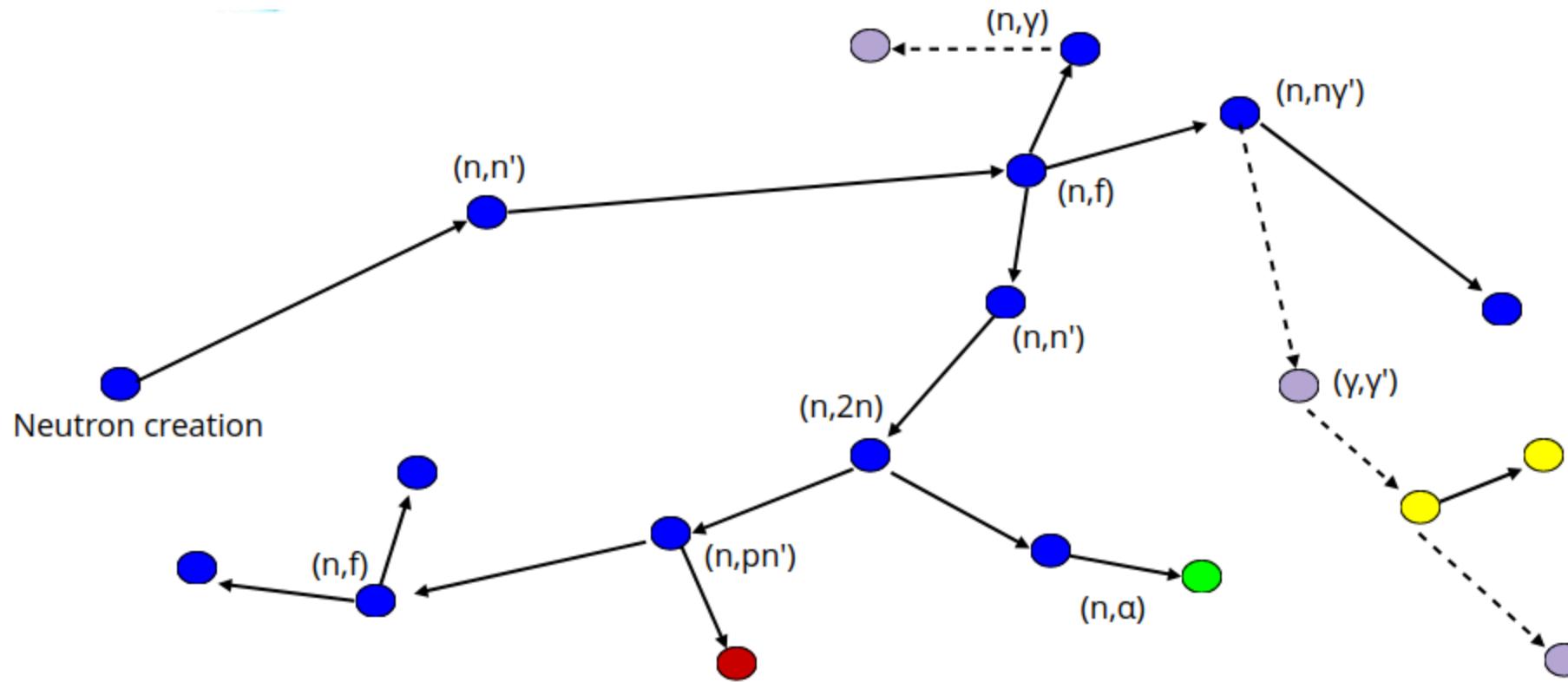


Neutronics Analysis of Fusion Systems



Slides available on GitHub

<https://github.com/fusion-energy/fusion-neutronics-presentation-slides>

Why is neutronics useful

- **Radioactivity** - Neutrons activate material, making it radioactive leading to handling and waste storage requirements.
- **Hazardous** - Neutrons are Hazardous to health and shielded will be needed to protect the workforce.
- **Produce fuel** - Neutrons will be needed to convert lithium into tritium to fuel the reactor.
- **Electricity** - 80% of the energy release by each DT reaction is transferred to the neutron.
- **Structural integrity** - Neutrons cause damage to materials such as embrittlement, swelling, change conductivity ...
- **Diagnose** - Neutrons are an important method of measuring a variety of plasma parameters (e.g. Q value).

Topics Covered

- Nuclear data
- Prompt responses
- Delayed responses
- Simulation approaches

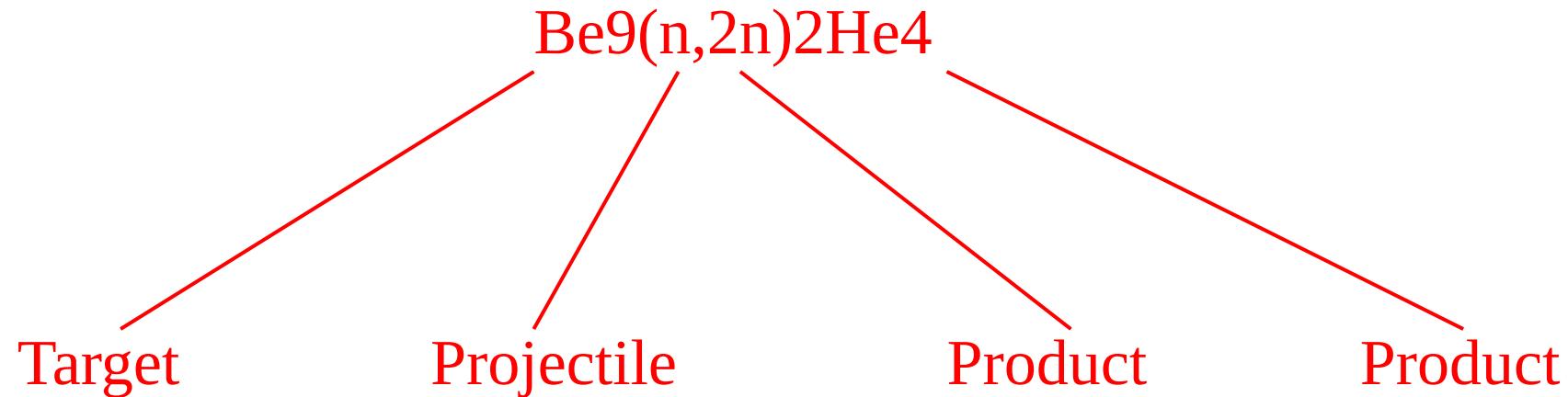
Nuclear data

- Reactions
- Isotope chart
- Transmutation reactions
- Q values
- Threshold reactions
- Fusion fuels (DT,DD ...)
- Energy distribution from DT
- Microscopic cross sections
- Experimental data
- Libraries (ENDF, TENDL, FENDL ...)
- Cross section regions
- Multigroup / continuous energy
- Group structures
- Reaction rate equation
- Macroscopic cross sections
- Scattering / thermalisation
- Decay data
- Photons
- Energy distribution from radioactive material

Reactions

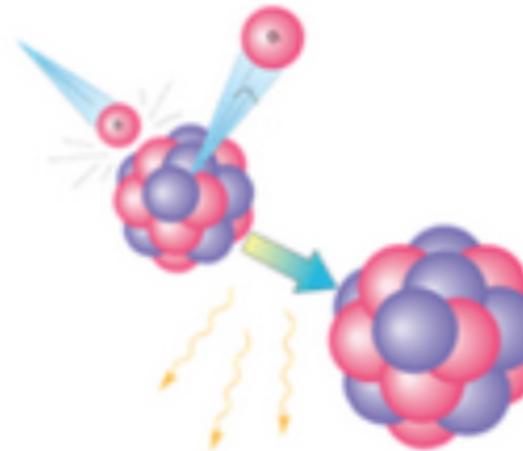
Nuclear reactions notation

Target nuclei (incident projectile, resulting fragments) resulting nuclei



Neutron induced reactions

- 999 reactions channels with unique reaction IDs (MT numbers)
- MT 3 is elastic scattering ($n, 'n$)
- MT 16 is neutron multiplication ($n, 2n$)
- MT 18 is neutron multiplication (n, f)
- MT 205 is tritium production (n, X_t) where X is a wild card
- MT 444 is damage energy



🔗 ENDF reaction numbers

Transmutation reactions

Reactions that result in a change of the isotope

No transmutation

(n, elastic)

(n, inelastic)

(n, heating)

Element transmutation

(n,p)

(n,alpha)

(n,fission)

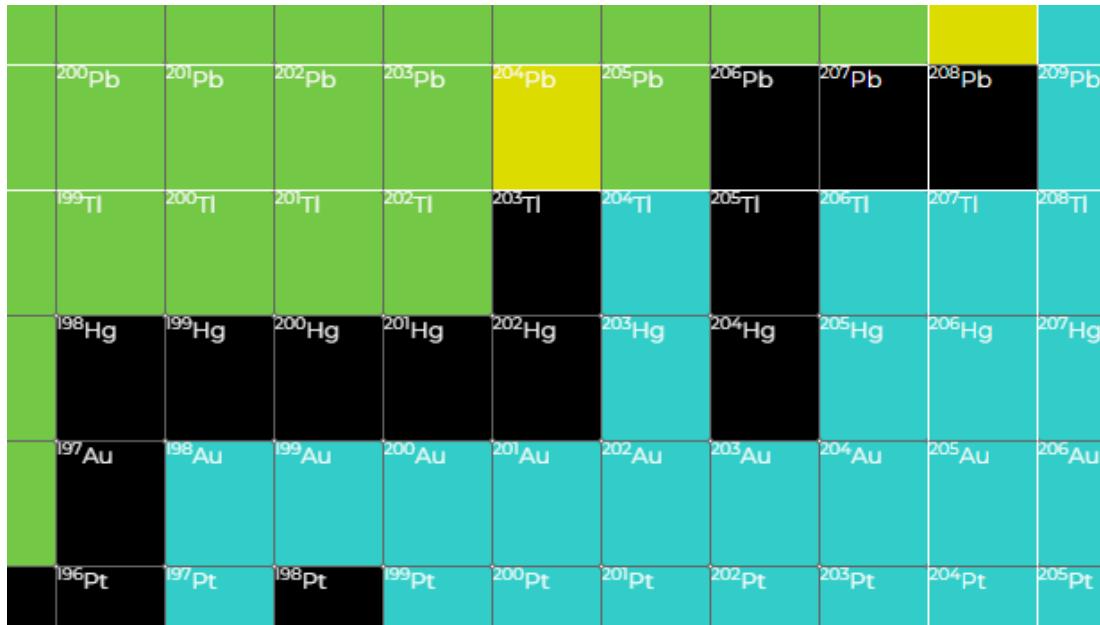
Be9(n,2n)2He4

Isotope transmutation

(n, gamma)

Pb208(n,2n)Pb207

Transmutation of lead to gold



- 1 stable isotope of gold Au_{79}^{197}
- 3 natural isotopes of lead
 - Pb_{82}^{204} -3 protons, -4 neutrons
 - Pb_{82}^{206} -3 protons, -6 neutrons
 - Pb_{82}^{207} -3 protons, -7 neutrons
 - Pb_{82}^{208} -3 protons, -8 neutrons
- 2 reactions for converting gold to lead
 - $\text{Pb}^{204} (\text{n},3\text{n}pa) \text{Au}^{197}$
 - $\text{Pb}^{204} (\text{n},\text{nta}) \text{Au}^{197}$
- No cross section data found in ENDF

Image source IAEA

Q values

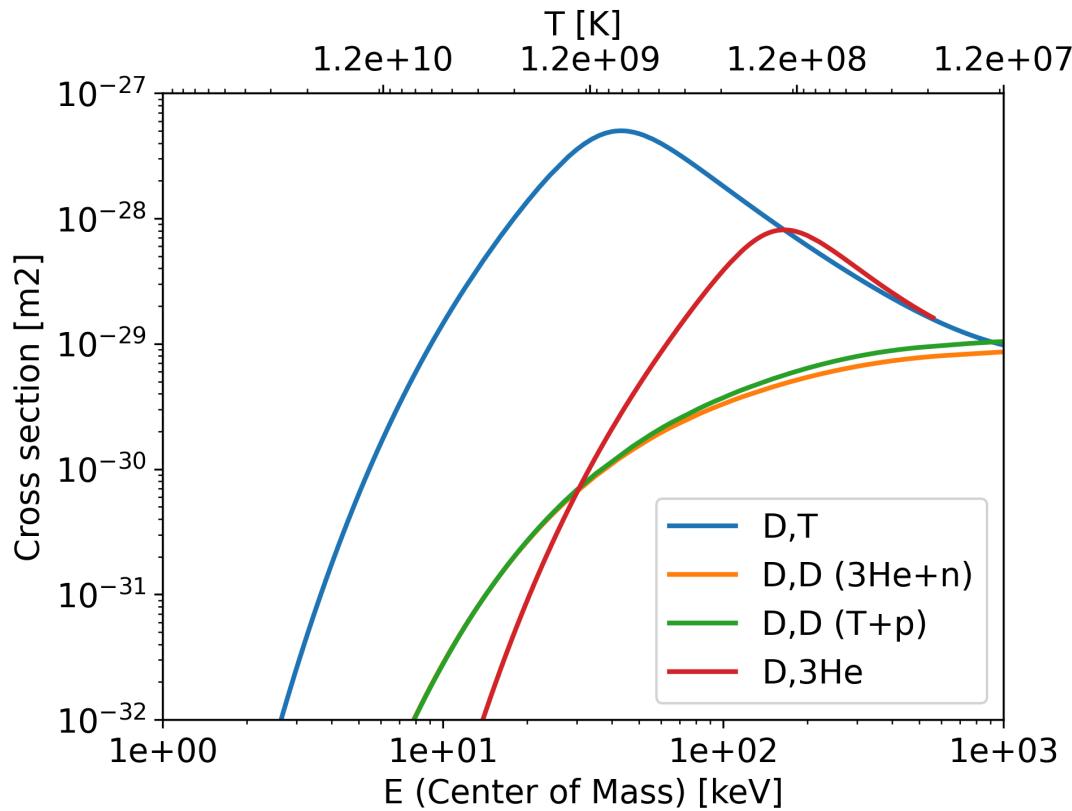
Amount of energy absorbed (-ve) or release (+ve) during the nuclear reaction

Reaction	Energy release [MeV]	Threshold reaction
Be9(n,2n)	-1.6	Yes
Pb208(n,2n)	-7.3	Yes
Li6(n,t)	4.8	No
Li7(n,nt)	-2.4	Yes

Mass and Binding energy converted to kinetic energy

Online Q value calculator at NNDC

Fusion fuels



Q values of fusion fuel reactions

Reaction	Energy release (MeV)
D + T → He ⁴ + n	17.6
D + D → He ³ +n	3.3
D + D → T + p	4.0
D + He ³ →He ⁴ +p	18.3 *

- No neutron emitted

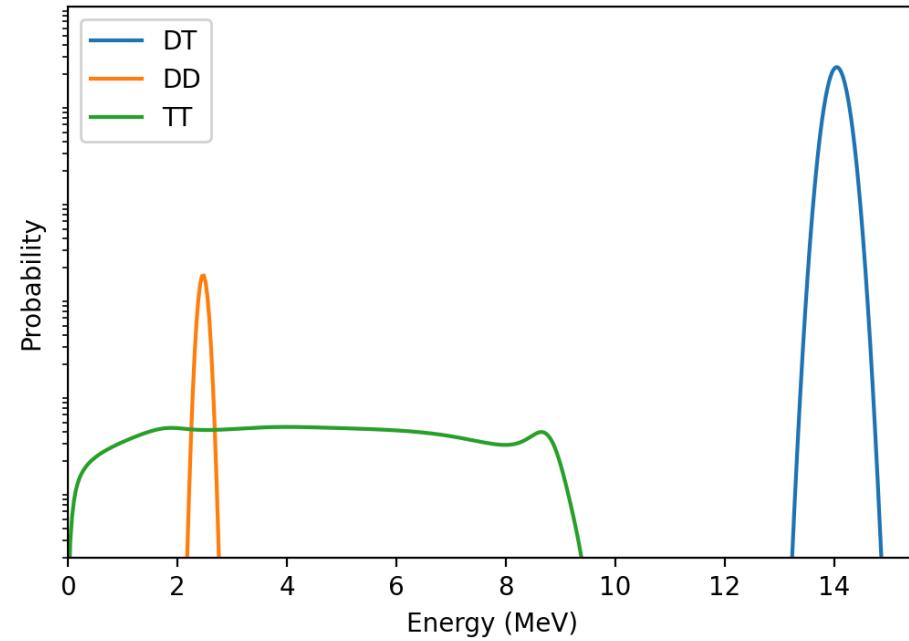
Aneutronic Fusion fuels

- Neutrons are not emitted in the primary fuel reaction
- Neutrons can be emitted by reactions with the products
- Energy capture via direct conversion or divertor?

Reaction	Energy release [MeV]
$D + Li^6 \rightarrow 2He^4$	22.4
$P + Li^6 \rightarrow He^4 + He^3$	4.0
$He^3 + Li^6 \rightarrow He^4 + p$	16.9
$He^3 + He^3 \rightarrow He^4 + 2p$	12.86
$p + Li^7 \rightarrow 2He^4$	17.2
$p \rightarrow B^{11} \rightarrow 3He^4$	8.7
$p \rightarrow N^{15} \rightarrow C^{12} + He^4$	5.0

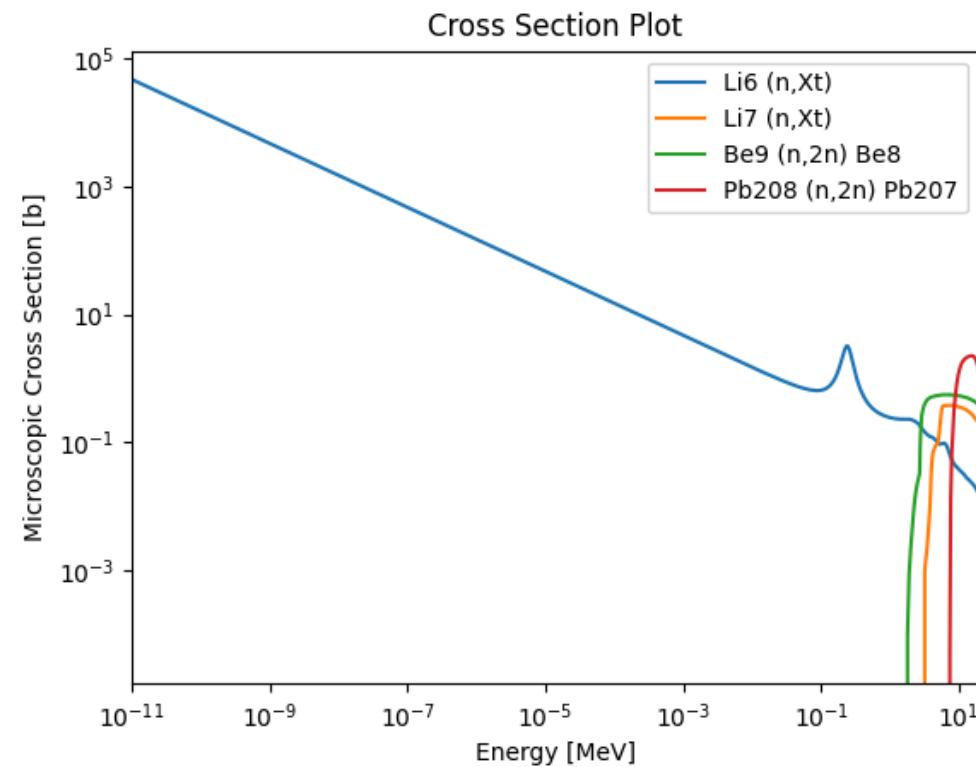
Energy of neutrons from DT fuel

- A DT plasma has several fusion reactions.
- DT is the most likely reaction.
- DD and TT reactions also occur with lower probabilities.
- All reactions emit different energy neutrons.



Microscopic Cross Section

- Measured in Barns ($1 \text{ barn} = 10^{-28} \text{ m}^2$)
- Energy dependant
- Cross section evaluations exist for:
 - different incident particles
 - different nuclides
 - different interactions.
- Important neutron reactions plotted
 - Tritium breeding
 - Neutron multiplication



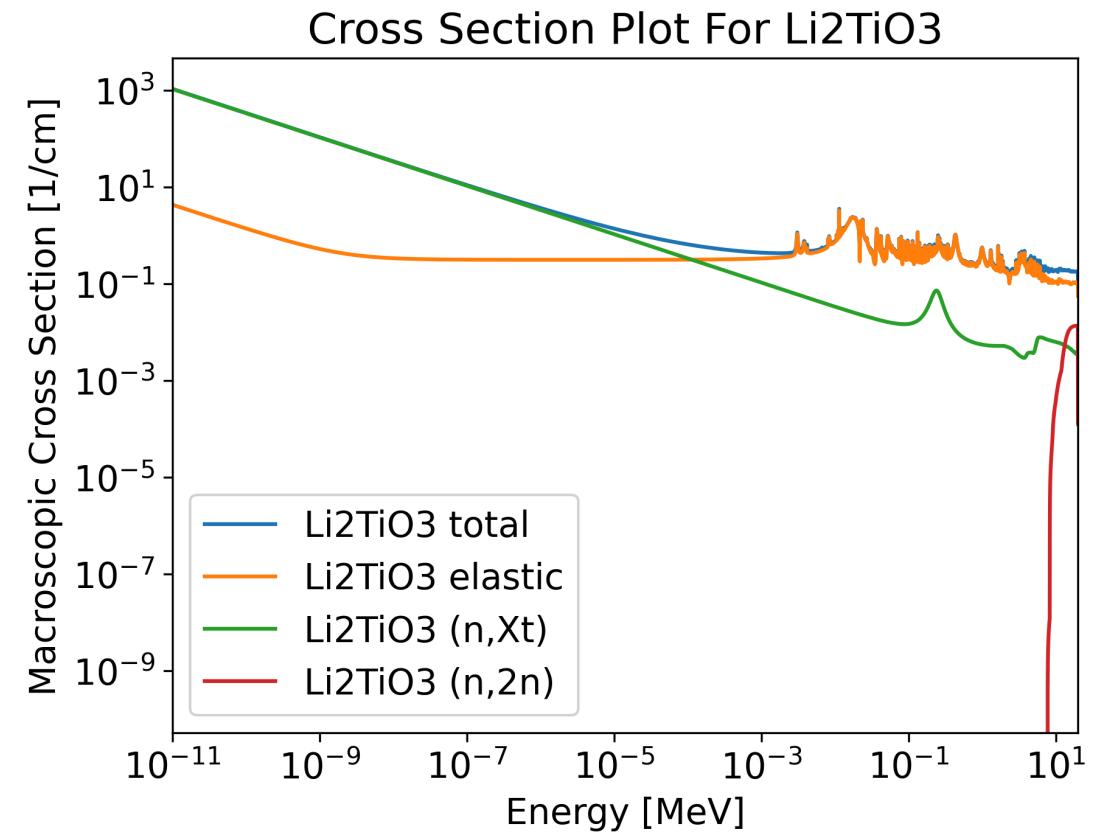
Reaction rate equation

- The reaction rate (RR) can be found by knowing the number of neutrons per unit volume (n), the velocity of neutrons (v), the material density (ρ), Avogadro's number (N_a), the microscopic cross section at the neutron energy (σ_e) and the atomic weight of the material (M).
- This reduces down to the neutron flux (ϕ), nuclide number density (N_d) and microscopic cross section σ_e .
- This can be reduced one more stage by making use of the Macroscopic cross section (Σ_e).

$$RR = \frac{nv\rho N_a \sigma_e}{M} = \phi N_d \sigma_e = \phi \Sigma_e$$

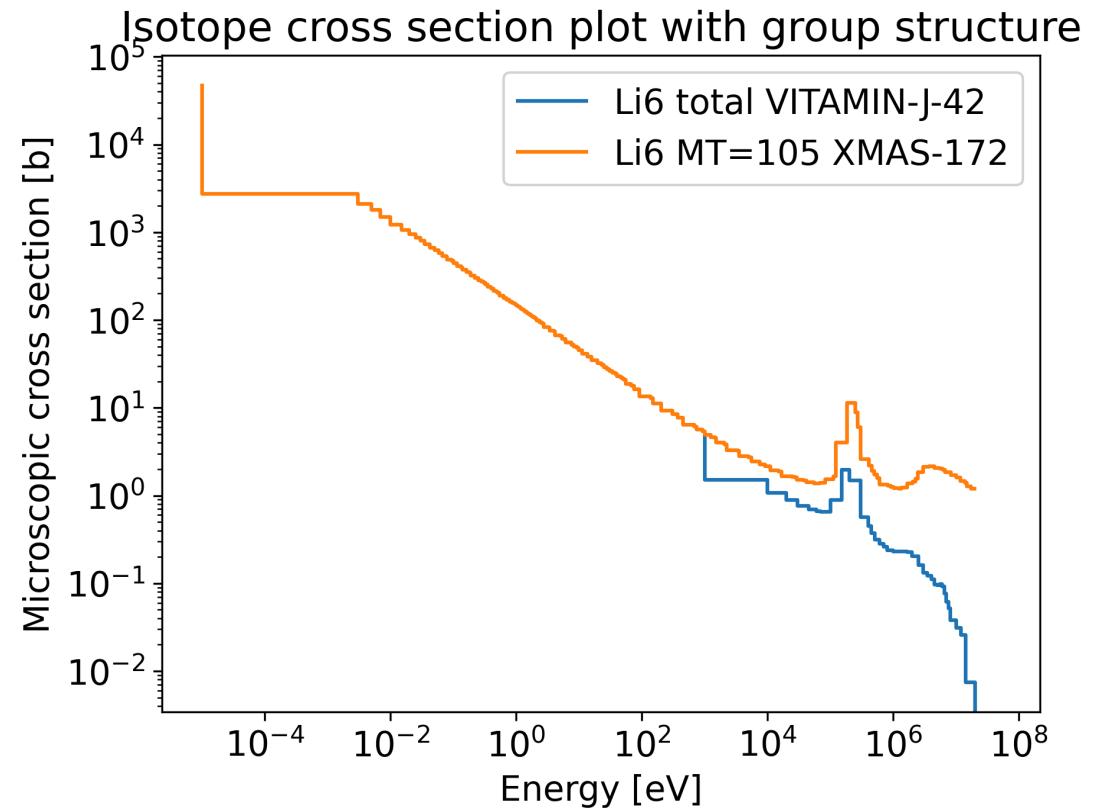
Macroscopic cross section

- Lithium metatitanate has a material density of 3.4 g/cm^3
- When plotting materials the Macroscopic cross section accounts for number density of the different isotopes
- Units for Macroscopic cross section are cm^{-1}



Multigroup cross sections

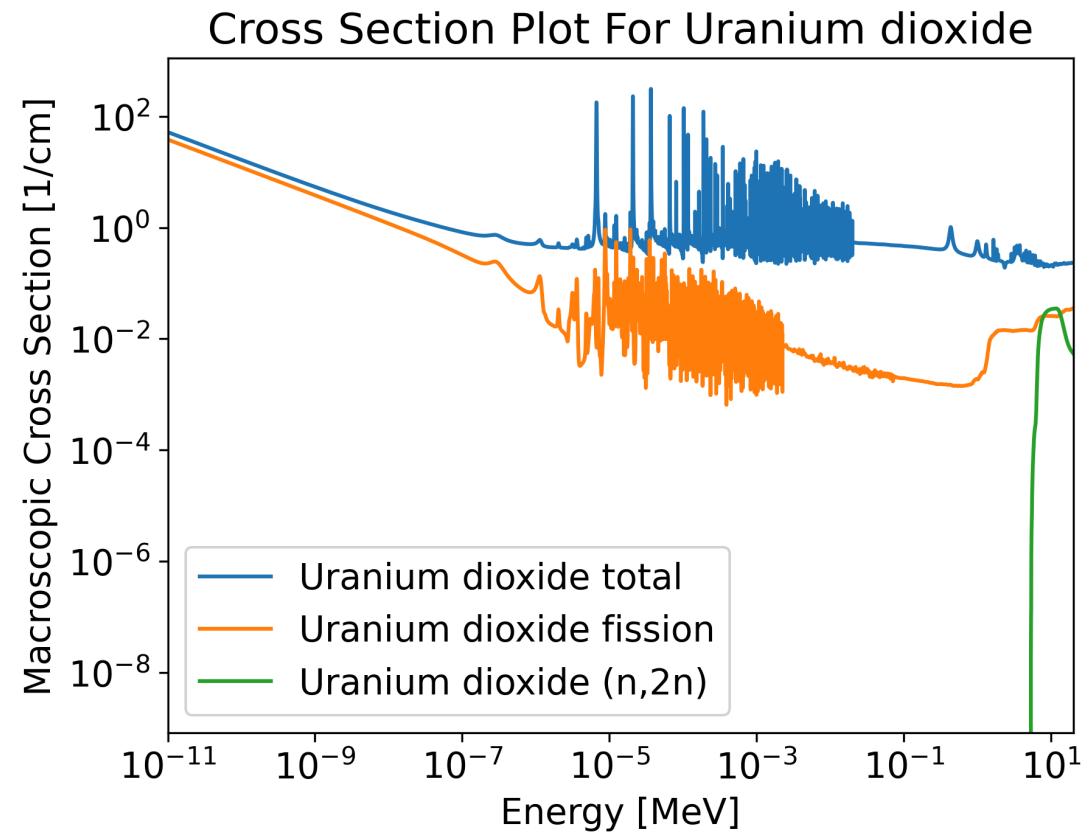
- Discretize a continuous distribution
- Histogram of average cross section in each energy bin
- Continuous cross section has rules for interpolation that can be accounted for.
- Groups are not equally spaced.
- Structures are optimized for different energy ranges (fission, fast fission, fusion etc)



Cross section regions

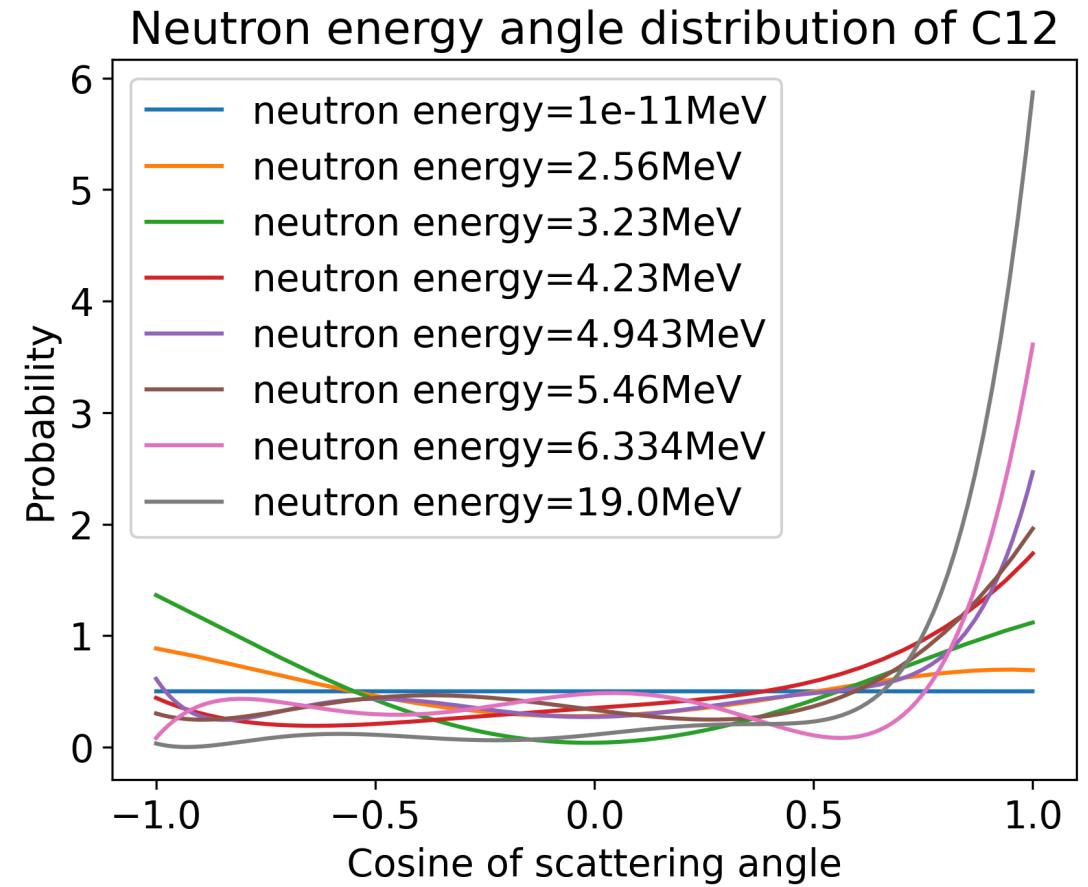
Reactions have characteristics

- resolved resonance
- unresolved resonance
- $1/v$ section
- thresholds
- scattering



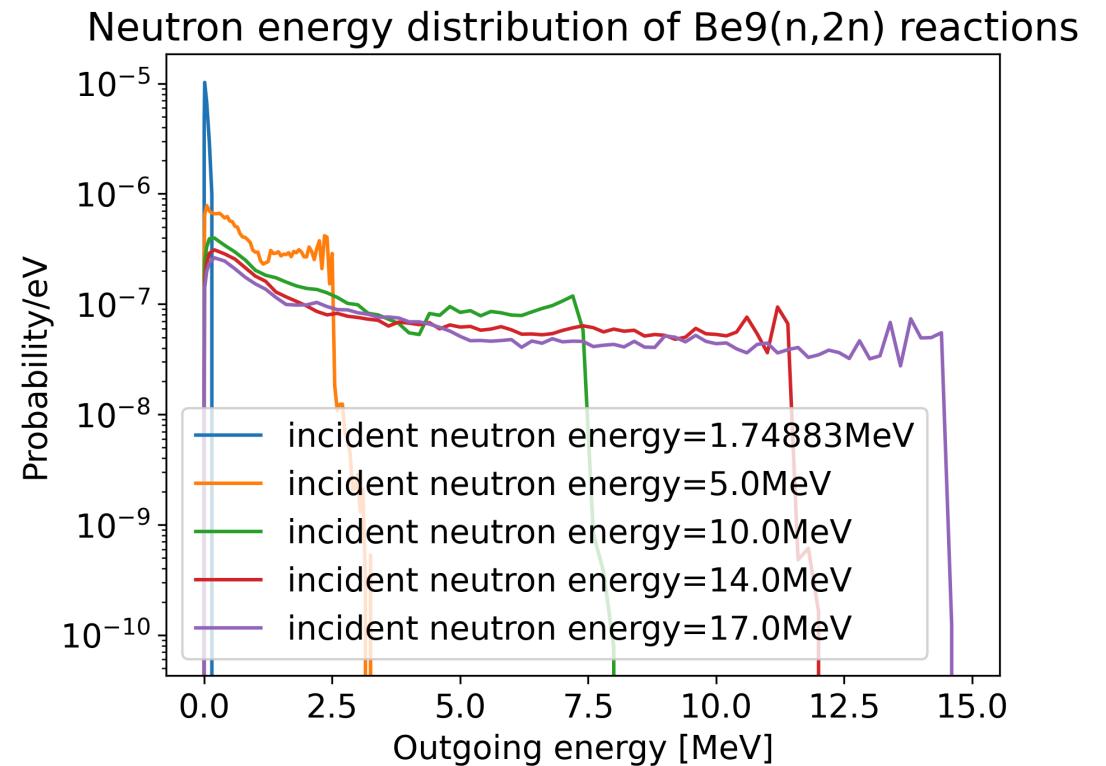
Angular distribution

- The scattering angle varies depending on the energy of the incident neutron
- Low energy neutrons have isotropic scattering (even probability in all directions)
- High energy neutrons are more likely to have a low deflection angle and are forwards bias.



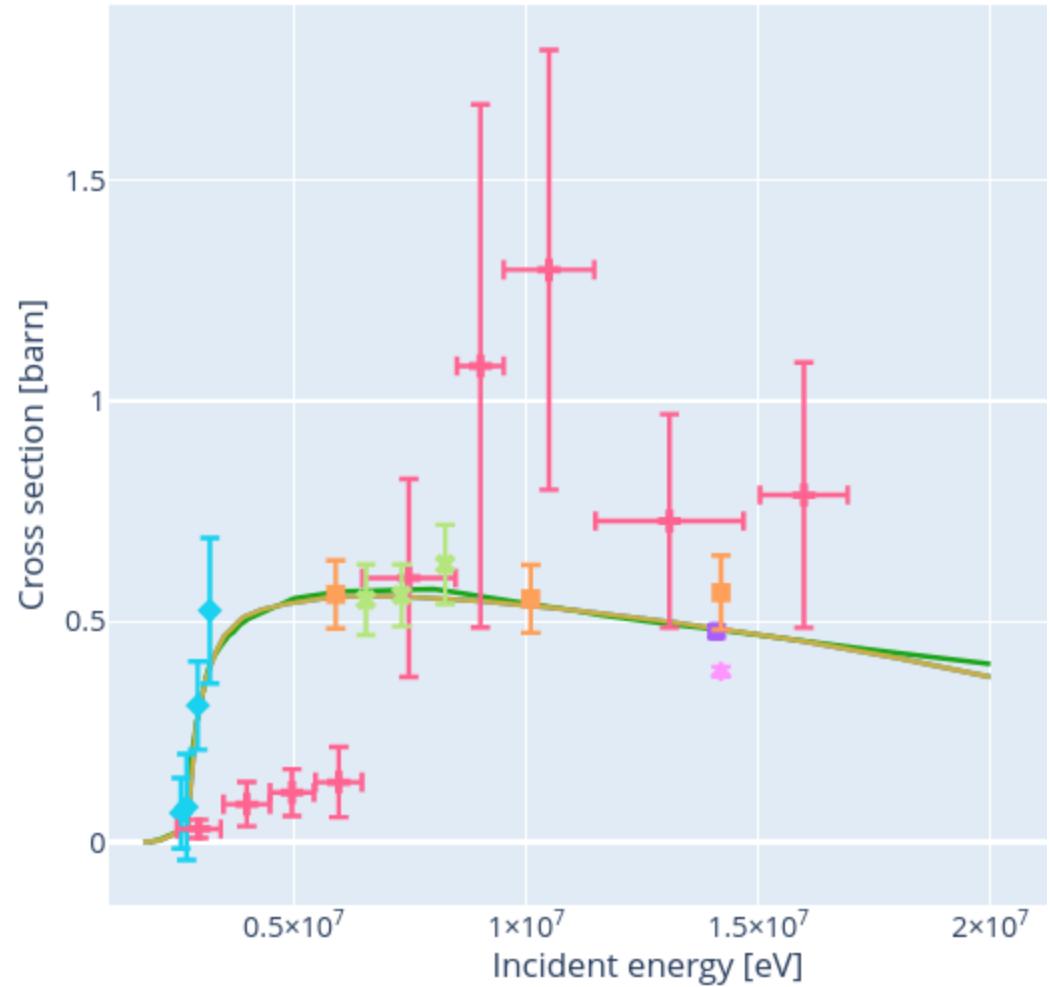
Energy distribution

- There is also data on neutrons released in reactions such as $(n,2n)$.
- The $(n,2n)$ reaction is a threshold reaction and requires energy.
- No run away chain reaction possible.



Experimental data

- Availability of experimental data varies for different reactions and different isotopes.
- Typically the experimental data is then interpreted to create evaluation libraries, such as ENDF, JEFF, JENDL, CENDL.



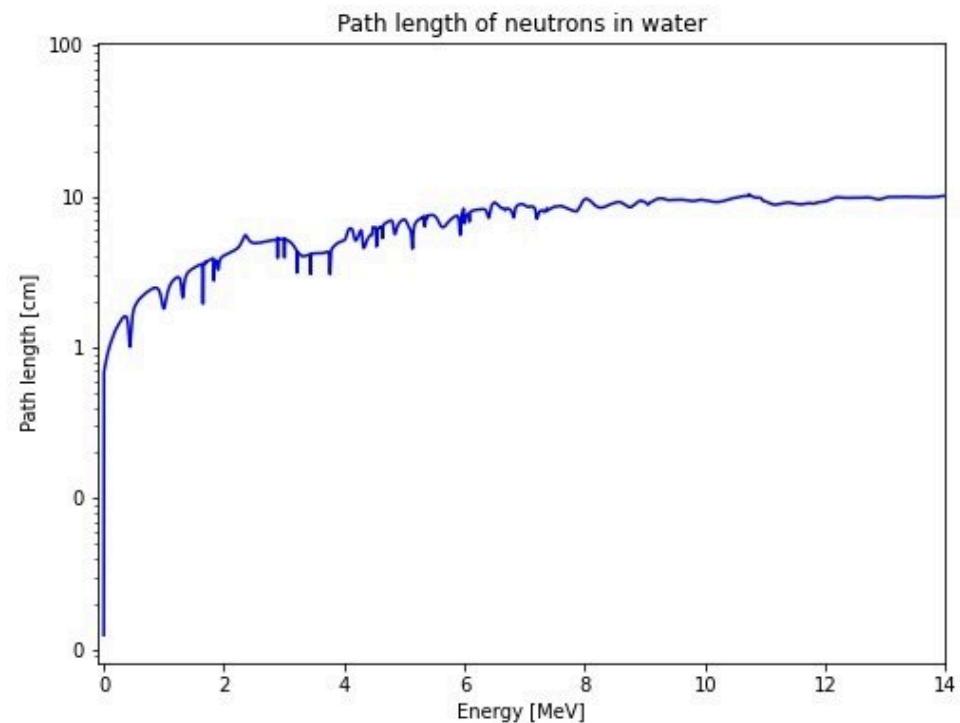
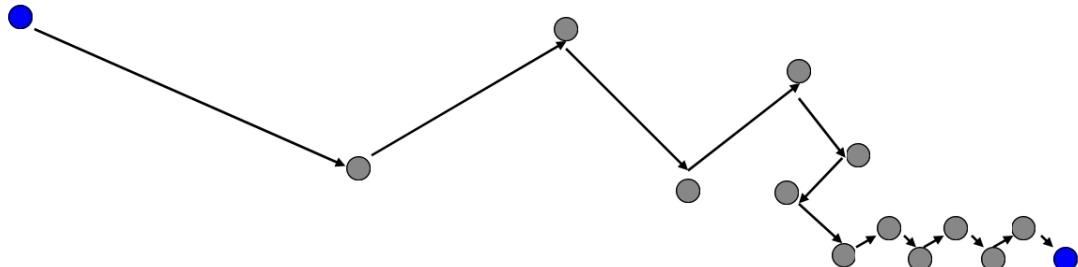
Nuclear data libraries

There are several groups that produce and distribute nuclear data

- TENDL 2023  2850 neutron
- JENDL 5  795 neutron
- ENDF/B-VIII.0  557 neutron
- JEFF 3.3  562 neutrons
- BROND 3.1  372 neutrons
- FENDL 3.2b  191 neutron
- CENDL 3.2  272 neutron

Path length

- Path length = $1 / \Sigma_T$
- A 14MeV neutron will lose energy via scattering interactions
- As the neutron energy decreases the path length also decreases
- Path length at thermal energy is more constant

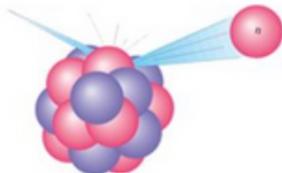


Energy loss

The average logarithmic energy decrement (or loss) per collision (ξ) is related to the atomic mass (A) of the nucleus

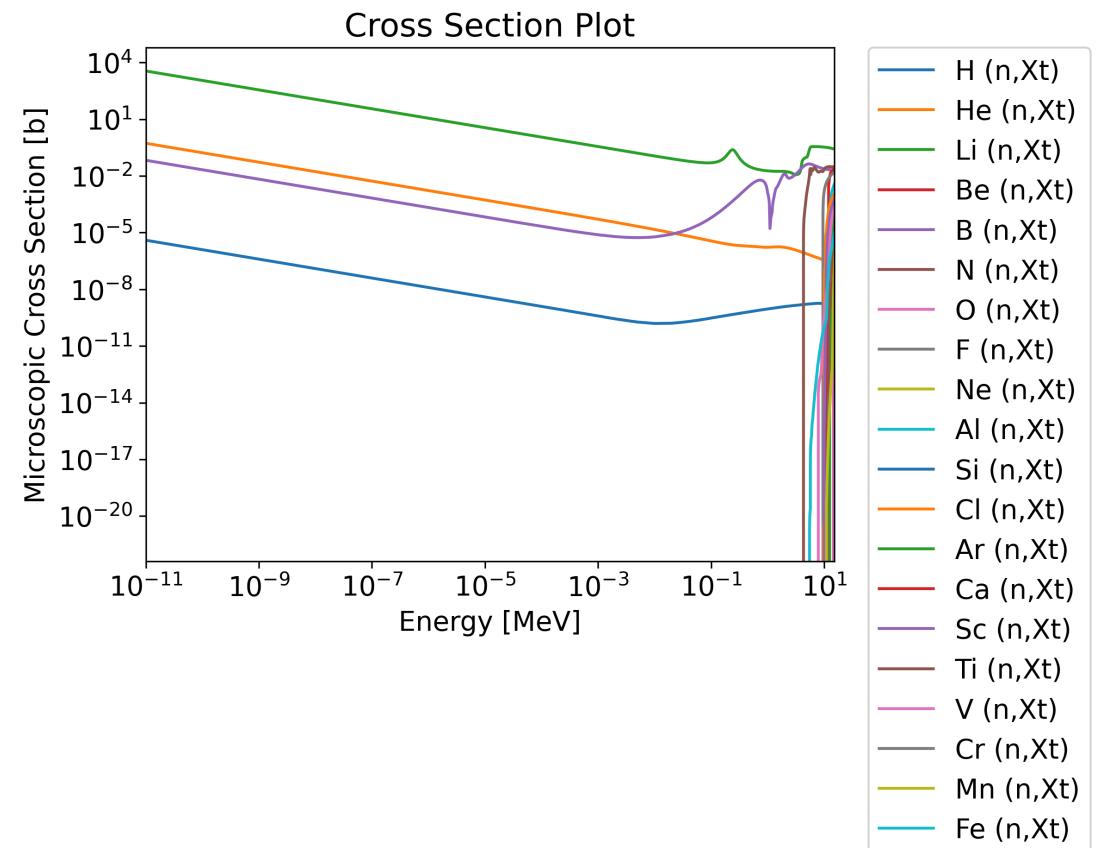
$$\xi = 1 + \frac{(A-1)^2}{2A} \ln \frac{(A-1)}{(A+1)}$$

	Hydrogen	Deuterium	Beryllium	Carbon	Uranium
Mass of nucleus	1	2	9	12	238
Energy decrement	1	0.7261	0.2078	0.1589	0.0084



Why lithium

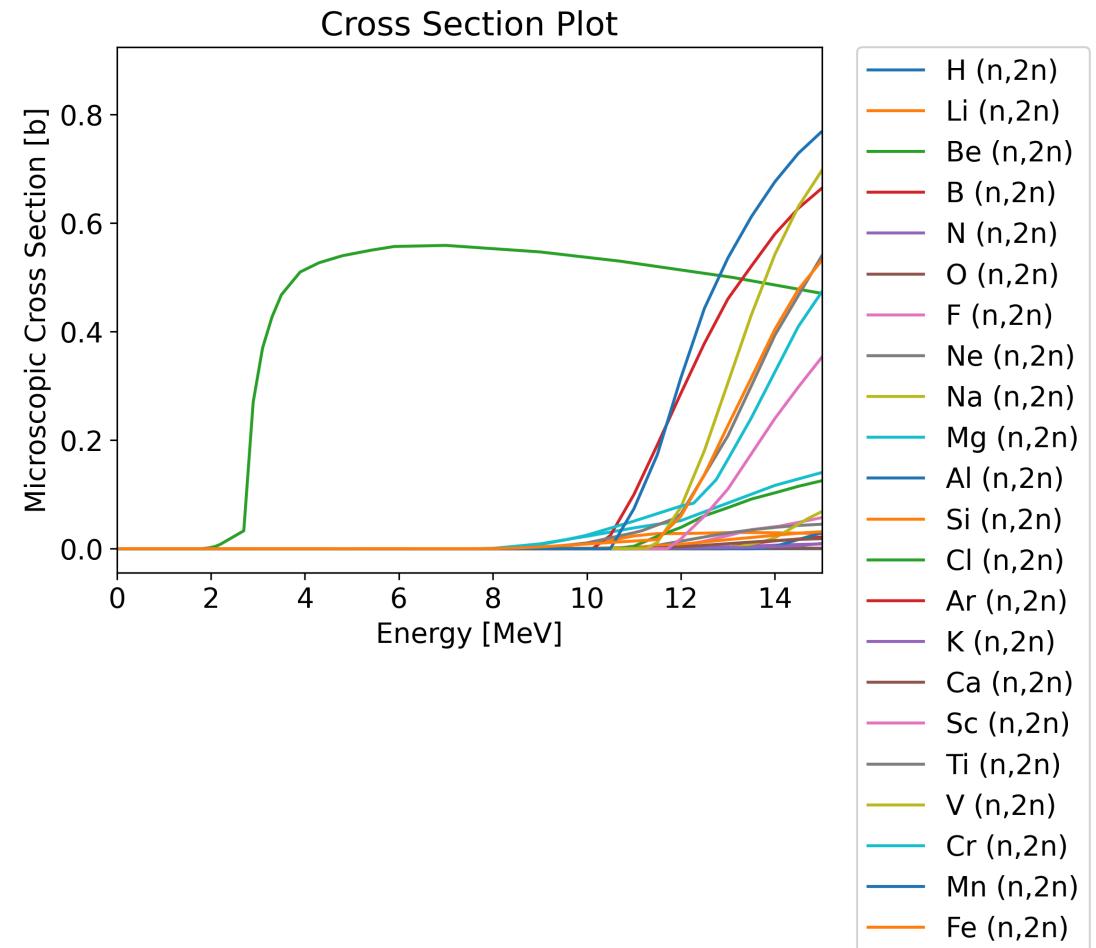
- Lithium has a particularly high cross section for tritium production
- Li6 has a very high cross section at low neutron energies
- Li7 has a reasonable cross section at high neutron energies
- Other reaction channels are relatively low
- Often alloyed with Si or other elements to improve material properties (e.g. flammability)



- Elements up to Iron plotted

Why beryllium

- Beryllium has the lowest threshold energy for any isotope with a $n,2n$ reaction.
- This means even low energy 3MeV neutrons can undergo $(n,2n)$ reactions.
- Often alloyed with Ti or other elements to improve material properties (e.g. swelling due to retention)
- Lead is also a popular choice for a neutron multiplier



- Elements up to Iron plotted

Other materials

Tungsten

- High atomic number = good gamma attenuation
- High neutron capture resonances = good neutron attenuation

Water

- High hydrogen content = excellent neutron moderator

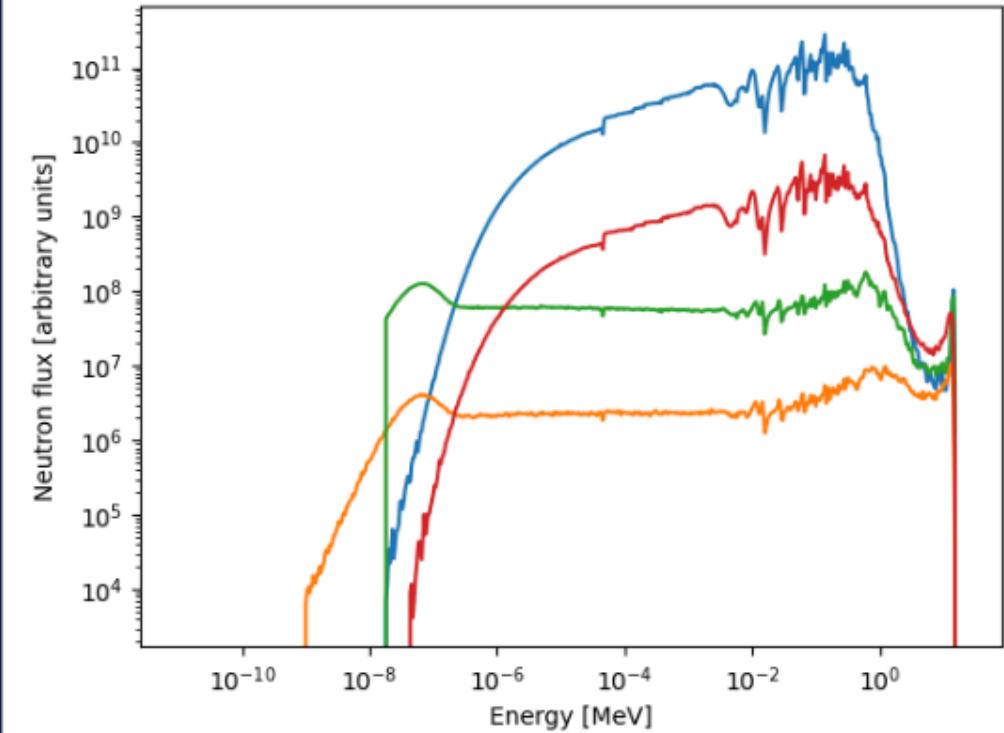
Helium 4

- Low interaction cross sections and low density = transparent to neutrons and gammas

Neutron spectra through materials

By knowing the materials present can you identify which blanket results in which spectrum

- FLiBe, Molten salt, typically 90% enriched Li6
- HCPB, helium cooled pebble bed, typically 60% enriched Li6
- HCLL, helium cooled lithium lead, typically 90% enriched Li6
- WCCB, Water cooled ceramic breeder, typically 60% enriched Li6
- WCLL, water cooled lithium lead, typically 90% enriched Li6
- Liquid Lithium, typically natural enrichment



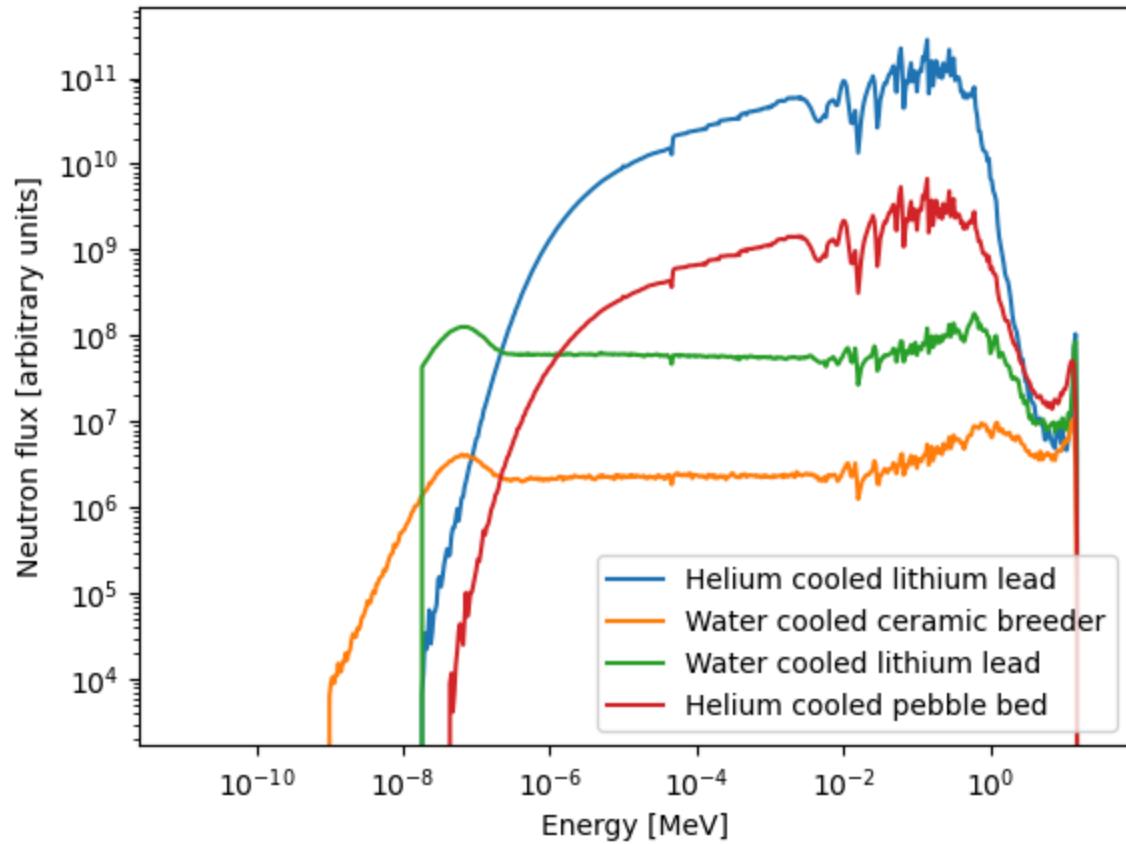
The neutron spectra at the back of HCPB, WCCB, WCLL, HCLL blankets are shown.
Which of the plots shows the neutron energy spectra at the back of a WCCB blanket

Blue

Orange

Green

Red

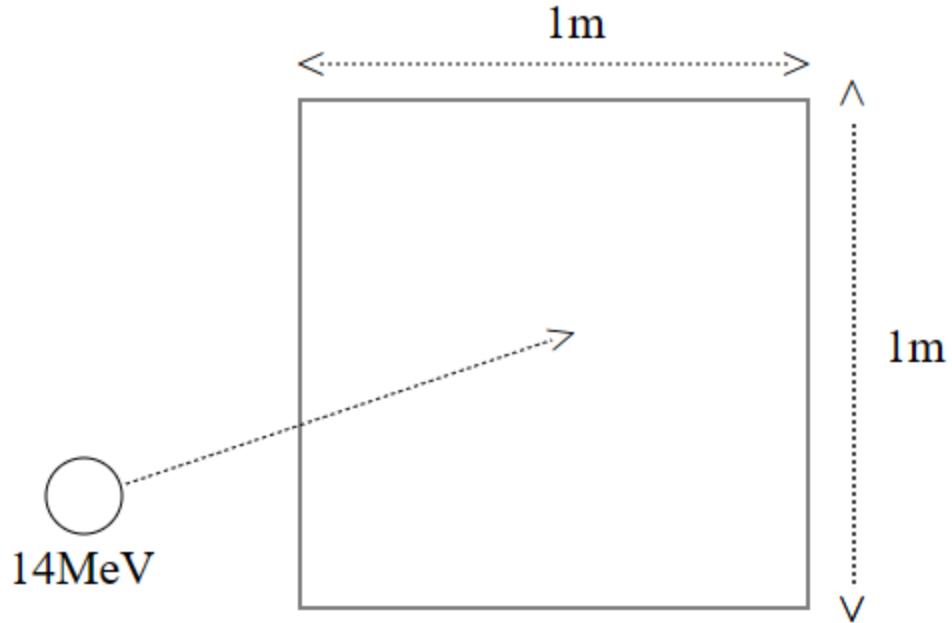


Prompt responses

- Neutron wall loading
- Heating
- Tritium breeding
- Dose

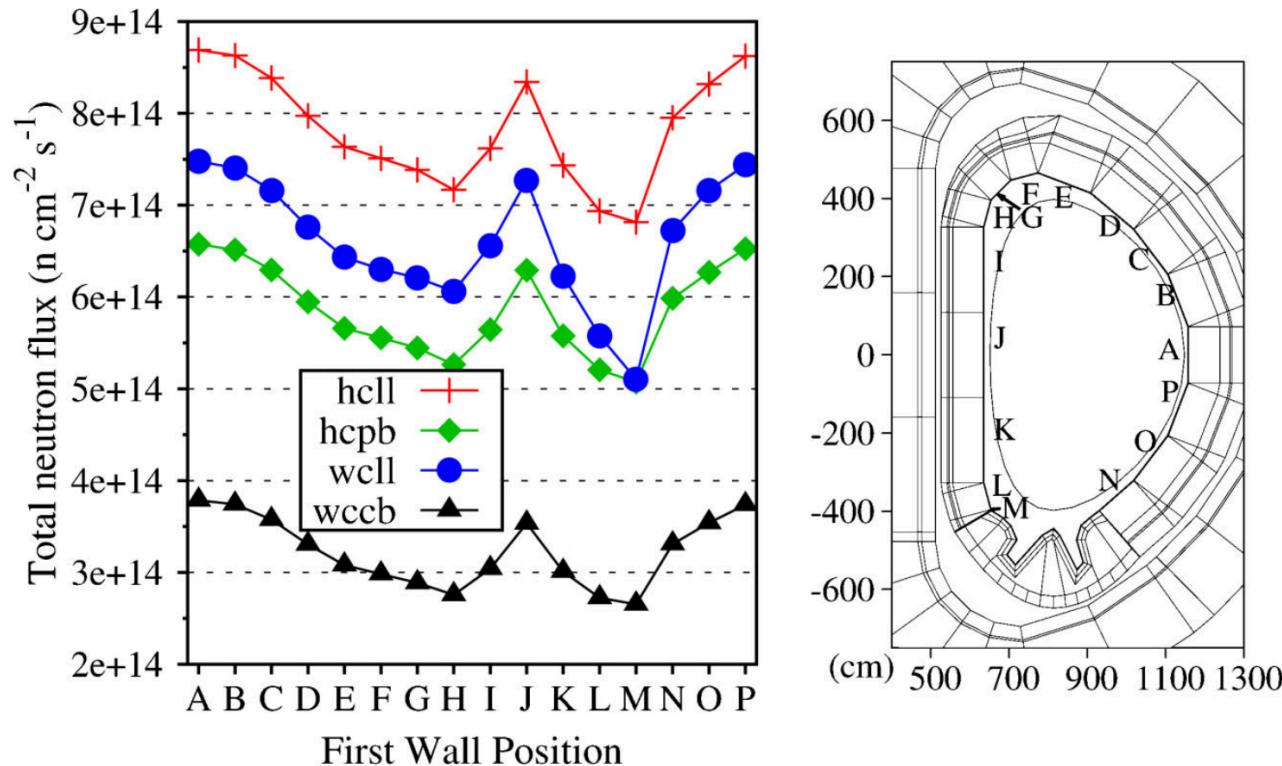
Neutron wall loading

- Energy carried by uncollided source neutrons incident on a unit area of first wall per unit time
- Units typically used MWm^{-2}
- Useful for estimating neutronics results and scaling or comparing results
- For simple source distributions and geometry, can calculate analytically
- Complex source distributions or geometries require more sophisticated methods (e.g Monte Carlo)



Neutron wall example

- Significant poloidal variation of neutron wall loading occur in toroidal magnetic confinement fusion reactors

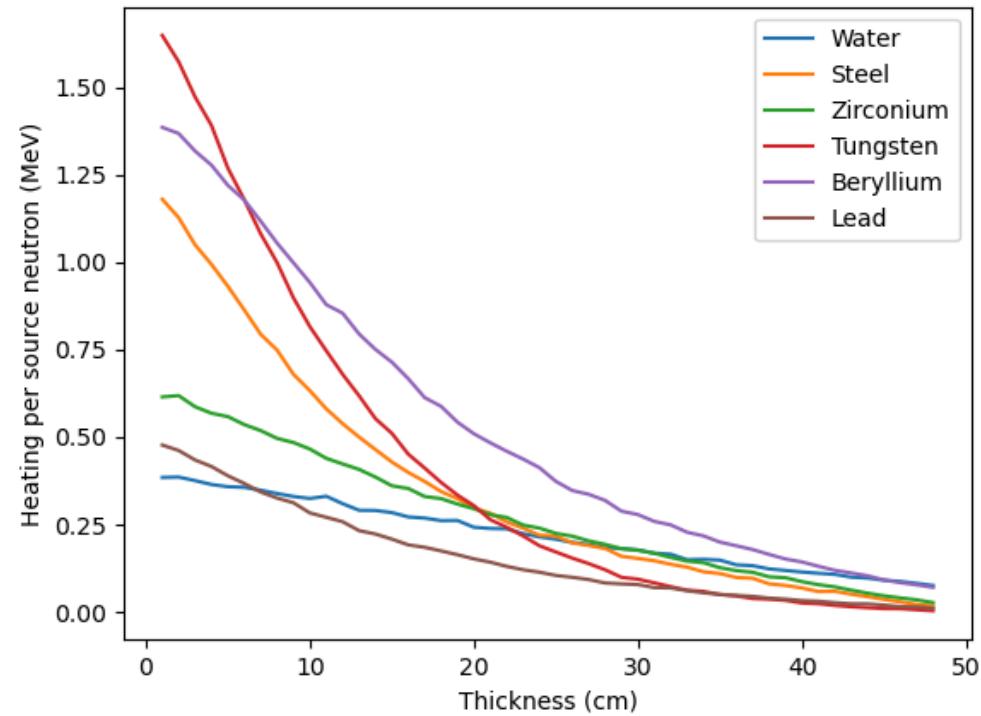


Nuclear Heating

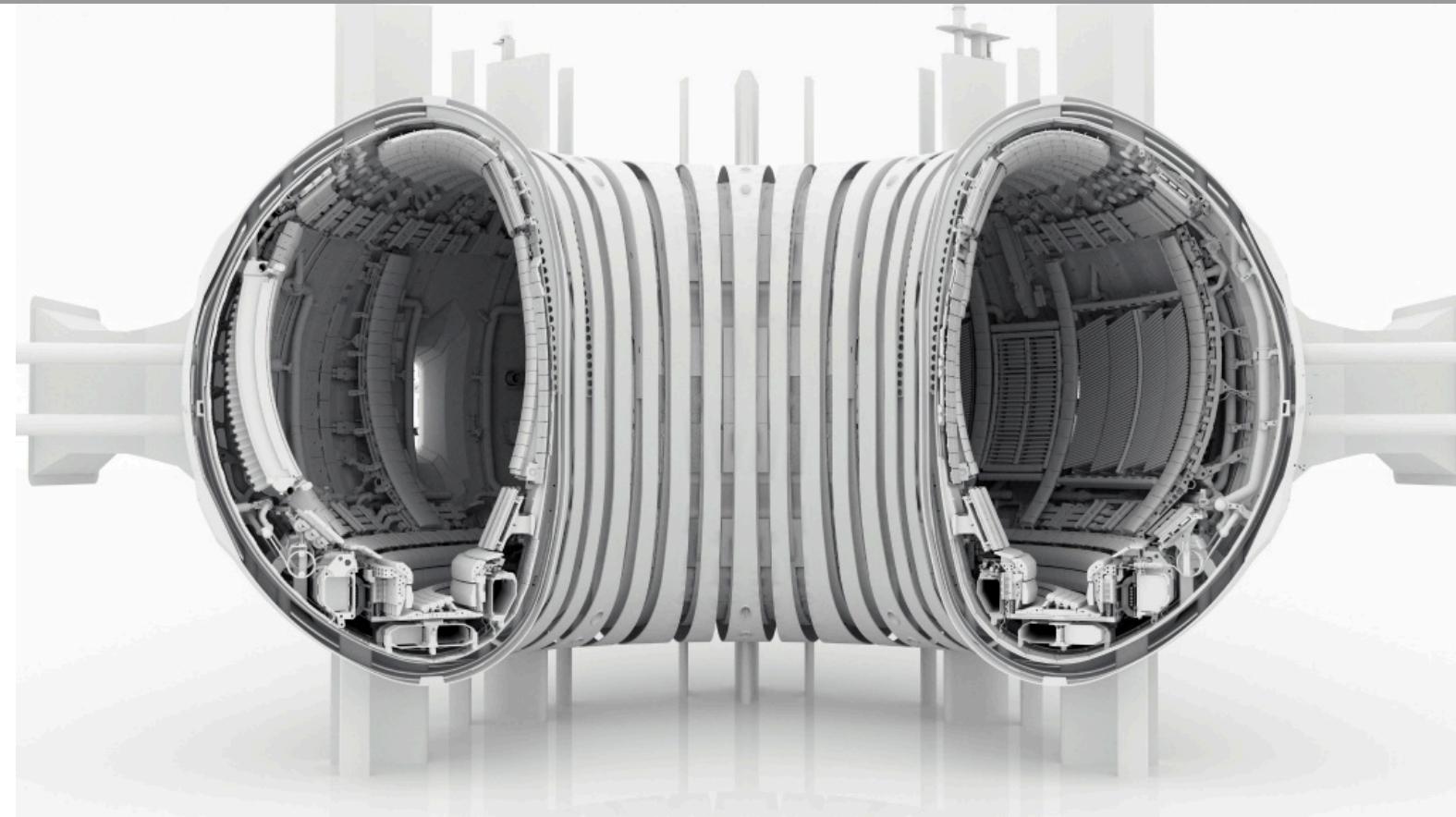
- Energy deposition calculated from the flux using “Kinetic Energy Released in MAterials” (KERMA) factors
- Energy lost by a neutron from a collision is assumed to be deposited locally
- Gamma photons produced by neutrons are transported to determine where their energy is deposited (need coupled neutron-photon transport)
- The power density distribution is used in thermal-hydraulics calculations and subsequent structural analysis (e.g. thermal stress)
- Total heating is used for sizing cooling systems
- Nuclear energy multiplication (M_n) is ratio of energy deposited by neutrons and gamma photons in the reactor to neutron energy incident on FW

Nuclear Heating depends on material and location

- At same location with same neutron flux, nuclear heating depends on material
- High-Z materials usually yield higher nuclear heating than low-Z materials
- Gamma heating represents ~85% of nuclear heating in high-Z materials and only ~40% in low-Z materials
- Nuclear heating drops rapidly as we move away from FW

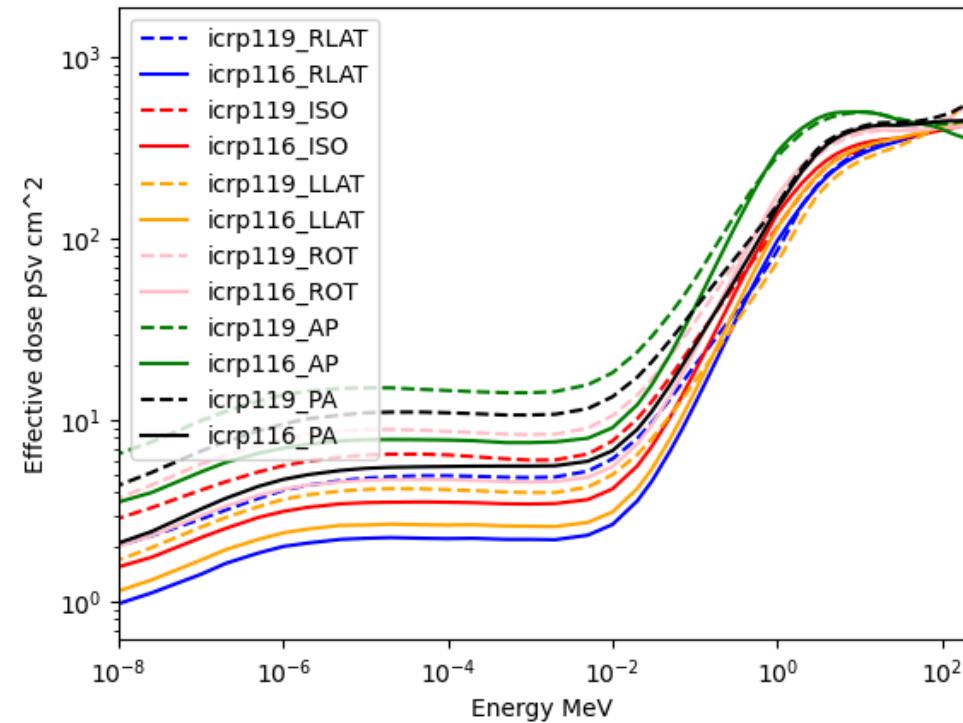


Tritium Breeding Ratio



Instantaneous Dose

- Different types of dose, absorbed, equivalent and effective.
- Effective dose is typically used for dose maps.
- Dose coefficients units of $Sv \cdot cm^2$
- Neutron flux ($particles \cdot cm^{-2} s^{-1}$)
- Resulting dose in Sv per second

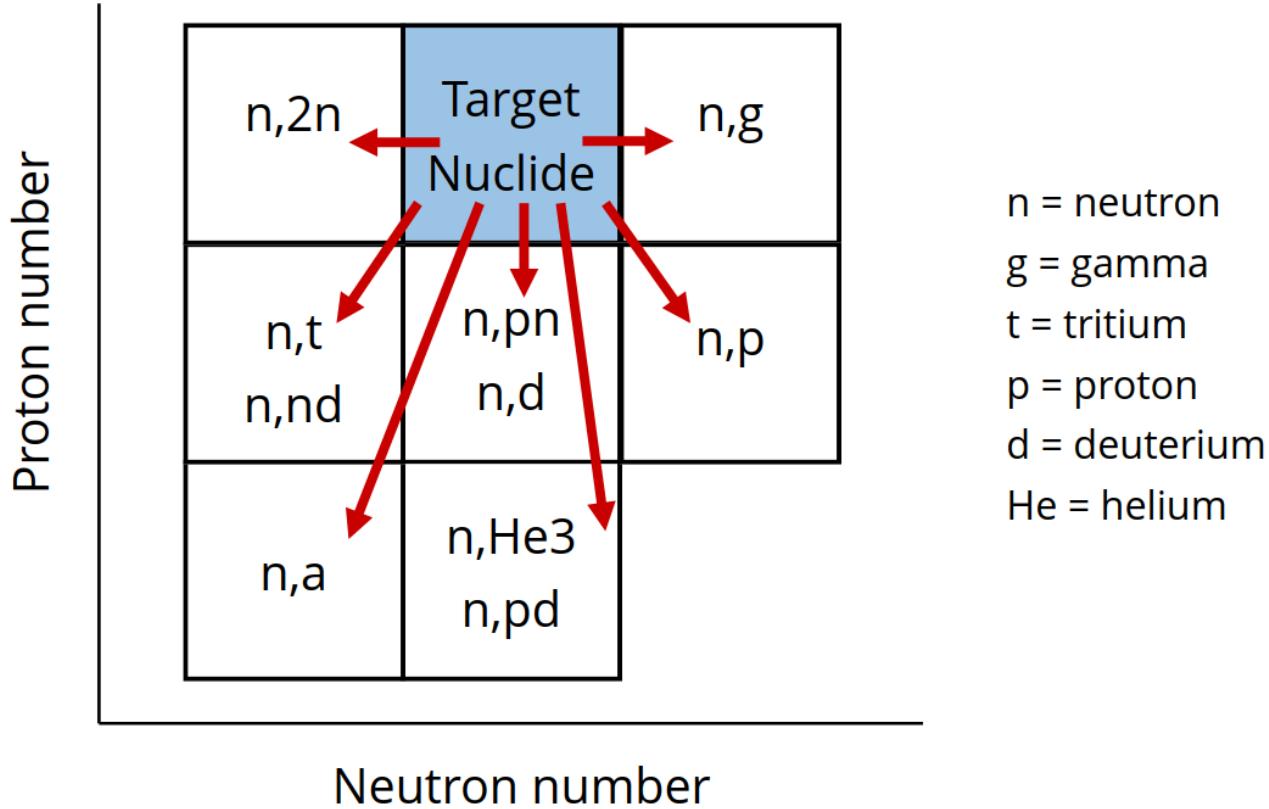


Delayed response

- Activation
- Activity build up and decay
- Emission spectra
- Shut down dose
- Analysis needed to lift or cool components
- Impact of burn up on TBR

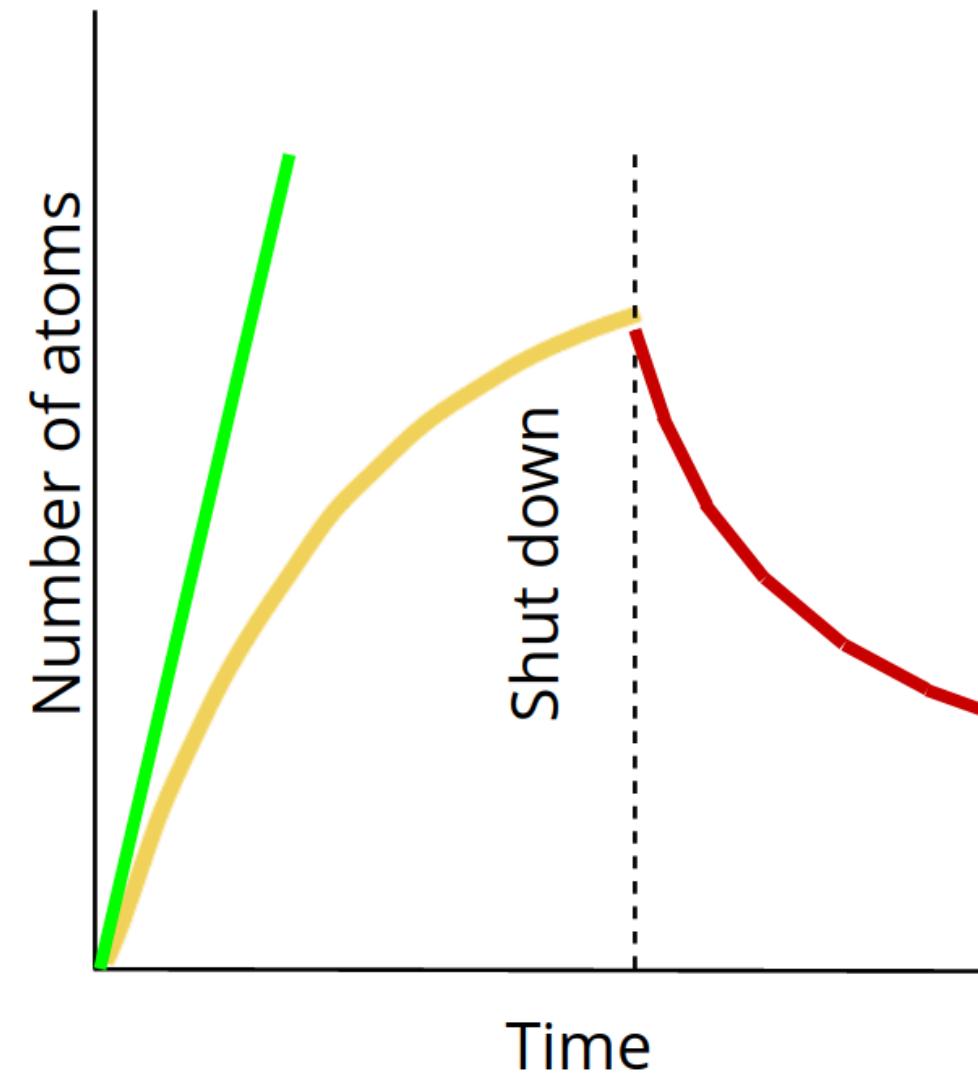
Activation reactions

Common neutron induced reactions

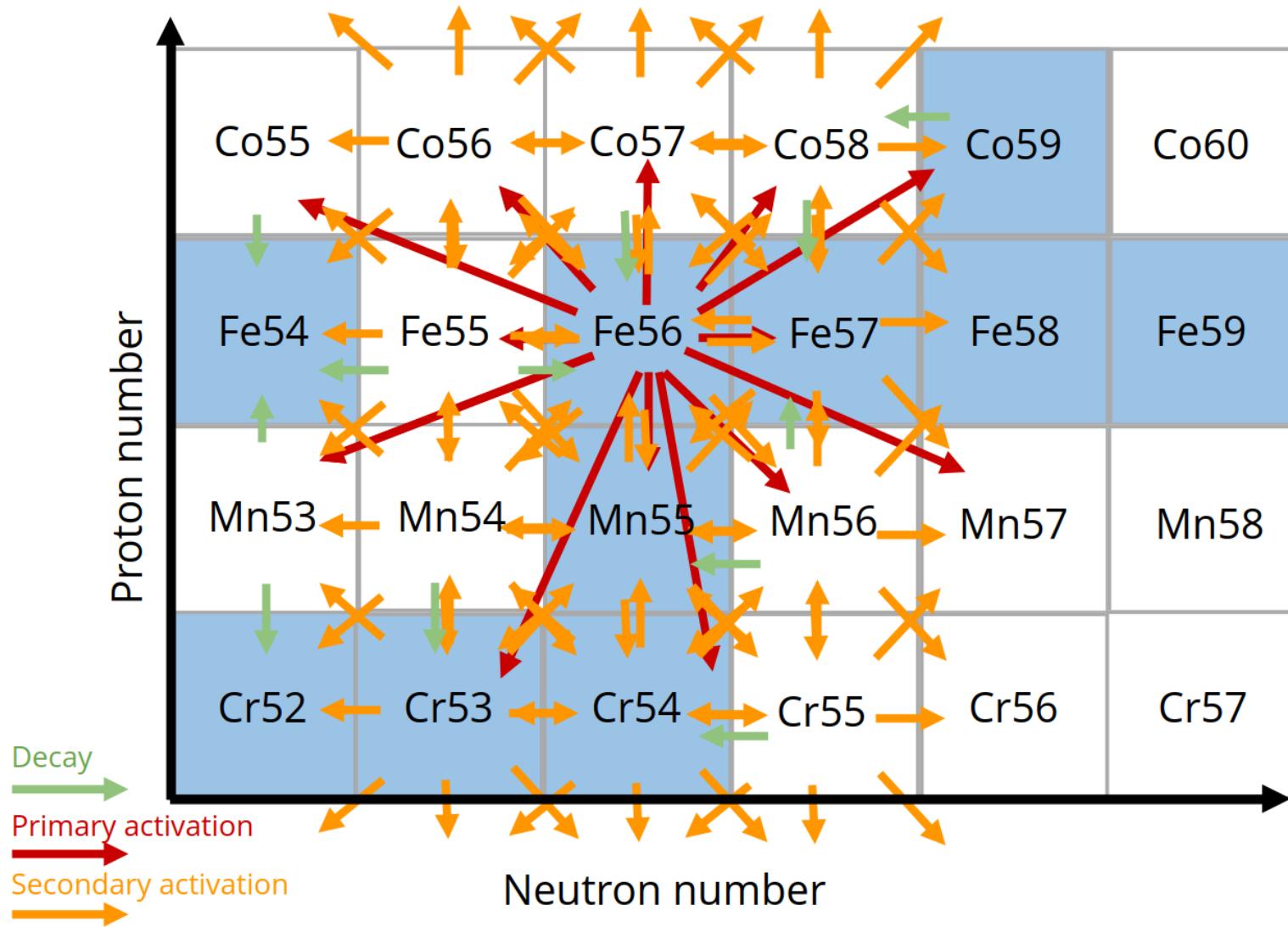


Build up and saturation

- New isotopes created during irradiation
- Radioactive isotopes decay and will eventually reach a point where decay rate is equal to activation rate.
- Decay is more noticeable once the plasma is shutdown.
- The activity is related to the irradiation time and the nuclide half life.

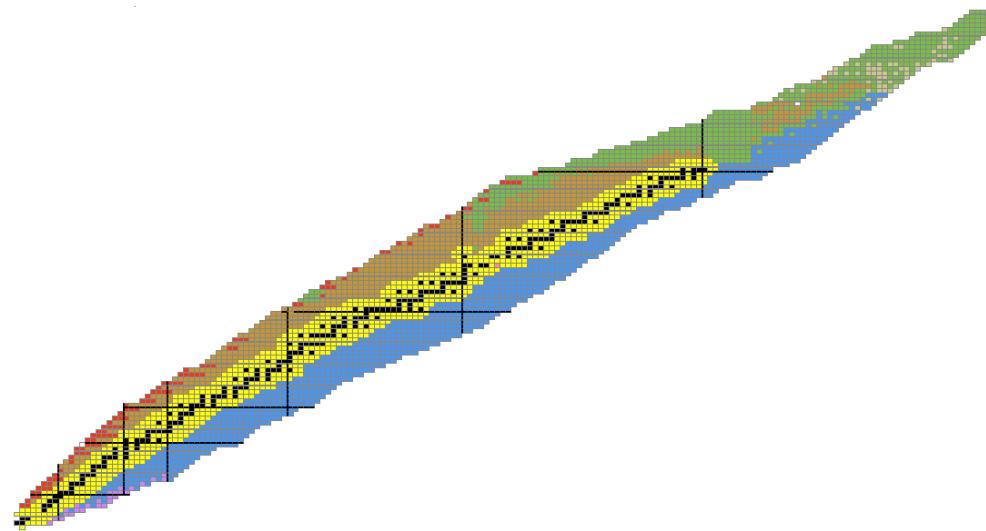


Activation pathways

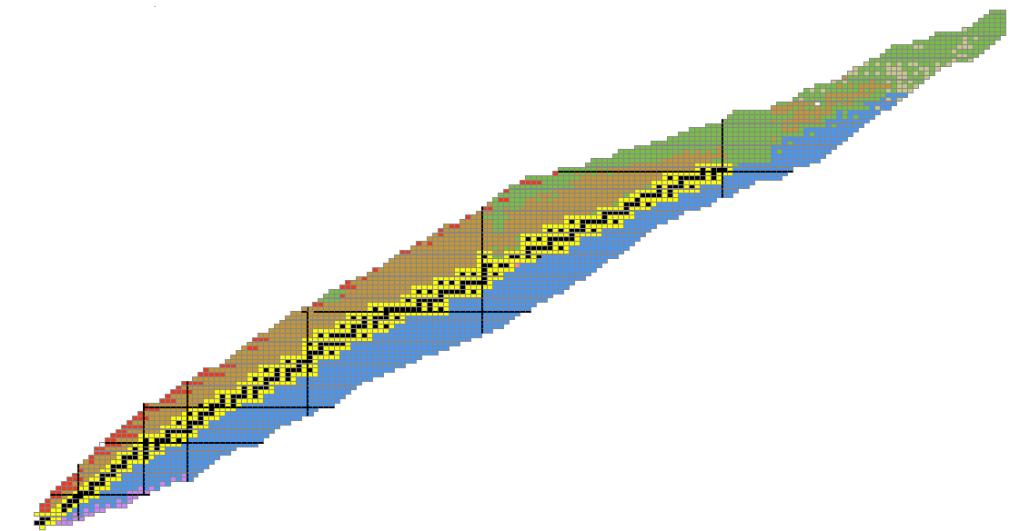


Activation products

- High energy neutron activation

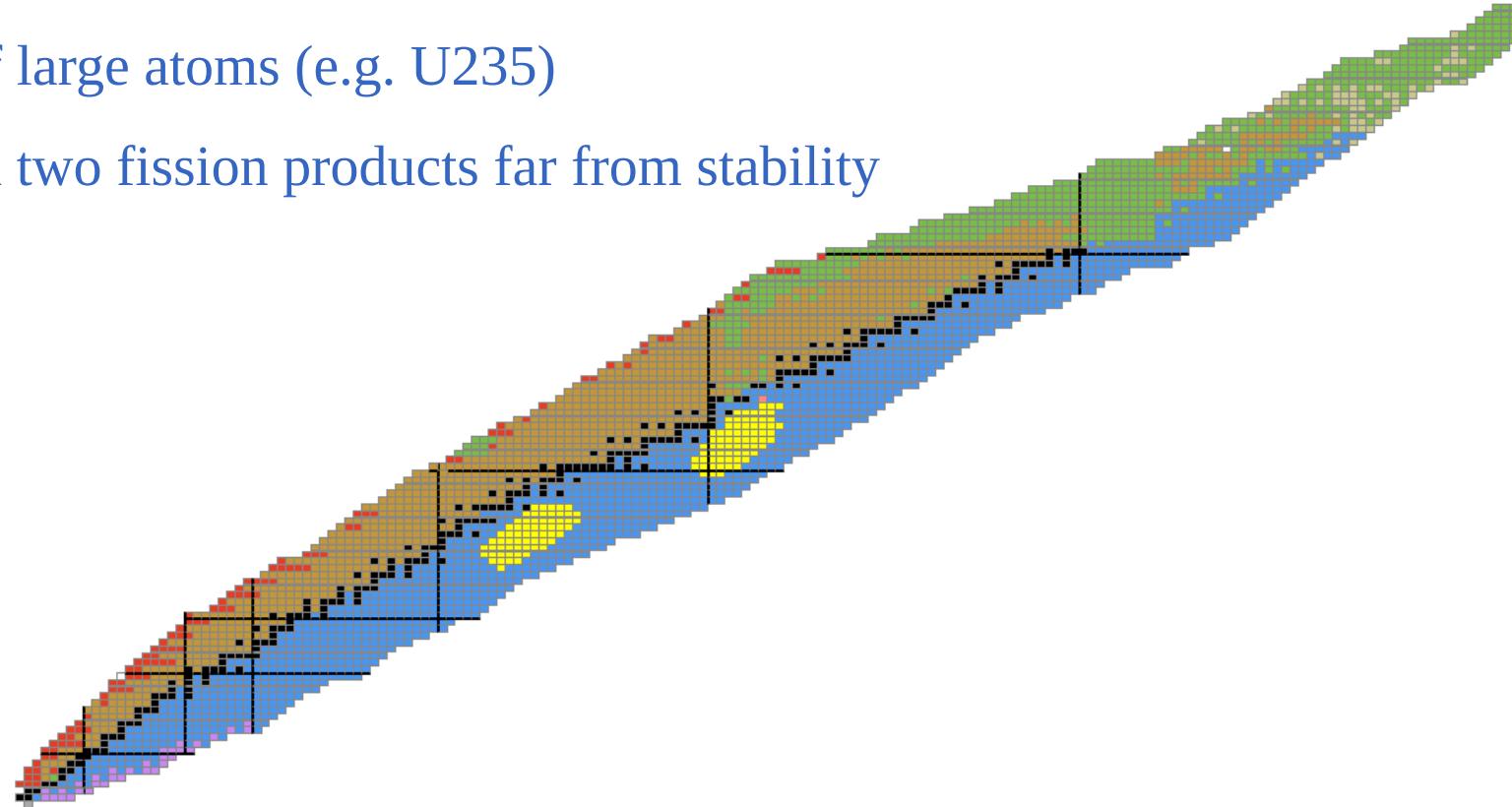


- Low energy neutron activation



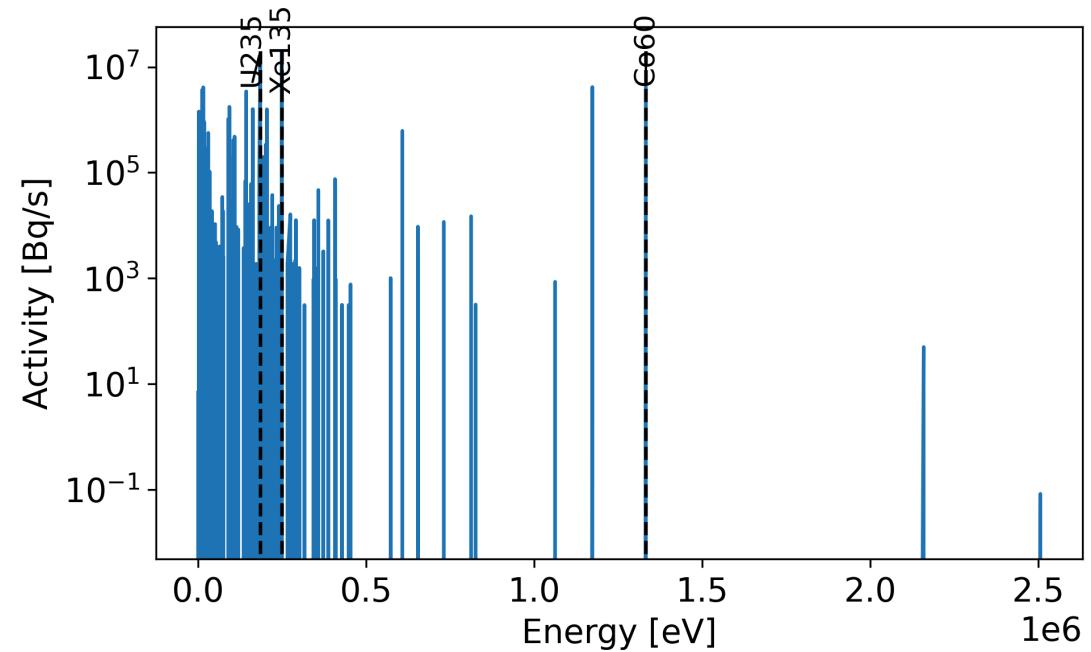
Activation products from fission

- Fission of large atoms (e.g. U235)
- Results in two fission products far from stability



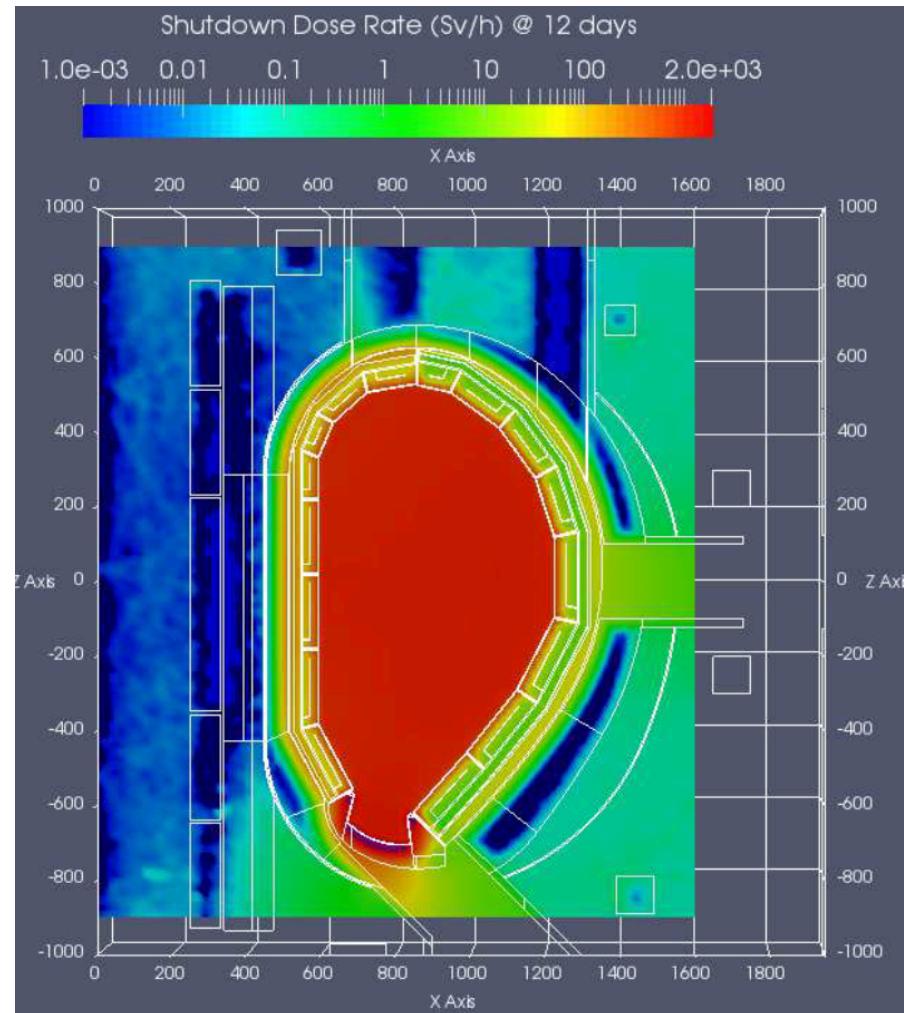
Emission during decay

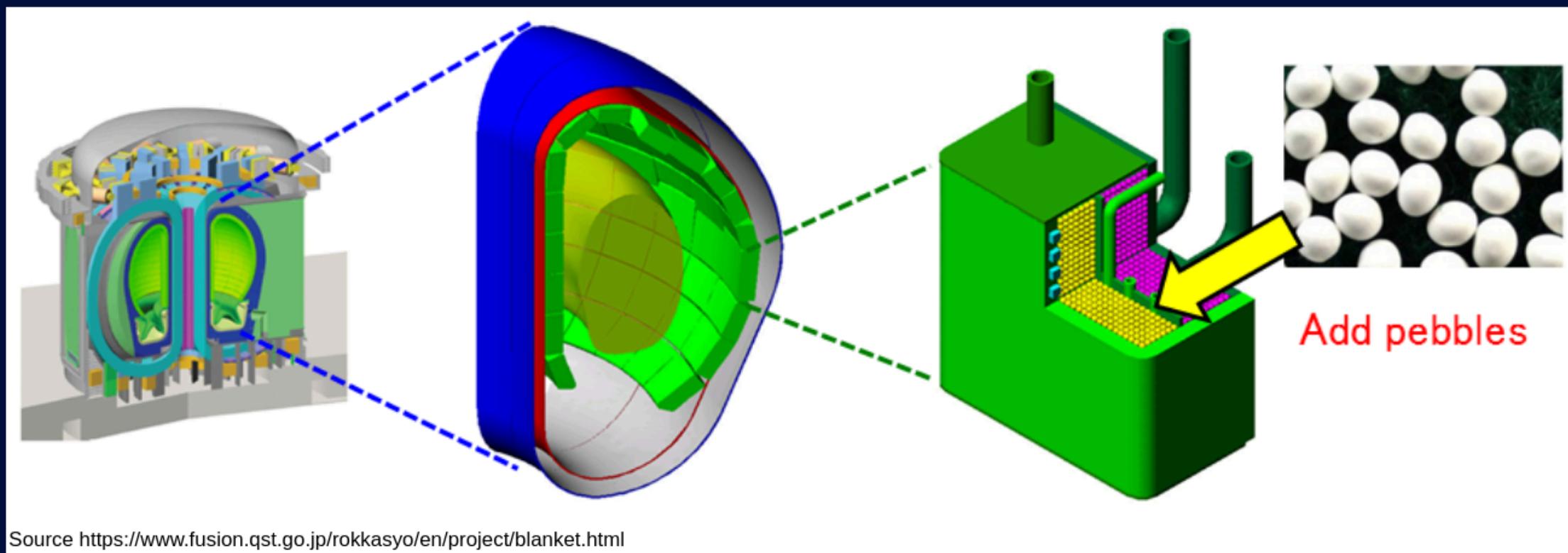
- Characteristic gamma energies and intensities emitted
- Reduces with half life of unstable isotope
- Problematic sources in fusion Co60
- Neutrons also emitted by isotopes such as N17 found which is formed by Oxygen irradiation in water



Shut down dose rate

- Post irradiation gamma and even neutron emission from radioactive isotopes continues.
- Gamma and neutrons emitted cause dose field that makes human maintenance difficult.
- This causes components to generate self heating
- Reduced strength of components due to temperature, lift carefully
- Activated coolant pumped outside of the bio-shield





Source <https://www.fusion.qst.go.jp/rakkasyo/en/project/blanket.html>

What happens to the tritium breeding ratio over the lifetime of a solid breeder blanket

It goes up a lot

It goes up a little bit

It goes down

It stays the same

Overview of neutronics simulation software

- Inventory codes
- Monte Carlo Radiation transport
- Geometry conversion software

Inventory codes (Bateman equation)

Name of software	Group / community / country
ACAB	UNED, Spain
ALARA	Wisconsin, US
Aburn	North China Electrical Power
OpenMC	MIT, ANL, community
Origen	LANL, US
Serpent	VTT, Finland
Fispact	CCFE / UKAEA
	Other codes

Radiation transport

Sampling the Boltzman transport equation

- Stochastic / Monte Carlo is most widely used method in fusion
- Track individual particle histories through phase space
- Random sampling of particle behavior at each event
- Accumulate contributions to the mean behavior from each history
- Variance reduction used to speed up simulation

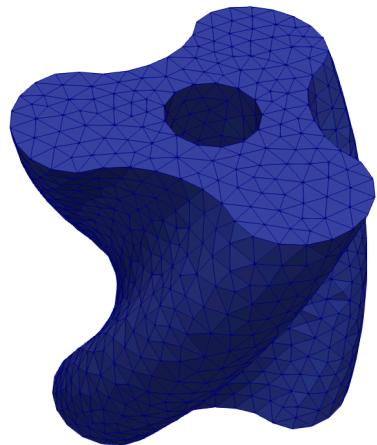
Monte Carlo Simulations

Name of software	Group / community / country
FLUKA	CERN
GEANT	CERN
MCNP	LANL
OpenMC	MIT, ANL and open source community
Serpent	VTT, Finland
TopMC	China
TRIPOLI	France
GEANT4	CERN, others

Geometry for Monte Carlo

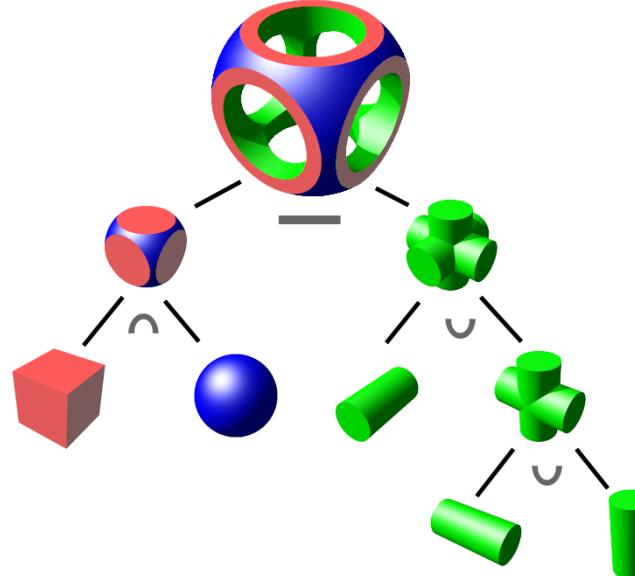
CAD to DAGMC convertors

- cad-to-dagmc
- cad-to-openmc
- stl-to-dagmc
- stellermesh
- Cubit



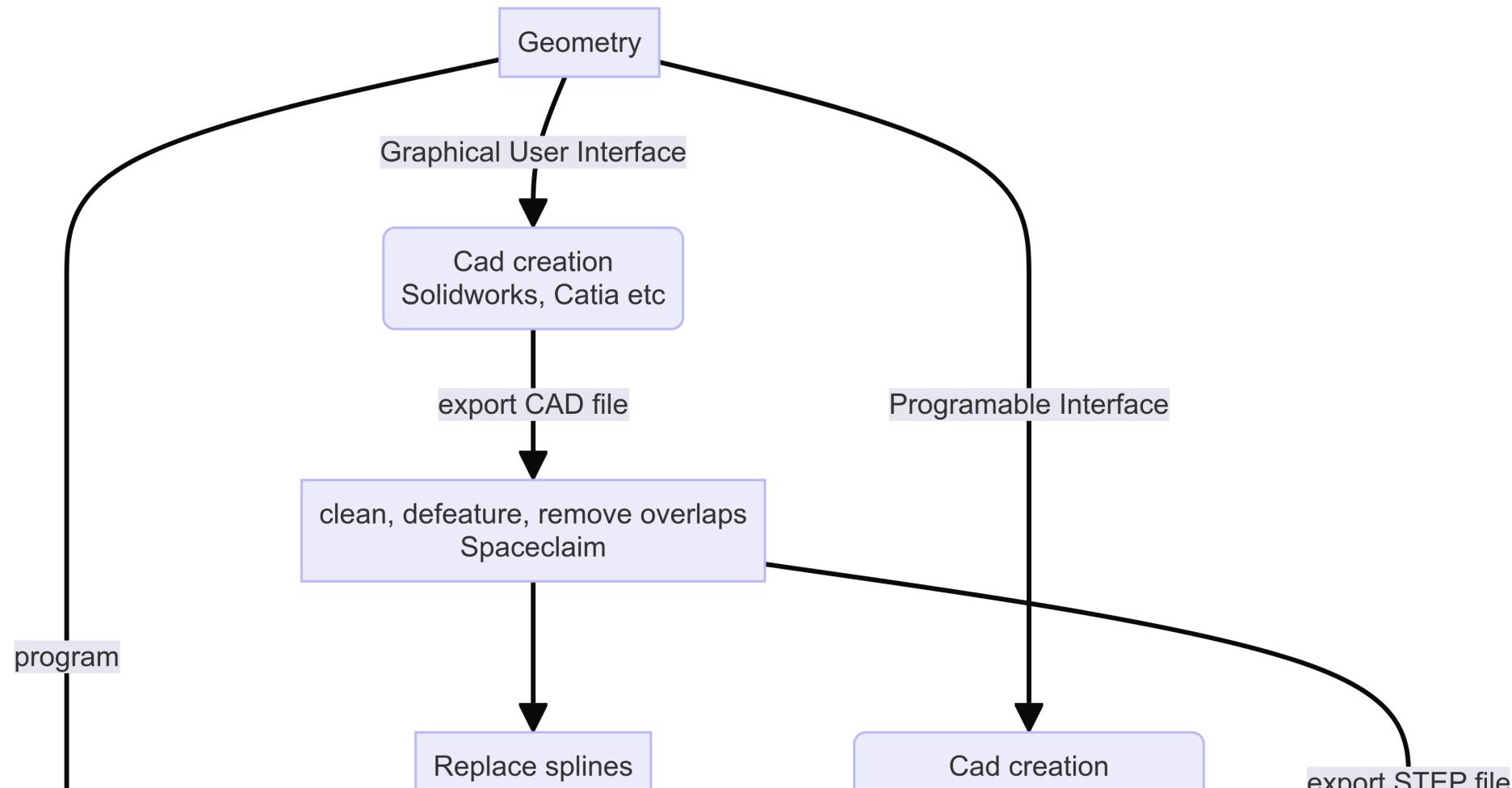
CAD to CSG convertors

- GeoUned
- McCAD
- TopMC



Geometry conversion

[Link to flowchat](#)



Software distribution

Open source codes such as OpenMC and DAGMC are distributed via GitHub, conda.

Some codes used in neutronics are controlled codes under export control

Distribution in the US by RSICC and in the EU by the NEA databank.

The screenshot shows the homepage of the Radiation Safety Information Computational Center (RSICC). The header features the RSICC logo (a globe with orbital paths) and the text "Radiation Safety Information Computational Center" with the subtitle "Delivering the Best Computational Tools for Nuclear Research". The header also includes the Oak Ridge National Laboratory logo. A navigation bar at the top has links for "Software Index" (with a dropdown menu for numbers 1-9 and letters A-Z), "HOME", "About RSICC", "Customer Service", "Registration Requests", "Software Catalog", "Contact Us", "Workshops", "Newsletters", and "Benchmarks". The main content area discusses the evolution of RSICC since 1962 and lists various software and services available, including Physics of the interaction of radiation with matter, Radiation production and sources, Criticality safety, Radiation protection and shielding, Radiation detectors and measurements, Shielding materials properties, Shields and shipping cask design, Radiation waste management, Radiological safety and assessment, Atmospheric dispersion and environmental dose, Radiation dose in medical applications, and Space shielding applications. A red banner at the bottom encourages users to request source versions of MCNP® or SCALE.

Since its inception in 1962, RSICC has evolved into one of the world's leading resources for a broad range of the best available nuclear computational tools and services. RSICC software and data collections provide in-depth coverage of radiation transport and safety topics encompassing, but not limited to:

- Physics of the interaction of radiation with matter
- Radiation production and sources
- Criticality safety
- Radiation protection and shielding
- Radiation detectors and measurements
- Shielding materials properties
- Shields and shipping cask design
- Radiation waste management
- Radiological safety and assessment
- Atmospheric dispersion and environmental dose
- Radiation dose in medical applications
- Space shielding applications

Requesting Source versions of MCNP® or SCALE

Individuals that need access to the SOURCE version of MCNP® or SCALE should submit a request for the executable version of the desired package and include a justification in either the end use statement or in the comments of their request that explains why the SOURCE version is needed. Individuals that fail to provide this information will only be provided the executable version.

Questions



mail@jshimwell.com



[@jshimwell](https://github.com/jshimwell)

