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LA-UR-05-2755

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Title: USING MCNP5 for Medical Physics Applications

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HSR-4, LANL, USA

Submitted to: American Nuclear Society Topical Meeting - Monte Carlo 2005
Chatanooga, TN, USA
April 17-21, 2005

SAVE PRINT CLEAR FORM

Los Alamos NATIONAL LABORATORY

Diagnostics Applications Group (X-5)
Health Physics Measurements (HSR-4)

LA-UR-05-2755

Using MCNP5 for **Medical Physics Applications**

Tim Goorley, X-5

Dick Olsher, HSR-4

Los Alamos National Laboratory

Schedule: 1 pm – 4:30 pm

1.	What can MCNP do?	TG – 15 min
2.	Overview of new MCNP5 features	TG – 30 min
3.	Geometries and Modeling	TG – 30 min
4.	Break	15 min
5.	Sources	DO – 45 min
6.	Tallies	DO – 45 min
7.	Misc (n scattering, VR, Benchmark)	TG – 30 min
8.	Additional References	

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.

MCNP5 New Features

• Mesh Tallies	1^{st} Release
• Radiography Tallies	1^{st} Release
• Photon Doppler Broadening	1^{st} Release
• More Detectors & Tallies	2^{nd} Release
• >2.1 Billion Histories & RAND #	3^{rd} Release
• Lattice Tally Enhancements	3^{rd} Release

- Mesh Tally Improvements (RSICC_1.40)
- Electron Improvements (RSICC_1.40)
- Stochastic Geometry (RSICC_1.40)
- Large Lattice Improvements
- FUTURE WORK for MCNP5 Teaser

Mesh Tallies

- Original release in MCNP5_RSICC_1.14
- 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.
- Can be used with TR cards (transformation).
- Particles must track through mesh to tally.

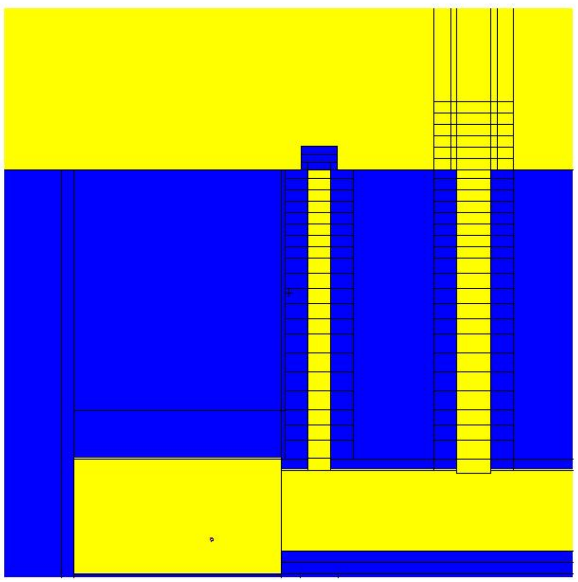
Mesh Tallies

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

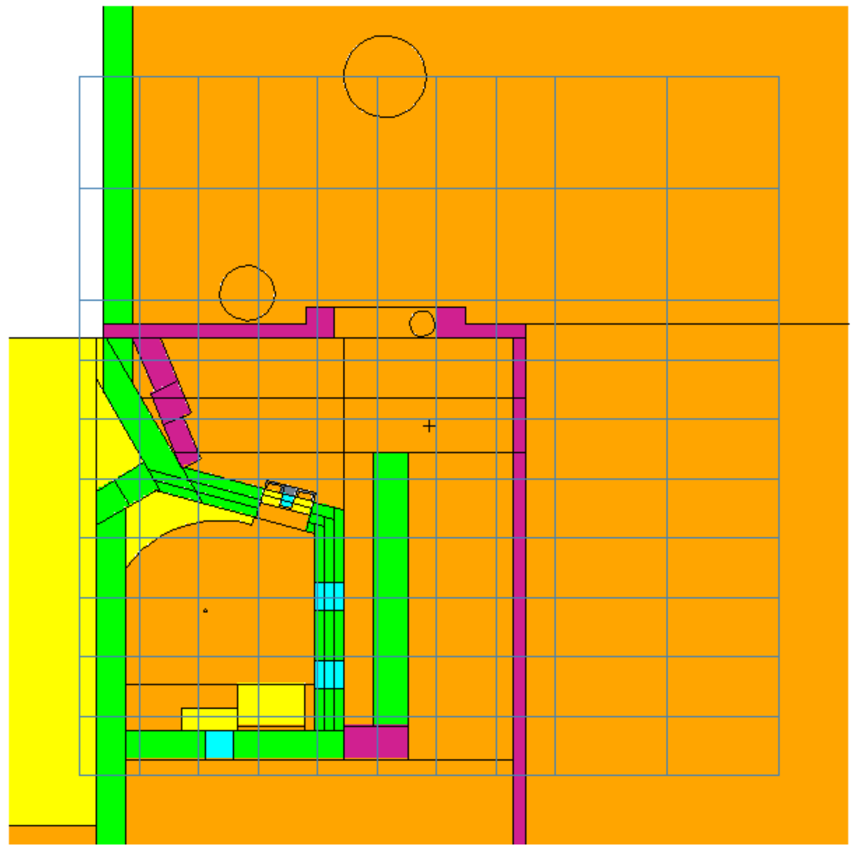
Geometry

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

Picture from Sabrina

Picture generated with results from MCNP calculation.

Simulated Radiograph

1 M pixels

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.

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: 2^{nd} 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:

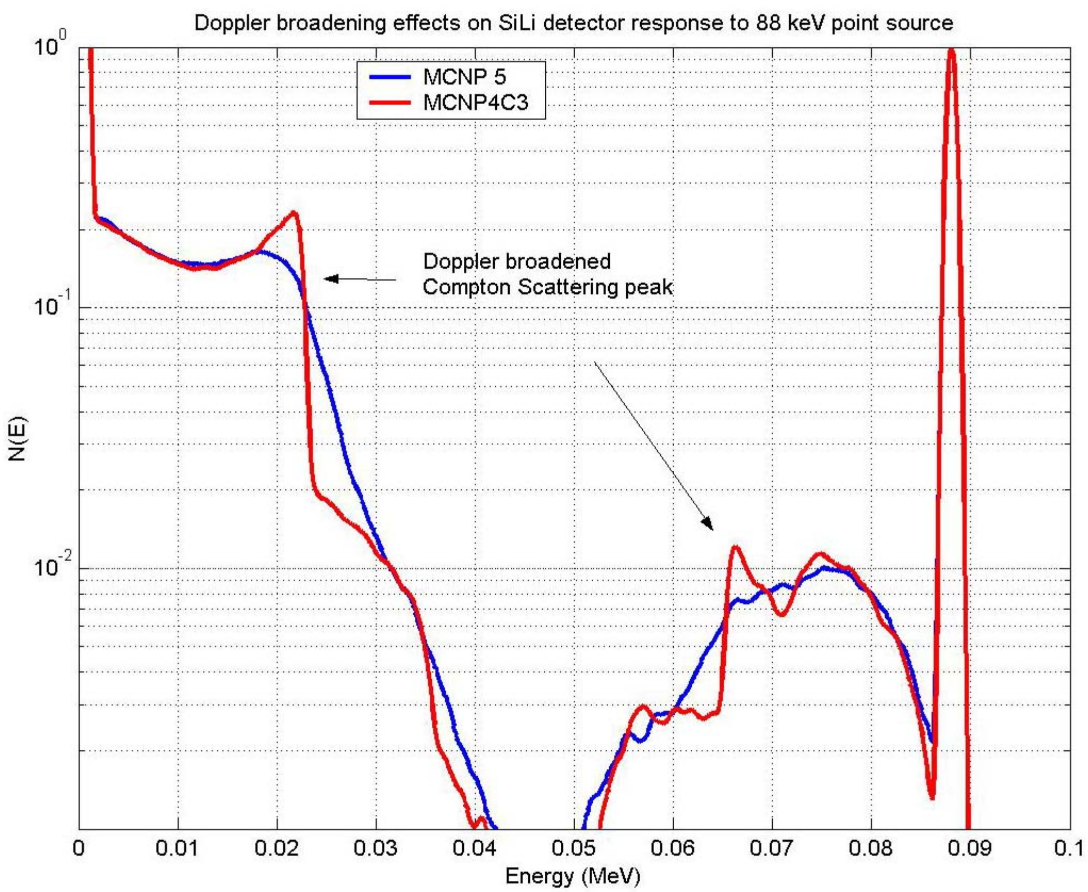
```
fir5:p 0 0 15. 0 0 0 -1000. 0 1e20 0
fs5 -55.0 999i 50.0
c5 -30.0 999i 30.0
notrn
talnp
```

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 : $\sim 10^{14}$ New Period: $\sim 10^{19}$

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, 1^{st} 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 appropraite.
- “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,
WARNING: # of histories does not correlate to simulated source strength!

MCNP5 Mesh Tally Plotting

- **Built-in plotter now plots mesh tally results on top of geometry outline**

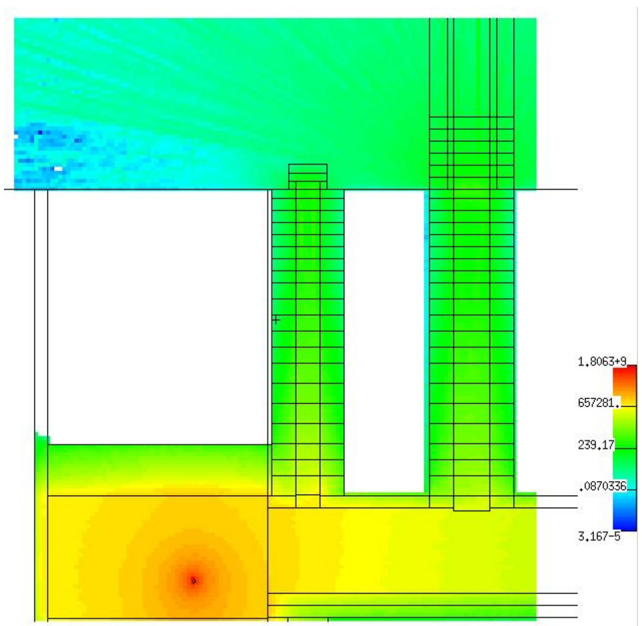
~Summer 2005

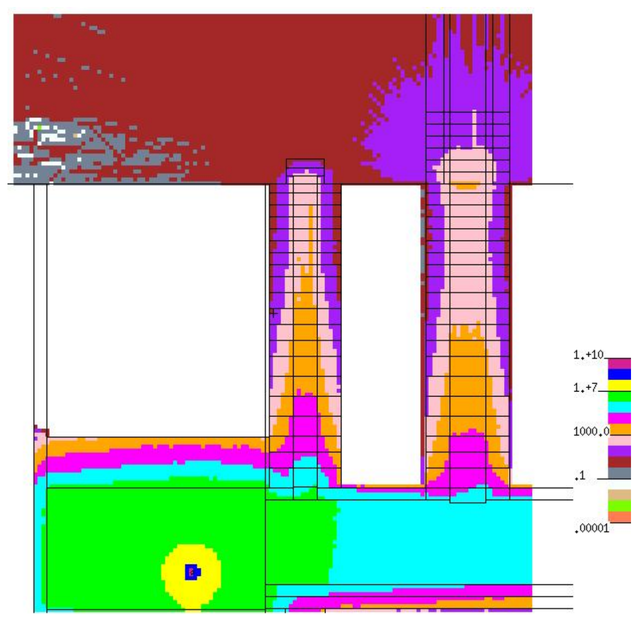
Proton Storage Ring

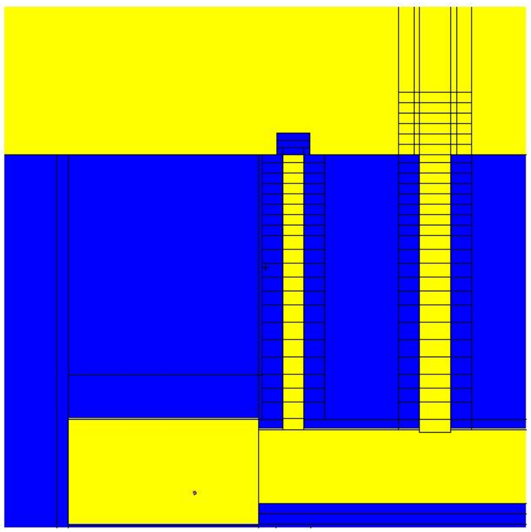
at LANSCE accelerator

Dose rate calculation for cable penetrations

Images from MCNP5 plotter





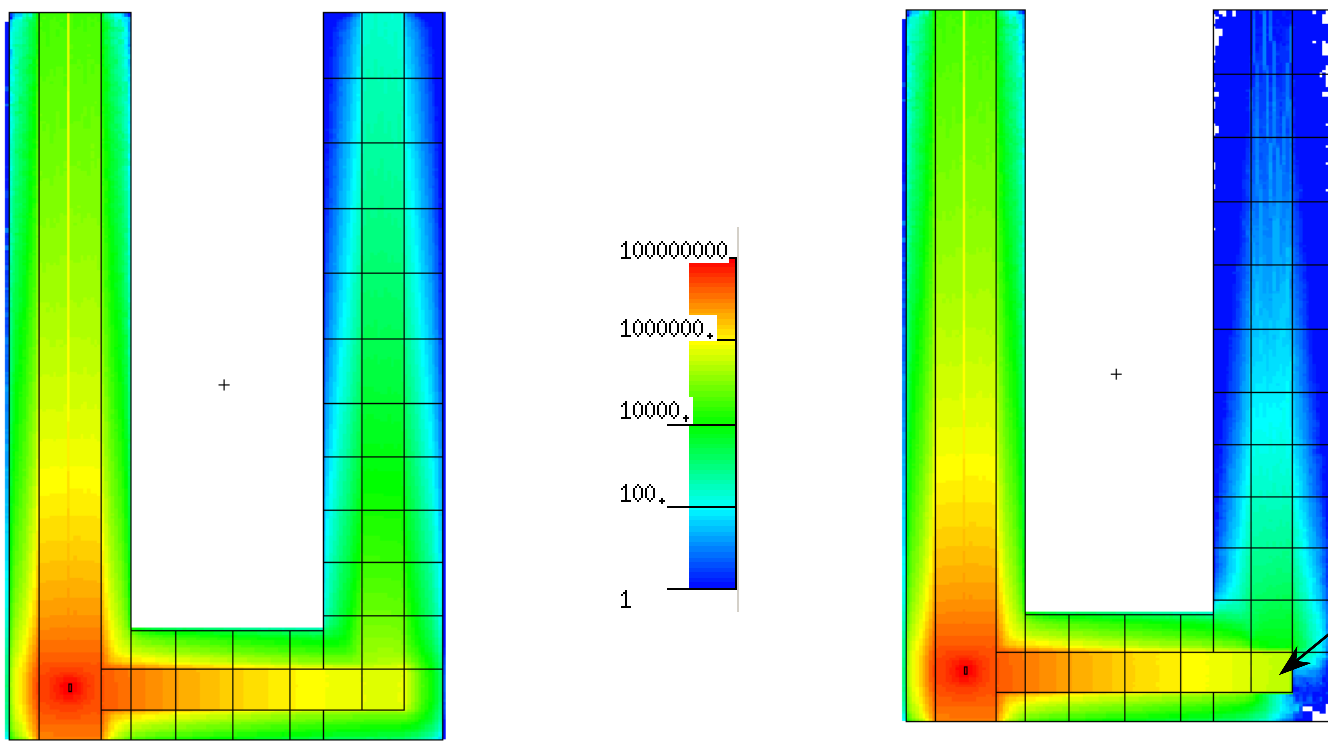


MCNP5 Mesh Tally Plotting

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

~Summer 2005

- **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**



Images from MCNP5 plotter

Anti-flagged cell

MCNP5 Mesh Tally Plotting

~Summer 2005

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

2000 x 1100 x 1 mesh

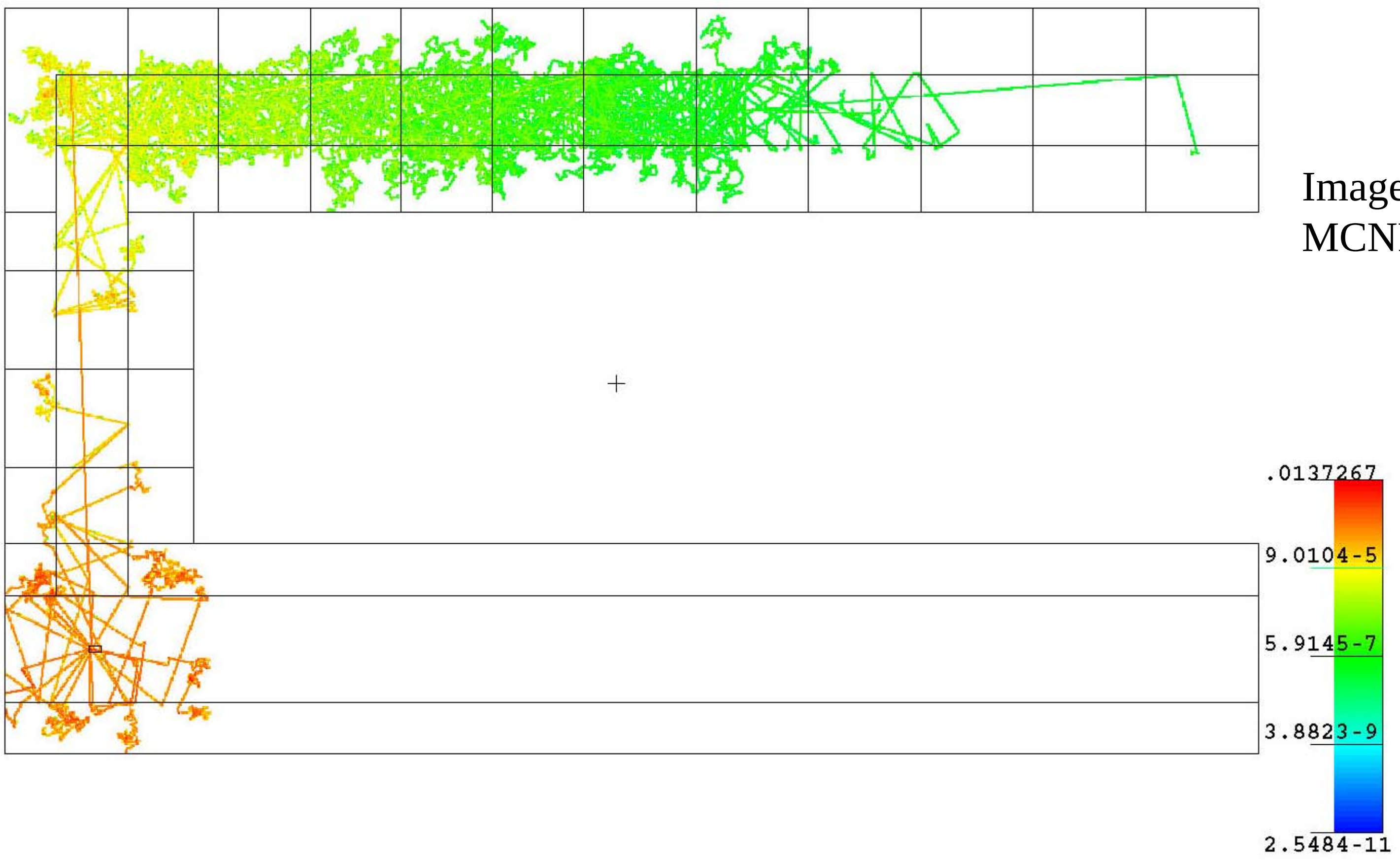


Image from MCNP5 plotter

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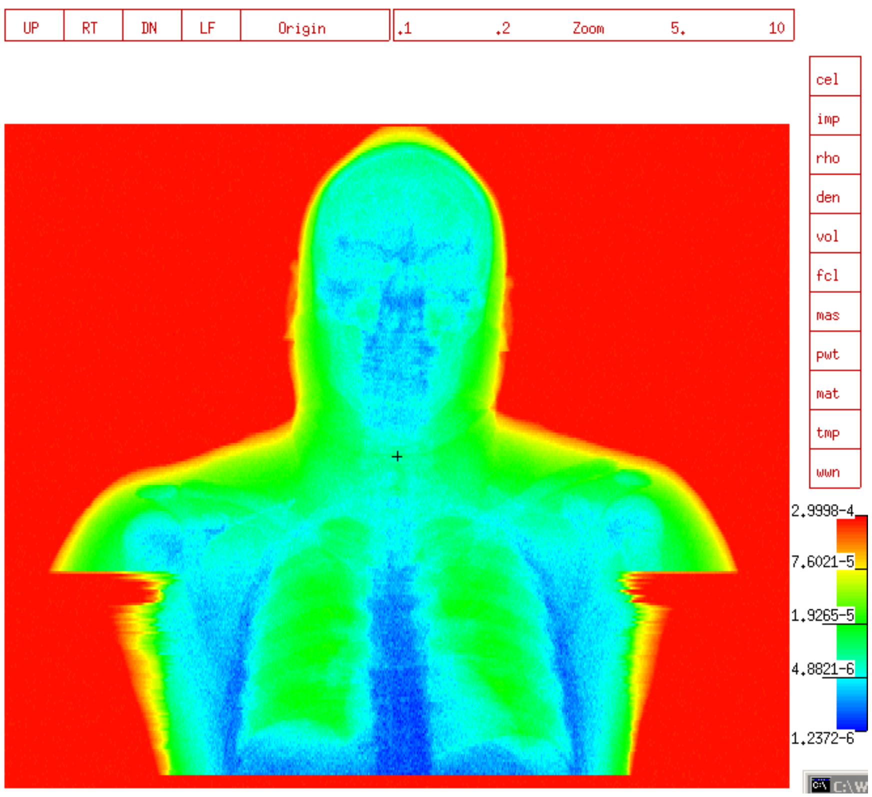
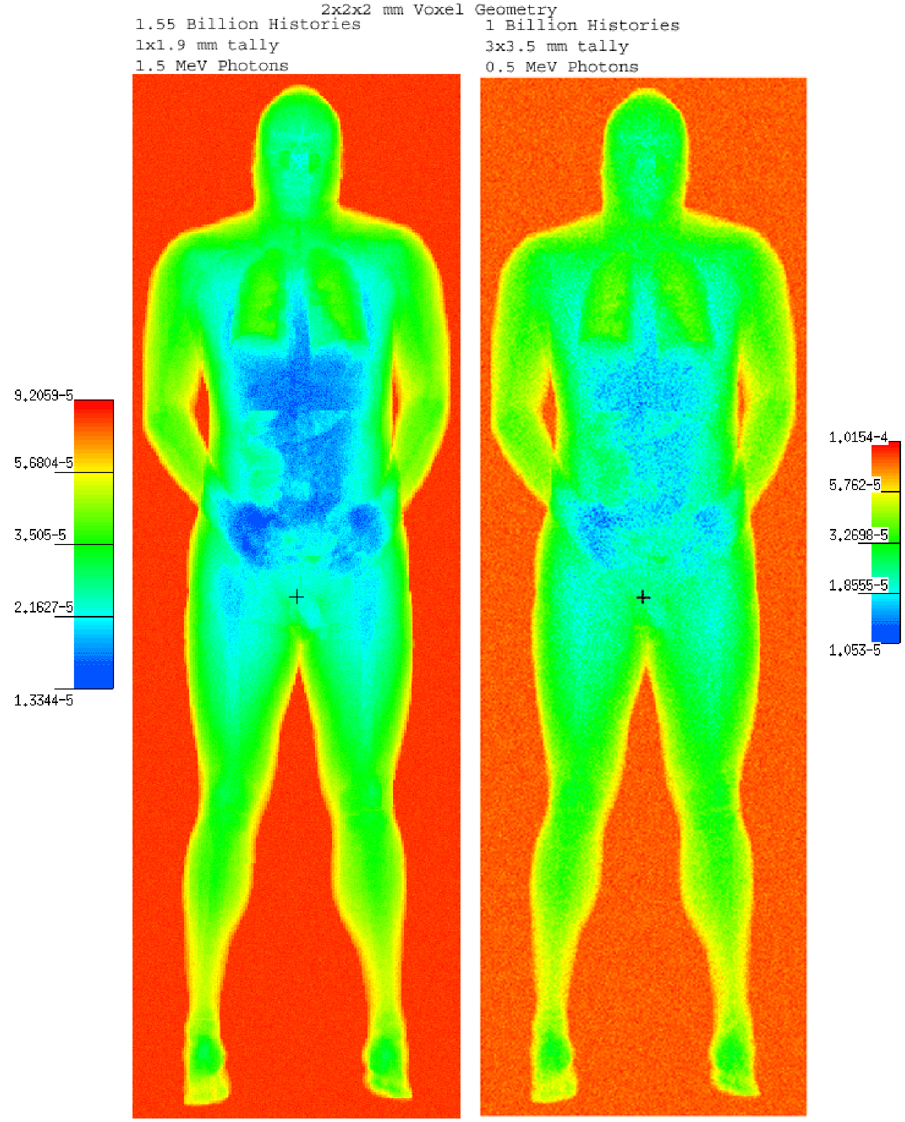
Health Physics Measurements (HSR-4)

MCNP5 Mesh Tally Plotting ~Summer 2005





Radiographs of VIPMan model,

1x1x1 mm voxels (above),

2x2x2 mm voxels (right)

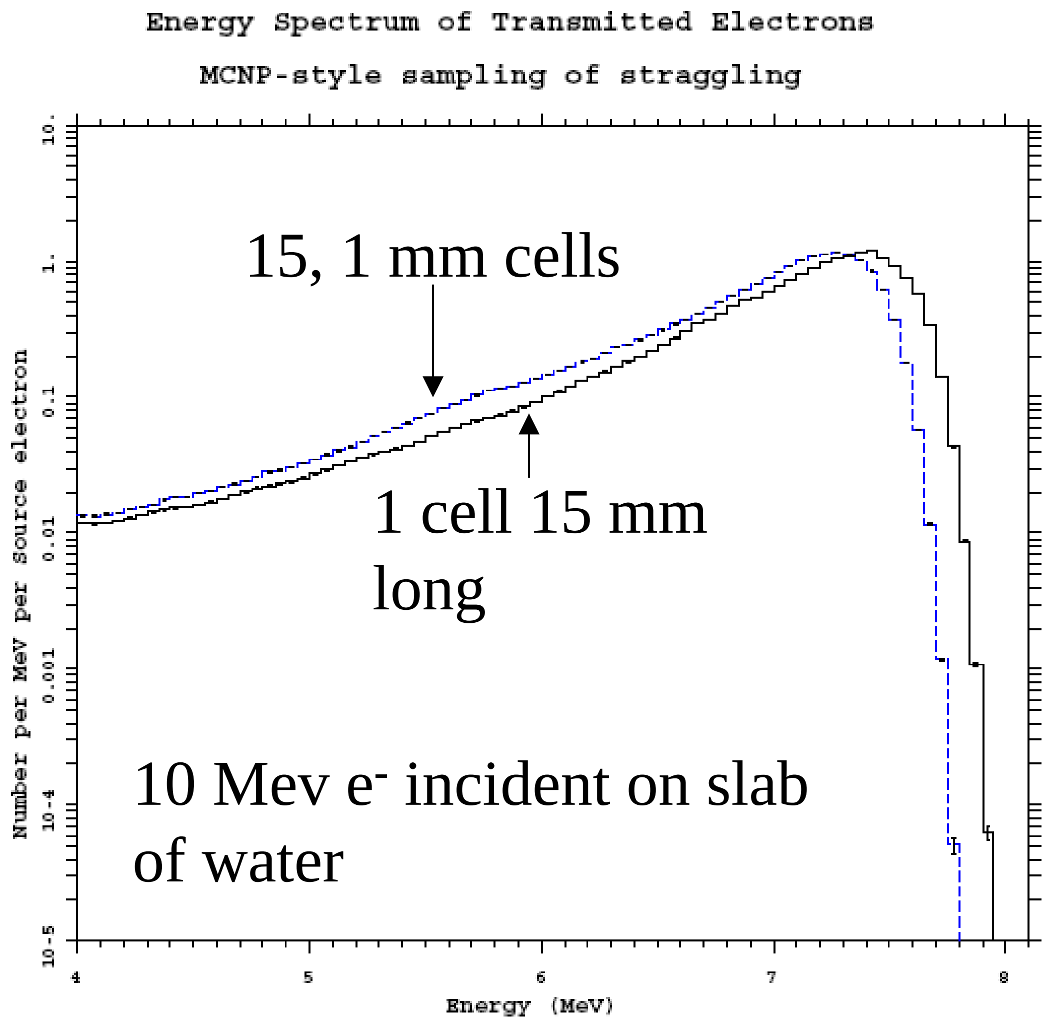
Images from MCNP5 plotter

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

Electron Improvements

~Summer 2005

- Positron Source
- 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



Hughes M&C 2005 Conference Paper

Stochastic Geometry

~Summer 2005

- On-the-fly random translations of embedded universes in lattice
- Developed for pebble bed reactors.
- Potential for medical physics applications?
  - Alveoli
  - Sinuses
  - Bone marrow

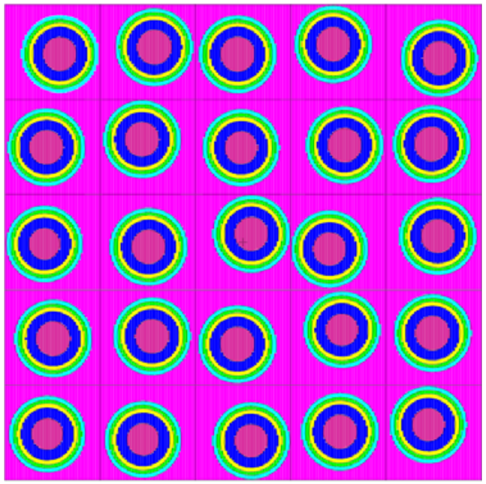


Image of the stochastic geometry of fuel kernels from MCNP5 plotter

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

Monte Carlo 2005 Plenary talk by Forrest Brown, Wed am

"**Monte Carlo Methods & MCNP Code Development**"

Los Alamos NATIONAL LABORATORY
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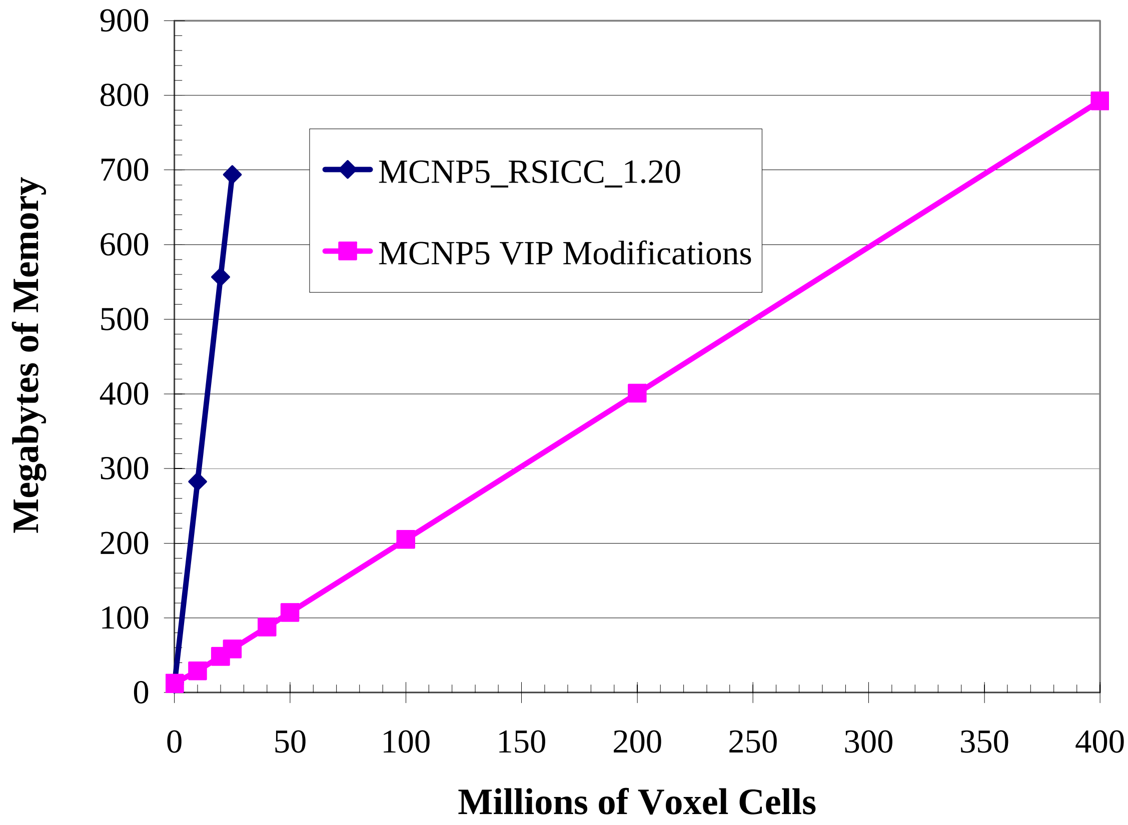
Health Physics Measurements (HSR-4)

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)

~Summer 2005?



Goorley, Tu 10:50 am, "Issues Related to the use of MCNP code for an Extremely Large Voxel Model VIP-MAN"

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Health Physics Measurements (HSR-4)

FUTURE WORK for MCNP5 Teaser

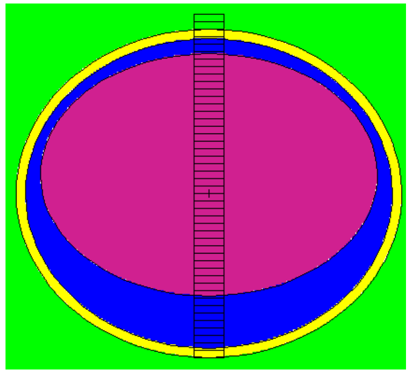
- Proton transport
  - Continuous-energy physics up to 50 GeV
  - Direct tracking through magnetic fields
  - COSY-map tracking through magnetic fields
- Many additional particle types
- ENDF/B-VII (Data Team)
- Improved electron transport
- Automated variance reduction, using deterministic adjoint
- Continuously varying tallies

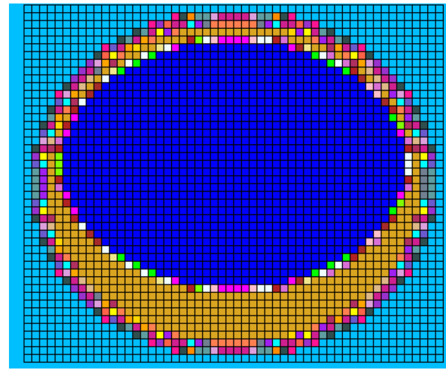
Monte Carlo 2005 Plenary talk by Forrest Brown, Wed am

"**Monte Carlo Methods & MCNP Code Development**"

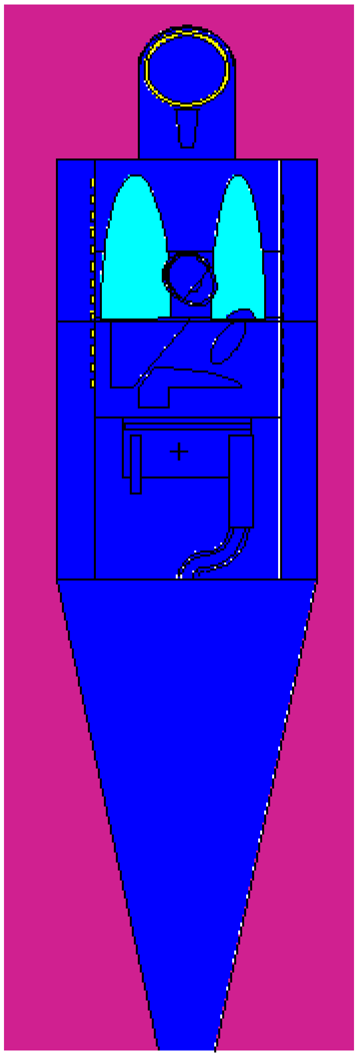
Diagnostics Applications Group (X-5)

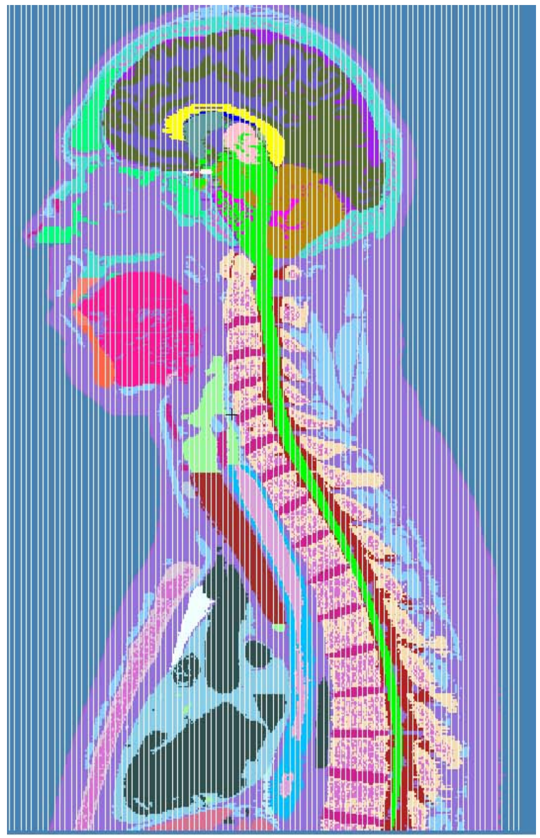
Health Physics Measurements (HSR-4)

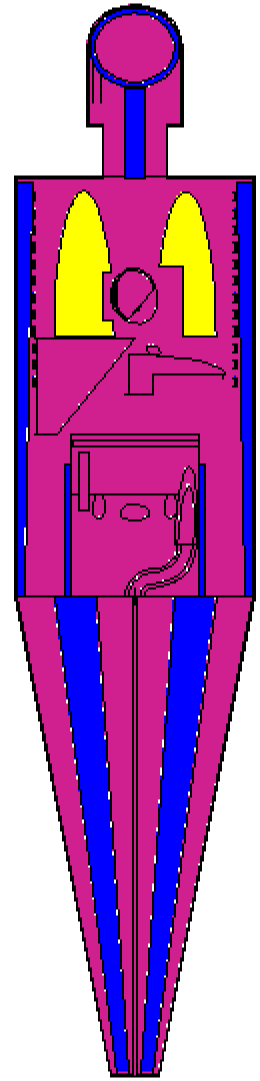




Geometries & Modeling

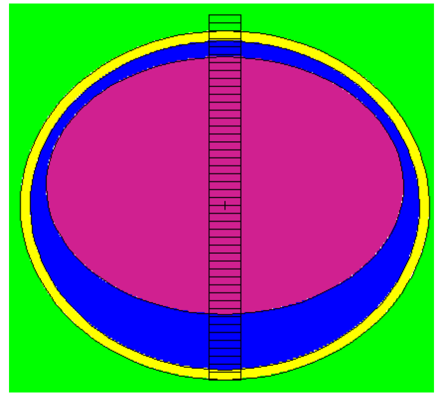


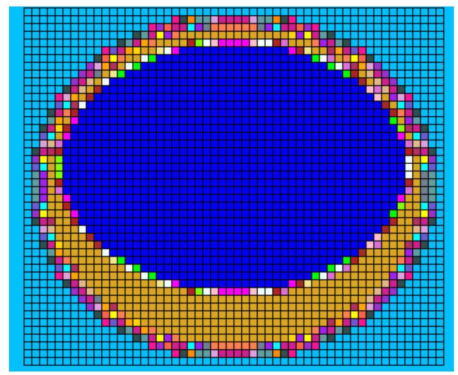




Geometries and Modeling

- 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.

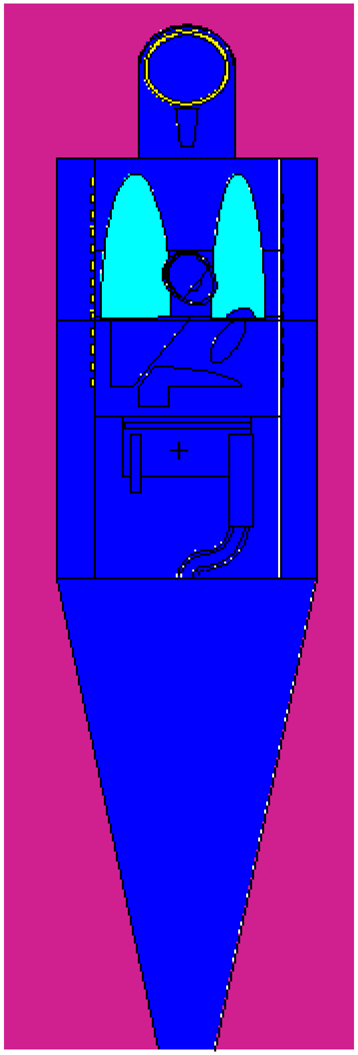
Analytical Models

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

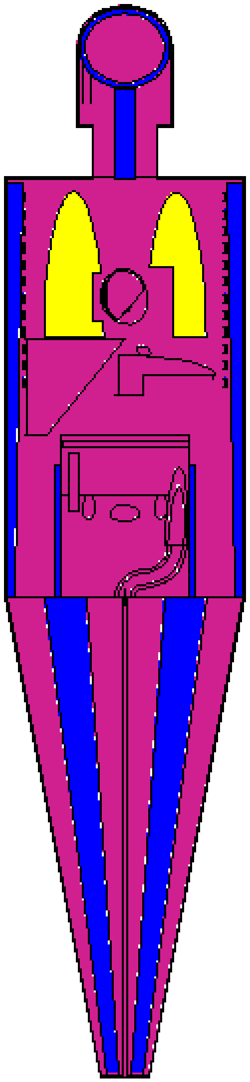
Analytical Models

Geometry plots from MCNP5 plotter

Observe differences in organs and materials.



Yanch - MIT



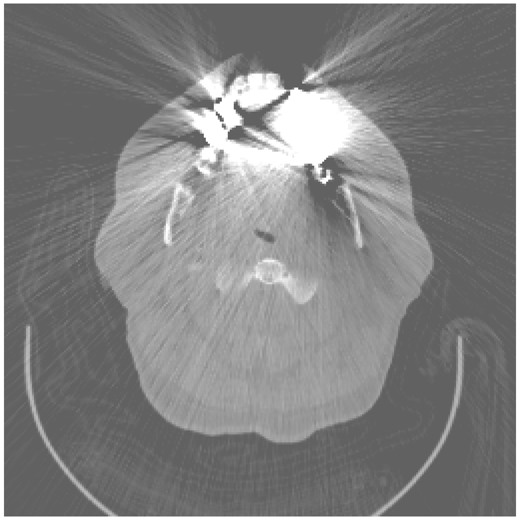
ORNL

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
  - 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.

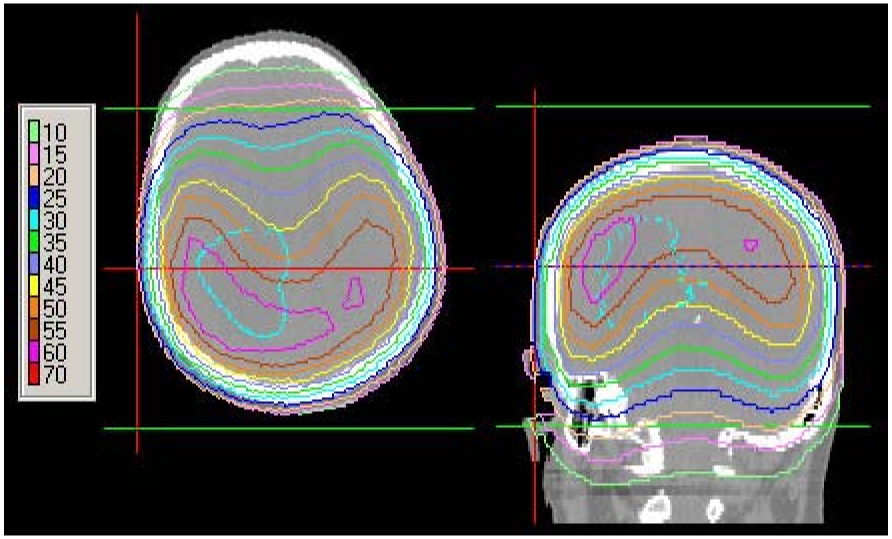


Image from clinical trials using NCTPlan (Harvard-MIT & CNEA)

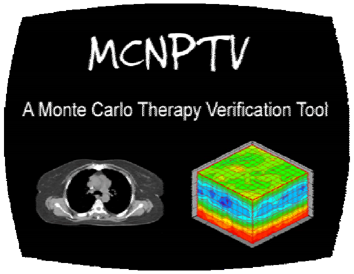
Voxel Models

- Can easily consume Gigabytes of memory
- Large input decks, difficult to modify
- Limit in MCNP5_RSICC_1.30 to ~20 million voxels (lattice locations)
- 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

- 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

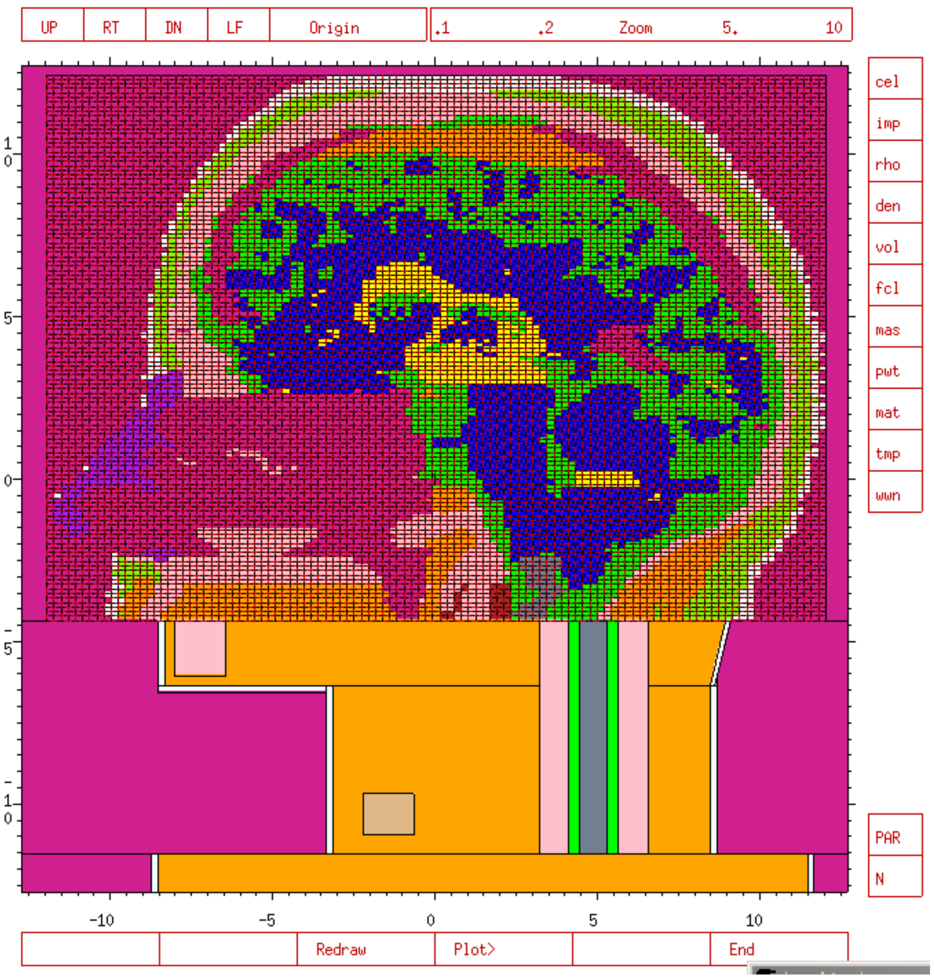


Image from
MCNP5 plotter

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 MP database

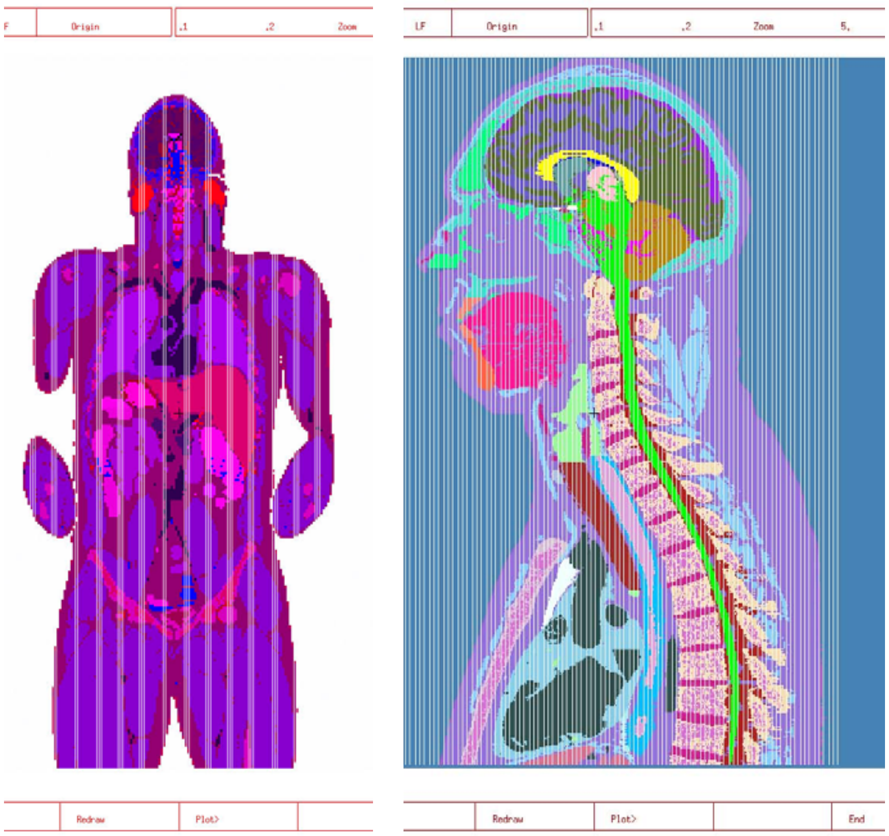


Image from MCNP5 plotter

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

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

MP Geometry Database

- A database of Medical Physics phantom input decks distributed with MCNP5.
- Analytical
  – Snyder Head, ORNL MIRD, MIT MIRD
- Voxel
  – Snyder Head, Water Cubes, Zubal Head
- Contributions Welcome!

Sources

Source Term Definition

- Powerful feature: allows specification of spatial extension, energy, direction, and particle type.
- Source extension need not coincide with the outline of an existing cell
- A large variety of sources may be defined by the user
  - point source
  - area source
  - volume source
  - multiple sources
- Biasing of one or more aspects of the source term is possible.

  For example, some directions or energies may be sampled more frequently (but with reduced weight) to improve transport efficiency.

Source Term Definition

The SDEF card (together with si and sp cards) allows complete specification of the source term.

Syntax: sdef variable$_1$ variable$_2$ variable$_3$ . . .

si card: information about the variable (bins, discrete values, distribution numbers)

sp card: probability of choosing a particular value (provide actual probability or use build-in functions)

- starting particle type & weight
- source spatial extension & location
- energy spectrum
- starting angular distribution (isotropic is default for point & volume sources)

Default: 14 MeV isotropic point source at position 0,0,0 with weight 1

Source Term Definition

(1)	explicit value:	erg = 2	monoenergetic source with energy equal to 2 MeV
(2)	distribution:	erg = d1	distribution “d1” is to be described by si1 and sp1 cards
(3)	function of another variable:		
	sdef	pos = d1	erg fpos d2

SDEF Card: Particle Type

The source variable “par” is used to designate starting particle type:

Par = 1	Neutron	(default)
Par = 2	Photon	
Par = 3	Electron	

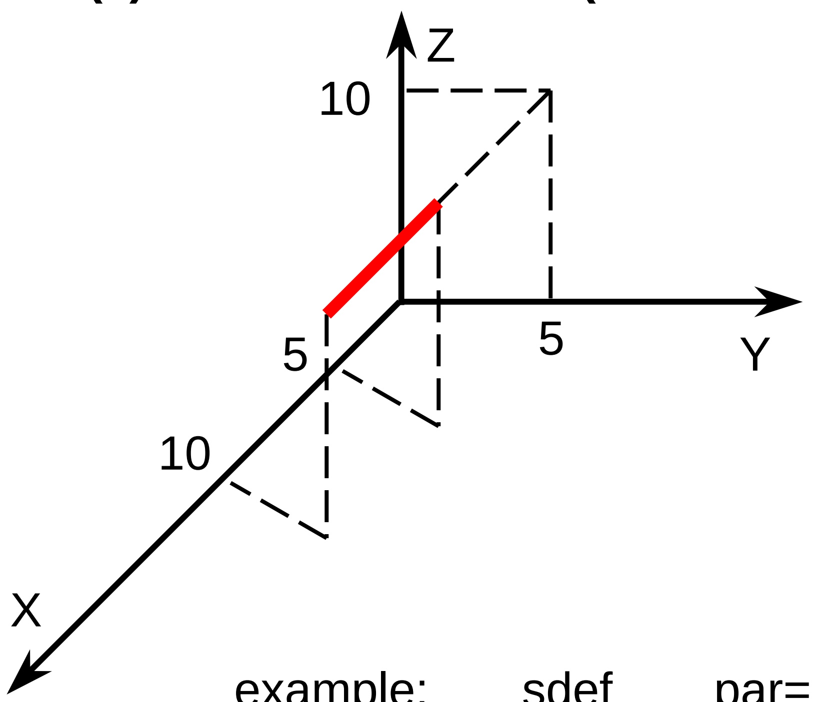
Note: Only one particle type may be started. However, one may start neutrons and electrons and tally for secondary photons.

SDEF Card:
Source Spatial Extension & Location

(1) Point Source

pos = x y z

(2) Line Source (uniform distribution)



Line source extending from

x = 5 to x = 10 at y =5 and z = 10

x = d1   y = 5   z = 10

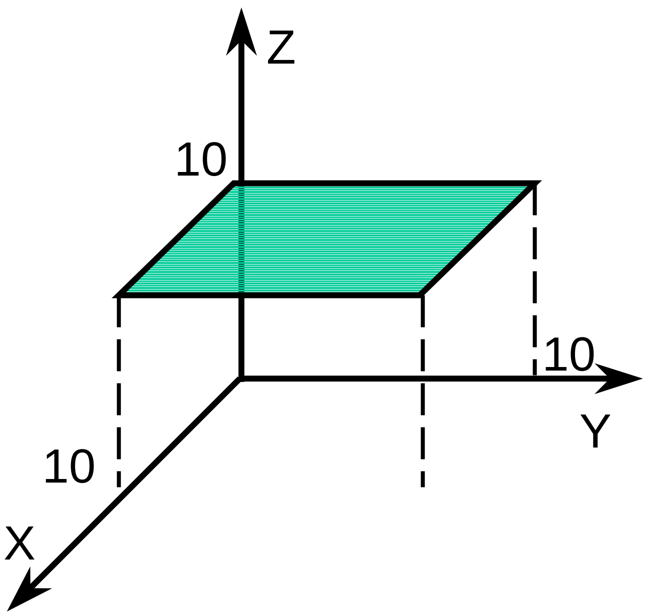
si1	h	5	10
sp1	d	0	1

h = histogram
d = discrete

```
example:    sdef    par=2  erg=1.0  x=d1  y=5  z=10
            si1     h  5  10
            sp1     d  0  1
```

SDEF Card:
Source Spatial Extension & Location

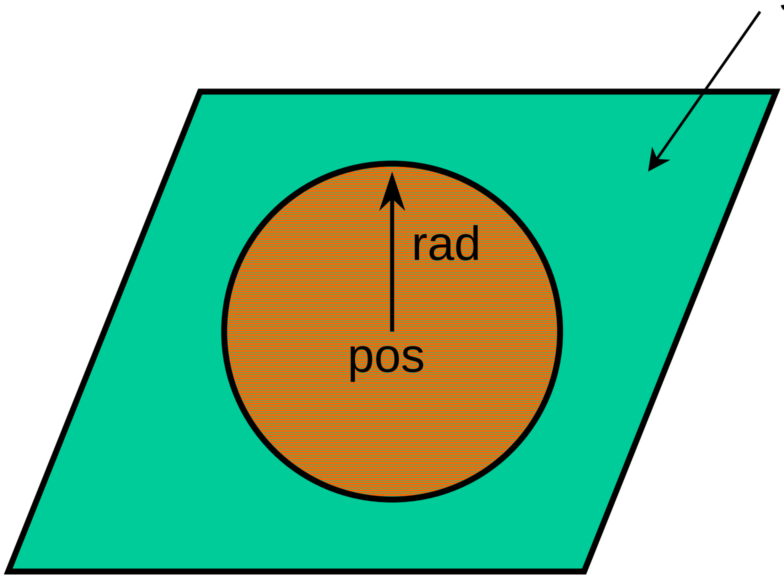
(3) Rectangular Area Source (uniform spatial distribution)



sdef	x = d1		y = d2	z = 10
si1	h	0	10	
sp1	d	0	1	
si2	h	0	10	
sp2	d	0	1	

SDEF Card:
Source Spatial Extension & Location

(4) Disc on a Plane Surface (uniform spatial distribution)



sdef	sur = 5	pos = x y z		rad = d1
si1	h	0	r	
sp1	-21		1	

uniform sampling in circle with radius = r

Probability of picking radius: P(r) = cr

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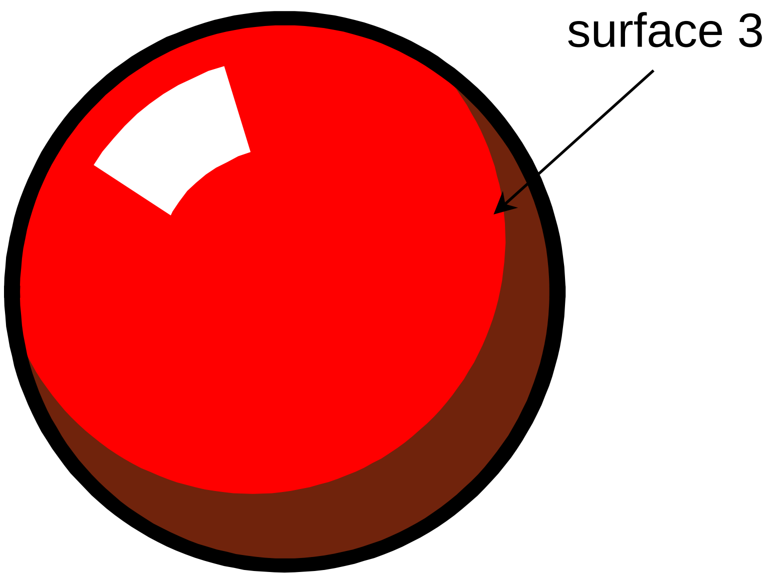
SDEF Card:
Source Spatial Extension & Location

(5) Spherical surface: Source on surface of a sphere

(uniform spatial distribution)



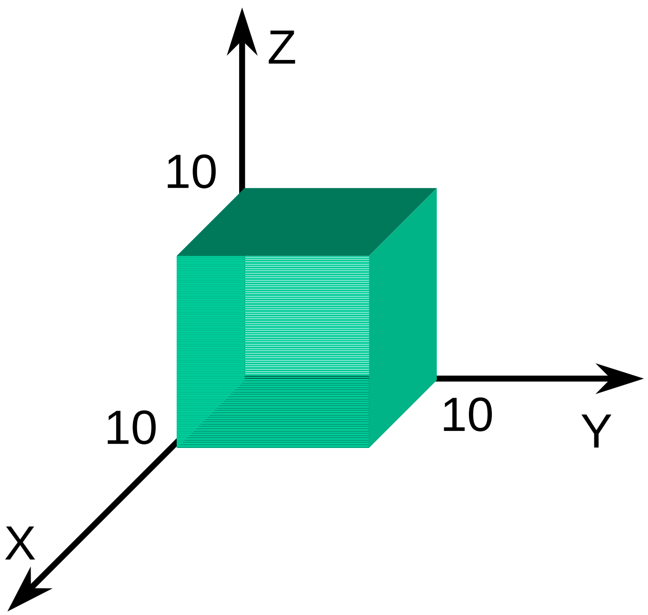
sdef sur = 3

sdef sur = 3 nrm = -1

gives inner directed source

SDEF Card:
Source Spatial Extension & Location

(6) Box Source (uniform volume distribution)



sdef	x = d1	y = d2	z = d3
si1	0	10
sp1	0	1
si2	0	10
sp2	0	1
si3	0	10
sp3	0	1

SDEF Card:
Source Spatial Extension & Location

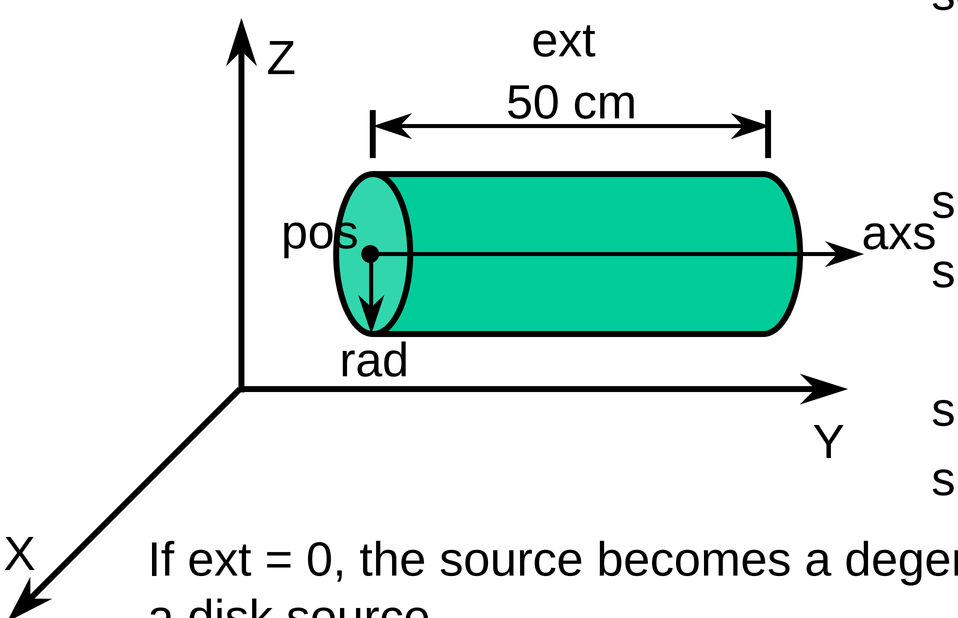
(7) Cylindrical Source (uniform volume distribution)

pos = reference point for sampling position

rad = radial distance from pos

ext = distance from pos along axs (cell case)

axs = reference vector for ext

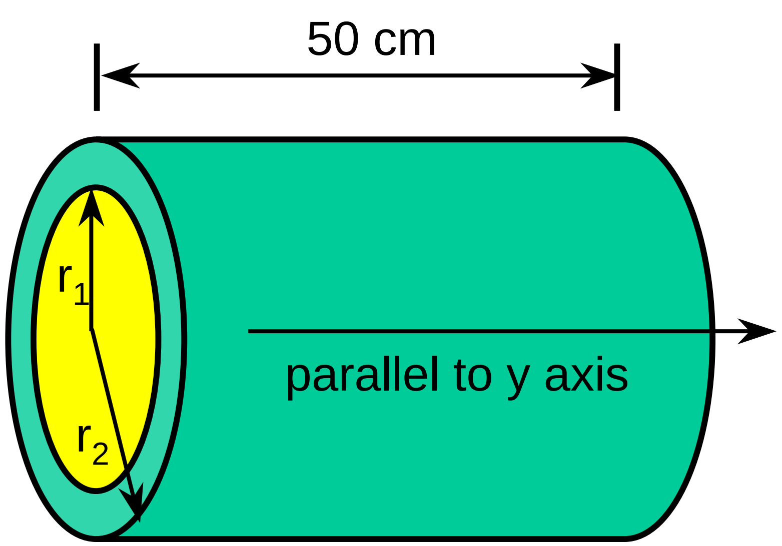


sdef	pos = 5 10 20			rad = d1
	axs = 0 1 0			ext = d2
si1	0	r		
sp1	-21	1	(sampling density is proportional to r)	
si2	0	50		
sp2	-21	0		

If ext = 0, the source becomes a degenerate volume source, or in effect, a disk source.

SDEF Card:
Source Spatial Extension & Location

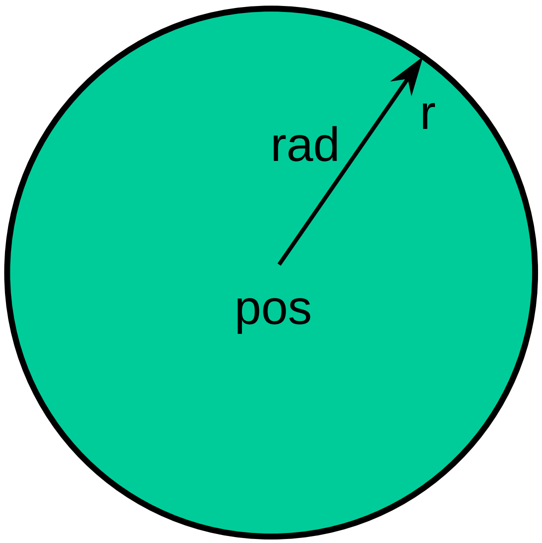
(8) Cylindrical Shell Source (uniform spatial distribution)



sdef	pos = 5 10 20	rad = d1	
	axs = 0 1 0	ext = d2	
si1	r_1	r_2	
sp1	-21	1	
si2	0	50	
sp2	-21	0	

SDEF Card:
Source Spatial Extension & Location

(9) Spherical Volume Source (uniform volume source)

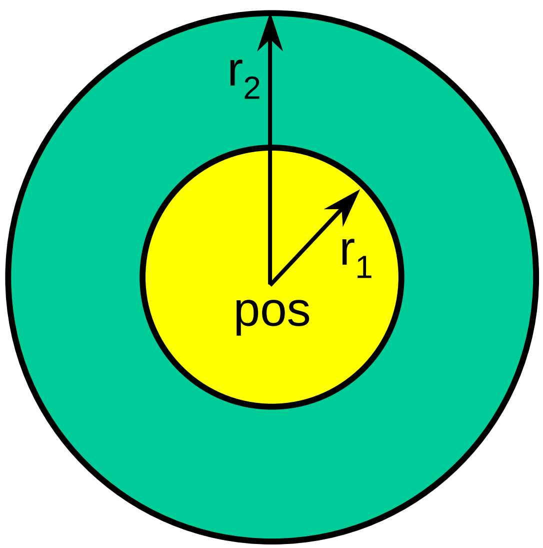


sdef	pos = x y z		rad = d1
si1	0	r	
sp1	-21	2	

Probability of picking radius: $P(r) = cr^2$

SDEF Card:
Source Spatial Extension & Location

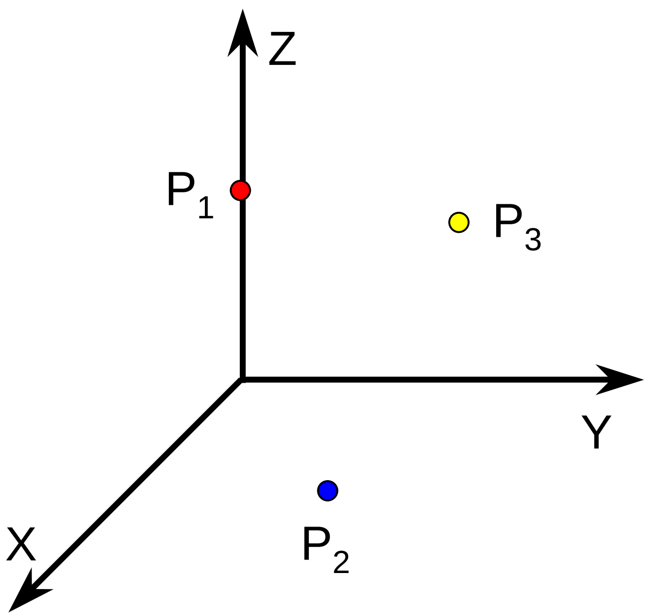
(10) Spherical Shell Source



sdef	pos = x y z		rad = d1
si1	r_1	r_2	
sp1	-21	2	

SDEF Card:
Source Spatial Extension & Location

(11) Multiple Point Source




```
sdef    pos = d1

si1     L    x1 y1 z1   x2 y2 z2   x3 y3 z3
sp1     d    0.25       0.25       0.5
```

Note: Entries on the sp1 card do not need to be normalized. You could simply enter the respective photon emission rates, assuming P_1, P_2, and P_3 are photon sources.

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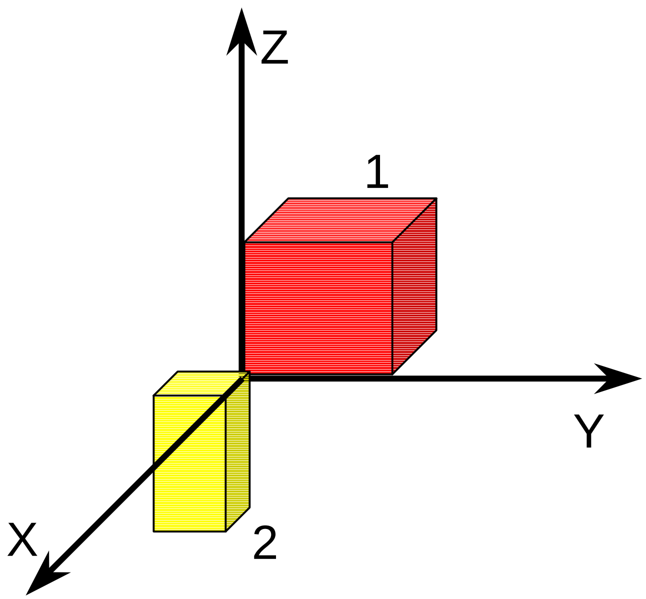
SDEF Card:
Multiple Rectangular Volume Sources

Box #1 = 10-cm cube

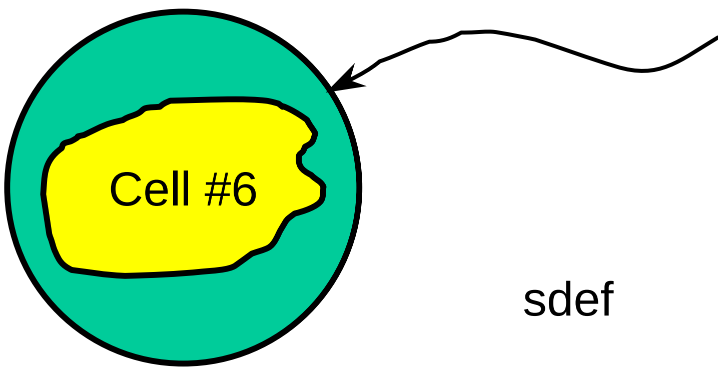
Box #2 = 5-cm x 5-cm x 10-cm




```
sdef    erg=0.662   x=d1   y=fx=d4
        z=fx=d7
si1     s  2  3
sp1     4  1
si2     -10  0
sp2     0  1
si3     0  5
sp3     0  1
ds4     s  5  6
si5     0  10
sp5     0  1
si6     -5  0
sp6     0  1
ds7     s  8  9
si8     0  10
sp8     0  1
si9     -10  0
sp9     0  1
```

SDEF Card:
Source Spatial Extension & Location

Arbitrary Volume Source



Specify a volume source (cartesian, spherical, or cylindrical) that completely contains a particular cell of arbitrary shape.

```
sdef    pos = x y z    rad = d1    cel = 6

si1     h     0     r
sp1     -21   2
```

defines spherical volume source

If the sampled point is found to be inside cell #6, it is accepted. Otherwise, it is rejected and another point is sampled.

SDEF Card: Energy Specifications

Use source variable ERG

(1) Monoenergetic Sources

erg = 10 energy equals 10 MeV

(2) Discrete Spectrum

```
erg = d1
si1     L   0.5  1.0  2.0
c   discrete energies of 0.5, 1.0, and 2.0 MeV
sp1     d   0.25     0.25     0.5
```

SDEF Card: Energy Specifications

(3) Histogram Distribution

erg = d1

si1 h 0.01 0.02 0.05 0.1 0.5

Histogram Distribution:

0.001 - 0.01 (or from Ec to 0.01) (1 keV = photon energy cutoff)

0.01 - 0.02

0.02 - 0.05

0.05 - 0.1

0.1 - 0.5

Entries on the si card define the top of each bin in increasing order.

Two possible sp1 cards:

sp1	d	0	0.2	0.2	0.3	0.3
sp1	c	0	0.2	0.4	0.7	1.0

↑

cumulative

SDEF Card: Energy Specifications

(3) Histogram Distribution (cont.)

Can also use vertical format for data entry:

#	si1	sp1
	0.01	0
	0.02	0.2
	0.05	0.2
	0.1	0.3
	0.5	0.3

“pound” sign
in column 1-5

useful for pasting in energy spectra.

SDEF Card: Energy Specifications

(4) Built-in Function

erg d1

sp1 -2 a

This invokes one of the built-in functions for energy

(See Table 3.3, P. 3-50)

-2 a: Maxwell fission spectrum where a is the temperature in MeV.

The Cf-252 neutron fission spectrum can be described in the energy range from 100 keV to 10 MeV by such a spectrum with a temperature of 1.42 MeV (ISO 8529).

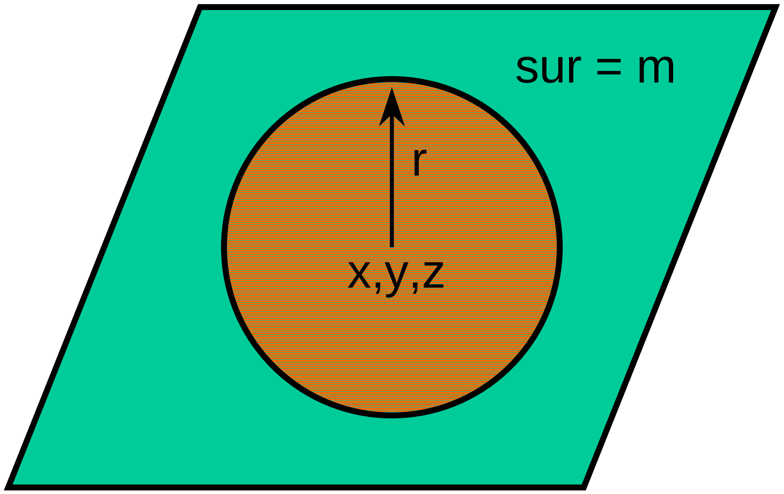
SDEF Card: Direction Specification

(1) For point and volume sources, the default is an isotropic distribution.

(2) The variable dir is used to specify direction and refers to μ, the cosine of the angle between a particle's line of flight and a reference vector VEC.

(3) For a surface source, the default is a cosine distribution. However, it is possible to specify an isotropic distribution.

Consider a disc source on surface m:



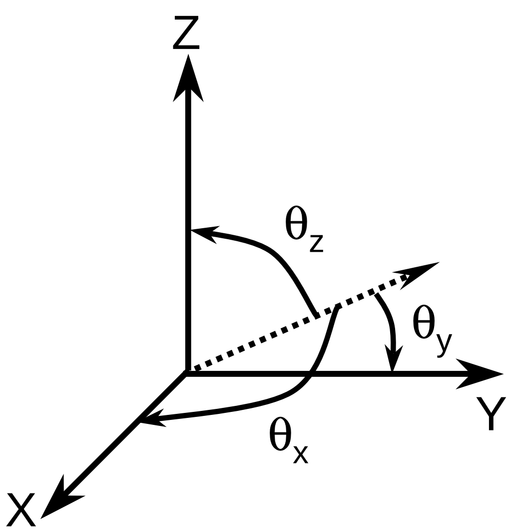

```
sdef    sur = m    pos = x y z    rad = d1
si1     h    0    r
sp1     -21    1
```

Default: p(dir) = 2 • cosθ

SDEF Card: Direction Specification

To modify direction to isotropic:

(a) First define a reference vector (vec) for direction. All particle lines of flight will be referenced relative to vec. This is a unit vector with x, y, and z components:



vec = $\cos\theta_x \; \cos\theta_y \; \cos\theta_z$

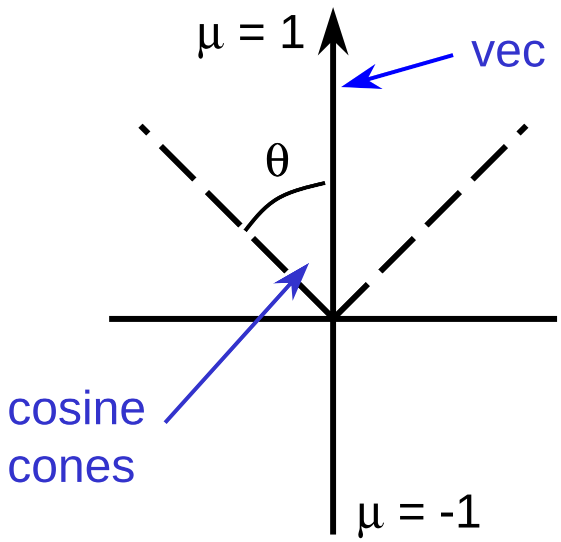
Example 1: vec = 0 0 1 is a unit vector pointing along the positive z axis

Example 2: vec = 0.707 0.707 0 is a unit vector in x-y plane at 45-deg. from x and y axis

Example 3: vec = 0.707 0.707 0.5 is a unit vector 60-deg. from z axis and 45-deg. from x and y axes

SDEF Card: Direction Specification

(b) Specify direction as a distribution (dir = d2) relative to vec:



θ = polar angle

0 = azimuthal angle - sampled uniformly between 0 and 2π

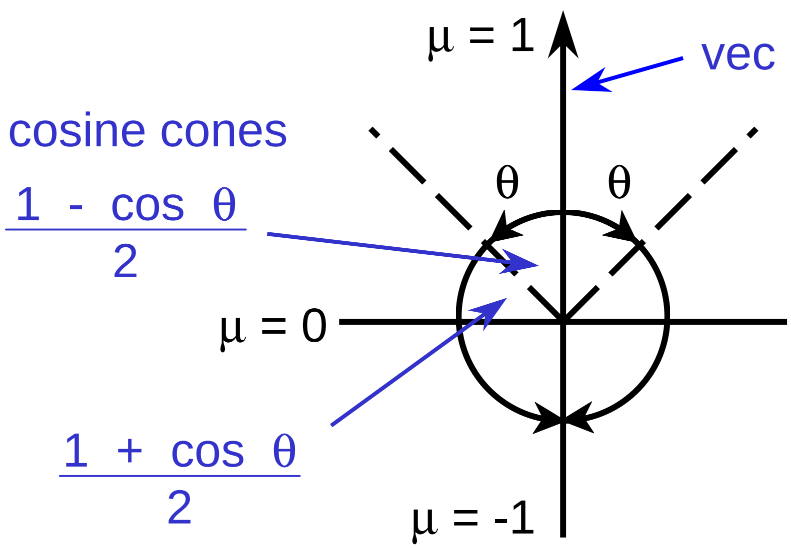
μ = cosθ (range is from -1 to 1)

```
si2    h  -1   0    1
sp2    d   0   0.5  0.5   $ equal probability for each cosine cone
```

This specifies an isotropic directional distribution

SDEF Card: Direction Specification

In general, polar angle space can be specified using direction cosine cones:



si1 -1 $\cos\theta$ 1

sp1 0 $\frac{1 + \cos\theta}{2}$ $\frac{1 - \cos\theta}{2}$

The sp card specifies an isotropic distribution.

SDEF Card: Direction Specification

Discrete Directions are Possible:

sdef   sur = m   pos = x y z   rad = d1   vec = 0 0 1   dir = 1

This gives a monodirectional parallel beam along the + Z axis.

sdef   pos = x y z   dir = 1   vec = 0 0 1

mono-directional point source.

Tallies

- Calculating dose w/ different tallies
- Flux to Kerma factors (DE DF cards)
- Calculating reaction rates

MCNP Tallies

- Standard Fluence Tallies are:

F2	Fluence averaged across a surface (#/cm^2)
F4	Fluence averaged across a cell (#/cm^2)
*F2	Energy Fluence (MeV/cm^2)
*F4	Energy Fluence (MeV/cm^2)

- Easy to use: syntax f4:p 8 ← tally cell no.

  ↑ particle type

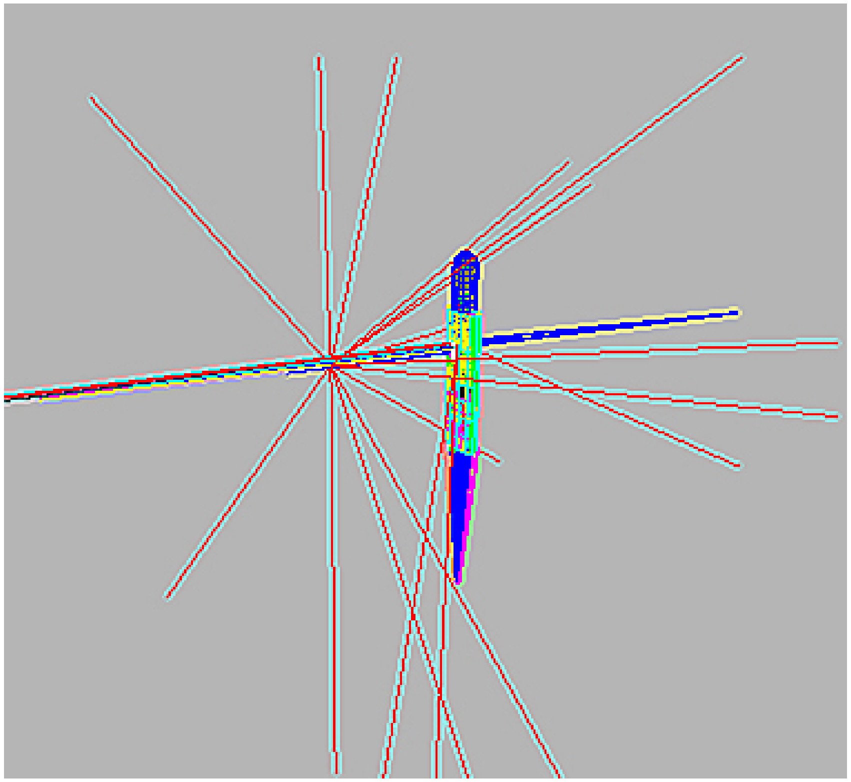
- f14, f24, f34, . . . are all type F4 tallies

  f12, f22, f32, . . . are all type F2 tallies

- Multiple tallies are allowed

  Several tally types may be mixed in the input deck

MCNP Tallies

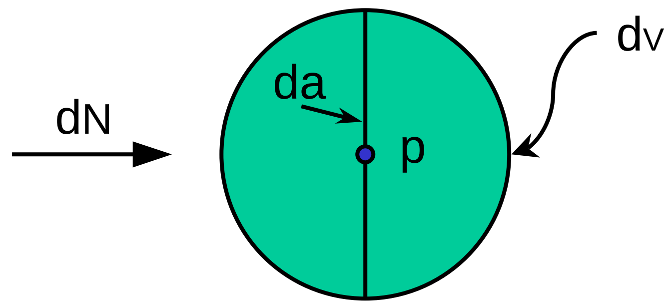


- By virtue of the nature of the simulation, MCNP builds up a picture of the radiation field:
  - Position
  - direction
  - energy
  - weight
- MCNP is ideally suited to determine current or fluence quantities.
- Fluence (particles/cm^2) is of paramount importance, because it can be converted into absorbed dose or dose equivalent if the differential distribution in energy is known.

Fluence Quantities

<u>Fluence at a Point</u>:

$$\phi\,(p) = \frac{dN}{da}$$

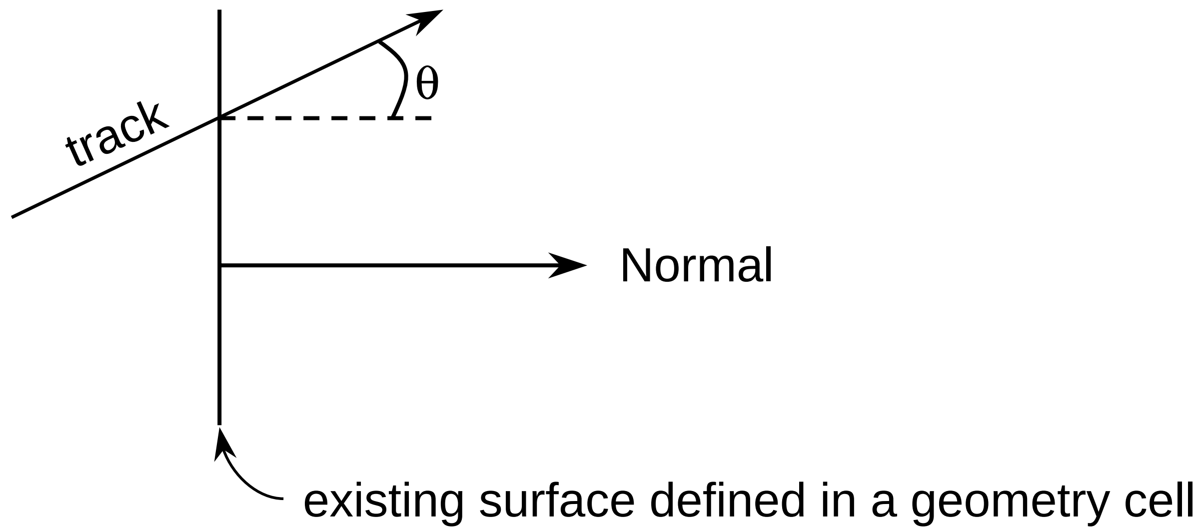


<u>Alternative definition</u>: $\phi\,(p) = \frac{dL}{dv}$

dL = sum of track lengths of all the particles traversing volume element dv

Track length definition is extended to surfaces and volumes in MCNP

Type F2 Tally: Surface (Planar) Fluence



<u>For each track:</u>

$$\text{Tally score} = \frac{W}{A|\cos\theta|} = \frac{W}{A|\mu|}$$

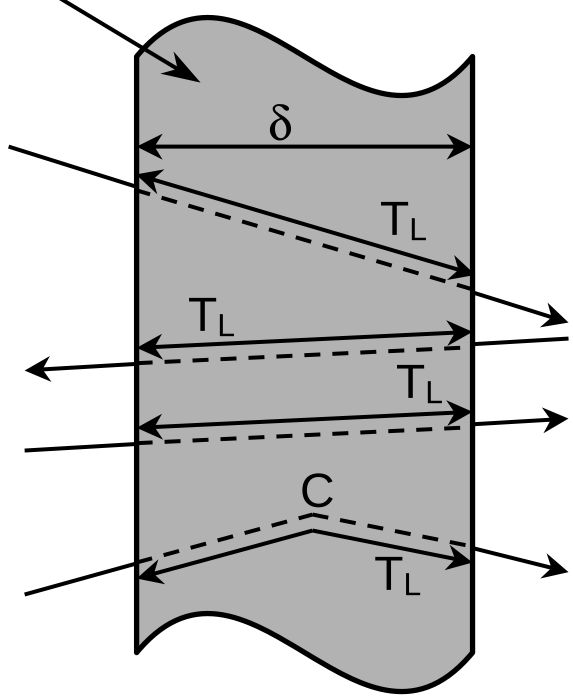
$$F2 = \frac{1}{N} \sum \frac{W}{A|\mu|}$$

For $|\mu| < 0.1$, MCNP sets $|\mu| = 0.05$

Type F4 Tally: Average Fluence in a Cell

MCNP uses track length (T_L) estimator for fluence averaged over a cell volume: $\phi = \sum \frac{T_L}{V}$

V = A • δ



V = cell volume

A = Surface Area

C = collision site

W = track weight

For each track:

tally score $= \frac{W\,T_L}{V}$

$$f4 = \frac{1}{N} \sum \frac{W\,T_L}{V}$$

where N = number of source particles

NOTE: Cell collisions are not necessary. A particle may cross the cell more than once during its history.

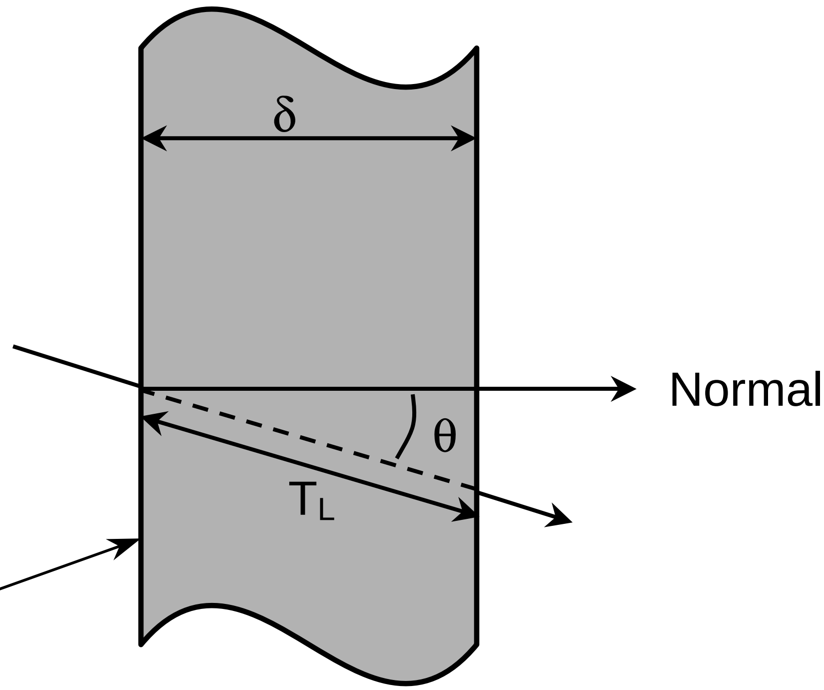
Type F2 & Type F4 Tally Relationship

In the limit, as the cell thickness (δ) goes to zero:

$$f4 = \frac{1}{N} \lim_{\delta \to o} \sum \frac{W\ T_L}{V}$$

$$T_L = \frac{\delta}{\cos \theta}$$

$$V = \delta A$$



$$f4 = \frac{1}{N} \sum \frac{(W)(\frac{\delta}{\cos \theta})}{\delta\ A} = \frac{1}{N} \sum \frac{W}{A\ |\ \cos \theta\ |} = \frac{1}{N} \sum \frac{W}{A\ |\ \mu\ |} = f2$$

Point Detector Tally (Type F5)

- A <u>deterministic</u> estimator of fluence at a point

  units: cm^{-2}

- Input card:

  f5:n   x y z R_0

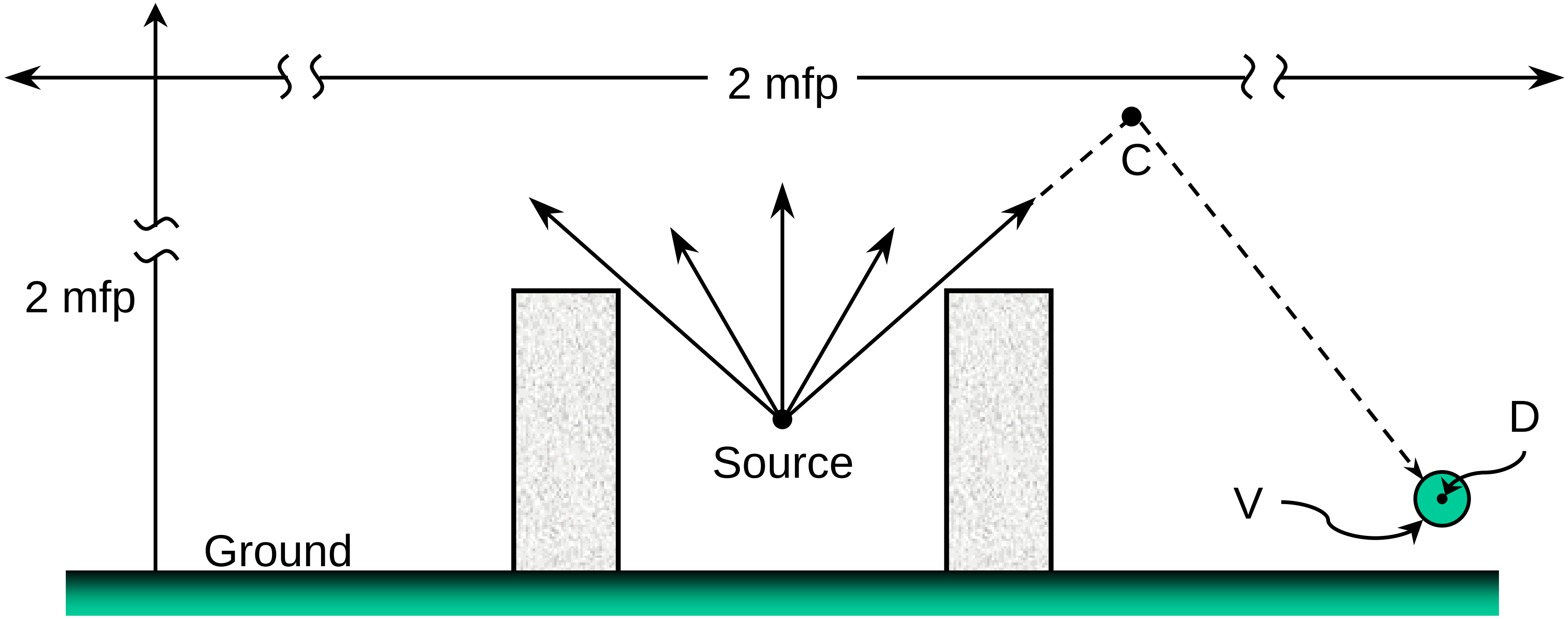
  n = particle type

  x y z = coordinates of tally point

  R_0 = radius of sphere of exclusion located around tally point

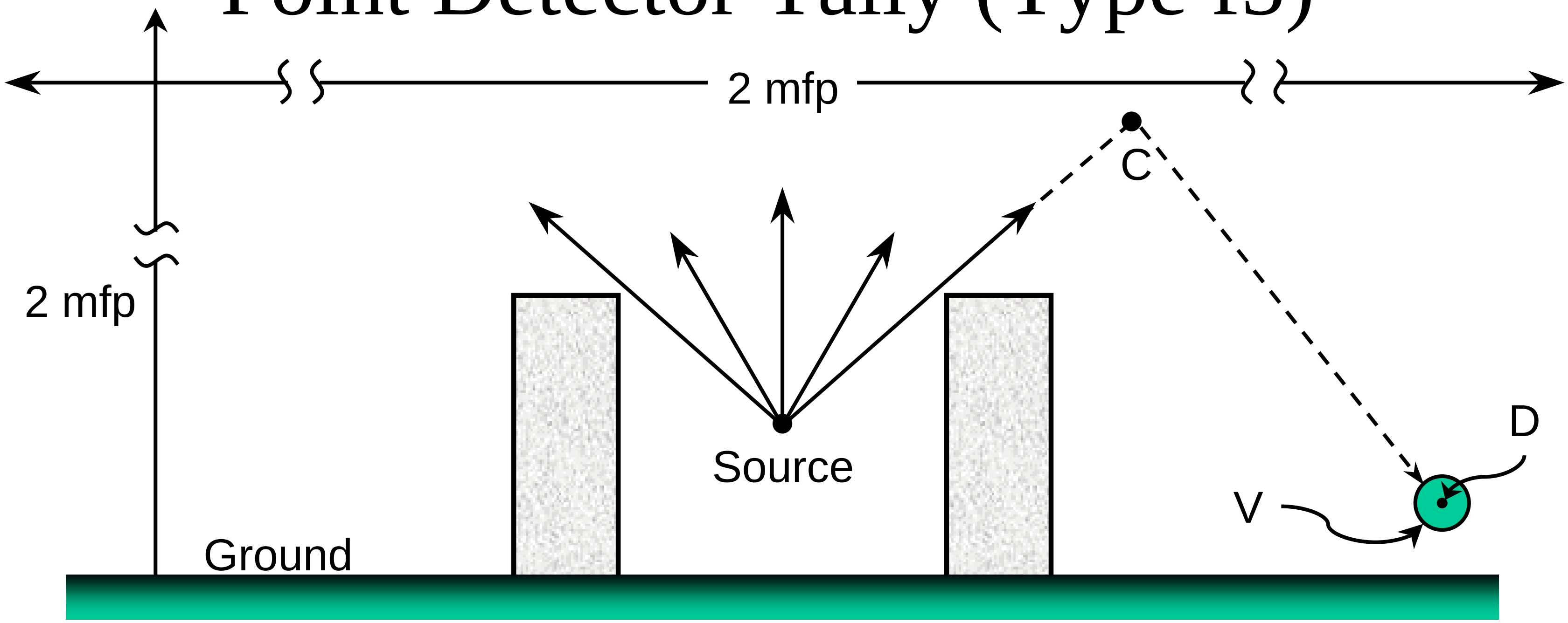
- A necessity in situations where the analog random walk is inefficient

Skyshine Calculation



- Silo restricts dose rate at point D to skyshine and groundshine
- We wish to calculate dose rate in a small volume element, V, around D
- F4 tally fails because V is poorly sampled by random walk

Point Detector Tally (Type f5)



At every collision site, we will calculate the probability of a particle scattering toward and penetrating to point D. Three factors affect this probability:

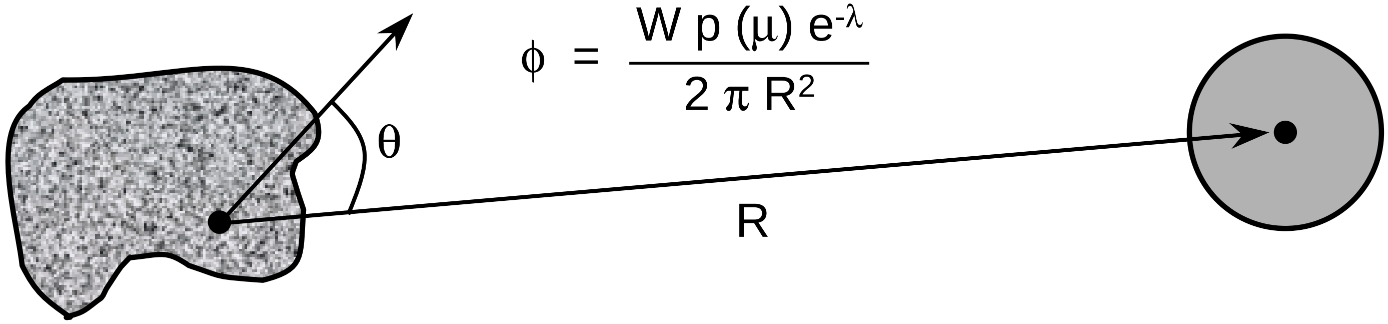
- distance between collision and detector
- probability of scattering toward D, rather than in original direction
- material absorption between the collision site and D

We then multiply the track weight by this probability and place the result at point D as a score.

Point Detector Tally (Type f5)

MCNP calculates the point detector contribution using the following equation:

$$\phi = \frac{W\, p(\mu)\, e^{-\lambda}}{2\pi R^2}$$



where:

μ = $\cos\theta$

W = particle weight

$\lambda = \sum_i \mu_i x_i$ = total number of mean free paths (mfp) integrated over the trajectory from the source or collision point to the detector

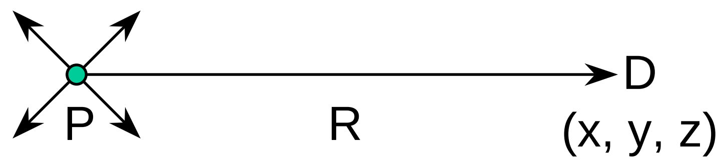
R = distance to detector from collision site or source creation event

$p(\mu)$ = value of probability density function at μ, the cosine of the angle between the particle trajectory and the direction to the detector

Point Detector Tally (Type f5)

For a point isotropic source in a void:

$e^{-\lambda} = 1$



$p(\mu) = 0.5$

$$\phi = \frac{W}{4\pi R^2}$$

The contribution to a point detector for each source creation event

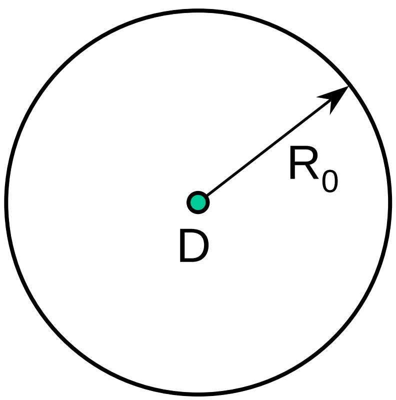
Point Detector Tally (Type f5)

Sphere of Exclusion

- For point detectors located in scattering media, collisions are possible very near the detector:

  As $R \to 0$, $\phi \to$ infinity, variance $\to$ infinity

- The sphere of exclusion eliminates the $1/R^2$ singularity of the point detector



For collisions within the sphere ($R<R_0$), the detector contribution becomes a volume average:

$$\phi = \frac{W\, p(\mu)\, (1-e^{-\Sigma_t R_0})}{2/3\, \pi\, R_0^3\, \Sigma_t} ,$$

where Σ_t = total macroscopic cross section

Point Detector Tally (Type f5)

Caveats

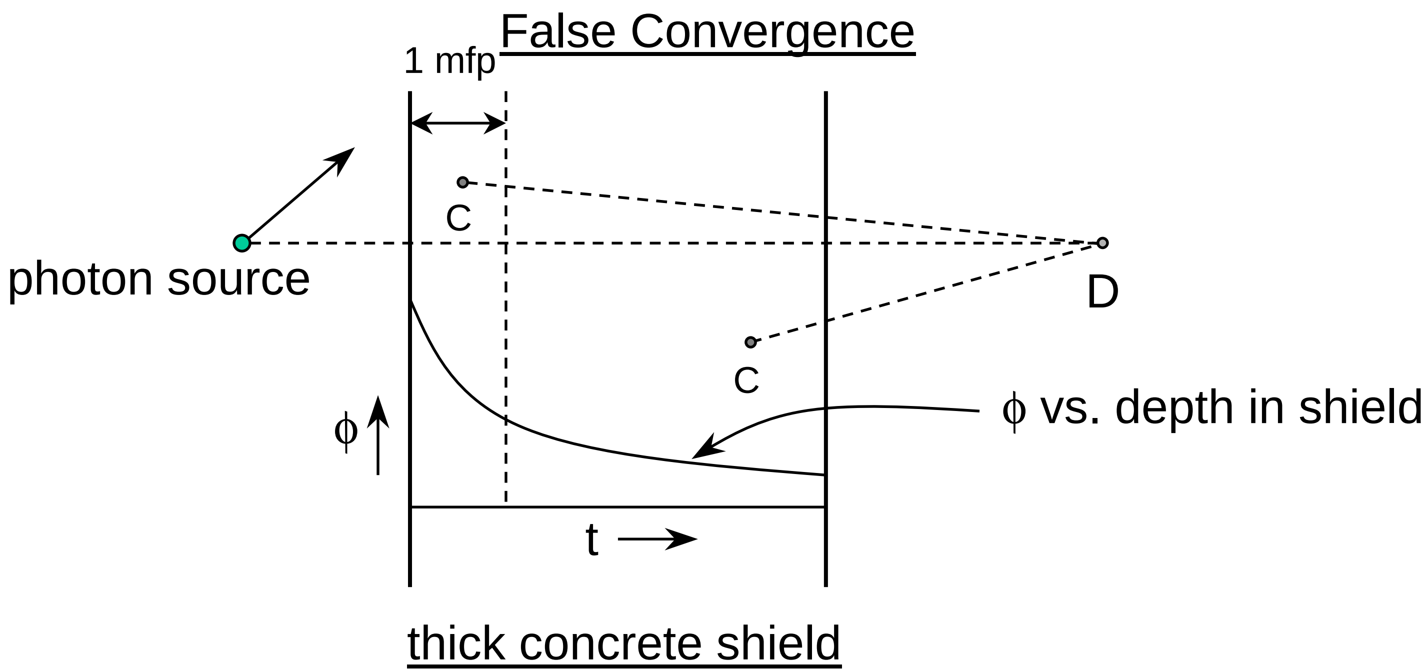
- Avoid use inside heavily scattering medium or very near a scattering surface. Use F2 or F4 type tallies instead

- It is permissible to locate a point detector in a lightly scattering medium such as air. But use an exclusion sphere with $R_0 \geq 50$ cm.

- Pay special attention to tally convergence:
  - slow convergence:

    Error may decrease $< \frac{1}{\sqrt{N}}$
  - False convergence

Point Detector Tally (Type f5)



- Early in transport, detector contributions are from source (uncollided flux) and collisions in 1st mfp. All of these scores are small and tightly bunched. Error appears to converge nicely.
- As more histories are followed, some photons reach deep in the shield and contribute very large scores. The variance jumps as does the standard error.
- The key is to adequately sample collisions deep in the shield.

Tally Normalization

- All MCNP tallies (except in criticality problems) are normalized to one starting source particle.
- The user is responsible for scaling the tally to the desired source strength.
- Tally normalization is tied to how the source strength is defined:

  - If the source strength for a given problem is specified in terms of photons/sec, normalize by multiplying the tally result by the actual photon emission rate - irrespective of whether the source is a point, area, or volume source.

  - If, however, the source strength is specified as C particles per unit area or per unit volume, then the normalization must include the actual area of volume of the source.

Tally Normalization

Example: a tally for a volume source of 100 cm^3 with an actual source strength of C photons/sec per cm^3, would be normalized as follows:

(Tally result)(100 cm^3)(C)

This is so, because the MCNP source strength is 1/V and the actual source strength is C photons/sec per cm^3.

Source Type	MCNP Source Strength	Desired Normalization	Normalization Factor
Point	1 particle	S particles/sec	S
Area	$\frac{1 \text{ particle}}{A}$	S particles/sec • cm^2	(A) (S)
Volume	$\frac{1 \text{ particle}}{V}$	S particles/sec • cm^3	(V) (S)

A = source area in cm^2
V = source volume in cm^3

Tally Modifier Cards

En	E_1 E_2 . . . E_k	MeV
Tn	T_1 T_2 . . . T_k	(shakes, 1 shake = 10^{-8} seconds)
Cn	C_1 C_2 . . . C_k	(-1 to 1)

n = tally number

E_i, T_i, C_i = upper bound of the i^{th} energy, time, or cosine bin in increasing magnitude.

A response function may be folded in with tally by specifying appropriate DE and DF cards.

DEn	A	E_1	E_2	. . .	E_k
DFn	B	F_1	F_2	. . .	F_k

E_i = energy points (MeV)

F_i = corresponding value of the dose function

A = LOG or LIN energy interpolation method (LOG is default)

B = LOG or LIN dose interpolation method (LOG is default)

FM Card (Tally Multiplier): $C \cdot \int \phi(E)\, R_m(E)\, dE$

FMn $(C_1\ m_1\ R_1)\ (C_2\ m_2\ R_2) \ldots$ T

n = tally number

C_i = multiplicative constant

m_i = material number identified on an Mn card

R_i = a combination of ENDF reaction numbers (a space = multiply and a colon = add)

Common R Values for Photons

-1 incoherent scattering cross section
-2 coherent scattering cross section
-3 photoelectric cross section
-4 pair production cross section
-5 total cross section
-6 photon heating number

Common R Values for Neutrons

1 = total cross section
-2 = absorption
-4 = heating (MeV/collision)
-6 = fission cross section
-8 = fission Q (MeV/fission)

Determination of Dose

- MCNP normally calculates absorbed dose on the basis of the KERMA approximation: Kinetic energy transferred to charged particles is assumed to be locally deposited.
- Conditions under which the KERMA approximation is valid:
  - Low-energy photons (secondary electrons have very short range)
  - Charged Particle Equilibrium (CPE) or at least transient CPE exists: range of primary radiation is >> than that of secondaries
  - Radiative losses in medium are negligible
- Under condition of CPE:

$$\text{Absorbed Dose} = \text{KERMA}$$

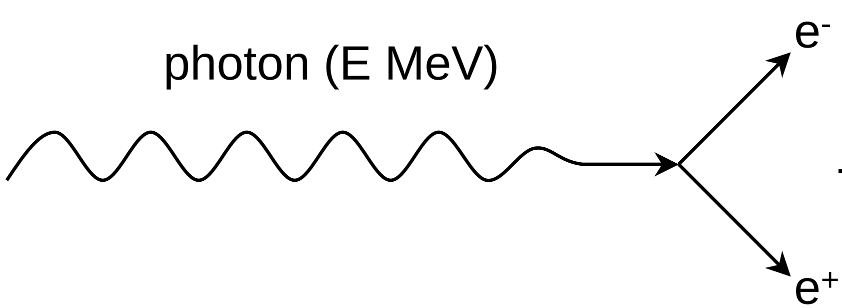
$$\text{Exposure in Roentgens} = \frac{\text{Dose (air)}}{0.876}$$

Determination of Dose

Charged - Particle Equilibrium (CPE)

In practical situations, the conditions for CPE or transient CPE are not fulfilled in the following cases:

- In the vicinity of an interface between two different materials - especially near an air - material boundary.
- Near the edges of a beam or in regions very close to a radiation source.
- Large change in photon spectrum with depth of penetration.
- High-energy photon beam incident on high-Z target.



<u>Pair Production</u>

– MCNP assumes that kinetic energy of electron-positron pair = E - 1.022

– Actual KE = E - 1.022 - E_x where E_x = energy of bremsstrahlung photons radiated away.

Dose Calculation

Three basic approaches:

(1) Track Length Heating Method:
F4/FM4 or type F6 Tallies (Kerma approximation) for a specified cell.

(2) *F8 Tally: rigorous dosimetry for situations where Kerma approximation does not hold (only for photons and electrons).

(3) Fold in Fluence to Dose Conversion Function using DE/DF cards.
Valid for irradiation geometry implicit in conversion function – typically whole body irradiation.

Track Length Heating Method

- H(E) = Average MeV per collision
- For photons: kinetic energy transferred to secondary electrons
- H(E) is part of cross section tables
- Step 1: Calculate collisions / gram in tally cell

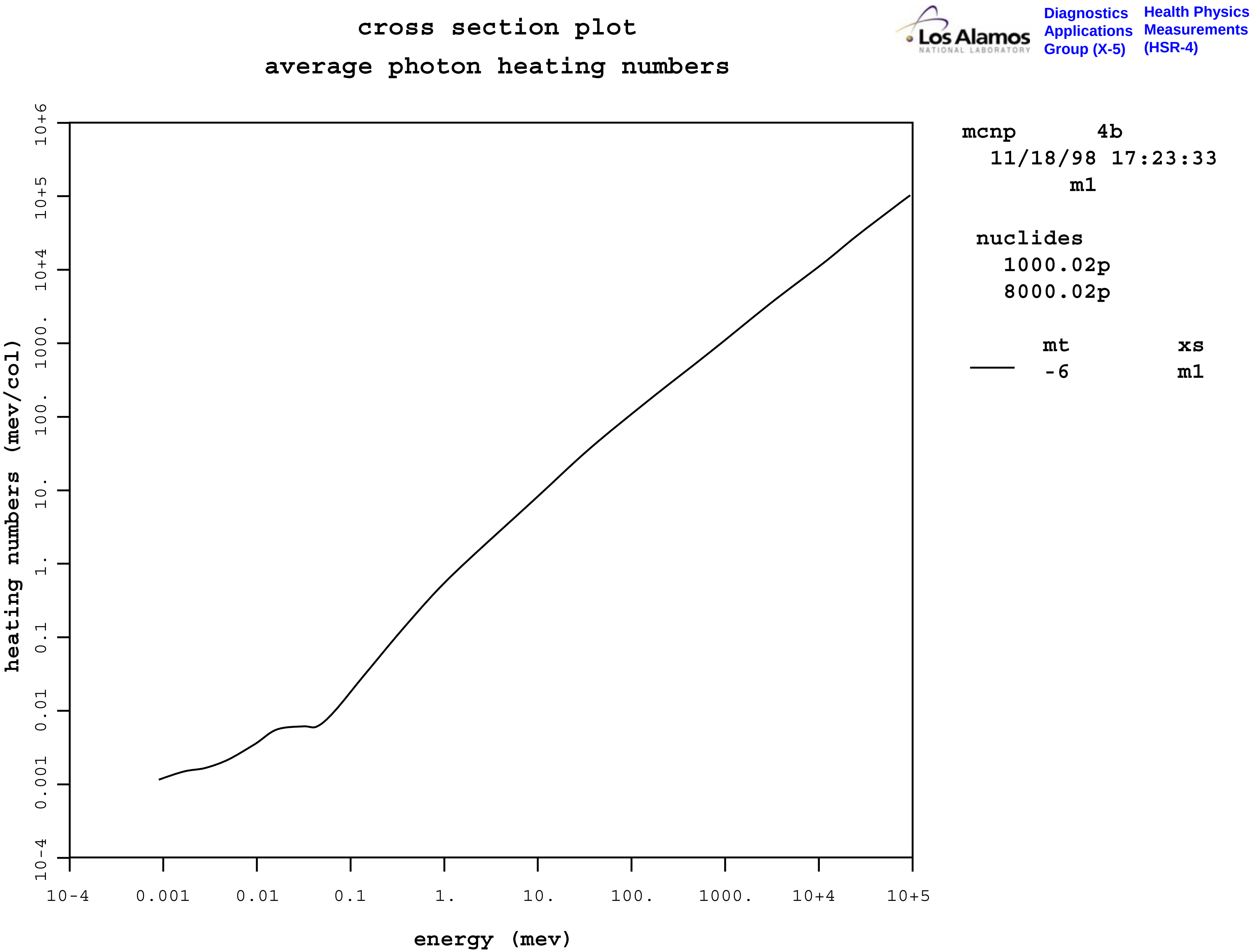
$$\left(\frac{\text{atoms}}{\text{gram}}\right)\left(\sigma_t \text{ barns}\right)\left(\phi \text{ cm}^{-2}\right)\left(1 \times 10^{-24} \frac{\text{cm}^2}{\text{barn}}\right)$$

- Step 2: Calculate MeV/gram in tally cell

$$\left(\frac{\text{collisions}}{\text{gram}}\right)\left(H \frac{\text{MeV}}{\text{Collision}}\right)$$

- Step 3: Convert to absorbed dose in rads

Multiply Step 2 by: 1.602×10^{-8} rad/MeV/g



cross section plot
average photon heating numbers
Diagnostics Applications Group (X-5)
Health Physics Measurements (HSR-4)
heating numbers (mev/col)
10+6
10+5
10+4
1000.
100.
10.
1.
0.1
0.01
0.001
10-4
10-4
0.001
0.01
0.1
1.
10.
100.
1000.
10+4
10+5
energy (mev)
mcnp 4b
11/18/98 17:23:33
m1
nuclides
1000.02p
8000.02p
mt xs
-6 m1

Track Length Heating Method

- Average dose per source photon: $D\ (\text{rads}) = \frac{C}{N} \sum_{i=1}^{N} \phi\, \sigma_t\, H$

where, $C = (1.6 \times 10^{-8})\,(1 \times 10^{-24}) \left(\frac{N_a\, \eta}{M} \right)$

N_a = Avogadro's Constant = $6.022 \times 10^{23}\ \text{mol}^{-1}$

η = number of atoms per molecule

M = molar mass of material in grams

ϕ = fluence score

σ_t = total atomic cross section for energy of scoring track in barns

H = Heating Number (MeV / Collision) at energy of scoring track

Track Length Heating Method

- The Tally Multiplier Card (FM) allows us to fold in the required cross section and heating number to obtain absorbed dose
- This method works in conjunction with F2, F4, and F5 fluence tallies.

```
F4:p     5    $ fluence averaged over cell No. 5
FM4      C    m   -5   -6
```

multiply each score by Heating Number at energy of scoring track

multiply each score by constant C

material No. of material for which dose is calculated

multiply each score by σ_T at energy of scoring track

- $C = \left(\dfrac{N_a\,\eta}{M}\right)(1.6\mathrm{x}10^{-8})\,(1\mathrm{x}10^{-24})$
- NOTE: Cell No. 5 may be void

Dose Calculations: Heating Tally (F6)

- Syntax

  cell #

  f6:p 2

  fm6 1.602e-8

- Basic tally units are MeV/g. The energy deposition is for the material filling the tally cell.

- Works identically to f4/fm4 dose calculations. The code generates the proper tally multiplier card. The tally cell must be filled with material.

- The constant (1.602e-8) on the fm6 card converts MeV/g to rads.

Fluence to Dose Conversion Functions

- Use DE/DF cards to fold in an appropriate conversion function

Example:

```
c       ICRP 51, Table 11: Fluence to Air Dose Conversion
c       Function, radcm^2
de5     0.01  0.015  0.02  0.03  0.04  0.05  0.06  0.08  0.1  0.15  0.2
        0.3  0.4  0.5  0.6  0.8  1  1.5  2  3  4  5  6  8  10
df5     7.43e-10  3.12e-10  1.68e-10  0.721e-10  0.429e-10
        0.323e-10  0.289e-10  0.307e-10  0.371e-10  0.599e-10
        0.856e-10  1.38e-10  1.89e-10  2.38e-10  2.84e-10
        3.69e-10  4.47e-10  6.12e-10  7.5e-10  9.87e-10
        12e-10  13.9e-10  15.8e-10  19.5e-10  23.1e-10
```

- Each fluence score is multiplied by a value of the conversion function corresponding to the energy of the scoring track:

$$(s\ \mathrm{cm}^{-2})\ (y\ \mathrm{rad\ cm}^{2}) = sy\ \mathrm{rad}$$

where s = score
y = value of the conversion function

- This method may be used with any of the fluence tallies:

F2, F4, or F5

Fluence to Effective Dose Conversion Functions

- Typically based on Monte Carlo simulations using anthropomorphic phantoms

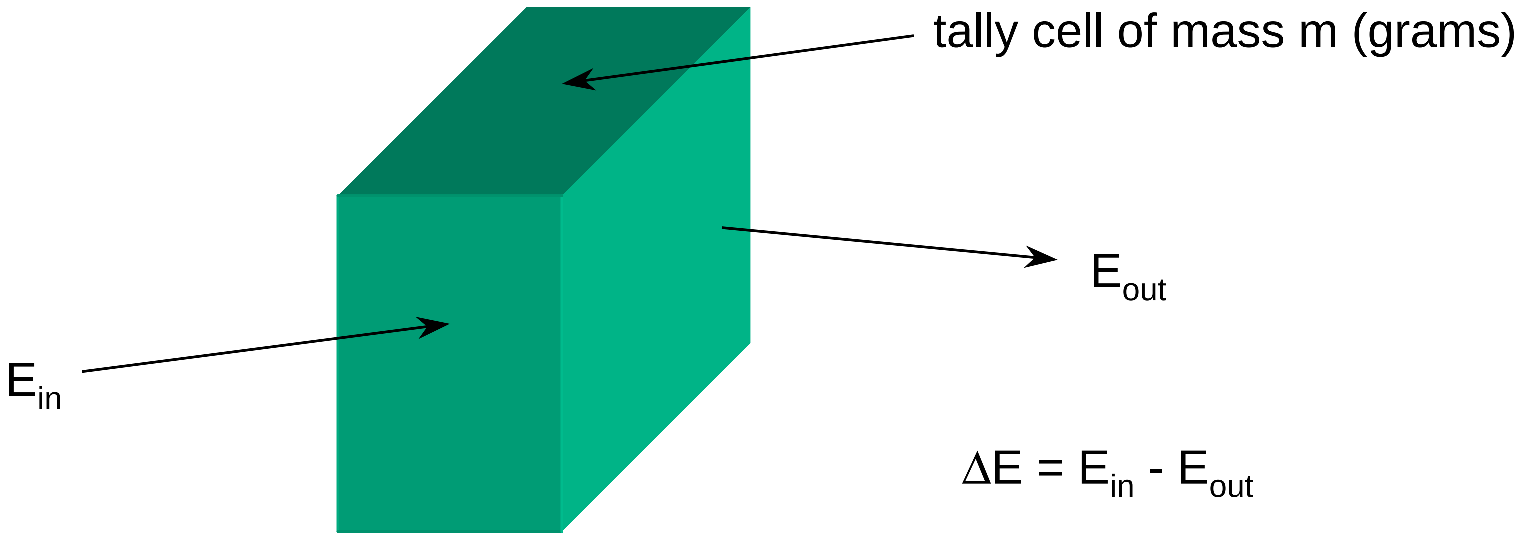
- Valid for a specific irradiation geometry:
  AP, PA, LAT, ROT, ISO

- Published ICRP functions are based on expanded and aligned field: do not apply for partial body irradiation

- Factors for partial body irradiation require calculations from first principles using anthropomorphic phantoms:
  - Voxelized human phantom
  - MIRD phantom: commercial version by White Rock Science

The *F8 Tally

Whenever CPE does not hold, absorbed dose may be calculated from first principles using the *F8 tally:



$$\Delta E = E_{in} - E_{out}$$

$$*F8 = \frac{1}{N}\sum_{i=1}^{N} \Delta E_i \text{ , N = number of histories}$$

$$D(\text{rads}) = \frac{*F8(\text{MeV})}{m} \bullet (1.602\text{x}10^{-6}\ \text{erg/MeV}) \bullet \left(\frac{1}{100\ \text{erg/g/rad}}\right)$$

$$D(\text{rads}) = \frac{*F8}{m} \bullet 1.602\text{x}10^{-8}$$

*F8 Tally: Mode Card Selection

- **mode p e**

  Most rigorous: Both photons and electrons are transported.

  Use when electron transport is important
  - small tally cells
  - high photon energies

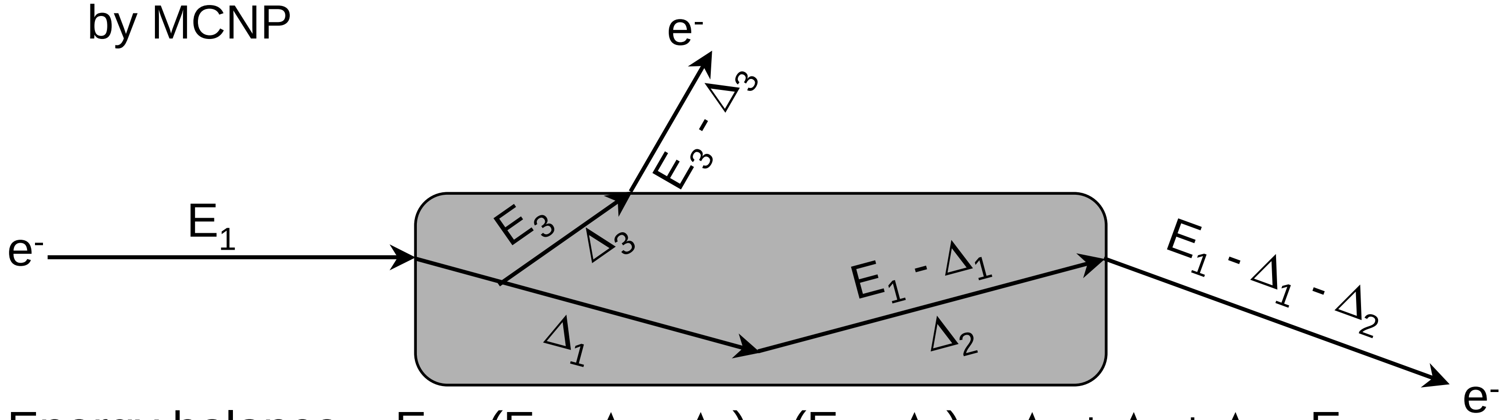
- **mode p**

  Only photons are transported. Electron energy is locally deposited.
  TTB is used by the code to generate secondary bremsstrahlung.
  Much faster than coupled electron - photon transport.

*F8 Type Tally: Physics

Negative Energy Score

- An artifact of the condensed-history electron transport model used by MCNP



Energy balance = $E_1 - (E_1 - \Delta_1 - \Delta_2) - (E_3 - \Delta_3) = \Delta_1 + \Delta_2 + \Delta_3 - E_3$

- E_3 is the energy of the delta ray produced during the first substep in the tally cell
- The energy of the primary electron is not correlated with the energy of the delta ray
- If $E_3 > (\Delta_1 + \Delta_2 + \Delta_3)$ the energy balance is negative
- Only an issue in thin detector cells where a fraction of the delta rays produced during electron transport may escape

Misc

- S(α,β) neutron scattering treatment
- Simple Variance Reduction
- 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
- What MCNP5 cannot do
  - High-Energy Particles (muons, pions, etc..)
  - Coincident Counting (lacks code and data)
  - Photon Polarization
  - Proton Transport (available with release of MCNP6)
- MCNP Help
- Obtaining MCNP

Neutron Scattering Treatment

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

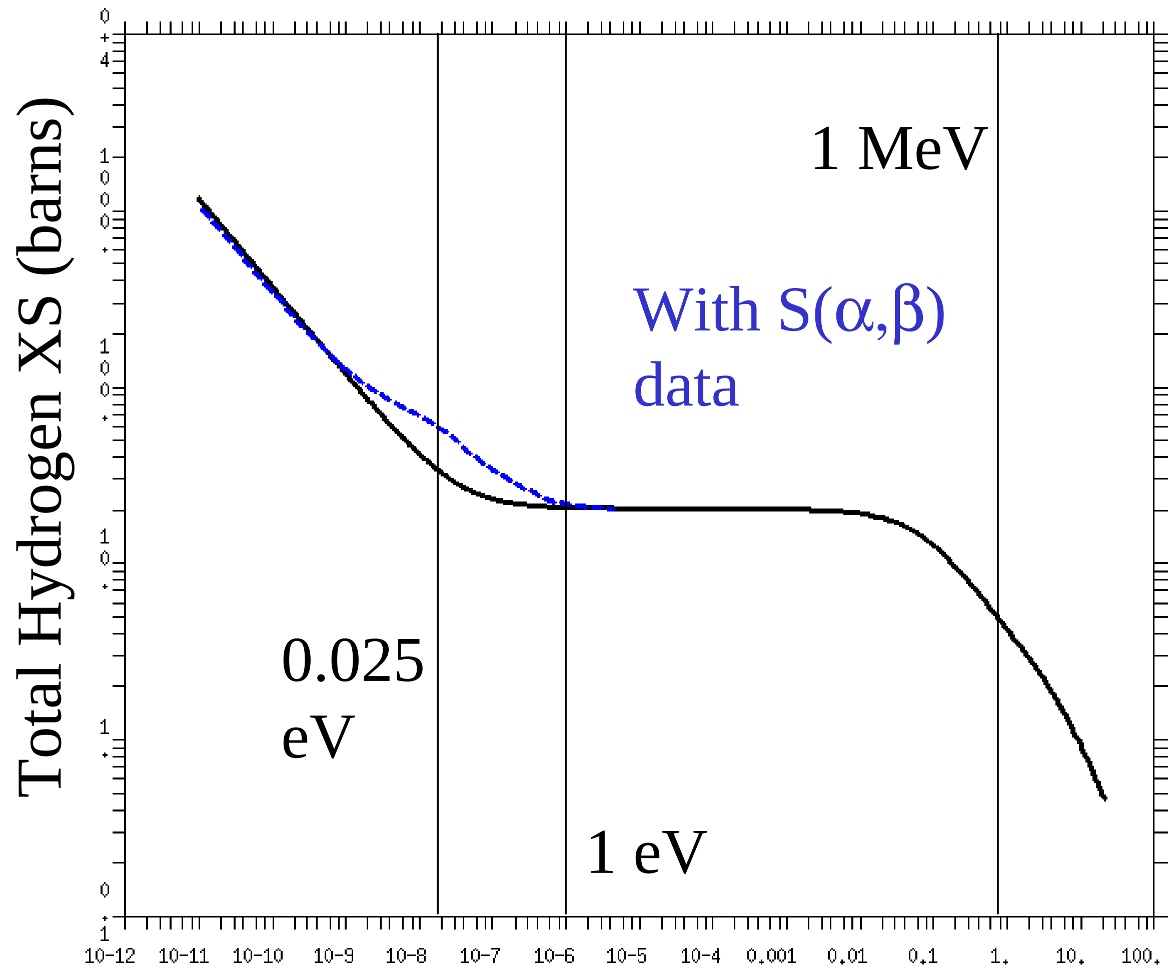


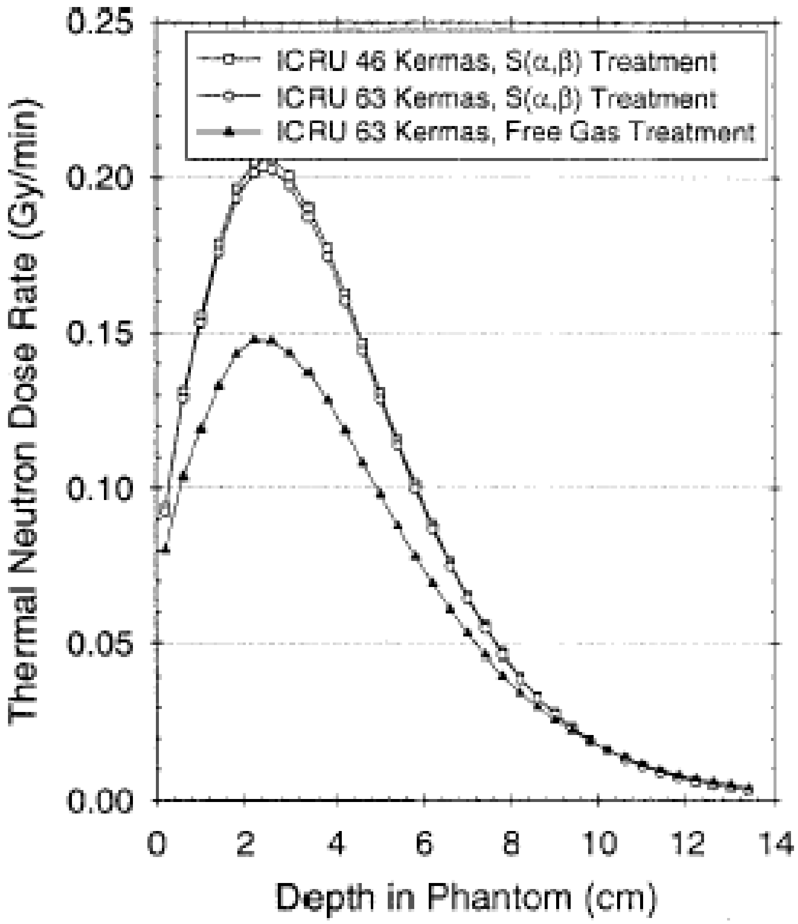
Image from MCNP5 plotter

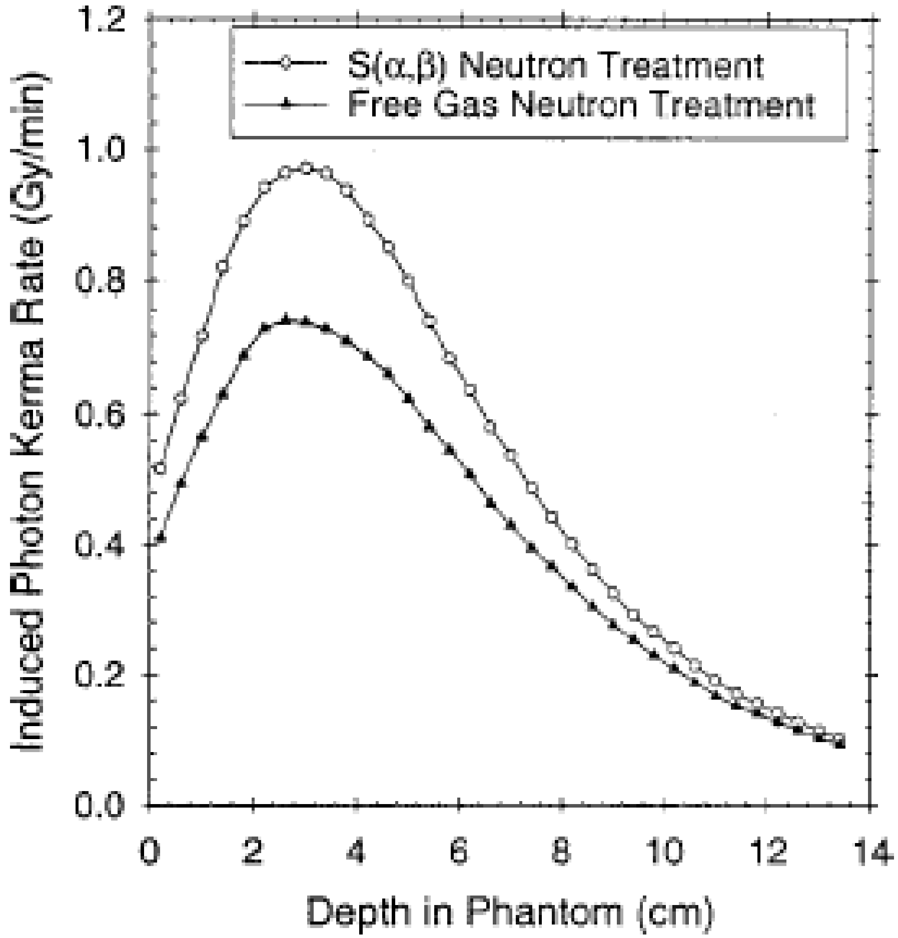
Neutron Energy

Neutron Scattering Treatment

- Use can cause significant differences.

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





Variance Reduction

- Exchange user time for computational time
- Few hours of user time often reduces computational time by 10-1000
- Truncation methods – truncates parts of phase space that do not contribute significantly
- Population control – use particle splitting and Russian roulette to control # samples in phase sp
- Modified sampling – alters statistical sampling of problem to increase # of tally contributions
- Partially deterministic methods – circumvent part of the random walk process by using know expected values.

Simple Variance Reduction

- Implicit Capture
  - Reduces weight of particle by probability of capture
  - Automatically on
  - WC1 parameter on PHYS card
  - Population control technique
- Geometry Splitting
  - Cause splitting or Russian Roulette when changing to cell of different importance
  - Change with the IMP card
  - Population control technique.

Simple Variance Reduction

- Point Detectors
  - Covered in Tally section of this workshop
  - F5 tally type
- Source Biasing
  - Sample from a fictitious density function instead of the true density function. This distortion must be corrected for by altering the particle’s weight.
  - SB card w/ SI SP cards
- Weight Cutoff
  - Kills particles whose weight falls below a certain limit
  - Automatically on
  - WC1 and WC2 parameters on CUT card

Verification & Validation

- Electron Benchmarks
- Computing Radiation Dosimetry - CRD
- QUADOS Code Comparison

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.

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

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

What MCNP5 Can't Do

- High-Energy Particles (muons, pions, etc.)
  - Will be available with MCNP6
- Proton Generation and Transport
  - Will be available with MCNP6
- Magnetic Field Tracking
  - Will be available with MCNP6
- Coincident Counting
  - lacks code and data
  - Monte Carlo 2005 Talk - An Upgraded Multidetector Pulse Height Tally For MCNP By Andriy Berlizov
- Photon Polarization

Obtaining MCNP

- Can be obtained from RSICC (even if outside US)
  - http://www-rsicc.ornl.gov/
  - Two CDROM versions
    - Executables, Source and Full Manual – limited release
    - Executables, no source, and Vol I & II of Manual – broader release
- All CDROMs Contain
  - 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

Help with MCNP

- Read the manual
- User forum: **mcnp-forum@lanl.gov**
- X-5 (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

Los Alamos National Laboratory
Diagnostics Applications Group (X-5)
Health Physics Measurements (HSR-4)

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

2005 MCNP Classes

- X-5:
- June 14-17: Introduction to MCNP - LANL
- June 27-July 1: Intermediate/Advanced - Tokyo
- Aug 23-25: Advanced Variance Reduction - LANL

- HSR-4:
- June 6-10: Practical MCNP for the Health Physicist, Medical Physicist, and Radiological Engineer - LANL

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

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- 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.