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Title: Using MCNP for Medical Physics Applications

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Using MCNP for Medical Physics Applications

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ANS Computational Medical Physics Working Group-II
http://cmpwg.ans.org/

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. MCNP5 has a wide range of abilities which make it useful for medical physics calculations. These abilities span its geometry representation, physics models, and source, tally and variance reduction capabilities. This workshop will demonstrate how MCNP5 can be used to calculate dose, simulate a radiograph, or even use CT data to create a voxel model of a human or phantom. A general review of MCNP5 source and tally capabilities, as well as new and future capabilities will also be included.

Schedule: x pm – x+2 pm

1. **What Can MCNP Do?** — **15 min**
2. **Overview of new MCNP5 features** — **30 min**
3. **Geometries and Modeling** — **30 min**
4. **Break** — **10 min**
5. **Medical Physics Sources** — **20 min**
6. **Medical Physics Tallies** — **15 min**
7. **MCNP5 Release – End of Oct** — **5 min**
8. **MCNP 6 / MCNPX Merger** — **5 min**
9. **Next Generation of Capabilities?** — **5 min**
10. **Additional References**

General Points

- **In this lecture, I will discuss:**
  - Specific Features
  - Input file commands for these specific features

- **Whole input decks can be found on the workshop CD:**
  - In the Medical Physics Geometry Database
    - Whole Body phantoms (both analytical & voxel)
    - CT image based phantoms for organs, portions
  - Medical Physics Primer
    - Sources
    - Tallies (Dose & Radiography)

What Can MCNP Do?

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, η, $\bar{\nu}$, $\bar{E}$ 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

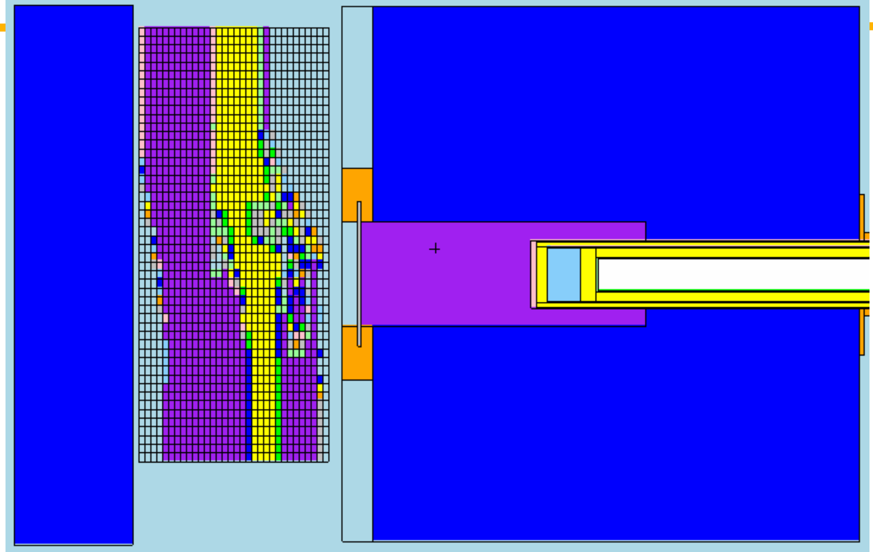
Examples

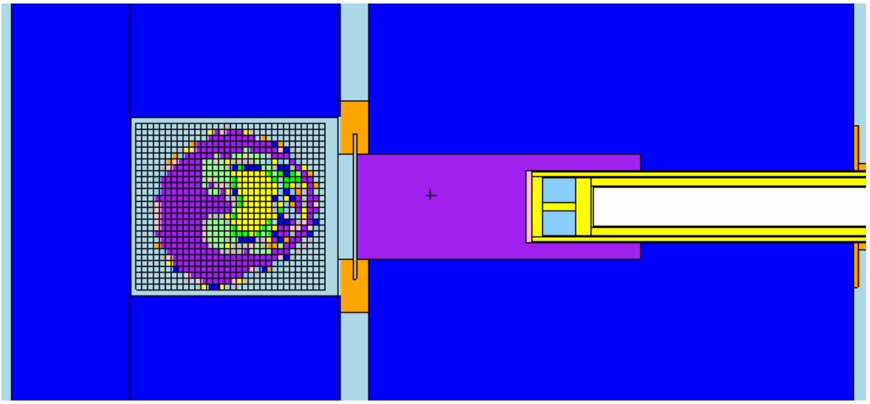
- **Following slides show examples of MCNP being used in many applications.**

  - 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.**





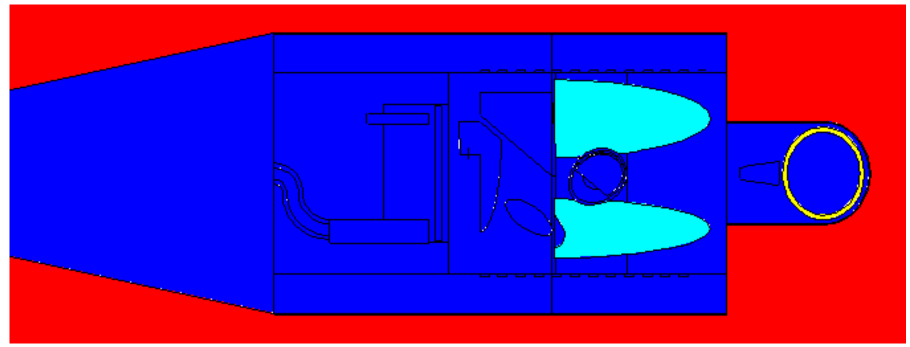
Pictures from mcnp plotter

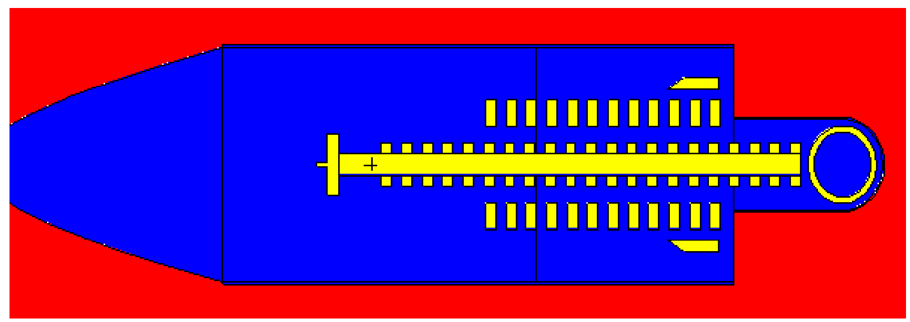
J. R. Albritton, ''Analysis of the SERA treatment planning system and its use in boron neutron capture synovectomy,'' M. S. thesis, Massachusetts Institute of Technology, 2001.

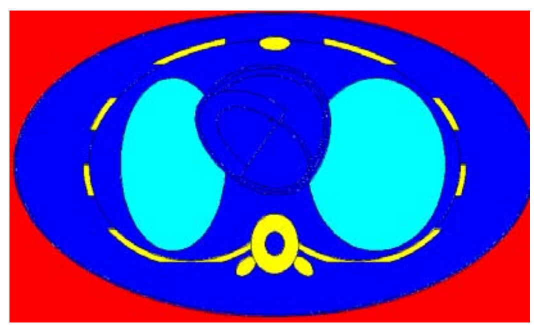
Gierga DP, Yanch JC, Shefer RE, "An investigation of the feasibility of gadolinium for neutron capture synovectomy", Med Phys. 2000 Jul;27(7):1685-92.

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.**







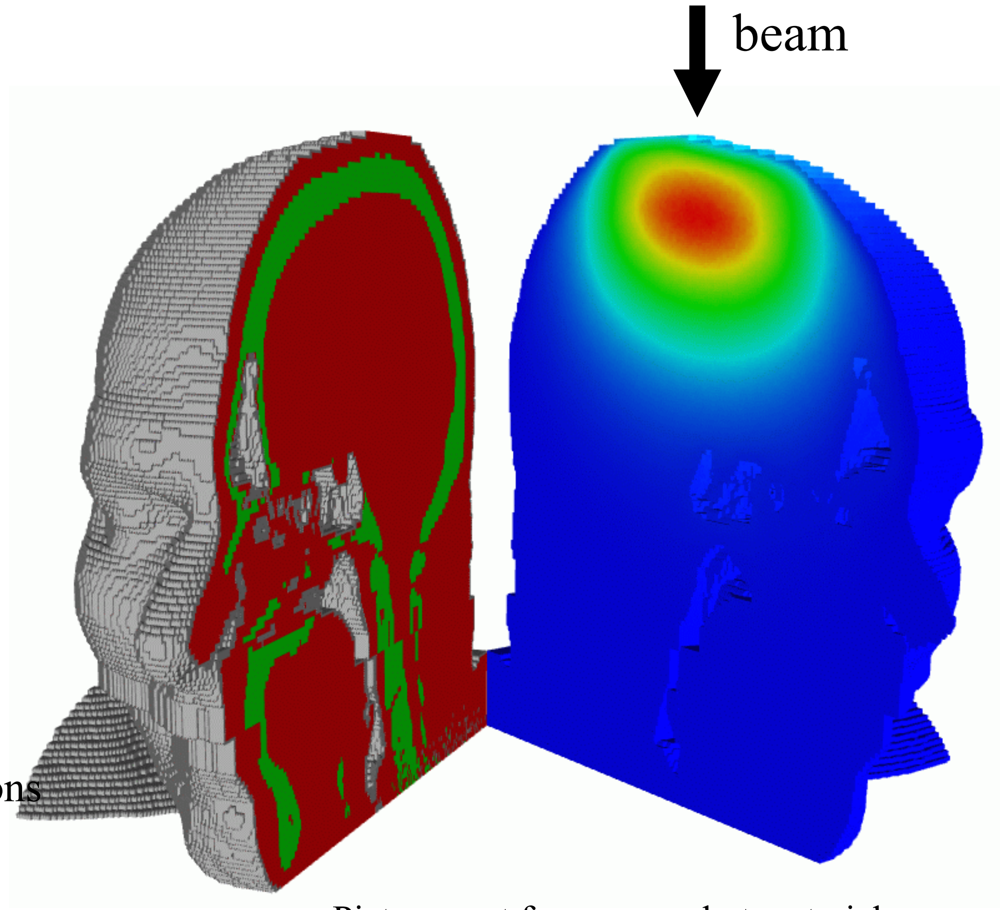
Pictures from mcnp plotter

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.**



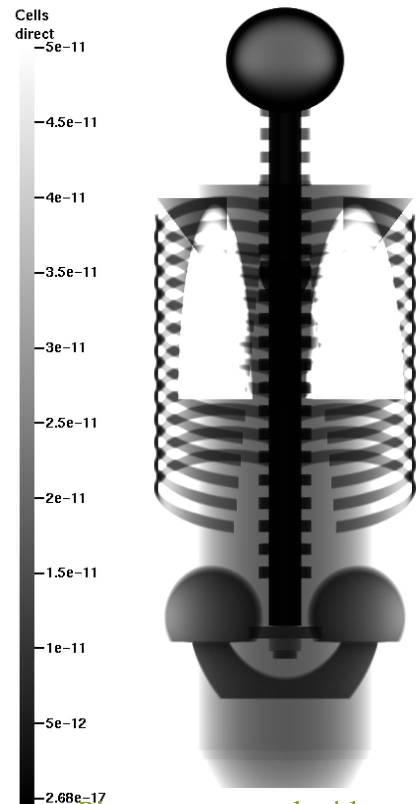
Pictures not from mcnp, but materials (left) and doses(right) from mcnp calculation.

- 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



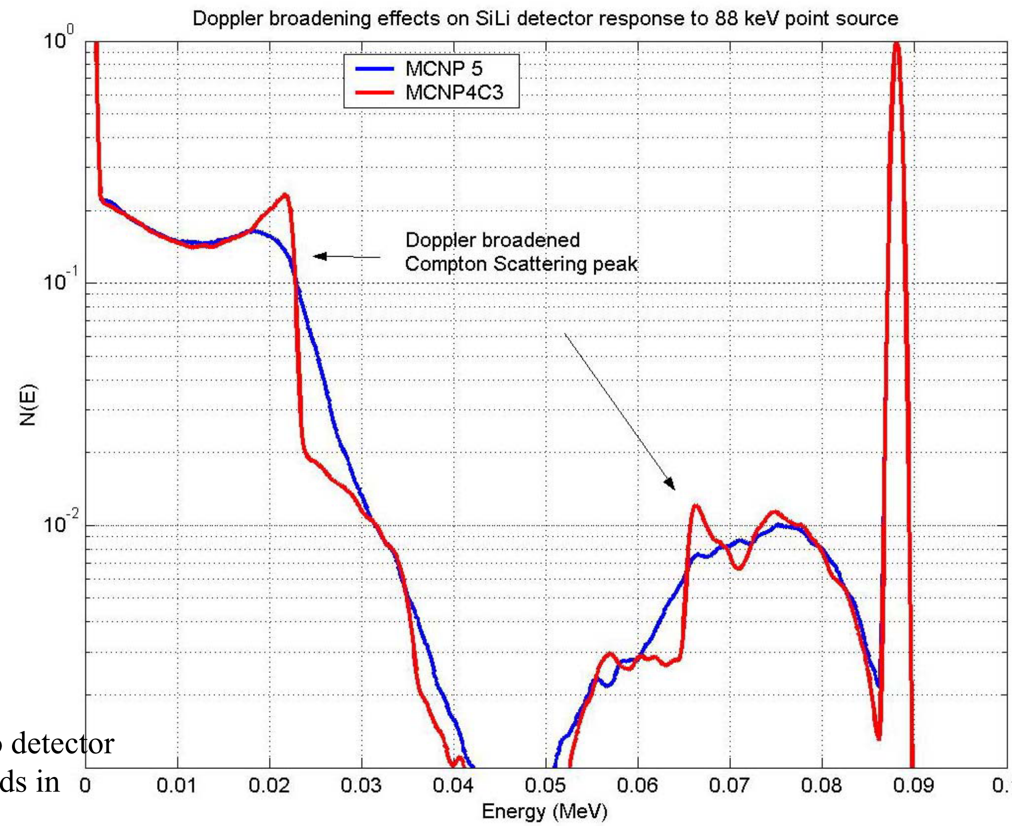
Picture generated with results from MCNP calculation.

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)**

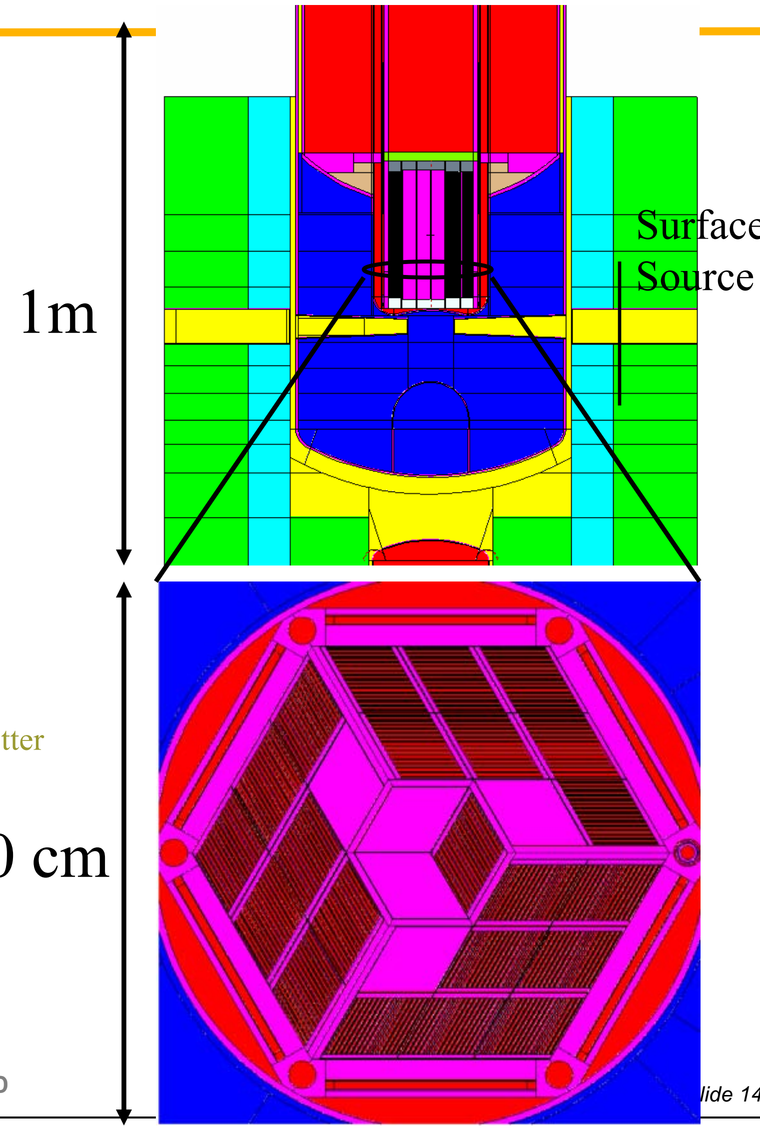


Sood, R. Gardner, "A new Monte Carlo assisted approach to detector response functions", Nuclear Instruments and Methods in Physics Research B, **213** (2004) 100-104.

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.**

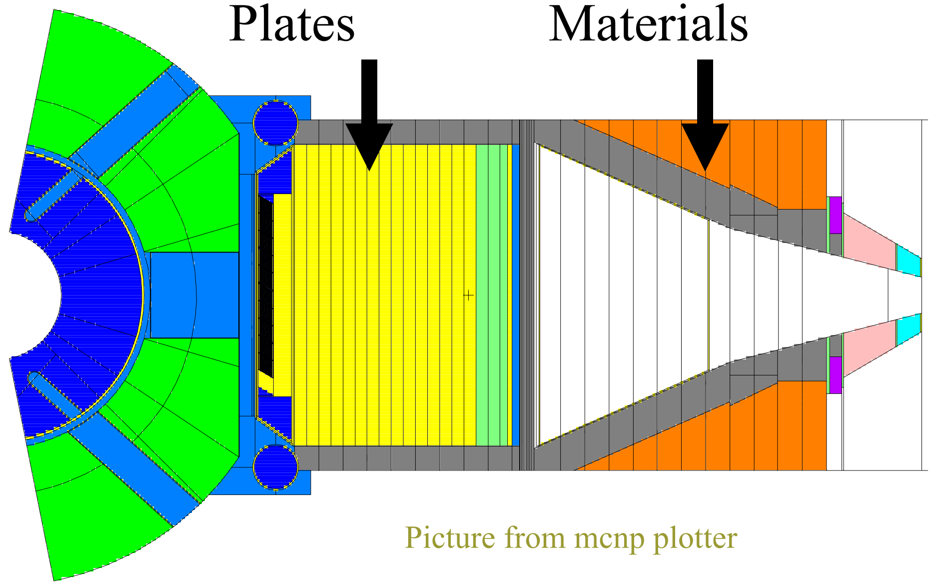


Picture from mcnp plotter

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.



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).

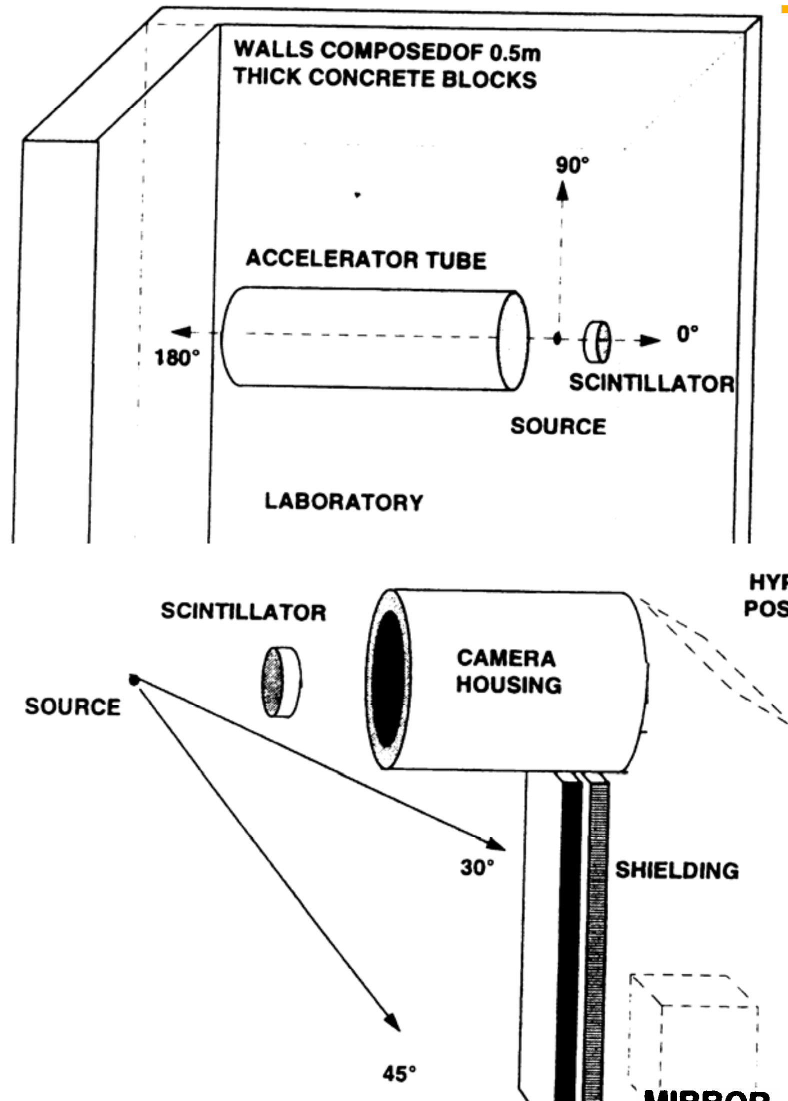
O.K. Harling, K.J. Riley, T.H. Newton, B.A. Wilson, J.A. Bernard, L.-W. Hu, E.J. Fonteneau, P.T. Menadier, S.J. Ali, B. Sutharshan, G.E. Kohse, Y. Ostrovsky, P.H. Stahle, R.L. Binns, W.S. Kiger III, and P.M. Busse, "The Fission Converter Based Epithermal Neutron Irradiation Facility at the MIT Reactor,"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



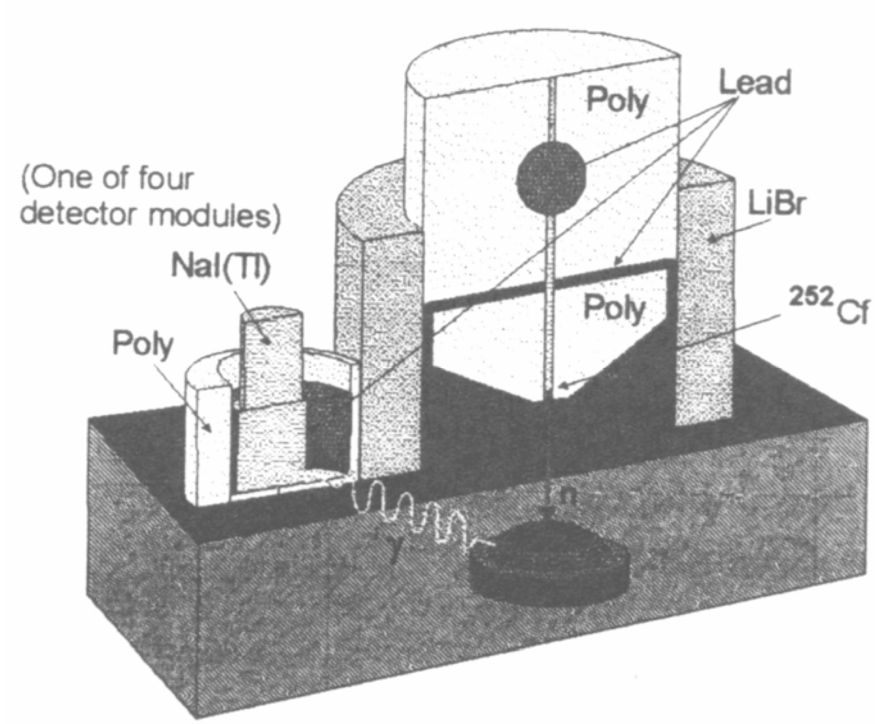
R.M. Ambrosi, J.I.W. Watterson, B.R.K. Kala "A Monte Carlo study of the effect of neutron scattering in a fast neutron radiography facility" NIMB **139** (1998) 286-292.

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.

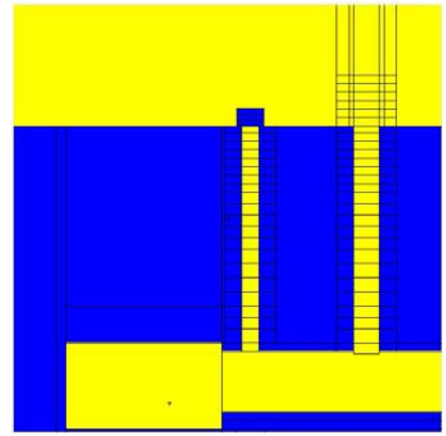


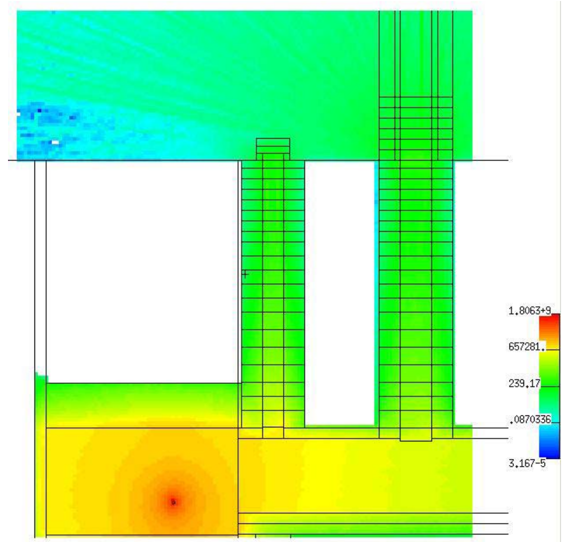
T. Cousins, T.A. Jones, et. Al. “The development of a thermal neutron activation (TNA) system as a confirmatory non-metallic land mine detector” J. Rad. Nucl. Chem. **235** (1998) 53-58.

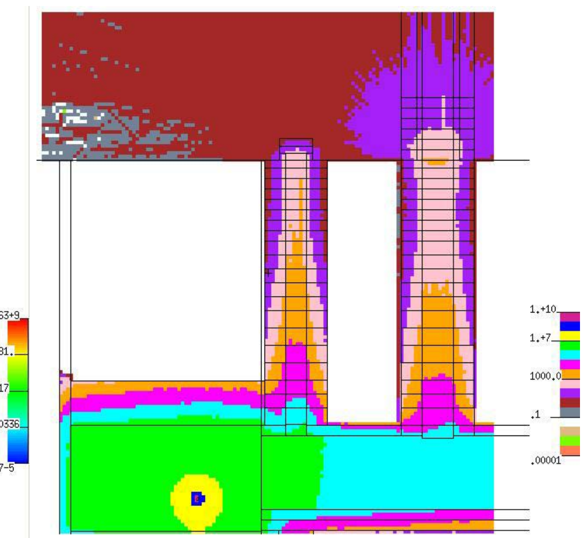
Calculate Dose – Health Physics

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

Geometry
Blue = concrete
Yellow = air
Picture from mcnp plotter



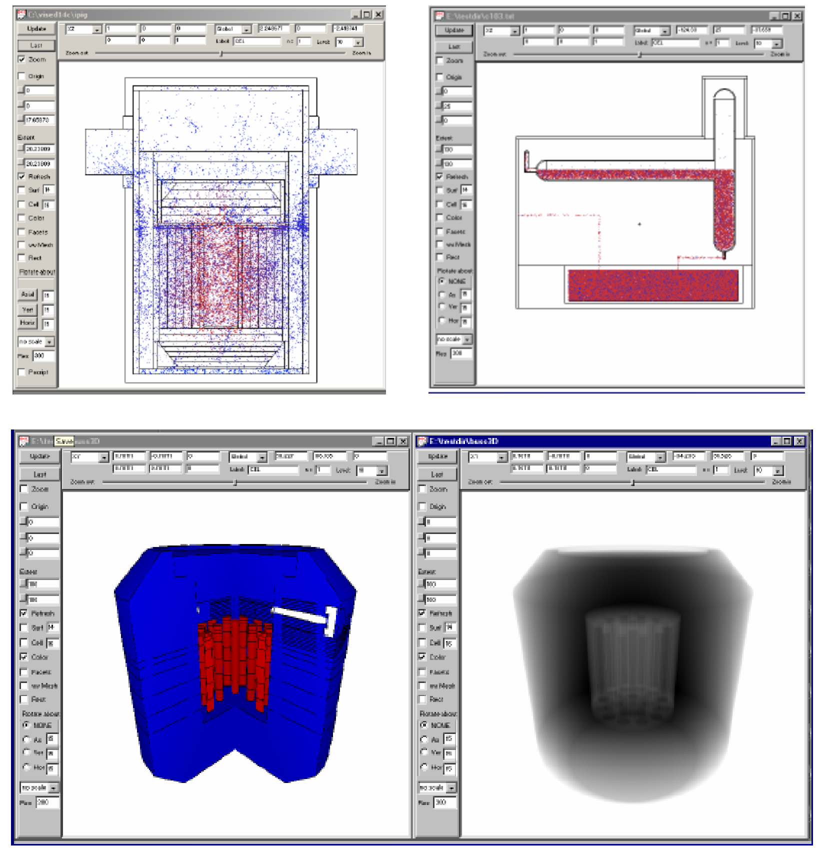




Pictures from beta version of mcnp6 mesh tally plotter

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.

What MCNP5 Cannot Do

What MCNP5 cannot do

- High-Energy Particles (muons, pions, etc..) — MCNPX/6
- Heavy Charged Particle Transport (protons, alphas, etc.) — MCNPX/6
- Magnetic Field Tracking
  - In Void — MCNPX/6
  - In Materials — MCNP6
- Coincident Counting (lacks code and data)
- Short Length-Scale (<100 micron) tracking (for DNA Damage)
- Photon Polarization

New MCNP5 Features

MCNP5 New Features for MP

Mesh Tallies	1st Release	1.14
Radiography Tallies	1st Release	1.14
Photon Doppler Broadening	1st Release	1.14
More Detectors & Tallies	2nd Release	1.20
>2.1 Billion Histories & RAND #	3rd Release	1.30
Lattice Tally Enhancements	3rd Release	1.30
Mesh Tally Improvements	4th Release	1.40
Electron Improvements	4th Release	1.40
Stochastic Geometry	4th Release	1.40
Large Lattice Improvements	5th Release	1.50
Pulse Height Tally Variance Reduction	5th Release	1.50
FUTURE WORK for MCNP5 Teaser		

Mesh Tallies

- **Geometry independent 3-D tally grid used to calculate volume averaged fluxes for each voxel in that grid.**
- **Cylindrical or rectangular mesh.**
- **Can be used with DE DF and FM cards to calculate volume averaged doses and reaction rates.**
  - Cannot yet be used to calculate dose for different materials that the mesh may cover
- **Can be used with TR cards (transformation).**
- **Particles must track through mesh to tally.**

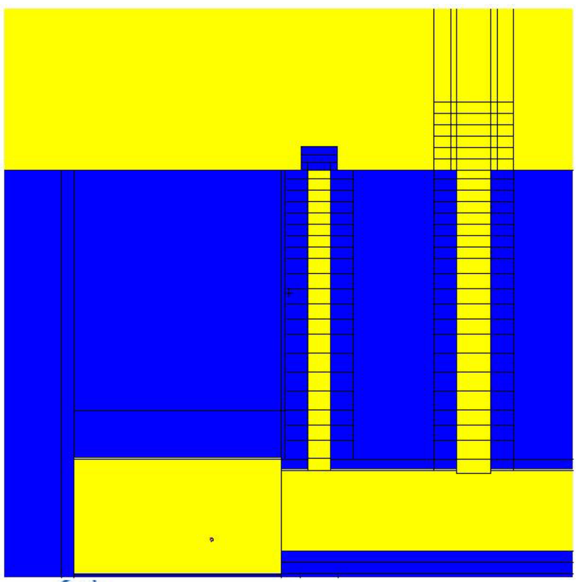
Mesh Tallies

- Built-in MCNP5 plotter now plots mesh tally grid superimposed over geometry

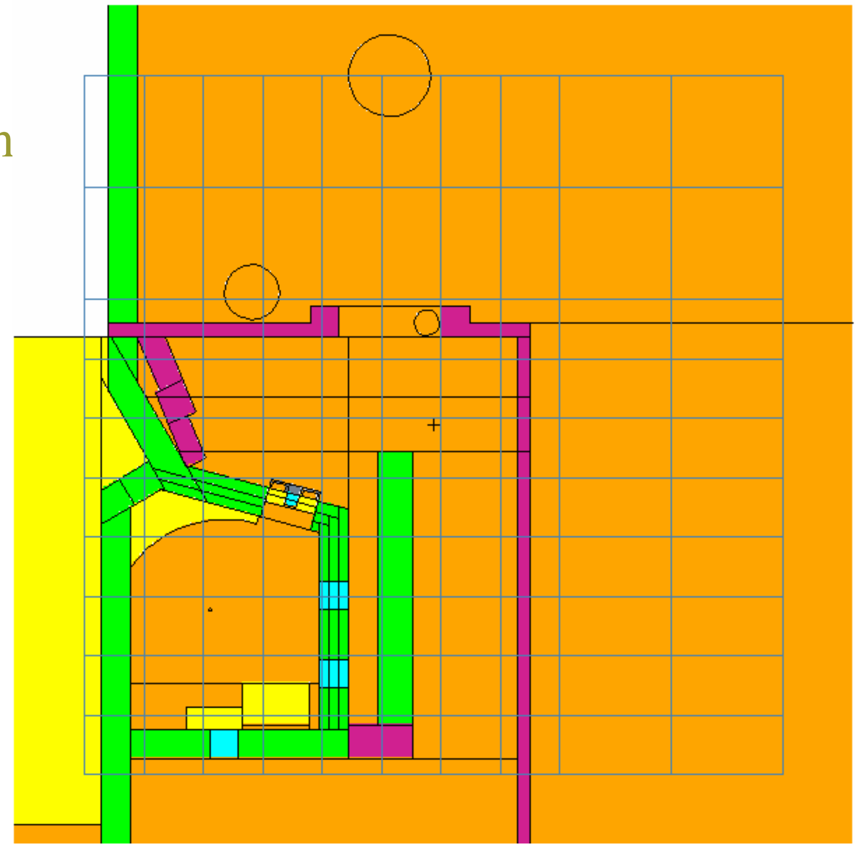
Geometry

Blue = concrete

Yellow = air



Images from mcnp5 plotter



Mesh Tally – Card Format

FMESHn:p create a mesh track-length tally where n is the tally number. Can be used with DEn, DFn, and FMn cards.

Caution: It is easy to create huge mesh tallies that can overflow computer memory.

Keywords GEOM{xyz} ORIGIN{0,0,0} AXS{0,0,1} VEC{1,0,0} IMESH IINTS{1} JMESH JINTS{1} KMESH KINTS{1} EMESH EINTS{1} FACTOR{1.} OUT(col} TR

GEOM = mesh geometry: Cartesian (“xyz” or “rec”) or cylindrical (“rzt” or “cyl”)

ORIGIN = x,y,z coordinates in MCNP cell geometry superimposed mesh origin

AXS = direction vector of the cylindrical mesh axis

VEC = direction vector, along with AXS that defines the plane for angle theta=0

IMESH = coarse mesh locations in x (rectangular) or r (cylindrical) direction

IINTS = number of fine meshes within corresponding coarse meshes

JMESH = coarse mesh locations in y (rectangular) or z (cylindrical) direction

JINTS = number of fine meshes within corresponding coarse meshes

KMESH = coarse mesh locations in z (rectangular) or theta (cylindrical) direction

KINTS = number of fine meshes within corresponding coarse meshes

EMESH = values of coarse meshes in energy

EINTS = number of fine meshes within corresponding coarse energy meshes

FACTOR = multiplicative factor for each mesh

TR = transformation number to be applied to the tally mesh

HINT: MCNP5 Manual Index – FMESH Card, Mesh Tally,

WARNING: MESH refers to weight windows mesh, used for variance reduction, not tally mesh.

Radiography Tallies

- Introduced in MCNP5_RSICC_1.14. Allows the user to generate images from neutral particles as one would expect from an x-ray or pinhole projections.

- FIR – Flux image radiograph
- FIP – Flux image pinhole
- FIC – Flux image cylinder
- Distinguish between scattered and unscattered flux
- Uses point detector methods.

Radiography Tallies

Radiograph of Anthropomorphic MCAT phantom

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

Picture from Sabrina

Picture generated with results from MCNP calculation.

Simulated Radiograph

1 M pixels

Radiography Tally – Card Format

- **General card format for FIR tally:**
  - FIRn:p X1 Y1 Z1 R0 X2 Y2 Z2 F1 F2 F3
- **NOTRN: Run only direct contribution to all point detector tallies**
- **TALNP: Eliminate tally prints with many bins from OUTP file**
- **NPS: 2nd entry controls the direct contribution for FIR tallies**
- **FSn and Cn cards control number of pixels in image plane**
- **Example for simulation of medical radiograph:**

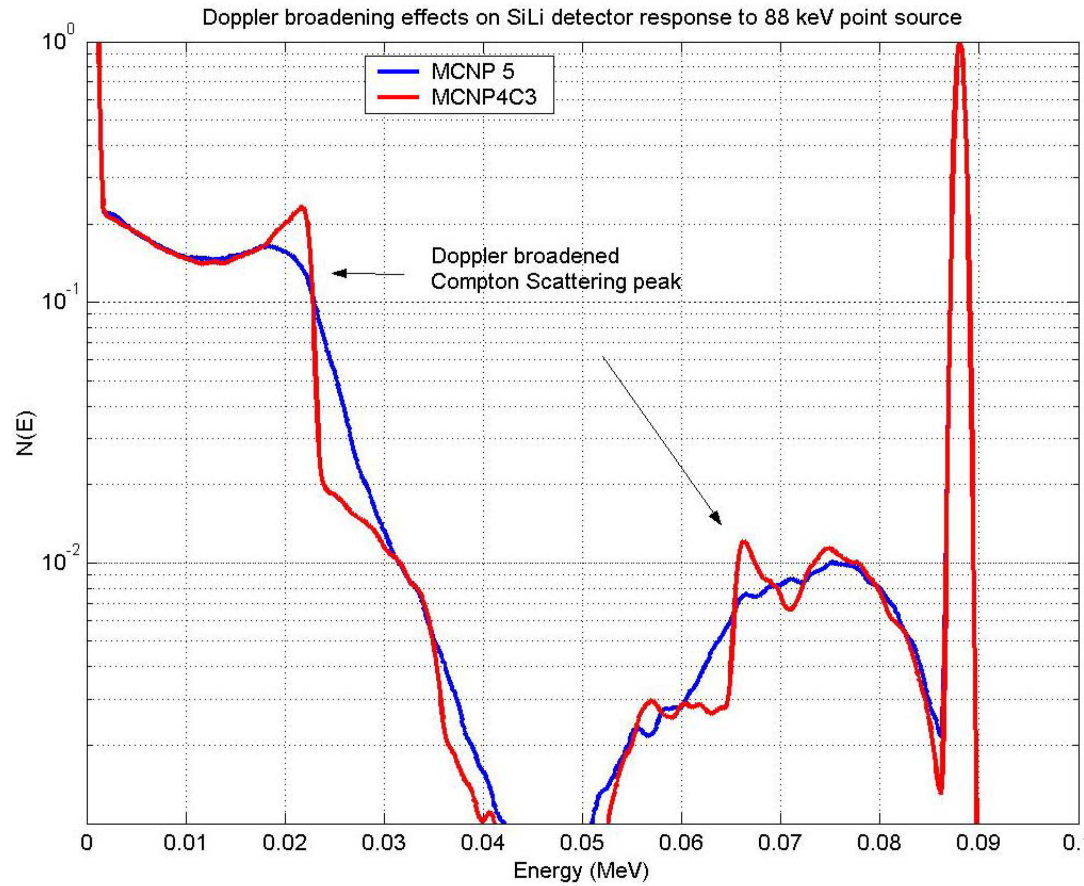
```
fir5:p 0 0 15. 0 0 0 -1000. 0 1e20 0
fs5 -55.0 999i 50.0
c5 -30.0 999i 30.0
notrn
talnp
```

HINT: MCNP5 Manual Index – Radiography Tallies, Pinhole, Flux Image Radiographs

HINT: Use with NOTRN card for faster calculations if scattered contributions are not needed.

Photon Doppler Broadening

- **Released in MCNP5_RSICC_1.14**
- **Incoherent Compton event, includes electron binding energy.**
- **Causes reduction of the photon's total scattering xs in the foreward direction.**
- **Causes broadening of photons' energy spectrum.**
- **Important E_p < 1 MeV.**
- **Bug fix in MCNP5_RSICC_1.40 release**



Doppler - Card Format

- By default, this option is on.
- Photon Doppler broadening will be used if appropriate data (xs library - #000.04p) is available. If xs library not available, comment is issued: “#000.0#p lacks Compton profile data for photon energy broadening”
- To turn off, set 4th entry of phys:p to 1.

HINT: MCNP5 Manual Index – Doppler Broadening, PHYS card

More Detectors & Tallies

- **With release of MCNP5_RSICC_1.20**
- **Maximum # of detectors increased from 20 to 100.**
- **Maximum # of tallies increased from 100 to 1000.**
- **Limit for a specific tally type still 100**

>2.1 Billion Histories

- **With MCNP5_RSICC_1.30, more than 2.1 billion histories can be run (<1E20)**
- **Done by explicitly declaring ~30 variables as 8 byte integers.**
- **Supported Cards: NPS, PRDMP, RAND, PTRAC, MPLOT**
- **Large PTRAC files also supported (250+ Gigabytes)**
- **Larger random # stride (not default): RAND card**
  - Prevent re-use of random numbers
  - Old Period : ~10^{14} New Period: ~10^{19}

HINT: MCNP5 Manual Index - NPS card, other card entries.

WARNING: # of histories does not correlate to simulated source strength!

Lattice Tally Speed Enhancement

- With release of MCNP5_RSICC_1.30, if certain conditions are meet, then runtimes can be significantly reduced (5-500 times shorter, depending on problem).
- Stringent Conditions: F4, DE DF, 1st level lattice.
- MCNP will attempt to determine if these conditions have been meet or not, and will attempt to use the enhancement if appropriate. Messages either way. Fast and slow runs will track.
- Card: SPDTL

SPDTL – Card Format

- In data card section: spdtl <force or off>
- "spdtl force" will cause the lattice tally enhancements to be used if at all appropriate.
- "spdtl off" will enforce the older (slower) tally routines.
- MCNP5 will automatically check for nearly all conflicts and respond.
- Documentation – LA-UR-04-3400 provided with MCNP5 distribution

HINT: MCNP5 Manual Index – SPDTL card

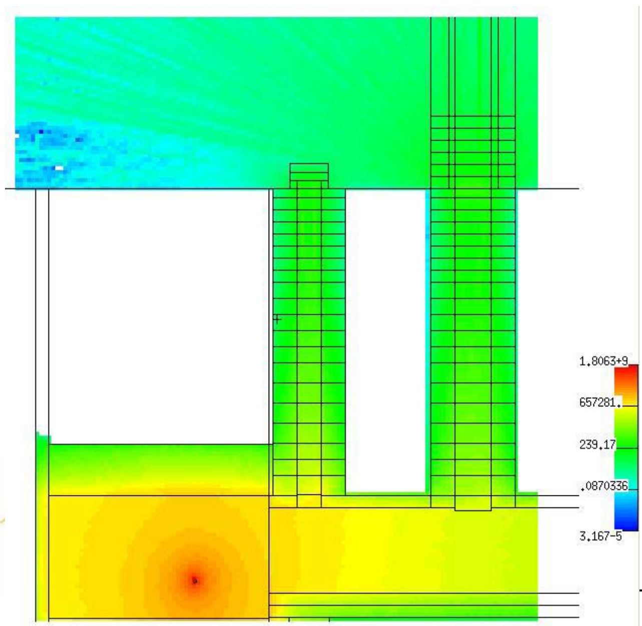
MCNP5 Mesh Tally Plotting

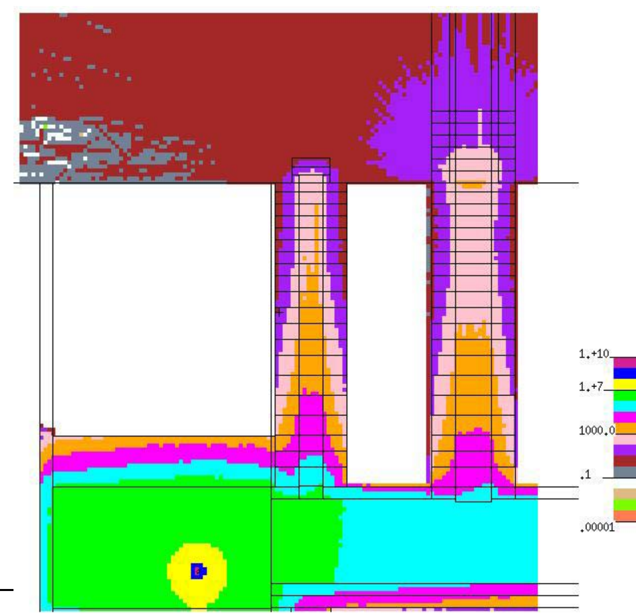
- Released in MCNP5_RSICC_1.40
- Built-in plotter now plots mesh tally results on top of geometry outline

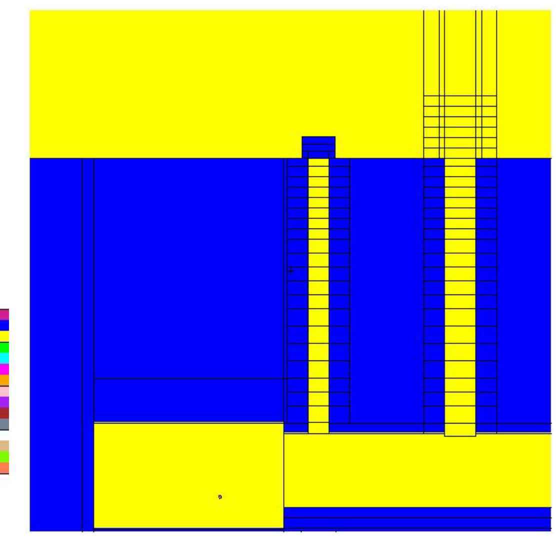
Proton Storage Ring at LANSCE accelerator

Images from MCNP5 plotter

Dose rate calculation for cable penetrations





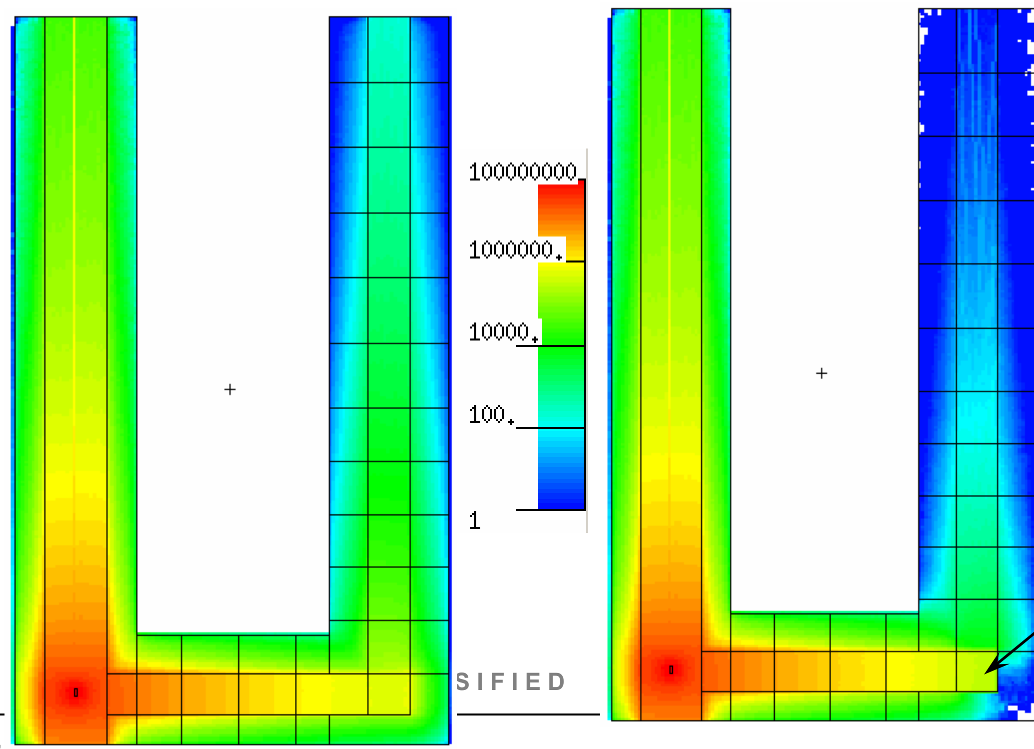


MCNP5 Mesh Tally Plotting

Use SF (Surface Flag) and CF (Cell Flag) cards as for a regular tally, except:

- **Only one tally (the flagged tally) is produced**
- **Negative cell or surface values interpreted as “anti-flag”. Scores only those particles that do not cross the surface or leave the cell**

Released in MCNP 5.1.40



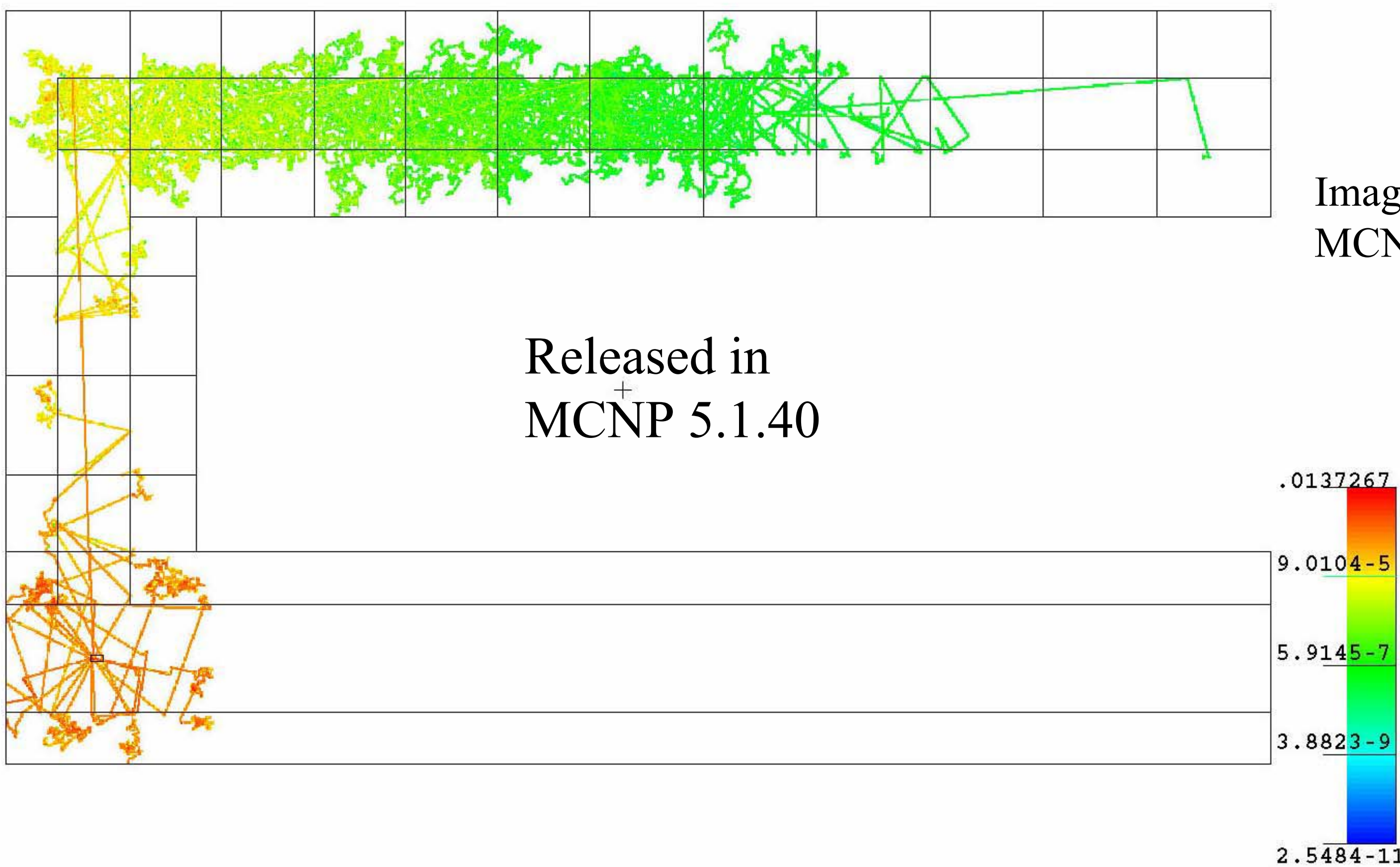
Images from MCNP5 plotter

Anti-flagged cell

MCNP5 Mesh Tally Plotting

By using a very fine mesh, particle tracks from individual histories can be plotted.

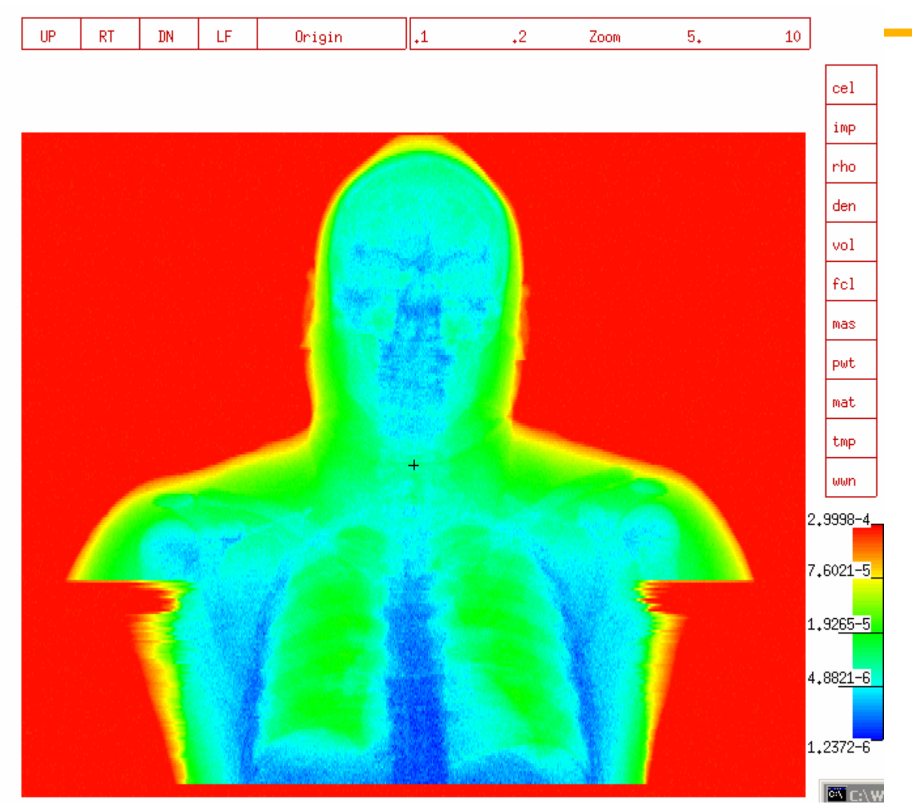
2000 x 1100 x 1 mesh

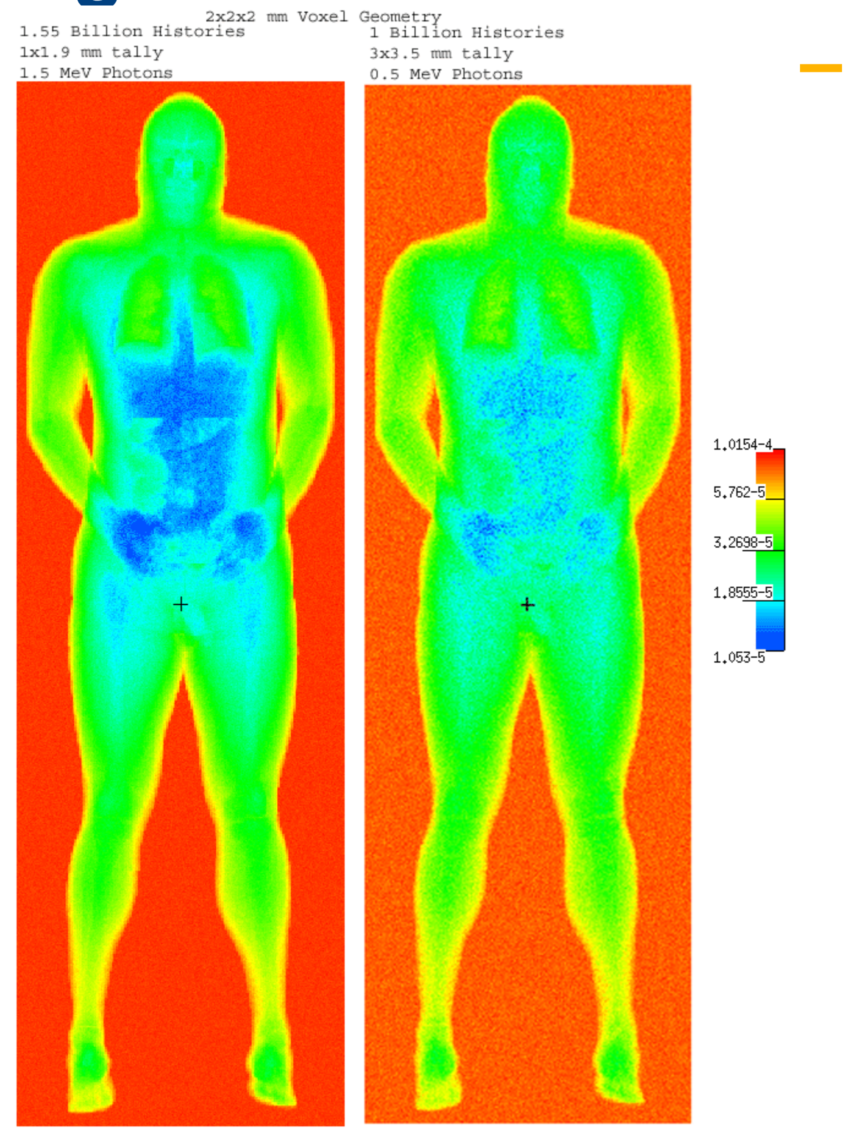


Released in
MCNP 5.1.40

Image from
MCNP5 plotter

MCNP5 Mesh Tally Plotting





Radiographs (Mesh Tally) of VIPMan model,

1x1x1 mm voxels (above),

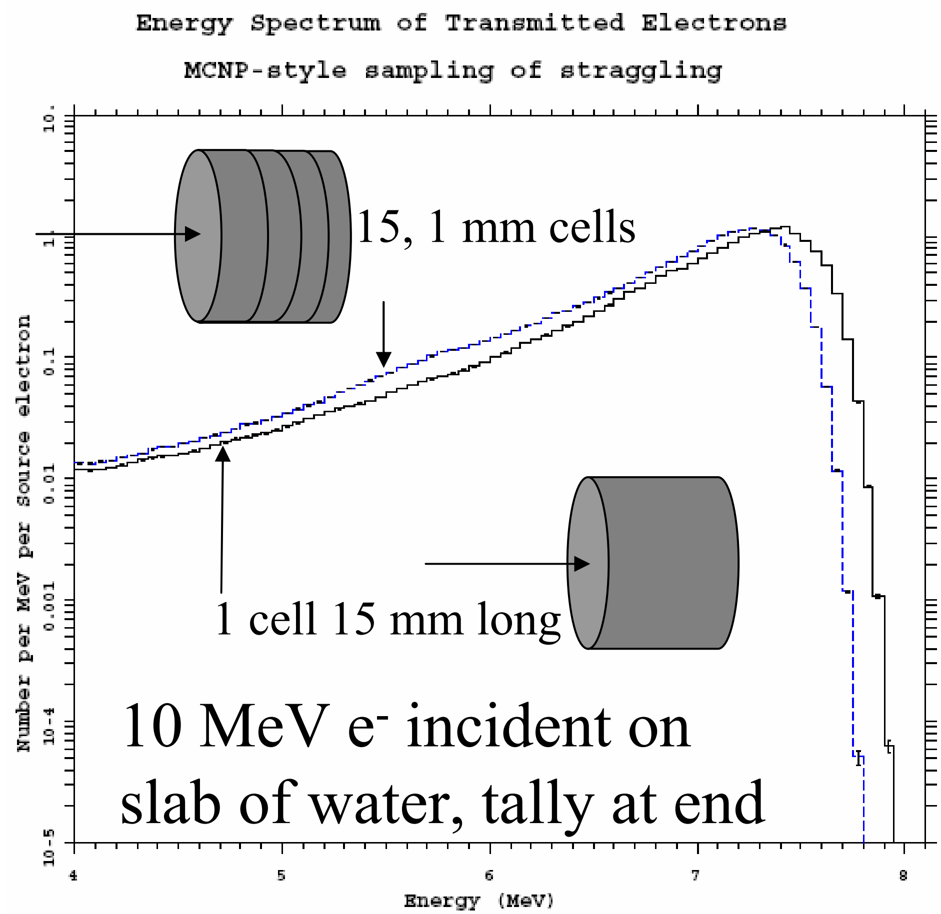
2x2x2 mm voxels (right)

Images from MCNP5 plotter

http://www.rpi.edu/dept/radsafe/public_html/home.htm

Electron Improvements

- **Released in MCNP 5.1.40**
- **Positron Source (SDEF par=4)**
- **For condensed-history electron transport, tables of Landau parameters were precomputed for a fixed step-size**
- **This could introduce errors for geometry with spacings less than the assumed Landau step-size**
- **Computing the Landau parameters on-the-fly for the current step-size & geometric distance eliminates these problems**
- **18th entry on DBCN card to 2**
- **DBCN 17j 2**

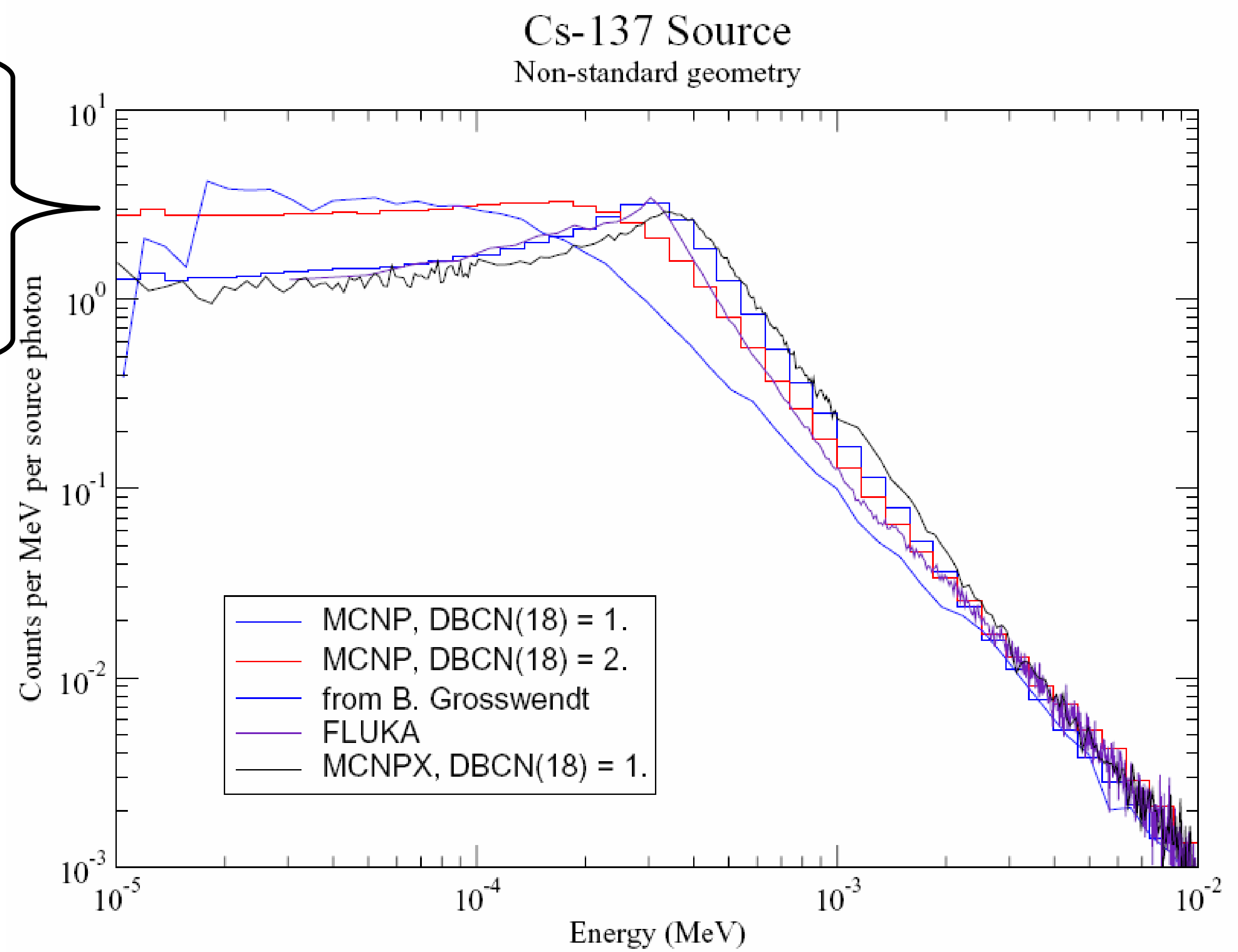


Hughes M&C 2005 Conference Paper

Electron Improvements

MCNP Improvements

& Grosswendt's code argee



Stochastic Geometry

- **Released in MCNP 5.1.40**
- **On-the-fly random translations of embedded universes in lattice**
- **Developed for pebble bed reactors.**
- **Potential for medical physics applications?**
  - Alveoli
  - Sinuses
  - Bone marrow
- **Use URAN card**
  - See MCNP5 Manual

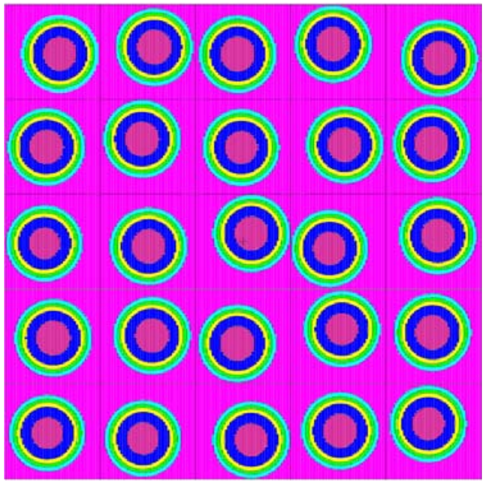


Image of the stochastic geometry of fuel kernels from MCNP5 plotter

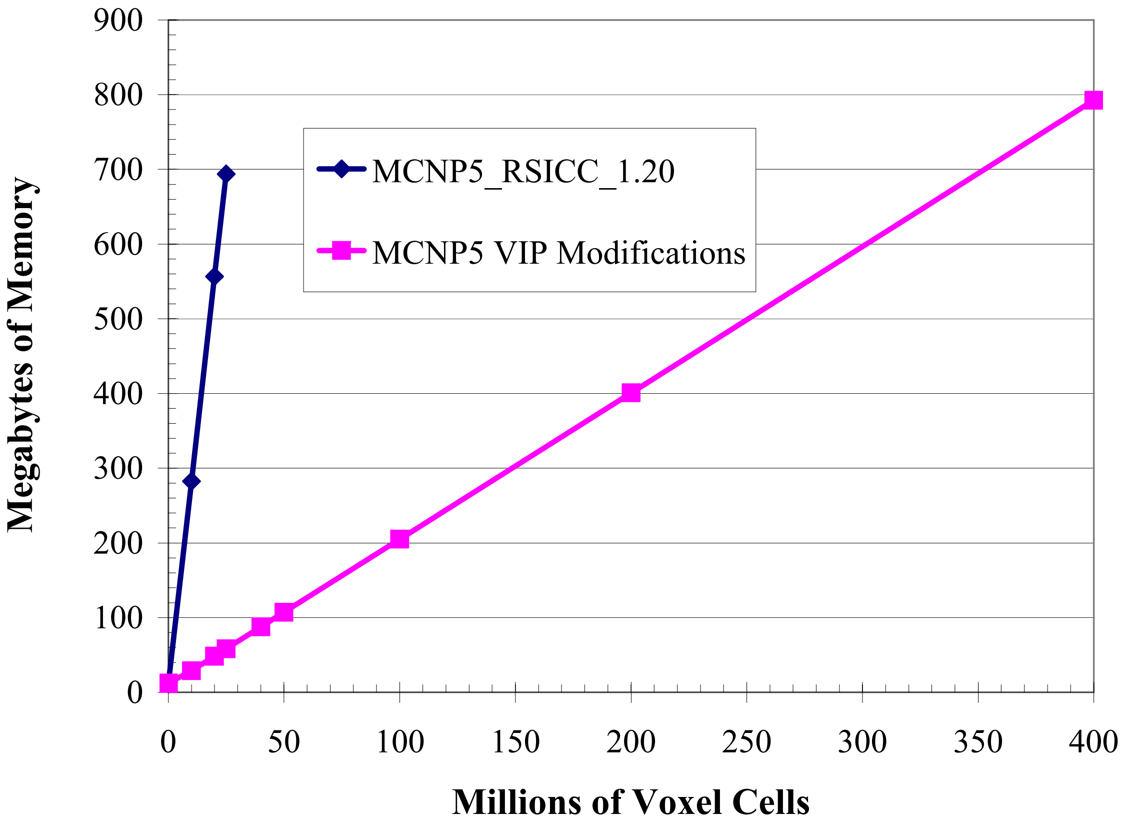
Fuel kernel displaced randomly within lattice element each time that particle enters

Forrest Brown, "**Monte Carlo Methods & MCNP Code Development**"
Monte Carlo 2005, Chatanooga, TN.

Large Lattice Improvements

- Increase limit on number of voxels from ~20 Million to ~200+ Million.
- Reduce startup times from hours or days to a few hours.
- Windows OS limit of 2 Gigabytes of Memory per program. (Use 64 bit chip & OS)
- Integrated into MCNP5 1.50
- BUT: Didn't implement full 2 byte Integers because not supported by MPI Standard

MCNP5 1.50

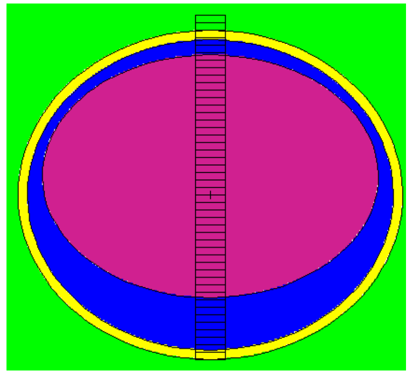


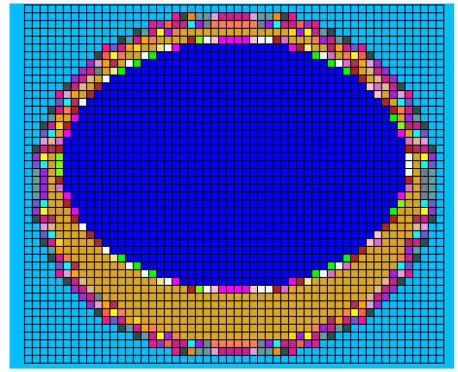
Goorley, "Issues Related to the use of MCNP code for an Extremely Large Voxel Model VIP-MAN" Monte Carlo 2005.

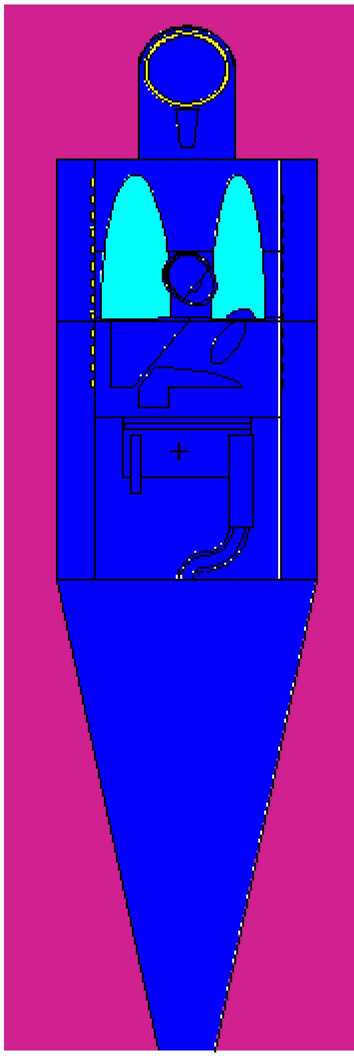
Anticipated next release – October 2007

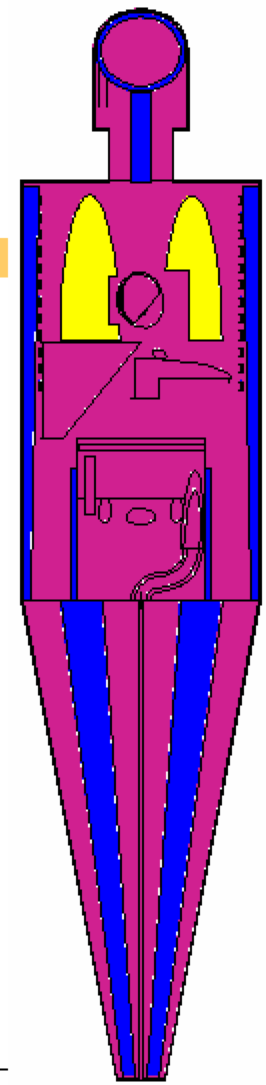
- Pulse Height Tally Variance Reduction
- Improved S(α,β) thermal neutron treatment
- Large Lattice Memory Improvements
- Long Path and File names
- Ignore tabs reading input deck
- Temperature adjusted neutron xs
- MCNP Medical Physics Primer
- ENDF/B – VII Nuclear Data

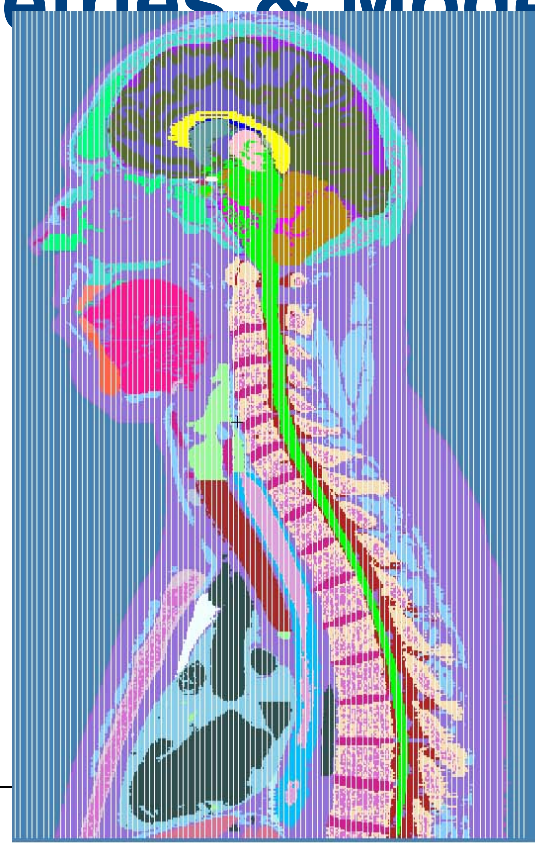
Geometries & Modeling





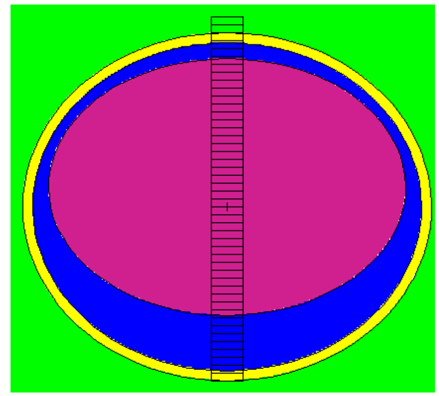




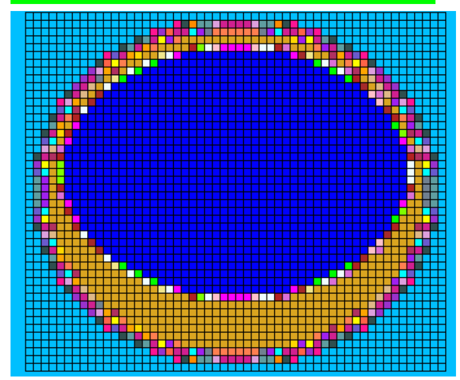


Geometries and Modeling

- **Analytical Phantoms**
  - MIRD Phantoms
- **Voxel Phantoms**
  - CT based Geometries
- **Phantom Database**
  - Set of MIRD and CT based Phantoms Distributed with MCNP5_RSICC_1.40



Images of Snyder Head Phantom from MCNP5 plotter.



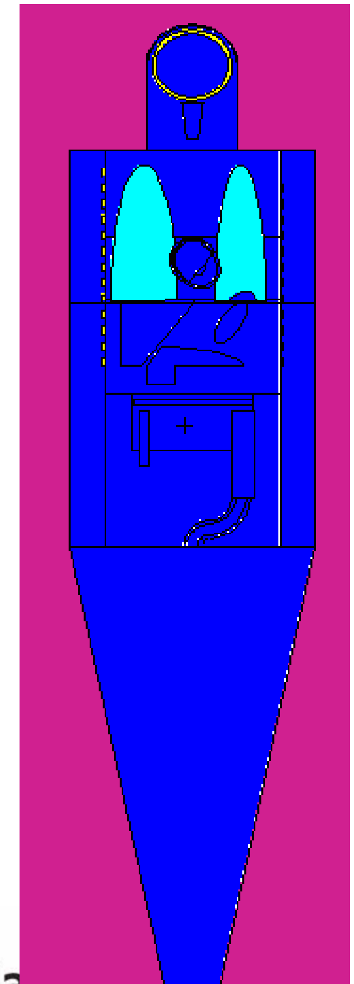
Input decks in MCNP5_1.40

Sample_Problems / Medical_Physics

Analytical Models

- **Conversion of equations into input deck, usually by hand. (sometimes tedious)**
- **MCNP Cells correspond to specific organs**
  - Easy to tally organ average
  - Easy to define materials (ICRU 46 for bio mats)
- **Calculate (flux/dose/reaction rate) distribution within organ with mesh tally or other user-defined surfaces**
- **Usually requires little memory**

Analytical Models

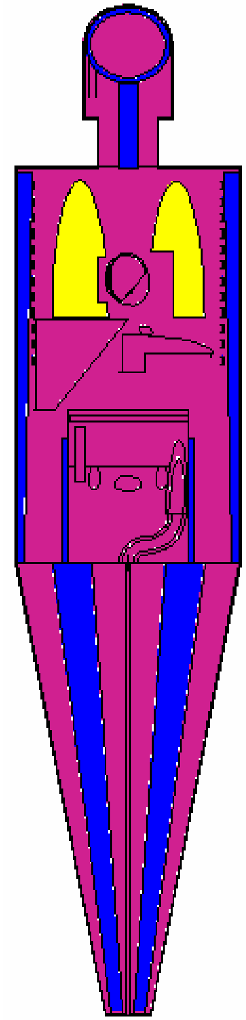


Yanch - MIT

Geometry plots from MCNP5 plotter

Observe differences in organs and materials.

Input decks in MCNP5_1.40 Sample_Problems/ Medical_Physics



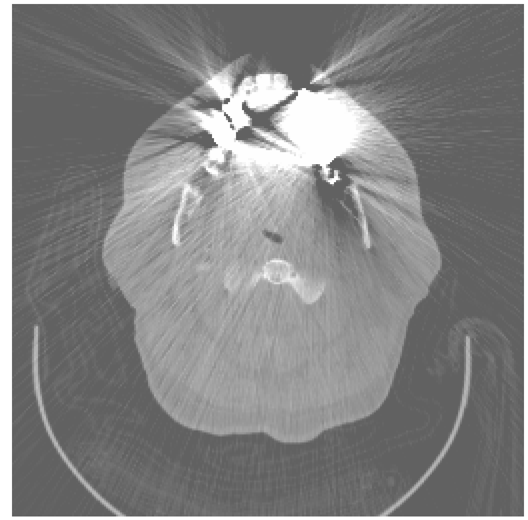
ORNL

Voxel Models

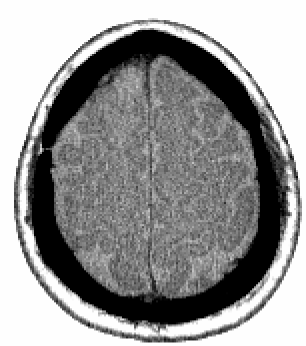
- **Obtain CT image data**
  - Can be patient specific
  - CTs preserve distances and volumes (better than MRI)
  - Can take CT of experimental phantom to compare calculations to experiments
    - (Reverse is possible - see talk by George Xu, where he starts with CT image and then build 3D phantom)
  - Possible use of CT contrast agent

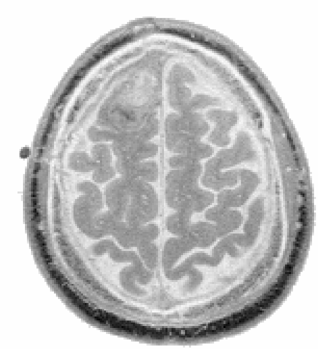
Voxel Models

- **Image manipulation**
  - Remove artifacts from CT (dental fillings, for example)
  - Align multiple data set with fiducial markers



Images from NIH Image, Data from Beth Israel Deaconess Medical Center





Voxel Models

- **Image conversion from DICOM or other medical format into MCNP input.**
  - Reduction in # of voxels and increase voxel size.
  - Homogenization of small voxels into large voxels.
  - Threshold Hounsfield # (12 bit) to correspond to materials (air, tissue, bone – or more complex)
  - Manually define certain regions (outline tumor and fill it with different material, for example).
- **Uses the MCNP lattice feature**
  - Each different material corresponds to different filling universes and at a lower level, different cells. If possible, different organs have different materials.
  - Example on following page.

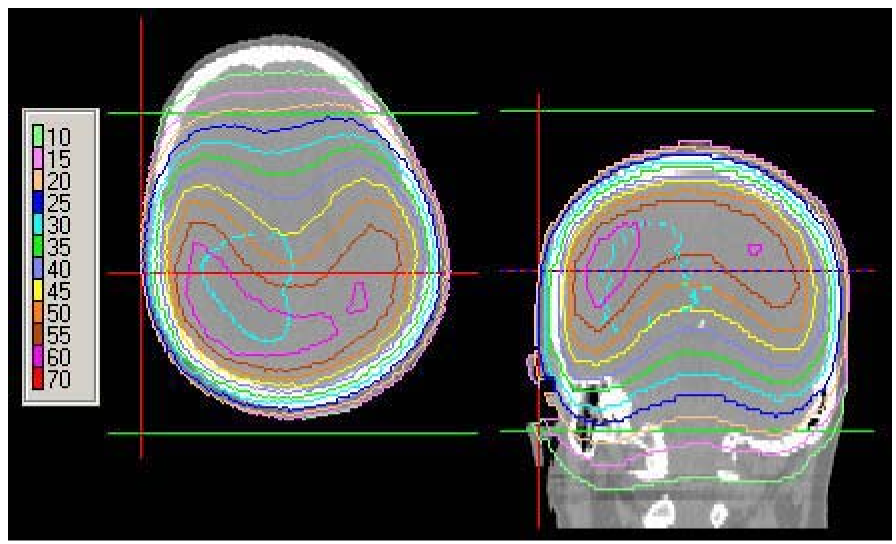
Memory Test of large lattices in MCNP5. 1K * 1K * 20 = 20,000,000 = 20M voxels.


```
1000   0  -11 10  -21 20  -31 30                $ Lattice Cell, bounding planes for single voxel
       lat=1 fill=  0: 999   0: 999   0: 19     $  fill=i1:i2 j1:j2 k1:k2, change k1,k2
       56  50 19999998r                         $  56 Xr, change X equal to (# voxels - 1)
       u=100                                    $  lattice cell is universe 100
  56   156  -1.29300E-03   -70  u=  56          $ Cell which fills each lattice voxel
  50   150  -1.29300E-03   -70  u=  50          $ Cell which fills each lattice voxel
1001  0   10  -12  20 -22  30 -32     fill=100  $ "Window" Cell, looking into lattice
1002  0  ( -10:  12:-20: 22:-30: 32) -1000      $ Outside window cell, inside bounding sphere
1003  0                      1000               $ Exterior of problem, particles die here
c   BLANK LINE

 10  px   -10.500000
 11  px   -10.479000    $  size to generate 1,000 lattice locations across x dimension
 12  px    10.500000
 20  py   -10.500000
 21  py   -10.479000    $  size to generate 1,000 lattice locations across y dimension
 22  py    10.500000
 30  pz   -12.500000
 31  pz   -11.250000    $ size to generate 20 lattice locations across z dimension
 32  pz    12.500000
c   Lattice entries = 1K * 1K * 20 = 20,000,000 = 20M voxels.
1000   so  10.0E+01
 70   so  5.0E+01
c   BLANK LINE

mode   n p
imp:n    1    3r  0
imp:p    1    3r  0
m156   7014 -0.77780  8016 -0.22220      $ Air
m150   1001 2  8016 1       $ Water
```

Voxel Models

- **Tally in regions of interest**
  - Tally over entire lattice (use of lattice speed tally capability possible)
  - Tally over cells (i.e. organs) of interest.
  - Use Mesh Tally to overlay geometry.
- **Possibly use post-processor to visualize isodose contours.**
- **If Mesh Tally is used, can plot dose contours in mesh plotter**

Image from clinical trials using NCTPlan (Harvard-MIT & CNEA)



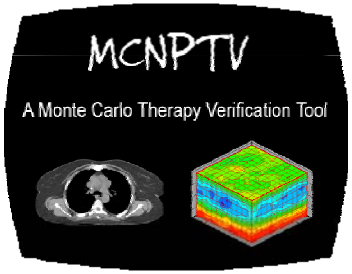
Voxel Models

- **Can easily consume Gigabytes of memory**
- **Large input decks 100s of MBytes, difficult to modify**
- **Limit in MCNP v 5.1.40 to ~20 million voxels (lattice locations) [Improved in MCNP v 5.1.50]**
- **Many users have created their own patches to speed up large voxel model calculations. (ORANGE, Speed Tally Patch)**
  - Monte Carlo 2005 Talk – Tues 4:45 Fast Monte Carlo Dose Calculations For All Particles: ORANGE By Steven Van Der Marck
- **Users are welcome to submit their patches for review and potential inclusion into MCNP.**

Conversion Programs

- **Currently available to the public:**
  - NCTPlan: Neutron Capture Therapy Plan. By Harvard-MIT & CNEA, Argentina (free – wskiger@mit.edu)*
  - Scan2MCNP: by White Rock Science (commercial - website)

- **Not ready for public release (but soon?)**
  - MiMMC: MultiModal Monte Carlo Treatment Planning System. By Harvard/Beth Israel Deaconess Medical Center.
  - MCNPTV: MCNP Therapy Verification. By Mark Wyatt (University of TN)
  - JCDS: JAERI Computational Dosimetry System.*
  - ImageJ & OEDIPE, by IRNS, France (irns.org)

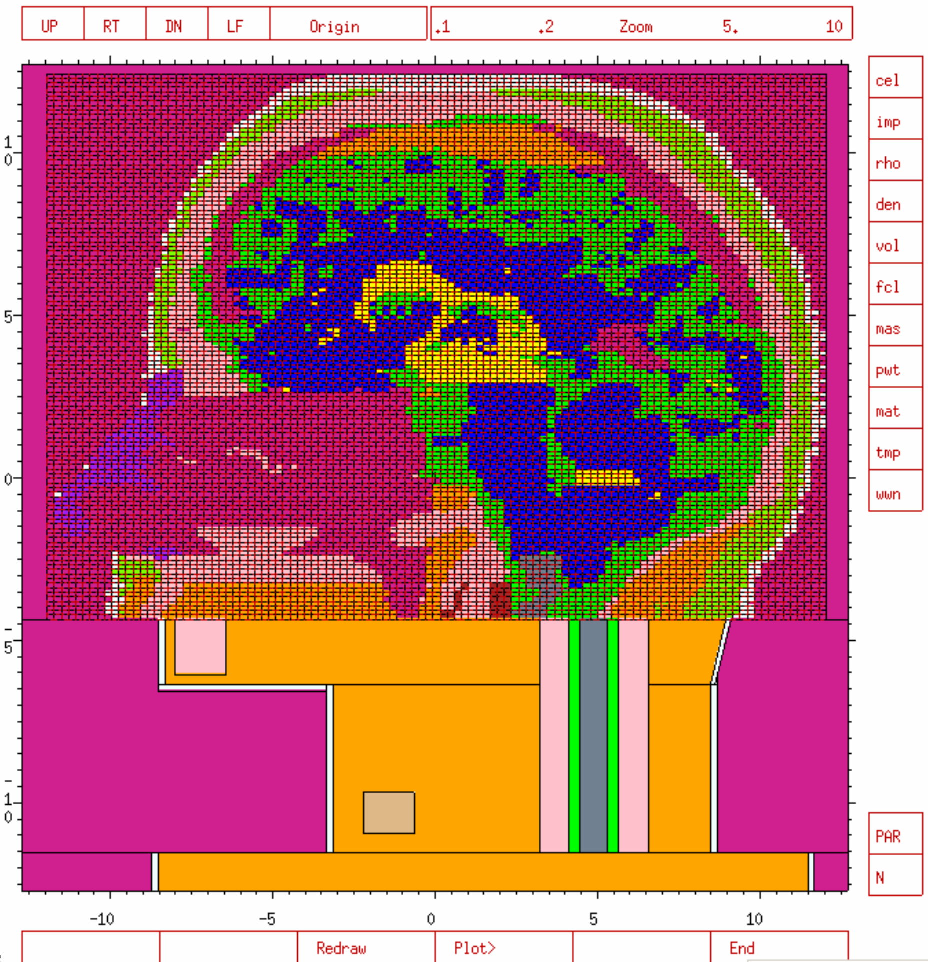


- **Not for public release?**
  - In-house versions at Ohio State, RPI.
  - THORPlan: By TsingHua University in Taiwan.

* Indicates use in human clinical trial irradiations.

Zubal Phantom

Image from
MCNP5 plotter

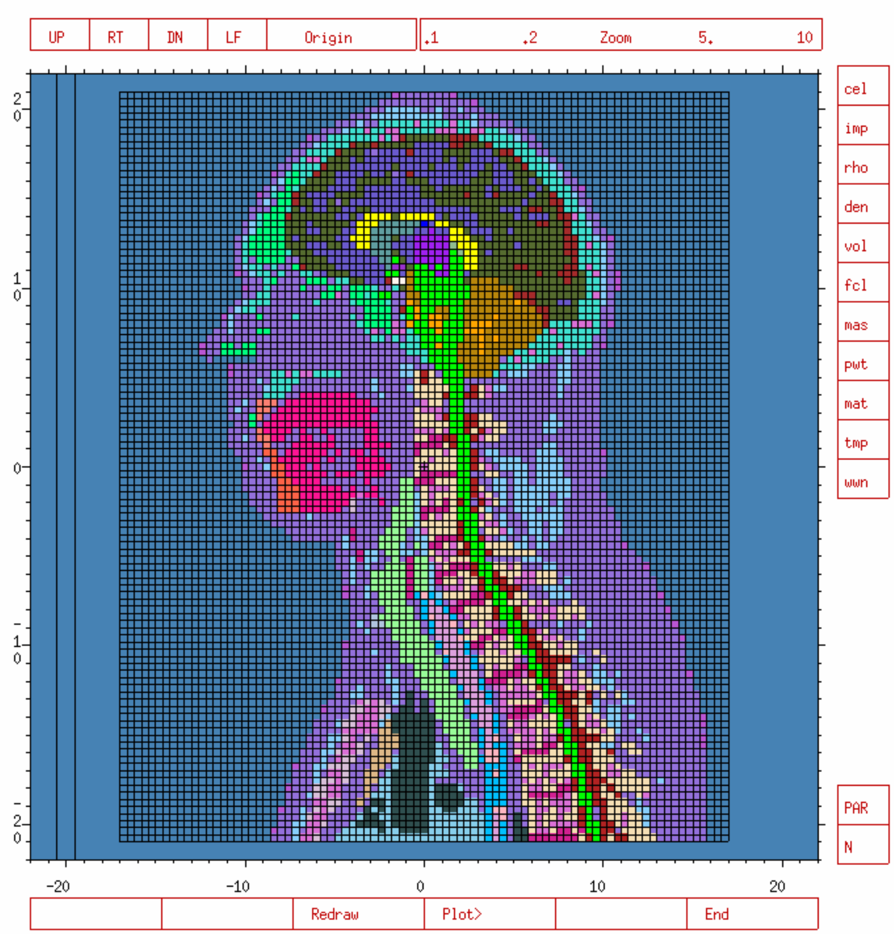


- **Voxel Phantom of Head**
- **85 x 109 x 120 voxels**
- **2.2 x 2.2 x 1.4 mm^3**
- **25 Brain structure tallies**
- **15 materials**
- **Jeff Evans, Ohio State**

Input deck in MCNP5_1.40
Sample_Problems/ Medical_Physics

VIP-Man

- Voxel Phantom of VIP-Man head and upper torso
- 147 x 86 x 105 voxels
- 2 x 2 x 2 mm
- 41 materials / organs
- By George Xu, RPI (xug2@rpi.edu)

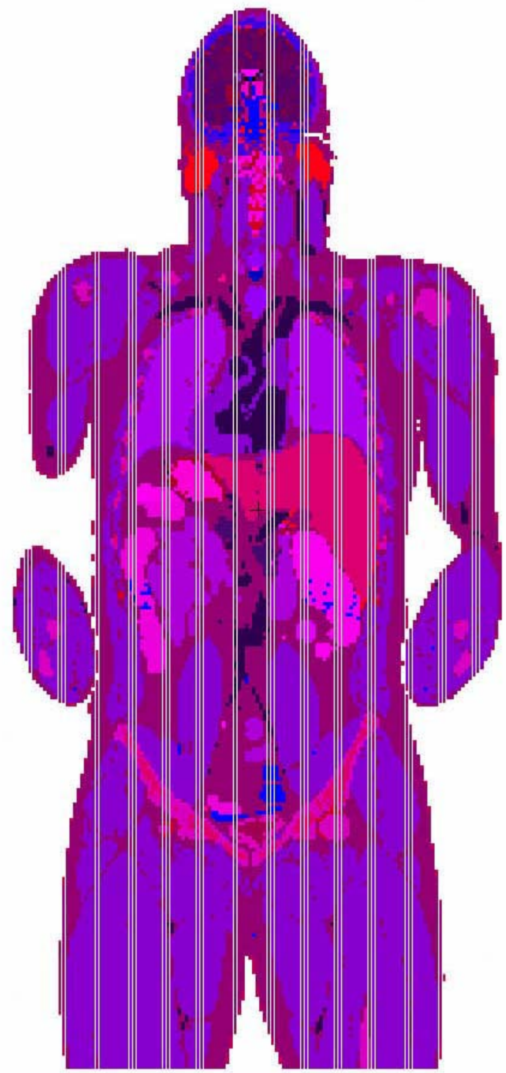


Input deck in MCNP5_1.50 Sample_Problems/ Medical_Physics

VIP Man

- Whole Body Phantom
- Based on NIH VIP-Man Project
- 6, 100, 300 Million Voxel Models
- 1 or 4 mm^3
- Available from Prof. Xu of RPI – not in this database

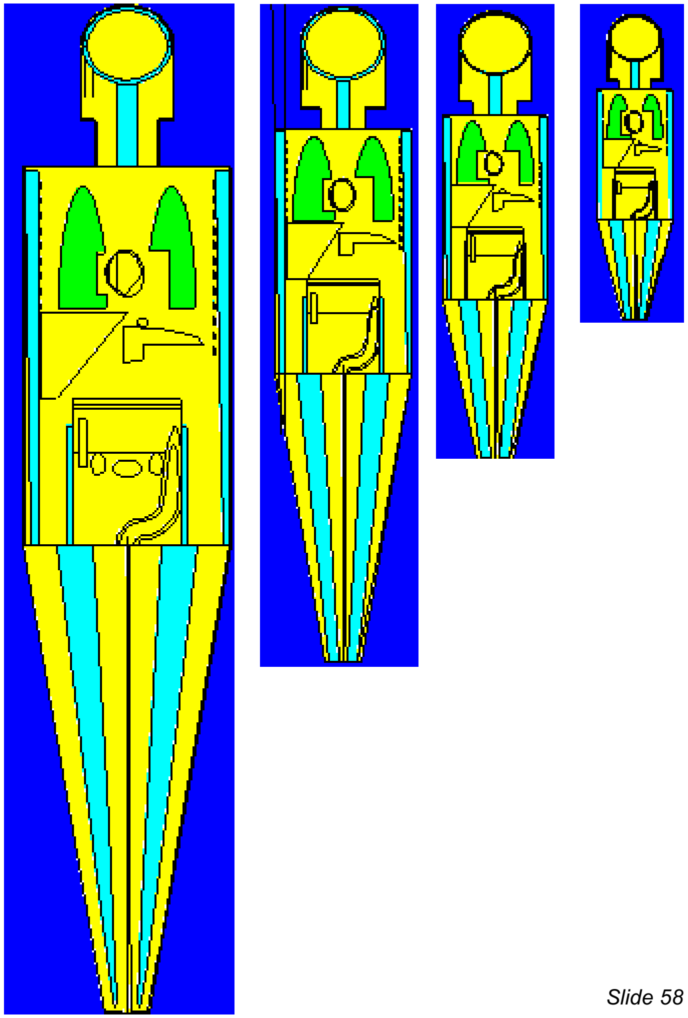
http://www.rpi.edu/dept/radsafe/public_html/home.htm



MIRD Humans

- Male, Female
- Children: 1, 5, 10, 15
- 40+ discrete cells
- 3 Materials
- D. Krstic and D. Nikezic, U. of Kragujevac, Serbia

Input deck in MCNP5_1.50
Sample_Problems/ Medical_Physics



MP Geometry Database

- **A database of Medical Physics phantom input decks distributed with MCNP5 or on MCNP website**

- **Analytical**
  - Snyder Head, ORNL MIRD, MIT MIRD, MIRD Female/Children
- **Voxel**
  - Snyder Head, Water Cubes, Zubal Head, Male Pelvis

- **Contributions Welcome!**

Break – 10 min

MCNP MP Sources

Modeling Radiation Source in MCNP

Every Radiation Source has:

- Location
  - **Point, surface or volume**
- Direction
  - **Isotropic, beam like, or angular distribution**
- Energy
  - **Single energy, multiple discrete lines, distribution**
- Particle type
  - **Neutrons, photons, electrons or positrons**
- Time distribution
  - **Constant, radioactive decay**

MCNP Sources

In this lecture, we will use the SDEF card to work the following:

A ^{99m}Tc (monoenergetic) point γ source in lung

- A ^{99m}Tc spherical γ source in Pb shield
- A ^{60}Co spherical γ source in Pb shield [optional]
- Two point gamma sources: ^{99m}Tc bottom, ^{38}S top
- Two spherical gamma sources: ^{99m}Tc bottom, ^{38}S top
- A neutron beam source [optional]

SDEF Data Card

Form: SDEF source variable=specification

Source Variable is an abbreviation for a physical description:

- ERG for Energy
- POS for Position (Location)
- VEC for Vector (Direction)
- Many More

Specification is a value or distribution, in one of three forms:

1. **explicit value:** **SDEF ERG=2.0**

   [default values; source energy = 2.0 MeV]

2. **distribution number:** **SDEF ERG=D1**

   [default values; source energy is a distribution ("D1" notation is explained later)]

3. **as a function of another variable:**

   SDEF POS=D1 ERG=FPOS=D2

   [default values; src position is a distribution; src energy depends on which position]

SDEF Source

When a physical description is omitted from the SDEF card, a default is assumed

Defaults:

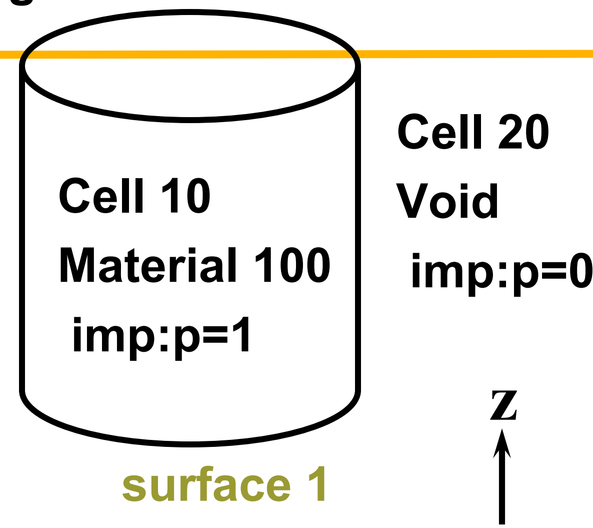
Energy	[ERG]	14.0 MeV
Position	[POS]	0.0 0.0 0.0
Direction	[VEC]	Isotropic
Time	[TME]	0.0
Particle Type [PAR]		neutrons if mode n, mode n p, mode n p e
		photons if mode p or mode p e
		electrons if mode e or mode e f

The mode data card is a listing of all particles to be used in the simulation.

Tc99m in lung -- using SDEF Sources

- Bare cylinder of (almost) ICRU lung, ρ = 1.06 g/cm3
- Tc99m emits one 0.14 MeV γ per decay
- Tc99m is not in geometry
- **Nuclide / Mass-fraction / ZAID**

Nuclide	Mass-fraction	ZAID
Hydrogen	.103	1001
Carbon	.105	6000
Nitrogen-14	.03	7014
Oxygen-16	.749	8016



(1) Create & edit file “source1"

(2) Use macrobodies, with center at (0.0, 0.0, 0.0) height = 10.0 rad=5.0

cylinder: RCC x0 y0 z0 Δx Δy Δz rad

(3) Add these data cards:

SDEF (**Add a point source of photons at center of cylinder**)
mode p $ for photon transport, mode p e for photon and electron transport
nps 100
print 110

Problem source1

```
Tc99m point source in lung
c  CELLS
10  100   -1.06      -1     $ lung
20    0               1     $ exterior

c SURFACES
1  RCC   0. 0. 0.   0. 0. 10.   5.0  $ center, heights, radius

c DATA
mode p                       $ or mode p e
imp:p   1  0
m100  1001 -0.103  6000 -0.105  7014 -0.03 $ Near ICRU lung
        8016 -0.749          $ Neg Fractions for mass fractions
sdef   pos 0.0 0.0 5.0      $ or x=0.0 y=0.0 z=5.0
       erg=0.14  par=p      $ 0.14 MeV, photons
nps 100                      $ run 100 source particles
print 110                    $ put print table 110 in output file
```

SI, SP, SB, and DS Cards

[source distribution cards] [dependent source card]

Usually, source variables are not single values.

The following cards are used in conjunction with the SDEF card to describe distributions in location, direction, energy, etc.

SI **information about the variable**
bins, discrete values, distribution numbers

SP **probability of choosing particular value**
true probabilities, built-in functions

SB **biased probabilities**

DS **dependent distribution**
values, distribution numbers

SI (source information) Card

FORM: SIn option entries

blank or

- **H** histogram bin boundaries
- **L** discrete values follow
- **A** points where probability density distribution is defined
- **S** distribution numbers follow

SI Card Examples

```
SDEF   ERG=D1
SI1    H  .01   .1   1.0   3.0   14.0     $  bins

SDEF   POS=D1
SI1   L   0.   0.   0.    10.   0.   0.    $  xyz values

SDEF   ERG=D1
SI1   S   3   4   5     $  other distribution#
```

Pb Shield -- SDEF Volumetric Source

Visual Editor Source plots

1) Copy file shield to source2

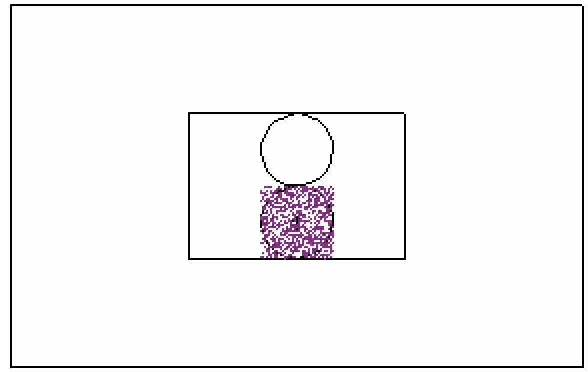
2) Change SDEF card to be a spatial distribution in xyz to surround bottom sphere.

3) Run. Look at starting cell locations in Table 110

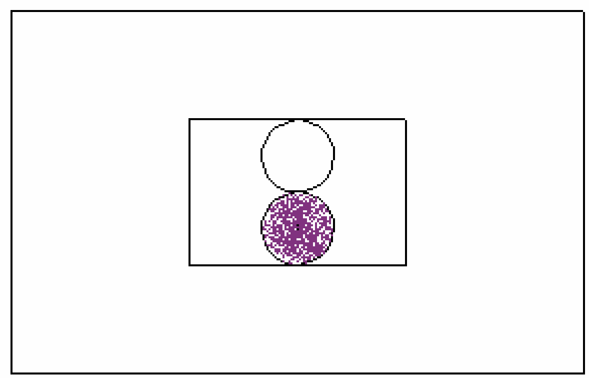
4) Add "cell=40" to sdef card for cell acceptance – source particles will only be in bottom sphere

5) Run. Look at starting cell locations in Table 110

Without SDEF Cell=40



With SDEF Cell=40



Problem source2


```
Tc99m [monoenergetic] photon spherical source [XYZ+rejection] in Pb shield
10   100    -11.4       -4 3        $ Lead Shield
20     0                -3 1 2      $ Void
30   200    -1.12       -2          $ Poly sphere, center at y=2
40   200    -1.12       -1          $ Poly sphere, center at origin
50     0                 4          $ Void, Exterior

1 SO                    1.0                 $ Sphere at origin with 1 cm rad
2 S    0.0  2.0  0.0    1.0                 $ Sphere at 0.,2.,0. with 1. cm rad
3 RCC    0. -1. 0.      0.   4. 0.     3.0 $ Right circular cylinder
4 RCC    0. -4. 0.      0. 10. 0.      8.0 $ Right circular cylinder

SDEF  X=D10 Y=D20 Z=D30     $ Source position @ X=0, Z=0, dist 20 for y
      erg=0.14              $ 0.14 MeV particles
      cell=40               $ accept point if in cell 40, otherwise reject
SI10  H  -1.0   1.0         $ Dist 10 has 1 bins, and -1 cm to 1 cm.
SP10  0.0   1.0             $ Probability below -1.0 cm is 0, -1 to 1 is 1.
SI20  H  -1.0   1.0         $ Even though the same as distribution 10,
SP20  0.0   1.0             $   these cards must be repeated, since
SI30  H  -1.0   1.0         $   each source variable must have a unique
SP30  0.0   1.0             $   distribution
imp:p   1 1 1 1 0
m100  82207 1.0                                         $ Lead Shield
m200  1000 -0.09677   6000 -0.38710   8000 -0.51613     $ Polyethylene
nps 1000
mode p
print 110
```

Dependant Source Distributions

- **Want to make the energy emitted a function of location?**

1) **Use FUNCTION of preceding Source Variable on SDEF card**

- **Example: SDEF Y=D20 ERG = F Y = D45**

2) Change its source information card (SI) to DS (dependant source) card

3) Remove SP card for the dependant source, since the probability of something is now correlated to the preceding source variable.

4) Must match number of selections on SI and DS cards

Dependant γ sources in Pb Shield

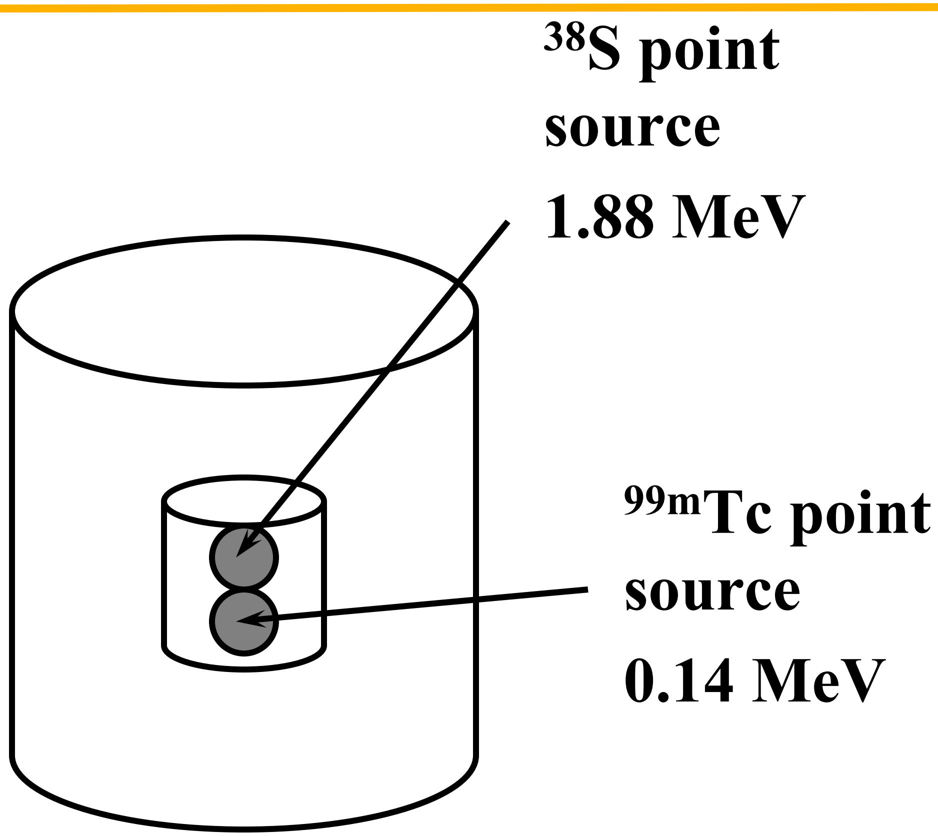
1) Copy shield to source4

2) Delete the SDEF, SI and SP cards.

3) Create a new SDEF:

2 point sources, each in the middle of the two spheres.

- Make the y=2 point emit 1.88 MeV photons (^{38}S) and the y=0 point emit 0.14 MeV photons (^{99m}Tc)
- Hint: Y should be a distribution with two discrete values.
- Hint: ERG is a dependant distribution of Y, and has two discrete values



Example source4


```
Two point gamma sources: Tc99m bottom, S38 top.
10  100   -11.4      -4 3        imp:p=1    $ Lead Shield
20    0              -3 1 2      imp:p=1    $ Void
30  200   -1.12      -1          imp:p=1    $ Poly sphere at origin
40  200   -1.12      -2          imp:p=1    $ Poly sphere at y=3
50    0               4          imp:p=0    $ Exterior

1 SO                 1.0                  $ Sphere at origin with 1 cm rad
2 S    0.0  2.0  0.0   1.0                  $ Sphere at 0.,2.,0. with 1. cm rad
3 RCC    0. -1. 0.     0.  4. 0.     3.0 $ Right circular cylinder
4 RCC    0. -4. 0.     0. 10. 0.     8.0 $ Right circular cylinder

sdef  x=0.0  y=D20  z=0.0    $ Source position @ X=0, Z=0, dist 20 for y
      erg=FY=D45             $ Source distribution 45 in energy
c sdef pos=D20 erg=FPOS=D45            $ Alternative way based on POS
c si20 L 0.0 0.0 0.0   0.0 2.0 0.0    $ Alternative based on POS, same sp20
si20 L 0.0    2.0          $ Two discrete values (L), not a line source
sp20   1.0    1.0          $ Equally probable
ds45 L 0.14   1.88    $ 0.14 MeV corresponds to 0.0 cm, 1.88 MeV to 2.0 cm
m100  82207 1.0                                       $ Lead Shield
m200  1000 -0.09677   6000 -0.38710   8000 -0.51613   $ Polyethylene
nps 1000
mode p
print 110
```

MCNP MP Tallies

On Electron – Photon

Energy Deposition

H. Grady Hughes

X-3 MCC

LA-UR-07-2996

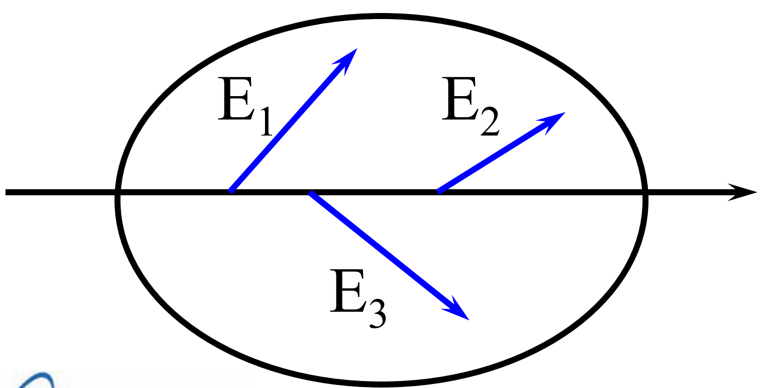
Abstract

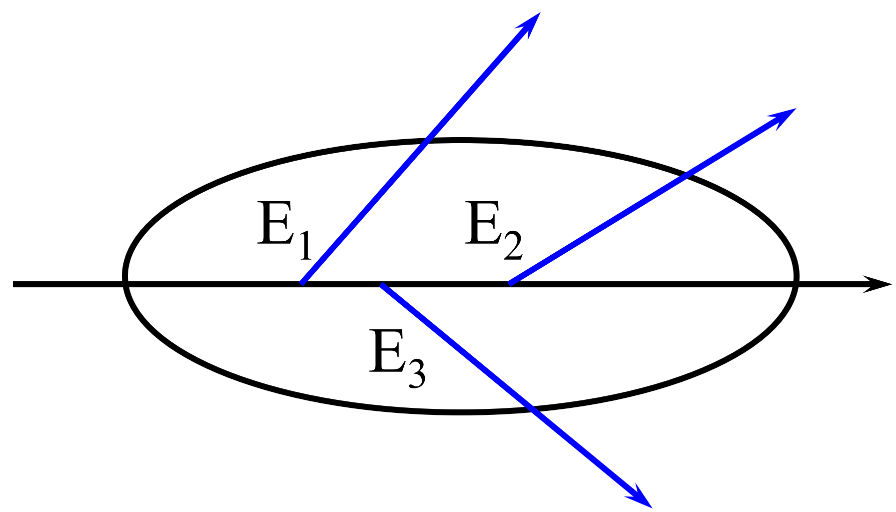
The presence in MCNP of two different tallies (F6:P and ∗F8:P,E) capable of estimating energy deposition in coupled photon/electron transport problems often causes some confusion. These slides provide heuristic descriptions of the two methods and thereby clarify the limitations on the validity of the F6 tally. An illustrative example is also given.

Energy deposition by F6 tally

mode	p		
f6:p	7	13	…
C	Typically no energy bins.		

This tally estimates energy deposition by integrating the track-length photon flux weighted by photon heating numbers. These numbers represent the average kinetic energy given to electrons along the photon path. Therefore, this tally is approximately valid only when most of the electrons are trapped in the tallied cells. If the cells are small (or dilute) enough that a significant amount of electron energy can escape, then the F6 tally will overestimate the energy deposition.

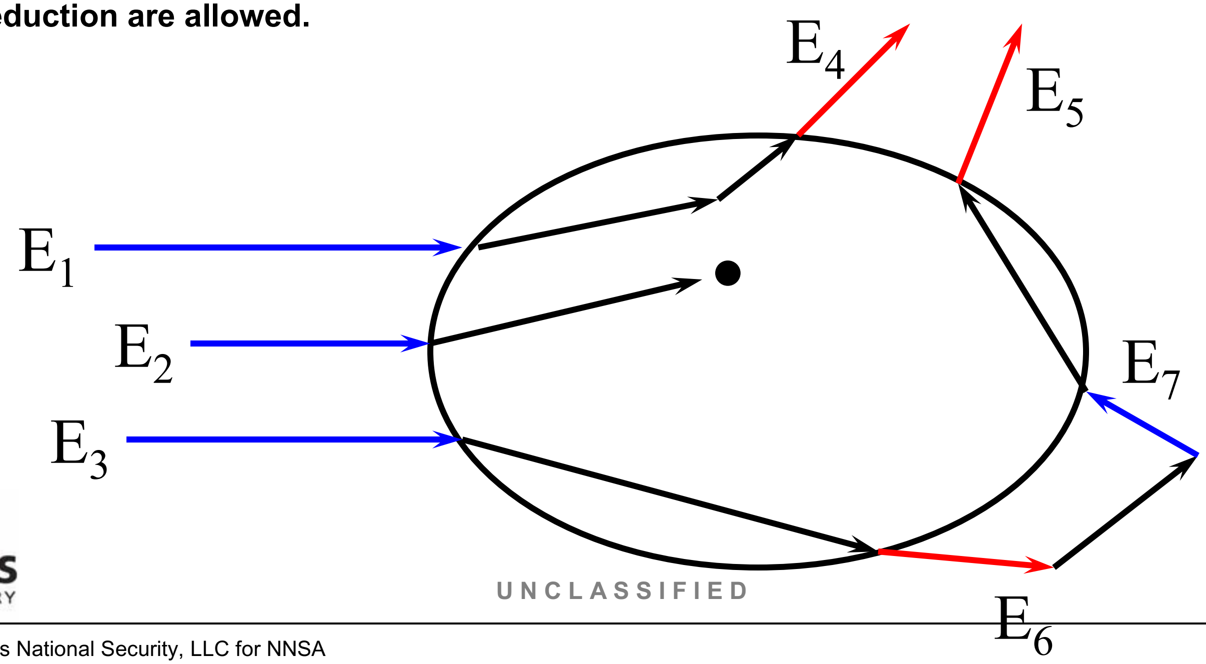




Energy deposition by ∗F8 tally

mode	**p**	**e**	
∗f8:p,e	**7**	**13**	**…**
C	**No energy bins.**		

This tally performs a detailed accounting of (energy entering a cell) minus (energy leaving a cell) for each history in a MODE P E problem. For example, DEPOSITION = $E_1+E_2+E_3-E_4-E_5-E_6+E_7$ for the three histories shown below. The tally is microscopically correct, except for the lack of correlation in the sampling of knock-on electrons or characteristic X-rays, which averages out over many histories. In contrast to the pulse-height tally, all forms of variance reduction are allowed.



An Example Problem

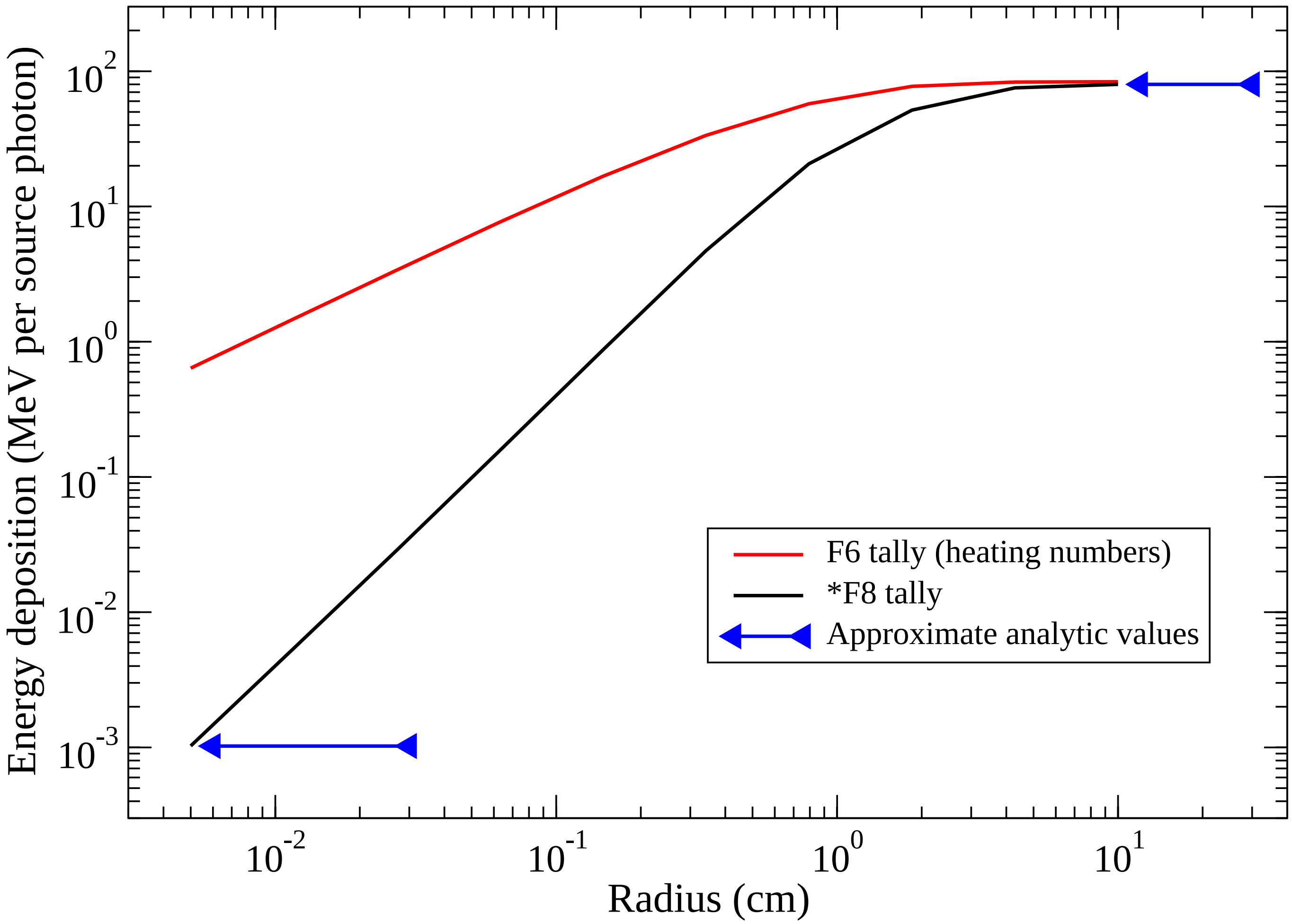
- 80-MeV photon point source at center of tungsten sphere
- 0.005 cm $\leq$ R $\leq$ 10 cm
- $\sigma_{total} \cong \sigma_{pair}$ = 25.03 barns
- N = 6.3218×10^{-2} nuclei/barn·cm
- ρ = 19.3 g/cm^3
- dE/dx(80 – 2mc^2) $\cong$ 1.342 MeV·cm^2/g

- $\therefore$ for R = 0.005 cm, $\Delta E \cong 1.025\times10^{-3}$ MeV per source photon.
- For R = 10 cm, $\Delta E \cong$ 80 MeV per source photon.

- Calculate energy deposition using F6:p and ∗F8:P,E tallies.

Comparison of F6 and *F8 Tallies

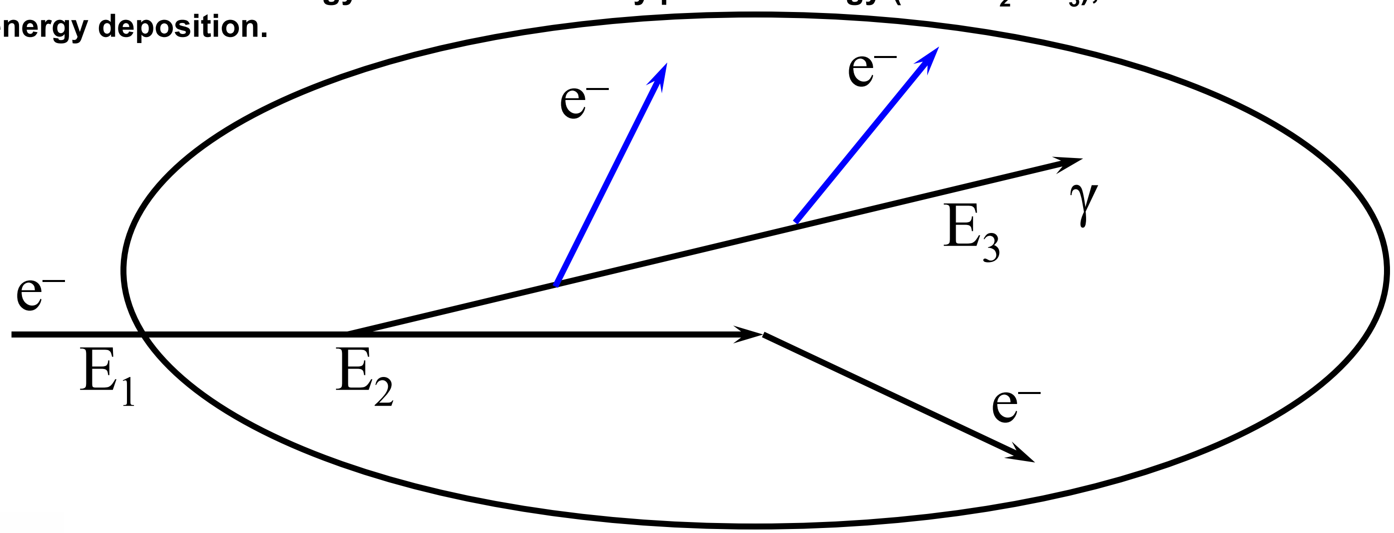
80-MeV Photons in Tungsten Sphere



When the F6 tally must not be used

mode	p	e
sdef	par = e	…
F6:p	7	$ Ignores initial electron energy loss and $E_e - E_p$

This photon tally ignores the electron energy loss prior to photon creation (here $E_1 - E_2$) and the difference between the electron energy and the secondary photon energy (here $E_2 - E_3$), and therefore underestimates the energy deposition.



Summary

mode	p	e	
*f8:p,e	7	13	…

This is the preferred method for MODE P E problems. All details of the transport are followed, and variance reduction is allowed.

mode		p	
f6:p	7	13	…

Track-length estimation originally developed for MODE P problems. It is valid only when electrons are mostly trapped in the cells where they are created.

mode		p	e
sdef		…	par = p
f6:p	7	13	…

This is allowed, but valid only when electrons are mostly trapped.

mode		p	e
sdef		…	par = e
f6:p	7	13	…

Allowed, but absolutely wrong!

MCNP Release
MCNP6/X Merger

MCNP Releases

- MCNP version 5.1.50 to be released to RSICC October 2007
- ~ 1-2 Months for RSICC V&V, then release to US users
- New Release should contain updated MCNP5, MCNPX, Nuclear Data
- Will cost $

- MCNP6 and MCNPX Merger Already underway for last year
- Spent ~2.5 Full Time Employees Already, Projected another 2-3
- Aka 2-3 million dollars for merger
- Merged code already tracks all particles through geometry
- Currently working on making sure physics interactions is correct.
- Dec 2007 beta release to users at LANL for testing

MCNP
Misc Topics & Reference

Misc MP Issues

- **S(α,β) neutron scattering treatment**
- **Benchmarking Studies**
  - Computing Radiation Dosimetry – CRD 2002, Sacavem, Portugal June 22-23 2002 (published by OECD)
  - QUADOS (EU intercomparison) Bologna, Italy July 14-16 2003 http://www.nea.fr/download/quados/quados.html
  - EURADOS & CONRAD (EU intercomparison) Deadline: Sept 2006 http://www.eurados.org/
  - ANS: Computational Medical Physics Working Group http://cmpwg.ans.org/
- **MCNP Help & Obtaining MCNP**
- **MCNP/X 2007 & 2008 Classes**

Neutron Scattering Treatment

- Accounts for molecular effects on target nucleus velocity for low energy (few eV) n scattering.
- Usually low Z, varies with molecule

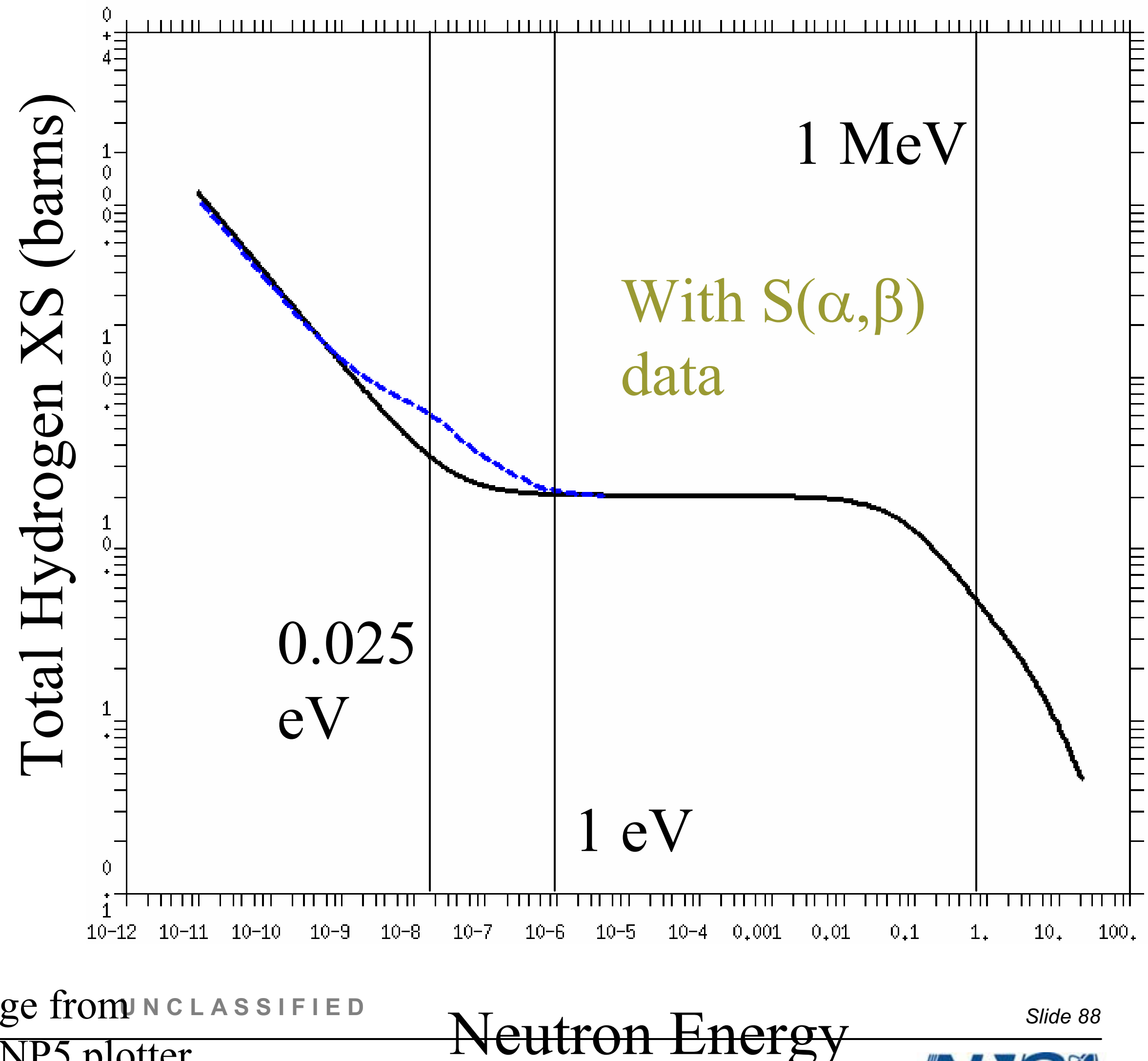
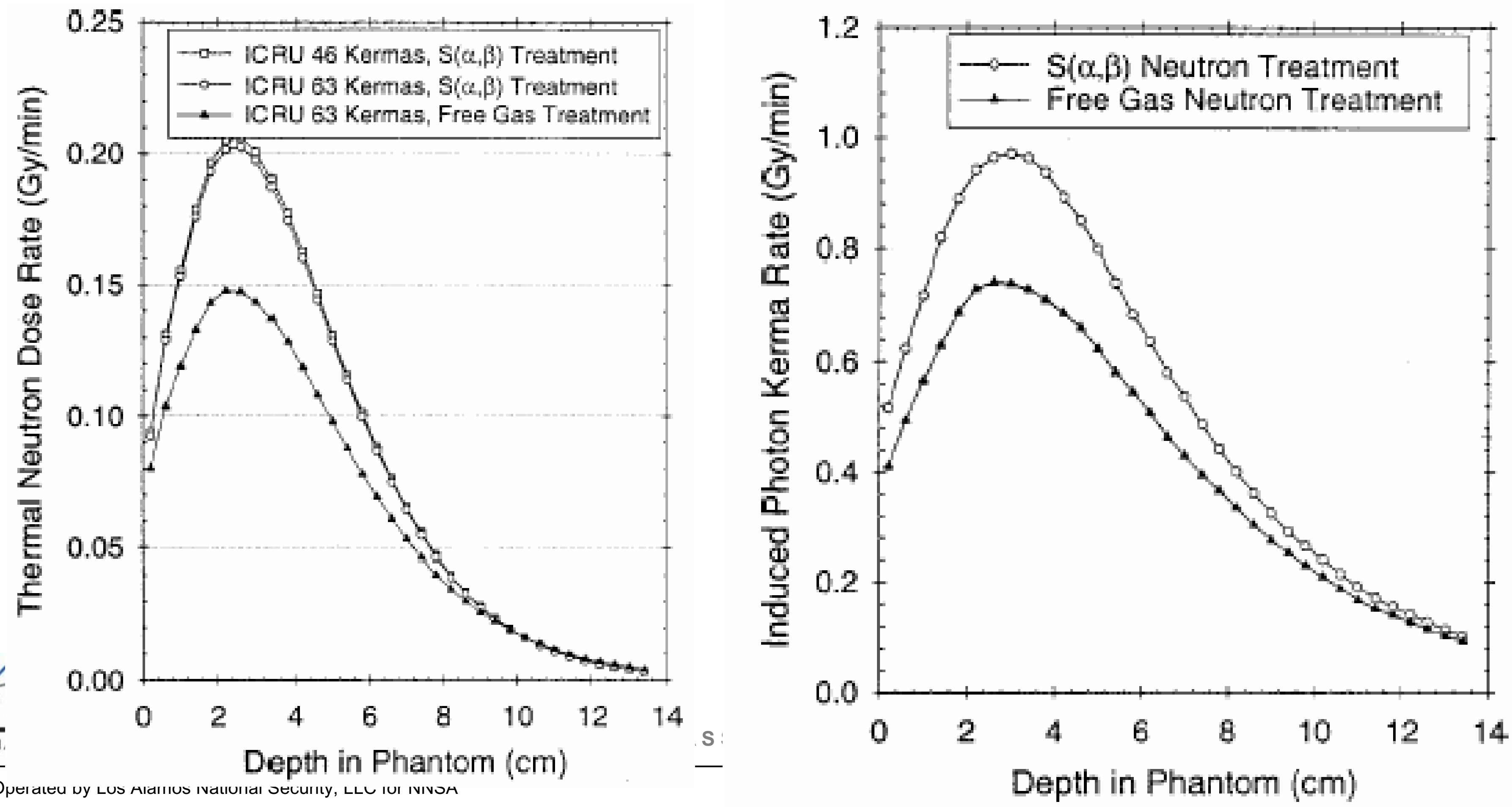


Image from MCNP5 plotter

Neutron Scattering Treatment

- **Use can cause significant differences.**

Goorley T, et. al. Med. Phys. 29 (2) 2002. pp. 145-156.



Verification & Validation

- **Electron Benchmarks - in resource section**
- **Computing Radiation Dosimetry - CRD**
- **QUADOS Code Comparison**
- **EURADOS - CONRAD Code Comparison**
- **ANS: Computational Medical Physics Working Group**
  - http://cmpwg.ans.org/
  - Additional Presentations
  - Code comparison effort

QUADOS

- **Quality Assurance of Computational Tools for Dosimetry**
- **Results presented June 14-16, 2004 Italy**
- **http://www.nea.fr/download/**quados/quados**.html**
- **8 Case Studies, some had 10+ participants**
- **Used MCNP5 for 6 cases, most good agreement**
- **Book of proceedings FREE! lrp@bologna.enea.it**

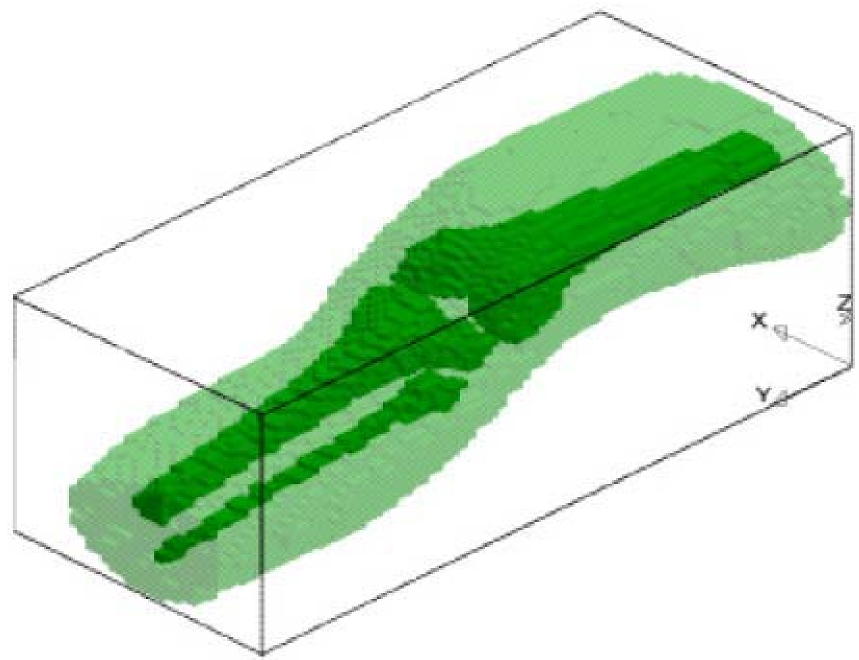
QUADOS

- **Brachytherapy – ^{192}Ir γ, dose distribution in H2O**
- **Endovascular – ^{32}P β-, dose in vessel wall**
- **Proton Therapy of Eye – 50 MeV p, depth dose**
- **TLD-Albedo Response – n + γ, 4 element TLD**
- **Phantom Backscatter – X ray ISO beams, slab**
- **Environmental Scatter – ^{252}Cf n, concrete room**
- **HPGe Detector – 15 keV – 1 MeV γ, pulse height**
- **Consistency check device – ^{241}Am-Be, ^{3}He detector**

- **Input decks available w/ MCNP5 1.40 Distribution**

EURADOS

- **European Radiation Dosimetry Group**
- **http://www.eurados.org/**
- **Active Code Comparison**
  - Monte Carlo modeling for in-vivo measurements of Americium in knee phantom
  - Deadline: November 2006
  - CONRAD - 4 Problems
  - Internal Dosimetry
  - Compex Rad Fields,
  - Medical Staff Dose
  - Computation Dosimetry
  - Results & uncertainties
  - Deadline: September 2006



Obtaining MCNP

- **Can be obtained from RSICC**
- **http://www-rsicc.ornl.gov/**
  - 2 DVD versions
    - Executables, Source and Full Manual – limited release
    - Executables, no source, and Vol I & II of Manual – broader release
- **All DVDs Contain**
  - MCNP5, MCNPX, and MCNP Data
  - MCNP5 executables for Linux, Mac, Windows
  - the latest data (pre ENDF/B-VII)
  - MCNPVisual Editor
  - Test Suite to ensure proper installation and compatibility
  - MCNP5 Manual and other documentation
  - Medical Physics Sample Problems

Help with MCNP

- **Read the manual**
- **User forum:** mcnp-forum@lanl.gov
- **X-3 (limited):** mcnp@lanl.gov
- **MCNP home page:**
  - **http://www-xdiv.lanl.gov/x5/MCNP/index.html**
- **RSICC e-notebook:**
  - **http://www-rsicc.ornl.gov/**
  - Go to eNotebooks tab

2007/8 MCNP Classes

- X-3:
- October 15-19, 2007: Introduction to MCNP - LANL
- January 7-19, 2008: Intermediate MCNPX – Las Vegas, NV
- February 4-8, 2008: Advanced MCNP5 - LANL
- April 7-10, 2008: Criticality Calculations with MCNP – LANL
- May 12-16, 2008: Intermediate MCNPX – Lisbon, Portugal
- June 2-6, 2008: Introduction to MCNP5 and MCNPX – LANL
- June 16-20, 2008: Introduction to MCNP5 and MCNPX – LANL

- HSR-4: Practical MCNP for the Health Physicist, Medical Physicist, and Radiological Engineer – LANL
- No Posted Dates: see http://drambuie.lanl.gov/~esh4/mcnp.htm

MCNP Next Generation of Capabilities

Next Generation of Capabilities?

- **In the ANS RPSD conference (Carlsbad, NM):**
  - Agreement of data and simulation < 3%.
  - Dose calculations ~ 2 mm tally grids or less

- **This will drive a new evolution in the codes.**

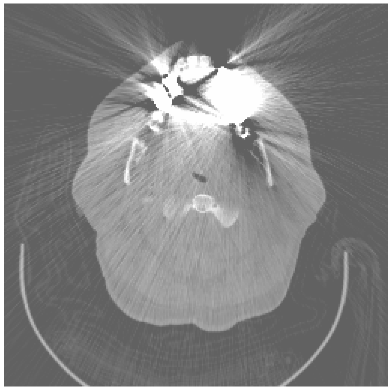
- **New physics processes that cause dose “blurring” on these scales will need to be added to get more accurate simulations.**

Medical Physics Brainstorming

- **Add into codes:**
  - Magnetic field (quadrapole) capabilities to model further upstream in beamline (bending magnets) to include slight beam spreading.
  - Better characteristic X-Ray production
  - Proton (& other heavy charged particles)
    - Proton recoil
    - Electron production from high energy protons as delta ray lengths exceed ~ few mm.
    - Inelastic collisions and subsequent gamma & conversion electrons
    - Very high fluxes: space charge effects

Medical Physics Brainstorming

- **Add into codes / develop methodology:**
  - Model CT scanner / MC simulation of CT images
    - Help create accurate geometric models when CT image is distorted.
    - Reconstruct Dose from CT imaging process:



  - Cross Section uncertainty / covariance
    - What is uncertainty in the dose due to uncertainty in the cross sections?

J J DeMarco et al. “A Monte Carlo based method to estimate radiation dose from multidetector CT (MDCT): cylindrical and anthropomorphic phantoms. Phys. Med. Biol. 50 (2005) 3989–4004

Additional References

- Electron Transport V&V papers
- Monte Carlo 2005 - Chattanooga
- MCNP V&V papers

STOP - Break

Electron Transport

- Gierga, DP, Adams KJ, Ballinger CT, Electron Transport using the macro Monte Carlo method for Medical Physics Applications, ANS Transactions 1997, vol 77, p. 356-7
- Gierga, DP, Adams KJ, Electron/Photon Verification Calculations Using MCNP4B. Los Alamos National Laboratory, LA-13440, 1999. 89 pages.
- Schaart, DR, Jansen JTM, Zoetelief J, de Leege, PFA, A Comparison of MCNP4C electron transport with ITS 3.0 and experiment at incident energies between 100 keV and 20 MeV: Influence of voxel size, substeps and energy indexing algorithm. Phys Med Biol, May 2002, vol 47 (9) p. 1459-84
- Chibani, O, Li, XA, Monte Carlo calculations in homogeneous media and at interfaces: A comparison between GEPTS, EGSnrc, MCNP and measurements. Medical Phys, May 2002, vol 29 (5), p. 835-47.

Monte Carlo 2005 MCNP Talks

- Mon 10:50 am Ballroom E - MCNP5 For Proton Radiography, H. Grady Hughes
- Tues 10:50 am Meeting Room 5 - Issues Related To The Use Of MCNP Code For An Extremely Large Voxel Model VIP-MAN, Tim Goorley
- Tues 3:30 Meeting Room 4 - Stochastic Geometry & HTGR Modeling with MCNP5, Forrest Brown, WR Martin, W Ji, J Conlin, JC Lee
- Wed 9:00 am Ballroom E - Monte Carlo Methods & MCNP5 Code Development, Forrest Brown
- Wed 9:25 am Meeting Room 6 - Analysis Of The Fourth Zeus Critical Experiment With MCNP5, Russell Mosteller
- Wed 10:50 am Meeting Room 5 - Comparison Of Phantom Models For External Dosimetry Computations, Richard Olsher

Voxel Model Talks at Monte Carlo 2005

papers available on conference CDROM

- Mon, 1:15 GSF Male And Female Adult Voxel Models Representing ICRP Reference Man By Keith Eckerman
- Mon, 1:45 Effective Dose Ratios For The Tomographic Max And Fax Phantoms By Richard Kramer
- Mon, 2:05 Reference Korean Human Models: Past, Present and Future By Choonsik Lee
- Mon, 2:25 The UF Family of Pediatric Tomographic Models By Wesley Bolch and Choonik Lee
- Mon, 2:45 Development And Anatomical Details Of Japanese Adult Male/ Female Voxel Models By Tomoaki Nagaoka
- Mon 3:25 Dose Calculation Using Japanese Voxel Phantoms For Diverse Exposures By Kimiaki Saito
- Mon 3:45 Stylized Versus Tomographic Models: An Experience On Anatomical Modeling At RPI By X. George Xu
- Mon 4:05 Use Of MCNP With Voxel-Based Image Data For Internal Dosimetry Applications By Michael Stabin
- Mon 4:45 Application Of Voxel Phantoms For Internal Dosimetry At IRSN Using A Dedicated Computational Tool By Isabelle Aubineay-Laniece
- Tues 10:45 Issues Related To The Use Of MCNP Code For An Extremely Large Voxel Model VIP-MAN By Tim Goorley
- Tue 2:40 Conversion Of Combinatorial Geometry To Voxel Based Geometry In Moritz By Kenneth Van Riper

Additional References

- Goorley T, Kiger WS III, Zamenhof RG. Reference Dosimetry Calculations for Neutron Capture Therapy with Comparison of Analytical and Voxel Models. Med. Phys. 29 (2) 2002. pp. 145-156.
- Goorley, T. "MCNP5 Tally Enhancements for Lattices (aka Lattice Speed Tally Patch)," Los Alamos National Laboratory report LA-UR-04-3400 (June 2004).
- J. H. Hubbell and S. M. Seltzer, ''Tables of x-ray mass attenuation coefficients and mass energy–absorption coefficients,'' http:/physics.nist.gov/ xaamdi, National Institute of Standards and Technology, Gaithersburg, MD, 1997.
- Hughes, H. Grady , "Improved Logic for Sampling Landau Straggling in MCNP5", Submitted to M&C 2005, ANS Mathematics and Computation Topical Meeting, Avignon, France, Sept 12-15, 2005.
- ICRU 46, ''Photon, electron, proton, and neutron interaction data for body tissues,'' International Commission on Radiation Units and Measurements, Bethesda, MD, 1992.
- Kiger WSIII, Hochberg HK, Albritton JR, Goorley T, "Performance Enhancements of MCNP4B, MCNP5 and MCNPX for Monte Carlo Radiotherapy Planning Calculations in Lattice Geometries", 11th International Symposia on Neutron Capture Therapy. Boston, USA, Oct 11-15, 2004.

Additional References

- Borisov, N; Franck, D; de Carlan, L; Laval, L. A new graphical user interface for fast construction of computation phantoms and MCNP calculations: Application to calibration of in vivo measurement systems. Health Physics; Aug. 2002; 83(2) p.272-9
- Franck, D; Borissov, N; de Carlan, L; Pierrat, N; Genicot, JL; Etherington, G. Application of Monte Carlo calculations to calibration of anthropomorphic phantoms used for activity assessment of actinides in lungs. Radiation Protection Dosimetry; 2003; vol.105, no.1-4, p.403-8 Conference: Internal Dosimetry of Radionuclides. Occupational, Public and Medical Exposure, 9-12 Sept. 2002, Oxford, UK
- Wyatt, MS, Miller, LF, Implementation of a Methodology for Converting CT Images to MCNP Input. 2004 ANS Winter Meeting,November 14 – 18, 2004, Washington, DC.