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Informal Report

MCNP—A General Monte Carlo Code for Neutron and Photon Transport

A Summary

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W. L. Thompson

FOREWORD

Beginning with the next paragraph, this report is a reproduction of the first few pages of the MCNP user's manual (LA-7396-M). The features and history of MCNP are summarized in these pages. For more information, contact any member of Group X-6.

The manual is written as a practical guide for the use of our general-purpose Monte Carlo code MCNP. The intent is that the second chapter describe the mathematics, physics, and Monte Carlo simulation found in MCNP. However, this discussion is not meant to be exhaustive - details of the particular techniques and of the Monte Carlo method itself will have to be found elsewhere. The third chapter shows the user how to prepare input for the code. The fourth chapter contains several examples, and finally the fifth chapter explains the output. The appendices show how to use MCNP on a particular computer system at the Los Alamos Scientific Laboratory and also give details about some of the code internals that those who wish to modify the code may find useful.

Neither the code nor the manual is static. The code is changed from time to time as the need arises (about once a year), and the manual is changed to reflect the latest version of the code. This particular manual refers to Version 2 of MCNP that was released on September 26, 1979.

MCNP and the manual are the product of the combined effort of the people in Group X-6 (formerly TD-6) of the Theoretical Applications Division (X Division) at the Los Alamos Scientific Laboratory.

Except for the figures, the manual was prepared entirely by the TRIX Report Editor and the REDPP post processor routines available through the LTSS operating system on the LASL CDC-7600 computers. The master for the manual is 35-mm film produced by the FR80 film recorders.

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MCNP - A General Monte Carlo Code
for Neutron and Photon Transport

LASL Group X-6

ABSTRACT

The general-purpose Monte Carlo code MCNP can be used for neutron, photon, or coupled neutron-photon transport, including the capability to calculate eigenvalues for critical systems. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces (elliptical tori).

Pointwise cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation (such as ENDF/B-IV) are accounted for. Thermal neutrons are described by both the free gas and $S(\alpha,\beta)$ models. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission following photoelectric absorption, and absorption in pair production with local emission of annihilation radiation.

MCNP includes an elaborate, interactive plotting capability that allows the user to view his input geometry to help check for setup errors. Provisions are also made to translate and/or rotate surfaces from one coordinate system to another. Cell volumes and surface areas are automatically calculated for use by the tallies.

Standard features which are available to improve computational efficiency include geometry splitting and Russian roulette, weight cutoff with Russian roulette, correlated sampling, analog capture or capture by weight reduction, the exponential transformation, energy splitting, forced collisions in designated cells, flux estimates at point or ring detectors, deterministically transporting pseudo-particles to designated regions, track-length estimators, source biasing, and several parameter cutoffs.

Extensive summary information is provided to help the user better understand the physics and Monte Carlo simulation of his problem. The standard, user-defined output of MCNP includes two-way current as a function of direction across any set of surfaces or surface segments in the problem. Flux across any set of surfaces or surface segments is available. Similarly, the flux at designated detectors and the average flux in any cell (track length per unit volume) are standard tallies. Reactions such as fissions, tritium production, absorptions, or any product of the flux times any standard ENDF reaction cross sections plus several nonstandard ones may be obtained in any cell, at a surface, or at a point. The heating tallies give the energy deposition in designated cells. In addition, particles may be flagged when they cross specified surfaces or enter designated cells, and the contributions of these flagged particles to the tallies are listed separately. The user is allowed to modify any of the standard tallies almost any way desired. All quantities tallied also have their relative errors calculated. All tallies are a function of time and energy as defined by the user.

CHAPTER 1
A SUMMARY OF FEATURES

If you are reading this Manual for the first time with the intent to set up and run a job with MCNP as soon as possible, see the Quick and Easy section of Appendix A.

MCNP is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron-photon Monte Carlo transport code. It may be used in any of three modes: neutron transport only, combined neutron-photon transport, or photon transport only. The capability to calculate eigenvalues for critical systems is also a standard feature of MCNP.

The code compiles under the CDC FTN compiler that has been slightly modified at the Los Alamos Scientific Laboratory (LASL), is largely compatible with the FORTRAN 77 standard except for a few nonstandard FTN features, and is developed and maintained by the LASL Group X-6 on the LASL CDC-7600 and CRAY-1 computers. MCNP has only 12697 lines of coding (this is with the COMMON blocks listed only once and not in every subroutine). Group X-6 also maintains MCNP on the Magnetic Fusion Energy (MFE) computers at the Lawrence Livermore Laboratory.

An attempt has been made to make MCNP as system independent as possible to enhance its portability. System dependencies that could not be avoided have been segregated as much as possible into system-dependent subroutines that will undoubtedly need to be replaced at other computer facilities. The manual is written for the CDC-7600 version of MCNP where the operating system is the *Livermore Time Sharing System (LTSS)*.

The various features and capabilities of MCNP are summarized in the rest of this chapter. More detail concerning each topic is available in later chapters or appendices.

I. GEOMETRY

The geometry of MCNP treats an arbitrary three-dimensional configuration of arbitrarily defined materials (using up to 40 different isotopes chosen from the MCNP cross-section libraries) in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces (elliptical tori). The cells are defined by the intersections and unions of the regions bounded by the surfaces.

MCNP does not have combinatorial geometry as it is commonly denoted. Rather than combining (i.e., taking the union of) several pre-defined geometrical bodies as in combinatorial geometry, MCNP gives the user the added flexibility to define his own geometrical bodies from all the available surfaces and then to combine them with a union operator.

Surfaces are defined by supplying coefficients to the analytic surface equations or for certain types of surfaces by supplying known points on the surfaces.

The code does extensive internal checking to find input errors. In addition, an elaborate plotting capability is available to aid the user.

II. NUCLEAR DATA AND REACTIONS

Pointwise cross-section data in considerable detail are used in MCNP. The data are tabulated in the MCNP cross-section libraries on an energy grid that is tailored to each isotope. Linear interpolation is used between energy points with a few hundred to several thousand points typically required. Cross sections are added at a sufficient number of points to insure that the linear interpolation constraint reproduces the original cross-section tabulation within a specified tolerance of a few percent (in fact, usually 0.1 to 0.5%). Resonance parameters, if they are given, are processed at several temperatures (but only at one temperature for a given cross-section evaluation) and added to the pointwise cross sections. Furthermore, the energies at which the cross sections are tabulated are shifted so that all reactions are given using the same energy grid - the grid on which the total cross section is tabulated. The total photon production cross section and neutron heating numbers are given on that same energy mesh.

MCNP is strictly a neutral particle transport code; any energy that would be carried by charged particles is deposited locally.

A. Neutrons

All reactions given in a particular neutron cross-section evaluation (such as ENDF/B-IV) are accounted for in the energy range from about 0.00001 eV to about 20 MeV. Users have the choice of three sources of neutron cross sections: ENDF/B,[1] Howerton's ENDL library from Livermore,[2] and the British (AWRE) library. Furthermore, there is the choice of using prompt or total fission $\bar{\nu}$ as well as the option of using discrete reaction cross sections in which the reaction cross sections are multigrouped.

The angular distributions for elastic and inelastic events are prescribed on a fine grid of incident neutron energies. Linear interpolation yields the angular distribution for the particular incoming neutron energy. This distribution is then sampled in a continuous fashion.

Similarly, the energy distributions for secondary neutrons from inelastic reactions (none needed for level scattering, of course) are also stored on a fine energy grid. These distributions are obtained from the laws prescribed in the particular cross-section evaluation. Linear interpolation yields the energy distribution for the incident neutron energy. Again, the distribution is sampled continuously.

If the total fission cross section alone is given, then that cross section is used with assumed behavior for the breakup into (n,n')f and (n,2n)f. However, if the reactions (n,f), (n,n')f and (n,2n)f are explicitly tabulated, then these cross sections along with the associated angular and secondary energy distributions are used directly.

The energy spectra for photons produced by neutron interactions are given in terms of 20 equally probable photon energies for each of 30 incident neutron energy groups.

There are two thermal neutron treatments in MCNP. One is the free-gas model in which, for elastic collisions, light atoms (for $Z = 1$ through 8) are assumed to be in a Maxwellian distribution with some thermal temperature that may be a function of time. Secondly, neutron thermal scattering can be modeled by the $S(\alpha,\beta)$ scattering model which includes chemical binding and crystalline effects that become important as the neutron's energy becomes sufficiently low. $S(\alpha,\beta)$ data are currently available for light and heavy water, beryllium metal, graphite, and polyethylene. Others will be added as they become available. Typically one will use the free-gas model from around 10 eV to 4 eV and will then switch to the $S(\alpha,\beta)$ model. $S(\alpha,\beta)$ effects are most significant below 2 eV.

B. Photons

The photon interaction cross sections come from the Storm and Israel[3] evaluation which covers the energy range 0.001 to 100.0 MeV. Below 0.001 MeV, MCNP allows only analog capture for photons which will rapidly terminate them.

MCNP takes account of incoherent (using an inverse fit rather than a rejection scheme on the Klein-Nishina distribution) and coherent scattering, the possibility of fluorescent emission following photoelectric absorption, and absorption in pair production with local emission of the annihilation quanta.

III. SOURCES

MCNP has five standard sources in addition to the provision to allow the user to provide his own source subroutine.

The five standard sources (which all allow energy and directional biasing) are:

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

IV. VARIANCE REDUCTION

Standard optional variance reduction schemes in MCNP include geometry splitting and Russian roulette, weight cutoff with Russian roulette, time and energy cutoff, correlated sampling, analog capture or implicit capture by weight reduction, the exponential transformation, energy splitting, forced collisions in designated cells, flux estimates at point or ring detectors, track-length estimators, and source biasing (both in energy and direction).

Flux estimates at point detectors are determined by two methods, the next-event estimator (or analog) scheme and the once-more-collided flux estimator (OMCFE). A scheme, DXTRAN, to improve sampling in the vicinity of detectors or other tallies is available. It involves deterministically transporting pseudo-particles on collision to some arbitrary, user-defined sphere in the neighborhood of a tally and then calculating contributions to the tally from these pseudo-particles. Contributions to the detectors or to the DXTRAN spheres can be controlled as a function of geometric cell or as a function of the number of mean free paths from a collision point to the detector or DXTRAN sphere.

V. TALLIES AND OUTPUT

The user-defined output of MCNP includes two-way current as a function of direction across any subset of surfaces (or surface segments) in the problem. Fluxes across any set of surfaces are available also. Similarly, the flux at designated detectors (points or rings) and the average flux in a cell (track length per unit volume) are standard tallies. The heating tallies give the energy deposition in specified cells. In addition, particles may be flagged when they cross specified surfaces or enter designated cells, and the contributions of these flagged particles to the tallies are listed separately. Reactions such as fissions, absorptions, tritium production (or any product of the flux times the approximately one hundred standard ENDF reactions plus several nonstandard ones) may be tallied with any of the MCNP tallies. In fact, any quantity of the form

$$C \int \varphi(E)\, f(E)\, dE$$

may be tallied where $\varphi(E)$ is the energy-dependent flux and $f(E)$ is any product of the quantities in the cross-section libraries or a function provided by the user. Tallies may be made for segments of cells and surfaces without having to build the desired segments into the actual problem geometry. All tallies are a function of time and energy and are normalized to be per unit starting particle.

Printed out with each tally is also its relative error corresponding to one standard deviation of the mean.

Standard summary information is also printed to give the user a better idea of how his problem ran. This information can give insight into the physics of the problem and the adequacy of the Monte Carlo simulation. If errors occur during the running of a problem, detailed debug prints are given.

VI. CRITICALITY

MCNP has the capability to calculate eigenvalues for both sub- and super-critical critical systems. The calculation is run as a series of generations of neutrons. At the end of each generation, k_{eff} is calculated for that generation as well as averaged over a specified number of preceding generations.

The source for the first generation is usually defined by the user at a set of spatial points in the system. The neutrons are then started at these points isotropically and with an energy sampled from standard fission distributions.

Three estimators (in various combinations) are used to calculate k_{eff}: absorption, collision, and track-length estimators.

VII. OTHER FEATURES

MCNP can be run interactively or in a batch mode.

An arbitrary cross-sectional view of the input geometry can be plotted on 35-mm film, 105-mm microfiche, on an interactive graphics display terminal, or on a 36-inch Versatec electrostatic plotter. A file, $PLOT$, of plotted points is produced to facilitate using other graphical devices.

A feature is available to allow the user to translate and/or rotate the entire geometry or a part of it from one coordinate system to another.

For use by some of the tallies, volumes of cells (regardless of axis of symmetry) are calculated if they are bounded by surfaces of revolution. Irregular volumes can be calculated stochastically. Surface areas of cells are also calculated.

Full restart capabilities, used for machine failure or continuing a run to obtain better statistics, are available. This includes periodically dumping to magnetic tape or disk automatically.

The user is allowed to specify almost any information desired to be written to a file for post-processing, such as for plotting results or to generate a source for a subsequent problem. Facilities also are included to read in information from a source or data file.

An event-log feature is available for debugging purposes that prints out the complete life history of a given particle.

CHAPTER 2
GEOMETRY, PHYSICS, AND MATHEMATICS

I. INTRODUCTION

The manual is written as a practical guide for the use of MCNP. This second chapter discusses in more detail the mathematics and physics of the MCNP geometry, cross-section libraries, variance reduction schemes, the Monte Carlo simulation of the neutron and photon transport, and the tallies. This discussion is not meant to be exhaustive; many of the details of the particular techniques and of the Monte Carlo method itself will have to be found elsewhere. Carter and Cashwell's book *Particle-Transport Simulation with the Monte Carlo Method*[4] is a good general reference on radiation transport by Monte Carlo and is based upon what is in MCNP.

A. History

Although the first use of Monte Carlo techniques is difficult to trace, the emergence of the Monte Carlo method as a radiation transport research tool springs from work done at Los Alamos during the Second World War. The credit for the so-called invention of Monte Carlo as a mathematical discipline is generally attributed to Fermi, von Neumann, and Ulam. Their ideas were developed by many followers in various laboratories, but the name of von Neumann seems to be attached to many, if not most, of the fundamental ideas and techniques. Metropolis and Richtmyer should also be credited with some of the early Monte Carlo development at Los Alamos.

In 1947 while at Los Alamos, Fermi invented a mechanical device to trace neutron movements through fissionable materials by the Monte Carlo method. The FERMIAC[5] is on display in the Bradbury Science Museum at LASL.

Monte Carlo at LASL has had, since its inception, a number of devoted and well-known disciples. In the early 1950's there was assembled at Los Alamos, under Ulam's guidance, a small band of scientists engaged in investigating transport methods and solving transport problems by Monte Carlo. At first the calculational effort was rudimentary, but with the completion of the MANIAC computer, the work gained momentum and rapidly became a factor in much of the research at Los Alamos. Much of the early work is summarized in the first book to appear on the subject of Monte

Carlo by Cashwell and Everett.[6] Shortly thereafter there appeared at LASL the Monte Carlo neutron transport code MCS[7] which incorportated a general geometry treatment.

The addition of several more features resulted in the Monte Carlo neutron code MCN[8] in 1965. These features included the improvement and expansion of the geometry treatment, provision for standard cross-section libraries, inclusion of standard variance reduction techniques, inclusion of general sources, treatment of thermalization of neutrons by the free-gas model, and more general tallies and output. The photon codes MCG and MCP[9] were then added to the LASL family of Monte Carlo codes. Both of these dealt with photon transport, MCG treating higher energy photons or gamma rays and MCP treating photons of energies down to 1 keV. MCN and MCG were merged to form MCNG (the predecessor of MCNP) in 1973. Gamma-production cross sections were added to enable the gamma rays produced by neutron interactions to be generated.

B. MCNP Structure

MCNP is the culmination of all the above work and codes plus many additions in mathematics, physics, and user-oriented features. The first version of MCNP was released in June, 1977. There have since been two revisions, versions 1A and 1B. Version 2 is the first major revision to MCNP, although as far as the user is concerned its input is almost completely backward compatible to the early MCN and MCG codes.

The general structure of MCNP is based on overlays. The main overlay MCNP calls up to four other overlays, depending upon the user's requirements. IMCN (overlay 1C0 and commonly called the initiation code) processes the problem input file and calculates volumes and areas. PLOT (overlay 2C0) plots, translates and/or rotates the problem geometry. XACT (overlay 3C0) processes cross sections for the specified materials. MCRUN (overlay 4C0 and commonly called the run code) does the actual particle transport. Reference is frequently made (mainly for historical reasons) to the MCRUN overlay (the run code) in three different ways: MCN refers to the neutron portion, MCG refers to the simple-physics photon treatment, and MCP refers to the detailed-physics photon treatment which includes fluorescence. The ERGP input card (see page 159) determines whether MCP or MCG is used. This terminology will be used in the rest of the manual. The criticality portion of MCRUN is frequently called the KCODE.

MCNP can be run in three modes: Mode 0 is neutron transport only, Mode 1 is combined neutron-photon transport, and Mode 2 is photon transport only. In Mode 1, photon histories are followed as part of their parent neutron histories rather than being saved and followed later after all the neutron histories have been completed as was the case with MCNG.

Variants of MCNP are available that offer a geometrical perturbation feature; a multigroup adjoint capability; a state-space splitting capability; flux, importance, and contributon flux data generation with subsequent plotting; and a calculation of the variance of the variance. Some of these features may become a permanent part of MCNP at a later date.

Future work includes a geometrical lattice capability, the probability table method for unresolved resonances, photo-neutron reactions, improved variance reduction schemes such as better source biasing and a weight window to control high and low weighted tracks, and higher energy neutron cross sections (perhaps up to 60 MeV in some cases). Furthermore, X-6 always strives to be current with the best cross-section data such as ENDF/B-V.

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