NOTE: Use these buttons to print or save the form. DO NOT use the browser tool bar. SAVE PRINT

LA-UR-04-3399

Approved for public release; distribution is unlimited.

Title: What Can MCNP Do?

Author(s): Tim Goorley, Forrest Brown, Jeff Bull, Russ Mosteller, Elizabeth Selcow, Avneet Sood
X-5
Los Alamos National Laboratory

Submitted to: Introduction to MCNP Class
June 1-4,2004
Los Alamos, NM

SAVE PRINT CLEAR FORM

Los Alamos NATIONAL LABORATORY

What Can MCNP Do?

Abstract

MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. Specific areas of application include, but are not limited to, radiation protection and dosimetry, radiation shielding, radiography, medical physics, nuclear criticality safety, Detector Design and analysis, nuclear oil well logging, Accelerator target design, Fission and fusion reactor design, decontamination and decommissioning. The following slides give examples of situations where MCNP has been used for some of these applications.

What Can MCNP Do?

- Monte Carlo coupled particle transport (n,γ,e) [no decay]

- Calculate
  - Flux, Current, Energy or Charge Deposition, Heating, Reaction Rates, Response Functions, Radiographs, Mesh Tallies (E, θ, t bins)
  - k_{eff}, prompt neutron lifetime, fission distributions, $\overline{\eta}$, $\overline{\nu}$, $\overline{E}$ of neutrons causing fission, neutron balance per cell and nuclide.

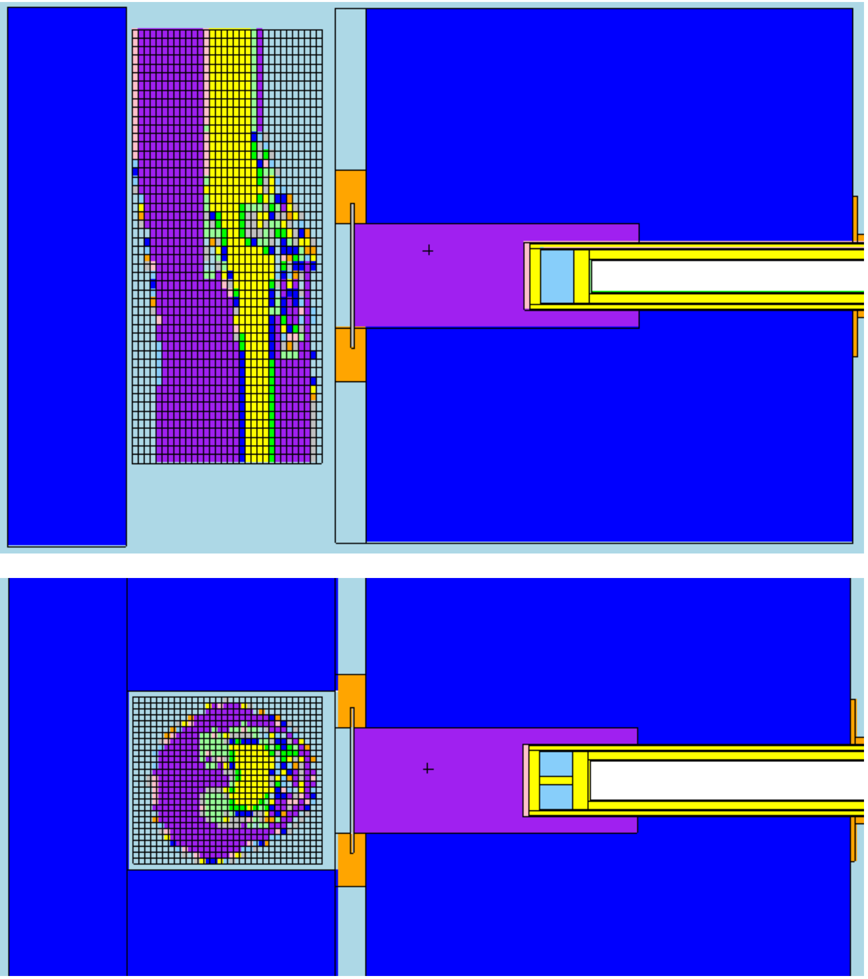
- With help of
  - Geometry construction techniques: macrobodies, trcl, u, lat
  - Surface sources for large & repetitive problems
  - Geometry, cross section, tally plotting (More with Visual Editor)
  - Many variance reduction techniques
  - Parallel calculation ability

Examples

- Following slides show examples of MCNP being used in many applications.
  - Medical Physics
  - Criticality / Shielding
  - Nuclear Engineering Design and Development

Calculate Dose - Investigate Therapies

- Patient-CT based knee model and end of accelerator in geometry.
- Need other code to determine neutron production in accelerator target.
- Calculate dose throughout knee.
- Study impact of moderating/ shielding materials & B^{10} conc. in knee.



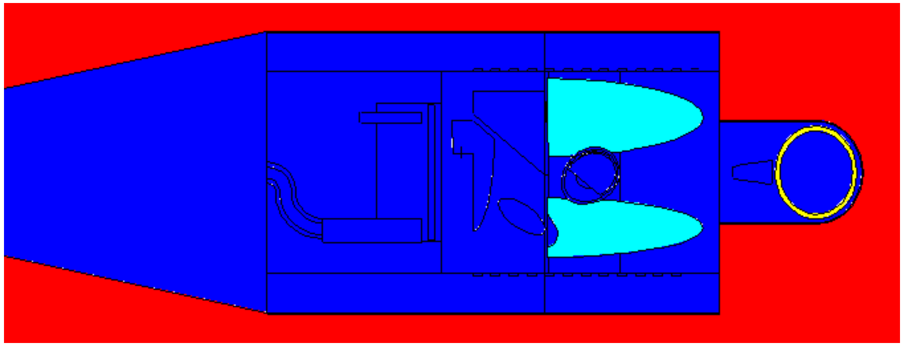
Pictures from mcnp plotter

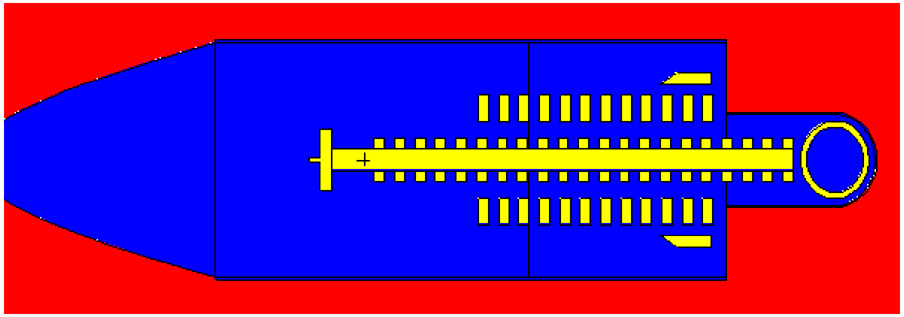
J. R. Albritton, ''Analysis of the SERA treatment planning system and its use in boron neutron capture synovectomy,'' M. S. thesis, Massachusetts Institute of Technology, 2001.

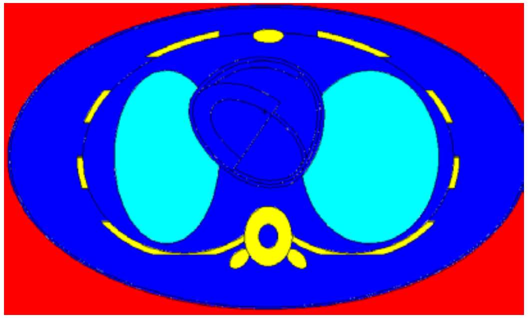
Gierga DP, Yanch JC, Shefer RE, "An investigation of the feasibility of gadolinium for neutron capture synovectomy", Med Phys. 2000 Jul;27(7):1685-92.

Calculate Dose - Investigate Therapies

- Use of MIRD-like whole body model for accelerator based X-ray or neutron therapies.
- Organ specific doses.
- Vary incident X-ray spectra, shielding.







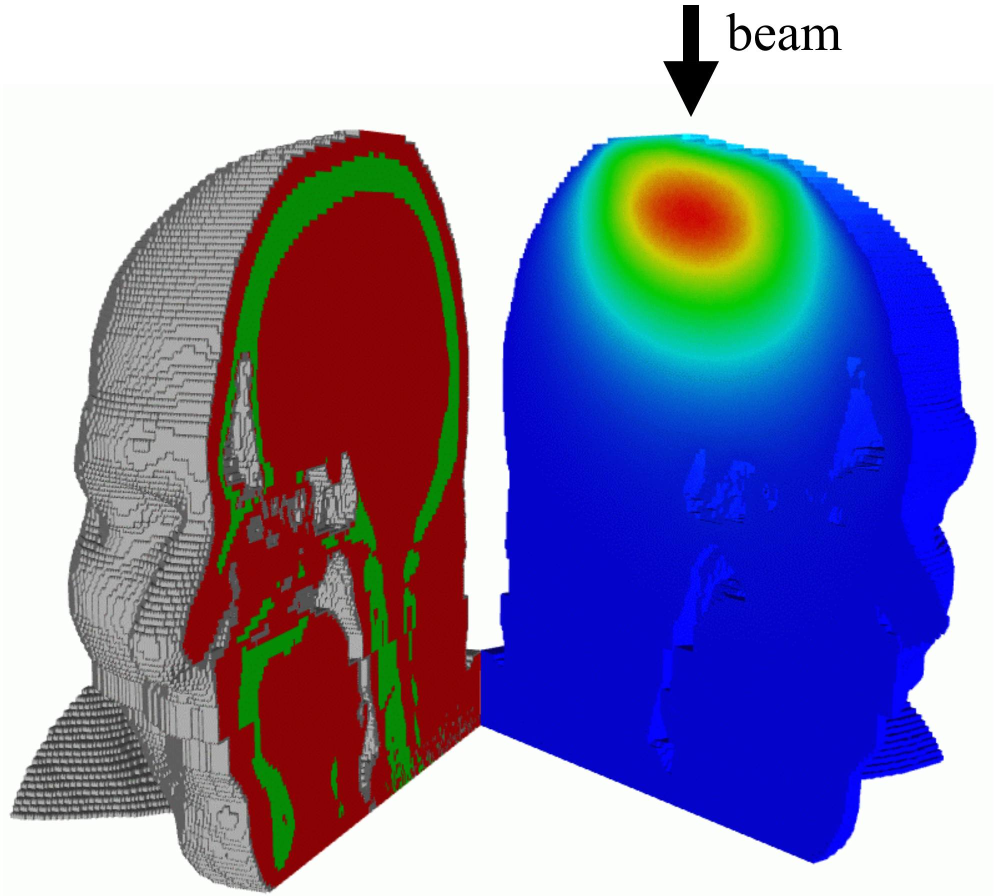
Pictures from mcnp plotter

Lambeth, Melissa. “Development of a computerized anthropomorphic phantom for determination of organ doses from diagnostic radiology.” Thesis, B.S., Massachusetts Institute of Technology, Dept. of Nuclear Engineering, 1997.

Gierga DP, Yanch JC, Shefer RE, “An investigation of the feasibility of gadolinium for neutron capture synovectomy”, Med Phys. 2000 Jul;27(7):1685-92.

Calculate Dose – Treatment Planning

- Use Patient-based CT geometry.
- Calculate dose throughout head, tumor.
- Change beam direction and look at differences in dose distributions.



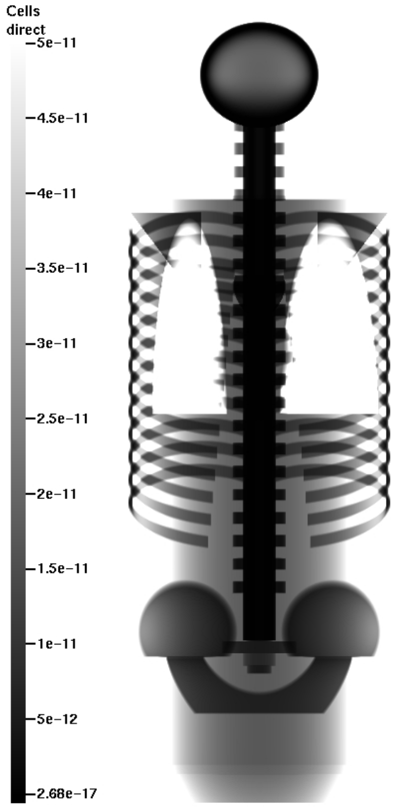
Pictures not from mcnp, but materials (left) and doses(right) from mcnp calculation.

- Larry Cox - Job Queuing & Execution
- Gregg McKinney - Input & Code Modifications
- Robby Russell - Graphics
- Tim Goorley - Input Generation
- ASCI Blue Mountain

Calculate Dose – Simulate Radiograph

- Neutron and photon radiography uses a grid of point detectors (pixels).
- Each source and collision event contributes to all pixels.
- Simulate X-ray, neutron radiographs. Investigate role of scatter in image.

Picture from Sabrina



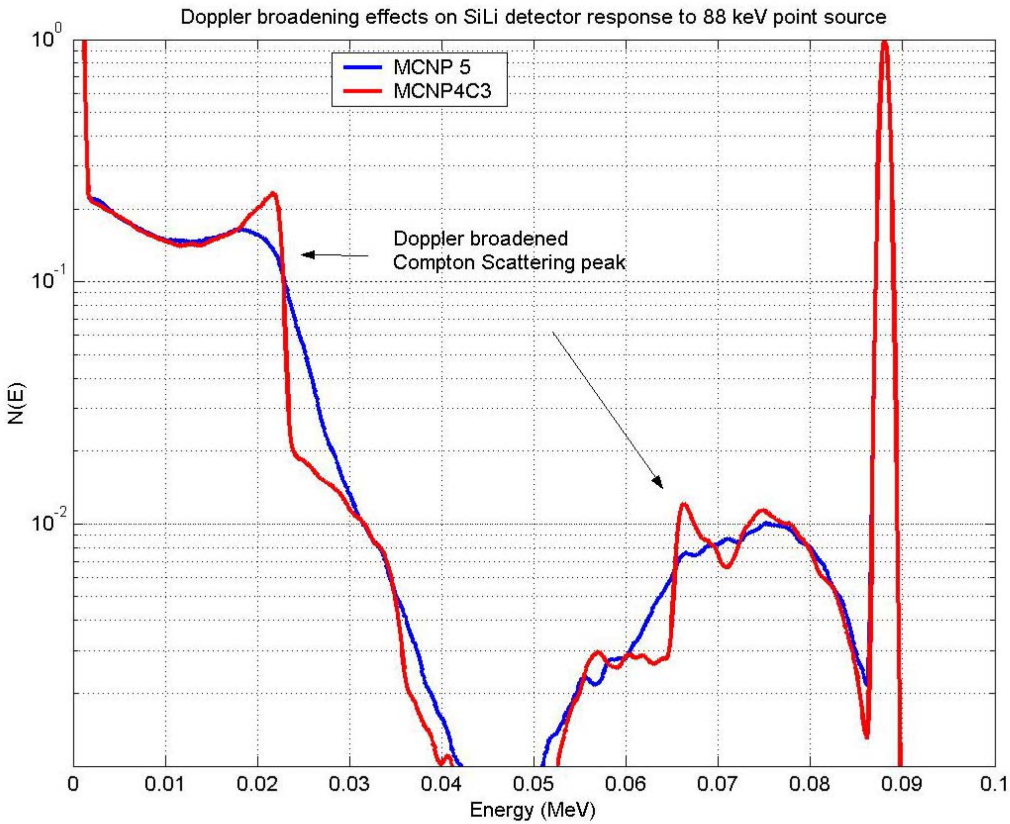
Picture generated with results from MCNP calculation.

Simulated Radiograph

1 M pixels

Calculate Detector Response

- Calculate SiLi detector response to 88 keV point source.
- Compare to experiment, look at scatter from various portions of geometry.
- Other detector response problems in QUADOS comparison. (prob #7)

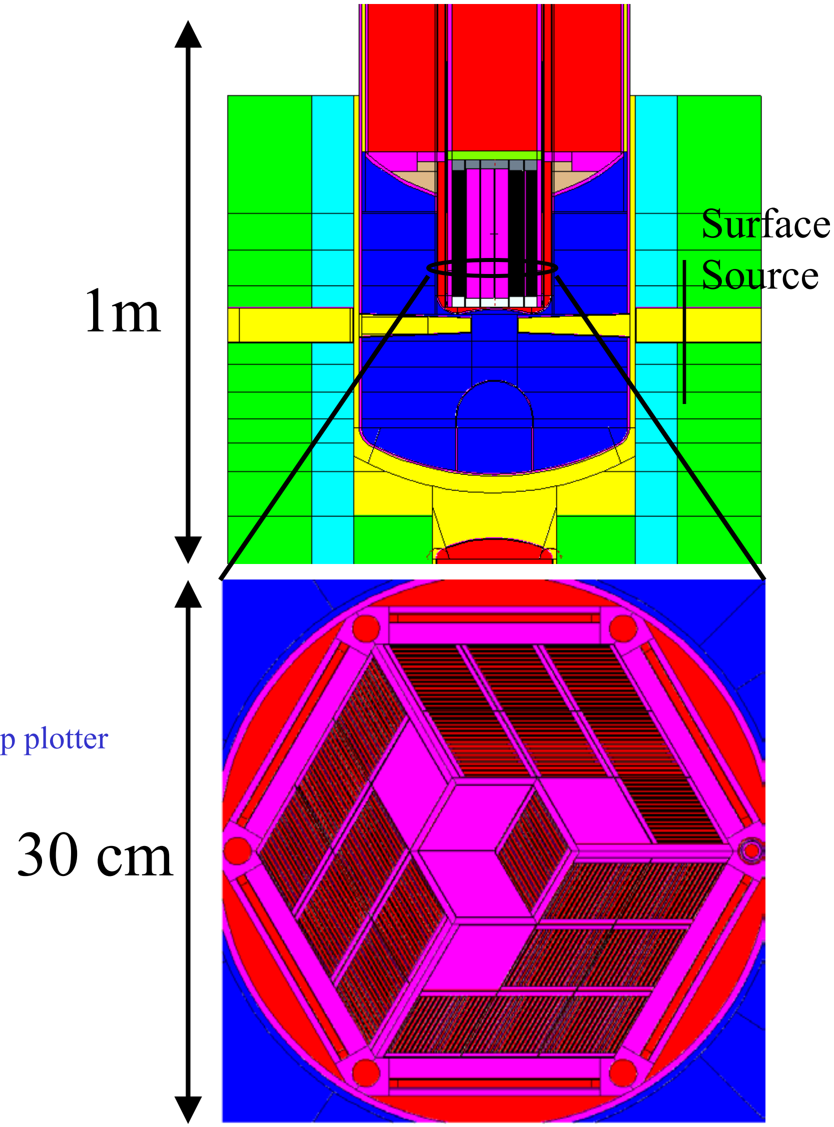


Sood, R. Gardner, "A new Monte Carlo assisted approach to detector response functions", Nuclear Instruments and Methods in Physics Research B, **213** (2004) 100-104.

http://www.nea.fr/download/quados/quados.html

Criticality & Surface Source

- Model research reactor core.
- Calculate surface source at beam port.
- Use surface source for further downstream calculations, like beam port design.
- Calculate different K_{eff} from different control rod insertions.



Picture from mcnp plotter

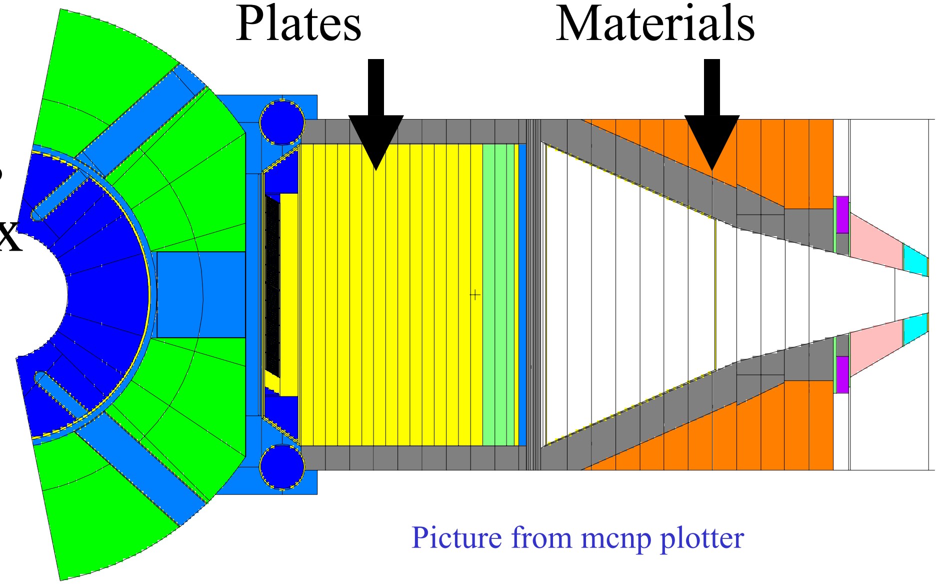
Redmond, E.L., II; Yanch, J.C.; Harling, O.K. “Monte Carlo simulation of the Massachusetts Institute of Technology Research Reactor.” Nuclear Technology; April 1994; vol.106, no.1, p.1-14

Criticality and Flux

- Development of MIT Reactor Fission Converter Beam.
- Change geometry & materials to find optimal epithermal flux
- Intent: Lower fast n and gamma dose but increase epithermal flux at patient position.
- Calculate K_{eff} of U plates.

Used U Fuel Plates

Filter/Moderator Materials



Picture from mcnp plotter

Reactor Core

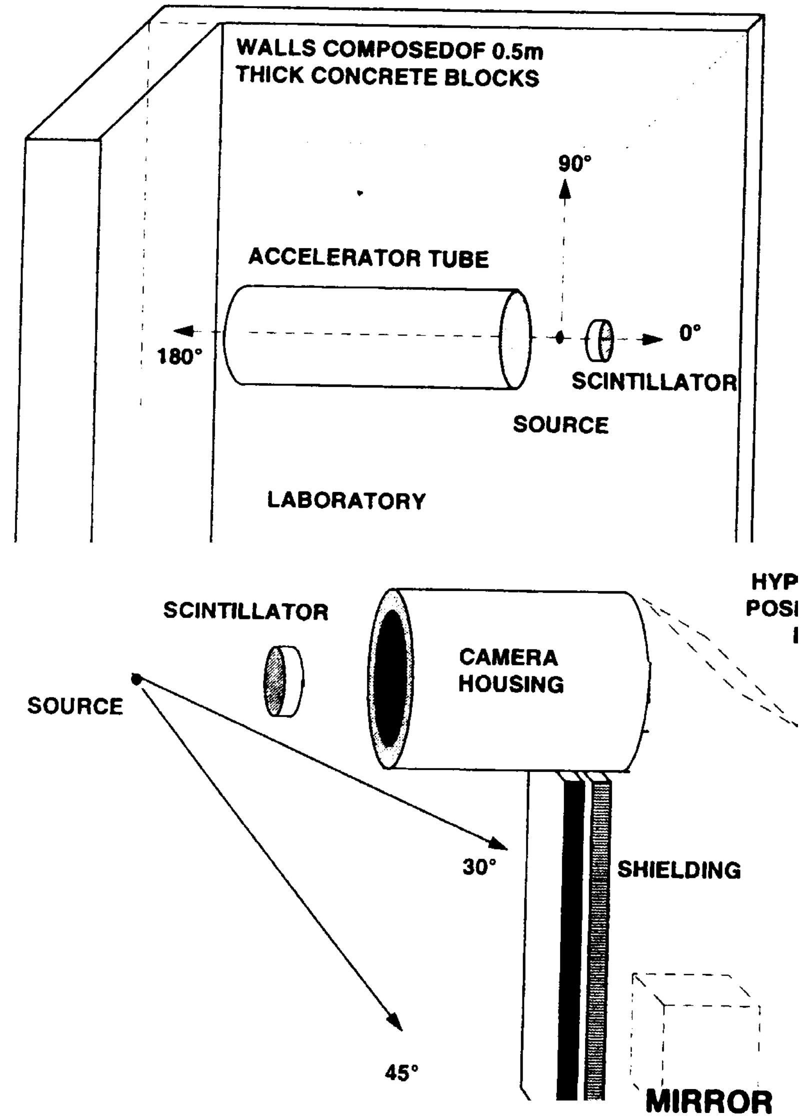
W.S. Kiger III, S. Sakamoto, and O.K. Harling, "Neutronic design of a fission converter-based epithermal neutron beam for neutron capture therapy," *Nuclear Science and Engineering*, **131**, 1-22 (1999).

K.J. Riley, "Construction and Characterization of a Fission Converter Based Epithermal Neutron Beam for NCT," Ph.D. Thesis, Massachusetts Institute of Technology (2001).

O.K. Harling, K.J. Riley, T.H. Newton, B.A. Wilson, J.A. Bernard, L.-W. Hu, E.J. Fonteneau, P.T. Menadier, S.J. Ali, B. Sutharshan, G.E. Kohse, Y. Ostrovsky, P.H. Stahle, P.J. Binns, W.S. Kiger III, and P.M. Busse, "The Fission Converter Based Epithermal Neutron Irradiation Facility at the MIT Reactor," *Nuclear Science and Engineering*, **140**, 223-240 (2002).

Calculate Flux

- Schonland Research Center wanted to design a fast n radiography facility
- Determine how scattered n's affects on image quality.
- Used MCNP4A to model electronic shielding, scintillator, camera casing and irradiation room



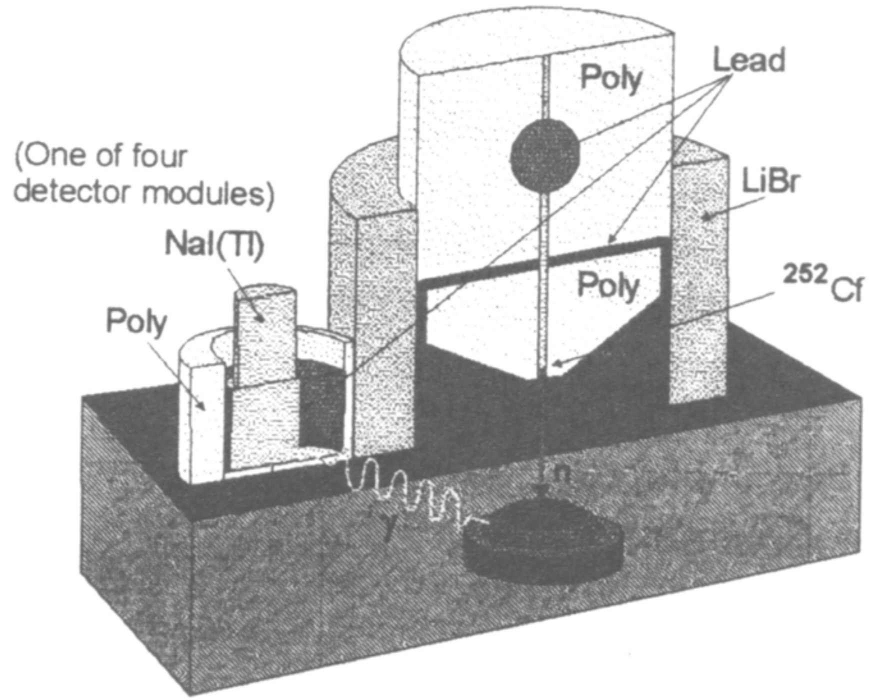
R.M. Ambrosi, J.I.W. Watterson, B.R.K. Kala "A Monte Carlo study of the effect of neutron scattering in a fast neutron radiography facility" NIMB **139** (1998) 286-292.

Calculate Flux & Dose

CDND designed a landmine detector system.

Needed to shield personnel and detector from 100 MBq ^{252}Cf source.

Used MCNP4A to vary shielding materials and dimensions.



T. Cousins, T.A. Jones, et. Al. “The development of a thermal neutron activation (TNA) system as a confirmatory non-metallic land mine detector” J. Rad. Nucl. Chem. **235** (1998) 53-58.

Calculate Dose – Health Physics

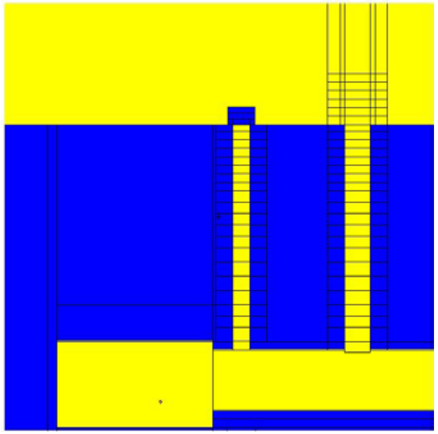
- Proton Storage Ring at LANSCE accelerator
- Investigate dose rates at certain locations.

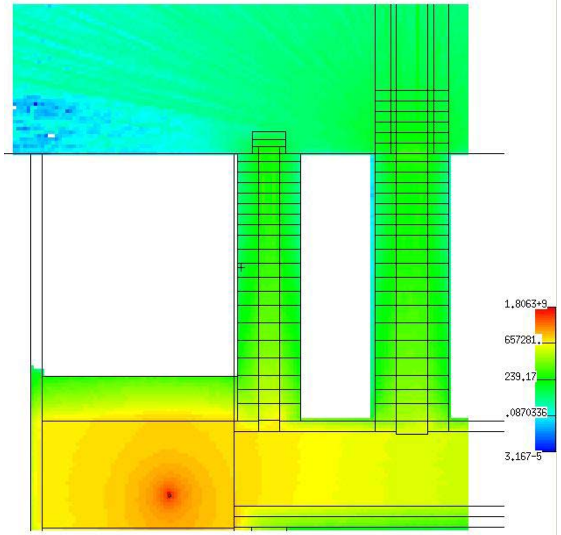
Geometry

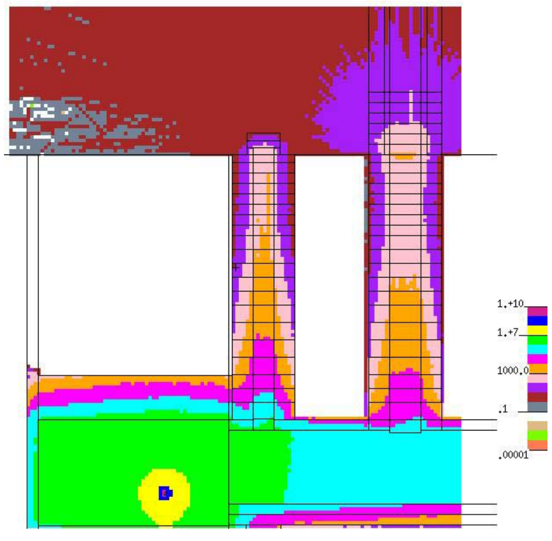
Blue = concrete

Yellow = air

Picture from mcnp plotter



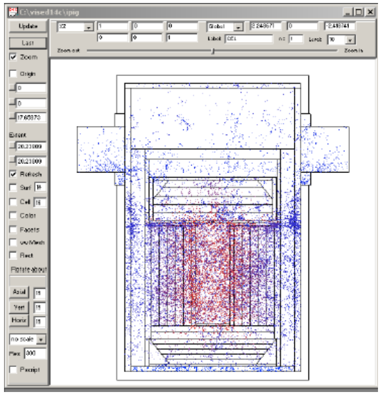


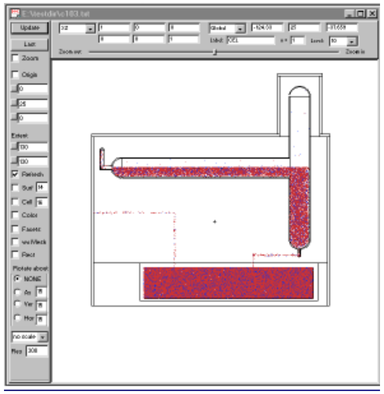


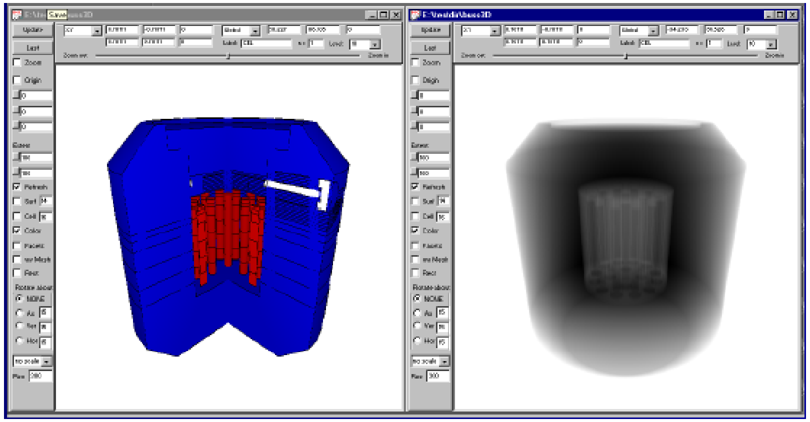
Pictures from beta version of mcnp6 mesh tally plotter

Visual Editor

- Plot
  - Tracks
  - Source points
  - 3D Geometry
- 2-D CAD to MCNP input







Picture from mcnp plotter

VisEd distributed on Windows MCNP5 CDROM.
See http:// mcnpvised.com
VisEd Training Classes offered frequently by Randy Schwarz.