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MCNP session 4

Steve Lilley

STFC Rutherford Appleton Laboratory
Steven.Lilley@stfc.ac.uk

Outline

- Recap
- Criticality
- Criticality input
- Criticality output
- Extras – State of the art & assessed exercise tips

Recap - theory

- Monte Carlo – based on sampling distributions using random numbers
- Calculate mean behaviour
- more samples = closer to true mean
- In ‘fixed source’ radiation transport a history is a single source particle and any subsequently produced particles
- Source distributions, distance to next collision, type of interaction, emission angle all sampled by random number,
- the sampled distributions are not random!

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Recap - MCNP

- Geometry – surfaces used to define cells using Boolean logical operators "intersect" (and,) & "union" (or, :)
- Materials using ZAID
- Sources using SDEF
- Tallies – F2 surface, F4 cell, F5 point

Criticality

- Criticality assessment is a very important area concerned with safety, accident prevention and regulatory compliance
- Majority of events involve liquids, for detailed review of accidents see “A Review of criticality accidents” link
- Monte Carlo calculations are useful for predicting scenario’s and geometries which may lead to criticality accidents

Criticality – analytic formula

- Four factor formula
  - $k_{\infty} = \eta f p \epsilon$
- Six factor formula
  - $k = \eta f p \epsilon P_{FNL} P_{TNL}$
- Where
    - η = thermal fission factor
    - f = thermal utilisation factor
    - p = resonance escape probability
    - Pfnl =fast non leakage
    - Ptnl = thermal non leakage
    - ϵ = fast fission factor

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Criticality

- So far we have considered ‘fixed source’ problems
- However for criticality the exact source distribution is not known for each generation
- Calculating K_{eff} is an eigenvalue problem
- $K_{eff} = \frac{\#\, neutrons\ in\ generation\ n}{\#\, neutrons\ in\ generation\ n{-}1}$
- Most codes use a power iteration technique i.e. iterate the source until the answer converges

Criticality in MCNP - input

KCODE Criticality Source Card

Form: KCODE NSRCK RKK IKZ KCT MSRK KNRM MRKP KC8

NSRCK	=	number of source histories per cycle
RKK	=	initial guess for k_{eff}
IKZ	=	number of cycles to be skipped before beginning tally accumulation
KCT	=	number of cycles to be done
MSRK	=	number of source points to allocate storage for
KNRM	=	normalize tallies by 0=weight / 1=histories
MRKP	=	maximum number of cycle values on MCTAL or RUNTPE
KC8	=	summary and tally information averaged over 0 = all cycles 1 = active cycles only

Defaults: NSRCK=1000; RKK=1.0; IKZ=30; KCT=IKZ+100; MSRK=4500 or 2* NSRCK; KNRM=0; MRKP=6500; KC8=1

Use: This card is required for criticality calculations.

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Criticality in MCNP - input

- Cells, surfaces, materials all the same as for fixed source problems
- In addition to the *KCODE* card also need to define an initial source distribution
- Can be an *SDEF* as for fixed source problems
- Or can use:

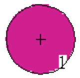
KSRC $x_1\ y_1\ z_1\ x_2\ y_2\ z_2\ \ldots\ldots\ x_n\ y_n\ z_n$

Criticality in MCNP- example

```
C simple U sphere
1 1 -19.2 -1 $ U sphere
99 0 1 $ grave yard

1 so 10 $ 10cm sphere on origin

Mode n
IMP:N 1 0
C Natural Uranium
M1 92238.42c 0.97 92235.42c 0.03
C crit cards
C 1000 neutron per gen, keff guess
C 0.8, skip 100 gen, run 1000 active gen
KCODE 1000 0.8 100 1100
KSRC 0 0 1
```



Criticality in MCNP

- Neutrons are tracked the same way as in fixed source problems,
  - Leakage, Moderation, Absorption, Fission
- The position of each fission event is saved
- This becomes a source point for the next generation
- Once Nsrck neutrons have been simulated, Keff is calculated and the next generation (cycle) is simulated

Criticality in MCNP - output

- Output is provided for each generation
- Although simulating generations of neutrons, it is not time dependant, there is only one value of Keff
- It should converge!
- 3 estimates of Keff are provided
  - From collision events
  - From absorption events
  - Track length estimate
- As well as Keff, standard deviation is provided to estimate the statistical uncertainty

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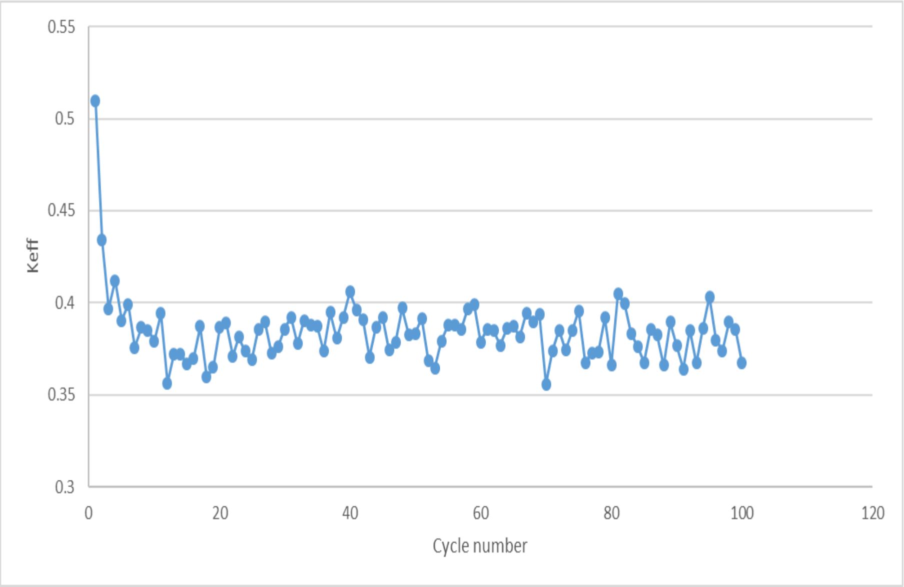
Parameter choosing

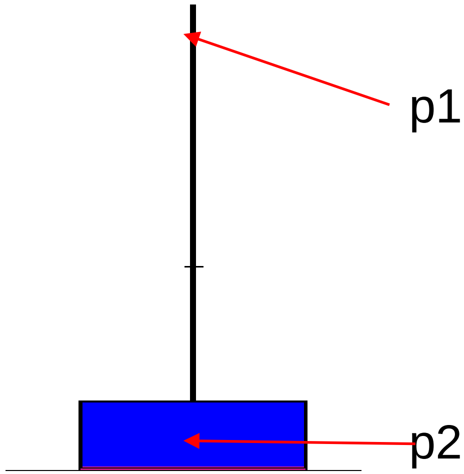
- Number of particles per cycle (Nsrck)
  - Shouldn't effect the answer unless very low only the time to convergence and final uncertainty
  - Consider size and distribution of fissile material
  - Typically at least 1000
- Number of cycles to skip and total number to run (IKZ & KCT)
  - Make a sensible estimate, run evaluate if enough and repeat
  - Need to know required accuracy



Choice of initial source distribution

- Eventually after enough generations the code will reach a stable value of Keff, the choice of starting source distribution and number of neutrons per generation can affect how fast it converges

- Try ensure largest fissile material cells are covered,
- Preferably put a point or distribution in all fissile cells



# generations	Keff, p1	Keff p2
500	0.033	1.041
1000	0.033	1.039
5000	0.033	1.038
100000	1.042	1.038

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Extras

- State of the art MCNP
- Hints and tips for assessed exercise

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Current state of the 'art'

- MCNP6 released 2014, 6.2 is current version
- Multiphysics is a current buzz word
- As are 'mesh geometry' and 'hybrid methods'
- As computers get more powerful the problems get 'bigger' and more complex

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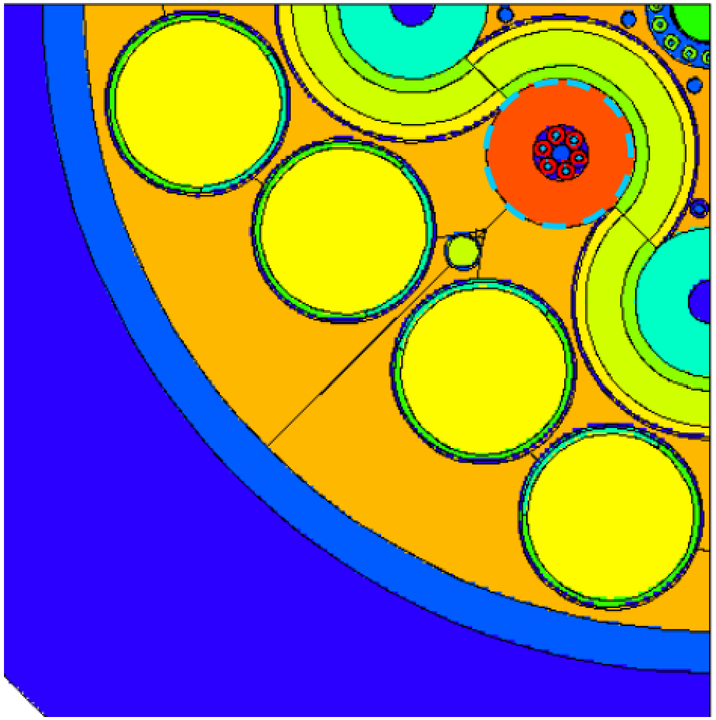
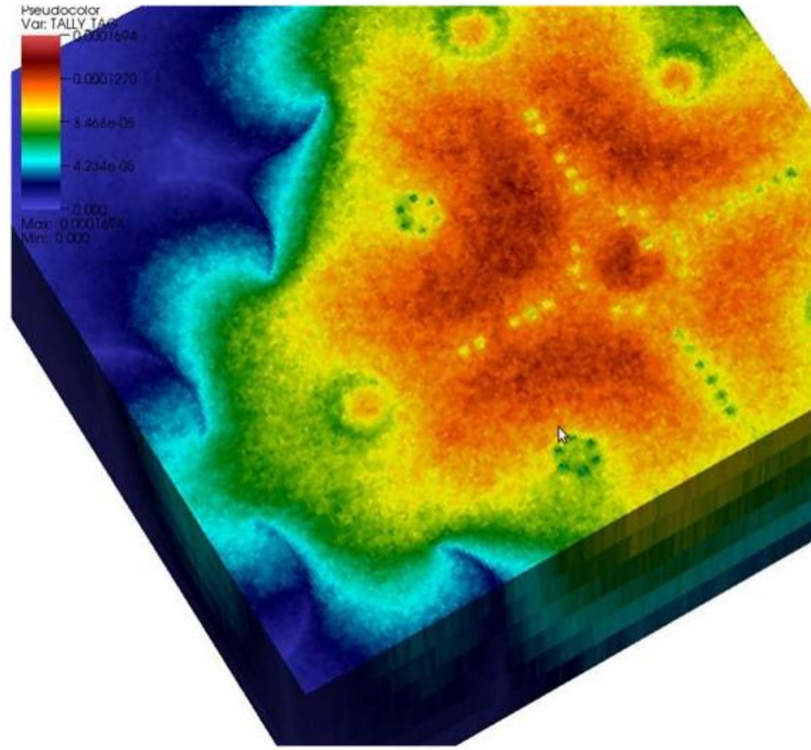
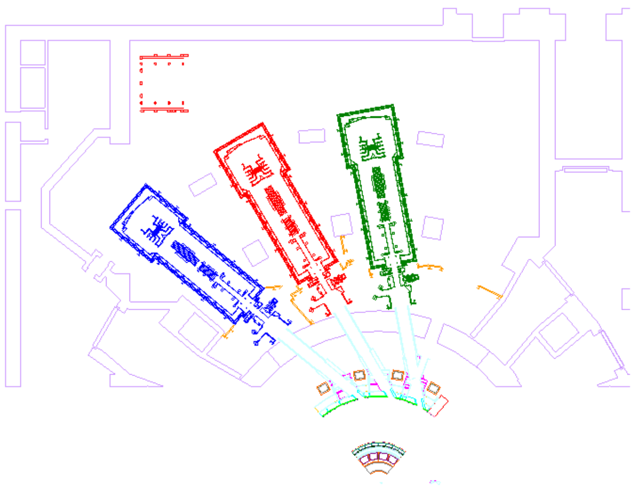


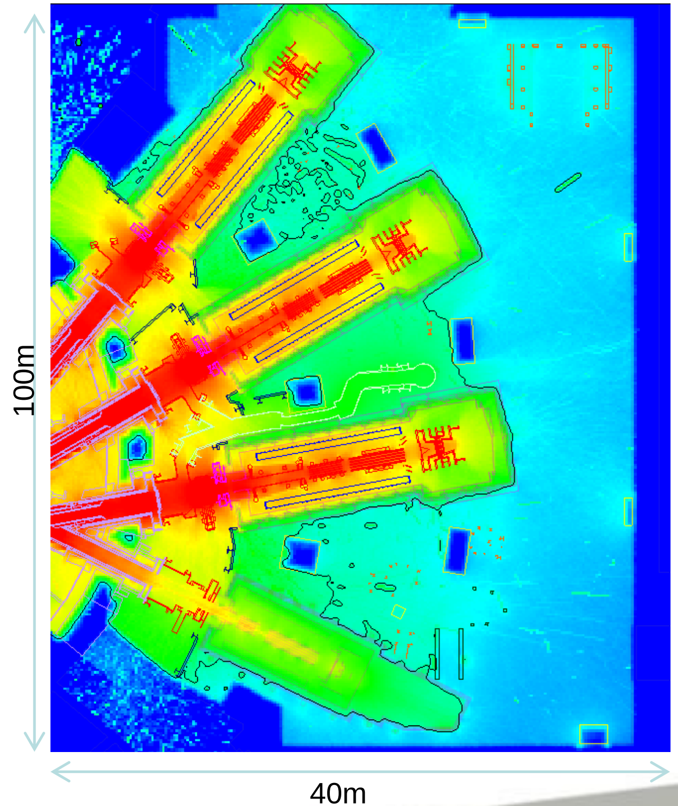


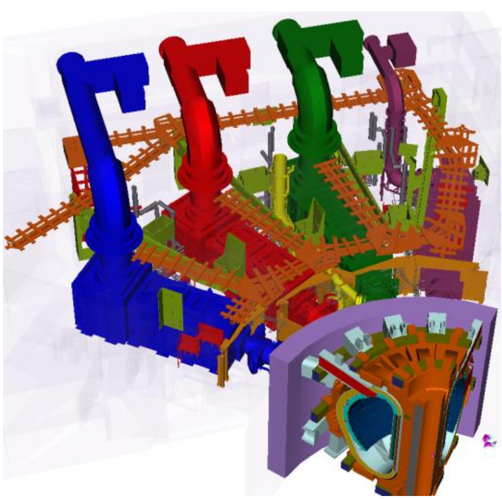
Fusion, ITER tokamak

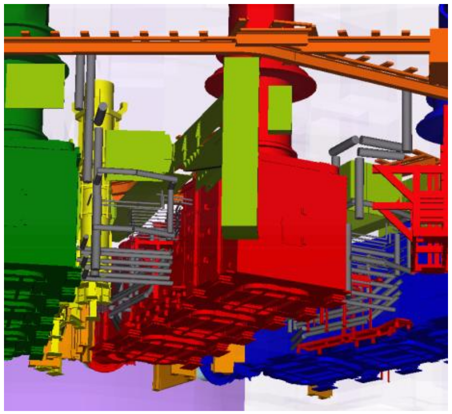


30k MCNP cells
100k surfaces

Work performed at CCFE







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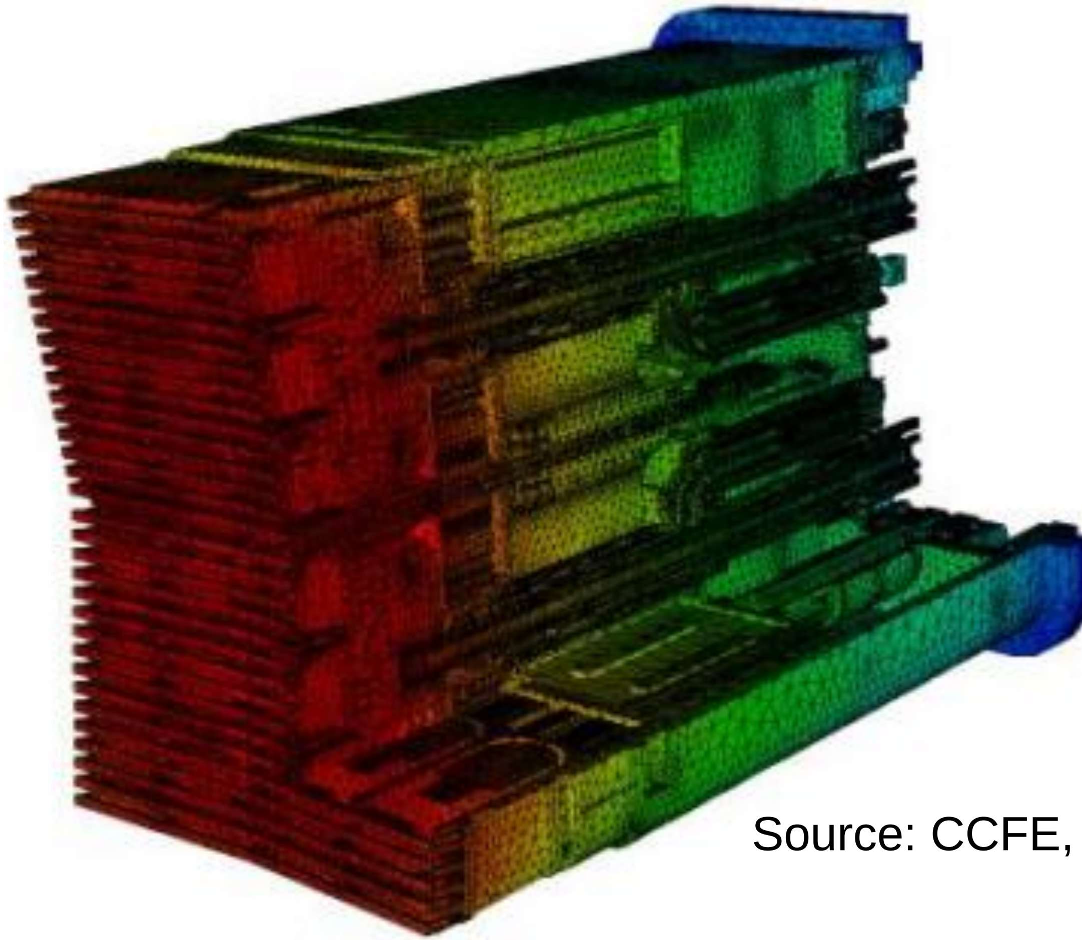
Mesh based geometry (MCNP6)

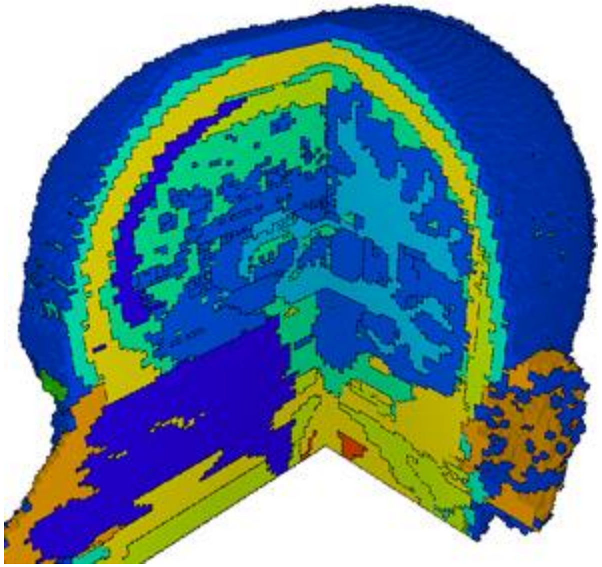
Mesh geometry enables better linking to engineering FEA codes.

This should enable multi-physics simulations in the future



Source: CCFE, A Turner

Human phantoms



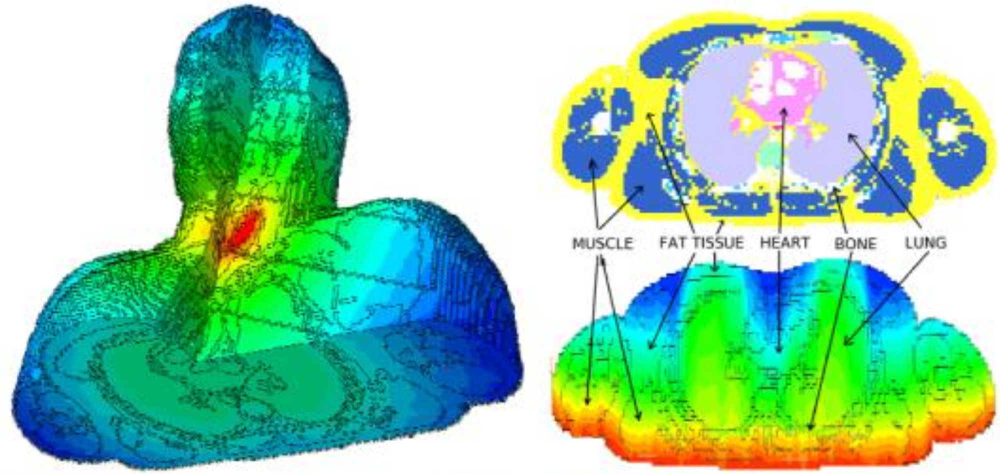
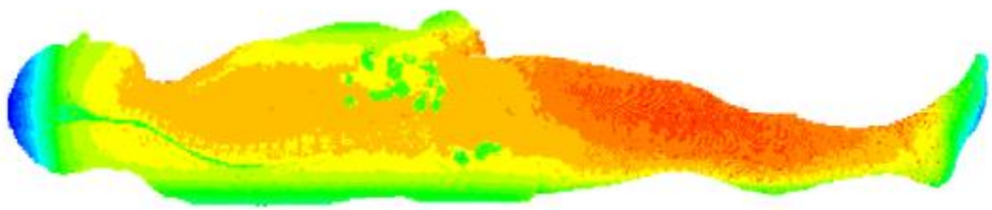


Figure 3: Simulating an Iodine source in the thyroid (left) and a posterior-anterior directional gamma source (right)

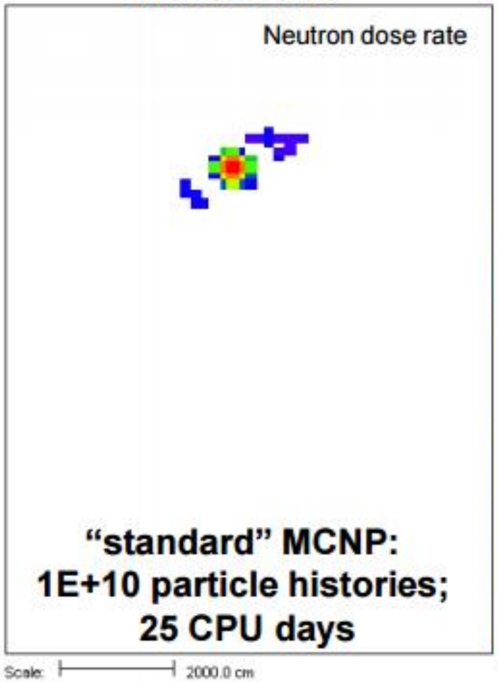
Source: Anderson et al, Mesh Human Phantoms with MCNP

Source: VISED website

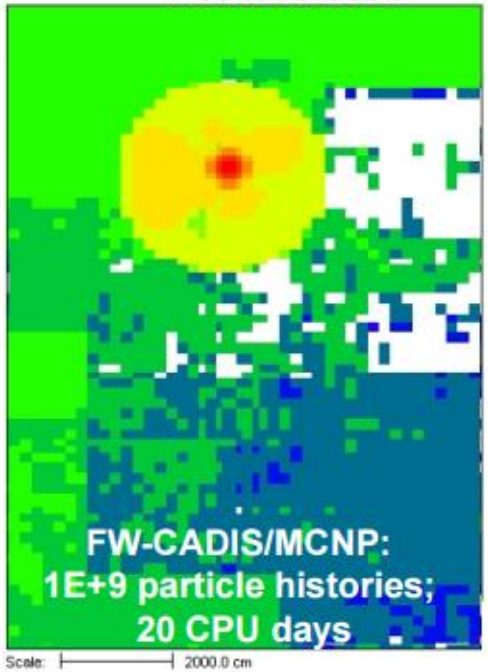


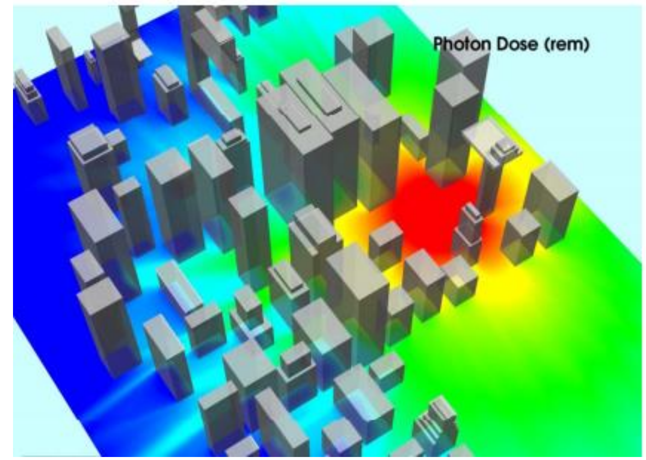
Hybrid simulations

Simulation not possible with "standard" Monte Carlo



Simulation enabled with hybrid methods/code





Source: ORNL radiation transport group

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Assessed Exercise tips

- Short paper style report – 5 page limit + figures
- Use section headings
- Figures are good, if they clearly show something worth showing!
- When comparing 2 similar values consider if a ratio would make the comparison clearer
- Remember answers are only as good as the uncertainty.
- Don't waffle – be precise

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Assessed Exercise tips

- As well as the report a MCNP input files should be provided.
- Please ensure it is clear that it is yours
  - add a comment block at top including your name
- Use comments
- Do exercise 1 and either 2 or 3

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Don’t forget to check your work

- Check dimensions and volumes, MCNP is in cm, most engineers deal in mm
- Check source position and distribution using 1st 50 particles
- Most F2 and F4 tallies cost very little computational time – add liberally, cheaper to add then need to recalculate
- MCNP is not a mind reader it doesn’t know what you mean and the defaults are rarely what you expect
- The manual and primer(s) are very helpful
- Google and you tube, not so helpful

Questions