L-L-T group notes

Monday, April 19, 2021

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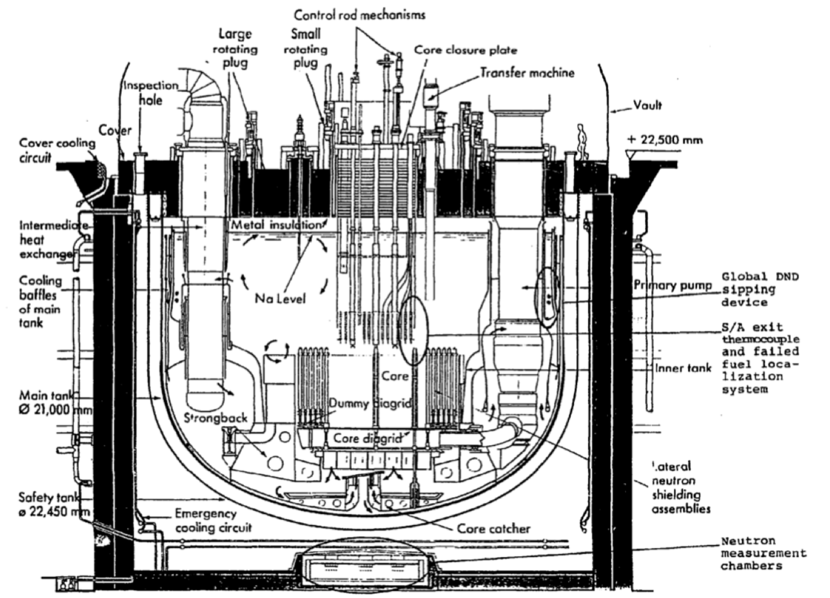
Superphénix Reactor

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



GENERAL VIEW OF SPX1 PRIMARY CIRCUIT
FIGURE 1 LOCATION OF CORE INSTRUMENTATION
23

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

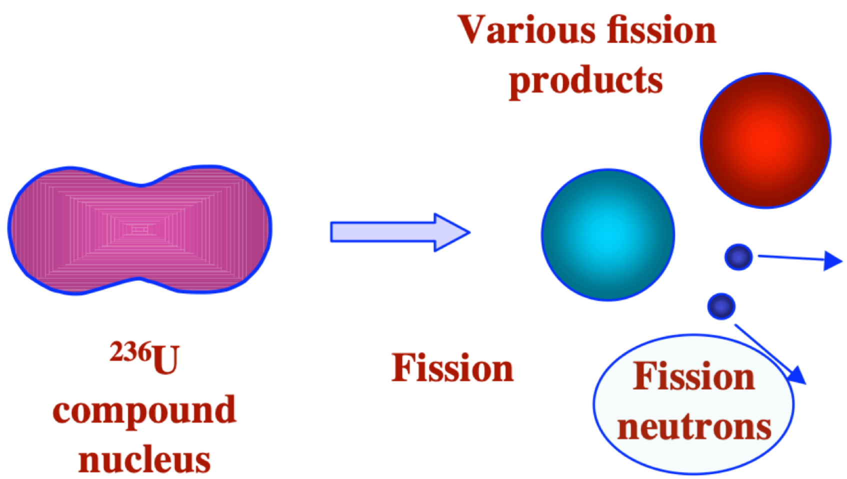
12 Jan. 2021

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 $^{16}_{8}O$

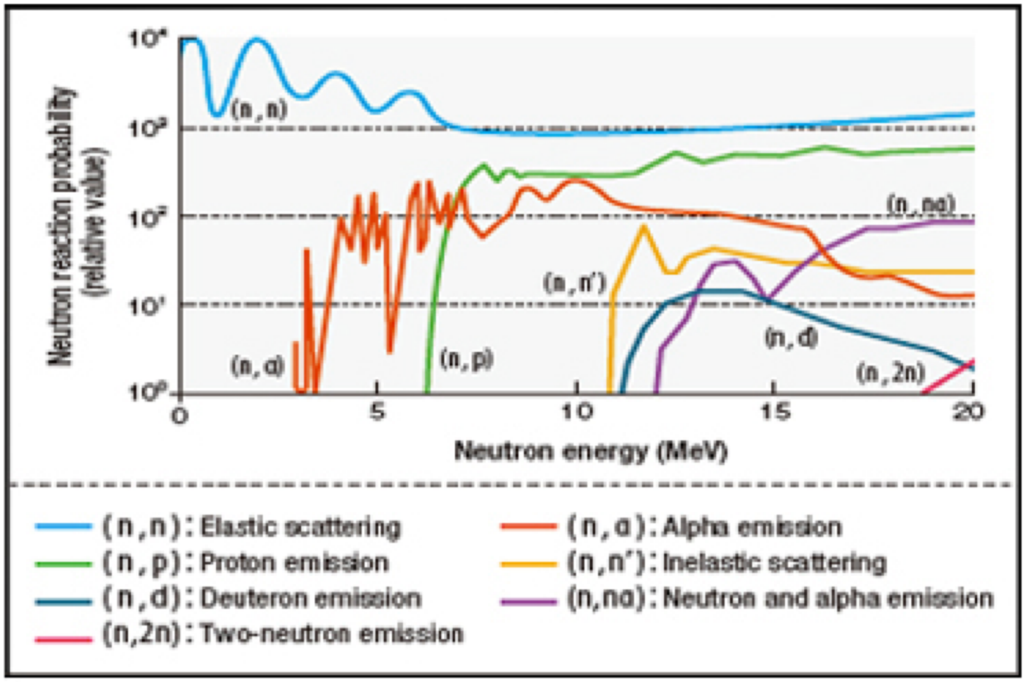
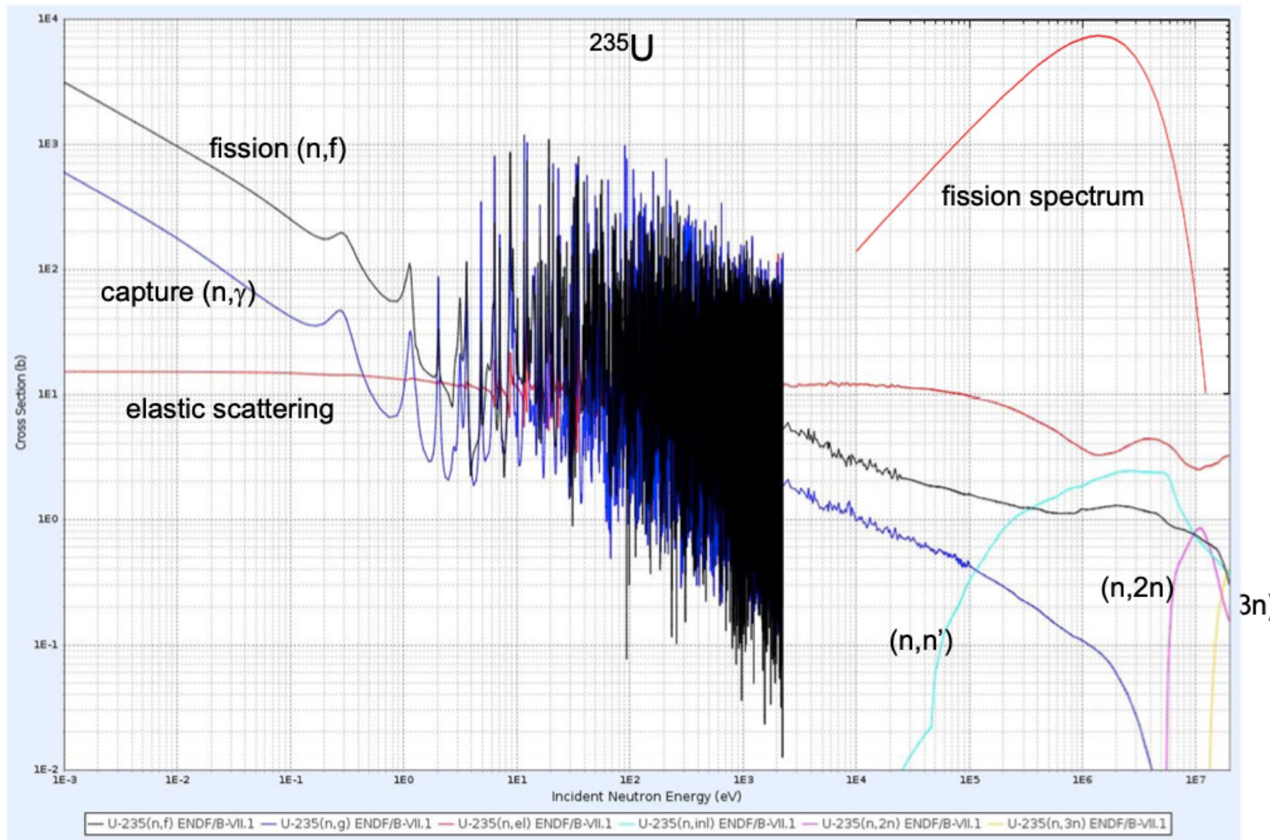


Figure 1. (for example)
Various Nuclear Reactions between ^{16}O and Neutrons

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): ${}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{Z}X)^* \rightarrow {}^{A-4}_{Z-2}Y + {}_2^4He$
Deuteron emission	(n,d): ${}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{Z}X)^* \rightarrow {}^{A-1}_{Z-1}Y + {}_1^2H$
Fission event	(n,f): ${}_0^1n + {}_Z^AX \rightarrow ({}^{A+1}_{Z}X)^* \rightarrow {}^{A_1}_{Z_1}Y_1 + {}^{A_2}_{Z_2}Y_2 + (\ldots){}_0^1n$

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

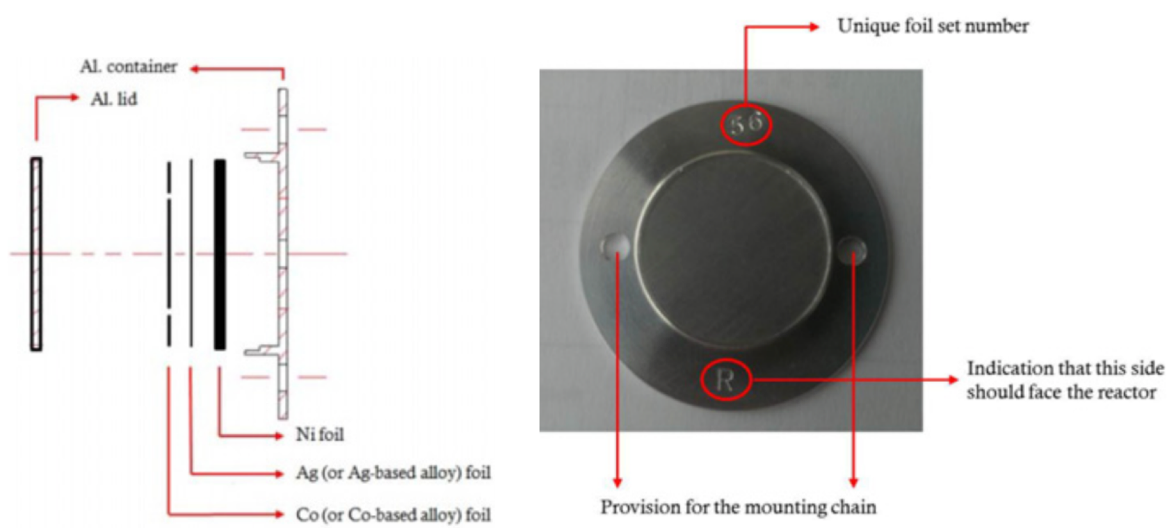


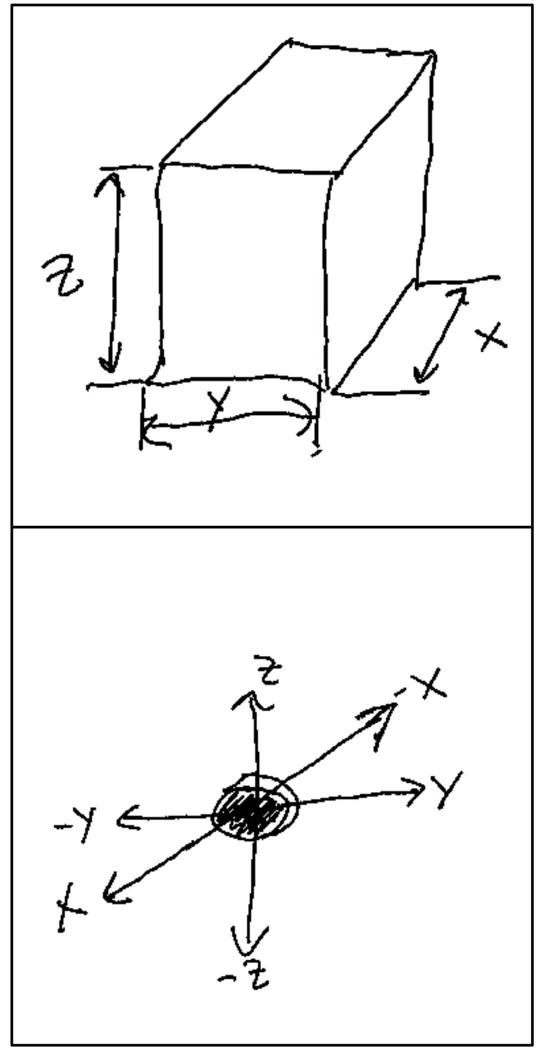
Figure 3. A general layout (left) and a picture (right) of an activation foil set used.

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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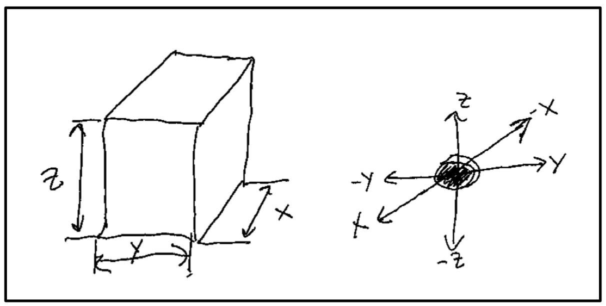
MCNP6.2-EXE

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:

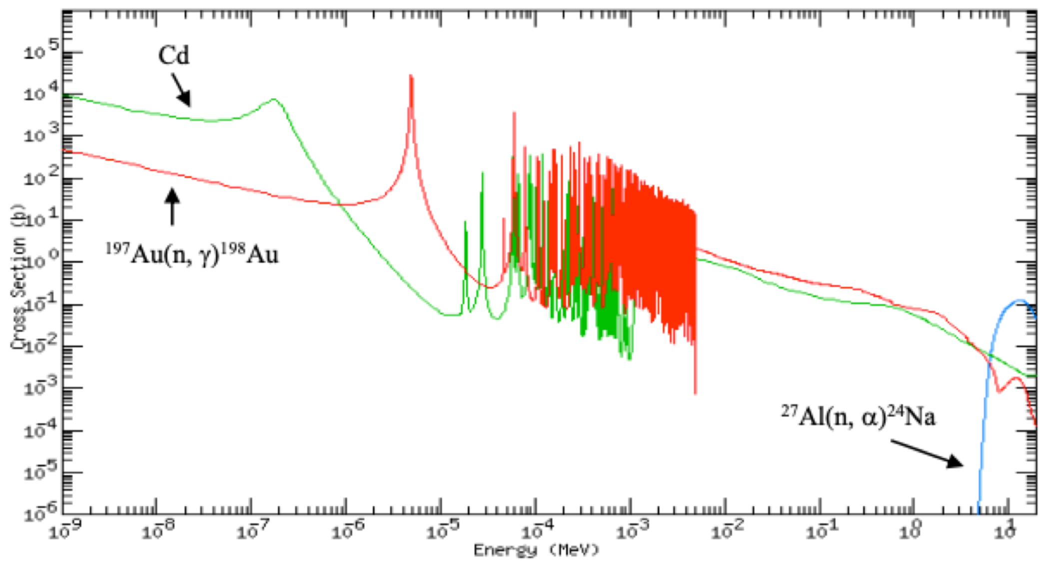


Figure 2. Cross sections for $^{197}Au(n, \gamma)^{198}Au$ and $^{27}Al(n, \alpha)^{24}Na$ reaction and Cd from ENDF/B-VII nuclear data library [2].

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

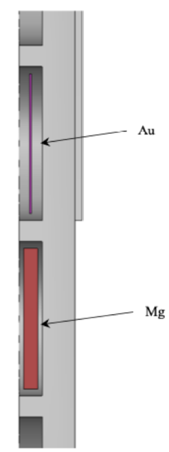


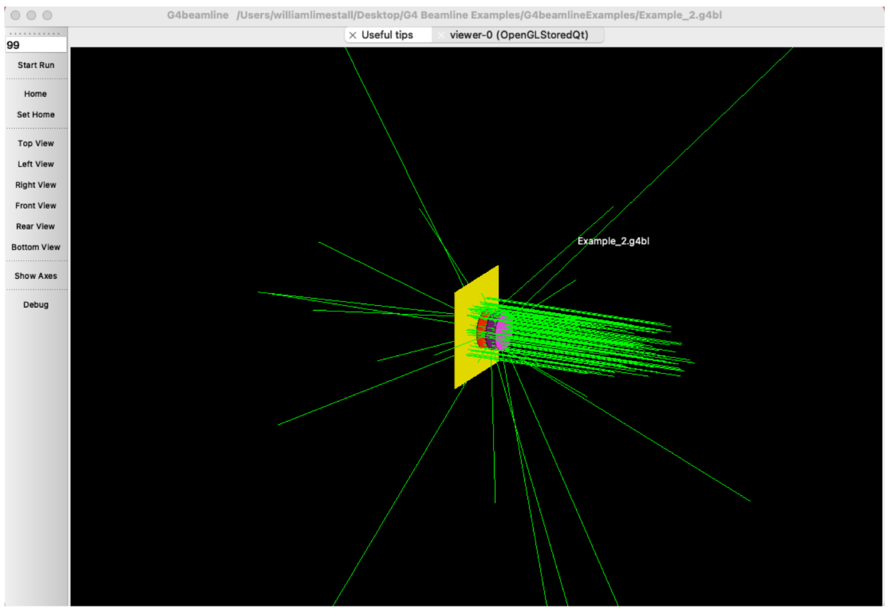
Fig. 3-33. XZ view of the neutron activation foils within the foil holders.

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

G4beamline

Material Selection:

- Multiple foils
- Model foil as a right circular cylinder

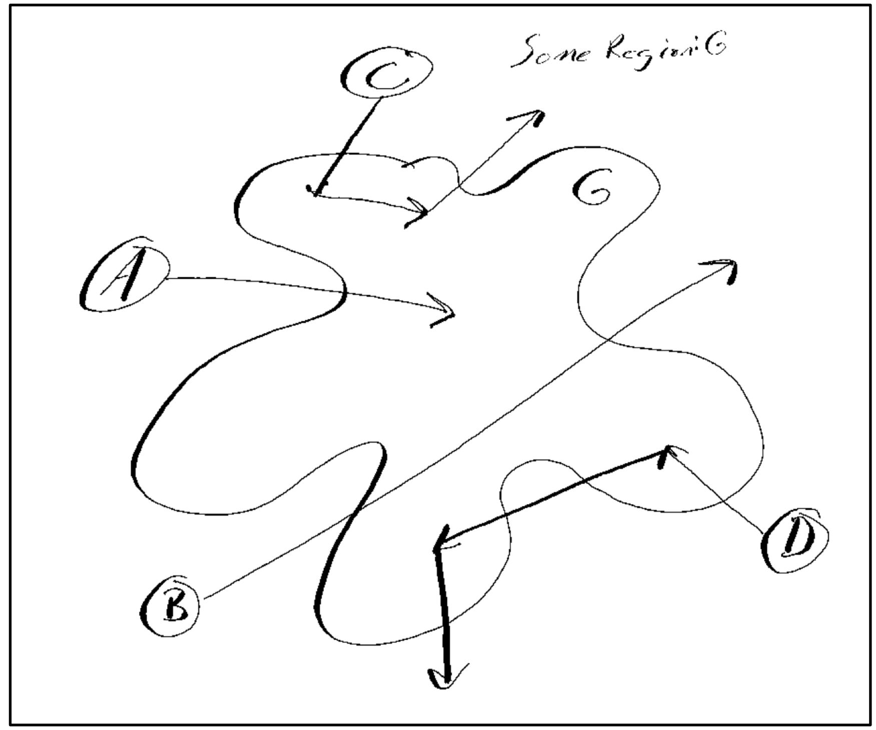


Addendum

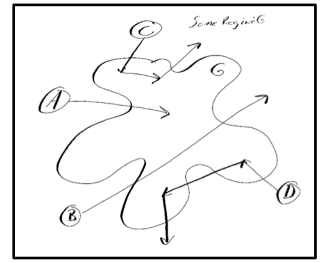
Basics of Monte Carlo

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.



Basics of Monte Carlo



Macroscopic Cross Section (Σ)

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

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

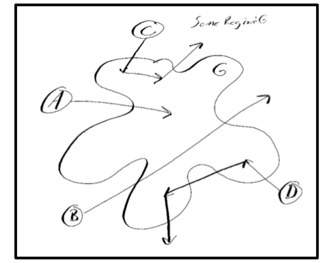
N^i is the nuclide density $\left(\frac{atoms}{barns \cdot cm}\right)$

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

Forrest Brown. “Monte Carlo Techniques for Nuclear Systems – Theory Lectures.” LA-UR-16-29043



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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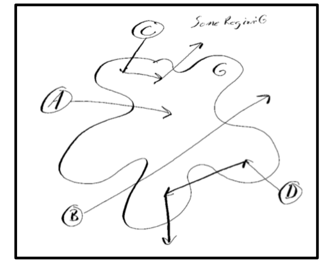
Basics of Monte Carlo



Macroscopic Cross Section (Σ)

- Total:

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

- Probability:

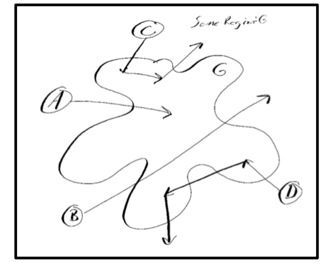
$$P_A = \frac{\Sigma_{absorbed}^{mat}}{\Sigma_{total}^{mat}} \qquad P_S = \frac{\Sigma_{scattered}^{mat}}{\Sigma_{total}^{mat}}$$

$$P_A + P_S = 1$$

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

Forrest Brown. "Monte Carlo Techniques for Nuclear Systems – Theory Lectures." LA-UR-16-29043

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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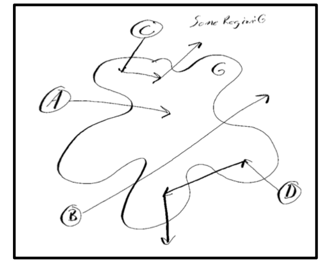
Basics of Monte Carlo



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

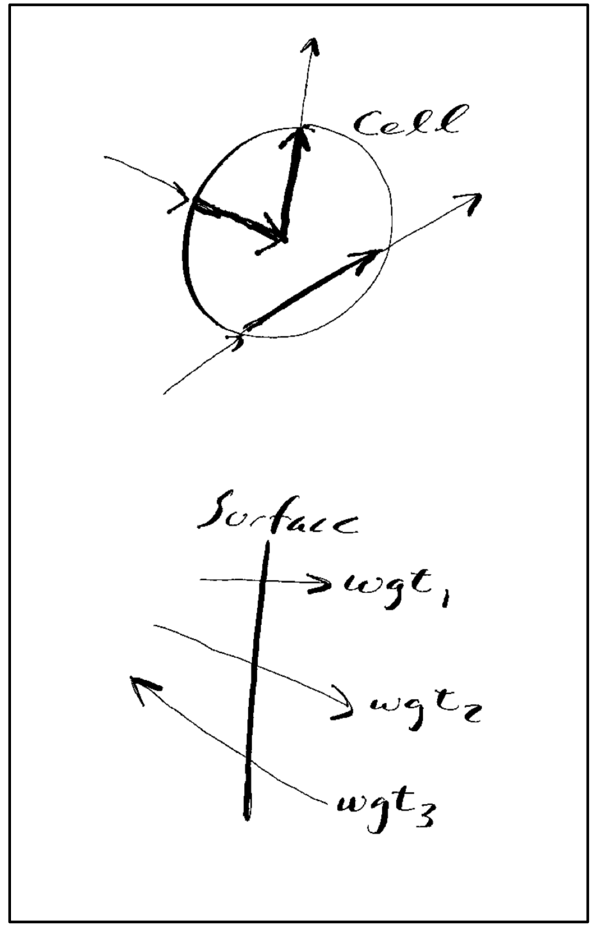
- **Flux in a cell**:

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

- **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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