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Title: The History of Monte Carlo and MCNP at Los Alamos

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The History of Monte Carlo and MCNP at Los Alamos

Art Forster, Michael E. Rising, and Avneet Sood

July 2021

LA-UR-21-XXXX

Abstract

The Monte Carlo method for radiation particle transport has its origins at LANL dating back to the 1940's. The creators of these methods were Drs. Stanislaw Ulam, John von Neumann, Robert Richtmyer, and Nicholas Metropolis. Monte Carlo methods for particle transport have been driving computational developments since the beginning of modern computers; this continues today. In the 1950's and 1960's, these new methods were organized into a series of special-purpose Monte Carlo codes, including MCS, MCN, MCP, and MCG. These codes were able to transport neutrons and photons for specialized LANL applications. In 1977, these separate codes were combined to create the LANL Monte Carlo N-Particle (MCNP) radiation particle transport code. In 1983, MCNP3 was released for public distribution to the Radiation Safety Information Computational Center (RSICC). MCNP6.3 will be released late in 2021. Each year, LANL has ~100 MCNP new users and RSICC ~1250 new licenses distributed. This talk will review the Los Alamos history of the development of the modern Monte Carlo method and MCNP.

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Outline

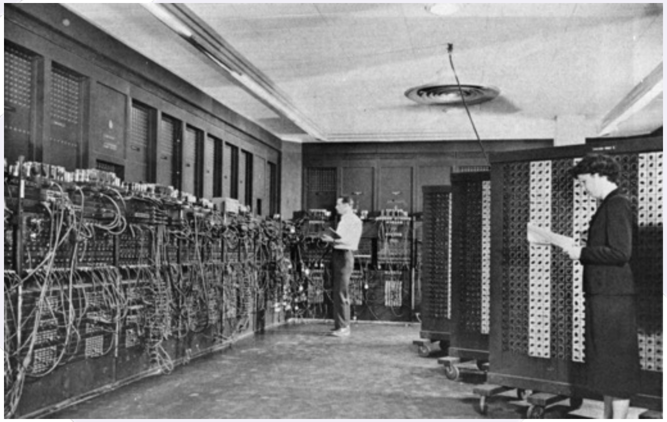
- Origins of the Monte Carlo method
  - Development of electronic computers and the Monte Carlo method occur simultaneously
  - Ulam, Von Neumann, Richtmeyer, Metropolis, Fermi
- Early Los Alamos Monte Codes MCS, MCN, and Others
  - several special purpose codes were developed
  - codes were merged to facilitate efficient code development
- Emergence of MCNP
- MCNP's history

The Origins of Monte Carlo – 1946 Stanislaw Ulam

- “*The year was 1945. Two earthshaking events took place: the successful test at Alamogordo and the building of the first electronic computer*” – N. Metropolis
- The method was invented by Stanislaw Ulam in 1946 playing Solitaire while recovering from an illness.
- “*After spending a lot of time trying to estimate success by combinatorial calculations, I wondered whether a more practical method…might be to lay it out say one hundred times and simply observe and count the number of successful plays*” – S. Ulam


Stanislaw Ulam


WITH HIS WIFE, Klara, Von Neumann stood outside their Princeton home just after he was named to AEC. Dog has mathematical name, Inverse.


ENIAC– the first electronic computer, University of Pennsylvania. Solved ballistic trajectory problems for Army Ballistics Research Lab. Used electron tubes instead of mechanical counters. Minutes instead of days. Declassified in 1946.

“Stan Ulam, John von Neumann, and the Monte Carlo Method,” R. Eckhardt, Los Alamos Science Special Issue 1987.

The Origins of Monte Carlo

J. Von Neumann invented scientific computing in the 1940's

- Stored programs now called software
- Algorithms/Flowcharts
- Hardware design

Klara von Neumann ran the earliest calculations on ENIAC

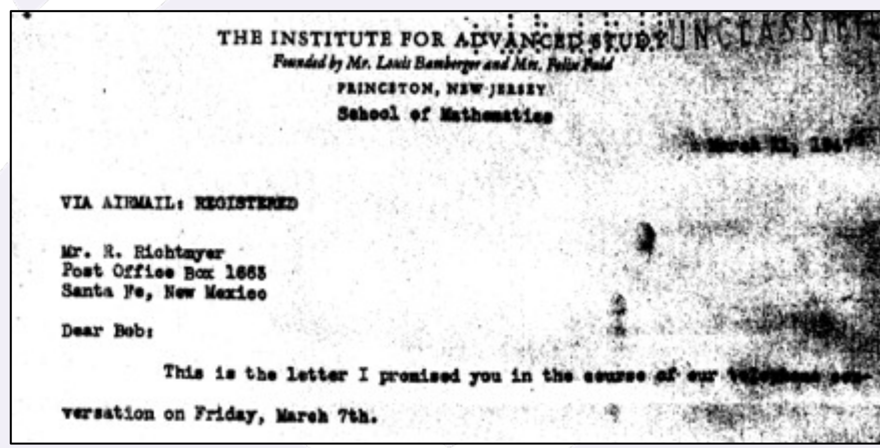

John von Neumann


Klara and John von Neumann

- Ulam describes this idea to John von Neumann in a conversation in 1946

- Von Neumann is intrigued
  - 1943: Electro-Mechanical computers solved non-linear diff. eq. via production line. Punch card used for every point in space/time
  - New computers could count/arithmetic and hence solve difference equations (BRL at Aberdeen, MD)
  - Statistical sampling on electronic computers
  - Especially suitable for exploring neutron chain reactions in fission – neutron multiplication rates

- R.D Richtmyer and J. von Neumann "Statistical Methods in Neutron Diffusion", Los Alamos (LAMS-557) April 9, 1947.
  - Detailed letter from John von Neumann to Robert Richtmyer describing a conversation in March 1947
    - "I have been thinking a good deal about the possibility of using statistical methods to solve neutron diffusion and multiplication problems in accordance with the principle suggested by Stan Ulam"
    - Letter contained 81-step pseudo code for using MC for particle transport


THE INSTITUTE FOR ADVANCED STUDY UNCLASSIFIED
Founded by Mr. Louis Bamberger and Mrs. Felix Fuld
PRINCETON, NEW JERSEY
School of Mathematics

March 11, 1947

VIA AIRMAIL: REGISTERED

Mr. R. Richtmyer
Post Office Box 1663
Santa Fe, New Mexico

Dear Bob:

This is the letter I promised you in the course of our telephone conversation on Friday, March 7th.

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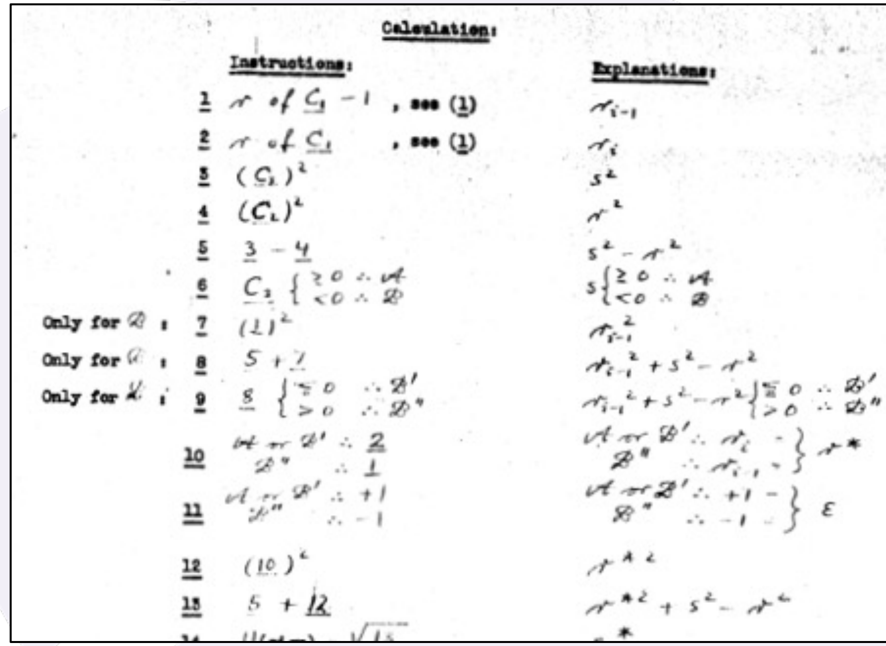
The First Monte Carlo (pseudo) Code - 1947

- Von Neumann's Assumptions:
  - Time-dependent, continuous energy, spherical but radially-varying, 1 fissionable material, isotropic scattering and fission production, fission multiplicities of 2,3, or 4
- Suggested 100 neutrons each to be run for 100 collisions
  - Thought these were too much
- Estimated time: 5 hrs on ENIAC
- Richtmyer's response:
  - Very interested in idea and proposed suggestions
    - Allow for multiple fissionable materials, no fission spectrum energy dependence, single neutron multiplicity, run for computer time not collisions
- ENIAC: first calculations run April/May 1948
  - Code finalized in **December 1947**;
  - Continuous energy neutrons, fission spectra and XS tabulated at interval mid-points, histogram energy-dependence of XS, pseudo-RN.

Thomas Haight, et al., "Los Alamos Bets on ENIAC: Nuclear Monte Carlo Simulations, 1947-1948, IEEE Ann. Of History of Comp July-Sept 2014



R.D Richtmyer and J. von Neumann "Statistical Methods in Neutron Diffusion", Los Alamos (LAMS-557) April 9, 1947.

April 2, 1947

Professor John vonNeumann,
The Institute for Advanced Study,
School of Mathematics
Princeton, New Jersey

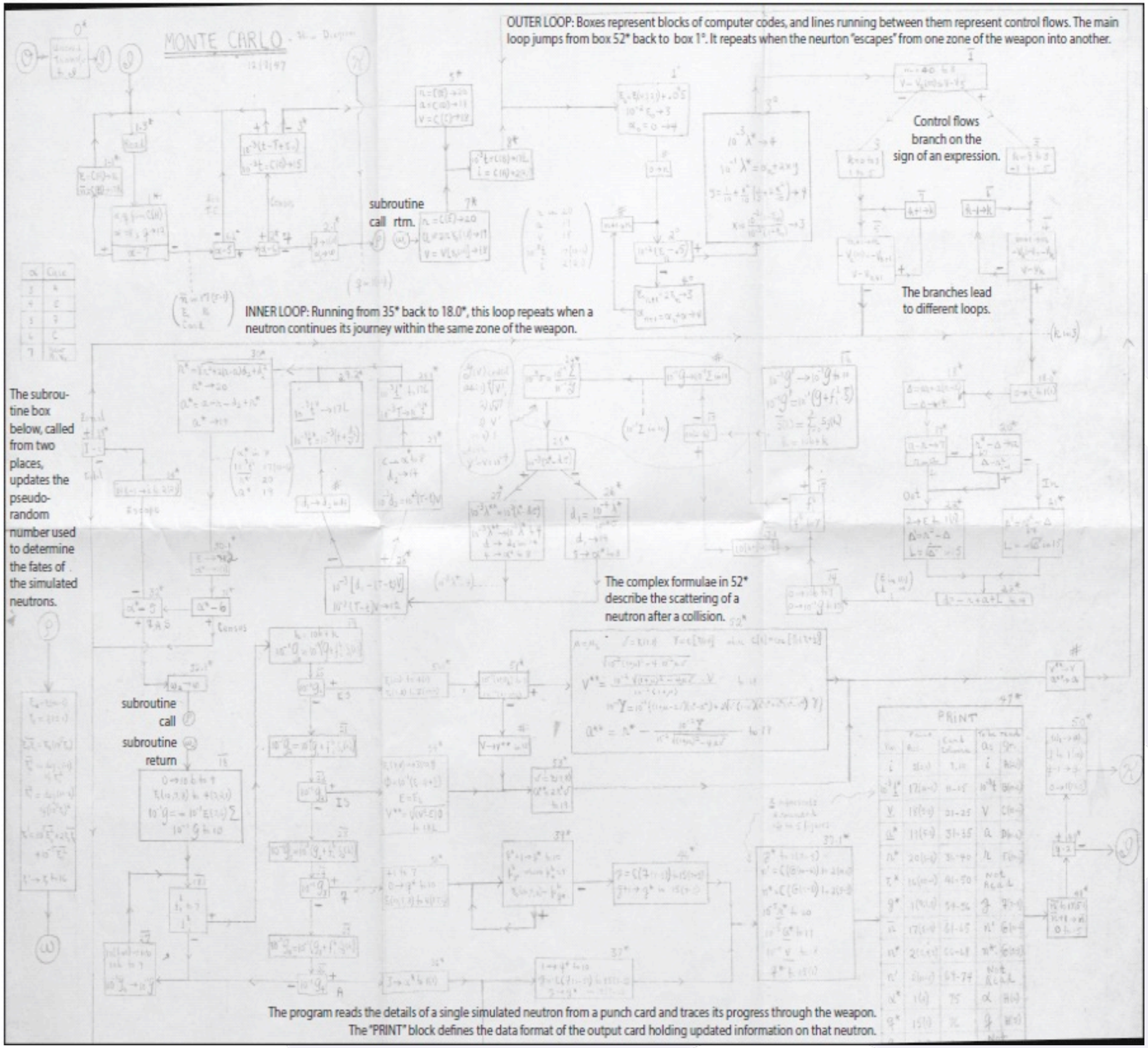
Dear Johnny:

As Stan told you, your letter has aroused a great deal of interest here. We have had a number of discussions of your method and Bengt Carlson has even set to work to test it out by hand calculation in a simple case.

ENIAC in Action: MC Program / flowchart

Boxes	Function
1* – 8*	Read a card and store neutron characteristics
1° – 4°	Calculate random parameter λ*
$\overline{1}$ – $\overline{7}$	Find neutron's velocity interval
18* – 23*	Calculate distance to zone boundary
$\overline{14}$ – $\overline{17.1}$, 24*	Calculate cross-section of material in zone
25* – 27*	Determine if terminal event is collision or escape
28* – 30*	Determine if a census comes first
31* – 35*	Discriminate between terminal events
Subroutine ρ/ω	Refresh random number
$\overline{18}$ – $\overline{27}$	Determine collision type
51* – 52*	Elastic scattering
53* – 54*	Inelastic scattering
36* – 39*, 46*	Absorption/fission
37.1*, 47* – 50*	Print card and restart main loop

Thomas Haight, Mark Priestley, and Crispin Rope, "ENIAC in Action: Making and Remaking the Modern Computer," MIT Press 2016



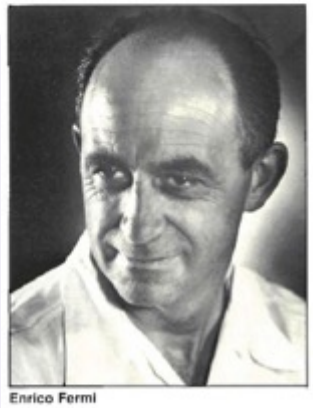
Enrico Fermi: Independently developed Monte Carlo!

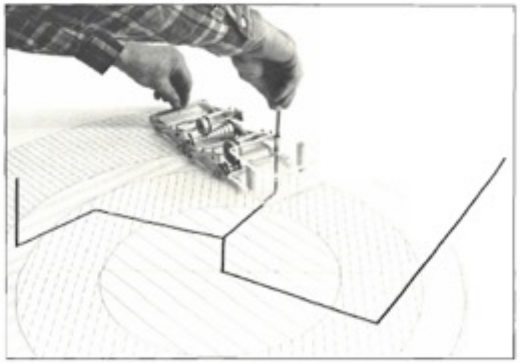
- Emilio Segre, Fermi's student and collaborator:
  - "*Fermi had invented, but of course not named, the present Monte Carlo method when he was studying the moderation of neutrons in Rome. He did not publish anything on the subject, but he used the method to solve many problems with whatever calculating facilities he had, chiefly a small mechanical adding machine*"

- Astonished Roman colleagues when he would predict experimental results remarkably accurately. He revealed that he used statistical sampling techniques whenever insomnia struck.

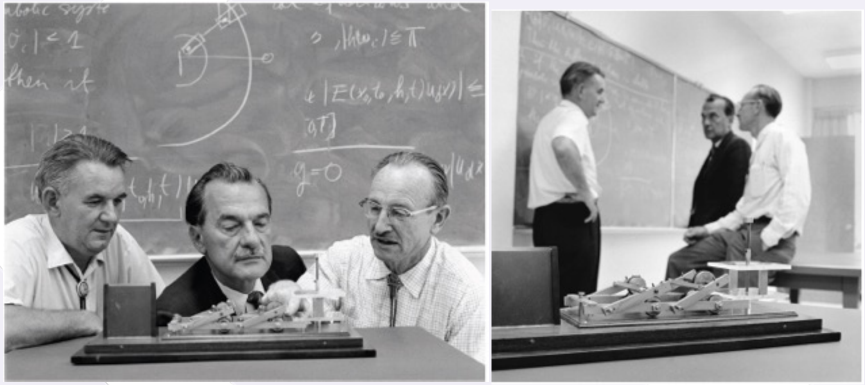
- 15 years prior to Ulam

- While in Los Alamos and awaiting ENIAC's move, he created an analog device to study neutron transport.
  - Called FERMIAC
  - Generated the site of next collision based upon characteristics of material; Another choice was made at boundary crossing; "slow" and "fast" neutron energies





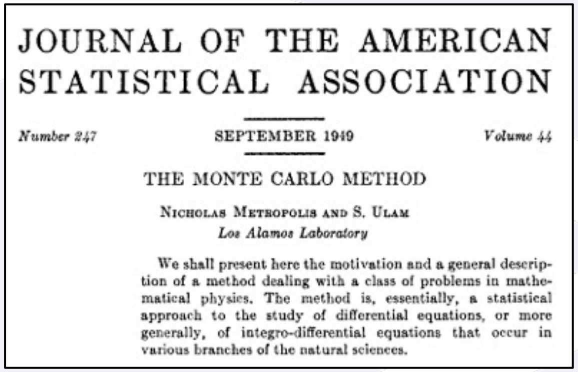
FERMIAC



Los Alamos Scientists: Bengt Carlson, Nicholas Metropolis, LDP King with Fermiac (1966)

MANIAC – Nicholas Metropolis

- Post-war ENIAC started a revolution that continues today
- MANIAC – Mathematical and Numerical Integrator and Computer
  - Was a product of Nicholas Metropolis at LANL; borrowed concepts from von Neumann's IAS, operational in 1952;
  - MADCAP – high-level language and compiler
  - Rapid growth of computing: AVIDAC (Argonne) ORACLE (Oak Ridge), ILLIAC (U of I)
  - Special effort that helped bind Von Neumann, Fermi, Bethe, Teller, Ulam, Feynman, others in post-war efforts. MANIAC was a fascination.
  - First time "Monte Carlo" appears in publication:
    - Nicholas Metropolis and S. Ulam , "The Monte Carlo Method," *Journal of the American Statistical Association* Vol. 44, No. 247 (Sep., 1949)
  - MC on MANIAC used for multiple problems other than radiation transport:

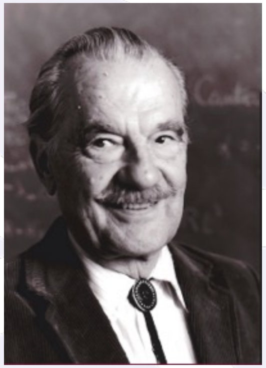

JOURNAL OF THE AMERICAN STATISTICAL ASSOCIATION

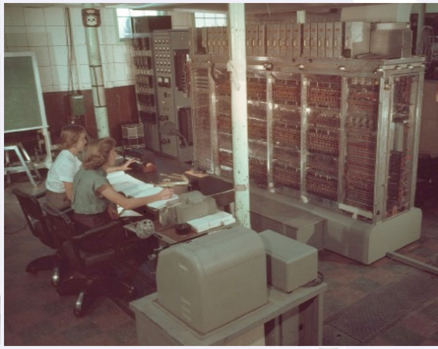
Number 247 — SEPTEMBER 1949 — Volume 44

THE MONTE CARLO METHOD

NICHOLAS METROPOLIS AND S. ULAM
Los Alamos Laboratory

We shall present here the motivation and a general description of a method dealing with a class of problems in mathematical physics. The method is, essentially, a statistical approach to the study of differential equations, or more generally, of integro-differential equations that occur in various branches of the natural sciences.





Pion-proton phase-shift analysis (Fermi, Metropolis; 1952)
Phase-shift analysis (Bethe, deHoffman, Metropolis; 1954)
Nonlinear coupled oscillators (Fermi, Pasta, Ulam; 1953)
Genetic code (Gamow, Metropolis; 1954)
Equation of state: Importance sampling (Metropolis, Teller; 1953)
Two-dimensional hydrodynamics (Metropolis, von Neumann; 1954)
Universalities of iterative functions (Metropolis, Stein, Stein; 1973)
Nuclear cascades using Monte Carlo (Metropolis, Turkevich; 1954)
Anti-clerical chess (Wells; 1956)
The lucky numbers (Metropolis, Ulam; 1956)

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The First Practical Monte Carlo Manual

- E.D. Cashwell and C.J. Everett, "A Practical Manual on the Monte Carlo Method for Random Walk Problems," LA-2120 (December 18, 1957)
  - Well described 228 page report specific to neutral particle transport
  - Detailed diagrams and flowcharts
    - Neutron collisions – (in)elastic scattering, fission, etc.
    - Photon collisions – Compton scattering, photoelectric, pair production
    - Particle direction after collision – direction cosines
  - Did not deal with thermal neutron collisions nor pseudo-random number generation
  - Appendix includes 20 neutron and photon problems run on the MANIAC I
    - Energy-dependent sources in various geometries and materials
    - Problems for neutrino detection and rocket motors
    - Bob Schrandt did most of the coding

- MCNP Website contains hundreds of archival references on Monte Carlo for particle transport
  - https://laws.lanl.gov/vhosts/mcnp.lanl.gov/references.shtml

Monte Carlo & MCNP History

ENIAC – 1945

30 tons

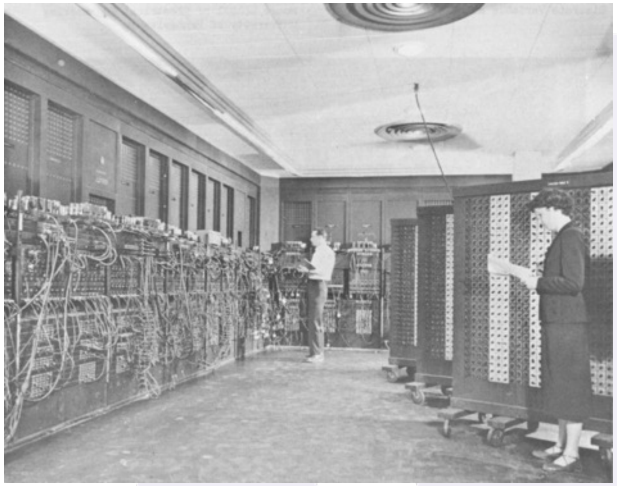
20 ft x 40 ft room

18,000 vacuum tubes

0.1 MHz

20 word memory

patchcords



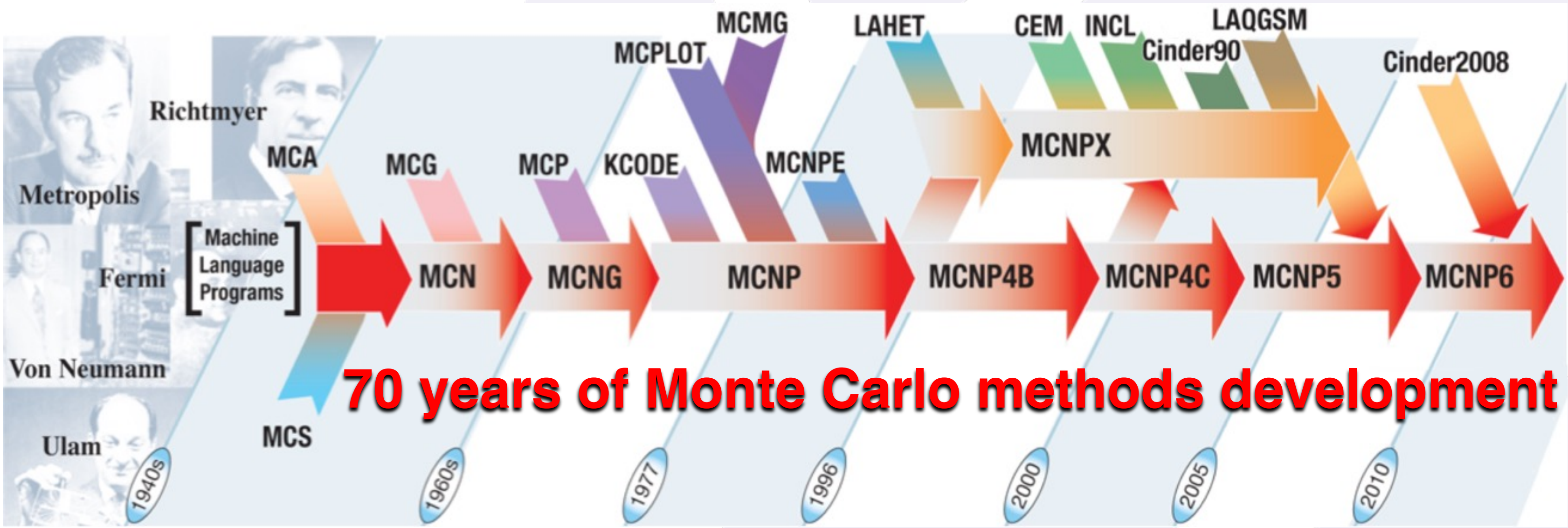
Manhattan Project – 1945...

Discussions on using ENIAC

Ulam suggested using the "method of statistical trials"

Metropolis suggested the name "Monte Carlo"

Von Neumann developed the first computer code



MCS: A General Monte Carlo Neutronics Code (1961)

LAMS-2856
UC-32, MATHEMATICS AND COMPUTERS
TID-4500 (19th Ed.)

LOS ALAMOS SCIENTIFIC LABORATORY
OF THE UNIVERSITY OF CALIFORNIA LOS ALAMOS NEW MEXICO

REPORT WRITTEN: March 1, 1963

REPORT DISTRIBUTED: May 13, 1963

A GENERAL MONTE CARLO NEUTRONICS CODE

by

Robert R. Johnston

Contract W-7405-ENG. 36 with the U. S. Atomic Energy Commission

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1

- R.R. Johnston, "A General Monte Carlo Neutronics Code," the MCS code, Los Alamos Scientific Laboratory report, LAMS-2856, 188 pages (1963).
  - MCS is a general Monte Carlo neutron shielding code for a time-independent geometry.
  - written in the FLOCO coding system (described in LAMS-2339) for the IBM 7090 calculator.
- MCS Code Capabilities
  - It is capable of treating an arbitrary three-dimensional configuration of first- and second-degree surfaces.
  - basic units are cm, shakes, MeV, 10^{24} atoms/cc, and barns
  - particle weight, cell importances $I_0 + I_1E + I_2E^2$, and exponential transform
  - MCA sets up problems to be run by MCS, MCH, and MCR, max of 432 surfaces and 2048 cells
  - nuclear data from cards for different laws for inelastic and elastic scattering
  - cell and surface flux tallies with variances as a function of E and t

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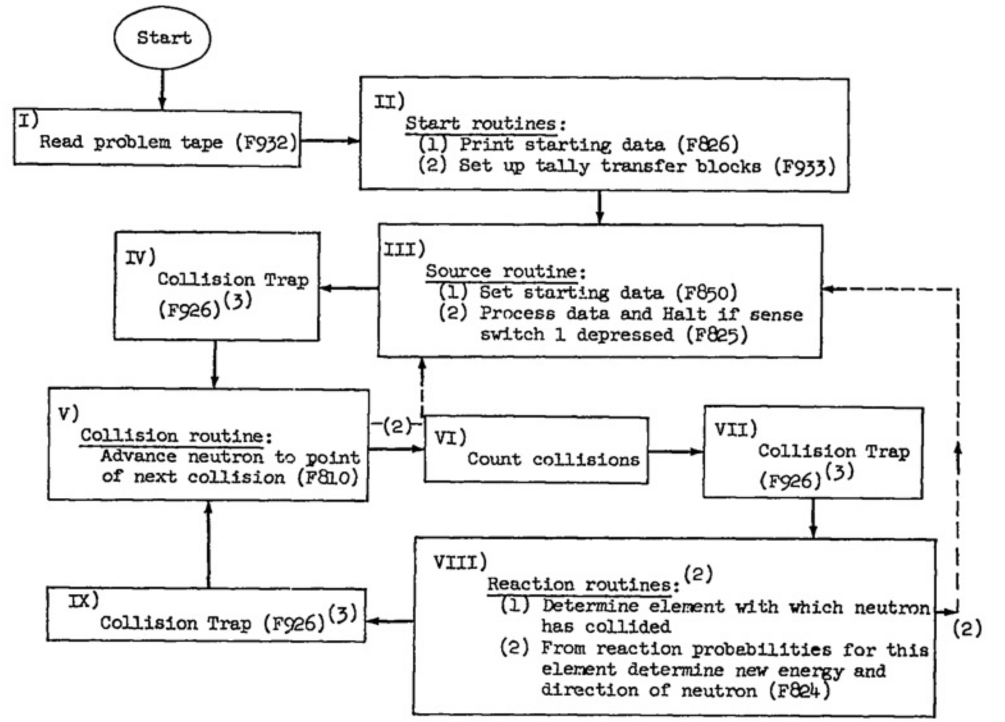
MCS: Flocode – controls overall flow of calculation

IBM 7090 (1958- 1969)

https://www.ibm.com/ibm/history/exhibits/mainframe/mainframe_PP7090.html



Notes:

(1) Two versions of F800 exist, labeled FC1MC1 and FC2MC1. Code FC1MC1 is as diagrammed above; FC2MC1 includes a collision tally routine between blocks V and VI above -- to tally collisions as a function of cell and energy (F833).

(2) When neutron is finished -- due to escape, time or energy cutoff -- the control returns from blocks V or VIII to the source routine to pick another neutron.

(3) Collision traps operative only if sense switch 5 is depressed and sense switch 3 not depressed. (See description of sense switches, Chapter VIIB, and F926.)

Early Monte Carlo Transport Codes at Los Alamos

LA-UR-74-897

Conf-74071

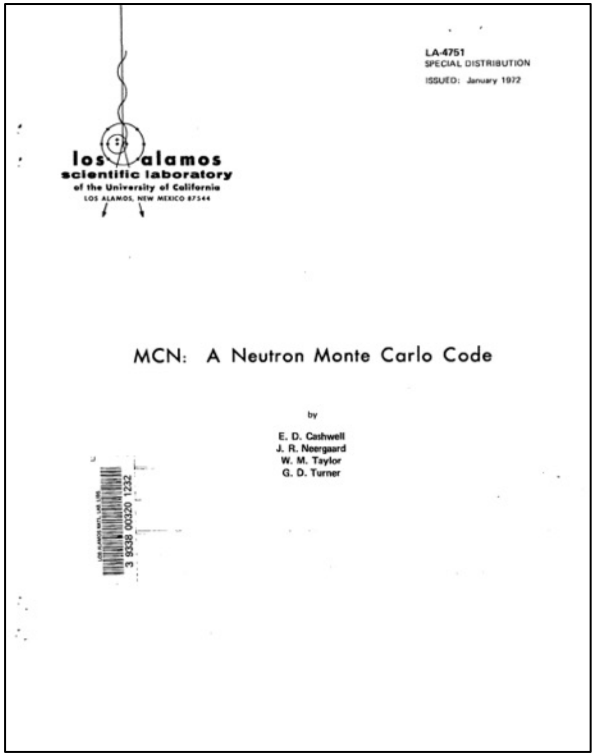
TITLE: MONTE CARLO CODE DEVELOPMENT IN LOS ALAMOS

AUTHOR(S): L. L. Carter, E. D. Cashwell, C. J. Everett, C. A. Forest, R. G. Schrandt, W. M. Taylor, W. L. Thompson, and G. D. Turner

SUBMITTED TO: Presentation at the Monte Carlo Conference at Argonne National Laboratory, Argonne, Illinois, July 1-3, 1974.

- MCN Monte Carlo Neutron
- MCNA Monte Carlo Neutron Adjoint
- MCG Monte Carlo Gamma-ray
- MCP Monte Carlo Photon
- MCNG combined neutron & gamma-ray
- MCK Monte Carlo Criticality
- MCMG multigroup Monte Carlo (based on MCNG)
- MCGE coupled electron-photon
- MCGB gamma-ray with bremsstrahlung

MCN: A Neutron Monte Carlo Code (1965)


LA-4751
SPECIAL DISTRIBUTION
ISSUED: January 1972

los alamos
scientific laboratory
of the University of California
LOS ALAMOS, NEW MEXICO 87544

MCN: A Neutron Monte Carlo Code

by

E. D. Cashwell
J. R. Neergaard
W. M. Taylor
G. D. Turner

Control Data Corporation (CDC)-6600 (1964)

- E.D. Cashwell, J.R. Neergaard, W.M. Taylor, and G.D.Turner, "MCN: A Neutron Monte Carlo Code," Los Alamos Scientific Laboratory report, LA-4751 , 32 pages (1972).
  - The general purpose Monte Carlo neutron code MCN is described in detail to help the user set up and run problems on the CDC-6600 and CDC-7600.
  - The code treats general three-dimensional geometric configurations of materials, and can use point cross-section data in either the Livermore (LLL) or the Aldermaston (AWRE) format.
- MCN Code Capabilities
  - basic units are cm, shakes, MeV, 10^{24} atoms/cc, and barns
  - 5 standard sources are available and a user source subroutine
  - 24 surface types are available to build a geometry
  - an improved random number generator (Lehmer)
  - particle weight, cell importances, I, and exponential transform

MCN: A Neutron Monte Carlo Code (1965)

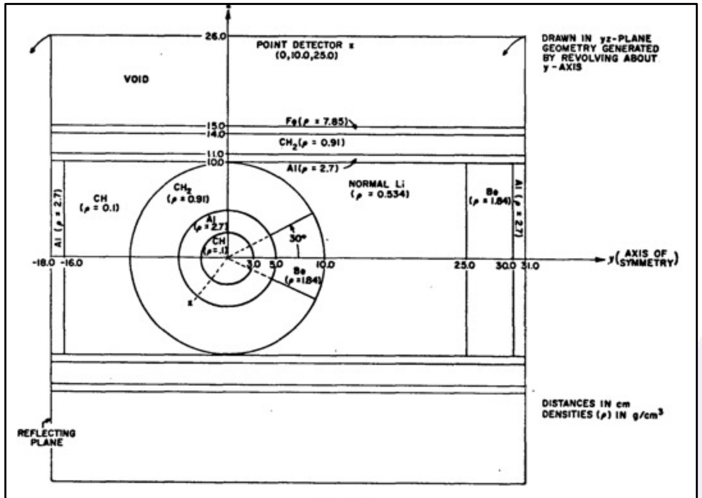


Fig. C-1.
Sample problem for the MCN code.

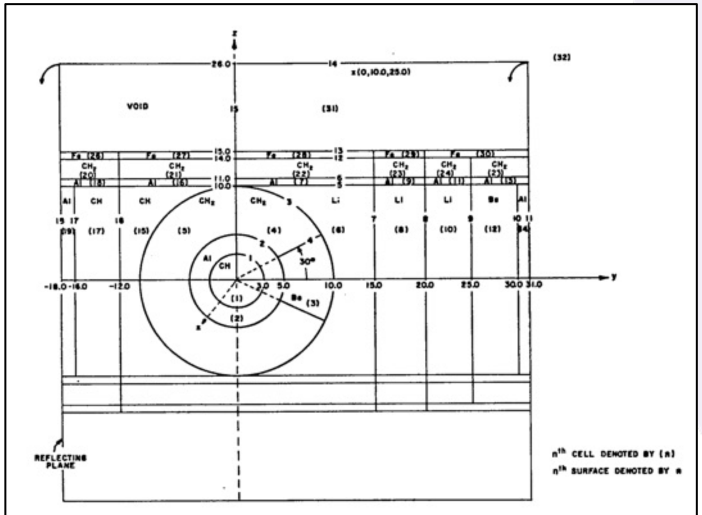


Fig. C-2.
Sample problem zoned for input to MCN.

- MCN Code Capabilities (continued)
  - cross sections from LLL or AWRE
  - free-gas thermal neutron treatment with spatially and time-dependent temperatures
  - numerous input warnings and fatal errors provided
  - input, tally structure, and output similar to MCNP
  - point detector tally is now available

  - free-gas thermal neutron treatment with spatially and time-dependent temperatures
  - numerous input warnings and fatal errors provided
  - input, tally structure, and output similar to MCNP
  - point detector tally is now available

- Testing and Versions
  - verification test sets are routinely used
  - code was periodically updated

MCN: Input and Output

```
                          SAMPLE PROBLEM
 1  45 .00926   -1,2
 2  43 .0603    1,1 -2,4,5,3
 3  44 .123    2,2 -4,4 -3,6
 4  46 .1173    2,2 4,3 15,5 -3,6
 5  46 .1173    2,2 -15,4 -3,15
 6  41 .0463    3,4,3 -5,7 -7,8
 7  43 .0603    5,6 15,16 -6,22 -7,9
 8  41 .0463    -5,9 7,6 -8,10
 9  43 .0603    5,8 7,7 -6,23 -8,11
10  41 .0463    -5,11 8,8 -9,12
11  43 .0603    5,10 8,9 -6,24 -9,13
12  44 .123    -5,13 9,10 -10,14
13  43 .0603    5,12,14 9,11 -6,25 -11,32
14  43 .0603    -5,13 10,12 -11,32
15  45 .00926     3,5 16,17 -5,16
16  43 .0603    5,15 16,18 -6,21 -15,7
17  45 .00926    -5,18 17,19 -16,15
18  43 .0603    5,17,19 18,32 -6,20 -16,16
19  43 .0603    -5,18 18,32 -17,17
20  46 .1173    6,18 18,32 -12,26 -16,21
21  46 .1173    6,16 16,20 -12,27 -15,22
22  46 .1173    6,7 15,21 -12,28 -7,23
23  46 .1173    6,9 7,22 -12,29 -8,24
24  46 .1173    6,11 8,23 -12,30 -9,25
25  46 .1173    6,13 9,24 -12,30 -11,32
26  42 .0847   12,20 18,32 -13,31 -16,27
27  42 .0847   12,21 16,26 -13,31 -15,28
28  42 .0847   12,22 15,27 -13,31 -7,29
29  42 .0847   12,23 7,28 -13,31 -8,30
30  42 .0847   12,24,25 8,29 -13,31 -11,32
31  0     13,26,27,28,29,30 18,32 -14,32 -11,32
32  0     14,31 -18,19,18,20,26,31 11,14,13,25,30,31

 1   SO 3.0
 2   SO 5.0
 3   SO 10.0
```

```
SOURCE NO.   TIME CUTOFF   WEIGHT CUTOFF  RUN TIME      PRINT CYCLE   DUMP CYCLE    DUMP NO.   CUTOFF CYCLE
    7        1.0000E+02    1.0000E-04     1.0000E+01      25000         25000          0            0
                    SAMPLE PROBLEM
SOURCE NO.   TIME CUTOFF   WEIGHT CUTOFF  RUN TIME      PRINT CYCLE   DUMP CYCLE    DUMP NO.   CUTOFF CYCLE
    7        1.0000E+02    1.0003E-04     1.0000E+01      25000         25000          2            0
                                   TIME= 11.568 MINUTES

  NUMBER OF    TOTAL         RANDOM        TOTAL         TOTAL         COLLISIONS    TRACKS        NEUTRONS
  NEUTRONS     NUMBER OF     NUMBERS       WEIGHT        ENERGY        PER NEUTRON   PER NEUTRON   PROCESSED
  STARTED      COLLISIONS    GENERATED     STARTED       STARTED       STARTED       STARTED       PER MINUTE
   18032        835948        8454601      1.8091E+04    4.7063E+04    4.6359E+01    4.5791E+00    1.5588E+03

  TOTAL        LOSS TO       LOSS TO       LOSS TO       LOSS          LOSS          TOTAL
  TRACKS       ENERGY        TIME          WEIGHT        TO            TO            TRACKS
  STARTED      CUTOFF        CUTOFF        CUTOFF        ESCAPE        SPLITTING     LOST
   82570             0         30930           146          9707        41787         82570

  WEIGHT       LOSS TO       LOSS TO       LOSS TO       LOSS          LOSS          WEIGHT
  STARTED      ENERGY        TIME          WEIGHT        TO            TO            LOST
  PER NEUTRON  CUTOFF        CUTOFF        CUTOFF        ESCAPE        CAPTURE       PER NEUTRON
  1.0033E+00   0.            7.3959E-01    3.8864E-07    2.3581E-01    3.6986E-02    1.0124E+00

  ENERGY       LOSS TO       LOSS TO       LOSS TO       LOSS          LOSS
  STARTED      ENERGY        TIME          WEIGHT        TO            TO
  PER NEUTRON  CUTOFF        CUTOFF        CUTOFF        ESCAPE        CAPTURE
  2.6100E+00   0.            7.8911E-05    5.2616E-11    7.2201E-01    6.1252E-02
```

```
SOURCE NO.   TIME CUTOFF   WEIGHT CUTOFF  RUN TIME     PRINT CYCLE   DUMP CYCLE     DUMP NO.   CUTOFF CYCLE
    7        1.0000E+02    1.0000E-04     2.0000E+00     25000         25000           0            0

                        SAMPLE PHOBLEM
```

NPS	X	Y	Z	IA	JA	U	V	W	TME	WT	DEL	ERG
1	2.2875E+00	3.7276E-01	-1.9899E-01	1	1	4.6154E-01	8.1828E-01	3.4264E-01	0.	6.6667E-01	0.	4.1427E-01
2	-3.4934E-01	2.2248E+00	-1.3003E+00	1	1	1.2991E-01	8.5080E-01	5.0917E-01	0.	6.6667E-01	0.	3.9221E-03
3	-1.5399E+00	8.7280E-01	-1.7711E+00	1	1	9.9863E-01	4.2836E-03	5.2221E-02	0.	6.6667E-01	0.	4.6492E-01
4	-5.5782E-01	-5.5362E-01	-1.2246E+00	1	1	-2.4954E-01	-9.6408E-01	-9.1046E-02	0.	2.0000E+00	0.	3.2640E-01
5	2.0899E+00	-1.3313E+00	1.1058E+00	1	1	-2.8440E-01	-4.0338E-01	-8.6971E-01	0.	2.0000E+00	0.	8.6086E-01
6	-4.8075E-01	2.4031E+00	4.9533E-01	1	1	-1.4351E-01	8.9441E-01	4.2359E-01	0.	6.6667E-01	0.	3.6066E-02
7	-2.8698E-02	-2.8978E+00	3.2509E-01	1	1	4.3334E-01	4.2601E-01	7.9418E-01	0.	6.6667E-01	0.	4.4488E-01
8	-2.3599E+00	8.1552E-01	-2.0618E-01	1	1	-5.1389E-01	2.6413E-01	-8.1618E-01	0.	6.6667E-01	0.	5.0756E-01
9	-7.8826E-01	7.2759E-01	-1.3609E+00	1	1	-4.3904E-01	-5.5958E-01	-7.0293E-01	0.	2.0000E+00	0.	6.2863E-01
10	-5.6906E-01	-1.7042E+00	1.0086E+00	1	1	9.6330E-01	-8.6596E-02	-2.5408E-01	0.	2.0000E+00	0.	8.6029E+00
11	1.4449E+00	-8.5310E-01	1.9067E+00	1	1	-6.9409E-01	7.0580E-01	1.1456E-01	0.	6.6667E-01	0.	4.3666E-01
12	3.8711E-01	-2.4355E+00	4.5961E-01	1	1	-1.9277E-01	1.0331E-01	-9.7577E-01	0.	6.6667E-01	0.	4.6801E-01

MCNP Version 1 – 06/21/1977

MCNP: MCNG + MCP

LA-7396-M
Manual

Special Distribution
Issued: July 1978

MCNP—A General Monte Carlo Code for

Neutron and Photon Transport

LASL Group TD-6

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

LASL Group TD-6

ABSTRACT

The general purpose Monte Carlo code MCNP can be used for neutron, photon, or coupled neutron-photon transport. 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. 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.

Standard optional variance reduction schemes include geometry splitting and Russian roulette, the exponential transformation, energy splitting, forced collisions in designated cells, flux estimates at point detectors, track-length estimators, and source biasing.

MCNP Version 2 - 09/26/1979

LA-7396-M, Revised

Manual

2

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

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

MCNP is Released to RSICC

MCNP Version	Release Month/Year	Some Significant New Features (For a more detailed description, see each version's release notes).
MCNP3	1983	First release through RSICC. Written in Fortran 77
MCNP3A	1986	
MCNP3B	1988	Plotting graphics, generalized source, surface sources, repeated structures/lattice geometries
MCNP4	1990	Parallel multitasking, electron transport
MCNP4A	10/1993	Enhanced statistical analysis, new photon libraries, ENDF-6, color X-Windows graphics, dynamic memory allocation
MCNP4B	4/1997	Operator perturbations, enhanced photon physics, PVM load balancing, cross-section plotting, 64-bit executables, lattice universe mapping, enhanced lifetimes
MCNPX 2.1.5	11/1999	First public release of MCNPX, based on MCNP4B with CEM INC, HTAPE3X, mesh and radiography tallies, and an improved collisional energy loss model.

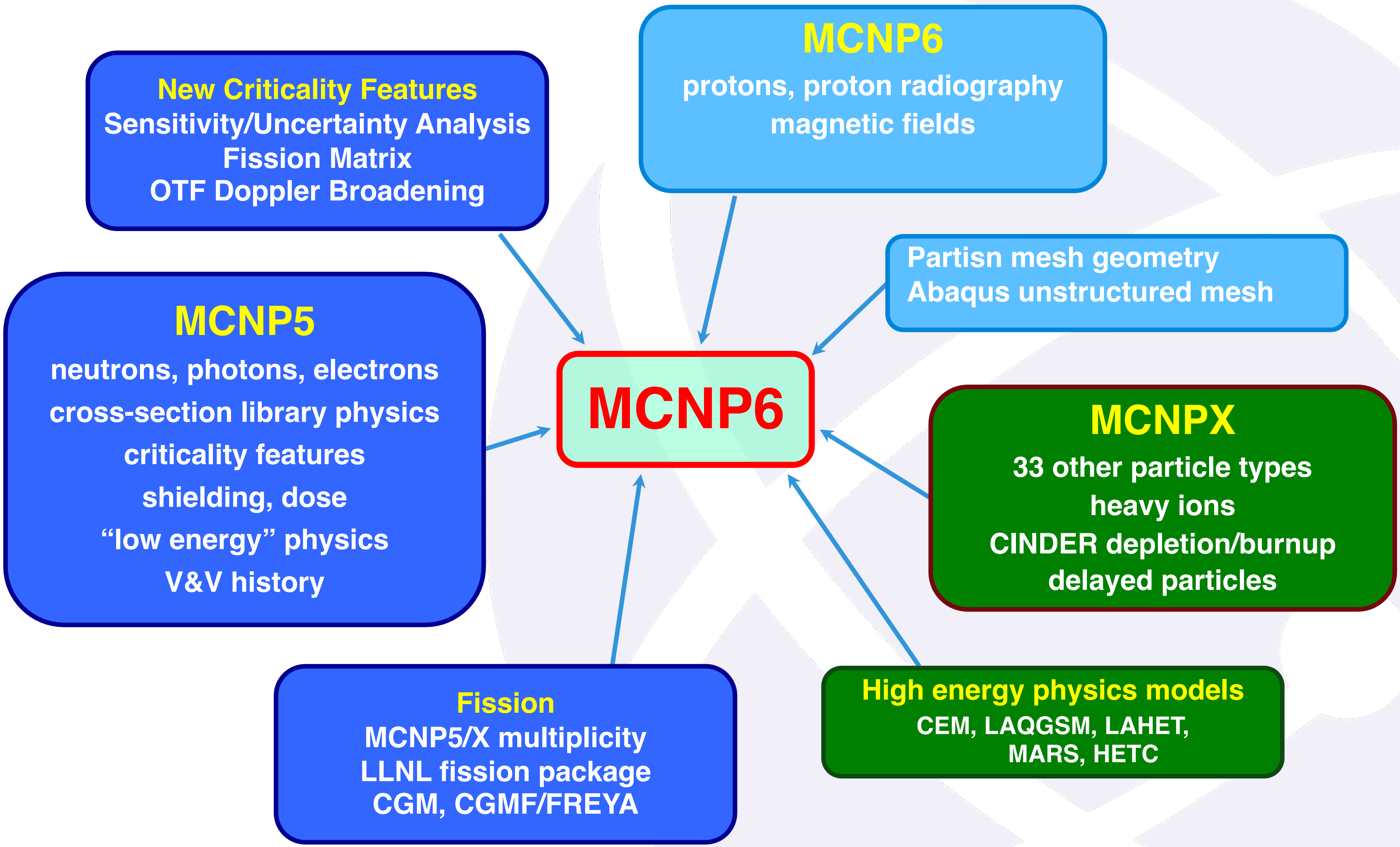
T. Goorley, et al. "Initial MCNP6 Release Overview," LA-UR-13-22934 (2013)

- **Key Value: MCNP provides a predictive capability that can replace expensive or impossible-to-perform experiments**
- **Used to design large-scale measurements providing significant time/cost savings**
- **MCNP represents a synergistic capabilities developed at LANL**
  - Evaluated nuclear data (ENDF) and data processing code NJOY
  - MCNP could not exist without this!
- **International user community's high confidence in MCNP's predictive capabilities are based upon its performance with verification and validation test suites.**

2000 – 2011: MCNP undergoes exponential growth

MCNP4C	4/2000	Unresolved resonance treatments, macrobodies, superimposed importance mesh, perturbation, electron transport, plotter and tally enhancements
MCNP4C2	1/2001	Photonuclear physics, interactive plotting, plot superimposed weight-window mesh, weight-window improvements
MCNPX 2.3.0	4/ 2002	LAHET 2.8 and some 3.0 extensions.
MCNPX 2.4.0	8/2002	Update to MCNP4C, build system for Windows OS, support for Fortran 90.
MCNP5 1.14	11/2002	Fortran 90, photonuclear collisions, geometry superimposed mesh tallies, time splitting, shared memory threading with OpenMP. Mac OSX support
MCNP5 1.20	10/2003	Increased number of detectors to 100 and number of tallies to 1000. Mostly a code defect fix release.
MCNP5 1.30	8/2004	Explicit 8-byte integers for nps > 2.1 billion, Lattice and fmesh tally enchantments. Support for MPI on Mac OSX.
MCNPX 2.5.0	4/2005	34 particle types, four light ions, mix and match nuclear data tables and model physics, CEM2k, INCL4/ABLA physics models, fission multiplicity, spontaneous fission sources, pulse height tallies with variance reduction, pulse height light tally, coincident capture tallies, variance reduction with model
MCNP5 1.40	11/2005	Lethargy plots, logarithmic data interpolation, neutron multiplicity distributions, stochastic geometry, source entropy, mesh tally plots, new electron energy loss straggling
MCNPX 2.6.0	4/ 2008	Depletion/Burnup, heavy ion transport, LAQGSM physics, CEM03 physics, delayed gamma emission, energy-time weight windows, charged ions from neutron capture, spherical mesh weight windows, spontaneous photons
MCNP5 1.51	1/2009	Photon Doppler broadening, variance reduction with pulse height tallies, annihilation gamma tracking, Doppler broadening in makxsf, large lattice enhancements
MCNP5 1.60	8/2010	Adjoint weighted tallies for point kinetics parameters, mesh tallies for isotopic reaction rates, up to 100 million cells & surfaces, up to 10 thousand tallies
MCNPX 2.7.0	4/2011	Tally Tagging, embedded sources, cyclic time bins, focused beam sources, PTRAC coincidence, LLNL fission multiplicity, Receiver-operator characterization (ROC) tally, NRF data in ACE libraries, triple & quadruple coincidence, LAQGSM 3.03 and CEM 3.03 physics.

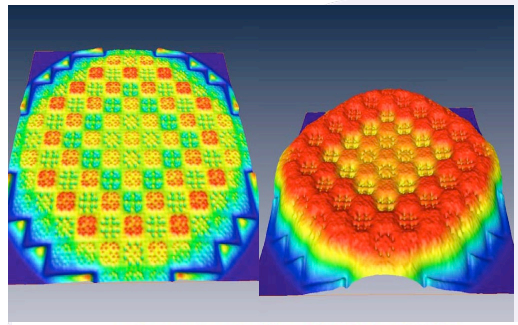
From MCNP5 & MCNPX to MCNP6



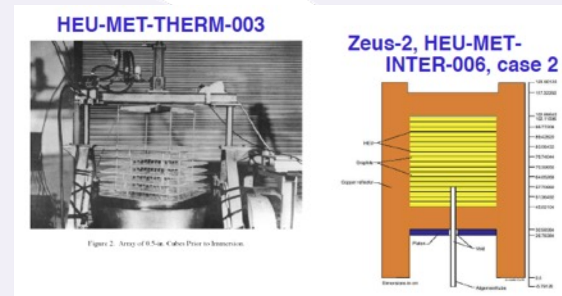
MCNP Capabilities

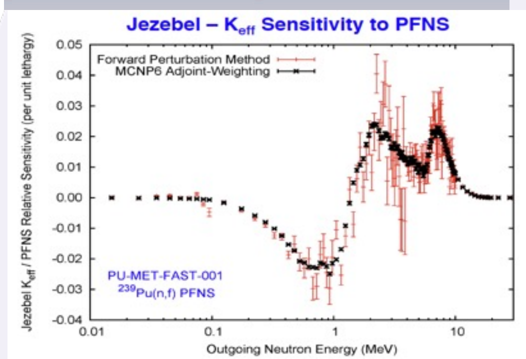
- Physics:
  - Continuous energy particle transport
  - Neutron, photon, electron, and many more particle types
- Algorithms:
  - k-eigenvalue calculations
  - Fixed source calculations
- Recently Implemented Features:
  - Unstructured mesh transport
  - Electric and magnetic field transport
  - High-energy physics models
  - 33 additional particle types
  - Reactor fuel depletion and burnup
  - Radiation source and detection capabilities
  - Sensitivity and uncertainty analysis for nuclear criticality safety
- Extensive Variance Reduction
  - Weight Windows
  - DXTRAN

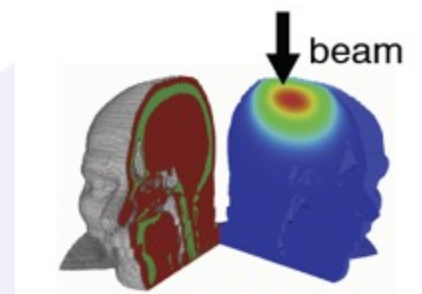
Whole-core Thermal & Total Flux

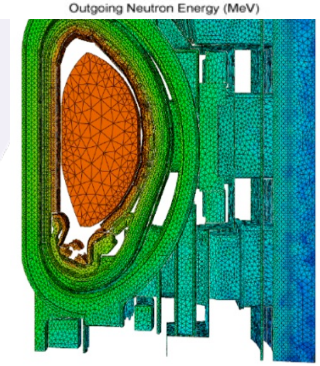


Experimental Benchmarks with Critical Assemblies

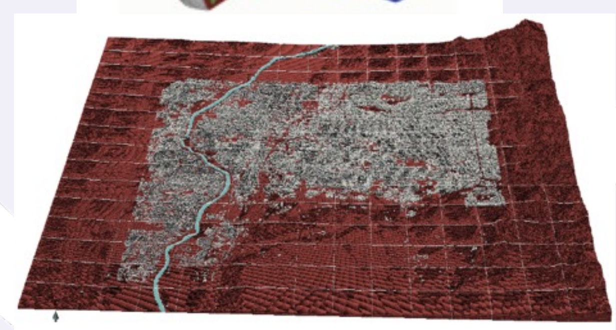








ITER Neutron Flux Calculations

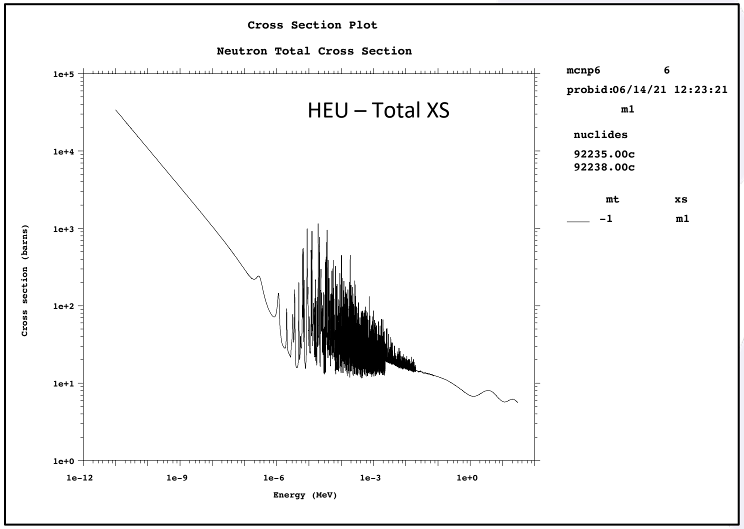


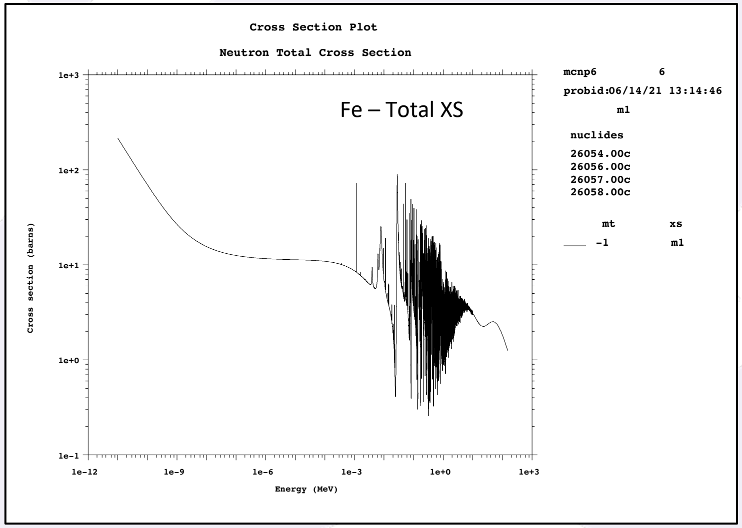
City model used to study nuclear weapon effects

Los Alamos NATIONAL LABORATORY

History of Monte Carlo Neutron Data Libraries (1)

- Code development linked with data libraries
  - MCS: treatment of nuclear data has been designed to represent accurately the experimental data, frequently at the expense of computing time
    - particle reaction sampling laws based on data evaluations
  - MCN: LLL and AWRE data maintained on the MANIAC
  - MCN: cross sections from ENDF/B-IV files became available 1975
    - 262-group discrete reaction cross sections and elemental evaluations often used
  - Current cross section libraries are based on ENDF/B-VIII containing 556 isotopes
  - MCNP makes no gross assumptions regarding data





History of Monte Carlo Neutron Data Libraries (2)

- History of Some Major Data Improvements
  - 1968: data libraries became available
  - 1975: first data library from ENDF became available
  - 1980: first $S(\alpha,\beta)$ capability for thermal neutrons
  - 1980: continual improvements in neutron-inducted photon production
  - 1998: unresolved resonance treatment for neutrons
  - 2000: delayed neutrons
  - 2003: photonuclear data from ENDF/B-VI
  - 2015: correlated neutron multiplicity models

- ACE nuclear data libraries from LANL
  - https://nucleardata.lanl.gov/ACE/index.html

Summary

- Modern Monte Carlo method developed at Los Alamos
  - technique of statistical sampling named Monte Carlo
  - successfully applied to problems on the ENIAC and MANIAC
  - paved the way for Monte Carlo calculations on computers
- Los Alamos Monte Carlo Theory, Physics, and Codes Have Evolved
  - particles, geometry, physics, variance reduction, computer architectures
  - cross-section libraries continually improve
  - more detailed information about calculation for user
  - MCNP was made available to users worldwide starting in 1983
- MCNP is the Monte Carlo particle transport code supported by LANL
  - rigorous verification and validation testing
  - new versions released every one to two years
  - used extensively at LANL for a variety of applications
  - used worldwide for broader range of application
  - MCNP website: https://mcnp.lanl.gov/

User Feedback and Suggestions Are Crucial to the Continued Success of MCNP® and its Data Libraries