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THE MCNP CODE

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I. Introduction

The MCNP code is the major Monte Carlo coupled neutron-photon transport research tool at the Los Alamos National Laboratory, and it represents the most extensive Monte Carlo development program in the United States which is available in the public domain. The present code is the direct descendent of the original Monte Carlo work of Fermi, von Neumaum, and Ulam at Los Alamos in the 1940s. Development has continued uninterrupted since that time, and the current version of MCNP (or its predecessors) has always included state-of-the-art methods in the Monte Carlo simulation of radiation transport, basic cross-section data, geometry capability, variance reduction, and estimation procedures. The authors of the present code have oriented its development toward general user application. The documentation,[1] though extensive, is presented in a clear and simple manner with many examples, illustrations, and sample problems. In addition to providing the desired results, the output listings give a wealth of detailed information (some optional) concerning each stage of the calculation. The code system is continually updated to take advantage of advances in computer hardware and software, including interactive modes of operation, diagnostic interrupts and restarts, and a variety of graphical and video aids.

Since MCNP was released to the public a few years ago, its use has become widespread, particularly in U.S. Fusion Energy programs. The code is distributed for Los Alamos by the Radiation Shielding Information Center (RSIC) at Oak Ridge in the U.S. and through RSIC by the NEA Data Bank at Saclay in France for the European Economic Community countries. Any inquiries concerning MCNP should be directed to either of these installations.

In addition to providing timely updates to the distributed versions of the code and its documentation, Los Alamos actively promotes MCNP in other ways. There is an MCNP newsletter issued occasionally which gives a variety of up-to-date information. Los Alamos has also sponsored seminar-workshops on MCNP, and RSIC has 25 one-hour video tapes of the latest sessions. (Much of this presentation is taken from this seminar.) In the literature MCNP is well represented for use in both practical applications and in verification of new Monte Carlo developments.[2] The review by Carter and Cashwell,[3] although written on general Monte Carlo methods, is closely tied to MCNP, and the 1980 RSIC seminar workshop proceedings[4] contain several informative papers on MCNP and its applications.

II. General Features of the Code

This section outlines the important features of MCNP. Most, but not all of these items are part of the standard code package distributed by RSIC and the Data Bank. Nonstandard items must be utilized as "patches" (updates), and the distribution centers should be contacted concerning their current state of availability.

II.A. Machine Hardware

The RSIC code package contains MCNP version 3 which is written in Fortran 77. The code is designed to be as machine independent as possible and runs on CRAY, CDC, IBM, and VAX computers, among others. There is a separate RSIC package for batch processing of MCNP on the IBM 3033. A Fortran 77 compiler is necessary. A systems guide is provided giving detailed instructions for implementation of the code on each machine.[5]

II.B. Modes of Particle Simulation

MCNP has three modes of operation: (1) neutron only, (2) gamma ray only, and (3) coupled neutron-secondary gamma ray. Problems may be run stationary or time dependent with fixed sources or, for neutrons, with subcritical, critical, or supercritical fission sources. Thermal neutrons may be treated with either the free gas or $S(\alpha,\beta)$ models. There are also two gamma-ray options: (1) a simple mode which includes only Compton (Klein-Nishina) scattering, pair production, and the photoelectric effect (absorption), and (2) a more detailed treatment which includes several low energy (to 1 keV) effects such as coherent and incoherent scattering and fluorescent emission.

The MCNP code solves only the forward transport equation. There exists a multigroup "patch" for MCNP (the MCMG code) which has both forward and adjoint modes of operation.

III. Cross-Section Data

The MCNP cross-section treatment is continuous in energy with linear interpolation between specific energies such that the original data are reproduced in most cases to within 0.5%. Resolved resonance data are Doppler broadened to a specific temperature. All reaction types from the basic data are included for neutron energies from 20 MeV to 10^{-5} eV and for gamma rays from 100 MeV to 1 keV. Processed cross-section files have been prepared by Los Alamos for distribution in the RSIC code package, and data from these files are accessed by a nuclide or material identification number for use in the MCNP problem setup. In addition to the continuous data, a discrete neutron cross-section file is also available. Here the continuous data have been averaged with flat weighting into 262 groups (increased recently from 240 groups for better thermal energy resolution). Only the cross sections are discretized, the scattering energy and angular distributions being the same as for the continuous data file. Computer memory requirements for the discrete data are usually less than half that for the more exact treatment. The disadvantage with the discrete data is that there could be significant self-shielding in resonance energy regions or other problems associated with cross-section averaging. It should be emphasized that MCNP uses this discrete data as if it were continuous data that just happens to be constant over some energy interval. Data such as this are sometimes referred to as "pseudo-multigroup" or "pseudo-point" and should not be confused with true multigroup cross-section data.

The nuclides and materials for which both continuous and discrete neutron data exist in the RSIC distribution of MCNP are shown in Table 1. Gamma-ray data are available for all nuclides through atomic number 94. All data are identified in Appendix E of the MCNP Systems Guide by the form

ZZZAAA.nnx

where ZZZ is the atomic number, AAA is the mass number, nn is an evaluation identifier (see Appendix F in the MCNP manual), and x is C for continuous neutron data, D for discrete neutron data, P for gamma-ray data, T for $S(\alpha,\beta)$ data, and Y for dosimetry data. The fission products are given some appropriate average values of ZZZ and AAA, and AAA=000 indicates the natural mixture of isotopes. The $S(\alpha,\beta)$ data are

LWTR.07T	^{1}H in light water ($^{1}H_2O$)
HWTR.01T	^{2}H in heavy water ($^{2}H_2O$)
POLY.03T	^{1}H in polyethylene (CH_2)
BENZ.01T	^{1}H and ^{12}C in benzene
GRPH.01T	^{12}C in graphite
BE.01T	^{9}Be in solid beryllium
BEO.01T	molecular BeO
H/ZR.01T	^{1}H in Zr H_x
ZR/H.01T	Zr in Zr H_x

Table 1. Nuclides, Elements, and Materials for Which Neutron Cross-Section Data are Available in the MCNP Code Package from RSIC

ZZZ	Material	ZZZ	Material
1	^{1}H, ^{2}H, ^{3}H	28	Ni, ^{58}Ni
2	He, ^{3}He, ^{4}He	29	Cu
3	^{6}Li, ^{7}Li	31	Ga
4	^{9}Be	40	Zr
5	B, ^{10}B, ^{11}B	41	^{93}Nb
6	C, ^{12}C	42	Mo
7	^{14}N	45	Fission Products, ^{235}U
8	^{16}O	46	Fission Products, ^{239}Pu
9	^{19}F	48	Cd
11	^{23}Na	50	Sn
12	Mg	50	Fission products, average
13	^{27}Al	56	^{138}Ba
14	Si	63	Eu
15	^{31}P	64	Gd
16	^{32}S	67	^{165}Ho
17	Cl	73	^{181}Ta
18	Ar	74	W, ^{182}W, ^{183}W, ^{184}W, ^{186}W
19	K	78	Pt
20	Ca	79	^{197}Au
22	Ti	82	Pb
		83	^{209}Bi
23	V	90	^{232}Th
		91	^{233}Pa
24	Cr	92	^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{237}U, ^{238}U, ^{239}U, ^{240}U
		93	^{237}Np
25	^{55}Mn	94	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu
26	Fe	95	^{241}Am, ^{242m}Am, ^{243}Am,
27	^{59}Co	96	^{242}Cm, ^{244}Cm

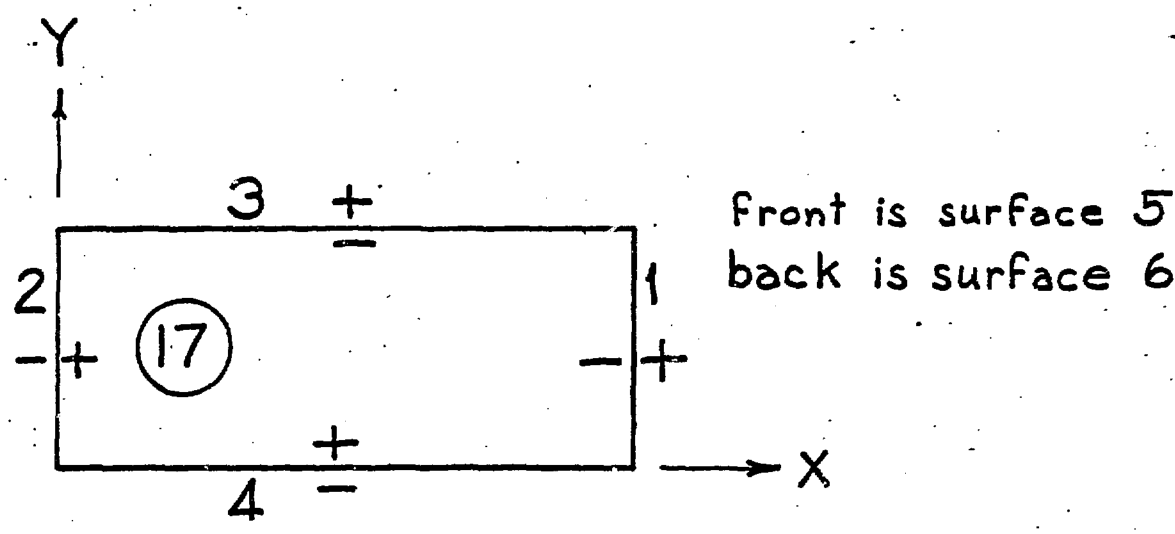
All MCNP cross-section data distributed outside the United States by RSIC are from ENDF/B-IV or earlier evaluations. Distribution of cross-section data for MCNP users in the United States includes a library based on ENDF/B-V. Users wishing any additional data must contact RSIC or the Data Bank or generate it themselves by running NJOY,[6] the basic processing code for MCNP cross-section data. This code and any available basic evaluated data sets can also be obtained from either distribution center. However, the degree of difficulty in the implementation and application of NJOY is comparable to that for MCNP itself, and any discussion of NJOY is more applicable to a course dealing with cross-section processing methods.[7]

IV. Geometry Description

The basic unit of MCNP geometry is a "cell." In addition to their use in specifying a particular mixture of nuclides, elements, or materials, cells are used to divide the system into importance regions for variance reduction and can be used to collect results (tallies). Thus, several adjacent cells may contain the same mixture but have different importance assignments, or one importance region may have several cells of different mixtures.

A cell is created from regions of space delimited by surfaces of first or second algebraic order and certain surfaces of fourth order (tori). In this manner, the MCNP geometry combines the features of other Monte Carlo code geometries by first creating regions of space with surfaces and then combining them combinatorially. A rectangular parallelepiped cell composed of six surfaces is shown in Fig. 1. The first surface, PX 20, in the MCNP input format, indicates a plane at $x=20$; the last, PZ 0, is a plane surface at $z=0$. The diagram gives the position of each surface and its relationship (+ or - depending on the value of the equation) to the interior and exterior of the cell, here numbered 17. The

RECTANGULAR CELL 17



Input to MCNP for cell 17

```
17   material   density   -1  2  -3  4  -5  6
```

Input for surfaces of cell 17

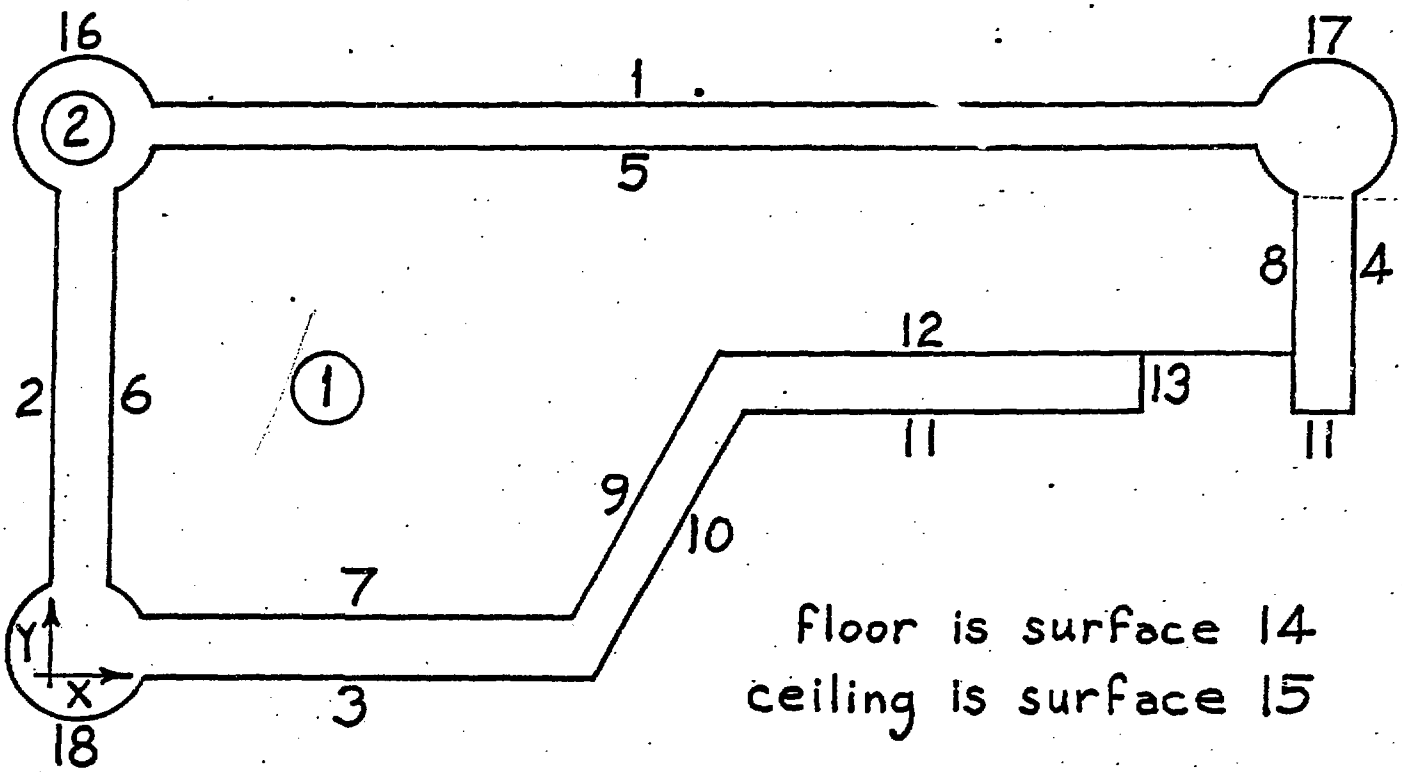
```
1   PX   20
2   PX    0
3   PY    8
4   PY    0
5   PZ    3
6   PZ    0
```

Fig. 1. Simple MCNP Geometry.

input card image for cell 17 gives first the cell number, the material number from the cross-section mixture, and the density of this mixture in cell 17. The next six numbers indicate that cell 17 is that space defined to the left of surface 1, to the right of surface 2, below surface 3, above surface 4, in front of surface 6, and behind surface 5.

A more complicated input is shown in Fig. 2. Cell 2 is the wall surrounding a room with three cylindrical columns, an offset wall, and an open door. Cell 1 is the room interior and is the space to the right of surface 6, below surface 5, to the left of surface 8, above surface 7, both to the left of surface 9 and above surface 12 (indicated by the : sign), outside (on the positive side of) the three cylinders, below the ceiling, and above the floor. The vertical cylinders are given as, for example, surface 16, $(X\text{-}15)^2 + (Y\text{-}290)^2 = 40^2$, and for a plane surface parallel to the z axis, number 9, $2X - Y = 540$. The #N symbol indicates outside of (or not in) cell N, and #() indicates outside of the space defined by the surfaces in the parenthesis. Then cell 2 is the space to the right of surface 2, below surface 1, to the left of surface 4, above surface 3, inside (all space on the negative side of) the three cylinders, outside cell 1, outside the door opening, not in the space left due to the wall offset, below the ceiling, and above the floor.

There are other ways to specify the geometry in Fig. 2, and the MCNP manual gives numerous other examples to demonstrate the flexibility of geometry input. Cells may be described in the simplest orientation possible and then rotated or translated to fit into the complete system. There are also many diagnostic and graphical (in the form of 2D geometry slices) aides for geometry debugging and verification and also a method for irregular volume calculation. A feature now under development is a general finite lattice array capability.



cells

```
1    m  d   6 -5 -8 7(12:-9)16 17 18 14 -15
2    m' d'  (2 -1 -4 3:-16:-17:-18) #1
            #(13 -12 -8) #(-11 10) 14 -15
```

surfaces

1	PY	300	10	P	2 -1 0 600
2	PX	0	11	PY	140
3	PY	0	12	PY	170
4	PX	690	13	PX	580
5	PY	280	14	PZ	0
6	PX	30	15	PZ	310
7	PY	30	16	C/Z	15 290 40
8	PX	660	17	C/Z	675 290 40
9	P	2 -1 0 540	18	C/Z	15 15 42.7

Fig. 2. Complicated MCNP Geometry.

V. Source Specification

MCNP has five standard source options. They are:

1. point isotropic,
2. outward cosine distribution on a spherical surface,
3. inward cosine distribution on a spherical surface,
4. uniform volume distribution, and
5. plane source.

Standard energy and time distributions are monoenergetic, linear, Gaussian, Maxwellian, and evaporation spectra.

For any source that cannot be described by these options, a special source routine must be supplied by the user. In this routine the position, energy, direction, time, weight, cell, and surface (if applicable) must be specified for each particle. For most problems this source routine will be the only user interaction necessary with the code other than standard input data. The manual gives examples of a source routine (Chapters 3 and 4) and its incorporation into the code (Appendix C). The March 18, 1982 (2C), December 20, 1982 (2D), and September 1, 1983 (Version 3) newsletters (included in the RSIC package) describe options added to MCNP since the manual was published. The MCNP systems guide gives general updating information for the different operating systems.

Figures 3 and 4 give an example of a nonstandard card input and patch for a uniform volumetric source. RANG gives a uniform random number $0 \leq RN \leq 1$. Isotropic directions are set before the source routine is called and in this case are not changed. The source energies are selected from the probabilities (see input cards at top of Fig. 4) by calling SRCSMP(K). The surface JSU is set to zero to indicate that particles are not started on a surface, but inside the cell.

Two rectangular sources
Cell 5 if $Y<0.0$
Cell 6 if $Y>0.0$

Both sources are isotropic emittors. They emit particles at different energies,

Cell 5: Two discrete energies, E1 and E2 with probabilities P1 and P2

Cell 6: Three discrete energies, E3 E4 and E5 with probabilities P3, P4 and P5

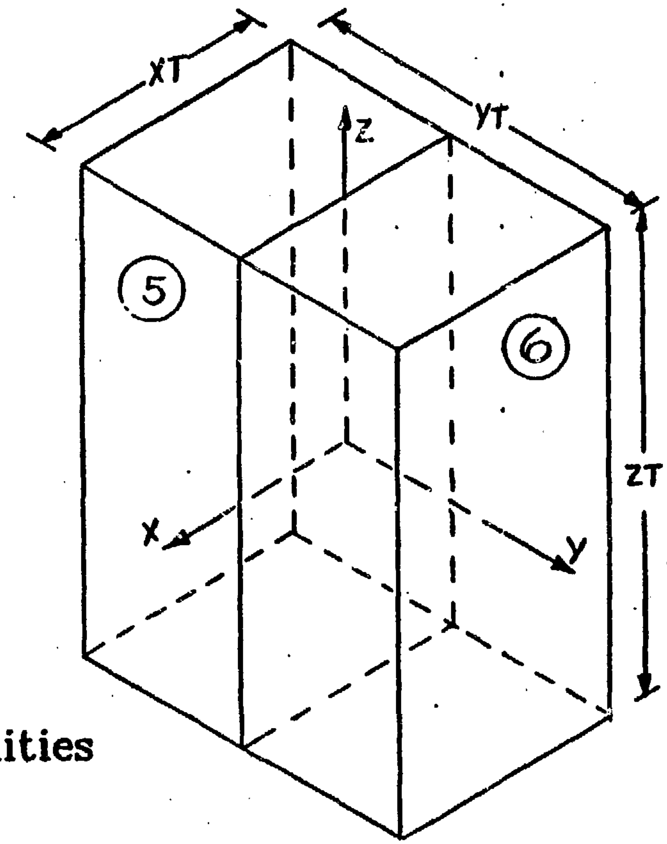


Fig. 3. Non-Standard MCNP Source Description.

INPUT DATA

```
SRC        XT   YT   ZT   1.0   0.0
SI2         L E1 E2
SP2         D P1 P2
SI3        L E3 E4 E5
SP3         D P3 P4 P5
```

For SOURCE patch

```
         RN=RANG()
         XXX=SRC(1)*(RN-0.5)
         RN=RANG()
         YYY=SRC(2)*(RN-0.5)
         RN=RANG()
         ZZZ=SRC(3)*(RN-0.5)
C       NOW SAMPLE ENERGY
         K=3
         IF(YYY.LE.0.0) K=2
         ERG=SRCSMP(K)
C       NOW SET CELL NUMBER
         ICL=6
         IF(K.EQ.2) ICL=5
         WGT=SRC(4)
         TME=SRC(5)
         JSU=0
```

Fig. 4. MCNP Source Patch.

Source energy and angular biasing are also available by input option for any source description. The energy input is the biased source spectrum. The angular input gives the biased probabilities of emission into fixed solid angles (cones) or as a continuous exponential function.

For criticality eigenvalue calculation of k_{eff} (KCODE), all source points must be input by card image or from a file from a previous calculation. None of the standard or user supplied sources are applicable here. A patch to MCNP exists for calculating the α eigenvalue.

If a user is supplying a nonisotropic angular source and any next-event estimation is used (point detectors or DXTRAN), a SRCDX routine must also be written to give the uncollided contribution from the source. The outline of the procedure is given in Chapter 4 of the manual.

A much more generalized source capability is available as a patch,[8] which is scheduled to become a standard part of Version 3A of MCNP early in 1985. This new source will replace the five standard sources now available and can be used to describe just about any source imaginable without having to modify the code.

VI. Variance Reduction

Although MCNP is a general code, its development, use, and documentation have been oriented toward deep penetration and other shielding calculations. As a result, there are many standard variance reduction techniques available by input option (these options are, in general, applicable to reactor calculation also). Individually, most of these techniques are simple in concept and simple to implement. The difficulty arises in deciding when and in what combination to use them, and an even greater difficulty is the determination of the input importance parameters.

Variance reduction can be applied in conjunction with the source, transport, collision, or estimation (tally) processes. In addition to the standard source biasing in the last section, the user can add any additional procedures he desires. Some estimation processes, such as for point detectors, are actually variance reduction schemes. But the main emphasis in MCNP has always been in the transport process with geometry splitting and Russian roulette. The real work here is in the geometry input where the system is first subdivided into cells describing the material mixtures and/or importance regions. Then each cell must be assigned a relative importance (for both neutrons and gamma rays) such that the ratio of importances between two adjacent cells determines the degree of splitting and Russian roulette a particle is subjected to when crossing from one cell to another. The objective here is to keep the particle population relatively constant as it moves through the system toward the regions of interest. An example of an MCNP printout illustrating this effect is given in Section VIII. This boundary procedure is probably the variance reduction technique a new user should first learn to perfect, and for systems without great geometric complexity this can usually be done with a few short calculations by observing the printout as described and adjusting the cell boundaries and their importances.

Boundary splitting and Russian roulette is one of several options which are subsets of a more general feature referred to in the MCNP literature as the "weight window." Here the particle weight is forced to remain within two spatial and energy dependent values (a window) by splitting and Russian roulette. This option, or one of its less general forms, is helpful in calculations where several weight altering schemes are used in combination, and particle weights can take on extremely large or small values. The small values contribute little to the results and waste computer time; whereas, large values can contribute to prohibitively high statistical uncertainty. The task of setting the weight window parameters as a function of space and energy can be difficult for complicated problems, but recent work on automatic generation of importance functions has had some success in alleviating this situation.[9] A method for sustaining the particle population far from the source other than boundary splitting and Russian roulette is the exponential transform (path stretching). It is recommended that this option be used only in combination with the weight window since the weight adjustment can produce large fluctuations in the particle weight.

Although work is being done by Los Alamos in the area of angle biasing,[10] there is no standard option for explicitly biasing the collision kernel in MCNP, i.e., altering the outgoing energy and direction. Experience with this technique has been discouraging due to the weight fluctuations introduced and the inability to consistently improve calculation performance. It has been the objective in the development and use of variance reduction methods in MCNP to keep particle weight variation under control without having to resort to the difficult task of determining general weight window input parameters.

Other variance reduction options are available which have usually been associated with reactor problems rather than deep penetration. These are implicit (or analog) capture, forced collisions, and correlated sampling. The capture option allows a particle to be captured and the history terminated or have its weight adjusted by the survival probability $(\Sigma_T - \Sigma_c)/\Sigma_T$ and continue. The collision option forces particles to undergo collisions by a splitting process in optically thin cells that otherwise would be transparent to particle transport. The correlated sampling option in MCNP is a fairly simple method of investigating small perturbations whose effect would be lost in the statistical uncertainty of a normal calculation. The random number sequence for corresponding pairs of particles in perturbed and unperturbed calculations is forced to be identical until the perturbation is encountered and the sequence diverges.

Additional standard variance reduction techniques are energy splitting with Russian roulette, weight cutoff with Russian roulette, time and energy cutoffs, and a scheme to reduce the number of unimportant contributions to point detectors and DXTRAN. Control of the birth rate of neutron-induced photons is also available which is often used to improve problem efficiency.

A unique option in MCNP which combines features of path-length stretching, angular biasing, and next-event estimation is called DXTRAN, which stands for "deterministic transport." As in the estimation process for each random walk event, "pseudo-particles" are deterministically transported to some region of interest where ordinary particles have a low probability of reaching. Here, these new particles enter a secondary random walk procedure completely independent of the original, producing the desired results. As in path-length stretching and next-event estimation, it is possible that important intermediate regions may be skipped. The method has been used as an angular scattering device to force particles along void streaming paths.[11] If a user written anisotropic source routine is used, a special SRCDX routine must also be written to account for pseudo-particles from source events. It is recommended that the DXTRAN option be used with caution.

VII. Tallies

The accumulation of results from any estimation process is called a tally in the MCNP literature. The tallies are collected on surfaces, in volumes, or at points, and the standard options should be sufficient for most calculations. The volumes and surfaces may be from all or any part of the cells used in the geometry input description. In addition, the user may supply a special routine for any desired nonstandard tally.

The standard tallies are:

1. particle current on a surface,
2. flux on a surface,
3. flux in a volume from particle track lengths,
4. flux at a point,
5. energy deposition in a volume from particle track lengths, and
6. energy deposition in a volume due to fission.

These standard tallies each have many options for the user such as tallying over any subset of surfaces and cells, energy, time, and, for the current tally, angle ranges. These results may be folded with a wide variety of response functions, nuclear data, dose functions, and normalization constants. The tallies may be segmented into subsets of cells and surfaces, flagged according to which ones have contributions from different spatial regions, and attenuated as desired. The flux at a point tallies are separated into the uncollided and collided parts, as well as the total, and are extensively diagnosed. Both energy and particle fluxes and currents may be tallied for the first four standard tallies as well. In addition, the tally output print format may be optionally rearranged by the user to produce more readable results.

The optional user-supplied subroutine is called TALLYX. Examples are given in Chapters 4 and 5 of the manual describing its possible construction. It is incorporated into an MCNP calculation in a manner similar to that described for SOURCE and SRCDX. Use of the user written TALLYX in conjunction with the output files creation feature (Chapter 3 in the manual) allows post-processing of any results for further analysis, plotting, or input to a subsequent calculation with a special source routine.

All the standard tallies except number 4 above are analog estimators and require an adequate particle population for reasonable results and statistics. In addition to the flux-at-a-point procedure, MCNP has two modifications of this method available as options. The once-more-collided-flux-estimator (OMCFE) eliminates the $1/R^2$ infinite variance problem, and the ring detector, where applicable, has been found to be more efficient than a point detector. The point detectors, often referred to in the MCNP literature simply as "detectors," require more care in their use than the analog estimators.

Although detectors can be beneficial where applicable, the results they produce are sometimes questionable and their use can greatly increase the problem running time. Figure 5 shows the results of a calculation of a deliberately designed pathological problem using different flux estimators (point detectors on a boundary) and, with the following two figures, is presented only to illustrate some of the characteristics of these tallies. The increase in the point detector answer is due to a collision very close to the detector point. The OMCFE is the same until this collision, and it is seen to nearly eliminate the $1/R^2$ effect. In Fig. 6 it is seen that because the entire answer from the point detector is dominated by this one collision, the relative error is almost 100%. In MCNP the figure of merit (FOM) is the inverse of the product of the estimated relative error squared ($\sigma^2/\bar{x}^2$) and the calculation time in minutes. The higher the FOM, the more efficient the estimator, as seen in Fig. 7.

A continuation of the calculation beyond the number of histories shown in Fig. 5 shows that the point detector mean gradually decreases and goes below that for the surface crossing estimator. Another collision close to the detector will, of course, produce another large contribution. The ring detector result approaches the analog surface flux as more histories are run but its FOM is substantially lower due to the increased computation time. The OMCFE is somewhat more erratic with mean, error, and FOM lying between those for the point and the ring.

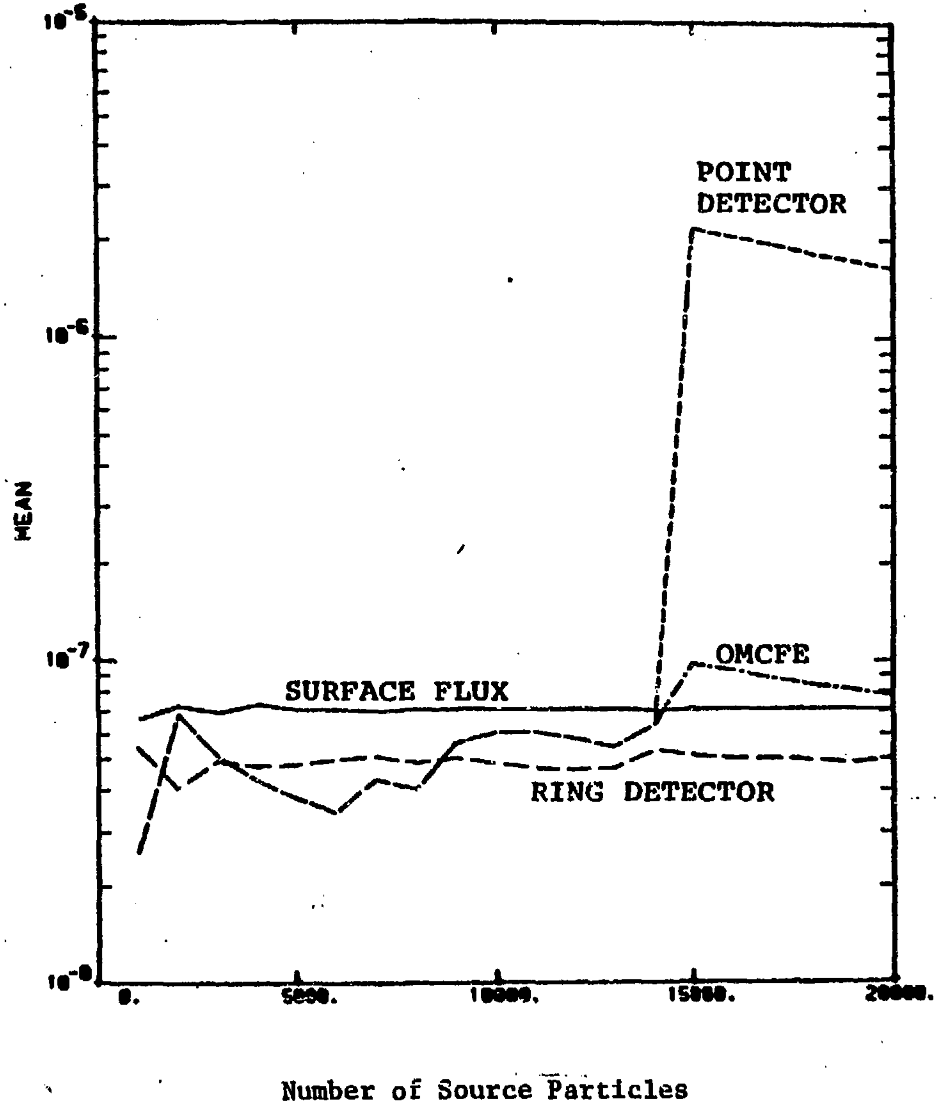


Fig. 5. Comparison of MCNP Estimators.

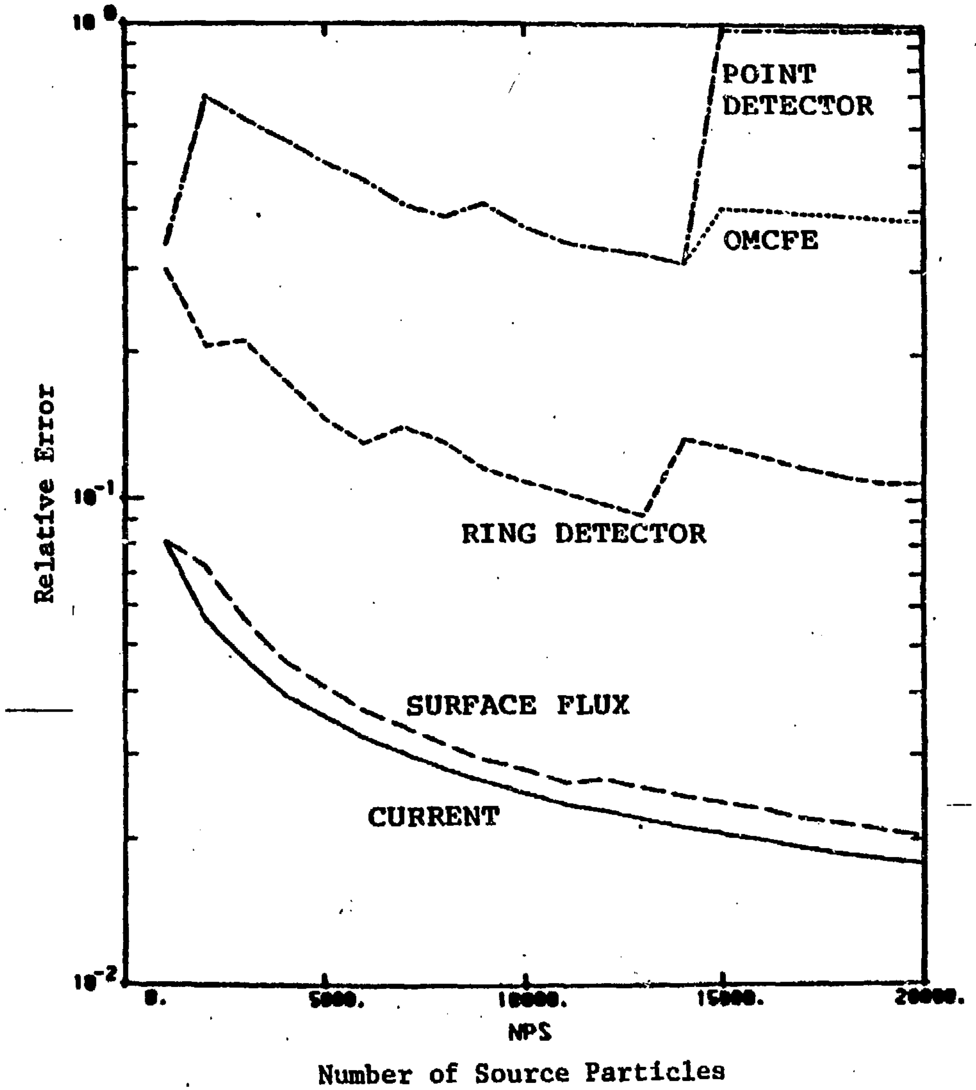


Fig. 6. Comparison of MCNP Estimator Relative Errors.

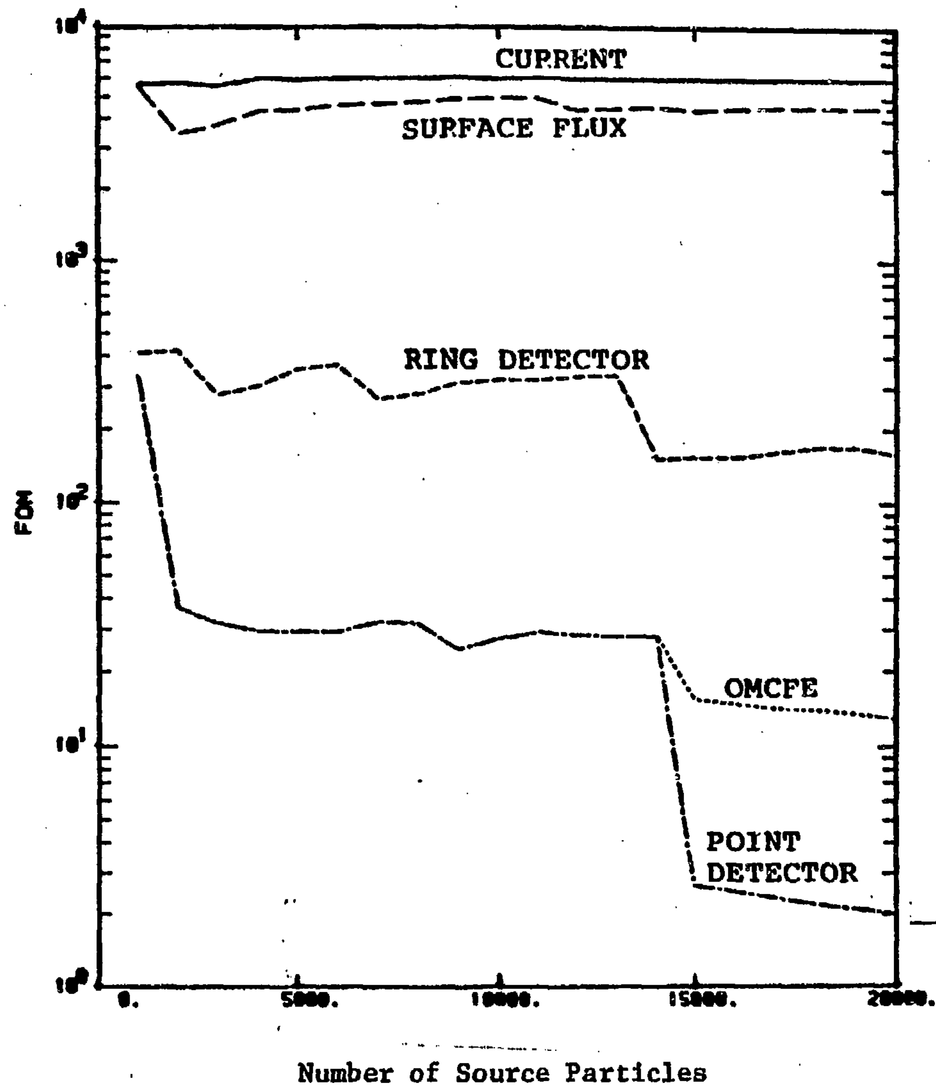


Fig. 7. Comparison of MCNP Estimator Figure of Merit.

These comparisons clearly indicate that it is necessary to examine the fluctuations in any Monte Carlo calculation and not just one result after some arbitrary number of histories. The information in these figures is available in tabular form in the standard MCNP output listings. These and other output quantities will be discussed in the next section.

VIII. Input-Output

The MCNP input is in a free form card image format with several shortcut features and a comment card capability. In addition to a detailed description of each input item, the manual provides a summary table of input cards, and the RSIC code package includes an alphabetic list of input parameters and a corresponding reference page in the manual.

The output of an MCNP calculation can be extensive, depending on the print options selected. The manual gives a detailed explanation of each item for a few sample problems. All source particle variables can be printed to aid in debugging special source routines, and if there is some trouble in the calculation, that particle is automatically restarted and the parameters from its entire history are printed in an event log for inspection. For illustrative purposes, Fig. 8 gives selected portions of a problem printout where cell 1 is the source cell and the tallies are in cell 12. Due to the correct application of surface splitting and Russian roulette, it is seen that the particle population remains approximately constant in the region of interest which, for this problem, is everywhere. The bottom of the figure shows a tally fluctuation chart for several tallies as a function of the number of processed source particles (NPS) for the estimated average answer (MEAN), the estimated fractional standard deviation (RELATIVE ERROR), and the figure of merit (FOM). Unlike the point estimator in the pathological problem of Fig. 5, these tallies are seen to be well behaved.

PROBLEM ACTIVITY IN EACH CELL, NEUTRONS ONLY

CELL PROGR NAME	PROBL NAME	TRACKS ENTERING	POPULATION	COLLISIONS	COLLISIONS * WEIGHT (PER HISTORY)	NUMBER WEIGHTED ENERGY	FLUX WEIGHTED ENERGY	AVERAGE TRACK WEIGHT (RELATIVE)	AVERAGE TRACK MFP (CM)
1	100	29137	16233	0	0.0000E+00	6.5576E-03	3.5114E+00	9.3449E+18	1.0000E+38
2	2	44337	17046	86823	8.2885E+19	4.6691E-03	2.1059E+00	1.0217E+19	4.1433E+00
3	3	36304	20390	226771	2.2322E+20	9.9992E-03	1.3432E+00	9.6505E+18	2.7778E+00
4	4	29407	20210	194935	1.3314E+20	9.3603E-03	6.6798E-01	9.9549E+18	2.7711E+00
5	5	24942	19252	174559	7.2061E+19	5.6234E-03	4.0178E-01	1.0791E+19	2.6205E+00
6	6	23915	19538	94670	2.4691E+19	1.0472E-03	2.6210E-01	1.4110E+19	1.8195E+00
7	7	22478	19245	89031	1.3204E+19	6.3870E-04	2.4963E-01	1.5454E+19	1.7144E+00
8	8	21856	19498	83528	5.9105E+18	4.6864E-04	2.7624E-01	1.5601E+19	1.6956E+00
9	9	21238	19840	91638	4.8333E+13	9.8514E-05	2.6480E-01	2.1389E+19	1.0389E+00
10	10	21356	20347	88465	1.7591E+18	8.5310E-05	4.3362E-01	2.1344E+19	1.0622E+00
11	11	20307	19394	67406	4.6795E+17	1.5570E-04	8.9557E-01	1.4895E+19	1.3951E+00
12	12	19553	19440	9779	1.0677E+16	8.4999E-04	2.2676E+00	7.2440E+18	2.4834E+00
TOTAL		314830	230433	1207605	5.6219E+20				

TALLY FLUCTUATION CHARTS

	TALLY 1			TALLY 2			TALLY 4		
NPS	MEAN	ERROR	FOM	MEAN	ERROR	FOM	MEAN	ERROR	FOM
1000	5.30921E+15	0.2011	26	2.82999E+13	0.0599	289	3.59301E+13	0.0513	393
2000	5.60374E+15	0.1486	22	2.99403E+13	0.0438	257	3.81510E+13	0.0370	361
3000	6.12753E+15	0.1161	24	3.02556E+13	0.0389	216	3.82429E+13	0.0304	353
4000	6.20226E+15	0.1008	24	3.11853E+13	0.0344	205	3.93352E+13	0.0267	340
5000	6.67259E+15	0.0897	24	3.06257E+13	0.0307	204	3.87111E+13	0.0237	342
6000	6.55475E+15	0.0810	25	3.07340E+13	0.0285	200	3.84403E+13	0.0220	338
7000	6.70984E+15	0.0760	24	3.12591E+13	0.0280	175	3.86851E+13	0.0206	322
8000	6.80111E+15	0.0703	24	3.11990E+13	0.0259	178	3.87176E+13	0.0193	320
9000	7.23829E+15	0.0651	25	3.12144E+13	0.0243	179	3.87741E+13	0.0183	317
10000	7.26587E+15	0.0617	25	3.09924E+13	0.0229	182	3.85927E+13	0.0176	310
11000	7.22008E+15	0.0586	25	3.12281E+13	0.0219	181	3.88839E+13	0.0169	306
12000	7.28298E+15	0.0560	25	3.10867E+13	0.0206	187	3.88577E+13	0.0161	309
13000	7.24620E+15	0.0537	26	3.11344E+13	0.0197	191	3.88679E+13	0.0154	309
14000	7.17882E+15	0 0517	26	3.12805E+13	0.0191	187	3.89423E+13	0.0149	307
15000	7.03058E+15	0.0503	25	3.12192E+13	0.0184	190	3.89040E+13	0.0145	306

Fig. 8. Example of MCNP Output.

Much more printout exists for particle weights in terms of reaction types and isotopes and also for weight production and loss for physical and variance reduction processes. Particles may be "flagged" as they traverse specified geometry regions to determine differential contributions to a tally. With the TALLYX routine, the user is free to produce any additional output desired.

IX. Sample Problems

In addition to sample problems given in the manual, the RSIC code package includes the input card images for 14 problems. These problems are not completely documented, but they contain examples of most of the MCNP input options. A very brief description of the MCNP features for each problem is as follows:

1. point source, point detector, concentric spheres,
2. DXTRAN and ring detector,
3. thermal neutrons (free gas), point detector, detailed tallies,
4. gamma-ray calculation,
5. toroidal geometry (TOKAMAK),
6. source angular biasing,
7. several variance reduction techniques,
8. lost particle (geometry debug print),
9. criticality calculation, complicated geometry,

10. complicated geometry, detector debug, volume source,

11. coupled neutron gamma ray, complicated geometry, $S(\alpha,\beta)$ data,

12. OMCFE and ring detector,

13. elaborate geometry check, and

14. time dependence, weight window, geometry transformations.

A complete listing of the calculation output for problem #1 is included in the code package.

X. Documentation

The basic document of the current MCNP version 3 code package (RSIC CCC-200) is more than 600 pages long and appears quite formidable at first glance. The core of this document is a manual[12] for an earlier version of the code with its own content pages, index, and page numbering for easy reference. The initial and final portions of the larger report contain information for implementing MCNP on various computer systems and have been largely superseded by the MCNP systems guide, also included in the package. The three MCNP newsletters from Los Alamos are included to help bridge the gap from the version 2B manual to the version 3 code.

Additions and updates to MCNP are made in such a manner so as to minimize deviations from the manual. The RSIC newsletter will from time to time give pertinent information on its code packages and documentation. In reviewing these newsletters, attention must be given to their chronology and to application of the information to specific computer systems. Both a new MCNP manual and sample problem package are in preparation and will be distributed as soon as they become available to RSIC. Some of the sample problems may still contain a few input items, such as nuclide identifiers, which are incompatible with the RSIC package. At Los Alamos the attempt is made to have the most recent cross-sections evaluations available in MCNP. However, RSIC must abide by any current U.S. Department of Energy regulations in this matter, and the standard code package will contain data from the latest ENDF/B release only for internal United States distribution.

XI. Future Work

Most presentations on MCNP contain references to ongoing and future work. The results first become patches, then a permanent part of the Los Alamos in-house code, and are finally given to the distribution centers. Some items of interest are:

1. generalized standard sources,
2. finite lattice geometry array capability,
3. interactive three-dimensional geometry plotting package,
4. unresolved resonance probability table method,
5. discrete ordinates coupling,
6. source coupling from a previous MCNP calculation,
7. coupling with high energy (>20 MeV) charged particle codes,
8. automatic generation of weight windows and importances,
9. angle bias,

10. continued work on more generalized importance function generation and its automatic implementation.

11. coupled electron photon transport capability, and

12. multigroup version (MCMG) which solves both the forward and adjoint transport equations.

RSIC has a list of MCNP patch updates, and any user contemplating a significant development in the form of a patch should contact RSIC to see if a similar one is available. The code developers would, of course, be interested in any new MCNP features created by outside users.

The fluid nature of MCNP development, documentation, and distribution may cause some occasional inconvenience. However, it is felt that this situation is preferable to the alternative, where the distributed, but unmaintained, code package lags a few years perhaps behind the most up-to-date version of the code.

XII. References

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