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TITLE: MCNP - A GENERAL MONTE CARLO CODE FOR NEUTRON AND PHOTON TRANSPORT

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MCNP - A GENERAL MONTE CARLO CODE FOR NEUTRON AND PHOTON TRANSPORT

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ABSTRACT

MCNP is a very general Monte Carlo neutron photon transport code system with approximately 250 person years of Group X-6 code development invested. It is extremely portable, user-oriented, and a true production code as it is used about 60 Cray hours per month by about 150 Los Alamos users. It has in its data base the best cross-section evaluations available. MCNP contains state-of-the-art traditional and adaptive Monte Carlo techniques to be applied to the solution of an ever-increasing number of problems. Excellent user-oriented documentation is available for all facets of the MCNP code system. Many useful and important variants of MCNP exist for special applications.

The Radiation Shielding Information Center (RSIC) in Oak Ridge, Tennessee is the contact point for worldwide MCNP code and documentation distribution. A much improved MCNP Version 3A will be available in the fall of 1985, along with new and improved documentation. Future directions in MCNP development will change the meaning of MCNP to Monte Carlo N Particle where N particle varieties will be transported.

INTRODUCTION

CAPABILITIES AND AVAILABILITY

The MCNP (Monte Carlo Neutron Photon) code system developed by Group X-6 is the workhorse at the Los Alamos National Laboratory for neutron, photon, and coupled neutron-photon calculations using the Monte Carlo method. MCNP transports neutrons and/or photons continuously in all phase-space variables for both fixed-source and criticality (k_{eff}) problems. MCNP contains detailed neutron and photon physics models which automatically link to data libraries that contain the most up-to-data cross-section and reaction information from the ENDF/B and other evaluations. MCNP transports particles in generalized three-dimensional geometries using 26 different kinds of first, second, and certain fourth-degree surfaces. State-of-the-art Monte Carlo methods are used in all phases of the particle transport including the source, geometry-tracking, variance-reduction and tally-estimation processes.

MCNP is a highly portable user-oriented production code for a wide range of applications. The code has been run successfully on ten different systems from small 32 bit machines to Crays. MCNP has been designed to be very user oriented in its input, error checking, graphical displays, data libraries, dynamic interrupts, and output. It is truly a production code because of its heavy useage by about 150 Los Alamos users for approximately 60 Cray hours every month. MCNP is extremely well documented in its 511 page manual[1] on the code, data bases, and test problems, as well as in a series of 25 video tapes from a national MCNP workshop held in Los Alamos in 1983. In the same year, MCNP was released to the Radiation Shielding Information Center (RSIC) in Oak Ridge for national and worldwide distribution of the code, data bases, test problems and documentation.

Version 3 of MCNP has been available for a year and a half and is the subject of this status report. Version 3A, which contains many important improvements described at the end of this report, is scheduled for release in the fall of 1985.

HISTORY AND PHILOSOPHY

MCNP has an investment of about 250 person years, dating back to the formulation and first uses of the Monte Carlo method at Los Alamos during the 1940s by Fermi, Metropolis, Richtmyer, Ulam, and Von Neumann. E. D. Cashwell and C. J. Everett were the two most important contributors to MCNP and its predecessors beginning with MCS which was written in 1963.[2] The first version of MCNP appeared in 1976 as a combination of MCN[3], MCG[4], and MCP.[4,5] Version 3 of MCNP was completely rewritten in ANSI standard FORTRAN 77 in 1983 to produce a one-source code that would run on many different computer systems.

The MCNP code development philosophy has always centered around a consensus of the views of Monte Carlo theorists, data specialists, MCNP users, and code developers. New features have always been well tested and understood before being incorporated into a newly released version of the code.

ARCHITECTURE

The MCNP architecture has been primarily dictated by two constraints: the limited amount of small core memory on the CDC-7600; and the lack of a dynamic storage capability in FORTRAN 77. The first constraint results in an overlay structure for the initiation, plotting, data, and transport portions of the code. The second constraint results in arrays that are referenced by offsets. Subroutines are used as building blocks for the MCNP structure. Those parts of the code which are system dependent are concentrated in a few locations to simplify the portability aspects.

SOFTWARE

MCNP is written in ANSI standard FORTRAN 77 and uses the Los Alamos common graphics system (CGS), PLOT10, and DISSPLA graphics. There are about 21000 lines of source code and 190 subroutines that are logically structured and extremely readable. Virtually all storage arrays are dynamically allocated which keeps each problem as small as possible when running on the computer. The MCNP software is designed for easy user modification and includes features such as user arrays and variables, user-input cards, user-file access, and easy differentiation between COMMON variables (> 2 characters) and local variables (< 3 characters).

PORTABILITY

To achieve portability, a preprocessor for the source code is required in the form of UPDATE, HISTORIAN, or MCNP's own portable preprocessor. The preprocessor generates a source deck with the appropriate COMMON decks and conditional compilation directives for the specified computer system. The data files and graphics are also portable. To date, MCNP has been compiled, loaded, and run on the Cray (CTSS and COS), CDC (LTSS and NOS), IBM, VAX, PRIME, SUN, APOLLO, and RIDGE machines.

PHYSICS AND DATA

NEUTRONS

The MCNP neutron physics is based on ENDF/B, ENDL, and AWRE data libraries for 47 atomic numbers and for energy ranges from 10^{-11} to 20 MeV. Table I lists the various elements and isotopes available. Detailed modeling is used for both elastic and inelastic collisions. All of the reactions and resolved resonances available from a specific evaluation are included. The unresolved resonance region is treated as a continuum. There are two thermal-neutron treatments available in MCNP. The free-gas model assumes that neutrons collide with light nuclei (atomic numbers from 1 to 8) that are in a Maxwellian distribution which can be a function of time. Thermal neutron scattering can also be modeled using the $S(\alpha,\beta)$ scattering model which includes both chemical binding and crystalline effects. Room temperature $S(\alpha,\beta)$ data is available for H_2O, D_2O, C, CH_2, Be, BeO, ZrH, and benzene.

The MCNP neutron-transport and reaction-data libraries are processed using the NJOY[6] code. The continuous and thinned-continuous libraries treat the cross sections as continuous in energy with linear interpolation between specific energies so that the original evaluations are reproduced to 0.1% and 1% respectively. A discrete-reaction version of these libraries is also available which treats the total, absorption, elastic, and inelastic cross sections as a constant with a flat weighting spectrum in each of 262 energy bins. The kinematics of a collision are the same for all three forms of the neutron libraries. Energy-dependent angular-distribution tables are used to sample the outgoing energy and angle of a neutron from a collision. The discrete-reaction sets require much less memory than the continuous sets with an accompanying loss of some of the details of the evaluation. Discrete reaction data are normally used for trace isotopes or initial studies. All of the detailed information available in the evaluations concerning photon production is included in the MCNP data libraries. Some Doppler-broadened temperature-dependent cross-section sets are available. If the user does not wish to select data evaluations for the problem, default selections are assigned automatically by MCNP. Figure 1 shows the total neutron cross section for ^{235}U from 10^{-6} to 20 MeV represented in both the continuous and discrete reaction forms. Figure 2 shows continuous natural-tungsten photon-production cross sections over the same energy range for various reactions and total photon production. These data are typical of the evaluations in the MCNP libraries.

The final type of library available for MCNP neutron transport problems is the dosimetry library. Selected isotopes are maintained in this library to calculate responses from MCNP neutron flux calculations.

PHOTONS

The MCNP photon library contains all continuous data and is based on the ENDF/B evaluations plus Storm and Israel data[7] for a few high atomic-number isotopes. The energy range for photon transport in MCNP is between 10^{-3} and 100 MeV. Both incoherent scattering, using an inverse fit rather than a rejection scheme on the Klein-Nishina distribution, and coherent scattering is included in MCNP. The highly-forward-peaked coherent scattering may be removed from consideration by using the simple photon physics option in MCNP. Pair production with the resultant emission of annihilation radiation is standard in MCNP, as is photoelectric absorption followed by one or two fluorescent photons above 1 keV. Figure 3 shows the total and partial photon cross sections of lead as an example of the MCNP photon transport data.

GEOMETRY

CELLS

The basic unit of MCNP geometry description is the cell. Surfaces are used to create cells in the ordinary Cartesian coordinate system using the sense of space on either side. The sense of a point to a surface is the sign of the result of inserting the point coordinates into the surface equation. The sense of a portion of space relative to a surface is either + or -.

Combinations of sense-signed surfaces are used to define regions of space encompassed by a cell.

Three Boolean geometry operators are used in MCNP to construct cells. The **intersection** of common regions of space with respect to surfaces is the default operator. The **union** of space with respect to surfaces is designated by a colon (:). A **complement** operator (#) also exists to remove portions of space defined by either cells or surfaces. Parentheses are used to control the order of execution of the three geometry operators.

MCNP allows easy specification of cells with skewed surfaces. Such a cell may be defined on any convenient auxiliary Cartesian coordinate system. When the user defines the relationship of the auxiliary system to the main problem coordinate system, MCNP will automatically transform the surfaces into the problem coordinate system.

MCNP cells are required for many aspects of the transport calculation, such as constructing the model and specifying the appropriate materials. Cells are also convenient for defining variance reduction parameters and specifying tallies. The volumes of many classes of cells are calculated automatically by MCNP. Other volumes and their statistical errors can be calculated stochastically using MCNP.

SURFACES

MCNP uses 26 different types of surfaces defined by first-, second-, and certain fourth-order equations to create three-dimensional geometries. These surfaces are separated into seven classes: four planes, five spheres, six cylinders, six cones of either one or two sheets, simple quadratics which have their axes parallel to a coordinate axis, general quadratics whose axes are not parallel to a coordinate axis, and elliptical tori with an axis parallel to a coordinate axis. Each of the 26 types of surfaces has a descriptive one- or two- letter mnemonic. All surfaces are defined by specifying the appropriate equation coefficients. Surfaces in the first five classes can also be defined by entering up to three coordinate points. Any surface may be designated a reflecting surface as an aid in defining symmetrical geometries. Most surface areas are calculated automatically by MCNP. The areas of other surfaces and their statistical errors can be calculated stochastically using MCNP.

GRAPHICS

MCNP contains plotting algorithms to produce any two-dimensional slice of a three-dimensional geometry interactively. Improperly specified portions of the two-dimensional slice will be plotted with dashed lines as an aid in finding geometry errors. Many parameters defined in the problem can be displayed on the plots, including surface and cell numbers, neutron and photon importances, weight windows, masses and densities, temperatures, and others. Figure 4 shows two different two-dimensional MCNP plots of a complicated test geometry.

A new interactive three-dimensional geometry set-up code called SABRINA has been developed recently for MCNP geometries. Both surfaces and bodies can be used to define the geometry. Three-dimensional line and color plots are produced interactively. The geometry is automatically checked and appropriate error messages displayed when a flaw is detected. When the geometry is satisfactory, an MCNP input file is produced. SABRINA can also read an MCNP input file as a starting point for geometry modifications. Figure 5 shows two three-dimensional SABRINA plots of the MCNP test geometry in Figure 4.

MONTE CARLO FEATURES

FIXED SOURCES

MCNP has four standard types of fixed source geometries: point, spherical surface, uniform distribution in a cell volume, and a plane source. The point and uniform volume sources are isotropic, the spherical surface source is either an inward or outward cosine distribution, and the plane source is monodirectional. There are seven standard energy and time distributions for which coefficients can be specified: line, linear, Cranberg, Gaussian, Maxwellian, Watt, and evaporation spectra. MCNP has standard options to bias the source in direction, energy, and angle. Both cone and exponential biasing are available for directional biasing.

A SOURCE subroutine is included in MCNP that allows a user to create any source. Input cards are available to simplify the input of data into this source. A second subroutine called SRCDX is available to define the user-source angular-emission probabilities if point detectors and/or DXTRAN spheres are used in the problem.

VARIANCE REDUCTION TECHNIQUES

Analog Monte Carlo calculations are usually effective only for very simple problems. Unbiased variance-reduction schemes split and/or Russian roulette particle tracks, as well as sample from modified probability density functions. Correct use of these schemes can increase the problem efficiency and decrease computer times by up to a factor of 10^x where x is literally unbounded, depending upon the complexity of the problem. Variance reduction techniques make the solution to difficult problems possible with the Monte Carlo method.

MCNP is rich in sophisticated variance reduction techniques which have been thoroughly tested on a wide variety of Los Alamos MCNP problems. **Geometry splitting and Russian roulette** is an easy-to-use and hard-to-abuse technique. Particles tend to flow in the direction of higher importance. **Energy splitting and Russian roulette** perform the same function in energy space. **Weight cutoff** by Russian roulette unbiasedly reduces the number of low-weight particles that are followed so that more computational effort is expended on the important higher-weight particles.

The **exponential transform** or path-length stretching is another standard MCNP variance reduction technique to allow particles to flow in a preferred direction. It is recommended that the transform be used with the **weight window** technique, which keeps particle weights in an appropriate cell-energy window. There can be as many as ten energy weight windows for each surface and/or collision point, for both neutrons and photons.

Another standard variance reduction technique in MCNP is the ability to create a variable number of **forced collisions** in any cell in the problem. **Implicit** or **analog capture** can be used to improve problem efficiency by prolonging or shortening a history. **Energy** and **time cutoffs** will improve a calculation when some regions of phase space are known to be completely unimportant to the problem solution. **Source biasing** of any or all of the particle variables from the source toward the region of interest is available and often can be useful. Another standard MCNP variance reduction technique is the **neutron-induced photon source weight control** by cell in a coupled transport problem to provide the optimum number of photons per neutron in each cell. To reduce the uncertainty in the difference between the tallies in two separate problems, **correlated sampling** is used in MCNP where the starting random number for the ith particle is always the same.

There are three different forms of the **point detector** which reduce the variance of this tally. In addition to the **standard** point detector tally, with a constant flux spherical neighborhood, MCNP provides the **once-more-collided flux estimator** (OMCFE) which samples important tracks close to the detector and produces an additional flux estimate. This 1/R singularity estimator can be an improvement for a point detector in a scattering medium. Another form of the point detector is the **ring detector** which assumes that the flux is a constant on a ring around any of the coordinate axes. Where applicable, the ring detector is usually much more efficient than a point detector because of tally symmetry and the fact that points on the ring are sampled with $1/R^2$ weighting relative to the collision point. It is recommended that all point detector tallies be used with caution because results can be misleading.

The final variance reduction scheme is called **DXTRAN** for deterministic transport and is similar in some respects to the point detector estimator. The DXTRAN method is a way of obtaining large numbers of particles on a user-specified "DXTRAN sphere." DXTRAN makes it possible to obtain many particles in a small region of interest that would oterwise be difficult to sample. Upon sampling a collision or source emission probability, DXTRAN estimates the correct weight fraction that should scatter or be emitted toward the sphere, and arrive without collision. The DXTRAN method then puts this correct weight on the sphere. The exit event is sampled in the usual manner, except that the particle is killed if it tries to enter the sphere because all particles entering the sphere have already been accounted for deterministically. As with point detectors, caution is recommended when using the DXTRAN variance reduction technique.

A new class of variance reducing techniques is currently being studied that adjusts the parameters as the problem learns about itself. As these adaptive techniques become more mature, they will be included in future versions of MCNP.

TALLIES

MCNP has many standard tallies to make it easy for the user to obtain the desired result as a function of cell or surface, energy, time, angle (for the current tally), cell or surface flagging, segmented portions of a cell or surface, tally multipliers, and any user modification to a tally using Subroutine TALLYX. Many tallies are usually allowed in a problem.

MCNP has five basic types of tallies:

1) surface current; 2) surface flux; 3) track length per unit volume flux; 4) flux at a point; and 5) heating tallies.

There are three types of point flux estimators, as discussed in the previous section. All point-detector tallies automatically list separately the direct contribution from the source, as well as the total result. Point detector tallies also include diagnostic tables about the number and sizes of the contributions as an aid to understanding the results.

In addition to the basic tallies themselves, there are several standard tally modifiers that allow easy tally conversion to the quantity of interest. Tallies can be either particles or particle energy. Energy-, time-, and angular-dependent user-specified multipliers are available. General reaction multipliers to any tally are available in MCNP that can be sums and/or products of any of the various reactions from the data tables. Segmented tallies can be made in a cell or on a surface by using additional surfaces that are not a part of the problem geometry. Summations of tallies over cells and surfaces are available as well. Tallies can be further examined by flagging those particles which have crossed one or more specified cell(s) or surface(s). This feature can be very useful in determining which particle geometry paths are the most important contributors to the result.

In the event that the tally desired is not available as a standard MCNP feature, the user may define the required tally by providing the appropriate FORTRAN to the user-defined Subroutine TALLYX. This subroutine is called just before the tally is actually made in MCNP. This feature is useful for such results as the tally by collision number, charged particle generation, and many others.

ESTIMATED TALLIES AND RELATIVE ERRORS

Each tally is estimated by $\bar{x} = \sum_{i=1}^{n} x_i/n$ where x_i is the tally from the ith history and n is the total number of histories. The estimated relative error (RE) of $\bar{x}$ is RE = $s_{\bar{x}}/\bar{x} = \sqrt{\Sigma x_i^2/(\Sigma x_i)^2 - 1/n}$ where the x_i and x_i^2 are updated at the end of each history in MCNP to include history correlations. This relative error is printed for each tally by MCNP. Experience has shown that reliable confidence intervals are generated when the relative error is less than 0.10. For point-detector tallies, it is recommended that the relative error be less than 0.05 because these tallies will typically have a much larger variance of the variance than other tallies.

TALLY FIGURE OF MERIT

Since the RE^2 is proportional to 1/n and the problem computer time T is proportional to n, a figure of merit (FOM) for a tally is defined to be FOM = $[(RE)^2T]^{-1}$. With this definition, more efficient tallies will have a larger FOM. The FOM is, therefore, useful to compare problem efficiencies when using different variance reduction techniques. The FOM should be approximately a constant for each tally in a problem and is thus also a tally reliability indicator. In addition, the FOM can be used for estimating the amount of computer time required to achieve a desired relative error. Tally fluctuation charts which contain the FOM for one user-specified tally bin of each tally as a function of the number of histories are automatically printed at the end of each MCNP problem.

CRITICALITY

MCNP has the capability to calculate k_{eff} eigenvalues and removal lifetimes for both sub- and super-critical systems, and the associated relative errors. The calculation is performed as a series of generations of neutrons, estimating both of the above quantities for that generation, as well as the average over a user-specified number of preceding generations. Fission is treated as either implicit or analog "capture." New source points for the next generation are based on the fission "captures" from the previous generation.

There are seven estimators for k_{eff}: collision, absorption, track length, and the four combinations of these single estimators. The three removal lifetime estimators are collision, absorption and a combined collision-absorption estimator. Covariances are included in the calculation of the relative errors of the combined estimators.

TALLY GRAPHICS

MCPLOT is a completely interactive post-processing graphics code for MCNP problems. MCPLOT uses CGS or DISSPLA graphics and can plot any tally and its associated relative error generated by MCNP. In addition to the standard two-dimensional line plots and spline fits of the tally results, MCPLOT can also graphically present results in both two-dimensional contour and three-dimensional plots. MCPLOT has a flexible command structure for ease of use. Figures 6 and 7 show typical MCPLOT graphics of MCNP results.

USER ASPECTS

A large amount of effort has gone into making MCNP as user-oriented as possible. All aspects of the code, including data bases, input, output, documentation, and the software itself, reflect this code philosophy and level of effort.

INPUT

The MCNP input file contains three major sections, each separated by a blank card: 1) cell definitions with material, density, and sense-labeled surfaces; 2) surface cards; and 3) data cards which define all other aspects of the problem. All cards have free-form input with a mnemonic or cell/surface number in columns 1 to 5 and data in columns 6 to 72. Useful input-card aids are available for interpolating, repeating, multiplying and jumping over data entries. There are approximately 100 types of data cards for complete problem specification and control. To minimize user-input errors, MCNP performs over 200 checks of the input file which result in warning and/or fatal error messages to the output file. A fatal error terminates the problem while a warning error should be closely examined by the user even though the problem will continue to run.

OUTPUT

MCNP produces a full description of all aspects of the completed problem in the output file. The entire input file is printed along with the results of input operations such as cell-volume and surface-area calculations, source parameters including biasing, variance-reduction parameters, and tallying structures. All of the appropriate information about the data sets selected for the problem are included in the output, as well as all warning and fatal error messages.

The starting parameters of the first fifty source particles are printed. If a history gets lost in the geometry, that history is automatically rerun to produce an event log in the output file of the sequence of events leading to the difficulty. At the conclusion of the problem, several summary tables are printed. A global summary table of net particle creation and loss by various physical- and variance-reduction categories is printed, followed by cell and nuclide activity tables and particle weight balance tables. Photon-production information by cell and isotope is also included for a coupled neutron-photon problem. Finally, the tallies and relative errors are printed, followed by the tally fluctuation charts.

USER FEATURES

MCNP is rich in features provided especially for the user. Besides the extensive input file checks and more than 200 possible error messages, the user is also allowed to put comments both in the input file and in various places in the output file. MCNP has a full restart capability for problems which need to be continued.

There are several input cards for the user to insert information into the MCNP subroutines and arrays. Certain unused variables are included in COMMON and in the banking list for the user's convenience in modifying the code.

The completely interactive geometry graphics capability is of great benefit in setting up geometries. A VOID card is available to remove material from the problem to check overall particle tracking in the geometry.

In the data area, the user has many neutron cross-section evaluations and data forms from which to choose. The user is encouraged to select those evaluations that are most appropriate for the problem being solved. In the event the user chooses not to make these selections of neutron cross sections, MCNP will select an evaluation for each unspecified element or isotope. If the user has generated cross sections in an acceptable MCNP format, it is easy to include them in an MCNP problem as well.

Another important MCNP feature is Subroutine TALLYX which allows a user to write FORTRAN for inclusion in this subroutine. Virtually any tally can be made in MCNP with this valuable user feature, including a user-defined multiple tally capability.

The output from an MCNP problem is the end result of all of the calculations. The user can control what appears in the output and can also change the order of printing of the various tally variables. If a particle gets lost, the output will contain a log of all of the events of that history. For point detectors and DXTRAN, diagnostic tables of the results from these techniques aid the user in understanding the tallies. The tally fluctuation charts at the end of the problem are of great value to check quickly and easily the tally reliability and efficiency.

MCNP has several diagnostic interrupts for the user to check how the problem is proceeding. MCNP has source efficiency and long history checks, either of which can terminate the problem. The user also has the ability to change the starting random number for a problem.

DOCUMENTATION

MCNP is an extremely well documented code system. Over the years, the work on MCNP has spawned two books:[8,9] 1) E. D. Cashwell and C. J. Everett, "A Practical Manual on the Monte Carlo Method for Random Walk Problems," Paragon Press, 1957; and 2) L. L. Carter and E. D. Cashwell, "Particle Transport Simulation with the Monte Carlo Method," ERDA Critical Review Series, TID-26607, 1975. The MCNP manual,[1] LA-7396-M (revised in April, 1981), is 511 pages long and contains chapters and appendices on Monte Carlo theory, physics, input, output, examples, data, and coding aspects of MCNP. The manual has been used as a textbook for Monte Carlo studies at several major universities in the United States. Additional instruction is available from 25 hours of video tapes from the 1983 MCNP Workshop at Los Alamos. Both the manual and the videotapes are available from RSIC in Oak Ridge, Tennessee.

Theoretical and applied developments of various aspects of MCNP have been reported in numerous refereed journal articles, Los Alamos reports, and professional society meetings. Two recent notable Los Alamos reports are:[10,11] 1) C. J. Everett and E. D. Cashwell, "A Third Monte Carlo Sampler," LA-9721-MS, 1983; and 2) T. E. Booth, "A Sample Problem For Variance Reduction in MCNP," LA-10363, 1985.

TYPICAL APPLICATIONS

MCNP is used at Los Alamos for many different types of calculations which is the main reason it is designed to be such a general transport code. MCNP is used by many technical divisions to analyze physics experiments, design nuclear safeguards non-destructive assay systems, perform criticality analyses, design radiation shields, address health-physics problems, calculate magnetic fusion neutronics, analyze reactor designs, calculate material activations, or design nuclear instrumentation.

MCNP is also used extensively outside Los Alamos by various professional institutions for magnetic fusion energy (MFE) calculations on the MFE network, well logging calculations, determinations of radiological doses, physics experiments, spacecraft radiation modeling, radiation damage studies, and Monte Carlo applications at major universities. The feedback that X-6 receives from these applications is invaluable for improving MCNP and its associated data bases.

FUTURE VERSIONS OF MCNP

MCNP has steadily improved in its capabilities over the years. This trend will continue with the new Version 3A of MCNP and future versions.

MCNP VERSION 3A

Version 3A of MCNP has been under development for one year and will incorporate over one-hundred significant improvements. The biggest single improvement in Version 3A will be a highly generalized source routine that will almost eliminate the need for Subroutine SOURCE, although it will still exist. Many complicated source distributions can be specified with the new source, along with source-parameter biasing functions if necessary. An optional quasi-random number generator for the source will be added to improve sampling of the source variables. The free-gas thermal neutron treatment will be improved and expanded to include all atomic numbers and higher temperatures. A weight-window generator, which calculates the scoring function by energy and cell, will be included to assist the user in setting optimum cell- and energy-dependent weight windows. A Monte Carlo surface-source capability will be added to MCNP to allow the linking of two or more MCNP problems while preserving the correct statistical error estimates. Both the input and structure of the code have been modified to aid in adding other types of particles as those physics routines become available. Some modification to the code structure has been done also in anticipation of multitasking.

Version 3A will contain many new user features such as column cell input, cell parameters on cell cards, combined neutron and photon tallies, improved graphics, and improved tally and problem information. The MCNP manual is being rewritten by seventeen different members of Group X-6. A welcome addition to the new manual is a primer to help the new user more quickly and easily understand the code capabilities. Both Version 3A and the new manual will be released to RSIC in the fall of 1985.

MCNP SPECIAL VERSIONS

There are several special-purpose versions of MCNP that are available for non-standard applications. These versions are patches to MCNP and are not considered to be production codes.

One such version is the weight-window generator patch which is being improved and incorporated into MCNP Version 3A. A multigroup version of MCNP called MCMG is used for making forward and adjoint calculations with multigroup data libraries. Another important version of MCNP uses two different but associated techniques of perturbation Monte Carlo to solve directly for the differences in problems that can be both vastly and minutely different. A thick-target bremsstrahlung generation model for MCNP has just been developed and is being tested for problems where this photon source is important.

Another patch to MCNP is used to calculate the response of neutron coincidence counters. A delayed-neutron version of MCNP exists as well. A patch to MCNP to calculate the variance of the variance exists for statistical studies. A version of MCNP with a repeated-structures capability exists for this class of problems. A multitasking version of a scaled-down MCNP is available and is being used as a FORTRAN 77 compiler checker for the Cray X-MP.

MCNP AFTER VERSION 3A

In the future, MCNP will stand for Monte Carlo N Particle where N particle varieties will be transported. The new particles to be added will be electrons and heavy ions, followed by high-energy physics particles generated with intranuclear cascade models. Future versions will incorporate some of the special MCNP versions mentioned above as needs dictate. Monte Carlo methods of the future, which will become much more adaptive, will be added to MCNP. The data libraries will be improved and expanded to meet more demanding needs. SABRINA and MCPLOT will be integrated into MCNP to produce a highly interactive code. The portability aspects of the code will be examined further and improved. As new computer architectures emerge, the MCNP code system will be modified to take advantage of these new developments as they apply to a general user-oriented production Monte Carlo code.

CONCLUSIONS

MCNP is a very general Monte Carlo neutron photon transport code system with approximately 250 person years of Group X-6 code development invested. It is extremely portable, user-oriented, and a true production code as it is used about 60 Cray hours per month by about 150 Los Alamos users. It has as its data base the best cross-section evaluations available. MCNP contains state-of-the-art traditional and adaptive Monte Carlo techniques to be applied to the solution of an ever-increasing number of problems. Excellent user-oriented documentation is available for all facets of the MCNP code system. Many useful and important variants of MCNP exist for special applications.

The Radiation Shielding Information Center (RSIC) in Oak Ridge, Tennessee is the contact point for worldwide MCNP code and documentation distribution. A much improved MCNP Version 3A will be available in the fall of 1985, along with new and improved documentation. Future directions in MCNP development will change the meaning of MCNP to Monte Carlo N Particle where N particle varieties will be transported.

REFERENCES

1. Radiation Transport Group X-6, "MCNP - A General Monte Carlo Code for Neutron and Photon Transport," LA-7396-M (Revised April 1981).

2. R. R. Johnston, "A General Monte Carlo Neutronics Code," LAMS-2856 (1963).

3. E. D. Cashwell, J. R. Neergaard, W. M. Taylor, and G. D. Turner, "MCN: A Neutron Monte carlo Code," LA-4751 (1972).

4. E. D. Cashwell, J. R. Neergaard, C. J. Everett, R. G. Schrandt, W. M. Taylor, and G. D. Turner, "Monte Carlo Photon Codes: MCG and MCP," LA-5157-MS (1973).

5. C. J. Everett and E. D. Cashwell, "MCP Code Fluorescence-Routine Revision," LA-5240-MS (1973).

6. R. E. MacFarlane, R. J. Barrett, D. W. Muir, and R. M. Boicourt, "The NJOY Nuclear Data Processing System: User's Manual," LA-7584-M (December 1978).

7. E. Storm and H. I. Israel, "Photon Cross Sections from 0.001 to 100 Mev for Elements 1 through 100," LA-3753 (1967).

8. E. D. Cashwell and C. J. Everett, "A Practical Manual on the Monte Carlo Method for Random Walk Problems," Pergamon Press, Inc., New York (1959). [Also LA-2120, (1957)].

9. L. L. Carter and E. D. Cashwell, "Particle-Transport Simulation with the Monte carlo Method," ERDA Critical Review Series, TID-26607 (1975).

10. C. J. Everett and E. D. Cashwell, "A Third Monte Carlo Sampler (A Revision and Extension of Samplers I and II)," LA-9721-MS (March 1983).

11. T. E. Booth, "A Sample Problem for Variance Reduction in MCNP," LA-10363 (to be published in 1985).

TABLE I

Neutron Cross-Section Data that are Available in the MCNP Code Package from RSIC

Z	Nuclide	Z	Nuclide
1	^{1-3}H	28	Ni, ^{58}Ni
2	He, ^{3-4}He	29	Cu
3	^{6-7}Li	31	Ga
4	^{9}Be	40	Zr
5	B, $^{10-11}B$	41	^{93}Nb
6	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, $^{182,184,186}W$
19	K	78	Pt
20	Ca	79	^{197}Au
22	Ti	82	Pb
23	V	90	^{232}Th
24	Cr	92	$^{233-240}U$
25	^{55}Mn	94	$^{238-241}Pu$
26	Fe	95	^{242}Am

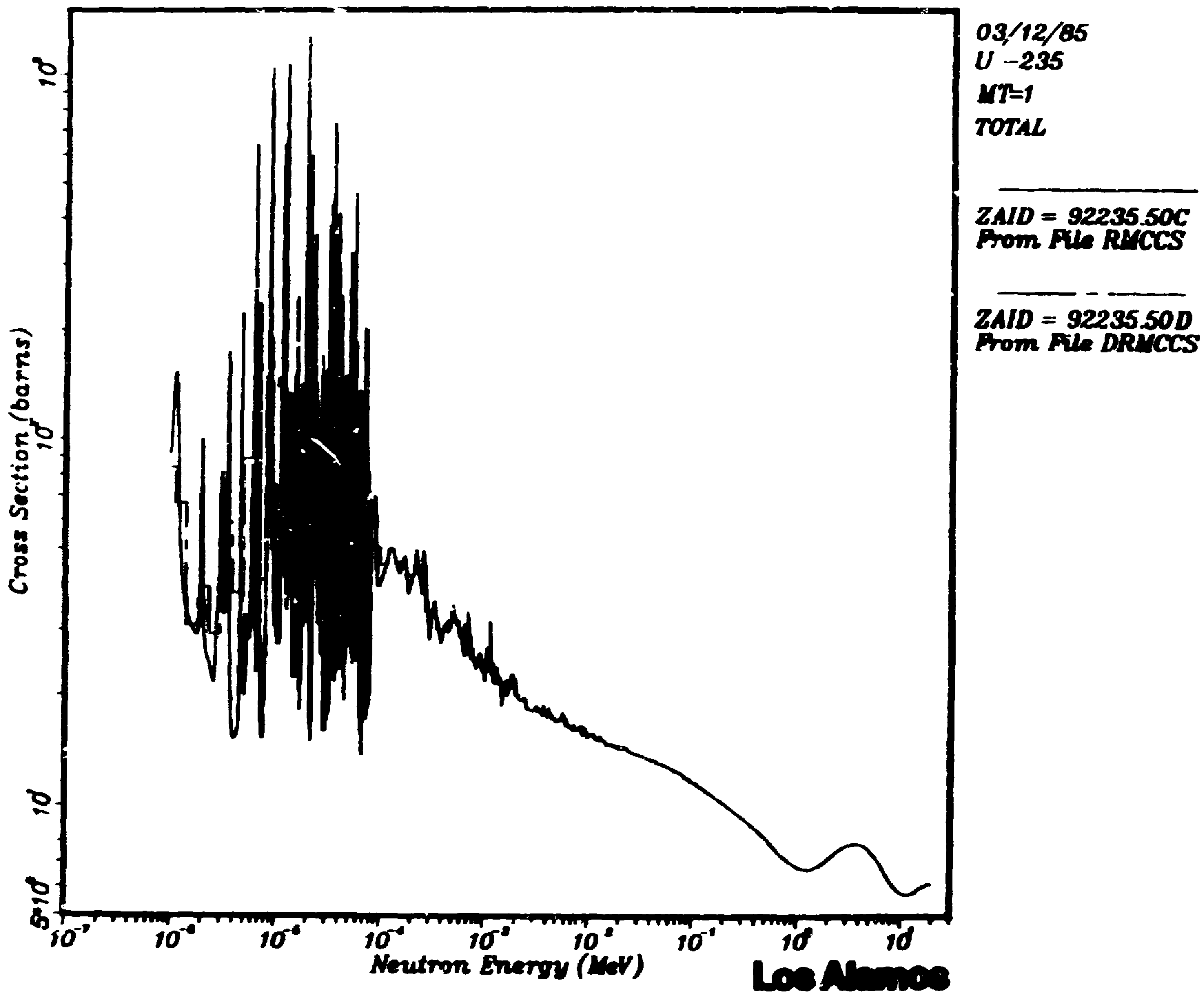


Fig. 1. Total ^{235}U neutron cross section from 10^{-6} to 20 MeV for both the continuous and discrete reaction (histogram) forms of MCNP data.

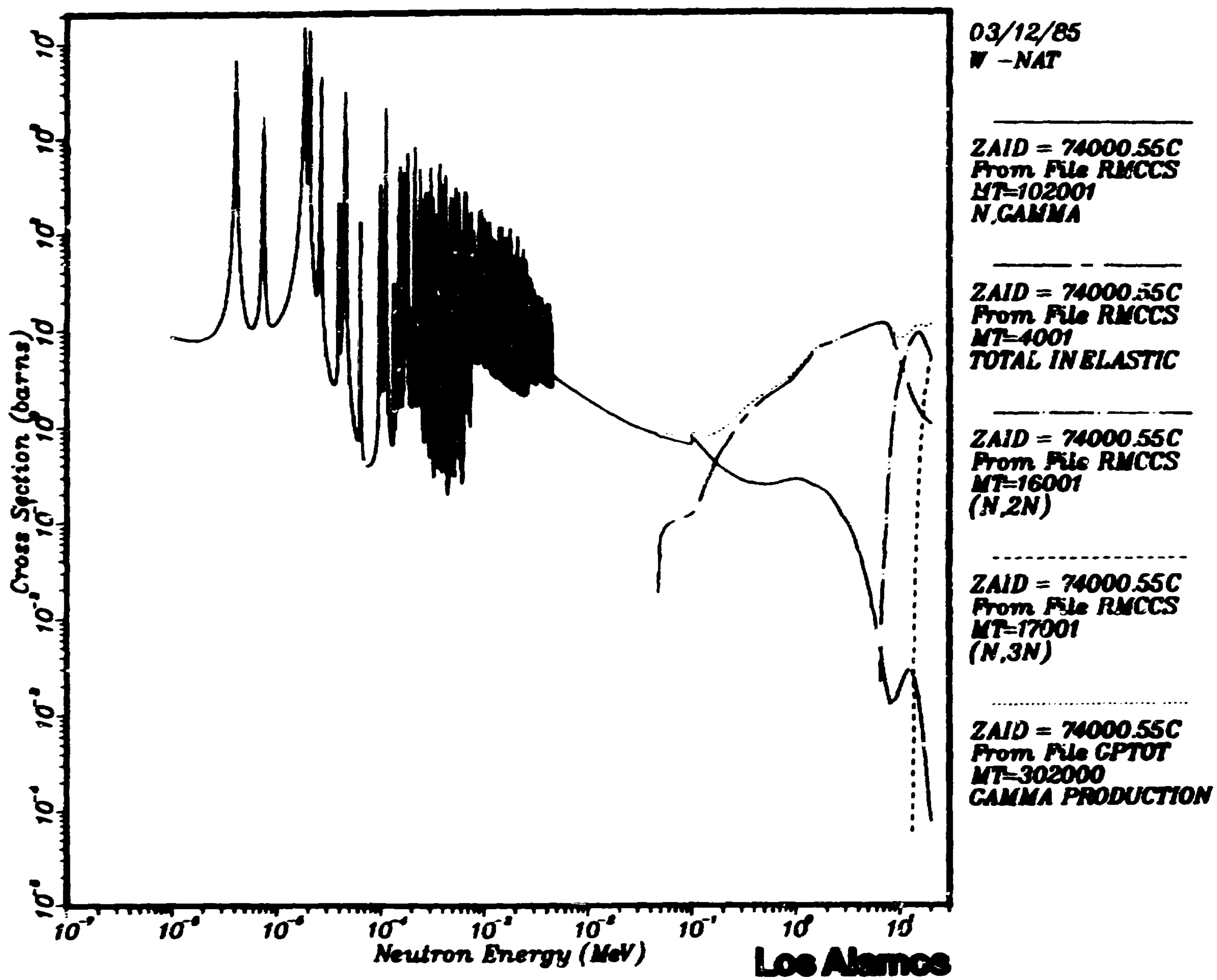


Fig. 2. Continuous natural-tungsten photon-production cross sections used in MCNP for various reactions and total photon production.

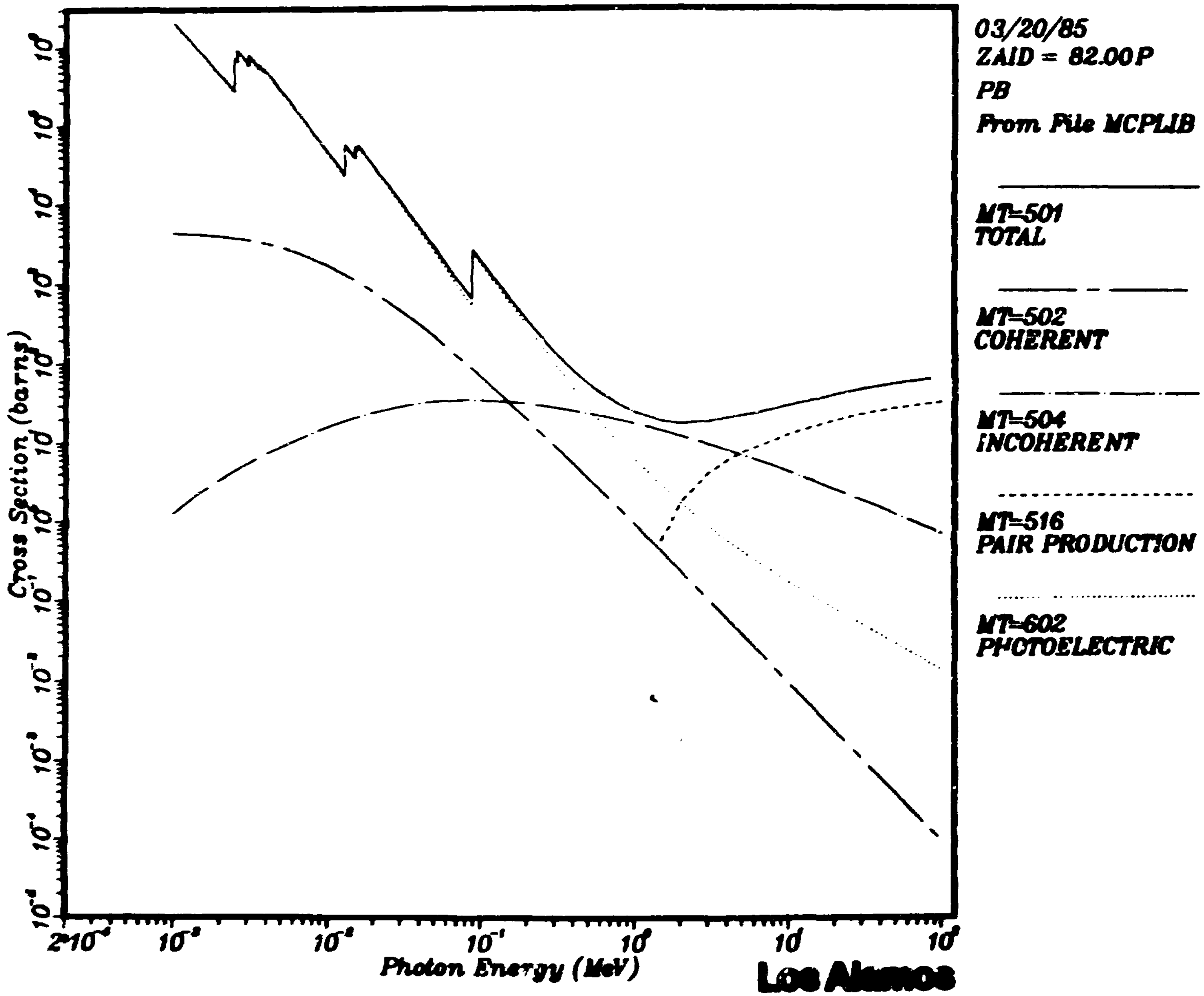


Fig. 3. Total and partial lead photon cross sections used in MCNP from 10^{-3} to 100 MeV.

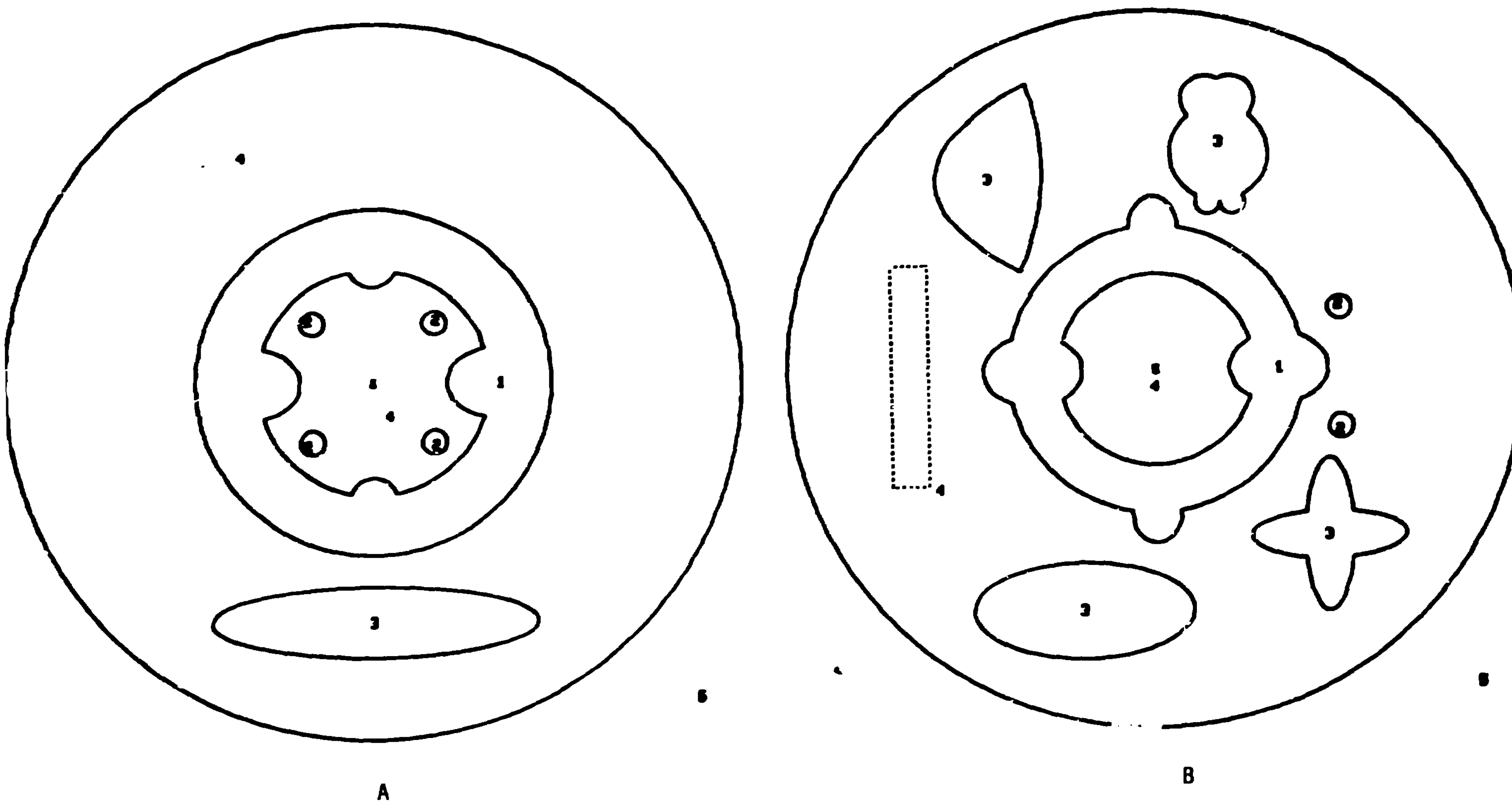


Fig. 4. MCNP two-dimensional plots of different slices of a complicated geometry with cell numbers. The dashed rectangle in B represents the "cookie-cutter" area of acceptance for source particles.

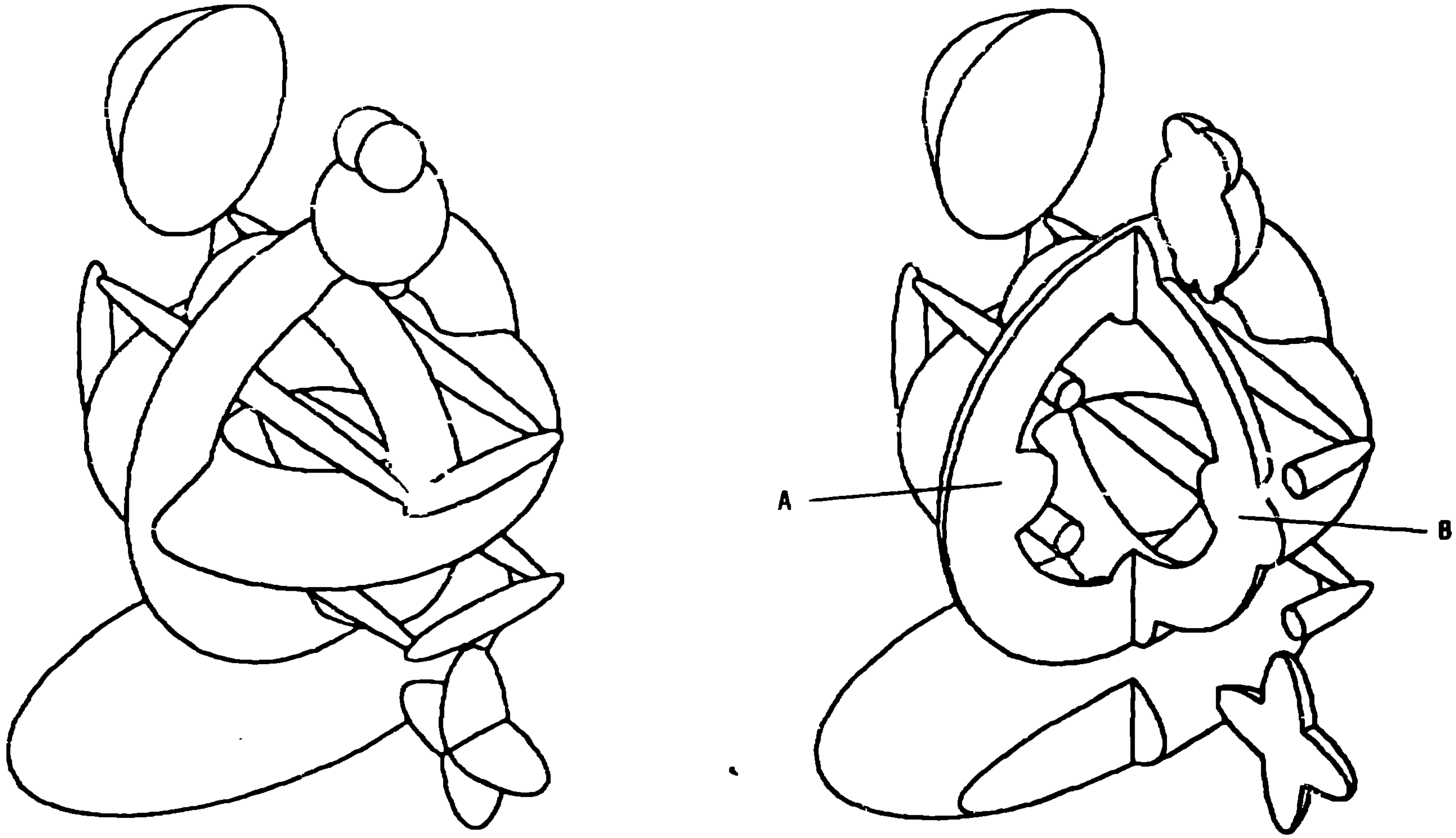


Fig. 5. SABRINA three-dimensional plots of a complicated MCNP geometry. The plot on the right is a cutaway plot of the body on the left. The A and B correspond to the MCNP plots shown in Fig. 4.

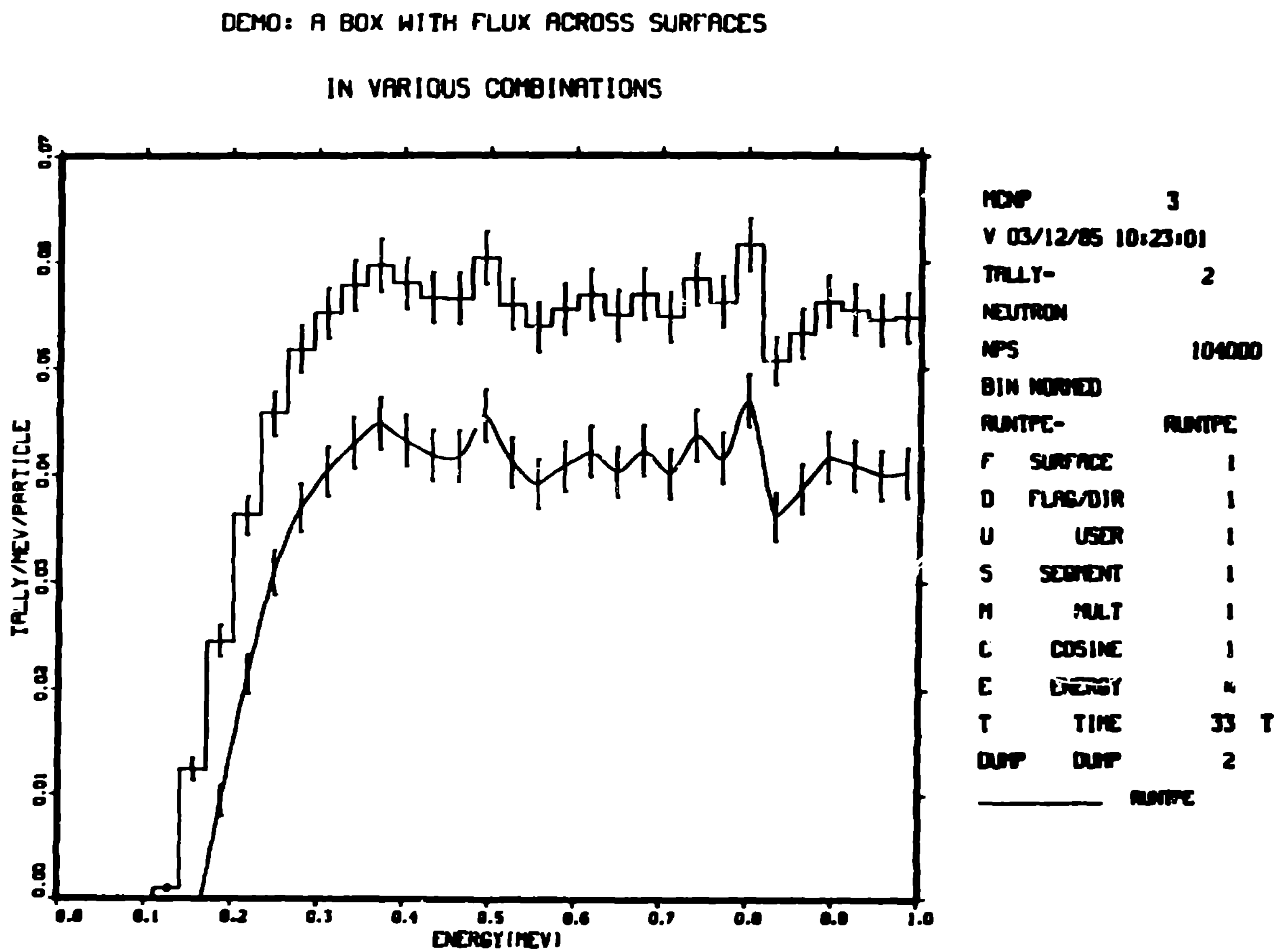
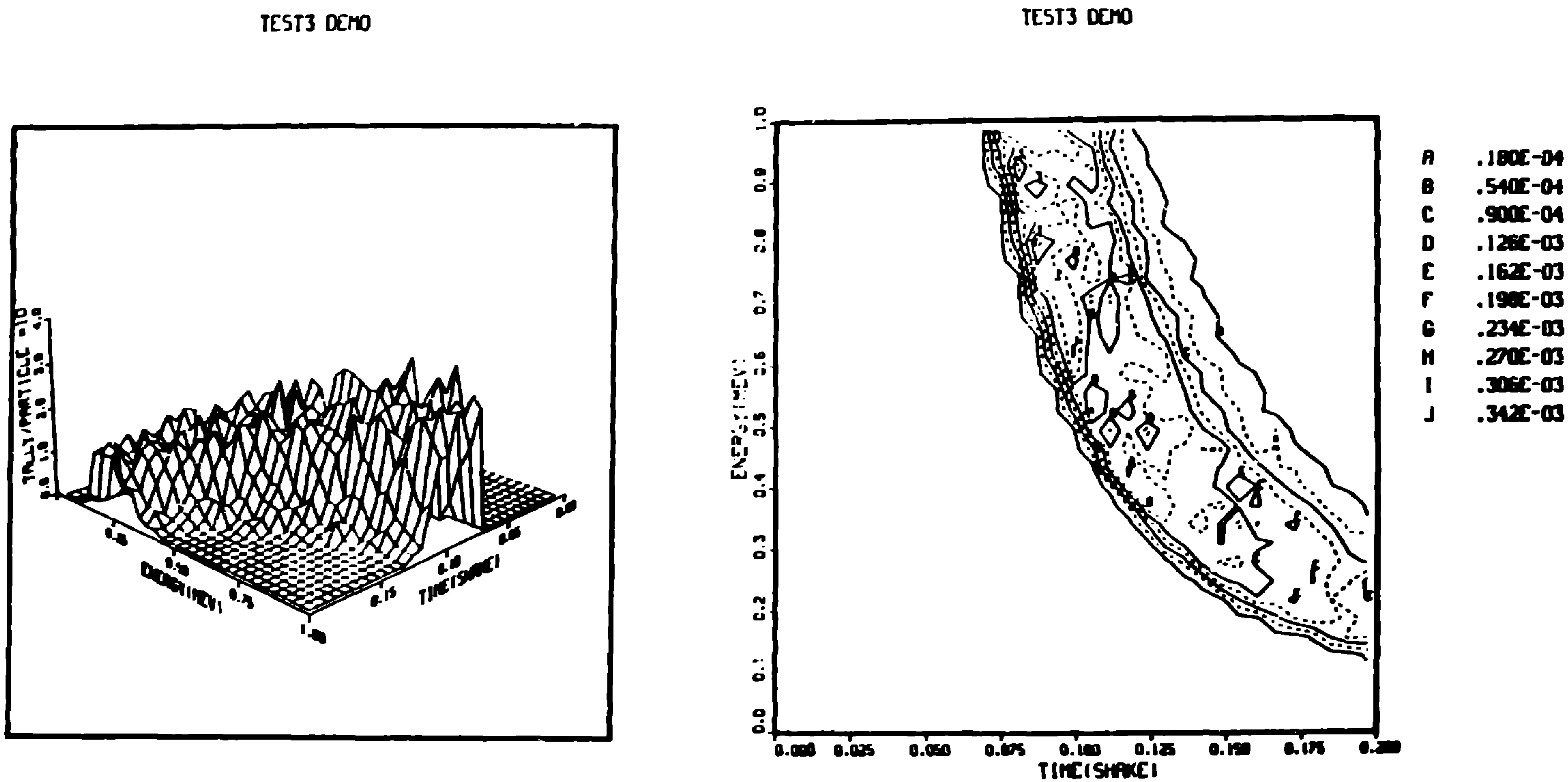


Fig. 6. MCNP output graphically displayed using MCPLOT. The lower curve is a scaled version of the upper histogram fit with a spline function.



A

B

Fig. 7. Plot A is a three-dimensional tally representation available from MCPLOT. Plot B is the same tally represented by the two-dimensional contours option in MCPLOT.