Low Energy Neutrons in FLUKA

Neutron simulations: two different "worlds"

Evaluated data files (E_{max}=20-150/200 MeV)

- Based on expert "evaluations" of available exp. data, often complemented by models
- "High" energy (> 20 MeV) evaluations based on complex (non MC) nuclear models, **(GNASH, Talys, Empire)** whose reliability becomes more and more unproven with increasing energy

Pros:

- ✓ E < 20 MeV: as good as our best knowledge
- ✓ Standard formats/processing tools available
- ✓ Little CPU (… but memory hungry)
- ✓ No real alternative below 20 MeV

Cons:

- ➢ No correlations!!
- ➢ Slow and complex to update when new data/improved models
- ➢ Sometimes incomplete or inconsistent

(MC) Models: 10-20 MeV - E_{max} up to TeV's

- MC nuclear models aimed at the description of particle production spectra by whichever projectile
- A large variety available (not necessarily all good)

Pros:

- ✓ Work for all proj/energies/targets
- ✓ They produce (at least the good ones) fully correlated physical events (eg conservation laws fulfilled event-by-event)
- ✓ Easy to update, just update the code and run again

Cons:

- ➢ As good as the physics inside, sometimes good for most apps, horrible for a few
- ➢ Not really usable below 10-20 MeV (or even higher for many)

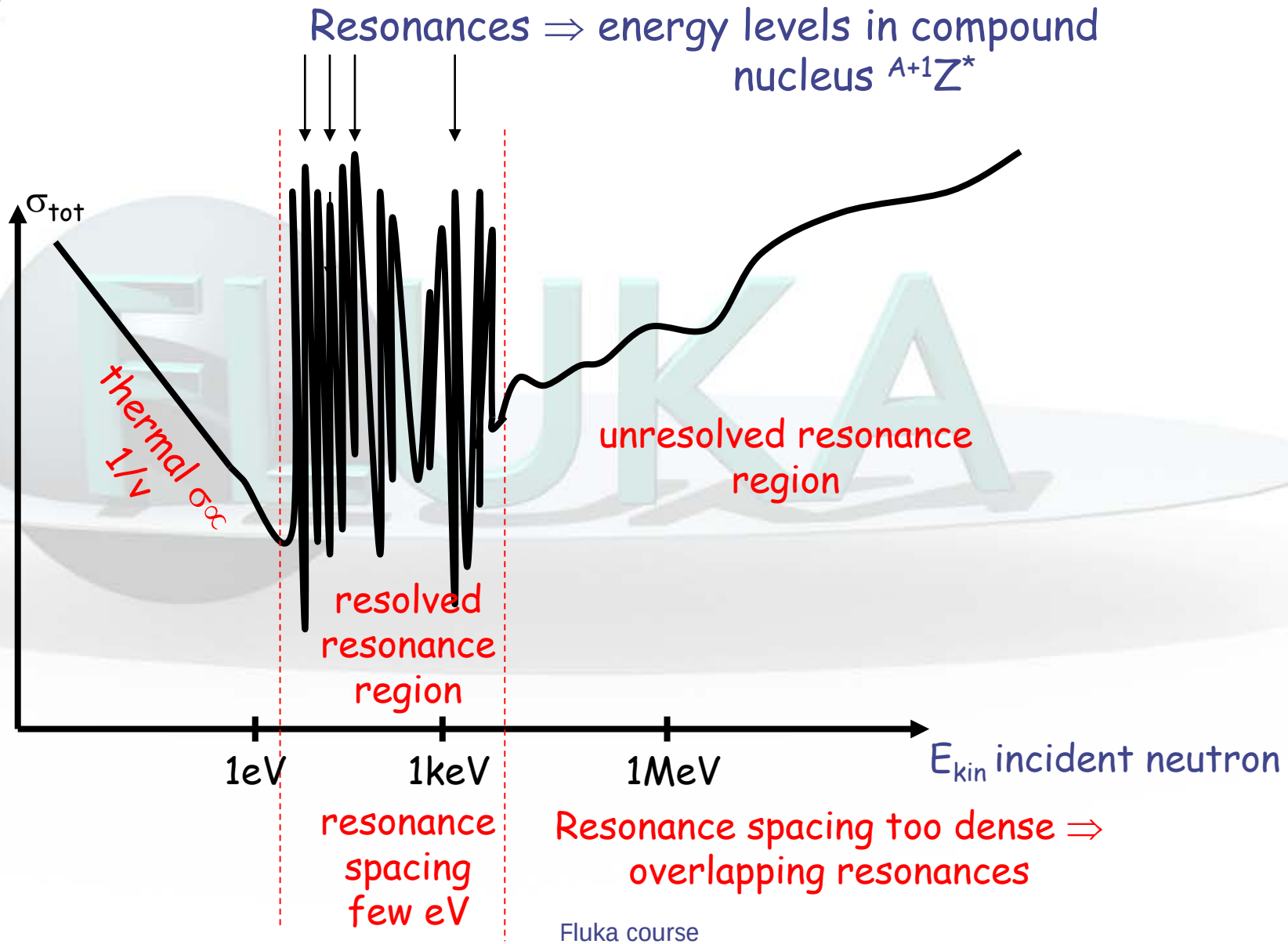
Introduction: low energy neutrons in Fluka

- In FLUKA we call neutrons below 20 MeV ***low energy neutrons***
- Neutron interactions at higher energy are handled by FLUKA nuclear models
- Transport and interactions of neutrons with energies below 20 MeV are handled by a dedicated library

Why are ***low Energy Neutrons*** special?

- The neutron has no charge and ~ ∞ lifetime ➔ can (only) undergo nuclear interactions even at very low energies, e.g. meV
- Neutron cross sections (σ) are complex and structure rich ➔ cannot be calculated by models ➔ we rely (like all codes) on evaluated data files
- Even at "thermal" energies neutrons can still generate several MeV's of γ's and/or charged particles through capture

Typical neutron cross section



Evaluated Nuclear Data Files

- Evaluated nuclear data files, used by all codes (ENDF, JEFF, JENDL…)
  - provide neutron σ (cross sections) and ***inclusive**** distributions for E<20MeV for all channels

Point-wise and Group-wise cross sections

- In neutron transport codes in general two approaches are used: point-wise ("continuous energy" cross sections) and group-wise ("energy group" averaged σ's) transport
- Point-wise follows cross section precisely but it can be (CPU) time and memory consuming
- Group approach is widely used in neutron transport codes because it is fast and gives good results for most applications

Complex programs (NJOY, PREPRO…) convert ENDF files to point-wise or group-wise cross sections, including Doppler broadening etc.

FLUKA uses group-wise cross sections with some exceptions

** As a consequence no event-by-event closed kinematics is possible*

Group Transport Technique

- The energy range of interest is divided in a given number of discrete intervals ("energy groups"). Each group is identified by a number increasing with decreasing energy. Group 1 is the group of highest energy (in the present FLUKA library, energies between 19.64 and 20 MeV)
- Elastic and nonelastic reactions are not simulated as exclusive processes, but by group-to-group transfer probabilities (downscattering matrix)
- Downscattering matrix: if a neutron in a given group undergoes a scattering event and loses energy, it will be transferred to a group of lower energy (each of the lower energy groups having a different probability)
- If the neutron does not lose enough energy to be in another group, it will stay in the same group (in-scattering)
- In the thermal region neutrons can gain energy. This is taken into account by an upscattering matrix, containing the transfer probability to a group of higher energy

Group wise treatment

- Convert σ to energy groups like histograms, the energy width is different for each group in order to better represent resonances
- Each group i contains the <average> σ_i:

$$< \sigma_i >= \frac{\int_{E_{i,low}}^{E_{i,high}} \sigma(E)\Phi(E)\,dE}{\int_{E_{i,low}}^{E_{i,high}} \Phi(E)\,dE}$$

where Φ is the a typical fluence shape (eg 1/E + Maxwellian)

- Advantage: fast
- Disadvantage: effects like self shielding require specific treatment (see later), angular distributions replaced by a discrete number of polar angles (3 in FLUKA for each group-to-group transfer) conserving their first n moments (6 in FLUKA)

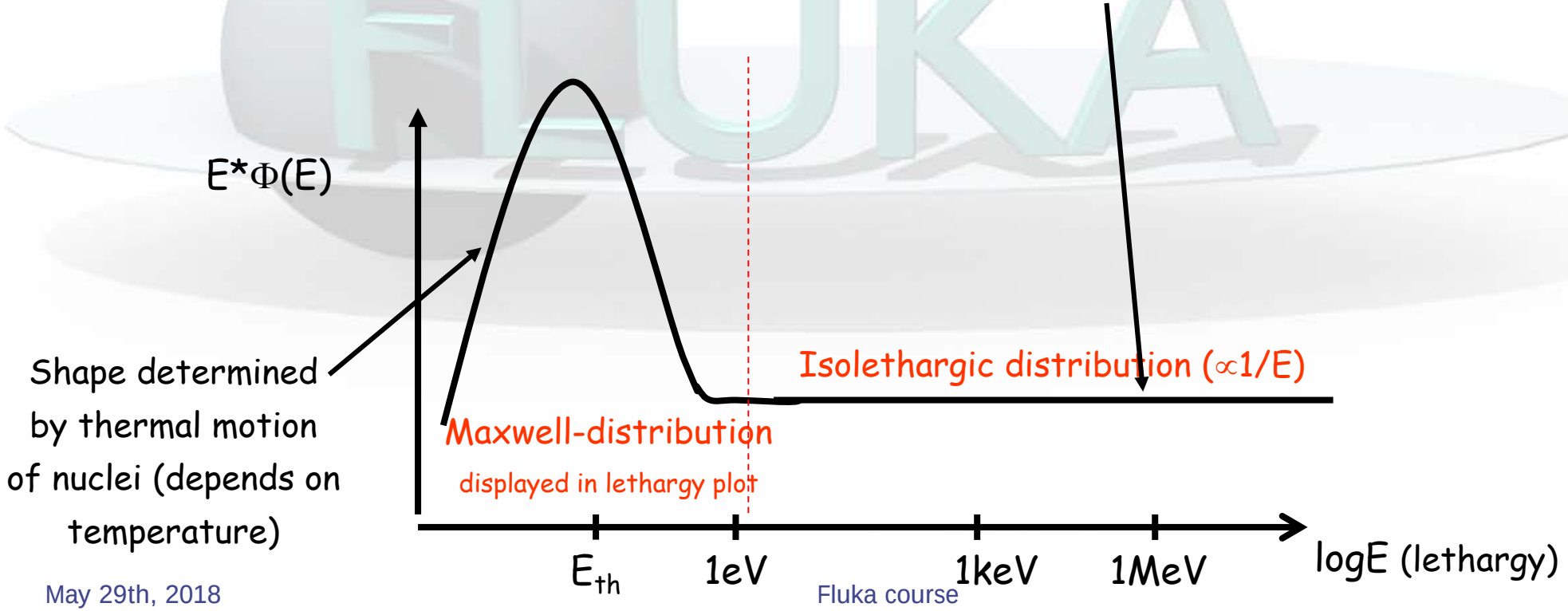
Fluence

- An assumption is needed about the neutron spectrum to be used as a weighting function for calculating the average cross section on each group
- For instance it can be shown that in most cases, between 1 eV and 1 MeV:

$$\Phi(E)dE = C\frac{dE}{E}$$

$\frac{1}{E}$ (spectrum)

- In any case, the error is small if the group width is small



Angular distribution

- The probability distribution of the scattering angle for each group-to-group transfer is represented by a Legendre polynomial expansion truncated at the $(N+1)^{th}$ term (N=5 in FLUKA)

$$\sigma_s(g \rightarrow g', \mu) = \sum_{i=0}^{N} \frac{2i+1}{4\pi} \mathrm{P}_i(\mu)\, \sigma_s^i(g \rightarrow g')$$

$\mu = \vec{\Omega} \bullet \vec{\Omega}' =$ cosine of the polar scattering angle

- This approach preserves the first 6 moments for each g→g′ transfer probability
- The result of this P5 expansion is a set of 3 discrete polar angle cosines and 3 corresponding probabilities, i.e. for a given transfer g→g′ only three values are possible for the polar angle

Self shielding

- The group structure is necessarily coarse with respect to the resonance structure in many materials
- A resonance in a material present in a dilute mixture or as a small piece cannot affect much a smooth neutron flux (so-called "infinite dilution'')
- But if an isotope exhibiting large resonances is very pure ("no dilution'') or is present with a large fractional abundance, it can act as a "neutron sink", causing sharp dips in the neutron spectrum corresponding to each resonance ➔ an apparent decrease in σ
- This effect, which results in a lower reaction rate, is called *self-shielding* and it must be accounted in the process of cross section averaging $\sigma\Phi$ over the width of each energy group, using a special correction which depends on the material dilution. FLUKA group cross sections are typically provided for "infinite dilution'' and "no dilution'' with a few exceptions

The FLUKA Low Energy Neutron Library [1/2]

- FLUKA uses the multigroup transport technique
- The energy boundary below which multigroup transport takes over depends in principle on the cross section library used. In the present library it is 20 MeV.
- Both fully biased and semi-analog approaches are available
- Number of groups: 260 of approximately equal logarithmic width, the actual energy limits of each group can be found in the manual (or can be printed to *.out file)
- N.B. the group with the highest energy has the number 1, the group with the lowest energy has number 260
- 31 thermal groups, with 30 upscattering groups
- Energy range of library: 0.01 meV - 20 MeV

The FLUKA Low Energy Neutron library [2/2]

- Based on recent versions of evaluated nuclear data files: ENDF/B-VI.8/VII.0/VII.1, Jendl-3.3/3.4/4.0, Jeff-3.1/3.2,…
- About 270 isotopes/materials available
- Almost all materials available at 2 temperatures: 87K, 296K
- Some also at 4K, 120K, 430K
- Doppler broadening at the relevant temperature is taken into account
- The library handles also gamma generation, energy deposition by kerma factors, residual nuclei production, secondary neutrons, fission neutrons, DPA and NIEL
- For some isotopes/materials: self-shielding, molecular binding, correlated gamma generation, point-wise transport
- NB: Because of the group technique the energy of a transported neutron below 20 MeV is only defined within the resolution of the groups

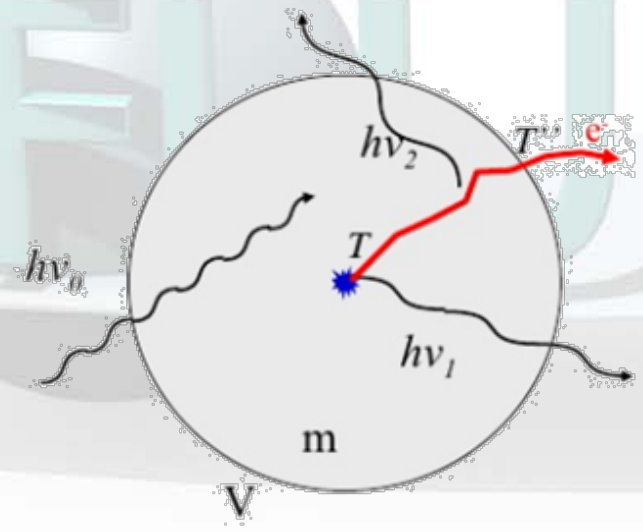
Gamma generation

- Gamma generation from (n,xγ) reactions is possible only for those elements for which data are available in the evaluated nuclear data files (see manual for a complete list)
- Gamma generation is performed by a multigroup scheme as well
- Number of gamma groups: 42
  - NB number of γ groups different from number of neutron groups!
- Energy range: 1keV - 50 MeV
  - NB γ energy range different from neutron energy range!
- The actual energy of the generated photon is sampled randomly in the energy interval corresponding to its gamma group
- Exceptions: 2.2 MeV from $H(n,\gamma)^2H$ reaction, 478 keV photon from $^{10}B(n, \alpha\gamma)$ and gamma cascades from Cd(n,γ) and Xe(n,γ)
- Capture gammas as well as gammas from inelastic reactions like (n,n') are included
- The neutron library only creates gammas, the transport is done by the EMF module (like all other gammas in FLUKA)

Kerma vs Absorbed dose

Kerma ("Kinetic Energy Released per unit MAss") is the amount of energy transferred from indirectly ionizing radiation (e.g. photons and neutrons) to directly ionizing radiation (e.g. electrons, atom recoils) in a medium, per unit mass.

The **absorbed dose** is the amount energy imparted by ionizing radiation in a medium, per unit mass.



$K = T / m = (h\nu_0 - h\nu_1) / m$

$D = (T - T' - h\nu_2) / m$

T = initial kinetic energy of the electron

T' = kinetic energy of the electron when crossing the boundary of volume V

Energy deposition for low energy neutrons:

- Energy deposition by neutrons below 20 MeV is estimated by means of kerma factors (in this case normalized as the average energy imparted to charged particles per interaction)
- As a consequence *no event-by-event energy deposition scoring is possible with low energy neutrons*, or, equivalently, *the energy deposited locally by a neutron of a given energy interacting in a given material is always the same* (and equal to the actual average one)*. The number and energy of outgoing neutrons and photons can instead vary from interaction to interaction
- For some materials with gamma production the kerma values of some groups (mainly at high energies) can be problematic (see manual). The reason is inconsistent data in the evaluated data files

** With the important exception of hydrogen, see later*

Secondary and Fission Neutrons

- Neutrons from (n,xn) reactions are taken into account implicitly by a group-dependent non-absorption probability (that is the average multiplicity of the outgoing neutrons)
- If the only possible processes are scattering and capture the non-absorption probability is smaller than 1. If also (n,2n), (n,3n), … are possible, the factor can be larger than 1 and more than neutron can be emitted. However there is no 1-to-1 correlation with the actual probability for (n,n), (n,), (n,2n), (n,3n) … due to the inclusive nature of the evaluated nuclear data files
- Fission neutrons are treated separately by a group-dependent fission probability
- Fission neutrons are emitted isotropically with an energy sampled from a fission spectrum appropriate for the target isotope and incoming neutron energy
- The fission neutron multiplicity was obtained separately from the evaluated data files
- The fission fragments are not transported, their energy is deposited at the spot by means of the kerma factors

Charged particle generation

- Recoil protons from hydrogen and protons from $^{14}N(n,p)$ are produced and transported explicitly (i.e. like other protons)
- That means that detailed kinematics of elastic scattering, continuous energy loss with energy straggling, delta ray production, multiple and single scattering, are all taken into account for those protons (allowing event-by-event scoring for neutrons on hydrogen)
- If point-wise transport has been requested, ^{3}H and α fragments from neutron capture in ^{6}Li and ^{10}B can also be transported explicitly
- Note that for this purpose you cannot use the cross sections of natural Li and B, but you must define a compound of ^{6}Li and ^{7}Li, or one of ^{10}B and ^{11}B
- All other charged secondaries produced in low energy neutron reactions are not transported but their energy is deposited at the point of interaction using kerma factors

Residual nuclei production

- Residual nuclei: nuclei that are the result of a reaction and are at rest, e.g. ^{28}Al after a neutron capture reaction of ^{27}Al
- For all materials data are available for estimating residual nuclei production by low energy neutrons. Command RESNUCLEi allows to request separately residual nuclei from low energy neutrons and from high energy particles
- For Ti, Ga the residual nuclei information is based on different evaluations than the transport

Using the Low Energy Neutron Library

Available Materials

- Section 10.4.1.2 of manual gives a list of available materials
- Example:

Material			Temp.	Origin	RN	Name	Identifiers			Gam
9Be		Beryllium 9	296K	ENDF/B-VIIR0	Y	BERYLLIU	4	9	296	Y
9Be		Beryllium 9	87K	ENDF/B-VIIR0	Y	BERYLLIU	4	9	87	Y
B		Natural Boron	296K	ENDF/B-VIIR0	Y	BORON	5	-2	296	Y
B		Natural Boron	87K	ENDF/B-VIIR0	Y	BORON	5	-2	87	Y
10B		Boron 10	296K	ENDF/B-VIIR0	Y	BORON-10	5	10	296	Y
10B		Boron 10	87K	ENDF/B-VIIR0	Y	BORON-10	5	10	87	Y
11B		Boron 11	296K	ENDF/B-VIIR0	Y	BORON-11	5	11	296	Y
11B		Boron 11	87K	ENDF/B-VIIR0	Y	BORON-11	5	11	87	Y
C		Free gas natural Carbon	296K	ENDF/B-VIIR0	Y	CARBON	6	-2	296	Y
C		Graphite bound nat. Carbon	296K	ENDF/B-VIIR0	Y	CARBON	6	-3	296	Y
C		Free gas natural Carbon	87K	ENDF/B-VIIR0	Y	CARBON	6	-2	87	Y
N		Natural Nitrogen	296K	ENDF/B-VIIR0	Y	NITROGEN	7	-2	296	Y
N		Natural Nitrogen	87K	ENDF/B-VIIR0	Y	NITROGEN	7	-2	87	Y
14N		Nitrogen 14	296K	ENDF/B-VIIR0	Y	NITRO-14	7	14	296	Y
14N		Nitrogen 14	87K	ENDF/B-VIIR0	Y	NITRO-14	7	14	87	Y
16O		Oxygen 16	296K	ENDF/B-VIR8	Y	OXYGEN	8	16	296	Y
16O		Oxygen 16	87K	ENDF/B-VIR8	Y	OXYGEN	8	16	87	Y
19F		Fluorine 19	296K	ENDF/B-VIR8	Y	FLUORINE	9	19	296	Y
19F		Fluorine 19	87K	ENDF/B-VIR8	Y	FLUORINE	9	19	87	Y
23Na		Sodium 23	296K	JENDL-3.3	Y	SODIUM	11	23	296	Y
23Na		Sodium 23	87K	JENDL-3.3	Y	SODIUM	11	23	87	Y
Mg		Natural Magnesium	296K	JENDL-3.3	Y	MAGNESIU	12	-2	296	Y
Mg		Natural Magnesium	87K	JENDL-3.3	Y	MAGNESIU	12	-2	87	Y
27Al		Aluminium 27	296K	ENDF/B-VIIR0	Y	ALUMINUM	13	27	296	Y
27Al		Aluminium 27 SelfShielded	296K	ENDF/B-VIIR0	Y	ALUMINUM	13	1027	296	Y

RN: residual nuclei

Gam: Gamma production

Name: name to be used in LOW-MAT card

Identifiers: to be used in LOW-MAT card

Using the Low Energy Neutron Library

How to activate low energy neutron transport?

- Implicit: giving a DEFAULTS card (except with default EM-CASCA), or not giving a DEFAULTS card at all

  That means: you are using the library in almost any simulation (unless you are using the default EM-CASCA or you have switched it off explicitly with a LOW-BIAS or PART-THR card)

- Explicit: giving a LOW-NEUT card (NB you don't need it in most practical cases!)

What must the user do?

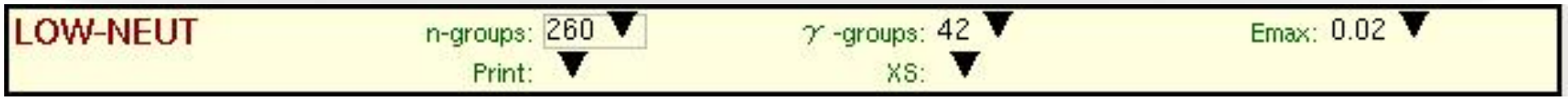
- To set correspondence between the actual material and the material in the low neutron library (LOW-MAT card), if not done by default (NB you don't need it in most practical cases!)
- To set transport thresholds with PART-THR, if defaults are not ideal for the actual problem (NB keep the 10^{-5} eV threshold for most problems!)
- To request special features like point wise cross sections or cross section printing (LOW-NEUT)

Input Cards: LOW-NEUT [1/2]

This card activates low-energy neutron transport.

- WHAT(1): number of neutron groups of the library
  - Default: 260
- WHAT(2): number of gamma groups of the library
  - Default: 42
- WHAT(3): maximum energy (in GeV) of the library
  - Default: 0.02
- WHAT(4): printout flag (see the manual)
- WHAT(5): number of thermal groups
  - Default: 31

```
LOW-NEUT   260.    42.    0.2    0.    31.     1.
```



- The defaults for WHAT(1) – WHAT(3) and WHAT(5) are fine. The only reason for changing them would be if using a different library (there was one until recently, but it has been suppressed)

➔ Don't use them!

Input Cards: LOW-NEUT [2/2]

- WHAT(6): $i_0 + 10 * i_1$:
  - i_0 = 1: available point wise cross sections used and explicit and correlated $^{6}Li(n,\gamma)^{7}Li$, $^{6}Li(n,t)^{4}He$, $^{10}B(n,\alpha\gamma)^{7}Li$, $^{40}Ar(n,\gamma)^{41}Ar$, $^{x}Xe(n,\gamma)^{x+1}Xe$ and $^{113}Cd(n,\gamma)^{114}Cd$ photon cascade requested
    - = 0: ignored
    - =< -1: resets to the default (point wise cross sections are not used)
  - i_1 = 1, fission neutron multiplicity forced to 1, with proper statistical weight to compensate for the actual multiplicity (it helps in avoiding to "explod" ~critical assemblies)
    - = 0, ignored
    - =< -1: resets to the default (normal fission multiplicity)
  - Default = -11., unless option DEFAULTS is present with **SDUM** = CALORIME, ICARUS, NEUTRONS or PRECISIO, in which case the default is 1.0 (point wise cross sections are used when available and fission multiplicity is not forced)

Input Cards: LOW-MAT [1/3]

- The LOW-MAT card sets the correspondence between FLUKA materials and the low energy neutron cross sections
- If a material has the same name as a name given in the list of low neutron materials, the correspondence between material and low energy neutron transport is set automatically, and a LOW-MAT card is not necessary. The first material with the right name is taken. This is always a material at room temperature.
- That means that for the predefined material HYDROGEN hydrogen bound in water is used, not the free gas one
- If you want to use low energy neutron transport in H_2 gas you have to do this explicitly by a LOW-MAT card

Input Cards: LOW-MAT [2/3]

- WHAT(1): Name of the material (single element/isotope only!)
  - In flair this can be chosen from a pull down menu
- WHAT(2), WHAT(3) and WHAT(4): the 3 identifiers from table 10.4.1.2 of the manual
- SDUM: name of the material from table 10.4.1.2 of the manual
- In flair there is only one pull down menu for all identifiers and the name
- If you want to use the predefined materials at a temperature different from 296K, it is mandatory to give a LOW-MAT card with the proper identifiers

Input Cards: LOW-MAT [3/3]

- Setting the correspondence between a material and low energy neutron transport cross sections:
  - First create the material with a MATERIAL card and give it a name in SDUM
  - Give a LOW-MAT card with WHAT(1) = the name you gave in the SDUM of the MATERIAL
  - Give in WHAT(2), WHAT(3) and WHAT(4) of the LOW-MAT card the numerical identifiers (table 10.4.1.2 in manual) of the material you want to use, be careful to use the one with the right temperature
  - Give in SDUM of the LOW-MAT card the name provided in the same table

Examples: Bonner sphere(s) with ^{3}He detector

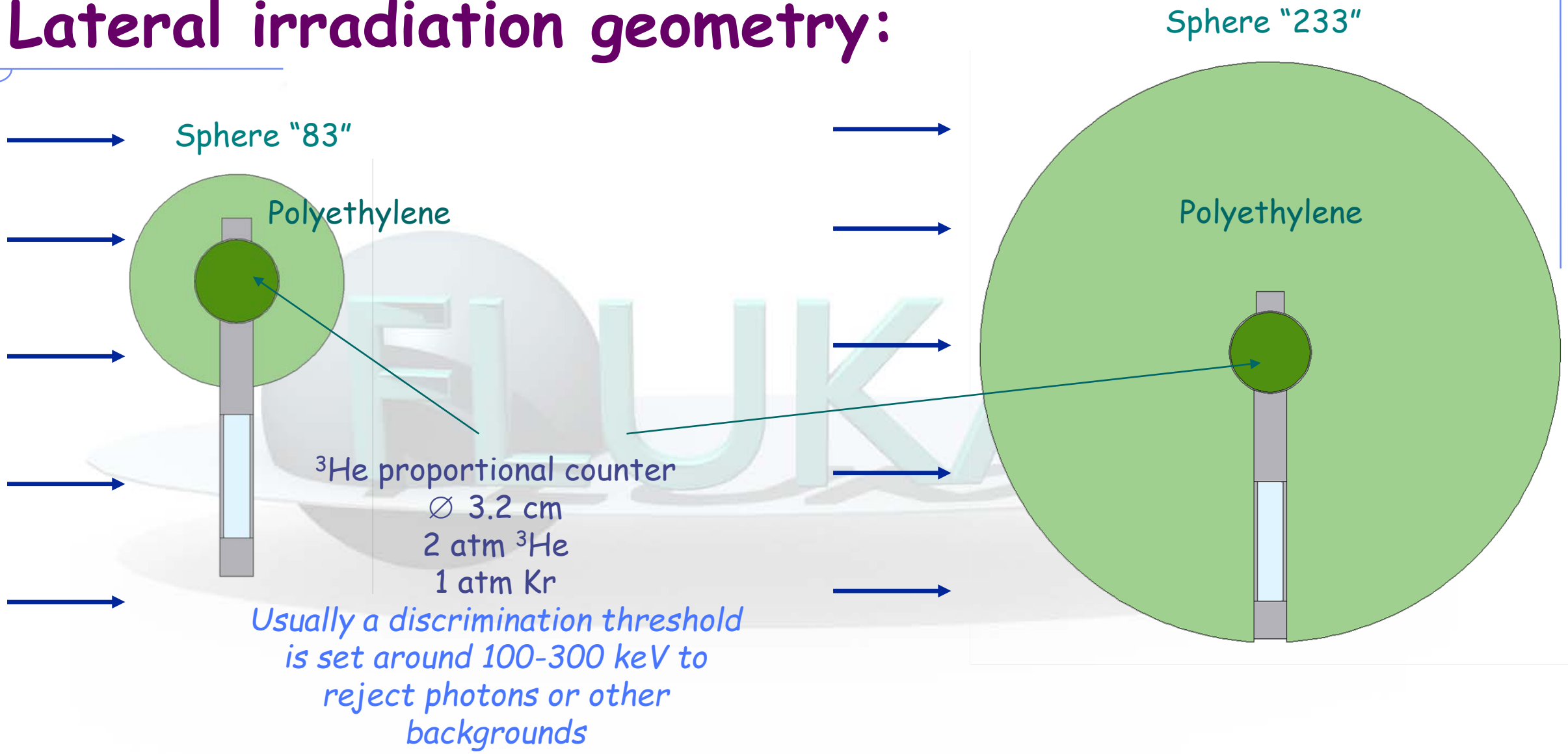
- Bonner sphere with ^{3}He detector at the centre ← Geometry description
- Assumptions:
  - Most of the counts will be due to low energy (~thermal) neutrons polyethylene
  - The vast majority of the counts will be due to the (n,p) reaction
- Compute response functions by:
  a) Folding with the cross section for ^{3}He(n,p)^{3}H only (and only for E < 20 MeV)
  b) As above but also adding the contribution of (n,el) and (n,d) when the resulting charged recoils are over a threshold set at 100, 200 and 500 keV

Track-length estimator in the gas volume, folding with ^{3}He σ's

- … assuming for irradiation geometries:
  - A parallel neutron beam impinging on the sphere side
  - (An uniform and isotropic neutron fluence surrounding the spheres, not today!)

Source term

Lateral irradiation geometry:



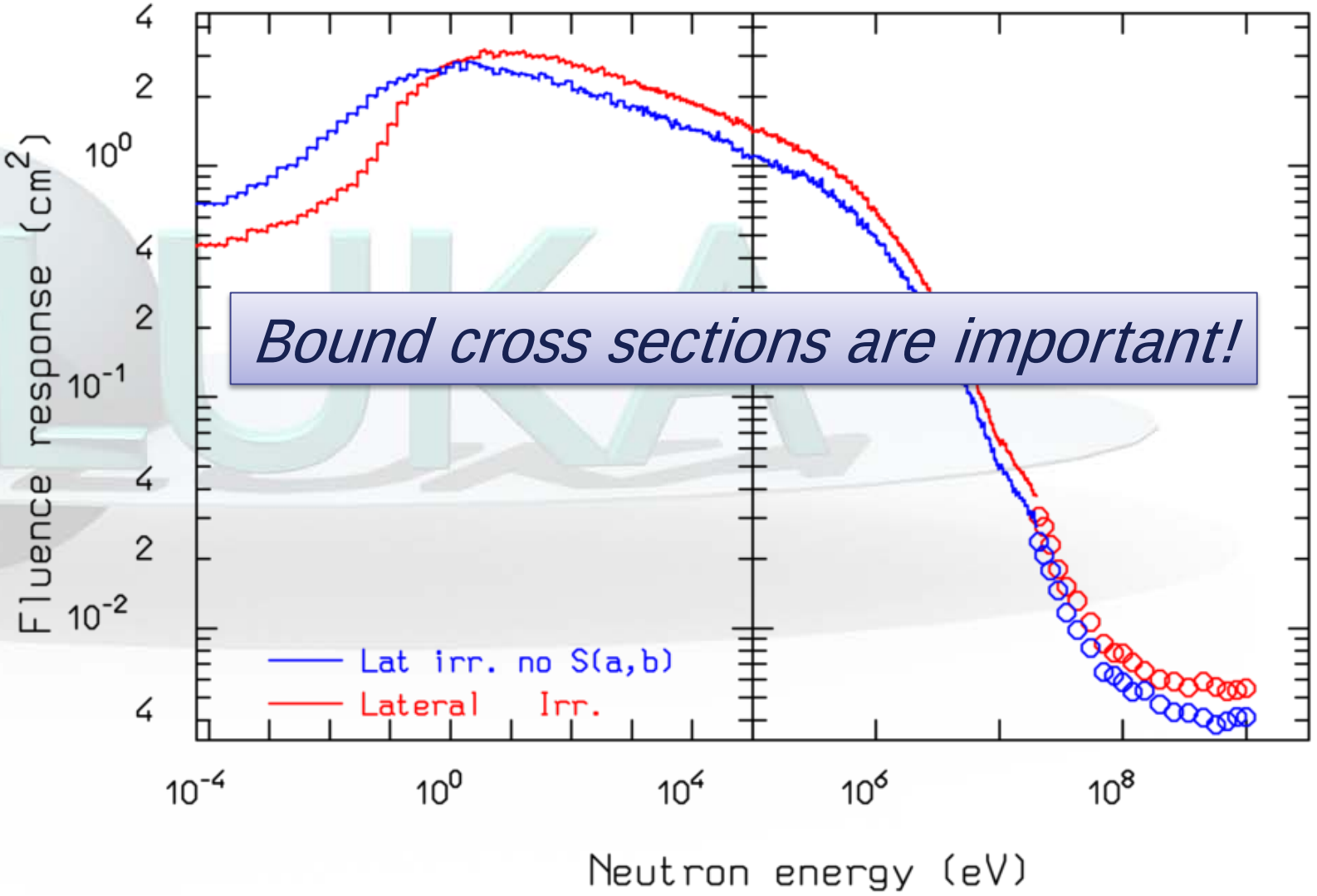
Usually a discrimination threshold is set around 100-300 keV to reject photons or other backgrounds

Materials with molecular binding in the Fluka library

- Available materials with molecular bindings at 296K:
  - H (natural isotopic amount) in H_2O, CH_2
  - ^1H in H_2O, CH_2
  - ^2D in D_2O
  - C in graphite
- Use of these materials makes the thermal neutron calculation more realistic and can affect the energy and spatial distributions
- Example: CH_2 (polyethylene) including molecular binding
  - Create a material hydrogen and give a corresponding LOW-MAT card that refers to H bound in CH_2
  - Give a COMPOUND cart that creates CH_2 as a compound of bound H and normal carbon
- For hydrogen, H bound in water is the default, because it is the first in the list of low energy neutron materials

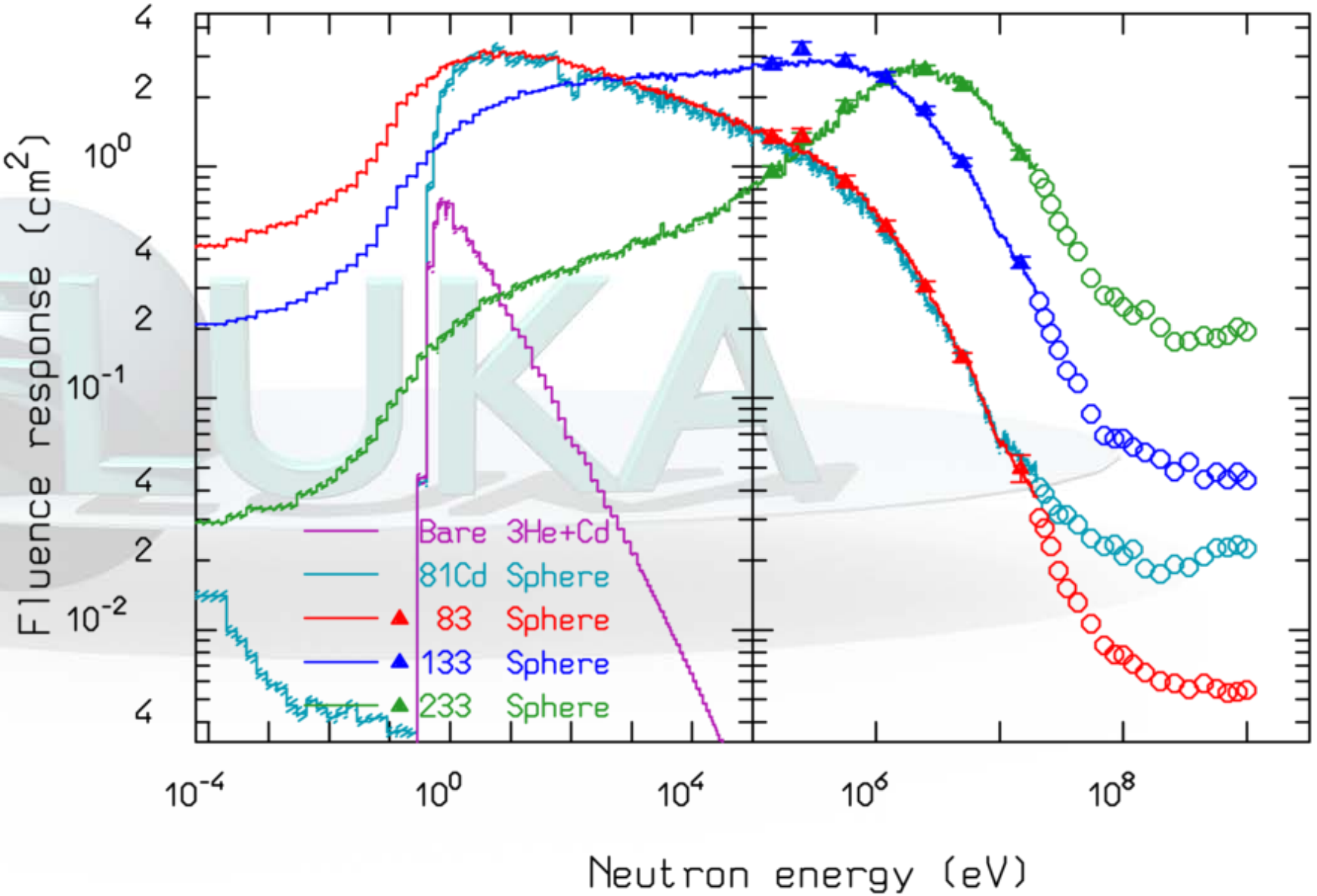
S(α,β) vs free gas for H: Sphere 83

S(α,β) is the acronym for the treatment of molecular binding for neutrons in the thermal region. In this case H bound in CH_2 (available in the FLUKA library) has been used for the red curve and for all curves in the next slide



Response functions:

Computed response functions by folding (option b)), using 200 keV as threshold for (n,el), (n,d). The symbols are exp. data obtained with mono-energetic neutron beams at PTB

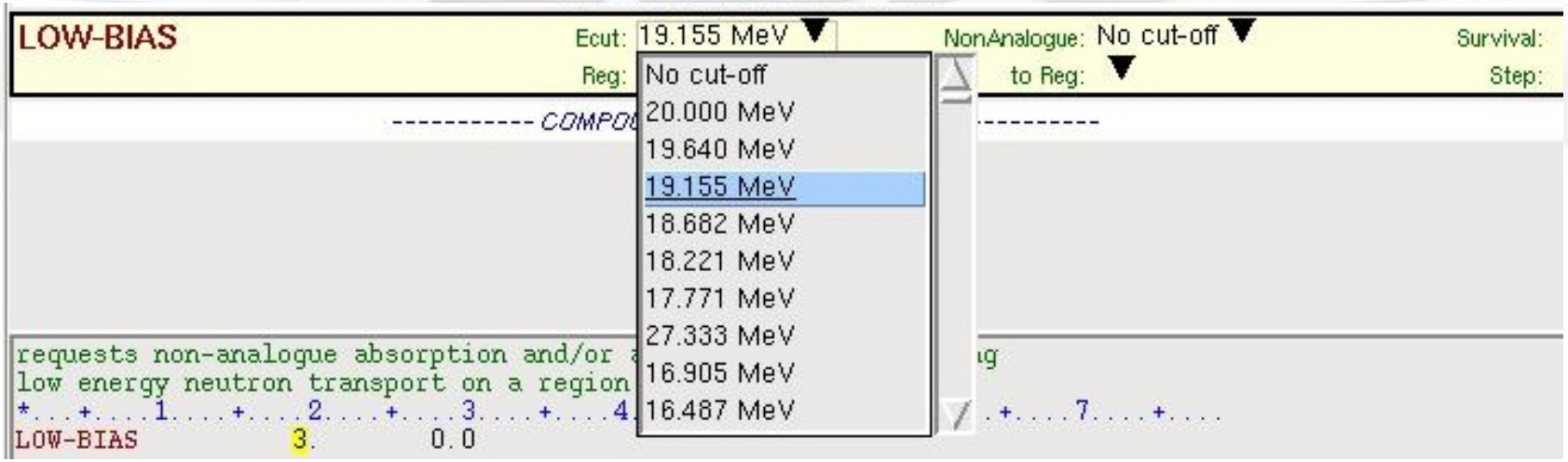


Input Cards: LOW-BIAS

This card sets an energy cut-off during low-energy neutron transport on a region by region basis and/or non-analog absorption.

However, for an overall energy cut-off, it is preferable to use PART-THR

- WHAT(1): number of the group to apply a transport cut-off, i.e. neutrons in groups the number of which >= WHAT(1) are not transported. Remember that the group with the highest energy has the number 1.
  - Default: 0.0 (no cut-off)
  - *flair* automatically matches the group number to the upper energy boundary of each group



Transport cut-offs

- Transport cut off: a particle is not transported if its energy is lower than a cut off energy
- Transport cut offs for neutrons can be set to save CPU time
- Use cut offs with care, you could miss important effects like activation, dose, secondary particles,…
- For activation, thermal neutrons are very important. If you are interested in activation never cut off low energy neutrons!
- To set a general transport cut off for neutrons, please give the energy of the cut off in the PART-THR card, no matter if high or low energy neutrons. That was different in previous versions: the card LOW-BIAS was needed for low energy neutrons. LOW-BIAS is still needed if one wants set a cut-off region by region
- If there is no interest in low-energy neutron transport, but that feature is implicit in the DEFAULTS option chosen, it is suggested to request LOW-NEUT, and to use PART-THRes with an energy cutoff WHAT(1) = 0.020

Self-shielding [1/3]

- Self-shielded materials in FLUKA:
  - ^{27}Al at 296K, 87K, 4K, 430K
  - ^{nat}A, ^{40}Ar at 296K, 87K
  - ^{nat}Fe at 296K, 87K, 4K, 430K
  - ^{nat}Cu at 296K, 87K, 4K, 430K
  - ^{181}Ta at 296K, 87K
  - ^{nat}W at 296K, 87K, 4K, 430K
  - ^{197}Au at 296K, 87K
  - ^{nat}Pb at 296K, 87K
  - ^{208}Pb at 296K
  - ^{209}Bi at 296K, 87K
    - Special case: cast iron (^{nat}Fe +5%C) at 296K, 87K, 4K, 430K (see example further on)

Self-shielding [2/3]

- When to use these materials?
  - Bulky (huge) pieces that are very pure (containing only one isotope)
- When not to use self-shielded materials?
  - "small" iron, copper, lead, aluminum pieces
  - Thin gold foils (but a self-shielded 100μm Au foil is available)
  - Diluted materials
- How to use self-shielded materials?
  - Define your material with a MATERIAL card
  - Give additionally a LOW-MAT card and give the proper identifiers in WHAT(2)-WHAT(4) and SDUM
  - If you have to use self-shielded and non self-shielded materials of the same element you need to define 2 different materials
  - Attention: predefined materials like iron, copper and lead are not self-shielded, you have to give a LOW-MAT card to use them self- shielded

Self-shielding [3/3]

- Cast iron is iron with a significant amount of carbon
- There is a self-shielded material cast iron in the low energy neutron library which is prepared to be used for creating a compound of iron and roughly 5% carbon. The amount of carbon doesn't need to be exactly 5%.
- How to create self-shielded cast iron?
  - Define a material iron called FeCarbSS (or any other name you like) with a MATERIAL card (parameters as for natural iron)
  - Insert a LOW-MAT card for FeCarbSS with the proper identifiers for cast iron in WHAT(2)-WHAT(4) and SDUM
  - Insert a MATERIAL card to declare a compound material called CastFe (or any other name you like)
  - Insert a COMPOUND card for defining CastFe as a compound of FeCarbSS and CARBON (predefined)

Artifacts of discrete angular distribution

- Artifacts can arise when a neutron is likely to scatter only once (thin foil, regions of low density like gases), due to the discrete angular distribution (only 3 angles are possible for each $g \rightarrow g'$)
- The user should be aware of such artifacts and interpret results of scattering at thin foils and gases carefully
- Because the 3 angles are different for each $g \rightarrow g'$ and the azimuthal angle is sampled from a continuous distribution, the artifact disappears when the neutrons have the possibility of scattering two or more times.
- Information about which angles and probabilities are used for each group can be obtained by setting WHAT(4) = 3 in the LOW-NEUT card. The information is then written to the *.out file (see manual chapter 9: Output)

Summary

- Most of the simulations in FLUKA use the low energy neutrons, implicitly via the DEFAULTS card
- Low neutron transport in a material is enabled by a LOW-MAT card, only needed if the material's name is not one of those in the neutron cross section library, or temperature, self-shielding or molecular binding are different from the default ones
- Use self-shielded materials properly for "bulky" and "pure" (in isotopic composition) materials
- Don't give a LOW-MAT card for compounds