L-L-T group notes

Monday, April 19, 2021

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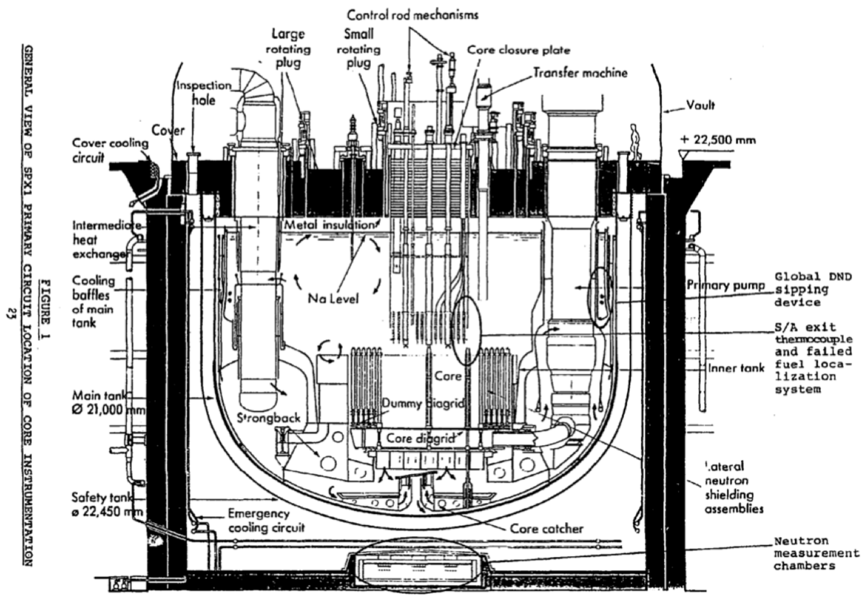
Superphénix Reactor

In Pile Neutron Monitoring

- For the future reactors it is intended to use high temperature fission chambers directly installed in the hot plenum above the core



IAEA Specialist Meeting on Instrumentation for supervision of core in fast breeder reactors 12-15 December 1989 - KALPAKKAM INDIA
https://inis.iaea.org/collection/NCLCollectionStore/_Public/22/085/22085482.pdf

In plenum Neutron Detectors
(Follow – up)

Proposed boron-carbide-based solid-state neutron detector

- **https://aip.scitation.org/doi/10.1063/1.1823579**
- boron-carbide-based thermoelectric device for the detection of a thermal-neutron flux

Development of a Neutron Flux Monitoring System for Sodium- cooled Fast Reactors

- http://uu.diva-portal.org/smash/get/diva2:1088118/FULLTEXT01.pdf
- Feasibility study of self-powered neutron detectors (SPNDs) with platinum emitters as in-core power profile monitors for SFR

In plenum Neutron Detectors
(Follow – up)

A class of boron-rich solid-state neutron detectors

- Appl. Phys. Lett. **80**, 3644 (2002); https://doi.org/10.1063/1.1477942 Submitted: 02 January 2002 . Accepted: 13 March 2001 . Published Online: 07 May 2002

Characterization of boron carbide thin films fabricated by plasma enhanced chemical vapor deposition from boranes

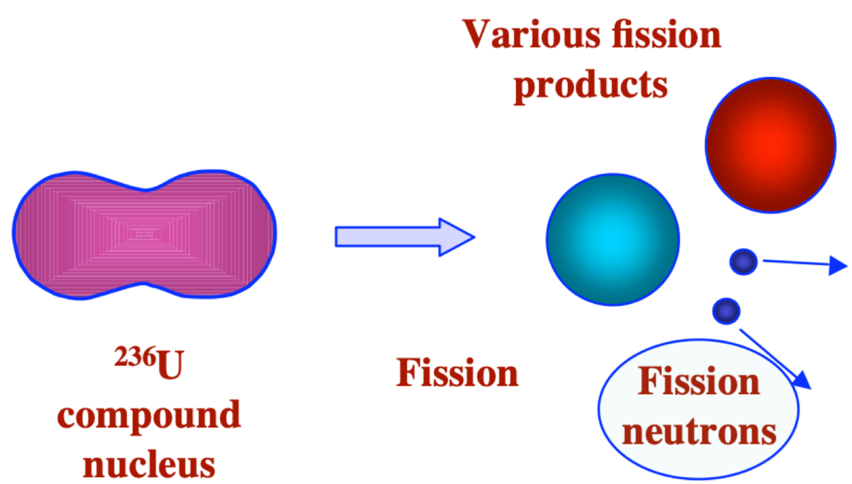
12 Jan. 2021

Reactor Theory

Reactor Theory

Fission products will obey fission **yield curve.**

- Generic $^{235}_{92}U$ fission event



More Next time

https://indico.cern.ch/event/145296/contributions/1381141/attachments/136909/194258/lecture24.pdf

Neutron Activation Analysis

"Prompt Fission Neutron Energy Spectrum of ${}^{1}_{0}n + {}^{235}_{92}U$

- Jason M. McGinnis
- University of Kentucky
- Theses and Dissertations--Physics and Astronomy. 63.
- https://uknowledge.uky.edu/physastron_etds/63

https://uknowledge.uky.edu/cgi/viewcontent.cgi?article=1065&context=physastron_etds

Neutron Activation Analysis

Using Reactor neutron spectrum

“**Beta minus decay of a neutron”**

Alone in a vacuum:

$$ {}^{1}_{0}n \rightarrow {}^{1}_{1}p + {}^{0}_{-1}e + {}^{0}_{0}\bar{\upsilon}_e. \quad (t_{1/2} \approx 10.2\ min) $$

http://www.jsnct.jp/e/about_nct/hannou.html

Neutron Activation Basics

Neutron Reactions with $^{16}_{8}O$

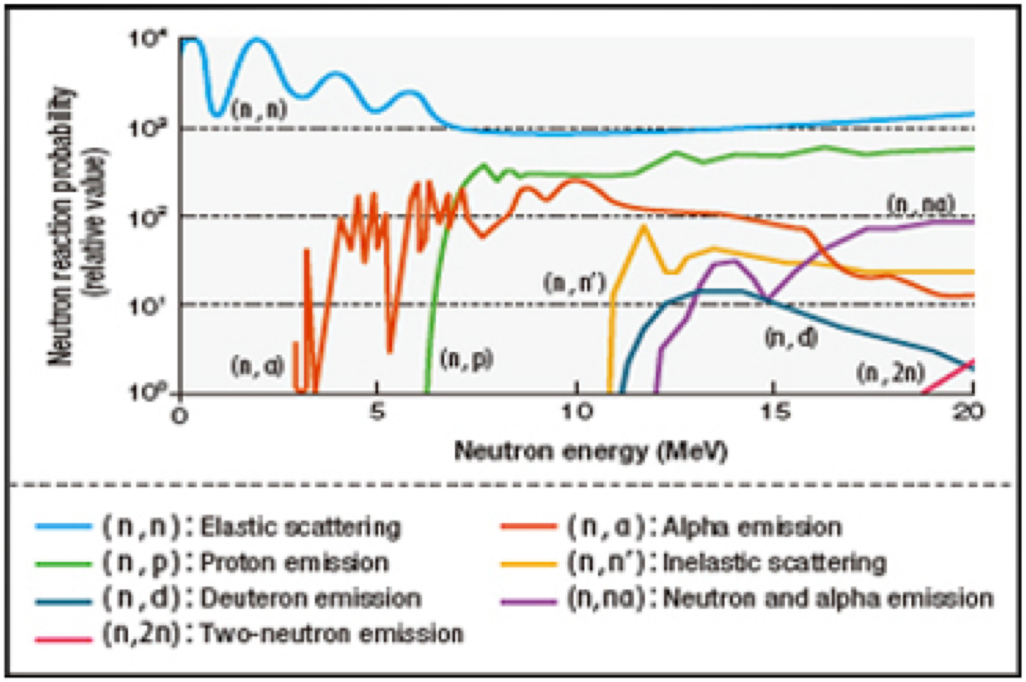
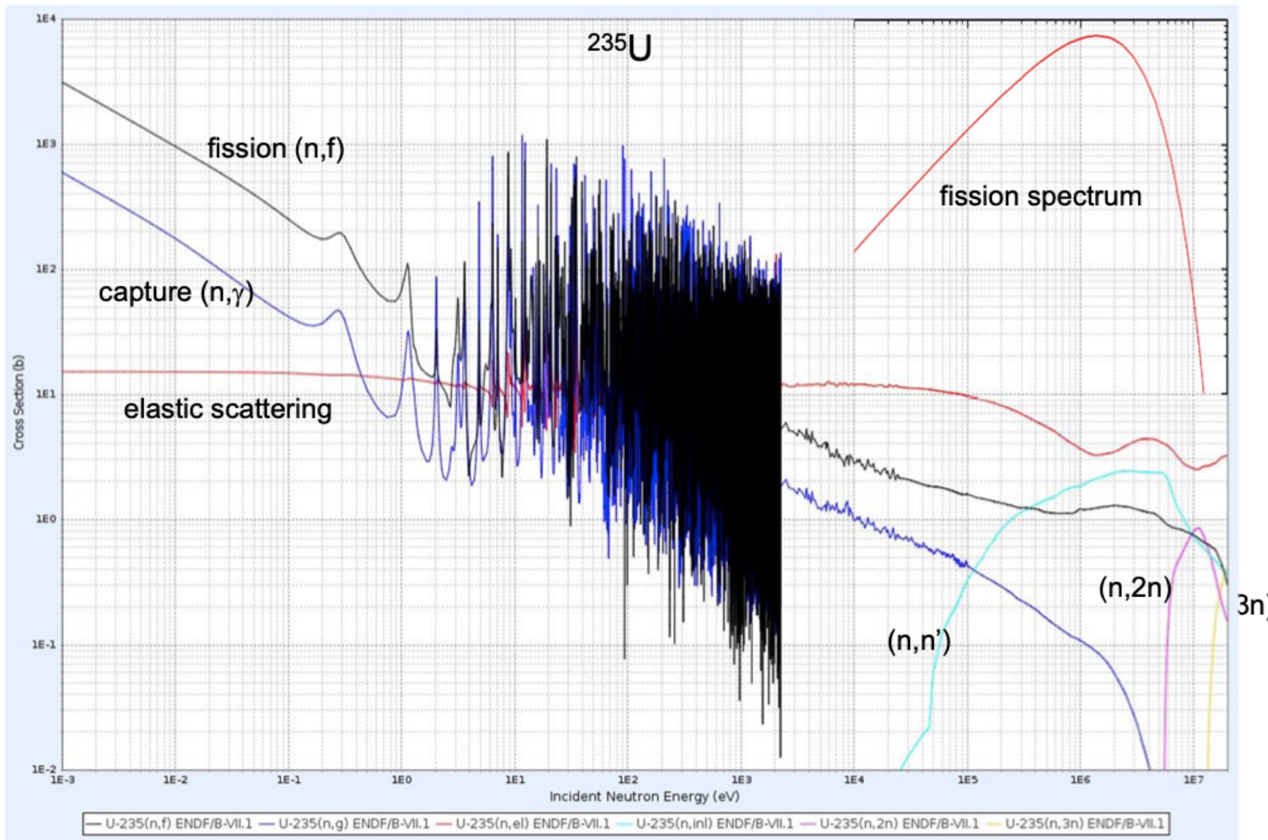


Figure 1. (for example)
Various Nuclear Reactions between ^{16}O and Neutrons

http://www.jsnct.jp/e/about_nct/hannou.html

Neutron Activation Basics

Neutron Reactions with ${}^{235}_{92}U$



http://kthn.dlu.edu.vn/Resources/Docs/SubDomain/kthn/2.NeutronInteraction.pdf
http://atom.kaeri.re.kr

Neutron Interaction Basics

$$(n, n)$$

$$(n, n')$$

$$(n, 2n)$$

$$(n, p)$$

$$(n, \alpha)$$

$$(n, f)$$

http://kthn.dlu.edu.vn/Resources/Docs/SubDomain/kthn/2.NeutronInteraction.pdf
http://atom.kaeri.re.kr

Basic Neutron Interaction Examples

Type of Event	**Reaction**
Elastic Scattering	(n,n): $\;{}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{\;\;\;Z}X)^* \rightarrow {}_Z^AX + {}_0^1n$
Inelastic Scattering	(n,n'): $\;{}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{\;\;\;Z}X)^* \rightarrow {}_Z^AX + {}_0^1n' + \gamma$
Capture gamma emission	(n,γ): $\;{}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{\;\;\;Z}X)^* \rightarrow {}^{A+1}_{\;\;\;Z}X + \gamma$
2 neutron emission	$(n,2n)$: $\;{}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{\;\;\;Z}X)^* \rightarrow {}^{A-1}_{\;\;\;Z}X + 2{}_0^1n$
3 neutron emission	$(n,3n)$: $\;{}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{\;\;\;Z}X)^* \rightarrow {}^{A-2}_{\;\;\;Z}X + 3{}_0^1n$
Proton emission	(n,p): $\;{}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{\;\;\;Z}X)^* \rightarrow {}_{Z-1}^{\;\;\;A}Y + {}_1^1H$
Alpha emission	(n,α): $\;{}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{\;\;\;Z}X)^* \rightarrow {}_{Z-2}^{A-4}Y + {}_2^4He$
Deuteron emission	(n,d): $\;{}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{\;\;\;Z}X)^* \rightarrow {}_{Z-1}^{A-1}Y + {}_1^2H$
Fission event	(n,f): $\;{}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{\;\;\;Z}X)^* \rightarrow {}_{Z_1}^{A_1}Y_1 + {}_{Z_2}^{A_2}Y_2 + (\ldots){}_0^1n$

Chang. "Neutron Interaction and Transport." http://kthn.dlu.edu.vn/Resources/Docs/SubDomain/kthn/2.NeutronInteraction.pdf

MCNP

Monte Carlo N–Particle® Transport Code System Version 6.2

- U.S. Department of Energy
- Los Alamos National Laboratory
- Oak Ridge National Laboratory
- Radiation Safety Information Computational Center (RSICC)

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MCNP6.2-EXE

- Use MCNP to accurately model foils in a fast fission spectrum
- Learn MCNP from ground up
- Utilize various features to examine how foils become activated over time
- Start simple and move to complex geometry
- Get useful information
- Process output

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MCNP6.2-EXE

Example of a foil mount used for activation analysis

15th ISRD

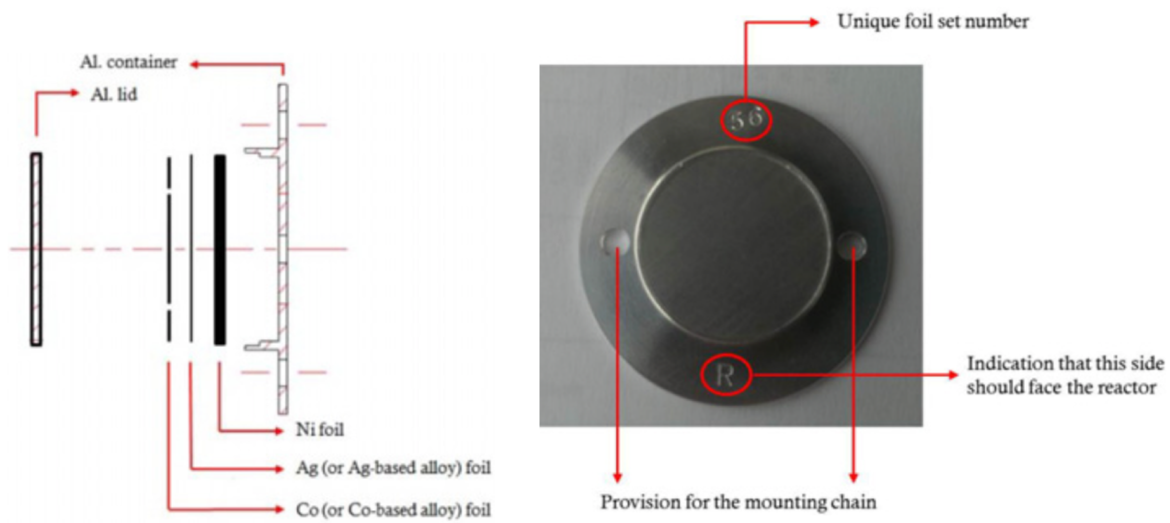


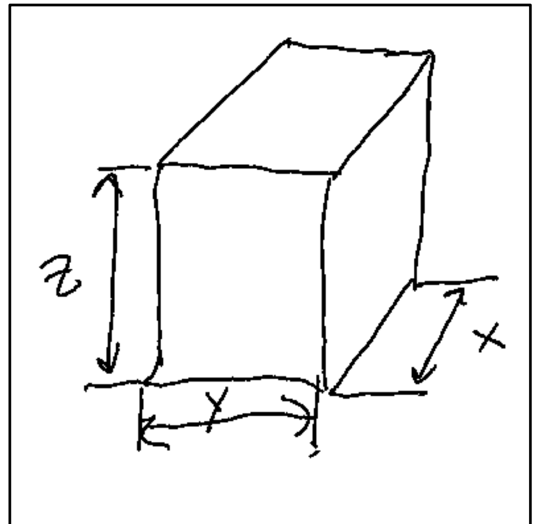
Figure 3. A general layout (left) and a picture (right) of an activation foil set used.

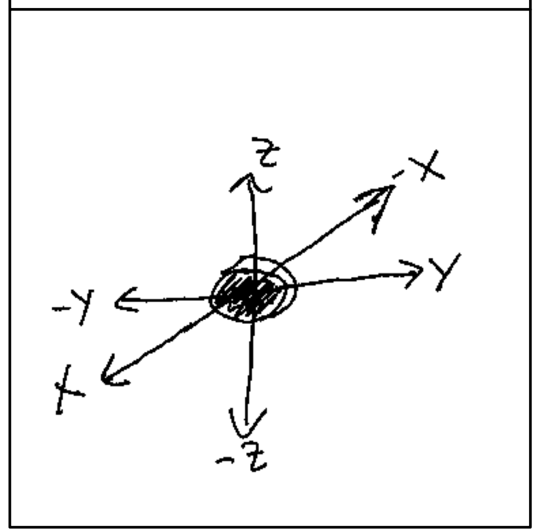
Validation of the MCNP computational model for neutron flux distribution with the neutron activation analysis measurement
https://iopscience.iop.org/article/10.1088/1742-6596/611/1/012007/pdf

MCNP6.2-EXE

Simplest geometry:

- The "Universe" as a box
- Single foil modeled as a right circular cylinder centered on origin
- Beam of neutrons from y-z plane from behind foil geometry.





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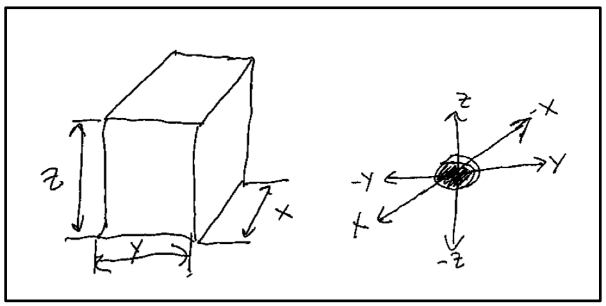
MCNP6.2-EXE

Simplest geometry

MCNP Input deck:




```
FoilTestInputDeck
c cell cards for sample problem
1 1 -7.874 -7
2 2 -8.908 -8
3 3 -1.60 1 -2 -3 4 -5 6 7 8
4 0 -1:2:3:-4:5:-6

c cube surfaces
1 PZ -5
2 PZ 5
3 PY 5
4 PY -5
5 PX 5
6 PX -5
c sphere surfaces
7 S 0 -3 -3 0.5  $ IRON SPHERE
c RCC surfaces
8 rcc  -0.0625 0 0  0.125 0 0  2.0  $ Nickel cylindrical foil

IMP:P,N       1 1 1 0
c centered on y-z plane cube at x=-5
SDEF POS=-5 0 0 x=-5 y=d1 z=d2 VEC= 1 0 0 PAR=1 DIR=1 ERG=10
SI1 -5 5
SP1 0 1
SI2 -5 5
SP2 0 1
F2:N 8.1 8.2 8.3 7 $ flux across Foil & sphere surfaces
F4:N 2 1 $ track length in foil (cell 2 & 1)
c comment out FM4 -7.68 2 2 102 $ activation information
E0 1 7I  10
M1 26000 1   $ Natural Iron
M2 28000 1   $ NATURAL Nickel
M3 6000 1    $ CARBON
PRINT 110 $PRINT first 50 neutrons from source
NPS 1000000
```

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MCNP6.2-EXE

More complex geometry:

- Multiple foils
- Model foil as a right circular cylinder

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MCNP6.2-EXE

Material Selection:

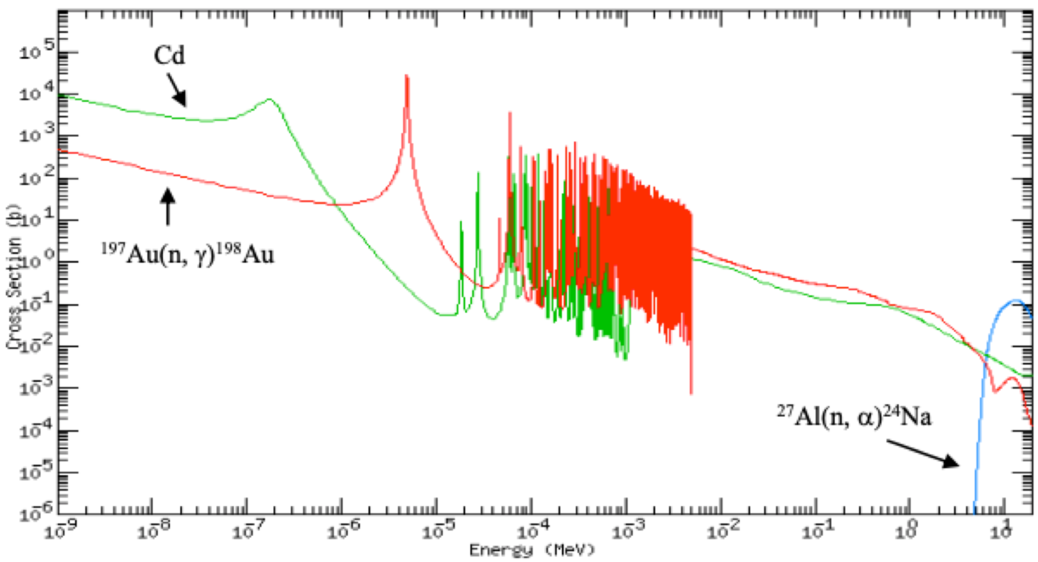


Figure 2. Cross sections for $^{197}Au(n, \gamma)^{198}Au$ and $^{27}Al(n, \alpha)^{24}Na$ reaction and Cd from ENDF/B-VII nuclear data library [2].

Validation of the MCNP computational model for neutron flux distribution with the neutron activation analysis measurement
https://iopscience.iop.org/article/10.1088/1742-6596/611/1/012007/pdf

MCNP6.2-EXE

Neutron Activation Foil And Thermoluminescent Dosimeter Responses To A Polyethylene Reflected Pulse Of The CEA Valduc SILENE Critical Assembly:

Table 3-2. Thickness of each benchmark model neutron activation foil

Location	Reference name	Thickness (mm)
Case 1 collimator A	Au007	0.25
	Ni019	2
	In13-A10	1
	Fe026	3
	Mg09-A10	2
	Co029	2
	Ti	2
Case 2 collimator B	Au003	0.25
	Ni010	2
	In14-A10	1
	Fe017	3
	Mg08-A10	2
	Co014	2
	Ti	2
Case 3 free field	Au002	0.25
	Ni008	1
	In12-A10	1
	Fe023	3
	Mg10-A10	2
	Co030	2
	Ti	2
Case 4 scattering box 1	Au005	0.25
	Ni030	2
	In001	1
	Fe022	3
	Mg07-A10	2
	Co035	2
	Ti	2
Case 5 scattering box 2	Au10-B10	0.25
	Ni032	2
	Co025	2
Case 6 scattering box 3	Au012	0.25
	Ni031	2
	Co032	2
Case 7 scattering box 4	Au004	0.25
	Ni026	2
	Co018	2

ALARM-TRAN-CH2-SHIELD-001

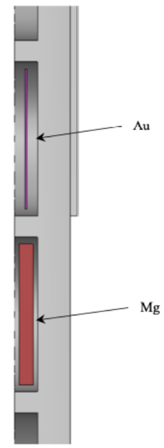
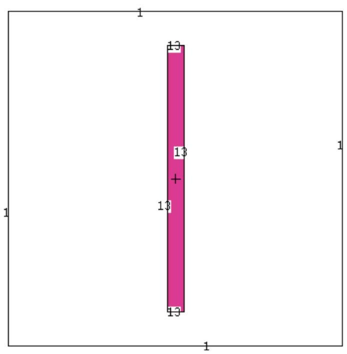


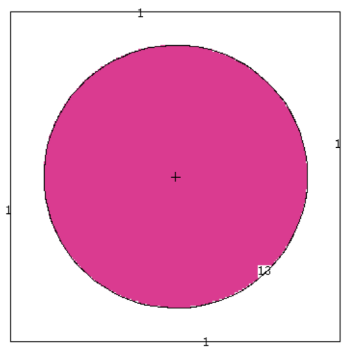
Fig. 3-33. XZ view of the neutron activation foils within the foil holders.

ALARM-TRAN-CH2-SHIELD-001
https://info.ornl.gov/sites/publications/files/Pub68537.pdf

MCNP6.2-EXE

MCNP Input deck:





Simplest geometry:

- Single/Multiple foils
- Model foil as a right circular cylinder
- Monoenergetic neutron source

WRL: example MCNP simulation input deck

```
message: O=OutputFiles/FoilTestOut1 RU=RUFiles/FoilTestRU1 C=ComOutFiles/FoilTestComOut1 PTRAC=ptracFiles/
ptracTestFoil1

TestFoilInputDeck
c cell cards for problem
c comment out:  1 1 -8.900 -11
c comment out:  2 2 -10.50 -12
3 3 -8.902 -13
4 0 -1 13 $comment out:  11 12 13  $ universe volume
5 0 1

c cube surfaces $ the "universe"
1  box  -5 -5 -5  10 0 0  0 10 0  0 0 10
c rcc cylinders surfaces as foils
c comment out:  11 rcc  -0.4000 0 0  0.200 0 0  2.0  $ cobalt cylindrical foil
c comment out:  12 rcc  -0.0500 0 0  0.100 0 0  2.0  $ silver cylindrical foil
C center surface 13 on origin
13 rcc   -0.2500 0 0  0.5 0 0  4.0  $ nickel cylindrical foil

IMP:P,N  1 1 0 $comment out:   1 1 1 1 0
c source centered on y-z plane cube at x=-5
SDEF POS=-5 0 0 x=-5 y=d1 z=d2 VEC= 1 0 0 PAR=1 DIR=1 ERG=10
SI1 -5 5
SP1 0 1
SI2 -5 5
SP2 0 1
c tally cards
c comment out surface flux:  F2:N 11.1 11.2 11.3 12.1 12.2 12.3 13.1 13.2 13.3 $ flux across Co, Ag, & Ni
surfaces
c comment out:  F14:N 1 $ track length in foil (cell 1)
c comment out:  F24:N 2 $ track length in foil (cell 2)
F34:N 3 $ track length in foil (cell 3)
c reaction rates for Co, Ag, Ni
c bin 1 (n,2n) + (n,3n) reaction rate
c bin 2 (n,0n) reaction rate
c comment out:  FM14  (-1 1 16:17)  &
c      (-1 1 -2)
c comment out:  FM24  (-2 2 16:17)  &
c       (-2 2 -2)
FM34  (-3 3 16:17)  $ comment out &(-3 3 -2)
c FM4 $ activation information
c Bin energy:
E0 1 8I  10
c Material Data
c comment out:  M1 27059.80c 1.000000 $ Cobalt (Co)
c comment out:  M2 47107.80c 0.518390 &
c comment out:     47109.80c 0.481610 $ Silver (Ag)
M3 28058.80c 0.680770 &
   28060.80c 0.262230 &
   28061.80c 0.011399 &
   28062.80c 0.036346 &
   28064.80c 0.009255 $ Nickel (Ni)
C comment out:  M4 06000.80c 1.000000 $ Carbon (C)
MPHYS ON
PRINT 110 $PRINT first 50 neutrons from source
c Particle Track Output
c PTRAC example from manual
c PTRAC EVENT=SUR NPS=1,50 TYPE=N CELL=3 FILE = ASC
c PTRAC for capture events
PTRAC TYPE=N CELL=3 FILE=ASC TALLY=34 WRITE=POS
C comment out:
NPS 1000000 $ stop after set number of particles
CTME 1.0 $stop after one minute
```

Analysis of MCNP output

ptrac text file:

```
   -1
mcnp     6                      02/20/18 03/31/21 10:25:27
TestFoilInputDeck
   1.4000E+01  1.0000E+00  1.0000E+02  1.0000E+00  3.0000E+00  0.0000E+00  1.0000E+00  1.0000E+00  0.0000E+00  1.0000E+00
   1.0000E+04  0.0000E+00  0.0000E+00  0.0000E+00  1.0000E+00 -1.0000E+00  1.0000E+00  1.0000E+00  0.0000E+00  1.0000E+00
   1.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
    5    5    3    6    3    6    3    6    3    6    3    1    4    0    0    0    0    0    0    0
   1    2    3    5    6    7    8    9   17   18   20   21   22    7    8   10   11   17   18   20   21   22    7    8   12   13   17   18   20   21
  22    7    8   10   11   17   18   20   21   22    7    8   14   15   17   18   20   21   22
        8388       1000          5         34  8.08830E-05
        3000          1         40          4          0
  -0.50000E+01 -0.56379E+00  0.83193E+00
        4000          2       13.3        179          3          1
  -0.25000E+00 -0.56379E+00  0.83193E+00
        3000          2      28060          2          3          1
  -0.70112E-01 -0.56379E+00  0.83193E+00
        3000          3       13.1         11          4          0
   0.22250E-01 -0.39029E+01  0.87590E+00
        5000          4        1.4          1          5          0
   0.52596E-01 -0.50000E+01  0.89034E+00
        9000          4          1          1          5          0
   0.52596E-01 -0.50000E+01  0.89034E+00
       25160       1000          5         34  1.17772E-04
        3000          1         40          4          0
  -0.50000E+01  0.15413E+01  0.24723E+01
        4000          2       13.3        179          3          1
  -0.25000E+00  0.15413E+01  0.24723E+01
        3000          2      28058          2          3          1
   0.90869E-01  0.15413E+01  0.24723E+01
        3000          3       13.2         88          4          0
   0.25000E+00 -0.38155E+01  0.13802E+00
        5000          4        1.4         23          5          0
   0.28519E+00 -0.50000E+01 -0.37814E+00
        9000          4          1          1          5          0
   0.28519E+00 -0.50000E+01 -0.37814E+00
       32046       1000          5         34  1.07736E-04
        3000          1         40          4          0
  -0.50000E+01  0.80240E+00 -0.36844E+01
        4000          2       13.3        179          3          1
  -0.25000E+00  0.80240E+00 -0.36844E+01
        4000          2      28058          2          3          1
   0.19748E+00  0.80240E+00 -0.36844E+01
        3000          2      28058          2          3          1
  -0.12012E+00  0.19908E+01 -0.19128E+01
        3000          3       13.3         88          4          0
  -0.25000E+00  0.17398E+01  0.25950E+01
```

WRL: example MCNP simulation input deck

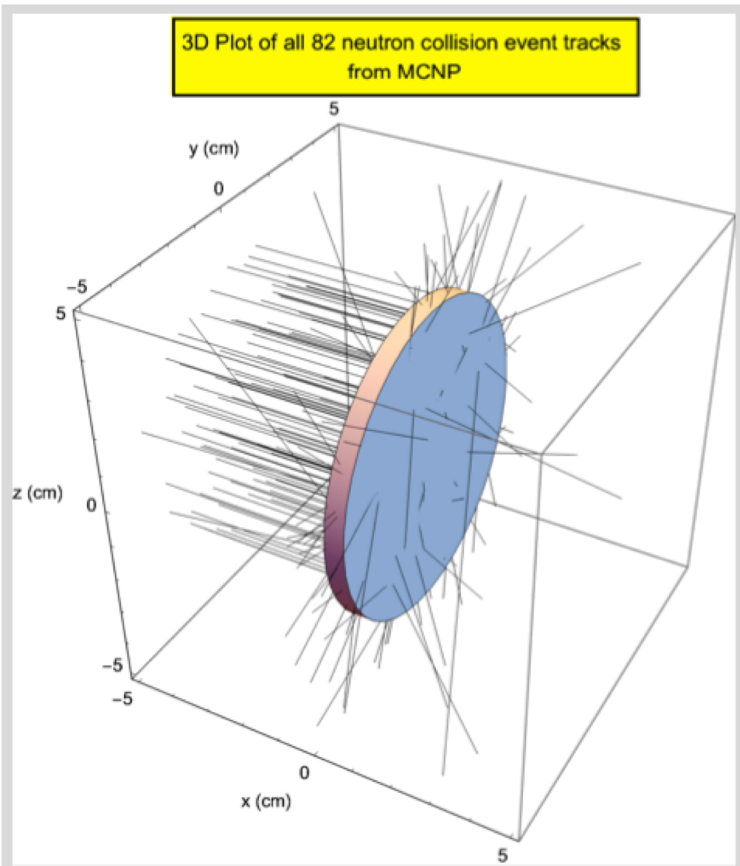
Analysis of MCNP output

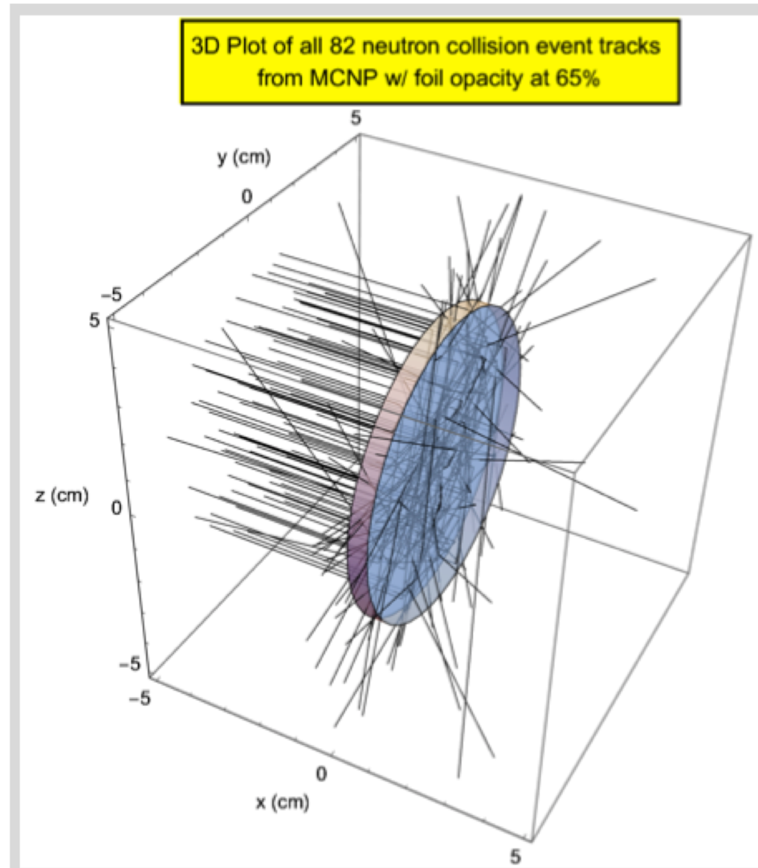
Parse ptrac text file for useful plotting information:

```
Total number of particle histories:  82
History: 1
{{-5., -0.56379, 0.83193}, {-0.25, -0.56379, 0.83193},
  {-0.070112, -0.56379, 0.83193}, {0.02225, -3.9029, 0.8759}, {0.052596, -5., 0.89034}}
History: 2
{{-5., 1.5413, 2.4723}, {-0.25, 1.5413, 2.4723},
  {0.090869, 1.5413, 2.4723}, {0.25, -3.8155, 0.13802}, {0.28519, -5., -0.37814}}
History: 3
{{-5., 0.8024, -3.6844}, {-0.25, 0.8024, -3.6844}, {0.19748, 0.8024, -3.6844},
  {-0.12012, 1.9908, -1.9128}, {-0.25, 1.7398, 2.595}, {-0.3193, 1.6059, 5.}}
History: 4
{{-5., -1.7488, -0.30272}, {-0.25, -1.7488, -0.30272}, {-0.14455, -1.7488, -0.30272},
  {0.055065, -0.40073, 0.057923}, {-0.077465, 2.3359, 2.517}, {0.24361, 2.8326, 2.8242}, {1.6446, 5., 4.1647}}
History: 5
{{-5., -0.37332, -1.0908}, {-0.25, -0.37332, -1.0908}, {0.18938, -0.37332, -1.0908},
  {0.21671, -0.33736, -0.99041}, {-0.11659, 0.44199, 3.9755}, {-0.18535, 0.60278, 5.}}
History: 6
{{-5., 1.0861, -3.2176}, {-0.25, 1.0861, -3.2176}, {0.021462, 1.0861, -3.2176},
  {0.19112, -1.6386, -2.4409}, {0.24982, -3.4713, -1.9875}, {0.29879, -5., -1.6093}}
History: 7
{{-5., -2.3022, 2.334}, {-0.25, -2.3022, 2.334}, {0.077326, -2.3022, 2.334},
  {0.020044, 1.4238, -0.56234}, {0.062515, 1.7276, -0.96147}, {0.25, 2.0545, -1.1158}, {1.9392, 5., -2.5063}}
```

WRL: example MCNP simulation input deck

Analysis of MCNP output





WRL: example MCNP simulation input deck

Analysis of MCNP output

```
source              1000000   1.0000E+00    1.0000E+01          escape              1000071   9.8794E-01    9.7289E+00
nucl. interaction         0   0.            0.                  energy cutoff             0   0.            0.
particle decay            0   0.            0.                  time cutoff               0   0.            0.
weight window             0   0.            0.                  weight window             0   0.            0.
cell importance           0   0.            0.                  cell importance           0   0.            0.
weight cutoff             0   0.            0.                  weight cutoff             0   0.            0.
e or t importance         0   0.            0.                  e or t importance         0   0.            0.
dxtran                    0   0.            0.                  dxtran                    0   0.            0.
forced collisions         0   0.            0.                  forced collisions         0   0.            0.
exp. transform            0   0.            0.                  exp. transform            0   0.            0.
upscattering              0   0.            0.                  downscattering            0   0.            1.5031E-01
photonuclear              0   0.            0.                  capture                   0   1.2124E-02    1.2028E-01
(n,xn)                  142   1.3508E-04    1.1535E-04          loss to (n,xn)           71   6.7540E-05    6.7498E-04
prompt fission            0   0.            0.                  loss to fission           0   0.            0.
delayed fission           0   0.            0.                  nucl. interaction         0   0.            0.
prompt photofis           0   0.            0.                  particle decay            0   0.            0.
tabular boundary          0   0.            0.                  tabular boundary          0   0.            0.
tabular sampling          0   0.            0.                  elastic scatter           0   0.            0.
    total           1000142   1.0001E+00    1.0000E+01              total           1000142   1.0001E+00    1.0000E+01

  number of neutrons banked                     71        average time of (shakes)              cutoffs
  neutron tracks per source particle     1.0001E+00         escape            2.3531E-01          tco   1.0000E+33
  neutron collisions per source particle 7.9782E-02         capture           1.1588E-01          eco   0.0000E+00
  total neutron collisions                   79782          capture or escape 2.3386E-01          wc1  -5.0000E-01
  net multiplication                1.0001E+00 0.0000       any termination   2.3385E-01          wc2  -2.5000E-01

computer time so far in this run     0.10 minutes         maximum number ever in bank         1
computer time in mcrun               0.09 minutes         bank overflows to backup file       0
source particles per minute            1.1679E+07
random numbers generated                  3133955         most random numbers used was          75 in history       129565
```

WRL: example MCNP simulation input deck

Analysis of MCNP output

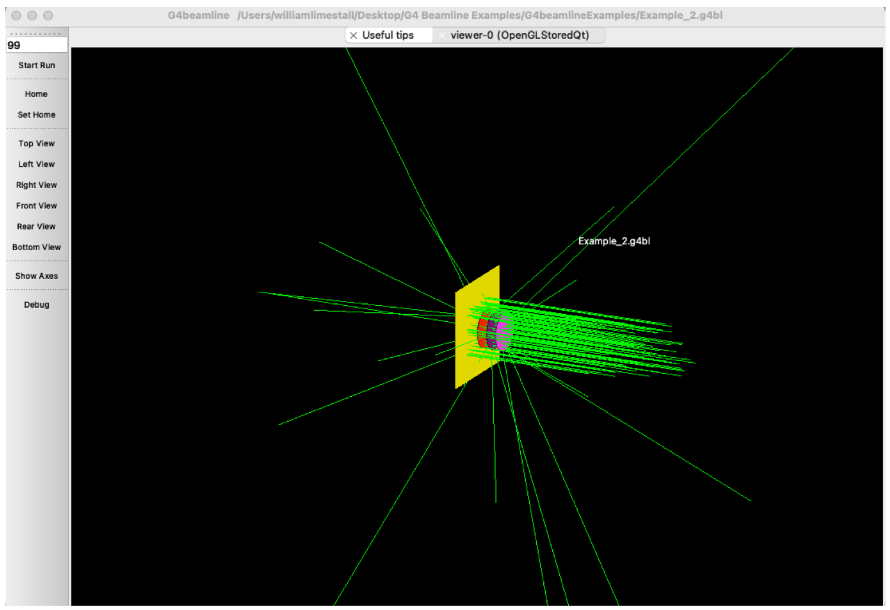
```
                      tracks     population   collisions   collisions     number        flux       average      average
              cell    entering                             * weight     weighted     weighted   track weight   track mfp
                                                          (per history)   energy        energy    (relative)      (cm)

       1        3      501941       502012        79782    7.8024E-02   8.8904E+00   9.5755E+00   9.7856E-01   3.3447E+00
       2        4     1502012      1000071            0    0.0000E+00   9.7860E+00   9.9229E+00   9.9380E-01   0.0000E+00

           total      2003953      1502083        79782    7.8024E-02

1tally       34        nps =     1000000
           tally type 4    track length estimate of particle flux.
           particle(s): neutrons

           volumes
                   cell:       3
                         2.51327E+01

 cell  3
 multiplier bin:  -3.00000E+00         3          16    :           17
      energy
    1.0000E+00   0.00000E+00 0.0000
    2.0000E+00   0.00000E+00 0.0000
    3.0000E+00   0.00000E+00 0.0000
    4.0000E+00   0.00000E+00 0.0000
    5.0000E+00   0.00000E+00 0.0000
    6.0000E+00   0.00000E+00 0.0000
    7.0000E+00   0.00000E+00 0.0000
    8.0000E+00   2.46864E-13 0.1758
    9.0000E+00   1.69714E-09 0.0388
    1.0000E+01   7.96880E-06 0.0010
      total      7.97050E-06 0.0010
```

WRL: example MCNP simulation input deck

G4beamline

Material Selection:

- Multiple foils
- Model foil as a right circular cylinder

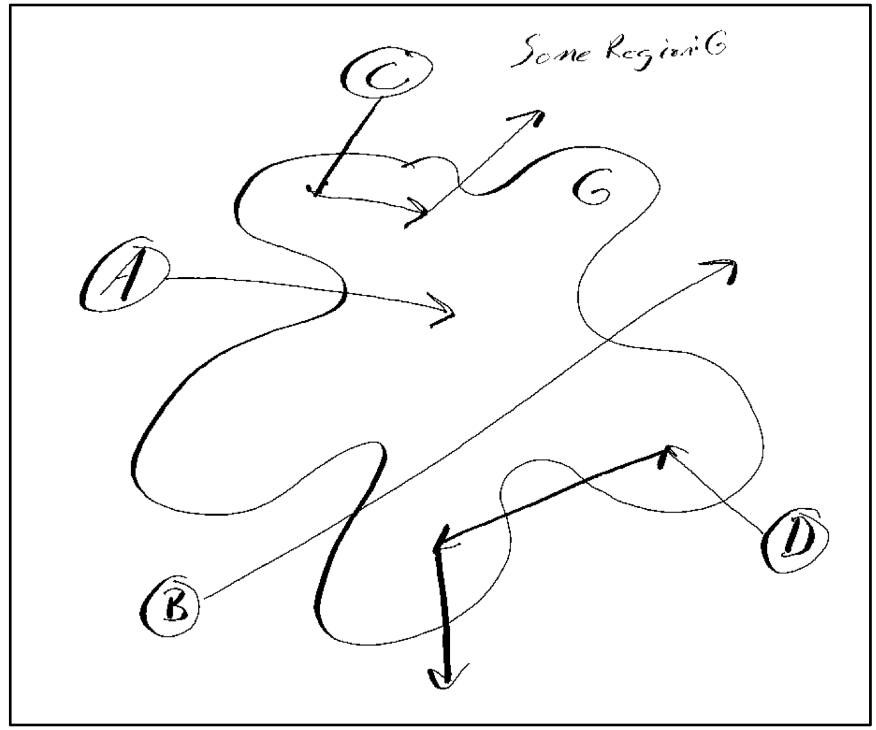


Addendum

Basics of Monte Carlo

Basics of Monte Carlo

- Follow all individual particles as they fly around and have collisions.
- Collect & tally information from all particles.
- Basis for a pathlength estimator.



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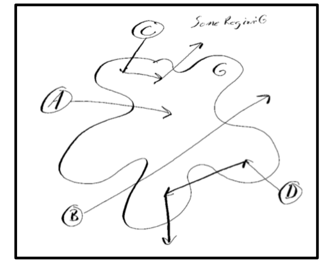
Basics of Monte Carlo



Macroscopic Cross Section (Σ)

- Probability of interaction with a material per unit distance traveled.

$$\Sigma_x^{mat} = \frac{1}{\lambda} = \sum_i N^i \cdot \sigma_x^i$$

N^i is the nuclide density $\left(\frac{atoms}{barns \cdot cm}\right)$

σ_x^i is an individual particle microscopic cross section $(barns)$

x is the type of interaction $\{(n, n'), (n, \gamma), (n, 2n), etc \ldots\}$

λ is the average distance to collision

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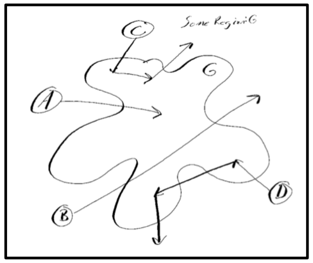
Basics of Monte Carlo



Macroscopic Cross Section (Σ)

- Example:

$$\Sigma_{(n,2n)}^{silver} = N^{Ag-109}\sigma_{(n,2n)}^{Ag-109} + N^{Ag-109}\sigma_{(n,2n)}^{Ag-107}$$

$$\Sigma_{(n,2n)}^{silver} = \Sigma_{(n,2n)}^{Ag-109} + \Sigma_{(n,2n)}^{Ag-107}$$

This must be done with all isotopes of a material contained in a particular region. Nickel for instance has 5 naturally occurring and stable isotopes.

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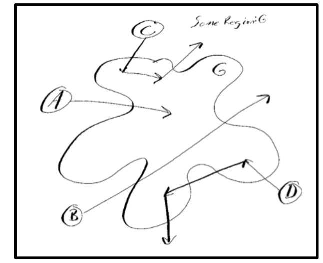
Basics of Monte Carlo



Macroscopic Cross Section (Σ)

- Total:

$$\Sigma_{total}^{mat} = \Sigma_{scattered}^{mat} + \Sigma_{absorbed}^{mat}$$

- Probability:

$$P_A = \frac{\Sigma_{absorbed}^{mat}}{\Sigma_{total}^{mat}} \qquad P_S = \frac{\Sigma_{scattered}^{mat}}{\Sigma_{total}^{mat}}$$

$$P_A + P_S = 1$$

Note: In MCNP lingo, absorption cross section is defined as capture plus fission, where capture is everything else NOT fission.

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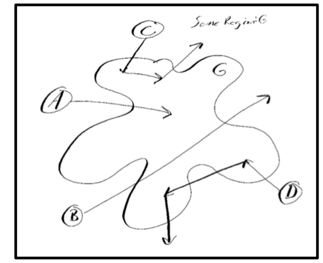
Basics of Monte Carlo



Reaction Rates from Macroscopic Cross Section (Σ)

- Collision Rate:

$$R_C = \Sigma_{total}^{mat}\,\phi$$

(Here ϕ is flux)

- Absorption Rate for isotope x:

$$R_A = \Sigma_{absorbed}^{x}\,\phi$$

Reaction Rate: $\left[\frac{reactions}{cm}\right] \cdot \left[\frac{total\ cm\ traveled}{cm^3 \cdot sec}\right] = \left[\frac{reactions}{cm^3 \cdot sec}\right]$

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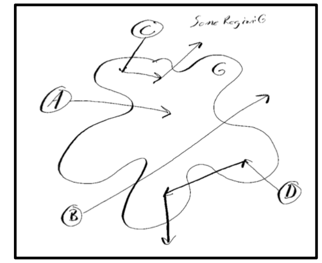
Basics of Monte Carlo



Flux (ϕ) **and Current** (J)

- **Flux**: the total distance traveled by all particles in a cm^3 per second

$$\phi = \left[\frac{distance}{cm^3 \cdot sec}\right] = \left[\frac{1}{cm^2 \cdot sec}\right]$$

- **Current**: number of particles crossing surface per second per unit area

$$J = \left[\frac{1}{cm^2 \cdot sec}\right]$$

Note also have partial current $\Longrightarrow J = J^+ + J^-$
With (+) or (–) directions

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Basics of Monte Carlo

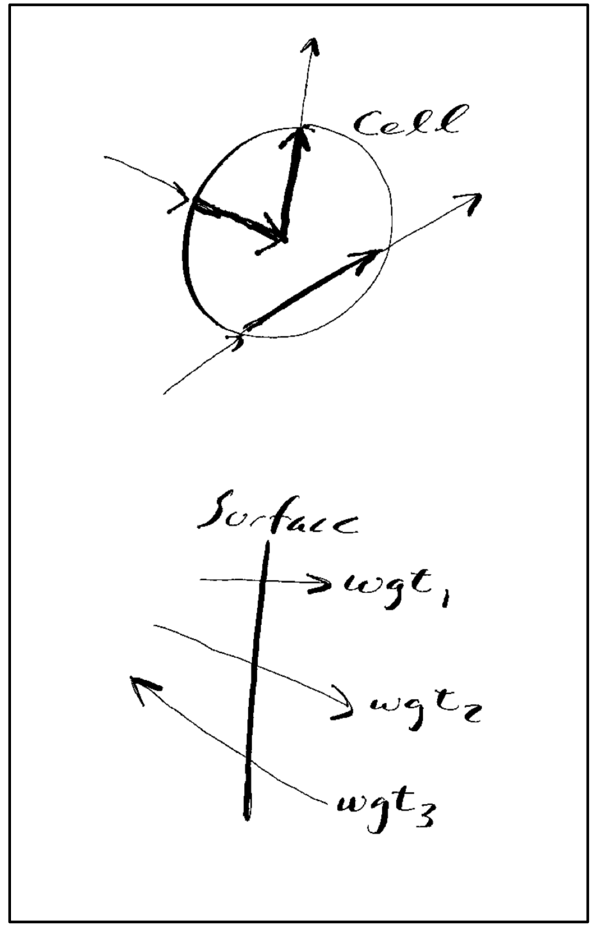
- **Flux in a cell**:

$$\phi = \frac{1}{V \cdot W} \sum_{\substack{all\ flights \\ in\ a\ cell}} wgt \cdot dist$$

- **Current across a surface**:

$$J = \frac{1}{A \cdot W} \sum_{\substack{all\ flights \\ crossing \\ surface}} wgt$$

W is total source weight
V is volume
wgt is individual particle weight
$dist$ is distance traveled
A is area



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