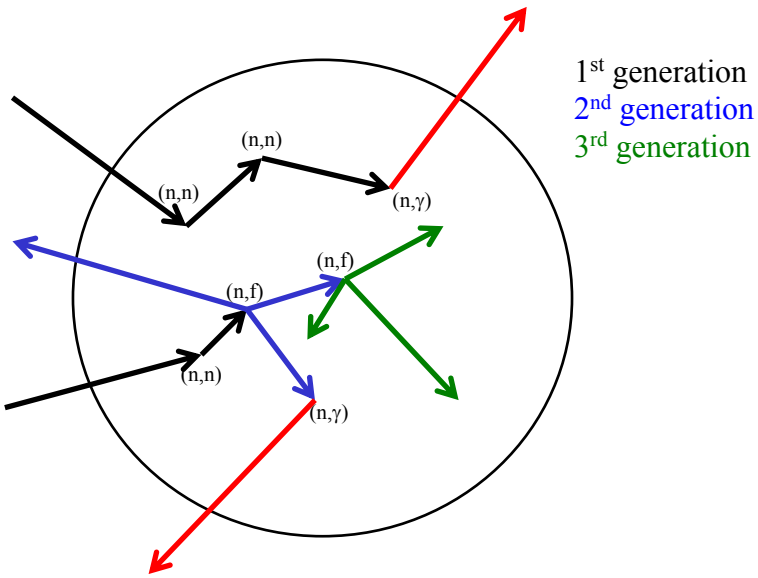
MCRT L9: Neutron Transport

- Outline of code for mono-energetic particles, if change particle energy must change cross sections
- Bank locations of fission neutrons for subsequent generations
- Criticality calculations, issues with regular MC neutron generations increasing or dying, method to keep same number of MC particles

Neutron Generations



Distance to interaction

- Usual optical depth sampling: $\tau = -\ln \xi$

$$\tau = \int_0^L n\,\sigma\,\mathrm{d}s$$

- n = number density of material
- Total cross section is sum of cross sections for absorption, scattering, and fission: $\sigma_T = \sigma_A + \sigma_S + \sigma_F$
- Depending on geometry, determine L analytically or numerically (e.g., using grid code)
- Uniform density uranium sphere: $\rho_U \sim 19$ g/cm^3, so $n_U \sim 4.8$ E22 cm^{-3}
- For 1MeV neutrons in U235: $\sigma_T \sim 5.1$b, $\sigma_A \sim 0.1$b, $\sigma_S \sim 4$b, $\sigma_F \sim 1$b
- $L = \tau / (n\,\sigma_T) \sim 4.1\,\tau$ cm

Scatter, absorb, or fission?

- Probability of fission = σ_F / σ_T, so use this algorithm to decide whether neutron is scattered, absorbed, or produces fission:

```
σT = σA + σS + σF
ran = ξ
if (ran .lt. σA/σT) then
  neutron absorbed
elseif(ran .lt. (σA+σS)/σT) then
  neutron scattered
else
  fission event
endif
```

Scatter, absorb, or fission?

- Scatter: assume isotropic scattering as with photon scattering simulation, so new directions for neutron are:

$$\theta = \cos^{-1}(2\xi - 1) \qquad n_x = \sin\theta \cos\phi$$
$$\phi = 2\pi\,\xi \qquad n_y = \sin\theta \sin\phi$$
$$n_z = \cos\theta$$

- Absorb: terminate neutron and start next neutron in the loop
- Fission: decide how many fission neutrons to be emitted and add their locations to the bank

If probabilities of emitting 1, 2, 3,.. fission neutrons are p1, p2, p3,.. can use this algorithm:

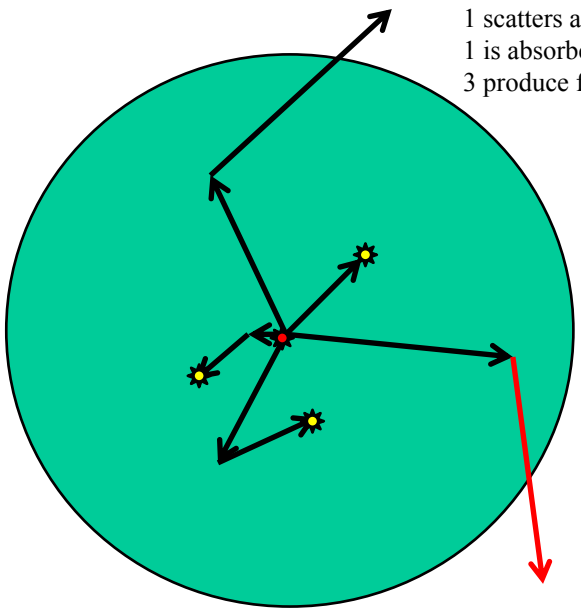
```
p = p1 + p2 + p3
ran = ξ
if (ran .lt. p1/p) then
  ! add one fission neutron to the bank
  ibank = ibank + 1
  xbank(ibank) = x    ! Same for y and z
elseif(ran .lt. (p1+p2)/p) then
  ! add two fission neutrons to the bank
  do j=1,2
    ibank = ibank + 1
    xbank(ibank) = x   ! Same for y and z
  end do
elseif(ran .lt. (p1+p2+p3)/p) then
  ! add three fission neutrons to the bank
  do j=1,3
    ibank = ibank + 1
    xbank(ibank) = x   ! Same for y and z
  end do
endif
```

ibank is a counter for number of fission neutrons. Set ibank = 0 at start of each generation loop over neutrons.

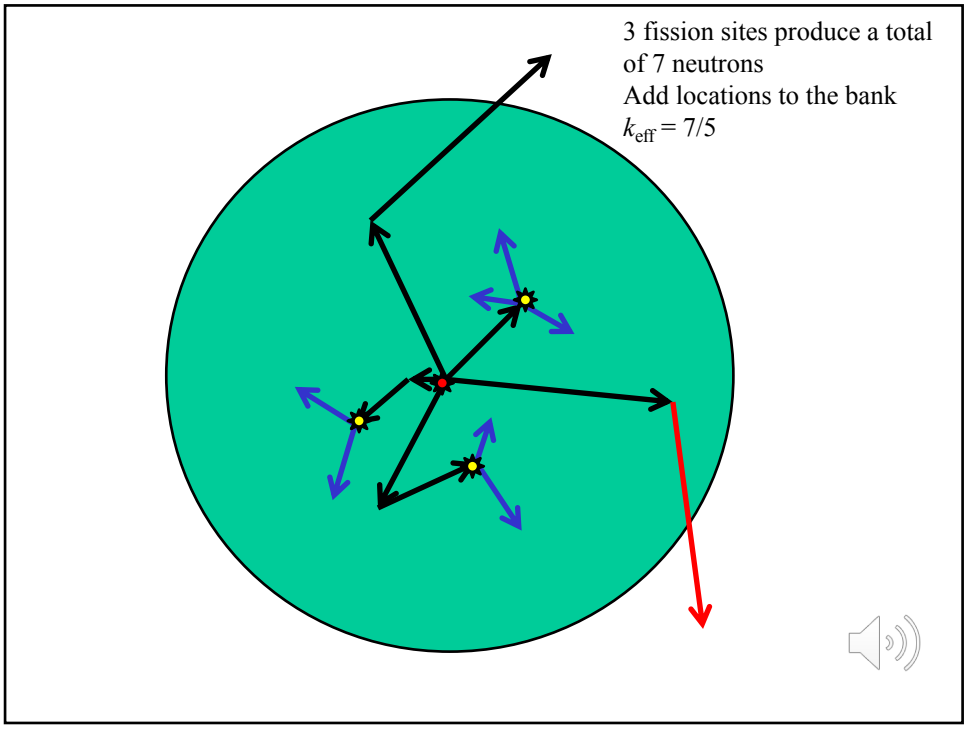
At end of the loop over the current neutron generation, the value of ibank is the number of fission neutrons produced. Can then compute k_{eff} = ibank / N, where N = number of neutrons emitted in the current generation.

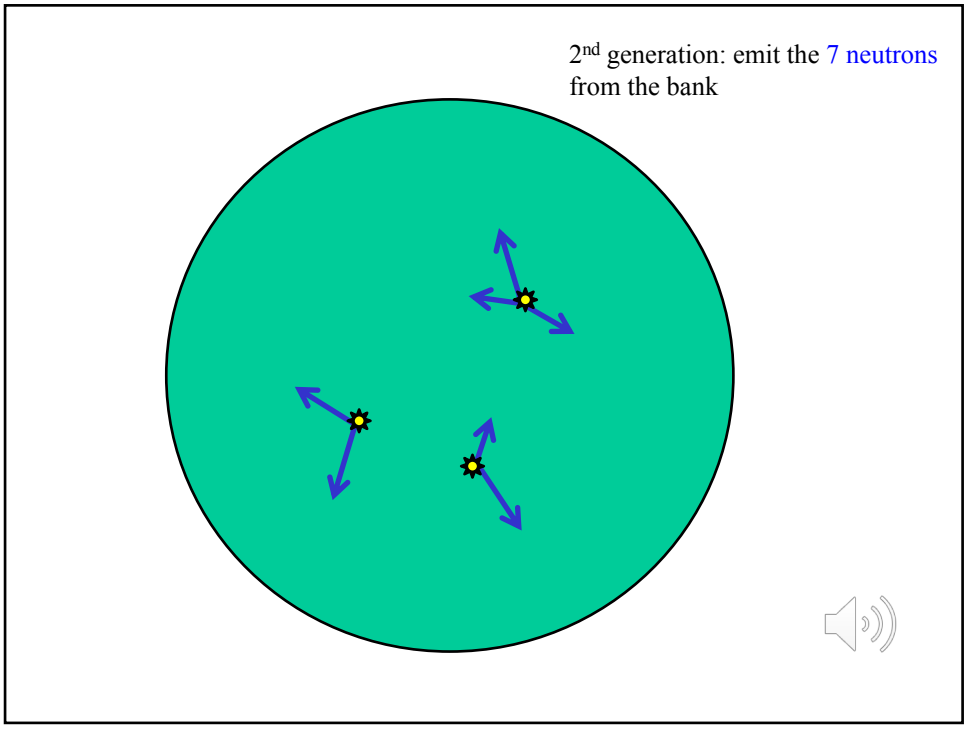
Neutron Transport & Criticality: Analog Monte Carlo

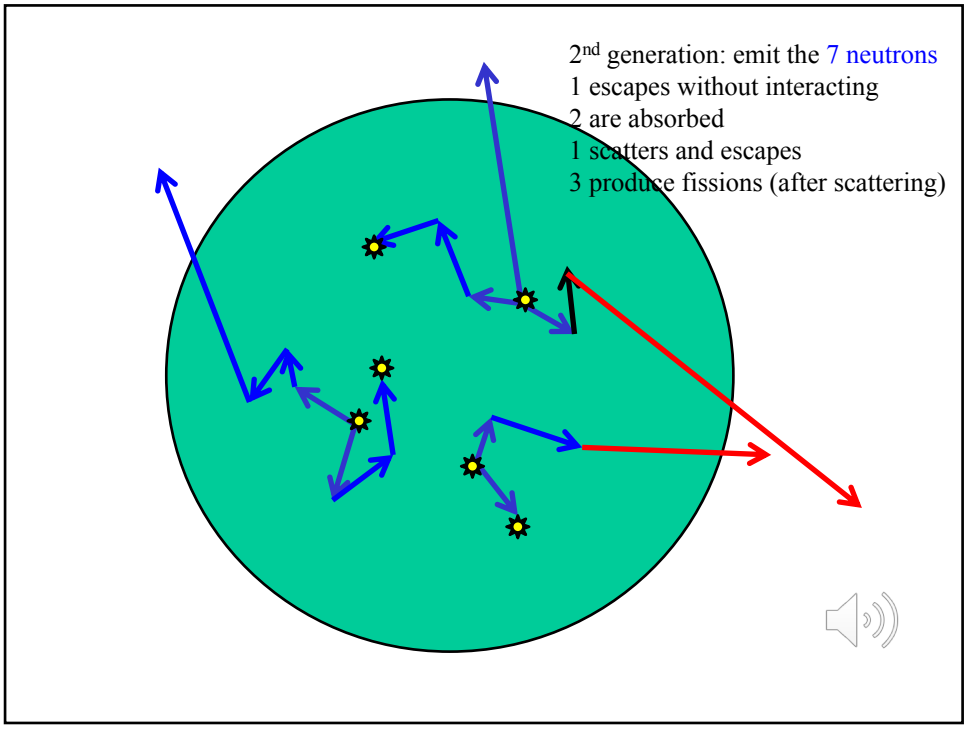
- Loop over initial generation of neutrons – either from a point or some guess of the final distribution, or uniformly throughout the medium
- Counter for how many neutrons go into the bank – ibank
- k_{eff} = ibank(generation i) / ibank(generation i-1)
- Loop over multiple generations of neutrons until exceed space you have allocated in the bank or bank is emptied (population dies out)
- In each generation, loop over all the neutrons in the bank, i.e., emit all neutrons in the bank for each generation

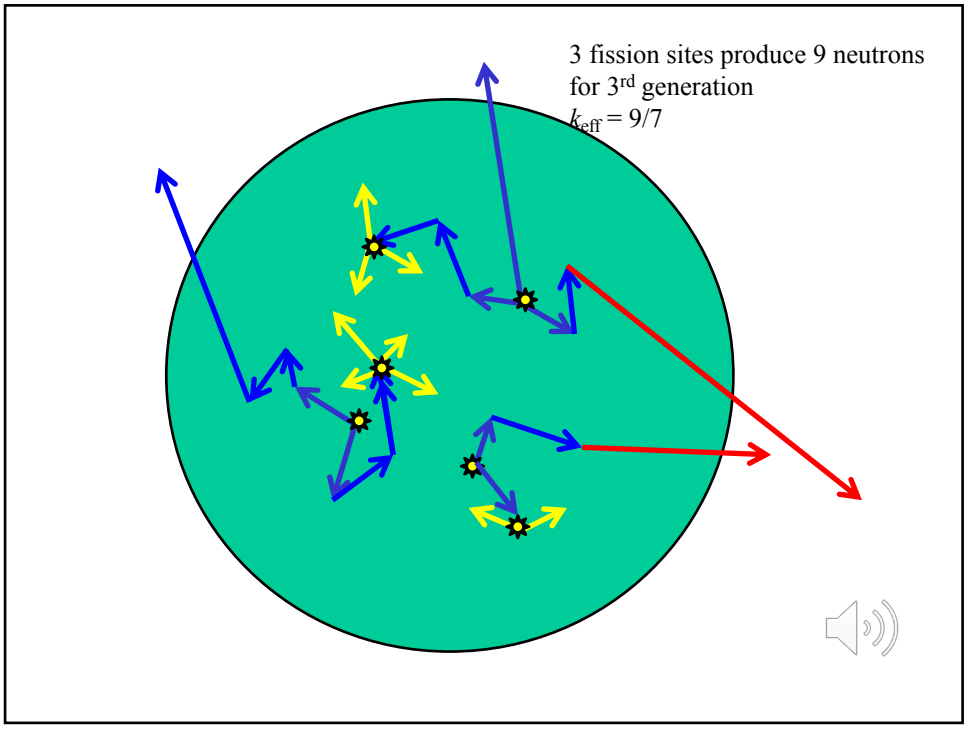


Launch 5 neutrons:
1 scatters and escapes
1 is absorbed (produces γ-ray)
3 produce fissions

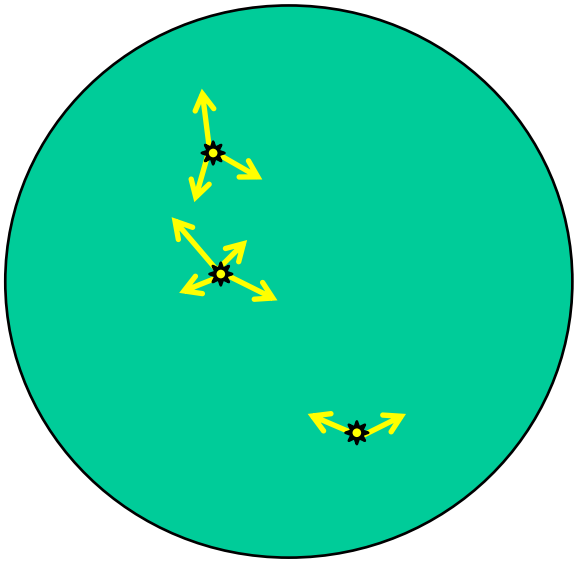

3 fission sites produce a total of 7 neutrons
Add locations to the bank
$k_{eff} = 7/5$


2nd generation: emit the 7 neutrons from the bank


2nd generation: emit the 7 neutrons
1 escapes without interacting
2 are absorbed
1 scatters and escapes
3 produce fissions (after scattering)


3 fission sites produce 9 neutrons
for 3rd generation
$k_{eff} = 9/7$

Track the 3rd generation neutrons…



Make the position bank 2D arrays: xbank(nbank, 2), ybank(nbank, 2), zbank(nbank, 2), where nbank = a large number (say one million)

First generation, store the locations of fission neutrons at xbank(ibank, 1) = x, recall we increment the counter ibank and at end of the generation ibank is the total number of fission neutrons produced

Second generation, emit i-th neutron from xbank(i,1), store fissions in xbank(ibank, 2)

Third generation, emit i-th neutron from xbank(i,2,) store fissions in xbank(ibank, 1)

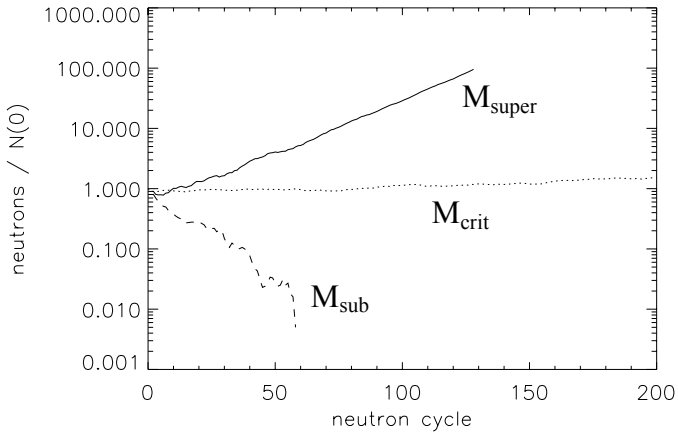
Need an algorithm so that the second index of the bank arrays for storing locations of fission neutrons cycles from 1 to 2 depending whether the generation is odd or even:

```
if(mod(gen,2) .eq.0) then
   current = 1
   next = 2
else
   current = 2
   next = 1
endif
```

When emitting i-th neutron for generation "gen", use source locations xbank(i,current)

When banking neutrons produced in generation "gen", store the locations in xbank(ibank,next)

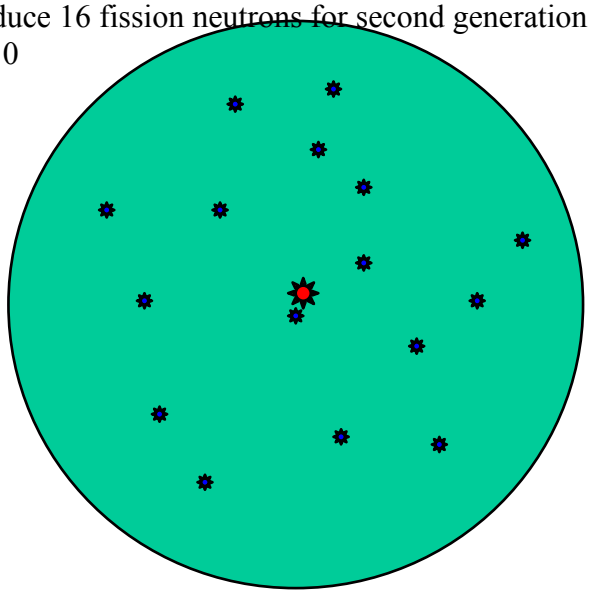
Critical Mass of Uranium Sphere



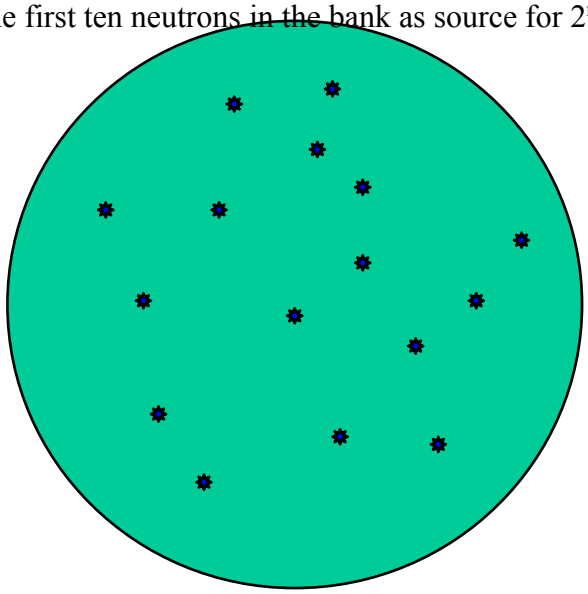
Alternative criticality calculation

- Criticality calculations, issues with number of neutrons in *analog* or *regular* MC generations increasing or dying out
- Algorithm to prevent runaway increase or decrease of number of particles being tracked – emit N Monte Carlo neutrons at every generation
- Emit N neutrons in first generation, bank the locations of fission neutrons as before
- Emit N neutrons in the 2nd generation – take the locations to be either the first N from the bank (if N < nbank) or take all neutrons in the bank and then take those from the start of the bank until have N for the 2nd generation
- Randomly sample from the "relaxed" distribution of neutrons – just take the first N from the bank (they are produced randomly anyway) if generate more than N. If generate less than N, then cycle through the bank multiple times until have N.

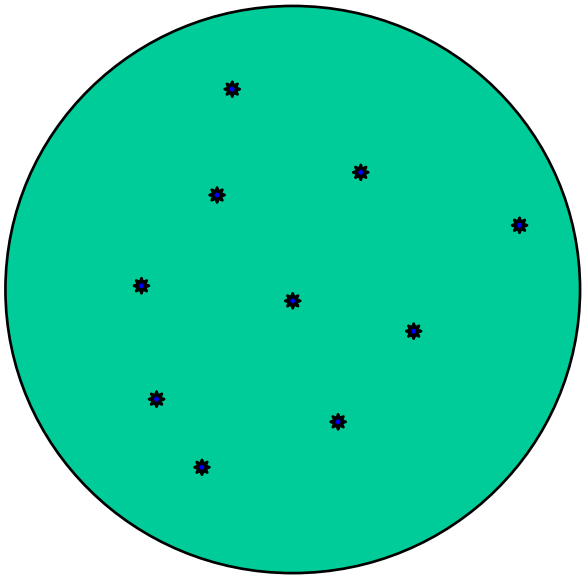
Emit 10 neutrons from centre of sphere
They produce 16 fission neutrons for second generation
$k_{eff} = 16/10$



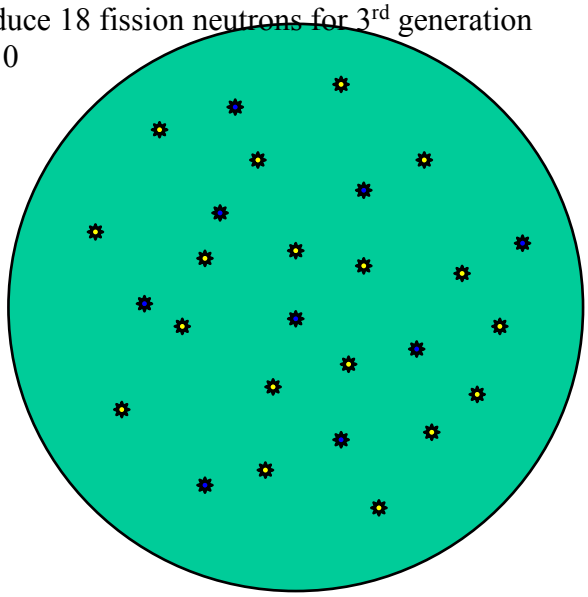
Have an increasing population of neutrons
Choose the first ten neutrons in the bank as source for 2nd generation



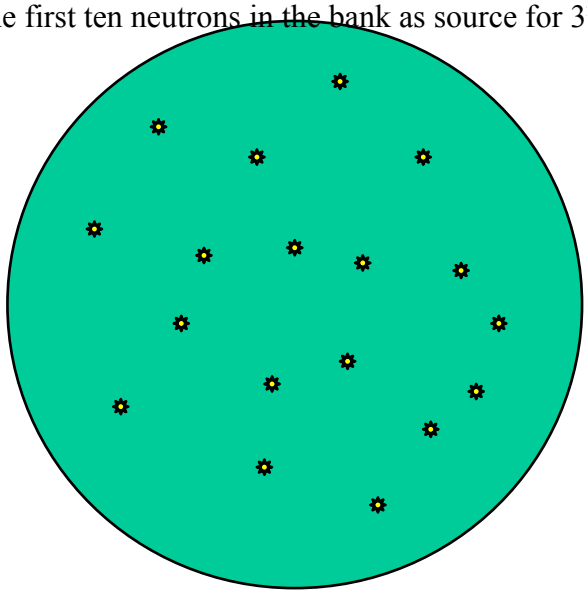
Choose the first ten neutrons in the bank as source for 2nd generation



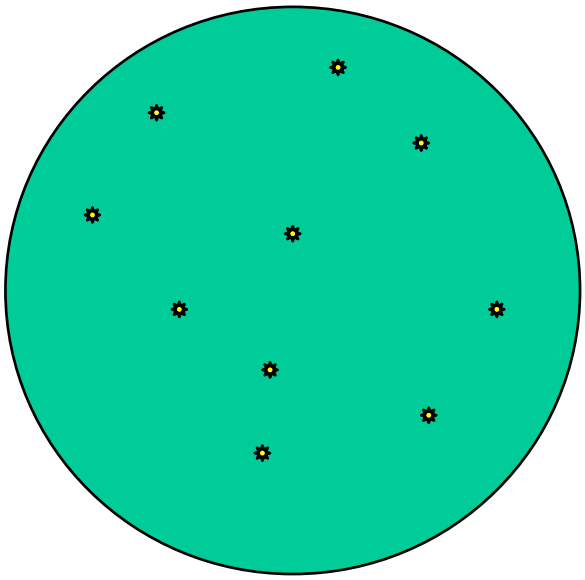
Emit 10 neutrons in 2nd generation
They produce 18 fission neutrons for 3rd generation
$k_{eff} = 18/10$



Have an increasing population of neutrons
Choose the first ten neutrons in the bank as source for 3nd generation

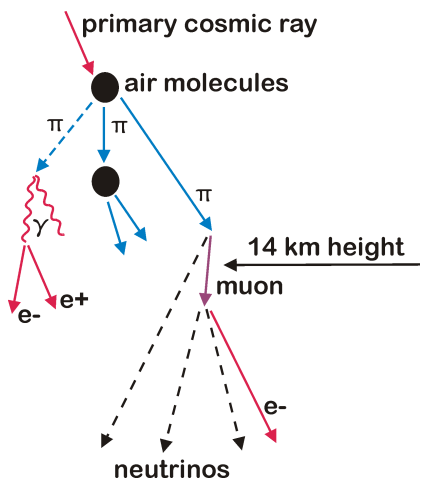


Choose the first ten neutrons in the bank as source for 3rd generation



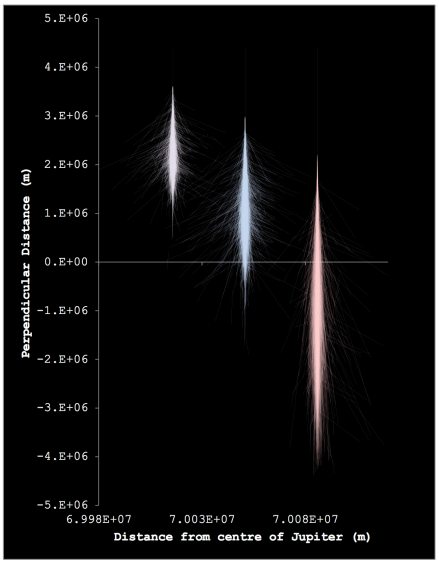
- In reality there are more than N neutrons in a real life situation, so our N Monte Carlo neutrons are a statistical sampling of the actual number of neutrons
- Neutron histories are all randomly generated, so it is ok to choose source locations from the bank as described above
- k_{eff} is then always ibank/N from generation to generation, where ibank is updated as in analog simulation for banking neutron locations

Cosmic Ray Air Showers



- Use similar techniques of generations of particles
- Need probabilities for producing different particles
- Need interaction cross sections to determine distances traveled by particles between interactions
- High energy cosmic ray can generate 10^{12} particles
- Can't follow all 10^{12} particles so use techniques to statistically sample particles in the shower: "thin sampling"

MPhys project in 2015



Starting from 3D grid code + ideas for banking from neutron code, developed MCRT code for cosmic ray air showers in Jupiter's atmosphere

He learned good…

28 September 2015

Hi Kenny,

How are you doing? I hope your work and everything else are doing well.

My new job as a trainee actuary is going excellently. I've come across more than a few Monte Carlo simulations where different economic scenarios are run in order to see the effects on a particular pension fund. I'm not quite at the level where I'd actually be writing these models, but perhaps in the a few years!

All the best,
Guy

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Criticality Safety Assessors Trainees

Provision of criticality safety advice and assessments to support the maintenance and delivery of the company's capability.

Sub-Discipline Criticality
Salary Range £21,000 - £42,000 dependent on experience.
Package Details Benefits include 28 days holiday and final salary pension
Location Aldermaston, nr Reading, Berkshire
Employment Basis Full Time
Job Reference 401262L

Key accountabilities

The delivery of criticality safety assessments and the provision of criticality safety support and advice to the operational facilities or project management teams.

Responsibilities

- Undertake criticality safety assessments and provide advice to existing facilities undergoing operational changes.
- Advise and support the design team for new facilities and projects in respect of criticality safety, including provision of analyses needed to support staged safety reports

- Participate in Review, Learn and Improve activities and implement the outcomes to ensure the continued development and implementation of best practice in the delivery of criticality safety
- Support external activities as required as part of the Criticality Safety Authority Outreach and Science Programme, i.e. through presentations to external conferences/symposia
- Acquire the core competences needed to ensure that work undertaken is in line with best practice in criticality safety.

Qualifications

- Minimum Upper 2nd Class Honours Degree in Physics, Mathematics, Chemistry or Engineering based degrees.
- MSc or PhD with a strong nuclear physics component will be advantageous.
- Membership of a Chartered Institute, desirable or clear evidence of working towards Chartered Status

Knowledge

- Good degree level understanding of nuclear physics.
- Ideally, knowledge of nuclear transport codes - Monte Carlo and deterministic methods
- Knowledge of the physics of criticality (and an understanding of the chemistry and metallurgy of fissile materials).

Experience