-x_0)^2 + B(y-y_0)^2 + C(z-z_0)^2 - K$	General boundary function

Determination of G_b . Remember that for a particle to be inside of a region that the product $G_b \cdot r$ must be negative for all of the boundaries of a region. To determine which sign to use for G_b just pick a point inside the region and solve the boundary functions for the region boundaries. For spheres and cylinders, G_b is positive inside and negative outside. For the x_0 , y_0 and z_0 planes, G_b is positive for the region below the plane and negative above.

4. Sample Geometry Description

A sample of the geometry description for a first flight collision probability calculation is given in Figure 4. The boundary data and region data are also given.

FIGURE 4

EXAMPLE OF GEOMETRY DESCRIPTION
FIRST FLIGHT COLLISION PROBABILITY CALCULATION

Material

1	Nickel
2	Graphite
3	Uranium
-1	Interior Void

Let the tally region number, f_h , be the same as the region number, I_h .

BOUNDARY DESCRIPTION

I_b	I_d	I_{om_b}	Boundary Function
1	3	3*	$r \equiv Z - 0$
2	3	3	$r \equiv Z - 8$
3	3	0	$r \equiv Z - 4$
4	5	0	$r \equiv x^2 + y^2 - (0.5)^2$
5	5	0	$r \equiv x^2 + y^2 - (0.75)^2$
6	5	0	$r \equiv x^2 + y^2 - (1.75)^2$
7	5	0	$r \equiv x^2 + y^2 - (2.00)^2$
8	5	0	$r \equiv x^2 + y^2 - (3.00)^2$

$I_{om_b} = 3$ for a mirror reflecting boundary.

FIGURE 4 (continued)REGION DESCRIPTION

I_h	N_{b_h}	$L_{s_{ph}}$	M_h	f_h	IGD	I_b	Q_1	Q_2	IGD	I_b	Q_1	Q_2
1	3	0	1	1	1 1	4 2	2 0	0 0	-1	1	0	0
2	4	0	-1	2	-1 -1	4 1	1 0	0 0	1 1	5 2	3 0	4 0
3	4	0	2	3	-1 -1	5 1	2 0	0 0	1 1	6 3	5 4	0 0
4	4	0	3	4	-1 -1	5 3	2 3	0 0	1 1	6 2	5 0	0 0
5	4	0	-1	5	-1 -1	6 1	3 0	4 0	1 1	7 2	6 0	7 0
6	4	0	3	6	-1 -1	7 1	5 0	0 0	2* 1	8 3	0 7	0 0
7	4	0	2	7	-1 -1	7 3	5 6	0 0	2* 1	8 2	0 0	0 0

*Kill the neutron-beam as it crosses boundary 8 and tally the leakage into leakage tally block 1.

5. Calculation of Distance to Boundary

The distance to boundary (S_b) is calculated using the equations shown in Table III. For boundary types 5 to 8 the distance to boundary is obtained by solving the quadratic equation $f(S_b) = hS_b^2 + 2eS_b + r = 0$. Now apply Sturm's theorem to this equation to see if there are any roots between zero and infinity. The three Sturm's functions for the quadratic equation $f(S_b)$ are:

$$f(S_b)$$

$$f'(S_b) = 2hS_b + 2e$$

$$F_1 = e^2/h - r.$$

The signs for the Sturm's functions at 0 and ∞ are the signs of the following values.

S_b	$f(S_b)$	$f'(S_b)$	$F_1(S_b)$
0	r	e	$\frac{e^2}{h} - r$
∞	h	h	$\frac{e^2}{h} - r$

Use is made of these equations to reduce the number of times the square root of $e^2 - hr$ must be determined.

The boundary with the smallest $S_b > 0$ is identified as the next boundary to be crossed.

6. Boundary Reflection Routine

There are two types of reflection presently allowed in the BMC code. They are isotropic and mirror reflection. The isotropic reflection is performed by selecting a set of isotropic random cosines and requiring the signs of the new α , β and γ to be correct for a reflection back into the region. The equations for a mirror reflection are derived from the vector equation for a mirror reflection,

TABLE III
EQUATIONS FOR CALCULATING DISTANCE TO BOUNDARY

I_d Equations for distance to boundary

1 $S_b = -r/\alpha = -(x-x_0)/\alpha$

2 $S_b = -r/\beta$

3 $S_b = -r/\gamma$

4 $S_b = -r/(\beta-\alpha \cdot a)$

For boundaries 5, 6, 7, and 8 the equation becomes

$$S_b = \frac{-e + G_b \sqrt{e^2 - hr}}{h}$$

where **e** and **h** are

5 $e = \alpha x + \beta y$

$h = \alpha^2 + \beta^2$

6 $e = \alpha(x-x_0) + \beta(y-y_0)$

$h = \alpha^2 + \beta^2$

7 $e = \alpha x + \beta y + \gamma z$

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

8 $e = A\alpha (x-x_0) + B\beta(y-y_0) + C\gamma (z-z_0)$

$h = A\alpha^2 + B\beta^2 + C\gamma^2$

$$\vec{n} - \vec{n}' = 2(\vec{n} \cdot \vec{n}_b)\vec{n}_b$$

where $\vec{n} = \alpha \hat{i} + \beta \hat{j} + \gamma \hat{k}$ original direction vector

$\vec{n}' = \alpha' \hat{i} + \beta' \hat{j} + \gamma' \hat{k}$ final direction vector

and $\vec{n}_b = \frac{\partial r}{\partial x} \hat{i} + \frac{\partial r}{\partial y} \hat{j} + \frac{\partial r}{\partial z} \hat{k}$ is the normal vector to the boundary.

Table IV gives the equation for mirror reflection.

The boundaries can have a reflection albedo associated with them so that the weight and beam strength is modified at the reflection by the albedo for that boundary. The albedos are input as a function of macro energy group.

TABLE IV

EQUATIONS FOR MIRROR REFLECTION FROM A BOUNDARY

I _d	Reflection equations
1	$\alpha' = -\alpha, \beta' = \beta, \gamma' = \gamma$
2	$\alpha' = \alpha, \beta' = -\beta, \gamma' = \gamma$
3	$\alpha' = \alpha, \beta' = \beta, \gamma' = -\gamma$
4	$\alpha' = \alpha + \frac{2a(\beta - \alpha a)}{a^2 + 1}, \beta' = \beta - \frac{2(\beta - \alpha a)}{a^2 + 1}, \gamma' = \gamma$
5	$\alpha' = \alpha - 2x \frac{e}{R^2}, \beta' = \beta - 2y \frac{e}{R^2}, \gamma' = \gamma$
6	$\alpha' = \alpha - 2(x - x_0) \frac{e}{R^2}, \beta' = \beta - 2(y - y_0) \frac{e}{R^2}, \gamma' = \gamma$
7	$\alpha' = \alpha - 2x \frac{e}{R^2}, \beta' = \beta - 2y \frac{e}{R^2}, \gamma' = \gamma - 2z \frac{e}{R^2}$
8	$\ell = A(x - x_0), m = B(y - y_0), n = C(z - z_0)$
	$G = \frac{2[\alpha\ell + \beta m + \gamma n]}{[\ell^2 + m^2 + n^2]}$
	$\alpha' = \alpha - \ell G, \beta' = \beta - m G, \gamma' = \gamma - n G$

7 The Special Square Lattice Geometry Routine

The special square lattice routine is used to locate and calculate distances to boundaries for a neutron-beam in a rectangular array of clad fuel rods. The cladding is assumed to all be made of the same material and the flux in the cladding is all tallied into the same tally region. The materials and tally regions for the fuels can all be different.

Up to four distances to boundary are calculated at a time thus saving two square root calculations for a neutron going from the moderator in through the cladding, fuel, and out through the cladding.

E. MFPTH - ROUTINE TO OBTAIN THE MATERIAL MEAN FREE PATH

The MFPTH subroutine is used to obtain the material mean free path. The mean free path (λ) is equal to the inverse of the total macroscopic cross section and is stored for each material and micro group. If the material contains resonances and the energy is in the resonance region, routine RESCLC is called to calculate the resonance cross sections. The resonance cross sections are then combined with the micro group cross sections to obtain the mean free path.

F. RESCLC - RESONANCE CALCULATION ROUTINE

The single level Breit-Wigner resonance equations used by RESCLC are

$$\sigma_s(E) = \sigma_{s_0} \psi(x, \theta) + 2\sqrt{\sigma_p g \sigma_{s_0}} \chi(x, \theta) + \sigma_p$$

$$\sigma_c(E) = \sigma_{c_0} \psi(x, \theta)$$

$$\sigma_f(E) = \sigma_{f_0} \psi(x, \theta)$$

where $\psi(x, \theta)$ and $\chi(x, \theta)$ are the Doppler broadening functions of the form

$$\psi(x, \theta) = \frac{1}{2\sqrt{\pi\theta}} \int_{-\infty}^{\infty} \frac{e^{-(x-y)^2/4\theta}}{1+y^2} dy$$

and

$$\chi(x, \theta) = \frac{1}{2\sqrt{\pi\theta}} \int_{-\infty}^{\infty} \frac{ye^{-(x-y)^2/4\theta}}{1+y^2} dy$$

where
$$x = \frac{2[E-E_0]}{\Gamma} \quad \theta = 3.4468 \times 10^{-4} \frac{ET}{a\Gamma^2}$$

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

$$\sigma_{s_0} = 2.607 \times 10^6 g \frac{\Gamma_n^2}{E_0 \Gamma^2} \left(\frac{1+a}{a} \right)^2$$

$$\sigma_{c_0} = 2.607 \times 10^6 g \frac{\Gamma_n \Gamma_\gamma}{\sqrt{E E_0 \Gamma^2}}$$

and
$$\sigma_{f_0} = \frac{\Gamma_f}{\Gamma_n} \sigma_{c_0}$$

The BMC code uses velocity so all of the above are converted to velocity coordinates. Also the constants are combined to cut down on the number of calculations. For each resolved resonance the following values are stored:

$$\sigma_{s_0}, \sigma_{c_0}, V_{o_0}, \sigma_{f_0}, V_{o_0}, 2\sqrt{\sigma_{s_0} g \sigma_{c_0}}, V_{o_0}^L, \text{ and } 2 \cdot 1.91322/\Gamma.$$

For unresolved resonances the contribution from the two nearest resonances are used. The lower resonance energy is either the top resolved resonance or

the energy $E_{o_a} = \left[\frac{E}{D} \right] D$ where $\left[\right]$ means the greatest integer. The next resonance has the Energy $E_{o_b} = E_{o_a} + D$ where D is the level spacing. Also

for the unresolved resonance the value for Γ_n is picked using a Portor-Thomas distribution⁽¹⁾ and Γ_f is picked using an exponential distribution.

Two other approximate corrections can be made for level spacing and for the $\ell > 0$ resonance in the unresolved resonance range. The corrections are

$$C = (1 + D^* V^2) (1 + \ell^* V)^2 \text{ and are applied to } \sigma_{s_o}, \sigma_{c_o}, \text{ and } \sigma_{f_o}.$$

Note that the unresolved treatment here is not the same as formulated for the ENDF/B cross section library. Usually group averaged cross sections are used in the unresolved region. However, one can read in parameters for the unresolved region and the group values will not be used,

Associated with the resolved resonance table is the maximum value of the width, x , and a table which has the number of the first resonance to use in each micro-energy group. This table greatly reduces the time needed to determine which resonance to use. If the table entry is 0 do not use any resonances, if the table entry is n , use the n^{th} resonance only, and if the entry is $-n$, use the resonances starting with the n^{th} which are within the maximum x width from the neutron energy.

The Doppler broadening functions $\psi(x, \theta)$ and $\chi(x, \theta)$ are calculated using the CHPSI routine. CHPSI uses either a convergent series or an asymptotic series depending on the values of x and θ . The conditions under which the convergent series is used are:

$$\begin{aligned} 0 \leq |x| < 1.2 \text{ and } 4\theta &\geq 0.275 x^2 + 0.05 \\ 1.2 \leq |x| < 10 \text{ and } 4\theta &\geq 0.120 x^2 + 0.28 \\ 10 \leq |x| < 100 \text{ and } 4\theta &\geq 0.105 x^4 + 1.75 \\ 100 \leq |x| < 200 \text{ and } 4\theta &\geq 0.095 x^2 + 100 \end{aligned}$$

The convergent series are

$$\psi(x, \theta) = e^{-(x^2/4\theta)} \sum_{q=0}^{Q+1} \left(\frac{x}{4\theta} \right)^{2q} \alpha_q(4\theta)$$

where

$$\alpha_q(4\theta) = \frac{\Gamma_q(4\theta)}{q!},$$

and

$$\chi(x, \theta) = \frac{x}{4\theta} e^{-(x^2/4\theta)} \sum_{q=0}^Q \left(\frac{x}{4\theta} \right)^{2q} \alpha'_q(4\theta)$$

where

$$\alpha'_q(4\theta) = \frac{\Gamma_{q+1}(4\theta)}{q!}$$

$\Gamma_0(4\theta)$ is the incomplete gamma function and is fit for speed by using a set of parabolas. The other gammas are obtained using the equality

$$\Gamma_{q+1}(4\theta) = \frac{2}{2q+1} \left[(4\theta)^q - \Gamma_q(4\theta) \right].$$

The series is terminated at the Q^{th} term and is the term for which the following equation is satisfied.

$$\left\{ \left[\left(\frac{x}{4\theta} \right)^{2q} \alpha_q(4\theta) \right] / \sum_{q=0}^Q \left(\frac{x}{4\theta} \right)^{2q} \alpha_q(4\theta) \right\} \leq 0.1$$

The asymptotic series are

$$\psi(x, \theta) = \sum_{q=0}^4 \frac{1 \cdot 3 \cdots (2q-1)}{2^q} (4\theta)^q (x^2+1)^{-(2q+1)/2} \sin [(2q+1)\alpha]$$

and

$$\chi(x, \theta) = \sum_{q=0}^4 \frac{1 \cdot 3 \cdots (2q-1)}{2^q} (4\theta)^q (x^2+1)^{-(2q+1)/2} \cos [(2q+1)\alpha]$$

where

$$\sin \alpha = \frac{1}{x} \quad \cos \alpha = \frac{x}{(x^2+1)^{1/2}}$$

$$\sin 2\alpha = \frac{2x}{x^2+1}, \quad \cos 2\alpha = \frac{x^2-1}{x^2+1}$$

$$\sin [(n+2)\alpha] = \sin 2\alpha \cos n\alpha + \cos 2\alpha \sin n\alpha$$

and

$$\cos [(n+2)\alpha] = \cos 2\alpha \cos n\alpha - \sin 2\alpha \sin n\alpha$$

■ COLLSN - ROUTINE TO PERFORM THE COLLISION MECHANICS

The COLLSN subroutine is used to process the results of a collision. The events which are considered are absorption, fission, inelastic scattering, and elastic scattering. The elastic scattering can be treated as isotropic or anisotropic slowing down scattering or as ideal gas thermal scattering

1. Neutron Absorption and Fission

The COLLN routine first reduces the neutron weight to account for absorption. The weight is then tested to see if a scattered neutron is to be produced. The fission probability is checked to see if any fission neutrons were produced. N fission neutrons are produced if $wt \cdot P_f + \xi \geq N$ where ξ is a random number and P_f is the ratio of the neutron fission production rate ($\nu \Sigma_f$) to the total reaction rate (Σ_T) in the material. Note that N will seldom be greater than 1. The fission neutrons are stored in the fission bank.

2. Selection of Scatterer

If the neutron survived absorption then the scattering isotope is selected. To save computer space and time use is made of a "heavy scatterer." More than one isotope in a material can be specified as heavy isotopes. All of the heavy isotopes elastic scattering cross sections are combined to form one elastic scatterer. A mass for the heavy scatterer is also specified. Scattering from a heavy scatterer is treated as if the scattering mass were infinite (no energy change) unless the neutron is in a region having an isotope with resonance parameters and the neutron has a velocity such that a resonance calculation was made. Then the special heavy scattering mass will be used.

The scatterer is picked by selecting a random number ξ and comparing it to the cumulative scattering probabilities. If $P_{S_{i,j+1}} \leq \xi < P_{S_{i,j}}$ then the j^{th} scatterer is selected where $P_{S_{i,j}}$ is the cumulative scattering probability for the j^{th} scatterers and the i^{th} microscopic energy groups. $P_{S_{i,j}}$ can be either inelastic or elastic scattering. If the total number of scattering

reactions in a material is J_S with the total number of inelastic scatterers J_{in} and the total number of elastic scatterers J_{el} then $P_{S_{i,j}}$ can be defined as

$$P_{S_{in_{i,j}}} = \frac{\sum_{k=1}^j (\Sigma_{in_{j,k}} + 2\Sigma_{n-2n_{i,k}})}{J_{el} \sum_{k=1} \Sigma_{el_{j,k}} + J_{in} \sum_{k=1} \Sigma_{in_{i,k}}}$$

and

$$P_{S_{el_{i,j}}} = \frac{\sum_{k=1}^j \Sigma_{el_{i,k}} + \sum_{k=1}^{J_{in}} (\Sigma_{in_{j,j}} + 2\Sigma_{n-2n_{i,k}})}{J_{el} \sum_{k=1} \Sigma_{el_{i,k}} + J_{in} \sum_{k=1} (\Sigma_{in_{i,k}} + 2\Sigma_{n-2n_{i,k}})}$$

where $\Sigma_{el_{j,k}}$ and $\Sigma_{in_{j,k}}$ are the macroscopic elastic and inelastic cross section respectively for the k^{th} isotope or the heavy scatterer. Note that if $j = J_S$ then $P_{S_{el_{i,J_S}}} = 1$ so it is only necessary to store $J_S - 1$ scattering probabilities.

3. Inelastic Scattering

Currently inelastic scattering is treated in only two ways, however, other models could be included. Also, n-2n scattering is combined with inelastic scattering and treated the same. The two models used for inelastic scattering are discrete level energy loss and the evaporation model. Both models assume that scattering is isotropic in the center of mass system.

Inelastic Spectrum Table. The inelastic spectrum table (discrete energy loss) uses a table of the inelastic excitation level energies along with a probability for each one. If the neutron energy is below a certain excitation energy that level can not be excited. An approximation is made in that it is assumed the cross section for excitation of a certain level is a constant in energy above the level energy. The way a velocity selection is made using the inelastic spectrum tables is to find the table entry V_i^2 such that $V_i^2 < V^2 \left(\frac{a}{a+1}\right) \leq V_{i+1}^2$. A cumulative probability is selected by $P = \sum P_{i+1}$. Now a j is selected such that $P_j \leq P < P_{j+1}$. The new neutron velocity in the center of mass system is then calculated as $V'' = [V^2 \left(\frac{a}{a+1}\right)^2 - V_j^2]^{1/2}$. If the scatterer is not a "heavy scatterer" the center of mass coordinates must be converted to the lab system. This is done by first picking an isotropic unit vector $\alpha'', \beta'', \gamma''$. Then the new lab coordinates are

$$V' = [\ell^2 + m^2 + n^2]^{1/2}$$

$$\alpha' = \frac{\ell}{V'}; \beta' = \frac{m}{V'}; \gamma' = \frac{n}{V'}$$

$$\ell = V''\alpha'' + \frac{V\alpha}{a+1}$$

$$m = V''\beta'' + \frac{V\beta}{a+1}$$

$$n = V''\gamma'' + \frac{V\gamma}{a+1}$$

Evaporation Model The evaporation model is used to select the emergent energy from the Maxwellian distribution

$$P(E') = \frac{E'}{T_n^2} e^{-E'/T_n}.$$

Actually $(V')^2 = 1.91322 E'$ is selected. T_n is the nuclear temperature and is calculated by $T_n = a_0 + a_1 V + a_2 V^2 + a_3 V^3$. The coefficients a_0, a_1, a_2, a_3 are fitted by least squares to the data on the ENDF/B library for T_n . The new velocity is then calculated as

$$V' = [T_n(\phi_1 + \phi_2)]^{1/2}$$

where ϕ_1 and ϕ_2 are random numbers selected from an exponential distribution, or "random logs." It is also required that $V' \leq V$. The Maxwellian is sampled correctly as shown in the equation.

$$P(x) = \int_0^x e^{-y} e^{-(x-y)} dy = x e^{-x}.$$

It is assumed that the evaporation model is used only for heavy nuclei and so the conversion from the C.M. to the lab system is not made.

4. Nonthermal Elastic Scattering

Nonthermal elastic scattering can be treated as isotropic or anisotropic. If the scattering is isotropic, a random set of direction cosines, $\alpha'', \beta'',$ and γ'' are used as the new direction cosines in the center of mass.

Presently there is only one method in the BMC code for treating anisotropic scattering. The cosine of the scattering angle, μ is selected by making use of the sample rejection technique. First, a random value of μ is selected from a uniform distribution between plus and minus one. The value for μ is accepted if

$$\xi < \frac{P(\mu)}{P_{\max}} = a_0 + a_1 \mu + a_2 \mu^2 + a_3 \mu^3 + a_4 \mu^4,$$

where $P_{\max}$ is the maximum value of $P(\mu)$ for $-1 \leq \mu < 1$ and ξ is a random number. The coefficients $(a_0, \dots, a_4)$ are obtained using the first four terms of the Legendre polynomials, $P(\mu)$.

Once the new scattering cosine μ is selected the new velocity and the direction cosines are selected. If the isotope is a heavy scatterer the velocity is not changed. Otherwise the new velocity V' and the value of μ is converted from the center of mass to the laboratory system by the following equations where A is the weight of the scatterer in neutron mass units.

$$V' = V \sqrt{\frac{1 + 2 A \mu_{\text{cm}} + A^2}{1 + A}}$$

$$\mu_L = \frac{1 + A \mu_{\text{cm}}}{\sqrt{1 + 2 A \mu_{\text{cm}} + A^2}}$$

The new direction cosines, α' , β' , γ' , are then selected using the equations.

$$\alpha' = \alpha \mu - \beta \sqrt{\frac{1-\mu^2}{1-\gamma^2}} \cos \psi - \alpha \gamma \sqrt{\frac{1-\mu^2}{1-\gamma^2}} \sin \psi$$

$$\beta' = \beta \mu + \alpha \sqrt{\frac{1-\mu^2}{1-\gamma^2}} \cos \psi - \beta \gamma \sqrt{\frac{1-\mu^2}{1-\gamma^2}} \sin \psi$$

and

$$\gamma' = \gamma\mu + (1-\gamma^2)\sqrt{\frac{1-\mu^2}{1-\gamma^2}} \sin \psi$$

However if $|\gamma| > 0.999$ the equations become

$$\begin{aligned}\alpha &= \sqrt{1-\mu^2} \cos \psi \\ \beta &= \sqrt{1-\mu^2} \sin \psi \\ \gamma' &= \mu \text{ if } \gamma > 0 \text{ and } \gamma' = -\mu \text{ if } \gamma < 0.\end{aligned}$$

where ψ is a random angle in the range $0 \leq \psi \leq 2\pi$.

5. Thermal Elastic Scattering

Thermal scattering may be treated using the ideal gas scattering model. When using the ideal gas model to represent bound scatterers it is the common practice to use a first order correction⁽⁶⁾ to the transport cross section. The approximation is to use a fictitious isotropic heavy scatterer which does not change the energy. The cross section of the fictitious isotope is picked so that the transport cross section of the bound scatterer is conserved, or

$\sigma_s'(E) = \sigma_{tr_b}(E) - \sigma_{tr_{ID}}(E)$ where $\sigma_{tr_{ID}}$ is the scattering cross section for the fictitious scatterer. Note that the transport cross section for the fictitious isotope is just its scattering cross section since $\mu = 0$.

The selection from the ideal gas model is accomplished by selecting a direction and velocity for the target atom and performing the necessary collision mechanics. This is done by first picking a nuclear or target velocity V from either the Maxwellian distribution

$$P(V) = \frac{4}{\sqrt{\pi}} \left(\frac{a}{2kT} \right)^{3/2} V^2 e^{-\frac{aV^2}{2kT}}$$

or the velocity weighted Maxwellian,

$$Q(V) = \frac{4}{\sqrt{\pi}} \left(\frac{a}{2kT} \right)^{3/2} V^3 e^{-\frac{aV^2}{2kT}} \text{ where } T \text{ is the temperature.}$$

The selection of ΔV is made from the P distribution if $\xi_1 \leq V / \left(V + 2 \sqrt{\frac{2kT}{\pi a}} \right)$; otherwise the selection is made from the Q distribution. After the target velocity ΔV has been obtained from the P or Q distribution, the direction cosines of the target vector ΔV , α'' , β'' , and γ'' , are selected from an isotropic distribution. The relative incident direction cosines and velocity then become

$$V_r = [(\alpha V - \alpha'' \Delta V)^2 + (\beta V - \beta'' \Delta V)^2 + (\gamma V - \gamma'' \Delta V)^2]^{1/2}$$

$$\alpha_r = \frac{\alpha V - \alpha'' \Delta V}{V_r}$$

$$\beta_r = \frac{\beta V - \beta'' \Delta V}{V_r} \quad \text{and}$$

$$\gamma_r = \frac{\gamma V - \gamma'' \Delta V}{V_r} .$$

The value V_R is only accepted if $V_R/(V + \Delta V) \leq \xi_2$; otherwise the whole selection process is repeated. Once the relative velocities are selected, the conversion from the center of mass to laboratory systems is accomplished using the equations:

$$V^1 = [(\alpha_R V_R + \alpha' \Delta V)^2 + (\beta_R V_R + \beta' \Delta V)^2 + (\gamma_R V_R + \gamma' \Delta V)^2]^{1/2}$$

$$\alpha^1 = \frac{\alpha_R V_R + \alpha' \Delta V}{V^1}$$

$$\beta^1 = \frac{\beta_R V_R + \beta' \Delta V}{V^1}$$

$$\gamma^1 = \frac{\gamma_R V_R + \gamma' \Delta V}{V^1}$$

H. FXILY - ROUTINE TO TALLY **FLUX**, LEAKAGE AND RESONANCE REACTION RATES

The **FXILY** subroutine is used to tally the results of the calculation. The quantities that are tallied are **flux**, leakage, and resonance **reaction** rates.

IV. THE BMC CODE: ORGANIZATION AND LAYOUT

The BMC code is comprised of three parts; input routines, the Monte Carlo routines, and the output routines. Each part of the code is maintained in different MAP segments (chain links) and are loaded as needed. The input routines read and process the problem input and the library cumulative cross section tape(CCT), preparing the information used by the Monte Carlo and output routines. This information is placed on drum and on a tape which can be used to restart the problem. The Monte Carlo routines are then loaded and executed. After the Monte Carlo calculation is complete, the output routines are loaded to process the Monte Carlo output.

A. DRUM AND TAPE USAGE

The BMC code uses four drum files labeled D1 to D4. The drum word count is used as a means of direct addressing of records. Table V contains the short outline of the usage and contents of the drums during the different parts of the execution phase.

The tapes used by the BMC code are the tape containing the BMC code, the CCT tape containing the micro-group cross section library, and the BMC code restart tape. Only the BMC code restart tape is needed for restarts.

B. RESTART CAPABILITY

The BMC code was designed so that a Monte Carlo calculation can be restarted and continued. The way this is done is to save the output from the Monte Carlo calculations on a restart tape along with the fission neutrons and the last random number used by the code. The restart tape may also contain the BMC code in absolute or symbolic form, the contents of drums D1 and D2, and the geometry input.

A control word, **NWRCD**, is used with all but the first two records on the restart tape. NWRCD is normally the number of words in the record. The end of a section, such as the contents of drum D1, is signaled by a record with NWRCD set equal to zero. The end of the restart tape is signaled by a record with NWRCD = 0 and the second word of the record set equal to -2. Note that a record written or read with the list ~~list--NWRCD, (NBUF(N) , N=1, NWRCD)--~~ will always have two words, even if NWRCD = 0.

TABLE V
DRUM USAGE AND CONTENT

<u>D_m</u>	<u>Execution Phase</u>		
	<u>Input</u>	<u>Monte Carlo</u>	<u>Output</u>
D1	Cross Sections from CCT Material-energy-cross section information in format for the Monte Carlo routines		Averaged batch cross sections and macro fluxes.
D2	Material-energy-cross section information for Output routines.		
D3	Batch output from the Monte Carlo routines.		
D4		Fission bank overflow.	Averaged Microgroup fluxes for plotting.

The layout of the BMC restart tape is:

It is not necessary to have the BMC code at the beginning of the restart tape. When the BMC code is given control, it is only required that the restart tape be positioned to read record R1. R1 and R2 are records which contain general information about the restart tape and about the problem.

Note that the tallies and restart information are written for each restart. This is a safety factor in case there is an error. If there is a slight error in any of the sections, it is possible to redo the input phase of the BMC code and still restart using the old restart information and saving the old flux tallies.

C. PROGRAM TIMING

The amount of time in seconds spent in various portions of the code is clocked and printed. The times which are written are:

- The compilation and load time.
- The time required in the input routines.
- The time required to process each batch.
- The total time for the Monte Carlo calculation.
- The time required to process the output.

The total time is also printed along with each of the above times,

The clock routines are also used to check the amount of time used for a batch. If there is not enough time left to do another batches and allow 1 minute to finish the output then the Monte Carlo is terminated and the output routines are called.

D. PROBLEM SIZE

The BMC code was designed to be flexible as to the size of problem which can be run in terms of energy, cross section, and geometry detail. The BMC code has been used for problems which contained as little as one material in one region and for problems which contained as many as 4 materials, 100 energy groups, and 135 geometry and tally regions.

The dimensions of the storage arrays are entered using PARAMETER cards and most of the storage arrays are maintained in a data element which is placed in the required subroutines using an INCLUDE statement. This makes it easy to change the dimensions for a problem.

The dimensions which are built into the standard version allow for 190 energy micro-groups, 20 macro-groups, 6 materials, 20 tally regions, 2 leakage tally set, 60 geometric regions, 60 boundaries, and 1000 fission particles in the fission bank. The storage bank for born particles is set to hold only 1 particle so if importance weighting were used the GETN routine would have to be recompiled.

The dimensioning parameters for the BMC code are defined in Appendix I.

V. THE BMC OUTPUT ROUTINES

The BMC output routines prepare the output and print **it** and/or plot **it**.

The two main tasks in preparing the output are preparing averaged values of fluxes, cross sections, reaction rates, and leakages along with some other cell parameters and obtaining statistics for the values which are calculated.

A. STATISTICS

Statistics are obtained by processing a series of equal sized batches and averaging the results for the batches. It is possible to leave some of the first batches out of the average. This might be done to damp out the effects of the source distribution. There are two ~~common~~ types of average values obtained from the Monte Carlo calculation. One type is a direct answer such as flux or reaction rate and the other type is a ratio such as flux averaged cross sections. The statistical error for a single averaged value $\bar{a}$ is $S_{\bar{a}}$ where $S_{\bar{a}}$ is the error for one standard deviation. Let N be the number of equal sized batches and a_n be the value of a for the n^{th} batch. The values of $\bar{a}$ and $S_{\bar{a}}$ then become:

$$\bar{a} = \frac{1}{N} \sum_{n=1}^N a_n$$

and

$$S_{\bar{a}} = \left[\left\{ \sum_{n=1}^N a_n^2 - N (\bar{a})^2 \right\} / N (N-1) \right]^{1/2}$$

$$= \left[\left\{ \sum_{n=1}^N a_n^2 - \left(\sum_{n=1}^N a_n \right)^2 / N \right\} / N (N-1) \right]^{1/2}$$

The statistical error for a ratio R is

$$S_R = \left[\sum_{n=1}^N a_n^2 - 2R \sum_{n=1}^N a_n b_n + R^2 \sum_{n=1}^N b_n^2 \right]^{1/2} / \sum_{n=1}^N b_n,$$

where

$$R = \sum_{n=1}^N a_n / \sum_{n=1}^N b_n.$$

Note that for a ratio the values of a and of b may have large statistical errors and the statistical error of the ratio can be very small or even zero. This is accounted for by the correlation term $(-2R \sum_{n=1}^N a_n b_n)$.

The statistical error is written out in a way which reduces the printing space. The statistical error is forced to have the same exponent as the value. If a value and its error were $1.0685 \times 10^4 \pm 5.2 \times 10^2$ then the numbers would be printed out as 1.0685+04 0.052.

B. BROAD ENERGY AVERAGING GROUPS

The output routines make use of one more type of energy group also called a broad group. These broad group energy boundaries must coincide with the macro-group boundaries. The broad groups do not need to cover the entire energy range or to have joining boundaries. In fact, the boundaries can overlap. To illustrate, suppose there were five macro-groups; then the four broad groups could contain groups 1-4, 1-3, 1-2 and 4 respectively. Note that group 5 is never included, group 4 is by itself, and the groups 1-4 are overlapping.

The broad groups are used for flux and reaction rate integrals. The broad group fluxes are also used to obtain statistics on the ratios between different

energy broad groups or between regions for a given broad group.

C. PRINTED OUTPUT

The output code prints out the results of the Monte Carlo calculation. First the averaged code check tallies are printed out. Next the macro-group and the broad group fluxes and leakages are printed. The macro-group average cross sections and the broad group reaction rates for each isotope in each region are printed along with the reaction rates for the region. The reaction rates for the entire cell along with an infinite and effective multiplication constant follow. Finally the micro-group fluxes for each region are printed and/or plotted.

D. THE DEFINITION OF FLUX

The flux as calculated by the Monte Carlo routines is really the flux-energy volume integral, which has the units of centimeters per second. This flux integral is what is used with cross sections to obtain reaction rates and is usually more useful than true flux.

The flux is printed out in the following three forms by the output routines:

- 1) Flux-volume-energy integral - Flux-cm/sec.
- 2) Flux-energy integral Flux/cm²/sec.
- 3) Flux Flux/eV/cm⁴/sec.

E. PLOTTING THE MICRO-GROUP FLUXES

The micro-group region and/or region averaged fluxes may be plotted. The plotting can either be a rough plot on the printed output or a neat plot using the Calcomp plotter. If the printer is used, only the flux points are plotted. If the Calcomp plotter is used, the flux points along with their error bars can be plotted.

The plots of the fluxes serve mostly as visual checks against gross errors or to observe how the flux spectrum changes from region to region. The fluxes can be normalized over part or all of the energy range and plotted on the same curve so that changes in spectral shape are readily visible.

F. SPECIAL OUTPUT

There may be special values which are desired which are not part of the regular output. These special values can often be obtained by simple changes to the BMC output routines. For some problems it might be better to rewrite the output routines.

VI. THE BMC INPUT ROUTINES

The BMC input routines are used to prepare the data needed by the BMC Monte Carlo and output routines. They also prepare much of the restart tape. The input routines receive information from two sources; card input and the CCT cross-section tape. The input routines also print the input information.

The card input is read by two different types of formats. The first type uses standard FORTRAN formats. The second type, used mainly for the geometry input, requires a special card reading routine.

A. SPECIAL CARD INPUT ROUTINE, CRDRD

The CRDRD* routines read cards of two format types. They are called P or U. The P or U is read in column 1 on the first card of a section of input called a record. The record is ended with an end of record marker, R. Cards with an asterisk in the first column can be used before the P or U type card as comment cards.

The P card indicates that the data will be intergers using 3 column fields (13). As many as 24 or as few as one data words can be on one card. Each field of three columns is checked. If a slash, /, is in any one of the three columns, that field along with the rest of the card is skipped. The record is ended with a R card.

"Written and coded by Kent B. Stewart, Battelle-Northwest Laboratory.

The U card indicates that the input will be a mixture of interger and floating point numbers. A number is considered to be floating point if it contains a decimal point and/or an exponent. Otherwise it is considered to be an integer. The data word must be contained in a field of not more than 12 columns. The data words do not have to be right adjusted. Fields of less than 12 columns can be used by placing a comma after the data word. The next field would then start right after the comma. Note, the routine assumes that the maximum field length is 12 columns so if a number uses more than 12 columns, it will be split into two numbers. If the columns are left blank, a zero will be entered.

Again, as with the P card, a slash in any column of a field will cause that field and the rest of the card to be skipped. The space after the slash may be used for comments as it is not examined.

Several identical words can be entered simply by putting the number of words followed by an asterisk. For example, 11*4.0 would enter the number 4.0 into 11 consecutive words of the record. The field of 12 columns starts right after the * when the * is used.

B. DESCRIPTION OF THE BMC CODE PROBLEM INPUT

The problem input numbers for each input card will be described along with their names and format. It will also be indicated when the input card is needed.

The first two input cards are always needed. The first card contains a BCD title which is printed at the top of each page of the output along with the date and page number. The second card contains execution control information. Card 3 and the input to the source routine are always required if a Monte Carlo calculation is done. The input for the output routines is always required if

the output calculations are done. The energy-material-geometry description input is not needed for a normal restart.

The restart tape is called IMCT when **it** is being written. **It** is called IMCO when **it** is being read. IMCO will always be on logical tape unit A while IMCT can be on unit A or B.

DESCRIPTION OF CARD INPUT TO THE BMC CODE

Card 1 Format (12A6) BCD title card

Card 2 Format (11I5, 5X, 2A6, 7XI1) program control

NPRBIP	Identification number of the Problem tape. If this is a restart, the number must match the number on the restart tape.
LSIRT	<p>0, Initial start. MCT = B.</p> <p>1, Restart, no changes, MCT = A</p> <p>2, Restart but do input calculations over.</p> <p>3, Restart but do the geometry input calculations over.</p> <p>(For LSIRT = 2 or 3, MCO = A, MCT = B)</p>
LCONT	<p>0 or 1, Do a Monte Carlo calculation.</p> <p>2, Go to the output routines after the input. Do not do a Monte Carlo calculation.</p>
LSTOP	<p>0, Do complete problem.</p> <p>1, Do input calculations only.</p> <p>2, Do input and Monte Carlo calculations but not the output processing.</p>
LCOPY	<p>If 1, copy the restart tape MCO to MCT and use IMCT.</p> <p>MCO = A, IMCT = B. (Active only if LSIRT = 1).</p>
LSKNS	<p>LSKNS is used when LSIRT > 0.</p> <p>0, Position restart tape to the normal (last) restart.</p> <p>If LSKNS > 0, then LSKSV must equal 77 or calculation will not be made.</p> <p>1, Start the tallies anew.</p> <p>2, Use the restart dated (DLSKA, DLSKB).</p>

LSKSV Safety factor for use of **LSKNS** > 0. **Must** equal 77 when **LSKNS** > 0.

The next three words are flags which make it possible to not save portions of the restart tape. They are normally zero.

ICXTSV 1, Do not save the output cross section tape (Drum D2).

IMCTSV 1, Do not save the Monte Carlo tape (Drum D1 and the Geometry Input).

IFXTSV 1, Do not save the flux tallies or restart information.

NERCV Number of recoverable errors before termination of calculatim. If **NERCV** = 0, it will be set to 10.

DLKA,DLKB The date for the restart under option **LSKNS** = 2. The dates are left adjusted and are of the form 18 ~~JAN~~ 69 or 07 FEB 69.

IDUMP A 1 in column 80 will cause a **dump** of the Monte Carlo blank **common** and geometry **common** data just before the Monte Carlo code is called.

Card 3 Format (215, 3E10.0, Ø15) Monte Carlo batch control. (Card 3 not needed if **LCØNT** = 2)

NBTCH The number of batches for this case. (**Not** total number of batches for the calculatim.)

NPTPB The number of neutrons (source or fission) to follow in each batch.

The next three input numbers are used only if the thermal tally option is used (see Card 4, ITALY = 2).

RTHRT The ratio of the weight leaving to that entering the thermal tally range.

ETHRT The energy to which neutrons must be slowed down before they are used in (enter) the thermal tally range.

ETHRTX The maximum energy a neutron may achieve in the thermal tally range.

RND RND can be used to change the initial random number. If RND is left blank, the octal number 343277244615 or the random number at the end of the last batch will be used. Otherwise, the octal number RND read in will be used.

Card 4 Format (1415,12) Problem description parameters. (Card 4 not needed if LSTRT = 1)

IDCCT Identification number of the CCT cross-section tape.

LPRB Problem type contro:
0, Fission descendant problem.
1, Directo source problem.

IMPRT	0, no importance weighting. 1, use importance weighting by zone. The zones will be the same as the tally regions. The zone importance is entered in CRDRD Record 4 (see page 64).
LESET	Energy set number. It is possible to have isotopes on the CCT tape with two different energy group sets. This tells which energy set to use. If LESET = 0 or 1, the first set will be used. If LESET = 2, the second energy set will be used.
NBGPX	The number of broad energy group::;. If NBGPX = 0, one broad energy group which includes the entire energy range specified on the CCT will be used.
JMAXX	Total number of Macro-energy groups in all of the broad groups.
NHMAX	Number of geometrical regions.
MMA	Number of materials. (Not including voids.)
NSPRG	Number of "Special" geometrical regions.
NFMAX	Number of tally regions.
NLKTLY	Number of leakage tally sets.
LTALY	0 or 1; Use only one tally set for each tally region. 2; use two tally sets (make special thermal tallies) for each tally region.
NBNDX	Number of boundaries used in the geometrical description.

NOMGX Number of boundary reflection albedo sets. Read in with CRDRD Record 3 (see page 64).

LESRC (Normal zero.) If LPRB is zero, a non-fission source for the first batch may be used by setting LESRC = 1.

The input from here to CRDRD Record 1 is not needed if LSIRT = 1 or LSIRT = 3.

Card 5 Format (6E12.0) (Not needed if NBPX = 0)

EBGM(NBPX+1) Enter the broad group energy boundaries starting with the highest energy. Note that NBPX+1 energies are required to define NBBX broad groups.

Card 6 Format (14I5) (Not needed if NBBX = 0)

ITBGM(NBPX) The type of each energy broad group. Presently only type 1 is allowed but code expansion could make this input meaningful.

Card 7 Format (6E12.0)

EJG(JMAXX) The lower energy limit for each Macro-group starting with the highest energy group.

Repeat Card 8 and its Card 9's for each material, Card 9 is repeated for each isotope in the material.

Card 8 Format (3I5,E10.0,5X,3A6)

NISO The number of isotopes in the material.

ITEMPM The material temperature in degrees Kelvin. Note that this is an integer. It should usually match the temperature associated with the isotopes on the CCT.

LSEIM The energy set on the CCT to use with this material.
If **LSEIM** = 0, it will be set to 1.

AWHVM The atomic weight in neutron masses to use with the
heavy scatterer (if used) in this material.

BCDM(3) Name of the material. Can use 18 letters.

Card 9 Format (I5,E15.0,2I5,E10.0,2I5)

ISOM Identification number for the isotope or material on
the CCT tape.

CONM The concentration in atoms per barn centimeter for this
isotope.

IHVYM If **IHVYM** = 1, treat this isotope as a heavy scatterer.

LTHRMM The type of scattering treatment to use for this isotope
in the thermal region.
0, normal slowing down collision mechanics
1, use the ideal gas model.

ETHRM The maximum energy for which thermal scattering will
be used.

LANMM 0, use anisotropic data as given on the CCT.
1, do not use anisotropic data for this isotope,
even if data is on the CCT.

INMM 0, use inelastic scattering data as given on the CCT.
1, do not use inelastic data for this isotope except
to combine the inelastic scattering cross sections
with the elastic data.

The next section of records contains the geometry and region data. A set of data which is read by the **CRDR** routines will be designated as **CRDR** Record. This section is not needed if **LSTRT = 1**.

CRDR Record 1, Format U (mixed floating point and integer)

Boundary description record.

The following information is repeated for each of the **NBNDX** boundaries.

IDB	The index telling the boundary type. If the first value of IDB is negative, all boundary dimensions are in inches. If the first value of IDB is positive, all boundary dimensions are in centimeters.
IOM	Albedo index. Gives type of reflection. 0, no reflection. 1, mirror reflection using albedo set 1. 2, mirror reflection using albedo set 2. 3, mirror reflection, unit albedo. 4, isotropic reflection, unit albedo.

Boundary Parameters

The boundary parameters depend on the boundary type. The parameters and their order are:

IDB	Parameters.	Dimensions in centimeters or inches, depending on IDB .
1	X_0	Plane $x = x_0$.
2	Y_0	Plane $y = y_0$.
3	Z_0	Plane $z = z_0$.
4	b, a	Plane $y = ax + b$.
5	R	R = radius of cylinder centered at origin

6	R,X ₀ ,Y ₀	cylinder centered at X ₀ ,Y ₀ .
7	R	sphere, centered at origin.
8	K,X ₀ ,Y ₀ ,Z ₀ , A,B,C	General boundary

For example, a record for 5 boundaries could be

```
P
3,3,-5.0,3,0,0,3,3,5.0,/
6,0,1.0,0.5,0.5,6,4,2.0,0.5,0.5,/
R
```

CRDRD Record 2, Format P (2413)

Region description record.

This record contains four words for each region plus four words for each boundary in each region.

NBH	Number of boundaries needed to describe this region.
LSPH	0, this is a normal region. L, the number of the "special" region to use in this region. (L ≤ NSPRG)
MATH	Material to use in this region. If MATH = 0, the region is an external void. A neutron in an external void is rejected. If MATH = -1, the region is an internal void. The neutron-beam will cross the region without collisions or alteration.
IFH	The tally region number for this region. If IFH = 0, no tallies will be made for a neutron -beam in this region. Note that the flux in an internal void can be tallied by setting IFH ≠ 0.

The following four parameters are entered for each of the NBH boundaries of the region.

IGDH

G is the sign of the boundary and

D is the leakage tally set index.

 $G = \text{sign of } \mathbf{IGDH}$ $D = \text{ABS } (\mathbf{IGDH}) - 1$ If $D = 0$, i.e., $\mathbf{IGDH} = \pm 1$, no leakage is tallied.If $1 \leq D \leq 50$, i.e., $\mathbf{IGDH} = \pm 2$ to ± 51 , the leakage will be tallied into the leakage tally set D and the neutron-beam will be killed.If $D \geq 51$, then the leakage across the boundary will be tallied into leakage tally set D-50 but the neutron-beam will not be killed.**IBHH**

Boundary index.

IQ1H

The region the neutron will most likely enter next if it leaves the region across this boundary.

IQ2H

The region to start the region search in if region IQ1H fails. (Usually this will be the lowest numbered region which the neutron could enter.)

IGDH to **IQ2H** are repeated for all of the boundaries of the region and then the data is repeated starting with **NEH** for the next region.

Cards 10 to 14 not needed if **NSPRG** = 0.

Card 10 Format (1415)

LTSPS(NSPRG)

LTSPS is the type of special region. Enter a value for each of the **NSPRG** special regions. Currently the only type of special region available is the square lattice of clad fuel rods (**LTSPS** = 1).

Cards 11 to 14 contain input for the special square lattice geometry routine. If a different special geometry routine were being used, the input would be different but the input cards would be in the same position.

Cards 11 to 14 are repeated for each special region.

Card 11 Format (1415)

NRX	Number of rows of rods in the X direction.
NRX	Number of rows of rods in the Y direction.
MATCL	Material number of the cladding.
IFTCL	Tally region of cladding.
MATFL	Material number of the fuel in the rods. If MATFL = 0, the material numbers for each rod will be read using Card 13.
IFTFL	Tally region of the fuel rods. If IFTFL = 0, the tally region numbers will be read using Card 14.

Card 12 Format (7E10.0) (All dimensions in centimeters or inches, depending on IDB.)

XC	X coordinate of the center of the lower left hand corner rod, rod (1,1).
YC	Y coordinate of the center of the lower left hand corner rod.
DXC	Lattice spacing in the X direction.
DYC	Lattice spacing in the Y direction.
RDF	Radius of the fuel rod.
RDC	Radius of outer edge of cladding.

Card 13 Format (14I5) Needed only if MATFL = 0.

MATXYS(NRX,J) The material numbers of all the rods in the J'th row.

Repeat card 13 for J = 1,2---,NRY.

Card 14 Format (14I5) Needed only if IFTFL = 0.

IFTXYS(NRX,J) The tally region number of all the rods in the J'th row.

Repeat Card 14 for J = 1,2,---,NRY.

CRDRD Record 3 Format U (Not needed if NEMGX = 0).

OMGAB(JMAXX,NEMGX) The boundary reflection albedos. Enter a value for each macro-energy group for the first albedo set. Then repeat for the next set until NEMGX sets have been entered.

CRDRD Record 4 Format U Russian Roulette and importance weight data for each region.

WTH(NHMAX) The minimum neutron weight before Russian Roulette is played. A value is needed for each region.

WTXH(NHMAX) Neutron survival weight.

BEMNH(NHMAX) Minimum beam strength before Russian Roulette is played.

BEMXH(NHMAX) Beam survival strength.

ZWTH(NHMAX) Importance weight of each region. (Not needed if IMPRT = 0.)

Source Routine Input

The source routine input is needed whenever a Monte Carlo calculation is made. It is not put on the restart tape. The input described here is for the BMC general source routine. If a different source routine is used, the input would go here.

Card S-0 Format (215) (Needed for any Monte Carlo calculation.)

NREG The number of source regions. If ~~NREG~~ is negative
a listing of the input description will be printed.

NCH 0 = dimensions in centimeters
-1 = dimensions in inches

No more source input is needed if NREG = 0. If NREG $\neq$ 0, read the following cards for each source region.

Card S-1 Format (412,2X,2E10.0)

LRGCS Type of geometrical coordinate system to use in this
source region.

- 1, Rectangular
- 2, Cylindrical
- 3, Spherical

NEGS Number of energy sources in this region. Card S-5
will be read for each of the ~~NEGS~~ sources.

LANGS	Type of angular distribution. 0 or 1, isotropic 2, mono-directional beam. Supply the direction cosines on Card S-6.
IQRGS	The geometrical region to look in first to locate a neutron born in this source region. If the neutron is not in region IQRGS or if IQRGS = 0, a region search will start with region IQRGS + 1.
PRBRS	The probability that a neutron is born in this source region relative to the other regions. PRBRS does not have to be normalized.
WTPRS	Weight of a neutron born in this source region. (Usually 1.0. If WTPRS = 0, it will be set to 1.0.)
Card S-2 Format (4I2,2X,6E10.0) Needed only for rectangular geometry, IQRGS = 1.	
LXD	Type of spacial distribution in the X direction. 1, Point at X_1 2, Uniform between X_1 and X_2 . 3, Cosine $\left[\frac{\pi}{2} \cdot \frac{(X-X_0)}{H_x} \right]$ between X_1 and X_2 .
LYD	Spatial distribution in the Y direction. (Same distributions as under LXD.)
LZD	Spatial distribution for the Z direction.
LHOL	The number of cylindrical holes to include in this source region. (It is often necessary to exclude a cylindrical region such as a fuel rod from a rectangular source region. This may be done by using "holes", which are infinite cylinders in the Z direction. The hole parameters are input on Card S-2B.

X1 Spatial coordinate limits.
Y1
Z1
X2
Y2
Z2

Card S-2A Format (6E10.0) Need if LXD,LYD, or LZD = 3.

XZ The values for X_0 and H_x used in the cosine distribution,
HX ($LXD = 3$). Same for Y and Z.
YZ
HY
ZZ
HZ

Card S-2B Format (6E10.0) Needed only if LHOL > 0.

XHOL The hole is a cylinder centered at (XHOL,YHOL) with
X radius RDA. Repeat as many cards as necessary, two
sets of parameters per card.

YHOL

ROA

Card S-3 Format (4I2,2X,6E10.0) Needed only for cylindrical geometry, (LRGGS = 2).

LRD The type of spatial distribution in the radial direction.
1, fixed radius, $r = r_c$.
2, uniform in volume between r_1 and r_2 .
3, $J_0(2.405 * r/r_c)$ between r_1 and r_2 .
The radius is centered at (X_0, Y_0) .

LZD The type of spatial distribution in the radial direction.

1, plane at $z = z_1$.

2, uniform between z_1 and z_2 .

3, cosine $\frac{\pi}{2} \frac{(z-z_0)}{H_z}$ between z_1 and z_2 .

IXPM IXPM and IYPM tell which quadrants the source neutrons may fall in. If IXPM = 0, then X can be plus or minus.

IYPM If IXPM = +1, then X can only be plus. If IYPM = -1, then Y can only be negative. IYPM controls Y in the same way.

XZ
YZ Center line of the cylinder.

R1 Spatial coordinate limits.

R2

Z1

Z2

Card S-3A Format (3E10.0) Needed if LRD or LZD = 3.

RC Extrapolation radius, r_c .

ZZ Z_0

HZ H_z

Card S-4 Format (412,2X,6E10.0) Needed for spherical geometry, IRGS = 3.

LRD Type of spatial distribution.

1, fixed radius, $r = r_1$.

2, uniform in volume between r_1 and r_2 .

3, (Sin U)/U where $U = \pi r/r_c$. Between r_1 and r_2 .

The radius is centered at X_0, Y_0, Z_0 .

IXPM Quadrant definition. See IXPM under Card S-3.

IYPM

IZPM

XZ Coordinate definitions.

YZ

ZZ

R1

R2

RC

Card S-5 Format (I2,8X,6E10.0) Use this card ~~NECS~~ times.

LEIS Type of this energy source.

1, point at $E = E_c$ (electron Volts).

2, Maxwellian distribution around $kT = E_c$.

3, fission distribution,

4, slowing down source past the energy E_c by a scatterer of Mass A.

PES Probability of using this energy source relative to the other energy sources in this region. (Normalization not necessary)

EC Energy E_c in electron Volts.

AWT Atomic weight, A. Used when LEIS = 4.

Card S-6 Format (3E10.0) Needed if LANGS = 2.

ALS Direction vector or cosines for the mono-directional

BIS beam. (Normalization not necessary)

GMS

The following input is read by the output routines and is not included on the restart tape so must be used with each restart.

Card 0-1 Format (6I5)

NBCHK Number of batches to skip before starting output processing. This could be used, for example, to get

several generations away from the source distribution when making a fission descendant calculation.

LCXST	<p>0 = do not process or print cross sections or reaction rates.</p> <p>1 = print all reaction rates and cross sections</p> <p>2 = print reaction rates only</p> <p>3 = print region reaction rates only</p>
LFXST	<p>0 = print all fluxes</p> <p>1 = suppress group flux ratio matrix</p> <p>2 = suppress group and region flux ratio matrix</p> <p>3 = print region flux ratio matrix only</p>
NEBG	Number of output broad energy groups. (Maximum of 4)
ICPLT	<p>0, do not process the micro-group fluxes.</p> <p>1, process the micro-group fluxes for printing and plotting.</p> <p>3, suppress printing.</p>
NPAV	Number of region averages for the micro-group fluxes. (Maximum of 10)

Card 8-2 Format (6E12.0)

VOL(NEMAX)	Tally region volumes or volume ratios.
------------	--

Card 9-3 Format (1415) Needed if NEBG >0.

JEBG1 (NEBG)	Lowest and highest numbered macro-group contained in each broad group. The order would be JEBG1(1), JEBG2(1), JEBG1(2), ..., JEBG1(NEBG), JEBG2(NEBG).
JEBG2 (NEBG)	

Card 0-4 Format (1415) Needed if $NPAV > 0$.

$NIPHA(NPAV)$ The number of tally regions to be included in each micro-group flux average.

Card 0-5 Format (1415) Needed if $NPAV > 0$, repeat $NPAV$ times.

$IPHAV(NIPHA, NPAV)$ The tally region numbers for each flux average.
There will be $NPAV$ Card 0-5's each containing a set of tally region numbers.

The rest of the input is only needed if the micro-group fluxes are to be printed or plotted. ($ICPLT > 0$.)

Card F-1 Format (515,5X,E10.0)

$NPL0T$ 0, no plotting.
N, ~~make~~ N plots using the Calcomp plotter.
-N, make N plots ~~on~~ the printed output. The printed output plots do not show the error bars, nor do they have the accuracy of the Calcomp plotter.

$IITYPE$ Type of flux.
0 or 1, the flux is $E \cdot \phi(E)$ or flux per unit lethargy. .
This is usually best for plotting because the slowing down flux is nearly flat.
2, $\phi(E)$, flux per unit energy.
3, $\phi(e)\Delta E$, this is the flux as calculated by the Monte Carlo and is the flux integral used to obtain reaction rates.

$INORM$ 0 or 1, normalize the fluxes so that the flux integrals between microgroups $IL0$ and IHI are equal for each region and normalized to $YNORM$.

IL0 Lowest microgroup (highest energy group) for flux normalization. If **IL0** = 0, it will be set to 1.
IHI Highest microgroup (lowest energy group) for flux normalization. If **IHI** = 0, it will be set to **IMAX**.
YNORM Normalization for flux. Will be set to 10.0 if $\text{YNORM} \leq 0$. This is where the flux plot will be normalized to.

Read the next three cards for each of the **NPLOT** plots. The plots are $\log \phi$ versus $\log E$.

Card F-2 Format (9A6)

TITLE(9) A 54 character title for the plot.

Card F-3 Format (8I5,2E10.0)

NCURV Number of curves on this plot.
LEBR 1, plot error bars on the fluxes.
IT0T 1, plot total cell averaged flux without error bars along with the other curves.
NRST 1, step normalization of each curve on the plot by a factor of 1.5.
IEL Lowest power of 10 for energy scale,
 If **IEL** = -8, then the following values will be used:
IEL = -3, **IEH** = 7, **IEYL** = 0, **IEYH** = 3.
IEH Highest power of 10 for energy scale.
IEYL Lowest power of 10 for flux scale.
IEYH Highest power of 10 for flux scale.
 Note, the flux is adjusted by the proper power of 10 so that all nonzero points are plotted.

DCDXI Number of inches per cycle (decade) for energy scale.

DCDYI Number of inches per cycle for the flux scale.

Card F-4 Format (14I5)

NRGN (NCURV) Tally region or flux average to be plotted. If the number is positive it refers to a tally region. If the number is negative it refers to a flux average.

End of the BMC code input.

VII. FIRST FLIGHT COLLISION PROBABILITY CALCULATIONS

First flight collision probabilities are used in resonance calculations to account for the space dependent contribution to the resonance integral. The BMC code can be used to calculate first flight collision probabilities and Dancoff correction factors. In this case, the code is not used to perform a Monte Carlo calculation but the geometry routines and ray tracing features are used to make the necessary numerical integrations.

The first flight collision probability, P_{ij} , is defined as the probability of a neutron being born in geometrical region i having its first collision in region j . It is usually assumed that the neutrons are uniformly distributed in the volume and have an isotropic angular distribution. It is usually necessary to make a numerical integration to obtain the values of P_{ij} as the analytical integrals usually can not be solved for any but the simplest geometries. The method used to make the numerical integrations will now be discussed.

Let a beam of π_0 neutrons start at a point going in the direction Ω and filling the volume cone $d\Omega$. After the beam goes a distance r_1 through a medium with a total cross section Σ_{t1} the number of neutrons left in the beam will be $\pi_1 = \pi_0 e^{-\Sigma_{t1} r_1}$. If the beam passes a second distance r_2 through a

region with total cross section Σ_{t2} , the neutrons left in the beam would be $\pi_2 = \pi_1 e^{-(\Sigma_{t1}r_1 + \Sigma_{t2}r_2)}$. The number of neutrons which collided in region 1 and 2, and hence the contribution of the beam to P_{11} and P_{12} are then $P_{11} = \pi_0 - \pi_1$ and $P_{12} = \pi_1 - \pi_2$, respectively.

Note that the beam strength π_i will never reach zero unless the beam escapes. Therefore, it is sometimes necessary or economical to discontinue the beam by some means. One way is to discontinue the beam when it is reduced below a minimum value. Another way is to play Russian Roulette with the beam when it is reduced below a minimum value.

The beam accounts for one solid angle from one space point. The first flight collision probability is the result of integrating over all angles and all space in the region. The approximate numerical integration is made by picking a finite number of volumes and using the center point of the volume as typical of all the volume. Then at each of these center points a number of solid angle center lines are picked and the beams for these angles are followed. The initial beam strength π_0 is picked to be proportional to the product of the volume and solid angle represented by the beam or $\pi_0 \propto \Delta V \cdot \Delta \Omega$.

The first flight collision probability for a neutron leaving a large region i and entering a small region j is usually calculated more accurately by application of the reciprocity theorem. The reciprocity theorem states that if the regions i and j have volumes V_i and V_j and total cross sections Σ_{Ti} and Σ_{Tj} , then

$$P_{ij} = \frac{V_j \Sigma_{Tj} P_{ji}}{V_i \Sigma_{Ti}}$$

where P_{ji} is the first flight collision probability for region j to region i .

VIII. BMCLIB: CROSS SECTION PREPARATION CODE

The **BMCLIB** code processes a nuclear data library tape written with the ENDF/B format and prepares the BMC cumulative cross section library tape (CCT). The following values are calculated and placed on the CCT:

- Micro-energy group averaged cross sections for:
 - Elastic scattering
 - Capture
 - Fission
 - Nu•fission
 - Inelastic scattering
 - n-2n scattering
 - Total
- Resonance parameter table information.
- Micro-energy group averaged anisotropic scattering data.
- Inelastic and n-2n scattering energy distribution data.

The cross-section averaging is done using a weighting function. The weighting function can be read in or selected from built in unity, $1/E$, f fission + $1/E$ or fission + $1/E$ + thermal Maxwellian weighting functions. The anisotropic scattering data is weighted by the product of the weighting function and the elastic scattering cross section.

The **BMCLIB** code is coded in Fortran V except for two minor subroutines which are coded in Sleuth. Three input-output binary tapes and four scratch drums or tapes are used. Many of the service routines and much of the coding came with but minor modification from the **ETOM**⁽⁷⁾ and **ETOE**⁽⁸⁾ codes. The sections for reading and processing the resonance and anisotropic data (ENDF/B files 2 and 4) came from the **ETOE** code while much of the rest came from the **ETOM** code.

TIMING

The **BMCLB** code is not real fast. The time spent in several phases of the calculation will be given for a case which processed the uranium-238 data from the ENDF/B library tape. There were 190 micro-energy groups. The weighting of the smooth cross sections was fission + $1/E$ + thermal Maxwellian. The resonance cross sections were weighted by $1/E \cdot (\sigma_T + \sigma_{IN})$. Uranium-238 would probably take the longest of any isotope to process. The times were:

Loading the BMCLB code	24 sec.
Initial input	2 sec.
Time to locate ^{238}U on library tape	68 sec.
Time for resolved resonance calculation	69 sec.
Time for unresolved resonance calculation	1 sec.
Time to process smooth cross sections	7 sec.
Time to process anisotropic data	2 sec.
Time to process inelastic and n-2n energy data	2 sec.
Total time for run was	3 min. 5 sec.

The time would be ~~much~~ shorter for an isotope near the beginning of the ENDF/B library tape which did not have resonance data.

B. RESONANCE CROSS SECTIONS

The resonance cross sections can either be group averaged and placed in with the smooth cross sections or they can be placed in a resonance parameter table with a "floor" being placed in with the smooth cross sections. The "floor" accounts for the resonances and resonance contributions not included in the resonance parameter table.

1. Resonance Parameter Table

The resonance parameter table contains resolved and unresolved resonance

parameters. The resolved resonances to be included in the resonance parameter table may be selected in several ways including maximum total or capture peak height. Usually the unresolved resonance parameters are not included in the resonance parameter table. If they are, the user must supply the parameters.

The resonance parameter table preparation routine determines what energy width around each resonance will be accounted for by the resonance parameter table. The resonance contribution outside of this energy width will be added to the smooth cross sections as a "floor." The potential scattering cross section is also included in the "floor."

If the unresolved resonances are to be included in the resonance parameter table the resonance parameters must be supplied by the user. The formulation is not the same as that on the ENDF/B library and will probably seldom be used.

2. Resolved Resonances

The resolved resonance contribution to the average group cross section is determined in one of two ways. If the resonance energy is more than 250 half widths outside the group, the contribution is integrated analytically. Otherwise, the resonance contribution is calculated for a number of points and a numerical integration is made.

The analytical integration* assumes that the flux is $1/E$ and that the resonance equations are not Doppler broadened. These should be good assumptions at more than 250 half widths away from the resonance energy.

The numerical integration is made by calculating the Doppler broadened resonance cross sections at a number of points using the resonances within 250 half widths of the micro-group boundaries. The point cross sections are then multiplied by the correct weighting function and numerically integrated.

*Derived by Dr. J. L. Carter, Jr., Battelle-Northwest Laboratories.

The points are selected using the subroutine EPT (ETQE) which picks the points so they are optimally placed for an integration of $(1 + x^2)^{-1}$ using logarithmic interpolation. Additional points are then added as needed so that the maximum energy spread between points is less than $\Delta E_i/N_p$ where $N_p = A + B \cdot E_{\min_i}$. The i is the group number and A and B are input numbers.

The weighting functions which can be used are either $1/E$ or $1/[E \cdot \sigma_T(E)]$ where $\sigma_T(E) = \sigma_{Tp}(E) + \sigma_{TIn} + \sigma_{TA}$. $\sigma_{Tp}(E)$ is the total cross section at each point, σ_{TIn} is the total cross section of the other isotopes in the material per atom of this isotope (an input number), and σ_{TA} is the analytical contribution from the other resonances of this isotope. Actually the point values are multiplied by unity or $1/\sigma_T(E)$ and the $1/E$ weighting is accounted for in the numerical integration. The integration assumes that the logarithm of the cross section varies linearly with the logarithm of the energy.

3. Unresolved Resonances.

The unresolved resonance contribution is calculated at a number of points using the equations as derived in the ETQE^() and MC2⁽⁻⁾ codes. The point values are then integrated numerically using linear log-log interpolation and assuming a $1/E$ flux. The point calculations account for Doppler broadening and use the narrow resonance approximation. The scattering from other isotopes in the material is again an input number.

C. ANISOTROPIC SCATTERING DATA

The BMCLB code first processes the anisotropic scattering data into group averaged Legendre coefficients in the center of mass system. The a 's as used by the BMC code are then obtained by matrix multiplication and normalized.

The normalization is the maximum value of $P(u)$ for $-1 \leq u \leq 1$ and is found by examining $P(u)$ at 101 equally spaced values of u . The a 's are also adjusted so that $P(u)$ is never negative and so that the number of random selections of u per acceptance will not average more than 20.

D. INELASTIC AND N-2N SCATTERING DATA

The BMC code uses the same energy distribution data for inelastic and n-2n scattering. However, the information for both inelastic and n-2n scattering are placed on the CCT tape by the **BMCLIB** code.

It is assumed that the energy distribution from inelastic scattering will be of two forms. The two forms are discrete excitation level and the evaporation model. One isotope can have both forms. The evaporation model is used above the energy range covered by the discrete energy level model.

The temperature coefficients for the evaporation model are calculated by fitting a cubic in velocity to the ENDF/B library values. The probability of exciting a discrete level is calculated by the **BMCLIB** code by integrating over energy the probability times the product of the inelastic scattering cross section and the flux weighting function.

E. Tape Requirements

The **BMCLIB** code requires the following binary tapes.

1. **BMCLIB** symbolic or absolute program tape.
2. The ENDF/B library tape or a library tape in the ENDF/B format. (Unit E)
3. The new CCT tape. (Unit C)
4. The old CCT tape. (Unit D)

An old CCT tape is not needed but will usually be used.

F. Card Input

The **BMCLIB** code requires a minimum of card input for most calculations. Several additional cards are needed when a resonance parameter table is being prepared. A number of options for the unresolved resonance calculation which were in the **ETOE** code are still available if desired. If these options are used a number of additional input cards are needed.

The card input will be listed along with the format and when the card is needed. The input cards all use Fortran formats.

Card 1	Format (12A6)	Needed for each isotope
	BCD(12)	BCD description of the isotope. Used as title for page headings.
Card 2	Format (10I5)	Needed only for first isotope.
	IDTAP	ENDF/B library tape identification number.
	IDCCT	Identification number for the new CCT.
	I0DCCT	OLD CCT tape number. If the old CCT tape is not
		material may be deleted from the old CCT.
	NEU	Number of energy sets to use. If I0DCCT $\neq$ 0 and NEU = -2 a second energy set will be added beside the one already on the old CCT. Otherwise NEU is inactive when IDCCT $\neq$ 0.

IMAX1 Number of energy points for the first energy set.
(the number of micro-groups +1)
If IEU1 = 1 and IMAX1 = 0, then IMAX1 will be
calculated.

(IMAX2 and IEU2 are active only if IBDCCT = 0 and NEU = 2, or if IBDCCT $\neq$ 2 and NEU = -2.)

IEU2 0, Read energies from cards.
 N, calculate the energies.

NMDLY The number of isotope materials to be deleted from
 the old CCT tape.

MATDLY_i $i=1, \dots, \text{NMDLY}$ ITMDLY_i	Material and temperature of the isotope-material to delete from the old CCT. If $\text{ITMDLY}_i = 0$, then any material with the number MATDLY_i will be deleted.
--	--

If two energy sets read card 3 again.

Card 4	Format (I5, 3E10.0)	Needed only if energy set to be calculated.
LEU		0, calculate energy points equally spaced in lethargy. >0, read LEU energy points from card 4A.
		Use ELU, EUU, and PTPUL if LEU = 0.
ELU		Lower limit of lethargy (highest energy) for this range.
EUU		Upper lethargy limit (lowest energy) for this range.
PTPUL		Number of points per unit lethargy.
Card 4A	Format (12A6)	Needed if LEU > 0
FØRIN (12)		Format for energies on card 4Bi
Card 4B	Format (FØRIN)	Needed if LEU > 0.
ESET (LEU)		The energies for this part of the energy range.
		Repeat cards 4, 4A and 4B for IEU energy ranges.
		If the last point of the previous range is lower or equal in energy to the first point of this range it will be removed.
		Use Cards 3 or 4 4A, 4B for energy set one and/or two.
Card 5	Format (I5, 1X, A6, 13, 515)	Always needed.
NATNØ		Material identification number on the ENDF/B format library. (NATNØ ≤ 0 signals the end of the case).
TNAME		6 character BCD name for this isotope - material.
MODE		0 or 1, library is on binary tape. 2, library is on cards as part of input.
MATID		The material-isotope identification number to use on the CCT tape. If MATID = 0 it will be set to MATNØ

LESET	0 or 1, use first energy set for this material. 2, use second energy set.
IW	The type of flux weighting to use for cross section averaging. 1, 1/E. 2, Unit 3, weighting function read in on cards 4, fission + 1/E. 5, fission + 1/E + thermal Maxwellian.
ITEMP	The temperature in degrees Kelvin. If ITEMP = 0 a temperature of 293°K will be used in BMC LIB but 0 will be put on the CCT tape as the temperature.
IPRNT	0, normal printout -1, short printout 1, long output 5, long output + resonance point values. (Warning, If 5 is used the printout and time are much longer)
Card 6 Format (1115)	Always needed
IRST	0, average resonance cross sections in with the smooth values. 1, prepare a resonance parameter table for the resolved resonances. 2, prepare a resonance parameter table for both the resolved and unresolved resonances.

MAXIP	The maximum group to allow inelastic or n-2n scattering in. If MAXIP = 0 then the largest group with data will be used. If MAXIP = -1 then no inelastic or n-2n data will be put on the CCT tape but will be accounted for in the elastic and capture cross section.
MXAN	Maximum group to allow anisotropic scattering in. 0 or -1 are the same as with MAXIP.
LTHRM	The method of calculating the thermal scattering cross sections. 0, use values on the library. 1, use the ideal gas model with temperature ITEMP below the energy ETHRM.
LANIS0	0, just write the a's for anisotropic scattering to the CCT tape. 1, also write the Legendre polynomials to the CCT Tape.
LINEL	Inactive
LN2N	Inactive
LNER	Number of points to fit to each resolved resonance for integration. (If LNER = 0, 51 points will be used.)
INT0P	0 or 1, use 1/E weighting for resolved resonance point intergration. 2, use $1/E \cdot [\sigma_T(E) + \sigma_{IN}]$

The next two input options are for the unresolved resonance calculation and are those which were already in the ETQE code

LGNO

Unresolved reduced neutron width option.

0, use ENDF/B library value.

1, read energies and Γ_n^0 for each energy and spin state.

2, also read the fission widths Γ_f .

LNEU

Energy point selection option for unresolved resonance calculation.

0, use the energies tabulated with the fission widths on the library or use 15 points equally spaced in lethargy.

1, read the energies.

N >1, use N energy points equally spaced in lethargy.

(If LGNO $\neq$ 0, then LNEU is set to 1.)

Card 7 Format (6E10.0)

Always needed.

APTPG, BPIPG

APTPG and BPIPG are used to determine the **maximum** distance between energy points for resonance integration.

$$(\Delta E_{\max} = \Delta E_{\text{group}} / \text{NPTPG},$$

$$\text{NPIPG} = \text{APIPG} + \text{BPIPG} * E_{\min \text{ group}}.)$$

SIGIN

The material cross section to add to the isotope resonance cross section when using INTOP = 2. (barns, $\Sigma_S \text{ material} / N_{\text{isotope}}$.)
If SIGIN = 0, it will be set to 10^5 .

SIGPIN

The material cross section to add to the total cross section for the unresolved resonance calculation. If zero it will be set to 10^5 .

ETHRM	Upper energy for the calculation of the ideal gas thermal scattering cross section when LTHRM = 1.
SIGFR	The free atom scattering cross section to use when LTHRM = 1. If SIGFR = 0, the library value at energy ETHRM will be used.
Cards 8 to 11 are used in preparing a parameter table.	
Card 8 Format (315)	Needed if IRST = 1 or 2.
LRSP	1, must be one. Type of resonance parameter table. 1 is the only option presently available.
LRSTU	<p>Index of how to select which resolved resonance on the library to place in the resonance parameter table. Only resonances with energies between ER2T and ER3T may be selected.</p> <p>0, use all the resonances (between ER2T and ER3T)</p> <p>1, use the NRSDU resonances nearest the energies specified on card 12.</p> <p>2, use all but the NRSDU resonances nearest the energies on card.</p> <p>3, use only the resonances which have a total peak height greater than PKUMN.</p> <p>4, use only the resonance which have a capture peak height greater than PKUMN.</p>
NRSDU	Number of resonance energies to read under options 1 or 2 of LRSTU.

Card9	Format (6E10.0)	Needed if IRST = 1 or 2.
ER3T		Lowest energy of the resolved portion of the resonance parameter table. (eV).
ER2T		Highest resolved and lowest unresolved energy.
ER1T		Highest unresolved energy.
XWDT		Number of half widths on each side of the resonance to be included in the resonance table. If XWDT = 0 it will be set to 200.
FNBT		The N_f value for fission to use in the resonance calculation.
PKVMN		The minimum peak cross section to accept a resonance under options 3 or 4 of LRSTU.
Cards 10 and 10A	Format (7E10.0)	Unresolved Parameters. Needed only if IRST = 2.
CNBT		$\bar{\Gamma}_{n_0}$, mean reduced neutron partial width (eV) ^{1/2} .
GGBT		$\bar{\Gamma}_{\gamma}$, mean radiative capture width (eV)
CFBT		$\bar{\Gamma}_f$, mean fission width
DBT		D, mean energy between resonances.
GBT		g, statistical weight
SPBT		a_p , potential scattering cross section (barns)
DSBT		D*, Level spacing correction term.
FLSBT		R*, spin state correction term.
Card 11,	Format (7E10.0)	Needed if LRSTU = 1 or 2.
ERUD (NRSDU)		The energies of resonances to use or not to use under options 1 or 2 of LRSTU. Need NRSDU values.

Cards 12 are used for the unresolved
resonance calculation under options of LGNO
and LNEU

Cards 12 and 12A **FORMAT** (I5/(6E12.0)) These cards needed if LNEU = 1
and/or LGNO $\neq$ 0.

NES1 Number of energy points to use for unresolved
resonance calculation.

NS1(NES1) The energies of the energy points. NES1 values.

Card 13 Format (6E12.4) Needed only if LGNO = 1.

GN0(NES1) The r_n^0 for each energy ES1.

Card 14 Format (6E12.4) Needed only if LGNO = 2

$\left. \begin{matrix} \text{GN0}_n \\ \text{GF}_n \end{matrix} \right\} n = 1, \text{NES1}$ The r_n^0 and r_f values at each energy ES1.
Read in pairs.

Repeat card 13 or 14 for each spin state.

Cards 15A, B and C are needed only if W = 3
and have the format of a ENDF/B TAB1 record.

Card 15 A Format (2E11.4, 4I11 , 14, I2, 13)

C1, C2, L1, L2 Not used.

N1 Number of interpolation table entries.

N2 Number of point value pairs E_n, Wt_n .

MAT, MF, MT Not needed

Card 15B Format (6I11)

$\left. \begin{matrix} \text{NBT} \\ \text{INT} \end{matrix} \right\}$ in pairs, Interpolation controls, see ENDF/B format
N1 pairs. specifications.

Card 15C Format (6E11.0)

$\left. \begin{matrix} x_i \\ y_i \end{matrix} \right\}$ in pairs Energy and weight values. See ENDF/B format
in pairs specifications.

G. Sample Input for the BMCLIB Code

The input needed to process the cross sections for isotopes ^{235}U and ^{232}U from the ENDF/B library is listed. A resonance parameter table would be prepared for ^{238}U . This input assumes that an old CCT tape is being used.

U-235 No Resonance Parameter Table.

300225162813	0	0	0	0	0	0	0
1044 U-235	0	1044	1	5	993	0	
0	5	6	5	6	0	1	0
						0	0
2.93	0.07	33027.0	33027.0	0.0	0.0		

U-238 With Resonance Parameter Table.

1047 U-238	0	1047	1	5	993	0	
1	5	6	5	6	0	1	0
						0	0
2.93	0.07	13.342	13.342	0.0	0.0		
1	4	0					
5.5	582.0	0.0	0.0	0.0	1000.0		

X. LIBR-LIBRARY UPDATE CODE

The purpose of this program is to make corrections or changes to the isotopes on the cumulative cross-section tape (CCT). The CCT is the library tape that is used by the BATTELLE MONTE CARLO CODE (BMC). Much computer time can often be saved by using the LIBR code to make corrections to the CCT tape instead of correcting the ENDF/B library and rerunning the BMCLIB code.

The format of the CCT tape is given here to help in making changes to it. The first three records on the CCT contain information about the tape and the energy used for the isotopes on the tape. These are called records A, B, and C. They only appear on the beginning of the tape. Following these, there is a maximum of 16 records for each isotope on the tape, depending on the isotope. The tape is ended with a special flag record and a file mark.

Tape Description and Energy Records

RECORD A	IDCCT	I.D. number of this tape.
	NEU	Number of energy sets used.
	IMAX1	Number of micro-group energy limits in the first energy set (NOTE: one more energy is needed than the number of micro-groups.)
	IMAX2	Same as IMAX1 but for energy group two.
RECORD B	ESET1(I), I=1, IMAX1	The micro-group energy limits in the first set.
RECORD C	ESET2(I), I=1, IMAX2	The micro-group energy limits in the second set.

A. ISOTOPE RECORDS

The first two words of each isotope record contain information about that record. These two words are:

	IRCD	= 1 First record of this isotope.
		=0 Present on all records except the first and last.
		= -1 Last record of this isotope
	NWRD	The number of words in this record.
RECORD 1	IRCD1	=1
	NWRD1	=11
	MATID	CCT material number of this isotope.
	MATNO	ENDF/B material number of this isotope.
	AWR	<i>Atomic</i> weight of isotope in neutron mass.
	ITEMP	Temperature in degrees K.
	LESET	Which energy set to use (1 or 2).
	MAXG	Number of energy groups.
	LFI	=0 non-fissionable isotope.
		=1 fissionable isotope.
	IRST	=0 resonance cross-sections averaged into groups.
		=1 special treatment for resolved resonances.
		=2 special treatment for both resolved and unresolved resonances.
	HOLN	6 holerith BCD name for this material.
	DATE A	6 holerith date when this isotope was processed by BMCLIB (30 JULY)

RECORD 2	DATE B	6 holerith date when this isotope was processed by BMCLIB (1969)
	SIGP	Sigma potential for this isotope
	(Present only if IRST $\neq$ 0)	
	IRCD2	0
	NWRD2	18
	LRSP	Type of resonance parameter table.
	IR1T	Lowest group number containing unresolved data
	IR2T	Lowest group number containing resolved data.
	IR3T	Highest group number containing resolved data.
	IFIST	=0 no fission =1 fission
	NEZT	Number of resolved resonances in the table.
	GNBT	Γ_{no}
	GGBT	Γ_{γ}
	GFBT	Γ_f
	FNBT	Ni value to use for resonance region.
	DBT	1.91322 D (average fuel spacing)
	GBT	$\sqrt{4g \sigma_p}$
	SPBT	σ_P
	CONBT	$3.297044 \times 10^{-4}/a$ (a = atomic wt.)

CONCT	$\sqrt{4.98173 \times 10^{-6} \left(\frac{a+1}{a}\right)^2} g$
DSBT	$1/2 D^*$
FLSBT	ℓ^*
XWDT	Bounds of half widths (usually 100)

RECORD 3

(Present only if IRST $\neq$ 0)

IRCD3	0
NWRD2	6 * NEZT
EZER(I)	$V_o^2 = 1.91322 E_o$
GNRR(I)	$4.9817303 * 10^6 g \left(\frac{a+1}{a}\right)^2 \frac{\Gamma_n^2}{V_o \Gamma^2}$
GGRR(I)	$4.9817303 * 10^6 g \left(\frac{a+1}{a}\right)^2 \frac{\Gamma_n \Gamma_\gamma}{V_o \Gamma^2}$
GFRR(I)	$\frac{\Gamma_1}{\Gamma_\gamma}$
GRR(I)	$\sqrt{4g \sigma_p} \text{ GNRR}$
GTRR(I), I=1, NEZT	$\frac{1.045358}{\Gamma}$

RECORD 4

(Present only if IRST $\neq$ 0)

IRCD4	0
NWRD4	IR3T-IR2T
IRPIT(I),	

I=IR2T, IR3T

Index of the resonance to use in group I.
If IRPIT(I) = N, use only the Nth resonance in the group. If IRPIT(I) = 0, do not use any resonance. If IRPIT(I) = -N, test the resonance starting with the Nth using XWDT.

RECORD 5	IRCD5	0
	NWRD5	8
	NOXS	=0 no scattering cross section >0 Scattering cross section present.
	NOXF	=0 no fission cross section >0 Fission cross section present
	NOXN	Number of groups with inelastic scattering
	IAN5	Maxim group number containing anisotropic data
	NLMAX	Number of Pl's at each energy
	LTANS	Type of anisotropic treatment
	LAANS	Number of A's at each energy (If LTANS = 1)
RECORD 6	IRCD6	0
	NWRD6	MAXG
	SIGT(I)	
	1=1,MAXG	Total cross sections
RECORD 7	IRCD7	0
	NWRD7	MAXG
	SIGS(I)	Scattering cross sectaons
	1=1,MAXG	
RECQRD 8	IRCD8	0
	NWRD8	MAXG
	SIGC(I)	Capture cross sections
	1=1,MAXG	

RECORD 9	(Present only if NOF $\neq 0$)
IRCD9	0
NWRD9	MAXG
SIGF(I)	Fission cross sections
	I=1,MAXG
RECORD 10	(Present only if NOF $\neq 0$)
IRCD10	0
NWRD10	MAXG
SIGFNU(I)	nu * fission cross sections
	1=1,MAXG
RECORD 11	(Present only if NOXIN $\neq 0$)
IRCD11	0
NWRD11	NOXIN
SIGIN(I)	Inelastic cross sections
	1=1,NOXIN
RECORD 12	(Present only if NON2N $\neq 0$)
IRCD12	0
NWRD12	NON2N
SIGN2N(I)	N2N cross sections
	1=1,NON2N
RECORD 13	(Present only if NLMAX > 0)
IRC(L)	0
NWR(L)	IANs * NLMAX
HL(I,L)	Legendre coefficients
	I=1,IANs,L=1,NLMAX
RECORD 14	(Present only if LAANS > 0)
IRCD14	0
NWRD14	LAANS * IANs

	AL(I,L)	AL's for BMC
	I=1,LAANS, L=1, LANS	
RECORD 15	(Present only if NOXIN $\neq$ 0)	
	IRCD15	0
	NWRD15	8 + (2*NVL)
	ITID1	555554
	ITIN1	Type of inelastic energy description
	NIST	Number of thetas for maximum calculations
	NVL1	Number of levels in discreet level table
	THETA1(I)	Terms of theta series
	I=1,4	
	ELVL(I)	Level energy
	PRB(I)	Probability for level I
	I=1,NVL	
RECORD 16	(Present only if NON2N $\neq$ 0)	
	IRCD16	-1 (only if last record)
	NWRD16	10
	ITID	555516
	ITIN	Type of N2N energy description
	NN2N	Number of theta coefficients
	NVL	Number of levels in discreet level table
	THETA(I)	Terms in theta series
	I=1,4	
	ELOW	Lowest energy
	DUMM	Dummy
<u>End of Isotope Record</u>		

Each isotope is ended with a three word record:

IRCF=-1, NWRC=1, (1).

End of Tape Record

The tape is ended with a 13 word record:

IRCF=-2, NWRC=11, (1,1,1,1,1,1,1,1,1,1,1,1,1).

This record is followed by a file mark.

The input to the LIBR code consists of 4 cards: A title card (Card 1), a control card (Card 2), a change card (Card 3, and a print control card (Card 4). The task which LIBR performs is controlled by the variable **CHNG** on card 2.

B. DESCRIPTION OF INPUT

CARD 1, the title card, is needed only once at the beginning of the input. It contains some description about the case.

CARD 2 is needed once for changing records A, B and/or C and once for each isotope that is to be changed or deleted.

CARD 3 is the change card and is needed only if **CHNG** is 1,2 or 5. As many cards may be used as are needed for making these changes. They use the namelist format. The namelist format is:

1. The first card must have the name \$CHANG in column 2 of the first of these cards.
2. Each number that is read in must be followed immediately by a comma. There may be no blank spaces between the number and the comma but there may be blanks between the comma and the next input name.
3. If a dimensioned variable is read in which the same value in consecution locations it may be read in as follows:

SIGS (2) = 10 * .56, where 10 is the number of consecutive locations which are to be filled with .56 starting at location 2.

4. If hollerith information is to be read in it must be as follows:

HOLN=6HIRON with no comma following the name and no other entries following on the same card.

5. After the last entry on the last card of namelist input for each case there must be a (\$) which signifies the end of namelist input for this case.

Card 4 is needed only if IKK is $\neq 0$. It is in namelist also and tells the names of the isotopes which are to be listed at the end of the run. **ICHNG** must be (0) for this option

Cards 2 and 3 are repeated until all changes and deletions have been made.

Changes must be made to the isotopes in the order that they appear on the tape.

Those records which are present depending on a condition of a certain variable are automatically deleted if that variable is changed.

(i.e., if **NONN** of record 5 is changed to (0), then records 12 and 16 are deleted.)

When using the option **ICHNG=4** no other option may be used in the same run.

C. TAPE USAGE

The LIBR code uses three tapes. The program tape on unit B, the old CCT library (CCT tape) on unit A, and the new CCT library (CCT tape) is written to unit E.

D. SAMPLE INPUT TO THE LIBR CODE

The following is a listing of input to the LIBR code which would perform a variety of tasks.

Change 1054, 2044. Delete 1051, 1020, 1013. Convert 304 to 399.

21054 0 Change Isotope 1054.

\$CHANG AWR=46.2, SIGP=1000.0, SIGS(4)=10*4.0, SIGF(4)=10*1.5,
ELVL(2)=7.5E+4, PRB(2)=2.5E-1, THETA(2)=133.0, .066, \$

3 1051 0 Delete Isotope 1051.

22044 0 ' Change Isotope 2044.

\$CHANG AWR=50.0, SIGP=5000.0, SIGC(15)=5*40.0, SIGFNU(3)=.11, \$

3 1020 0 Delete Isotope 1020.

3 1013 0 ' Delete Isotope 1013.

5 304 1 Convert Isotope 304 to 399.

\$CHANG MATID=399, HOLN=6HZIRC

0 0 3 No changes. List the 3 isotopes.

\$CHANG MATN(1)=1054, 2044, 1021, \$

INPUT TO LIBRPROBLEM I.D. AND CONTROL

CARD1 (72H) 72 hollerith case identification with a (0) zero in column one for program control of the printer.

CARD2 (315) ~~CHNG~~ =0 No changes to be made. A listing of the cross-section tape will be made (depending on the value of IKK).

=1 Records A, B, and C to be changed.

=2 Records 1-16 will be changed.

=3 An isotope will be deleted from the OCT tape.

=4 The isotopes from the CCT tape will be put onto the OCT tape. The OCT tape can either be a blank tape (IKK=-1) or another CCT tape (IKK=0). If a blank tape is used (IKK=-1) the energy records from the CCT tape are copied to the OCT tape. If OCT is a CCT tape (IKK=0) it is positioned to the end of tape record and the energy records on the CCT tape are skipped. The isotopes on the CCT tape are then copied to the OCT tape. If MATNAM is negative then only the isotope MATNAM is copied. If MATNAM is positive, all the isotopes starting with isotope MATNAM are copied.

=5 Isotope MATNAM will be put on the OCT tape and will also be converted to one or more different isotopes depending on IKK and these will be put

onto **OCT** tape also.

=6 Two or more isotopes will be combined into one isotope (inoperative at the present).

MATNAM

I.D. number of the isotope to be changed or deleted.

IKK

>0 The number of isotopes to be printed from the cross-section tape. If **ICHNG** = 5, **IKK** is the number of isotopes to be converted from isotope **MATNAM**.

=0 All the isotopes on the cross-section tape will be printed.

<0 Used only with **ICHNE** = 5 If **IKK** is minus then **MATNAM** will be changed before being written to tape and before conversions are made.

CHANGE CARDS

CARD 3

(NAMELIST)

These cards contain all the changes needed for **ICHNG** = 1, or **ICHNG** = 2 beginning with the name **\$CHANG** starting in column 2 of the first of these cards and a (\$) following the last change on the **last** card. As many cards may be used as is necessary.

CARD 4

(NAMELIST)

(Need only if **IKK** of the last control card is $\neq 0$)

MATN(I), I = 1, IKK

Material **I.D.** number of the isotopes to be listed.

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APPENDIX A

DIMENSIONING PARAMETERS FOR THE BMC CODE

APPENDIX A

DIMENSIONING PARAMETERS FOR THE BMC CODE

Most of the storage array dimensions used in the BMC Code are defined using PARAMETER variables. Also, most of the storage is defined in common blocks which are inserted into the subroutines using the INCLUDE statement.

The BMC code was designed so that the input routines except for subroutine MTPLC would seldom need the dimensions changed. The Monte Carlo routines and the output routines are more sensitive to problem size. To help in changing the dimensions to meet the requirements of different problems, the parameters used by the BMC code are listed here along with the minimum value they may have and the value included in the BMCI code.

Blank Common Parameters

<u>Parameter</u>	<u>BMC-I Value</u>	<u>Definition of Minimum Value</u>
PRI	190	IMAX, Number of energy micro-groups.
PRZIX	381	$Z \cdot \text{IMAX} + 1$
PRIE	190	Number of micro-graoups in talley set 1 (usually IMAX)
PRIET	64	Number of micro-groups in talley set 2.
PRJX	20	JMAX, Number of energy macro-groups.
PRJRX	8	Number of macro-graoups which contain resonance tallies (talley set 1)

<u>Parameter</u>	<u>BMC-I Value</u>	<u>Definition of Minimum Value</u>
PRJRX	2	Number of macro groups which contain resonance tallies. (Tally set 2)
PRFT	20	NFMAX, number of tally regions.
PRTTB	20	Total number of resonance isotopes in all tally regions.
PRL	4	NLKTLY, number of leakage tally sets.
IRM	6	MMAX, number of materials
PRFS	3	Number of materials containing fissionable isotopes.
PREL	22	Total number of elastic scatterers in all materials. (Heavy isotopes are combined in each material to form only 1 scatterer.)
PRFS	15	Number of elastic scattering probability sets. Obtain by adding for all materials the number of scatterers in the material minus 1.
PRANSX	2500	Total number of all anisotropic scattering coefficients.
PRIN	16	Total number of inelastic scatterers in all materials.
PRINV	56	Maximum micro-group which contains inelastic scattering cross sections.
PRINX	300	Sum of all items in all the inelastic scattering tables.

<u>Parameter</u>	<u>BMC-I Value</u>	<u>Definition of Minimum Value</u>
PRRIB	4	Number of resonance parameter tables.
PRRIS	8	Number of resonance isotopes in all materials.
PRNRX	60	Sum of the resonances in all resonance parameter tables.
PRIRX	250	Sum of all the micro groups encompassed by all of the resolved resonance parameter tables.

The total number of words in blank common (NBC) is given by the equation.

$$\begin{aligned}
 \text{NBC} = & 8 + \text{PR2IX} + 15 \cdot \text{PRM} + 9 \cdot \text{PREL} + 3 \cdot \text{PRIN} + \text{PRFT} + 3 \cdot \text{PRTTB} \cdot (\text{PRJRX} + \text{PRJRX}) \\
 & + 9 \cdot \text{PRRIS} + 19 \cdot \text{PRRTB} + \text{PRIRX} + 6 \cdot \text{PRNRX} + \text{PRINX} + \text{PRANSX} + \text{PRI} \cdot (2 \cdot \text{PRM} + \text{PRFS} + \text{PRES}) \\
 & + \text{PRIN} \cdot \text{PRINV} + \text{PRFT} \cdot (\text{PRIE} + \text{PRIET}) + 2 \cdot \text{PRJX} \cdot \text{PRL}
 \end{aligned}$$

INCM Parameters

PAR7	50	NHMAX, number of geometrical regions.
PAR8	20	JMAX
PAR9	2	NOMGX, number of surface reflection albedo sets.
PAR10	20	NFMAX or PRFT.
IRBG	3	The number of broad energy groups.

Note that BCDM assumes a maximum of 10 materials.

GMCM Parameters

PAR1	50	NHMAX, number of geometrical regions.
PAR2	200	Sum of all boundaries in all regions.
PAR3	50	NBNDX, number of boundaries.
PAR4	50	The last boundary of type 4 or 6.

<u>Parameter</u>	<u>BMC-I Value</u>	<u>Definition of Minimum Value</u>
PAR5	60	The last boundary of type 8.
PAR6	60	The last boundary of type 8.
PRSP	1	Number of special geometry regions.
PRNR	7	The dimensions of square array. Maximum of NRXS and NRYS
PRWSP	109	$PRSP \cdot (11 + 2 \cdot PRNR \cdot PRNR)$
<u>WRKN Parameters</u>		
PRCHT	40	The number of code check tallies.
<u>INOT Parameters</u>		
PNBH	50	Total number of batches in run which are not skipped.
PNIPM	10	Maximum number of isotopes per material.
PNMAT	10	MMAX
PJMXX	20	JMAX
PJMX1	21	JMAX+1
PRBBG	4	Number of putput averaging broad groups. (4 is maximum allowed for by FORMATS)
PIMX	191	IMAX+1
<u>Subroutine OUTPRP Parameters</u>		
PRRST	20	The same definition as PRTTB in blank common.
PITMX	64	The same definition as PRIET in blank common.
PRNLT	4	NLKTLY or PRL of blank common.

<u>Parameter</u>	<u>BMC-I Value</u>	<u>Definition of Minimum Value</u>
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Subroutine PHLST Parameter

PNLKT NLKTLY or PRL.

Subroutine PHPLOT Parameter

PIMX 191 IMAX+1

Subroutines BMCIN and CXPRP Parameters

PRBG1	4	NBGPX+1
PRIS	10	Maximum number of isotopes in any material.
PRMT	10	M MA X
PRIC	20	The total number of different isotopes in all materials.
PRJX1	21	JMAX+1
PRRT	6	Number of resonance parameter tables.
PRRP	250	Total number of resolved resonances.
PRRIT	1000	Total number of micro groups covered by resolved resonances.
PRINS	500	Sum of all inelastic scattering table entires .
PRANS	4500	Sum of all anisotropic scattering coefficients.

Note: The parameters for BMCIN and CXPRP were set large so they would seldom have to be changed.

Subroutine GETN Parameters

MXFSP	1000	NPTFB, number of particles per batch when running fission descendant problem.
MXBSP	5	Maximum size for Born Storage.

APPENDIX B

BMC CODE SAMPLE PROBLEM

APPENDIX B**BMC CODE SAMPLE PROBLEM**

This appendix contains the input and output for a sample problem run on the BMC code. The results are not meant to be final values but are used to show the input to the BMC code and the different values calculated by the BMC code.

The test problem is a cell containing a single clad fuel rod in light water. The cell is square with a lattice spacing of one inch. The outer cell boundaries are all mirror reflecting.

D H THOMSEN ROB-1 BMC SAMP

*X ASG, W-B=U0728 (IMTL)

~~SECRET~~ XQT CUR

IN B

~~XQT BMC~~

BMC SAMPLE PROBLEM. PCTR .9 PU-UO2, .5 IN ROD, 1 IN SQ. LAT. BNWL-88

728	0	9	9	9	9	9
10	20	U.	0	U.	U.	0
2813	0	0	0	18	3	3
1054000.	639280.	387740.	183160.	111090.	52475.	0
24788.	9118.8	3354.6	961.12	214.45	78.893	0
29.023	17.603	8.3153	2.3824	0.68256	U.	0

6 293 0 238. 0.86 PU-UO2 FUEL

1051 .00016964	1	0 0.	0	0
----------------	---	------	---	---

-----1053 .00001332-----1 0 0.-----0 0-----

1054	.00000109	1	0 0.	0	0
------	-----------	---	------	---	---

1044 .00004968 1 0 0. 0 0

2047	.02118	1	0 0.	0	0
------	--------	---	------	---	---

1013 .042827 0 1 2.0 0 0

1 293 0 238..... ZIRC CLAD .03 IN

1022 0,041418 1 0 0 0 0

4 293 0 238. WATER, 276.3 PPM B

1001	0.06686	0	1	2.	0	0
------	---------	---	---	----	---	---

1013 0.03343 0 1 2. 0 0

2001 0.06686 1 0 0. 1 1

1009	0.0000030448	1	0 0.	1	1
------	--------------	---	------	---	---

U-----BOUNDARY DATA

-5,0,0.2525, 5,0,0.2825, 1,3,-0.5, 1,3,0.5, 2,3,-0.5, 2,3,0.5, /

R

P

1 0 1 1 1 1 2 0 1

$$\begin{array}{cccccccccccc} 1 & 0 & 1 & 1 & 1 & 1 & 2 & 0 & 7 & & & & \\ 2 & 0 & 2 & 2 & 1 & 2 & 3 & 0 & -1 & 1 & 1 & 0 & / \end{array}$$
$$\begin{array}{ccccccccccccccccccccccccccccc} \cdot & 5 & 0 & 3 & 3 & -1 & 2 & 2 & 0 & -1 & 3 & 3 & 0 & 1 & 4 & 3 & 0 & -1 & 5 & 3 & 0 & 1 & 6 \end{array}$$

R

11

3*0.15, 3*0.30, 3*0.1, 3*0.2, /

— R —

1	0								
2	0	0	1	1.		1.			
2	1	0	0	0.		0.		0.	1.2827 0.

0	1	0	4	1	0
0.200296		0.050423		0.749281	

1	3	13	17	1	17	18	18
-1	0	1	50	110	0.		

PLOT OF FLUXES FOR SAMPLE PROBLEM - PCTR

$$2 \quad 1 \quad 1 \quad 0 \quad -8 \quad 0 \quad 0 \quad 0 \quad 2.5 \quad 3.0$$

13

```

*****
*                                     *
*               B.M.C.               *
*                                     *
*   BATTELLE MONTE CARLO   *
*                                     *
*****

```

THE COMPILATION AND LOAD TIME WAS 0 MIN. 9 SEC.

CARD 2	NPRBTP	728	IDENT. NO. OF PROBLEM TAPE.
LSTRT	0		0 = INITIAL START, IMCT=B
			1 = RESTART, NO CHANGES, IMCT=A (IF LCOPY = 1, IMCT=B)
			2 = RESTART BUT DO INPUT CALCULATIONS, IMCT=B
			3 = RESTART BUT DO INPUT CALCULATIONS EXCEPT FOR MATERIAL INFORMATION, IMCT=B
LCONT	0		0 OR 1 = GO ON TO MONTE CARLO AFTER BMCIN.
LSTOP	0		2 = GO ON TO BMCOUT AFTER BMCIN.
			0 = DO COMPLETE PROBLEM.
			1 = DO INPUT ONLY.
			2 = DO INPUT AND MONTE CARLO ONLY.
LCOPY	0		1 = COPY TAPE A TO TAPE B AND USE B. (ACTIVE ONLY IF LSTRT = 1.)
LSKNS	0		0 = FOR NORMAL POSITIONING TO THE LAST RESTART CASE. (ONLY IF LSTRT DOES NOT EQUAL 0.)
			1 = START THE TALLIES ANEW.
			2 = USE RESTART DATED (DLSKA AND DLSKB, THE TAPE DATES.)
LSKSV	0		MUST = 77 IF LSKNS DOES NOT = 0 OR LSKNS WILL NOT BE PERFORMED.
ICXTSV	0		0 = CROSS-SECTION TAPE WILL BE SAVED.
			1 = CROSS-SECTION TAPE WILL NOT BE SAVED.
IMCTSV	0		0 = MONTE CARLO TAPE WILL BE SAVED.
			1 = MONTE CARLO TAPE WILL NOT BE SAVED.
IFXTSV	0		0 = FLUX TALLY TAPE WILL BE SAVED.
			1 = FLUX TALLY TAPE WILL NOT BE SAVED.
NERCV	10		NO. OF RECOVERABLE ERRORS BEFORE TERMINATING. (IF EQUAL TO 0 NERCV WILL BE SET TO 10)

CARD 3	NBTCH	10	NUMBER OF BATCHES.
	NPTPB	250	NUMBER OF PARTICLES IN EACH BATCH.
	RTHRT	0.000000	THE RATIO OF WEIGHT LEAVING THE THERMAL-TALLY RANGE TO THAT OF ENTERING.
	ETHRT	0.000000	THE ENERGY TO WHICH NEUTRONS MUST BE SLOWED DOWN BEFORE THEY ARE USED FOR THE THERMAL TALLY RANGE.
	ETHRTX	0.000000	THE MAXIMUM ENERGY NEUTRONS MAY ACHIEVE WHILE CONTRIBUTION TO THE THERMAL FLUX TALLY.

CARD 4	IDCCT	2813	I.D. NO. OF THIS CCT.
	LPRB	0	0 = FISSION DESCENDENTS PROBLEM.
	IMPRT	0	1 = DIRECT SOURCE PROBLEM.
	LESET	0	0 = NO IMPORTANCE WEIGHTING.
	NBGPX	0	1 = IMPORTANCE WEIGHTING BY ZONE.
			1 = FIRST ENERGY SET USED.
			2 = SECOND ENERGY SET USED.
			NUMBER OF BROAD ENERGY GROUPS.
			(IF ONLY ONE, USE NBGPX = 0.)
	JMAXX	18	NUMBER OF MACRO ENERGY GROUPS.
	NHMAX	3	NUMBER OF REGIONS.
	PMAX	3	NUMBER OF MATERIALS.
	NSPRG	0	NUMBER OF SPECIAL REGIONS.
	NEMAX	3	NUMBER OF TALLY REGIONS.
	NLKTLY	0	NUMBER OF LEAKAGE TALLY SETS.
	LTALY	0	NUMBER OF SETS TO BE TALLIED.
	NBNDX	6	NUMBER OF BOUNDARIES.
	NOMGX	-0	NUMBER OF ALBEDO SETS.
	LESRC	-0	NORMALLY 0 IF LPRB IS 0, A NON-FISSION SOURCE CAN BE USED IF LESRC IS 1.

MINIMUM ENERGY OF EACH MACRO ENERGY GROUP.

GROUP	ENERGY	GROUP	ENERGY
1	1.054000+06	10	9.611200+02
2	6.392800+05	11	2.144500+02
3	3.877400+05	12	7.889300+01
4	1.831600+05	13	2.902300+01
5	1.110900+05	14	1.760300+01
6	5.247500+04	15	8.315300+00
7	2.478800+04	16	2.382400+00
8	9.118800+03	17	6.825600-01
9	3.354600+03	18	0.000000

MATERIAL DESCRIPTION

PU-UO2 FUEL

MATERIAL NO. 1, NO. OF ISOTOPES 6, TEMP, 293, ENERGY SET 1, HEAVY AWT. 238.0

ISOTOPE NO.	CONCENTRATION	IHVYM	LTHRMM	ETHRM	LANMM	LINMM
1051	1.696400-04	1	0	0.000000	0	0
1053	1.332000-05	1	0	0.000000	0	0
1054	1.090000-06	1	0	0.000000	0	0
1044	4.968000-05	1	0	0.000000	0	0
2047	2.118000-02	1	0	0.000000	0	0
1013	4.282700-02	0	1	2.000000+00	0	0

CLAD .03 IN

MATERIAL NO. 2, NO. OF ISOTOPES 1, TEMP. 293, ENERGY SET 1, HEAVY AWT. 238.0

ISOTOPE NO.	CONCENTRATION	IHVYM	LTHRMM	ETHRM	LANMM	LINMM
1022	4.141600-02	1	0	0.000000	0	0

6.3 PPM B

MATERIAL NO. 3, NO. OF ISOTOPES 4, TEMP. 293, ENERGY SET 1, HEAVY AWT. 238.0

ISOTOPE NO.	CONCENTRATION	IHVYM	LTHRMM	ETHRM	LANMM	LINMM
1001	6.686000-02	0	1	2.000000+00	0	0
1013	3.343000-02	0	1	2.000000+00	0	0
2001	6.686000-02	1	0	0.000000	1	1
1009	3.044800-06	1	0	0.000000	1	1

 THE FOLLOWING ISOTOPES HAVE BEEN LOADED FROM 881 TAPE NO. 2813

ISOTOPE NO.	ENDF/B NO.	TEMP.	NAME	DATE	SIGIN
1051	1051	293	PU-239	19 JUN 70	2.000000+03
1053	1053	290	PU-240	19 JUN 70	3.000000+04
1054	1054	0	PU-241	19 JUN 70	5.000000+05
1044	1044	293	U-235	12 JAN 70	2.500000+03
1013	1013	293	O-16	12 JAN 70	0.000000
2047	1047	293	U-238	03 SEP 69	7.970000+00
1022	1022	293	ZIRC	19 JUN 69	0.000000
1009	1009	293	B-10	31 DEC 68	0.000000
2001	2001	0	FIC-1	10 OCT 68	0.000000
1001	1001	293	H-1	27 SEP 68	0.000000

GEOMETRY INPUT

DIMENSIONS ARE IN IN.

BOUNDARY DATA

BOUNDARY	TYPE	ALBEDO	DESCRIPTION
1	5	0	CYLINDER, RADIUS= 2.525000-01, CENTERED AT THE ORIGIN.
2	5	0	CYLINDER, RADIUS= 2.825000-01, CENTERED AT THE ORIGIN.
3	1	3	PLANE, X=-5.000000-01
4	1	3	PLANE, X= 5.000000-01
5	2	3	PLANE, Y=-5.000000-01
6	2	3	PLANE, Y= 5.000000-01

REGION DATA

REGION	BOUNDARY	SPECIAL BOUNDARY	MATERIAL	TALLY SET	G*D	BH	01	02
1	1	0	1	1	1	1	2	0
2	2	0	2	2	1	2	3	0
					-1	1	1	0
3	5	0	3	3	-1	2	2	0
					-1	3	3	0
					1	4	3	0
					-1	5	3	0
					1	6	0	0

RUSSIAN ROULETTE DATA

REGION	MINIMUM WEIGHT	SURVIVAL WEIGHT	MINIMUM BEAM STRENGTH	SURVIVAL STRENGTH	IMPORTANCE WEIGHT
1	1.500000-01	3.000000-01	1.000000-01	2.000000-01	
2	1.500000-01	3.000000-01	1.000000-01	2.000000-01	
3	1.500000-01	3.000000-01	1.000000-01	2.000000-01	

THE RANDOM NO. USED TO START THIS CASE WAS 343277244615
THE BMC INPUT TIME WAS 0 MIN. 14 SEC., THE TOTAL TIME HAS BEEN 0 MIN. 23 SEC.

THE SOURCE ROUTINE INPUT.

DIMENSIONS ARE IN CENTIMETERS.

REGION 1 LRGGS= 2 NEGS= 0 LANGS= 0 IGRGS= 1 PRBRS= 1.0000+00 WTPRS= 1.0000+00
LRD= 2 LZD= 1 IXPM= 0 IYPM= 0 XZ= 0.0000 ZZ= 0.0000 R1= 0.0000 R2= 1.2827+00 Z1= 0.0000 Z2=-0.0000

MONTE - CARLO CODE CHECK TALLIES.
FOR BATCH NO. 1
LAST RANDOM NO. USED 214306650551

CTIFF	2.500000+02	INITIAL FISSION PARTICLES.
CTIW	2.500000+02	INITIAL WEIGHT.
CTIC	2.500000+02	INITIAL COORDINATES.
CTALR	3.912392+00	WEIGHT LOST BY RUSSIAN ROULETTE.
CTPL	0.000000	PARTICLES LOST, LEAKAGE.
CTLL	0.000000	WEIGHT LOST, LEAKAGE.
CTALR	6.023682+00	BEAM STRENGTH LOST BY RUSSIAN ROULETTE.
CTPTC	2.361500+04	PARTICLES TO COLLISION.
CTWTC	1.295039+04	WEIGHT TO COLLISION.
CTPFC	2.336500+04	PARTICLES FROM COLLISION, SCATTERING.
CTWFC	1.270431+04	WEIGHT FROM COLLISION, SCATTERING.
CTFP	2.400000+02	FISSION PARTICLES PRODUCED.
CTFPE	2.430688+02	FISSION PARTICLES EXPECTED.
CTIFI	3.300000+01	PARTICLES FROM INELASTIC SCATTERING.
CTWFI	3.080314+01	WEIGHT FROM INELASTIC SCATTERING.

THE TIME USED FOR BATCH 1 WAS 1 MIN. 6 SEC. THE TOTAL TIME HAS BEEN 1 MIN. 29 SEC.

JMC SAMPLE PROBLEM. PCTR .9 PU-U02, .5 IN ROD, 1 IN SQ. LAT. BNWL-887

25 JUN 70

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MONTE - CARLO CODE CHECK TALLIES.
FOR BATCH NO. 10
LAST RANDOM NO. USED 244644572161

CTIFP	2.500000+02	INITIAL FISSION PARTICLES.
CTIw	2.500000+02	INITIAL WEIGHT.
CTIC	2.500000+02	INITIAL COORDINATES.
CTWLR	3.213571+00	WEIGHT LOST BY RUSSIAN ROULETTE.
CTPLL	0.000000	PARTICLES LOST, LEAKAGE.
CTALL	0.000000	WEIGHT LOST, LEAKAGE.
CTBLR	-7.363991-01	BEAM STRENGTH LOST BY RUSSIAN ROULETTE.
CTPTC	2.320800+04	PARTICLES TO COLLISION.
CTATC	1.276831+04	WEIGHT TO COLLISION.
CTPFC	2.295800+04	PARTICLES FROM COLLISION, SCATTERING.
CTWFC	1.252152+04	WEIGHT FROM COLLISION, SCATTERING.
CTFP	2.600000+02	FISSION PARTICLES PRODUCED.
CTFPE	2.565890+02	FISSION PARTICLES EXPECTED.
CTPFI	4.000000+01	PARTICLES FROM INELASTIC SCATTERING.
CTWFI	3.774978+01	WEIGHT FROM INELASTIC SCATTERING.

THE TIME USED FOR BATCH 10 WAS 1 MIN. 4 SEC. THE TOTAL TIME HAS BEEN 11 MIN. 43 SEC.

THE TOTAL TIME IN THE MONTE CARLO WAS 11 MIN. 0 SEC.

MONTE - CARLO CODE CHECK TALLIES.

AVERAGED OVER 10 BATCHES

	AVERAGE	ERROR	
CTIFP	2.500000+02	.000	INITIAL FISSION PARTICLES.
CTIW	2.500000+02	.000	INITIAL WEIGHT.
CTIC	2.500000+02	.000	INITIAL COORDINATES.
CTWLR	-5.177671-01	13.408	WEIGHT LOST BY RUSSIAN ROULETTE.
CTPL	0.000000	.000	PARTICLES LOST, LEAKAGE.
CTLL	0.000000	.000	WEIGHT LOST, LEAKAGE.
CTBLR	1.128917+00	2.475	BEAM STRENGTH LOST BY RUSSIAN ROULETTE.
CTPTC	2.469630+04	.036	PARTICLES TO COLLISION.
CTATC	1.327004+04	.012	WEIGHT TO COLLISION.
CTPFC	2.444630+04	.036	PARTICLES FROM COLLISION, SCATTERING.
CLFC	1.301953+04	.012	WEIGHT FROM COLLISION, SCATTERING.
CTFP	2.488000+02	.028	FISSION PARTICLES PRODUCED.
CTFPE	2.486578+02	.023	FISSION PARTICLES EXPECTED.
CTPFI	4.210000+01	.299	PARTICLES FROM INELASTIC SCATTERING.
CTAFA	3.982251+01	.281	WEIGHT FROM INELASTIC SCATTERING.

LISTING OF INPUT NEEDED FOR THE OUTPUT CODES.

```

CARD 1  NBCHSK      0      NO. OF BATCHES TO SKIP.
        LCXST       1      >0= PROCESS THE CROSS-SECTIONS.
        ----- 0= DO NOT PROCESS THE CROSS-SECTIONS.  --
        ----- 1= PRINT ALL REACTION RATES.
        ----- 2= SUPPRESS CROSS-SECTION AVERAGES.
        ----- 3= PRINT REGION REACTION RATES ONLY.
        LFXST       0      0= PRINT OUT ALL FLUXES.
        ----- 1= SUPPRESS GROUP FLUX RATIO MATRIX, ---
        ----- 2= SUPPRESS GROUP AND REGION FLUX RATIO MATRIX,
        ----- 3= PRINT REGION FLUX RATIO MATRIX ONLY*
        NEBG        4      NO. OF BROAD GROUPS.  --
        ICPLT       1      >0= CALL PLOTTING CODE.
                        (IF +3 IS USED THE FLUXES WHICH
                        ARE USED WILL NOT BE PRINTED.)
        NPAV        0      NUMBER OF FLUX AVERAGES.
CARD 2  VOL          REGION VOLUMES. (NFMX OF THEM.)
        1  2.002960-01  2  5.042300-02  3  7.492810-01
CARD 3  JEBG1,JEBG2  LOWER AND UPPER LIMITS OF THE GROUPS. (NEBG OF THE; )
        GROUP 1      1      3
        GROUP 2      13     17
        GROUP 3       1     17
        GROUP 4      18     18
    
```

MINIMUM ENERGY OF EACH MACRO ENERGY GROUP.

GROUP	ENERGY	GROUP	ENERGY
1	1.054000+06	10	9.611200+02
2	6.392800+05	11	2.144500+02
3	3.877400+05	12	7.889300+01
4	1.831600+05	13	2.902300+01
5	1.110900+05	14	1.760300+01
6	5.247500+04	15	8.315300+00
7	2.478800+04	16	2.382400+00
8	9.118800+03	17	6.825600-01
9	3.354600+03	18	0.000000

FLUX IN CM. PER SEC. FOR REGION 1, MATERIAL NO. 1, WITH A VOLUME OF 2.002960-01

J	PHI	ERROR	PHI/VOL.	PHI/VOL./EV.	J	PHI	ERROR	PHI/VOL.	PHI/VOL./EV.
1	4.092906+02	.126	2.04343+03	2.28418-04	10	5.823241+01	.219	2.90732+02	1.21468-n1
2	1.307224+02	.050	6.52646+02	1.57370-03	11	6.400111+01	.301	3.19533+02	4.27944-n1
3	8.323540+01	.503	4.15562+02	1.65207-03	12	3.987373+01	.112	1.99074+02	1.46856+00
4	1.026464+02	.031	5.12473+02	2.50500-03	13	4.310441+01	.197	2.15204+02	4.31529+00
5	4.726816+01	.269	2.35992+02	3.27448-03	14	2.097380+01	.124	1.04714+02	9.16935+00
6	5.454948+01	.349	2.72344+02	4.64632-03	15	3.471210+01	.128	1.73304+02	1.86595+01
7	4.744723+01	.201	2.36886+02	8.55584-03	16	4.500172+01	.189	2.24676+02	3.78695+01
8	5.034142+01	.344	2.51335+02	1.60401-02	17	5.312081+01	.194	2.65212+02	1.56021+02
9	5.079608+01	.280	2.53665+02	4.39966-02	18	5.926438+02	.057	2.95884+03	4.33492+03

FLUX IN CM. PER SEC. FOR REGION 2, MATERIAL NO. 2, WITH A VOLUME OF 5.042300-02

J	PHI	ERROR	PHI/VOL.	PHI/VOL./EV.	J	PHI	ERROR	PHI/VOL.	PHI/VOL./EV.
1	9.294142+01	.247	1.84323+03	2.06040-04	10	1.583128+01	.058	3.13969+02	1.31177-n1
2	2.922287+01	.085	5.79554+02	1.39746-03	11	1.724504+01	.061	3.42007+02	4.58044-n1
3	1.947634+01	.130	3.86259+02	1.53558-03	12	1.046861+01	.033	2.07616+02	1.53158+00
4	2.330539+01	.065	4.62198+02	2.25925-03	13	1.133736+01	.086	2.24845+02	4.50862+00
5	1.194582+01	.050	2.36912+02	3.28725-03	14	5.394673+00	.404	1.06988+02	9.36851+00
6	1.308690+01	.089	2.59542+02	4.42792-03	15	9.020269+00	.487	1.78892+02	1.92612+01
7	1.177357+01	.042	2.33496+02	8.43342-03	16	1.295939+01	.037	2.57013+02	4.33200+01
8	1.254042+01	.080	2.48704+02	1.58722-02	17	1.336746+01	.058	2.65106+02	1.55960+02
9	1.249687+01	.060	2.47841+02	4.29965-02	18	1.590430+02	.020	3.15418+03	4.62110+03

FLUX IN CM. PER SEC. FOR REGION 3, MATERIAL NO. 3, WITH A VOLUME OF 7.492810-01

J	PHI	ERROR	PHI/VOL.	PHI/VOL./EV.	J	PHI	ERROR	PHI/VOL.	PHI/VOL./EV.
1	1.215310+03	.028	1.62197+03	1.81306-04	10	2.252765+02	.047	3.00657+02	1.25615-n1
2	3.742447+02	.099	4.99472+02	1.20436-03	11	2.595137+02	.032	3.46350+02	4.63860-n1
3	2.728298+02	.099	3.64122+02	1.44757-03	12	1.672235+02	.034	2.23179+02	1.64638+00
4	3.231264+02	.058	4.31249+02	2.10797-03	13	1.656477+02	.030	2.21076+02	4.43304+00
5	1.578275+02	.030	2.10639+02	2.92270-03	14	8.235218+01	.223	1.09908+02	9.62419+00
6	1.899380+02	.062	2.53444+02	4.32472-03	15	1.242767+02	.029	1.65861+02	1.78582+01
7	1.097074+02	.031	2.26494+02	8.18051-03	16	2.000372+02	.038	2.66972+02	4.49986+01
8	1.916296+02	.040	2.55751+02	1.63219-02	17	2.055299+02	.047	2.74303+02	1.61370+02
9	1.834506+02	.040	2.44836+02	4.24752-02	18	2.713577+03	.040	3.62157+03	5.30587+03

FLUX PER CM.**2 PER SEC. AS A FUNCTION OF REGION FOR MACRO-GROUP 1

REGION	FLUX	ERROR	FLUX/EV.	REGION	FLUX	ERROR	FLUX/EV.
1	2.04343+03	.063	2.28418-04	3	1.62197+03	.037	1.81306-04
2	1.84323+03	.049	2.06040-04				

FLUX PER CM.**2 PER SEC. AS A FUNCTION OF REGION FOR MACRO-GROUP 2

REGION	FLUX	ERROR	FLUX/EV.	REGION	FLUX	ERROR	FLUX/EV.
1	6.52646+02	.252	1.57370-03	3	4.99472+02	.133	1.20436-03
2	5.79554+02	.169	1.39746-03				

FLUX PER CM.**2 PER SEC. AS A FUNCTION OF REGION FOR MACRO-GROUP 3

REGION	FLUX	ERROR	FLUX/EV.	REGION	FLUX	ERROR	FLUX/EV.
1	4.15562+02	.251	1.65207-03	3	3.64122+02	.133	1.44757-03
2	3.86259+02	.258	1.53558-03				

FLUX PER CM.**2 PER SEC. AS A FUNCTION OF REGION FOR MACRO-GROUP 4

REGION	FLUX	ERROR	FLUX/EV.	REGION	FLUX	ERROR	FLUX/EV.
1	5.12473+02	.156	2.50500-03	3	4.31249+02	.077	2.10797-03
2	4.62198+02	.129	2.25925-03				

FLUX PER CM.**2 PER SEC. AS A FUNCTION OF REGION FOR MACRO-GROUP 5

REGION	FLUX	ERROR	FLUX/EV.	REGION	FLUX	ERROR	FLUX/EV.
1	2.35992+02	.134	3.27448-03	3	2.10639+02	.041	2.92270-03
2	2.36912+02	.099	3.28725-03				

FLUX PER CM.**2 PER SEC. AS A FUNCTION OF REGION FOR MACRO-GROUP 6

REGION	FLUX	ERROR	FLUX/EV.	REGION	FLUX	ERROR	FLUX/EV.
1	2.72344+02	.174	4.64632-03	3	2.53494+02	.083	4.32472-03
2	2.59542+02	.176	4.42792-03				
1	2.72344+02	.174	4.64632-03	3	2.53494+02	.083	4.32472-03
2	2.59542+02	.176	4.42792-03				

TOTAL FLUX CM/SEC
9.63092+03 .034

REGION FLUX/CM**2/SEC

REGION	TOTAL		GROUP 1		GROUP 2		GROUP 3		GROUP 4	
1	9.62556+03	.092	3.11164+03	.085	9.83109+02	.210	6.66672+03	.106	2.95884+03	.029
2	9.54835+03	.061	2.80905+03	.047	1.03285+03	.023	6.39418+03	.071	3.15418+03	.040
3	9.63790+03	.028	2.48556+03	.036	1.03812+03	.005	6.01633+03	.040	3.62157+03	.053

REGION FLUX RATIO MATRIX

REGION	REGION 1	REGION 2	REGION 3
1	-1.00000 .000	1.00809 .006	.99872 .009
2	.99198 .008	-1.00000 .000	.99071 .006
3	1.00128 .009	1.00938 .006	-1.00000 .000

REGION FLUX RATIO MATRIX / GROUP 4

REGION	REGION 1	REGION 2	REGION 3
1	-1.00000 .000	.93807 .005	.81700 .006
2	1.00602 .006	-1.00000 .000	.87094 .009
3	1.22398 .010	1.14818 .012	-1.00000 .000

 GROUP FLUX RATIO MATRIX / REGION 1

GROUP	GROUP 1	GROUP 2	GROUP 3	GROUP 4
1	-1.0000+00 .000	3.1651+00 .089	4.6674-01 .069	1.0516+00 .032
2	3.1595-01 .089	-1.0000+00 .000	1.4747-01 .027	3.3226-01 .071
3	2.1425+00 .032	6.7813+00 .123	-1.0000+00 .000	2.2532+00 .049
4	9.5089-01 .290	3.0097+00 .064	4.4382-01 .097	-1.0000+00 .000

GROUP FLUX RATIO MATRIX / REGION 2

GROUP	GROUP 1	GROUP 2	GROUP 3	GROUP 4
1	-1.0000+00 .000	2.7197+00 .080	4.3931-01 .060	8.9058-01 .204
2	3.6769-01 .109	-1.0000+00 .000	1.6153-01 .032	3.2745-01 .066
3	2.2763+00 .031	6.1908+00 .122	-1.0000+00 .000	2.0272+00 .040
4	1.1229+00 .026	3.0539+00 .062	4.9329-01 .097	-1.0000+00 .000

GROUP FLUX RATIO MATRIX / REGION 3

GROUP	GROUP 1	GROUP 2	GROUP 3	GROUP 4
1	-1.0000+00 .000	2.3943+00 .038	4.1314-01 .036	6.8632-01 .179
2	4.1786-01 .067	-1.0000+00 .000	1.7255-01 .015	2.8665-01 .036
3	2.4205+00 .021	5.7954+00 .051	-1.0000+00 .000	1.6612+00 .032
1	-1.0000+00 .000	2.3943+00 .038	4.1314-01 .036	6.8632-01 .179
2	4.1786-01 .067	-1.0000+00 .000	1.7255-01 .015	2.8665-01 .036
3	2.4205+00 .021	5.7954+00 .051	-1.0000+00 .000	1.6612+00 .032
4	1.4570+00 .038	3.4886+00 .044	6.0196-01 .117	-1.0000+00 .000

 CROSS-SECTIONS AND REACTION RATES FOR REGION 1

PU-UO2 FUEL

** CROSS-SECTIONS FOR ISOTOPE 1051, PU-239, MATERIAL 1, REGION 1, WITH A CONCENTRATION OF 1.696400-04 *

GROUP	TOTAL			SCATTER			CAPTURE			FISSION			NU * FISSION			INELASTIC			N-2N		
1	7.44585+00	.008		4.38209+00	.006		7.18848-03	.169		1.94040+00	.003		6.26469+00	.012		1.12257+00	.005		3.59465-03	.617	
2	7.44310+00	.007		4.91648+00	.008		6.76159-02	.033		1.68562+00	.001		5.02671+00	.004		7.73385-01	.005		0.00000	.000	
3	8.45344+00	.016		6.07334+00	.019		1.09153-01	.005		1.56107+00	.002		4.59000+00	.007		7.09876-01	.015		0.00000	.000	
4	1.00265+01	.001		7.88068+00	.014		1.66896-01	.005		1.47999+00	.000		4.30400+00	.001		4.98918-01	.017		0.00000	.000	
5	1.13216+01	.001		9.25315+00	.008		2.07310-01	.003		1.50423+00	.001		4.34702+00	.002		3.56938-01	.007		0.00000	.000	
6	1.25699+01	.002		1.06434+01	.002		2.27764-01	.006		1.42007+00	.002		4.08573+00	.005		2.78663-01	.009		0.00000	.000	
7	1.32091+01	.002		1.13095+01	.001		4.47669-01	.041		1.22074+00	.016		3.50617+00	.045		2.31193-01	.003		0.00000	.000	
8	1.45738+01	.004		1.17913+01	.000		8.44303-01	.027		1.77094+00	.037		5.08351+00	.105		1.67236-01	.010		0.00000	.000	
9	1.56888+01	.003		1.23461+01	.001		1.38622+00	.010		2.15649+00	.027		6.18917+00	.076							
10	1.93135+01	.006		1.29379+01	.000		2.74468+00	.018		3.63094+00	.045		1.04207+01	.013							
11	3.11526+01	.035		1.39310+01	.005		6.92590+00	.107		1.02958+01	.021		2.95489+01	.061							
12	4.22121+01	.035		1.28795+01	.007		8.92754+00	.257		2.04051+01	.018		5.85626+01	.050							
13	7.85144+01	.152		1.45152+01	.016		2.82512+01	.055		3.57480+01	.105		1.02597+02	.030							
14	6.69554+01	.101		1.09653+01	.009		2.40613+01	.019		3.19288+01	.105		9.16366+01	.303							
15	1.58728+02	.058		1.13151+01	.014		5.37762+01	.234		9.36369+01	.346		2.68743+02	.099							
16	5.03129+01	.375		9.87465+00	.025		1.62945+01	.188		2.41437+01	.185		6.92943+01	.531							
17	5.34702+01	.096		1.15005+01	.004		1.14685+01	.029		3.05011+01	.064		8.75423+01	.183							
18	9.79963+02	.066		1.13698+01	.002		2.89456+02	.023		6.79138+02	.043		1.94912+03	.012							

REACTION RATE FOR ISOTOPE 1051, PU-239

REACTION TYPE	TOTAL		GROUP 1		GROUP 2		GROUP 3		GROUP 4	
TOTAL	1.03501+02	.006	8.02095-01	.218	2.61296+00	.088	4.97899+00	.099	9.85217+01	.069
SCATTER	3.00712+00	.018	4.99040-01	.135	3.90803-01	.081	1.86405+00	.025	1.14307+00	.011
CAPTURE	3.01330+01	.021	3.53979-03	.125	8.36595-01	.343	1.03227+00	.034	2.91008+01	.022
ABSORPTION	1.00371+02	.007	1.97688-01	.054	2.22216+00	.082	2.99216+00	.084	9.73786+01	.068
FISSION	7.02377+01	.045	1.94148-01	.053	1.38556+00	.049	1.95989+00	.051	6.82778+01	.047
N*FISSION	2.01638+02	.013	6.11252-01	.169	3.97665+00	.140	5.68036+00	.148	1.95957+02	.014
INELASTIC	1.22534-01	.031	1.05117-01	.030	0.00000	.000	1.22534-01	.031	0.00000	.000
N2N	2.49584-04	.474	2.49584-04	.474	0.00000	.000	2.49584-04	.474	0.00000	.000
FLUX	1.92796+03	.018	6.23248+02	.170	1.96913+02	.042	1.33532+03	.021	5.92644+02	.057

TOTAL REACTION RATE FOR REGION 1, VOLUME 2.002960-01 PU-UO2 FUEL

REACTION TYPE	TOTAL	GROUP 1	GROUP 2	GROUP 3	GROUP 4
TOTAL	8.16840+02 .050	1.78071+02 .049	8.43335+01 .198	4.73576+02 .062	3.40265+02 .028
SCATTER	6.08442+02 .042	1.39895+02 .040	6.71996+01 .158	4.07547+02 .052	2.00396+02 .020
CAPTURE	8.35061+01 .073	1.36295+00 .042	1.53546+01 .059	2.29774+01 .069	6.00287+01 .042
ABSORPTION	1.71736+02 .011	5.44013+00 .164	1.71339+01 .062	2.93665+01 .076	1.40369+02 .010
FISSION	8.82295+01 .052	4.07719+00 .130	1.77927+00 .052	6.38913+00 .165	8.10404+01 .057
N*FISSION	2.47308+02 .015	1.16780+01 .039	4.94322+00 .147	1.80875+01 .049	2.20220+02 .016
INELASTIC	3.64343+01 .089	3.25080+01 .089	0.00000 .000	3.64343+01 .089	0.00000 .000
N2N	2.28294-01 .515	2.28294-01 .515	0.00000 .000	2.28294-01 .515	0.00000 .000
FLUX	1.92796+03 .018	6.23248+02 .170	1.96913+02 .042	1.33532+03 .021	5.90644+02 .057

REGION REACTION RATE / UNIT VOLUME FOR REGION 1

REACTION TYPE	TOTAL	GROUP 1	GROUP 2	GROUP 3	GROUP 4
TOTAL	4.07817+03 .025	8.89041+02 .247	4.21044+02 .099	2.36438+03 .031	1.70379+03 .014
SCATTER	3.03772+03 .021	6.98441+02 .200	3.35501+02 .079	2.03472+03 .026	1.00299+03 .010
CAPTURE	4.16913+02 .036	6.80466+00 .212	7.66597+01 .297	1.14717+02 .035	3.00196+02 .021
ABSORPTION	8.57409+02 .052	2.71605+01 .082	8.55429+01 .309	1.46615+02 .038	7.10793+02 .049
FISSION	4.40496+02 .026	2.03558+01 .065	8.88322+00 .259	3.18984+01 .082	4.00527+02 .028
N*FISSION	1.23471+03 .007	5.83038+01 .197	2.46796+01 .074	9.03036+01 .247	1.10441+03 .008
INELASTIC	1.81902+02 .044	1.62300+02 .044	0.00000 .000	1.81902+02 .044	0.00000 .000
N2N	1.13978+00 .257	1.13978+00 .257	0.00000 .000	1.13978+00 .257	0.00000 .000

TOTAL REACTION RATE FOR REGION 3, VOLUME 7.492810-01, 276.3 PPM B

REACTION TYPE	TOTAL		GROUP 1		GROUP 2		GROUP 3		GROUP 4	
TOTAL	1.23226+04	.012	6.42469+02	.082	1.16729+03	.005	4.05138+03	.019	8.27126+03	.126
SCATTER	1.22454+04	.012	6.41802+02	.082	1.16537+03	.005	4.04857+03	.019	8.19683+03	.125
CAPTURE	7.69982+01	.117	4.31071-01	.324	1.92916+00	.019	2.56714+00	.035	7.44310+01	.118
INELASTIC	2.36284-01	.570	2.36284-01	.570	0.00000	.000	2.36284-01	.570	0.00000	.000
FLUX	7.22150+03	.021	1.86238+03	.027	7.77844+02	.036	4.50792+03	.030	2.71358+03	.040

REGION REACTION RATE / UNIT VOLUME FOR REGION 3

REACTION TYPE	TOTAL		GROUP 1		GROUP 2		GROUP 3		GROUP 4	
TOTAL	1.64460+04	.016	8.57447+02	.109	1.55789+03	.007	5.40702+03	.025	1.10389+04	.017
SCATTER	1.63429+04	.016	8.56557+02	.109	1.55531+03	.007	5.40328+03	.025	1.09396+04	.017
CAPTURE	1.02763+02	.016	5.75313-01	.433	2.57468+00	.025	3.42613+00	.046	9.93366+01	.157
INELASTIC	3.15348-01	.761	3.15348-01	.761	0.00000	.000	3.15348-01	.761	0.00000	.000

CELL REACTION RATE

REACTION TYPE	TOTAL		GROUP 1		GROUP 2		GROUP 3		GROUP 4	
TOTAL	1.32698+04	.012	8.54940+02	.112	1.26513+03	.006	4.61419+03	.023	8.65558+03	.129
SCATTER	1.29793+04	.012	8.12762+02	.106	1.24604+03	.006	4.54152+03	.022	8.43779+03	.127
CAPTURE	1.62041+02	.016	1.82887+00	.069	1.73134+01	.060	2.60892+01	.068	1.35952+02	.015
ABSORPTION	2.50271+02	.020	5.90606+00	.190	1.90926+01	.062	3.24783+01	.076	2.17792+02	.020
FISSION	8.82295+01	.052	4.07719+00	.130	1.77927+00	.052	6.38913+00	.165	8.18404+01	.057
N*FISSION	2.47308+02	.015	1.16780+01	.039	4.94322+00	.147	1.80875+01	.049	2.29220+02	.016
INELASTIC	3.99667+01	.097	3.60404+01	.097	0.00000	.000	3.99667+01	.097	0.00000	.000
N2N	2.31609-01	.525	2.31609-01	.525	0.00000	.000	2.31609-01	.525	0.00000	.000

JMC SAMPLE PROBLE . PCTR .9 PU-U02, .5 IN ROD, 1 IN SQ. LAT. BNWL-887

25 JUN 70

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THERMAL (BG-4) ETA * F 1.05247+00 .0044

K-INFINITE= 9.89075-01 .0472

K-EFFECTIVE= 9.89075-01 .0472

LISTING OF INPUT NEEDED FOR THE FLUX CALCULATIONS.

CARD 1 NPLOT -1 0= NO PLOTTING.
 +1= PLOTTING WILL BE DONE ON THE CALCOM.
 -1= PLOTTING WILL BE DONE BY PRINTER.
 ITYPE 0 TYPE OF NORMALIZATION. (AT PRESENT ONLY 0 OR 1 IS USED.)
 INORM 1 >0= NORMALIZE FLUXES TO YNORM.
 ILO 50 LOWEST ENERGY GROUP FOR AVERAGING FLUX.
 IHI 110 HIGHEST ENERGY GROUP FOR AVERAGING FLUX.
 YNORM 1.000000+01 NORMALIZATION FACTOR, (IF .LE. 0 10.0 WILL BE USED.)

NORMALIZATION FACTORS FOR EACH REGION

REGION	RNORM
1	4.524592+00
2	1.179131+00
3	1.750581+01

MICRO-GROUP FLUXES.

GROUP	EMID	REGION 1	REGION 2	REGION 3
1	9.12500+06	2.64647+00	1.182	1.80092+00 .838 1.75760+00 .784
2	8.306500+06	5.75762+00	1.601	4.63316+00 1.503 4.61070+00 1.361
3	7.330450+06	1.05620+01	.282	8.16742+00 1.858 7.32204+00 1.839
4	6.469100+06	1.05069+01	.238	8.89412+00 1.892 8.80421+00 2.005
5	5.706950+06	1.54199+01	.311	1.27382+01 .252 1.25710+01 .229
6	5.038150+06	2.31555+01	.389	2.08533+01 .371 1.73561+01 .302
7	4.446150+06	3.12432+01	.524	2.80620+01 .417 2.35303+01 .361
8	3.923700+06	3.42000+01	.288	3.07755+01 .243 2.88706+01 .226
9	3.462650+06	3.48133+01	.271	2.98170+01 .249 2.64741+01 .184
10	3.055750+06	4.62649+01	.420	3.82510+01 .285 3.32215+01 .260
11	2.896700+06	6.80271+01	.575	5.75215+01 .605 5.04983+01 .377
12	2.379850+06	6.39332+01	.617	5.82910+01 .628 5.06195+01 .443
13	2.100200+06	6.37167+01	.453	5.36601+01 .329 4.79126+01 .267
14	1.853400+06	5.76202+01	.402	5.16540+01 .354 4.31894+01 .256
15	1.635600+06	7.04757+01	.308	5.86596+01 .249 5.17287+01 .234
16	1.443450+06	6.79878+01	.350	5.72608+01 .378 5.18243+01 .286
17	1.273850+06	6.09145+01	.469	5.70262+01 .379 4.71470+01 .181
18	1.124150+06	5.73793+01	.305	5.33370+01 .219 4.86791+01 .157
19	9.926700+05	5.39829+01	.348	4.53496+01 .292 3.66780+01 .185
20	8.754950+05	5.74473+01	.331	4.76871+01 .274 4.26804+01 .197
21	7.726250+05	6.16169+01	.404	5.43340+01 .255 4.82554+01 .113
22	6.618400+05	5.83842+01	.290	5.11529+01 .201 4.36343+01 .235
23	5.617200+05	4.56615+01	.384	3.77111+01 .279 3.65623+01 .145
24	5.310150+05	4.28742+01	.256	3.88657+01 .347 3.54969+01 .258
25	4.686200+05	2.91968+01	.307	2.88248+01 .331 2.69571+01 .177
26	4.135500+05	2.96266+01	.283	2.69088+01 .263 2.58250+01 .137
27	3.649600+05	3.40763+01	.290	2.95549+01 .204 2.75521+01 .110
28	3.226750+05	3.82192+01	.165	3.29276+01 .163 2.94375+01 .174
29	2.842300+05	2.65322+01	.215	2.59240+01 .125 2.35448+01 .143
30	2.506350+05	3.15083+01	.262	2.69045+01 .219 2.49072+01 .065
31	2.213600+05	2.88631+01	.170	2.24909+01 .183 2.23173+01 .121
32	1.953500+05	2.25313+01	.152	2.05270+01 .120 2.01029+01 .085
33	1.723950+05	2.27976+01	.204	2.13671+01 .196 1.88744+01 .111
34	1.521350+05	2.23464+01	.257	2.16437+01 .205 1.75349+01 .107
35	1.342600+05	1.84498+01	.267	1.89319+01 .164 1.78603+01 .084
36	1.184850+05	2.00865+01	.265	1.92076+01 .185 1.79469+01 .071
37	1.045635+05	2.16216+01	.236	1.73454+01 .149 1.68636+01 .097

LISTING OF INPUT NEEDED FOR THE PLOTTING CODE.

CARD 1	TITLE	PLOT OF FLUXES FOR SAMPLE PROBLEM - PCTR	
CARD 2	NCURV	3	NUMBER OF CURVES TO PLOT.
	LEBR	1	>0= PLOT ERROR BARS.
	ITOT	1	>0= PLOT TOTAL CELL AVERAGES ALSO.
	NRST	0	>0= STEP NORMALIZATION BY 1.5
	IELX	-3	LOWEST POWER OF 10 FOR ENERGY.
			(IF IELX IS -8, IELX= -3)
	IELH	7	HIGHEST POWER OF 10 FOR ENERGY.
			(IF IELX IS -8, IELH= 7)
	IEYL	0	LOWEST POWER OF 10 FOR Y SCALE.
			(IF IELX IS -8, IEYL= 0)
	IEYH	3	HIGHEST POWER OF 10 FOR Y SCALE.
			(IF IELX IS -8, IEYH= 3)
	DCDXI	2.50	NUMBER OF INCHES PER CYCLE FOR X SCALE.
	DCDYI	3.00	NUMBER OF INCHES PER CYCLE FOR Y SCALE.
CARD 3	NRGN	REGION NUMBERS TO PLOT.	
		1	2
		3	

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