



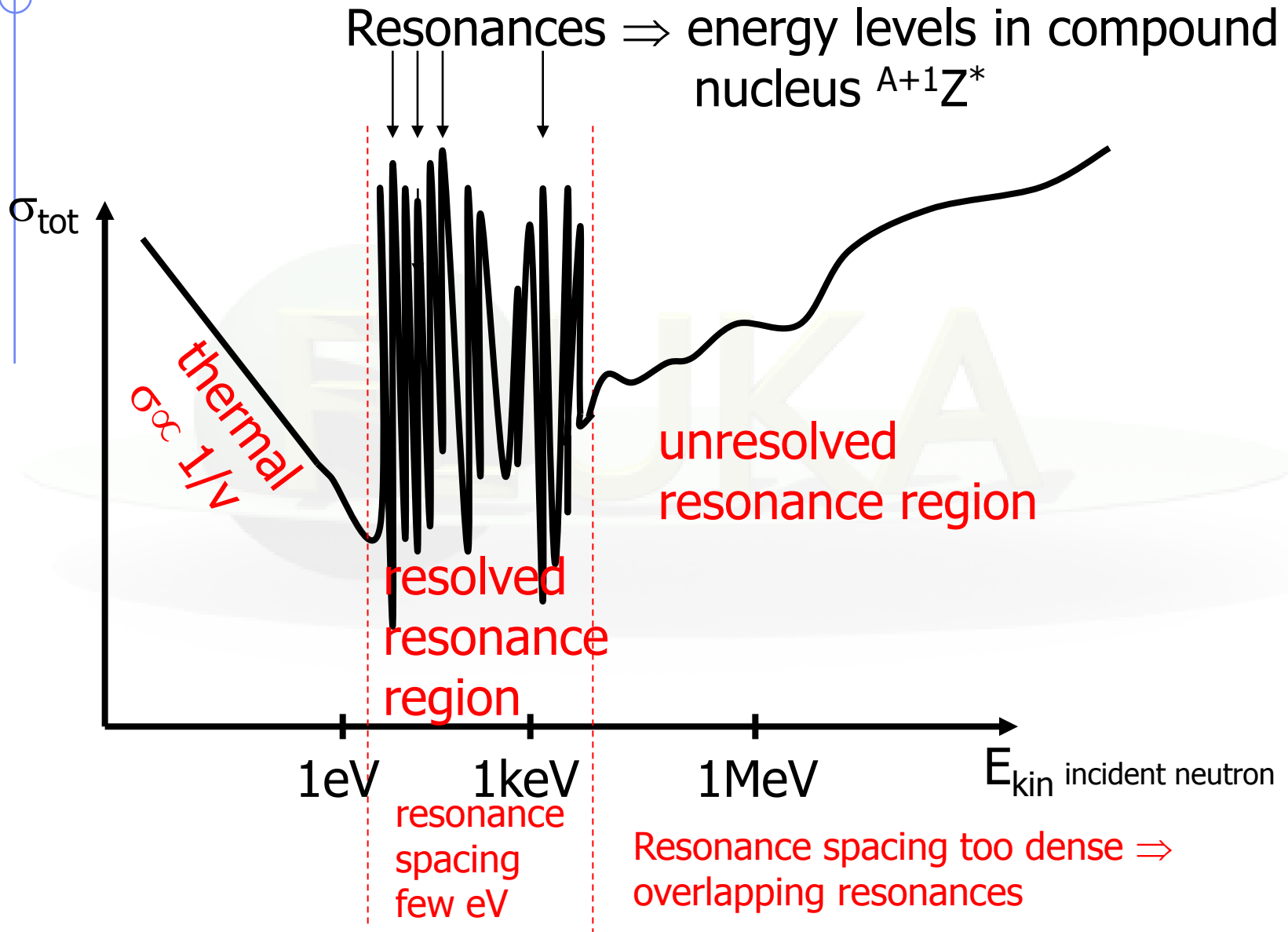
Low-Energy Neutron Treatment in FLUKA

Beginners' FLUKA Course

Why low Energy Neutrons are special

- The neutron has no charge \Rightarrow can perform nuclear-interactions at low energies e.g. meV
- **Neutron cross sections (σ) are complicated \Rightarrow cannot be calculated by models \Rightarrow we rely on data files**
- Data files
 - typically having n- σ for $E < 20\text{MeV}$ for all channels
 - σ is stored as continuum + resonance parameters
 - Complex programs like NJOY, PREPRO convert the ENDF file to P-ENDF (point wise cross section), or G-ENDF (group wise) including Doppler broadening etc.
- 2 typical ways of using the data files
 - 1) As point wise P-ENDF
 - 2) Group wise G-ENDF

Typical neutron cross section



Point wise

- P-ENDF – convert to scatter data $[E_i, \sigma_i(E)]$ for each channel, assuming a loss in linearization of $< 1\%$
- During tracking \Rightarrow load all data (for each isotope):
 - ◆ Total σ
 - ◆ Elastic
 - ◆ Capture
 - ◆ Inelastic
 - ◆ ...
- For each neutron on each step perform a search in the data and try to interpolate the σ (for each isotope in the material and for each channel for the selected target isotope)
- Advantage point wise treatment: more accurate tracking, especially self-shielding on resonances (will be explained later)
- Disadvantage: slow tracking, CPU is wasted for scanning the tables

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 contains the <average> σ :

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

- The group structure (widths of each group) are suggested by NJOY etc. to minimize the error on the most commonly used isotopes.
- Advantage: fast
- Disadvantage: effects like self shielding are not accurately reproduced

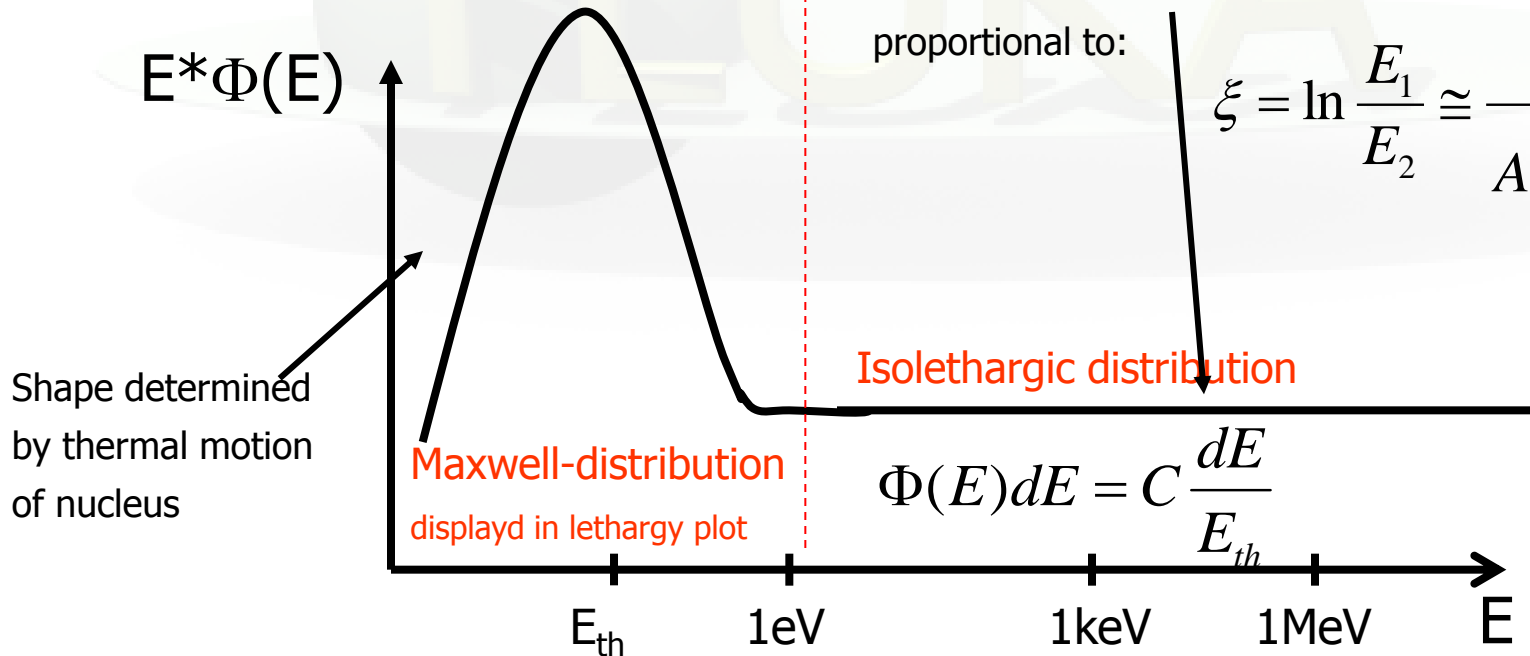
Fluence

- Problem when library is created: we need to know a-priori the neutron fluence $\Phi(E)$ the user will use in his simulation
- Solution: Make an assumption (not far from reality) on $\Phi(E)$
- The fluence assumption is needed as a weighting function for calculating the average cross section on each group

$$\Phi(E)dE = e^{-E/E_{th}} \frac{E}{E_{th}} dE$$

If we neglect absorption, elastic interacting neutron loses constant fraction of energy proportional to:

$$\xi = \ln \frac{E_1}{E_2} \cong \frac{2}{A + \frac{2}{3}} = \text{const}$$



Self shielding

- Problem: What if we have a thick material or huge absorption σ ?
- A resonant material will cause sharp dips in the neutron fluence corresponding to each resonance. The reaction rate will be reduced and the resonance is said to be “self-shielded”
- Solution: The magnitude of self shielding is a complicated function of the geometry and composition. A simple model “Bondarenko Method”:
The fluence $\Phi(E)$ is assumed to vary inversely as the total macroscopic cross section

$$\Phi_l^i(E) = \frac{C(E)}{[\sigma_0^i + \sigma_l^i(E)]^{l+1}}$$

$C(E)$ = smooth part like in isoethargic

σ_0^i = background σ (effect of all other isotopes)

l = dependence on large system “isotropic scattering”



*Description of the
Low Energy Neutron Library*

Introduction

- Transport and interactions of neutrons with energies below 20 MeV are handled by a dedicated library
- Neutron interactions at higher energy are handled by FLUKA nuclear models
- In FLUKA we call neutrons below 20 MeV **low energy neutrons**
- The low energy neutron library uses a multigroup approach
- About 230 materials at different temperatures are available
- The library handles also gamma generation, energy deposition by kerma factors, residual nuclei production, secondary neutrons, fission neutrons
- For some isotopes/materials: self shielding, molecular binding, correlated gamma generation, point-wise transport

Group Transport Technique

- In neutron transport codes in general two approaches used: point-wise (“continuous” cross sections) and group-wise transport
- Point-wise follows cross section precisely but is can be time and memory consuming
- Group approach is widely used in neutron transport codes because it is fast and gives good results for most applications
- Group-wise transport: energy range of interest is divided into a given number of discrete intervals: **energy groups**
- Elastic and inelastic reactions are not simulated as exclusive processes but by a **group to group transfer probability: downscattering matrix**
- Downscattering means that a neutron in a given group undergoes a scattering event where it loses energy. Afterwards it is in a group of lower energy. If the neutron does not lose enough energy to be in an other group it stays in the same group (in-scattering).
- In thermal region neutrons can gain energy, which is taken into account by an **upscattering matrix** which contains the transfer probability to be scattered into a group of higher energy

The FLUKA Low Neutron Library

- FLUKA uses the group transport technique
- Number of groups: 260 of approximately equal logarithmic with, the actual energies limits of each group can be found in the manual (or can be printed to *.out file), NB the **group with the highest energy has the number 1**, the group with the lowest energy has number 260
- Thermal groups: 31, with 30 upscattering groups
- Energy range of library: 0.01 meV - 20 MeV
- Based on recent versions of evaluated nuclear data files: ENDF/B-VI.8, ENDF/B-VII.0, Jendl-3.3, Jeff-3.1,...
- About 230 isotopes/materials available
- Almost all materials available at 2 temperatures: 87K, 296K
- Some also at 4K and 120K
- NB: Because of the group technique the energy of a neutron below 20 MeV is only defined within the accuracy of the groups

Gamma generation

- Gamma generation from $(n,x\gamma)$ reactions is done for the elements where data is available in the evaluated nuclear data files (c.f. manual for a complete list)
- Gamma generation is done also by a multigroup scheme
- Number of groups: 42, NB the number of gamma groups is different from the number of neutron groups!
- Energy range: 1keV - 50 MeV, NB the gamma energy range is different from the neutron energy range
- The actual energy of the generated photon is sampled randomly in the energy interval corresponding to its gamma group, exception: 2.2 MeV transition of deuterium and 478 keV photon from $^{10}\text{B}(n,\alpha)$ and gamma cascades from $\text{Cd}(n, \gamma)$ and $\text{Xe}(n, \gamma)$
- 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)

Energy deposition

- Energy deposition by neutrons below 20 MeV is estimated by means of kerma factors
- For some materials with gamma production the kerma values of some groups (mainly at high energies) are problematic (c.f manual). The reason is inconsistent data in the evaluated data files. Effort was addressed to apply corrections to improve the situation but it there are still some materials with problematic kermas.
- The user should check carefully the results of simulations with these materials. However, the effect should vanish in a typical simulation.

Secondary and Fission Neutrons

- Neutrons from (n, xn) reactions are taken into account implicitly by a group dependent **non-absorption probability**, i.e. a factor by which the weight (or the multiplicity) of a neutron is multiplied after a collision
- If the only possible processes are scattering and capture the non-absorption probability is smaller than 1, if also $(n, 2n)$ is possible the factor is bigger than 1
- 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 isotope and neutron energy
- The fission neutron multiplicity was obtained separately from the evaluated data files
- The fission fragments are not transported, the energy is deposited at the spot

Charged particle generation

- Recoil protons from hydrogen and protons from $^{14}\text{N}(n,p)$ 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
- If point-wise transport has been requested α and ^3H fragments from neutron capture in ^6Li (soon also ^{10}B) can also be transported explicitly
- All other charged secondaries produced in low energy neutron reactions, including fission fragments, are not transported but their energy is deposited at the point of interaction using kerma factors

Residual nuclei production

- Definition 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 except Zinc (Zn) data are available for estimating residual nuclei production by low energy neutrons
- 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 used in
LOW-MAT card

Identifiers: used in
LOW-MAT card

Using the Low Energy Neutron Library

- How to activate low energy neutron transport?
 - Explicit: giving **LOW-NEUT** card
 - Implicit: most **DEFAULT** cards (except EM-CASCA), or by not giving a **DEFAULT** card
 - 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 card)
- What does the user have to do?
 - Setting correspondence between the actual material and the material in the low neutron library (LOW-MAT card), if not done by default
 - Setting transport thresholds with PART-THR, if defaults are not ideal for the actual problem
 - Requesting special features like point wise cross sections (LOW-NEUT), biasing downscattering matrix (LOW-DOWN)

Input Cards: LOW-NEUT [1/4]

This card activates low-energy neutron transport.

- WHAT(1): number of neutron groups of the library
 - Default: 260, the FLUKA2006 library had 72 groups
- WHAT(2): number of gamma groups of the library
 - Default: 42, the FLUKA2006 library had 22 groups
- WHAT(3): maximum energy (in GeV) of the library
 - Default: 0.02, this value was 0.0196 in the FLUKA2006 library
- WHAT(5): number of thermal groups
 - Default: 31

LOW-NEUT

n-groups: 260 ▼

Print: ▼

γ -groups: 42 ▼

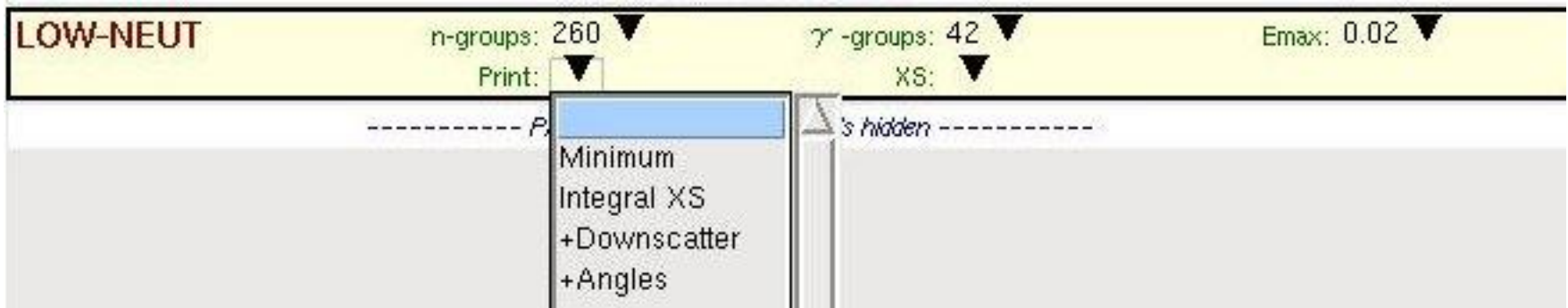
X8: ▼

E_{max}: 0.02 ▼

- The defaults for WHAT(1) – WHAT(3) and WHAT(5) are fine. The only reason for changing them is when you use the FLUKA2006 library. N.B. WHAT(5) has no corresponding input in *flair*.

Input Cards: LOW-NEUT [2/4]

- WHAT(4): printing flag, neutron cross section information is written to *.out file
 - ◆ 0.0 Minimum
 - ◆ 1.0 integral cross sections, kerma factors and probabilities
 - ◆ 2.0 additionally downscattering matrices and gamma matrices
 - ◆ 3.0 additionally scattering probabilities and angles
 - ◆ 4.0 residual nuclei information
- Default: 0.0 (minimum)



Input Cards: LOW-NEUT [3/4]

- The output for WHAT(4) = 1
 - Group energy limits, average energies, velocities and momenta, thermal velocities, gamma group limits
 - For each material: availability of residual nuclei information (the line: "RESIDUAL NUCLEI INFORMATIONS AVAILABLE" indicates the possibility to use option RESNUCLEi with WHAT(1)= 2.0
 - for each neutron energy group in each material:
 - ◆ SIGT = total cross section
 - ◆ SIGST = "scattering" cross section: $\sigma(n,n) + 2\sigma(n,2n) + 3\sigma(n,3n)$
 - ◆ PNUP = upscatter probability, is 0.0 in non thermal groups
 - ◆ PNABS = Probability of Non-ABSorption (= scattering)
PNABS = SIGST/SIGT, and can sometimes be > 1 because of (n,xn) reactions
 - ◆ GAMGEN = GAMma GENERation probability = gamma production cross section divided by SIGT and multiplied by the average number of γ per (n, γ) reaction
 - ◆ NU*FIS = fission neutron production = fission cross section divided by SIGT and multiplied by ν , the average number of neutrons per fission
 - ◆ EDEP = Kerma contribution in GeV per collision
 - ◆ PNEL, PXN, PFISS, PNGAM = partial cross sections, expressed as probabilities (*i.e.*, ratios to SIGT). In the order: non-elastic, (n,xn), fission, (n,gamma)

Input Cards: LOW-NEUT [4/4]

- WHAT(6): $i_0 + 10 * i_1$:
 - $i_0 = 1$: available point wise cross sections used and explicit and correlated ${}^6\text{Li}(n,\gamma){}^7\text{Li}$, ${}^6\text{Li}(n,t){}^4\text{He}$, ${}^{40}\text{Ar}(n,\gamma){}^{41}\text{Ar}$, ${}^{\text{x}}\text{Xe}(n,\gamma){}^{\text{x}+1}\text{Xe}$ and ${}^{113}\text{Cd}(n,\gamma){}^{114}\text{Cd}$ 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 weight to compensate for the "wrong" multiplicity
 - ◆ = 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)

The screenshot shows the LOW-NEUT input card interface. At the top, there are several parameters: "n-groups: 260", "Print:", "γ-groups: 42", "XS:", and "Emax: 0.02". A dropdown menu is open for the "XS:" parameter, showing the following options: "Default", "Pt-wise XS", "Fission mult", "Pt-XS+Fission", and "reset". The "Default" option is currently selected. Below the menu, there is a line of text: "----- PART-THR ... STDP : 2 cards hidden".

Input Cards: LOW-MAT [1/3]

- **The LOW-MAT card sets the correspondence between FLUKA materials and the low energy neutron transport**
- If a material has the same name as the name given in the list of low neutron material, the correspondence between material and low energy neutron transport is set automatically, 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 **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 H₂ you have to do this explicitly by a LOW-MAT card

Input Cards: LOW-MAT [2/3]

- WHAT(1): Name of the material
 - ◆ 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 87K 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) is 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

Creating a material at 87K

- Creating a material at 87K works the same way as for 296K. If a material is used at **several temperatures in one simulation** the user has to **define 2 different materials**. These materials have to have different names.
 - Create the material with a MATERIAL card and give it a name in SDUM. You don't have to do this if the material is already predefined and you don't want to use it also at an other temperature.
 - Give a LOW-MAT card with WHAT(1) is 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 87K material you want to use
 - Give in SDUM of the LOW-MAT card the name provided in the same table

Example compound at 87K

- Example: water at 87K
 - Create a material hydrogen and give it some name (HYDR_87), do the same with oxygen (OXYG_87)
 - Give a LOW-MAT card for HYDR_87 chose the right cross sections (those for 87K) identifiers and name (cf. list in manual), do the same for oxygen
 - Create a material WATER_87 by giving first a MATERIAL card and then a corresponding COMPOUND card with the right composition

MATERIAL z: 1.	Name: HYDR_87 Am: 1.00794	# A:	ρ : 8.988E-5 dE/dx: ▼
LOW-MAT	Mat: HYDR_87 ▼	LowMat: H. Free gas natural Hydrogen, 87K ▼	
MATERIAL z: 8.	Name: OXYG_87 Am: 15.9994	# A:	ρ : 0.001492 dE/dx: ▼
LOW-MAT	Mat: OXYG_87 ▼	LowMat: 16O. Oxygen 16, 87K ▼	
MATERIAL z:	Name: WATER_87 Am:	# A:	ρ : 1. dE/dx: ▼
COMPOUND r1: 2.0 r3:	Name: WATER_87 ▼ M1: HYDR_87 ▼ M3: ▼	Mix: Atom ▼ r2: 1.0	Elements: 1.3 ▼ M2: OXYG_87 ▼

Input Cards: LOW-BIAS [1/2]

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

However it is preferable to use the PART-THR

- WHAT(1): number of the group to apply a transport cut-off, i.e. neutrons in groups with numbers \geq WHAT(1) are not transported. N.B. that the groups 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

The screenshot shows the LOW-BIAS input card interface. The title bar is "LOW-BIAS". The "Ecut:" field is set to "19.155 MeV" with a dropdown arrow. The "NonAnalogue:" field is set to "No cut-off" with a dropdown arrow. The "Survival:" field is empty. The "Reg:" field is set to "No cut-off" with a dropdown arrow. The "to Reg:" field is empty. The "Step:" field is empty. A dropdown menu is open, showing a list of energy values: 20.000 MeV, 19.640 MeV, 19.155 MeV (highlighted), 18.682 MeV, 18.221 MeV, 17.771 MeV, 27.333 MeV, 16.905 MeV, and 16.487 MeV. Below the interface, there is a text description: "requests non-analogue absorption and/or low energy neutron transport on a region". At the bottom, there is a table with columns 1, 2, 3, 4, 7 and values 0.0, 3, 0.0.

1	2	3	4	7
0.0	3	0.0		

Input Cards: LOW-BIAS [2/2]

- Definition analogue absorption: a neutron does not exist any more after an absorption process
- Definition non-analogue absorption: the neutron is not killed after an absorption process but it lives on with a lower weight, capture gammas are created with a weight corresponding to the surviving neutron
- WHAT(2): Group limit for non-analogue absorption (neutrons in groups \geq WHAT(2) undergo non-analog absorption)
 - Default: 230
- WHAT(3): non-analogue survival probability
- **WARNING: Only experts should modify the non-analogue absorption survival probability!**
- If no LOW-BIAS card is given non-analogue absorption depends on the DEFAULT card cf. manual
- The change of weight of the particle is taken into account (cf. lecture about biasing)

Transport cut-offs

- Definition 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 transport cut off for neutrons 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.

Self shielding [1/2]

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

Self shielding [2/2]

- 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 must not 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
 - Give additionally a LOW-MAT card and give the proper **identifiers for cast iron** in WHAT(2)-WHAT(4) and SDUM
 - Give a MATERIAL card to create a material called CastFe (or any other name you like)
 - Give a COMPOUND card for CastFe to composed a compound of FeCarbSS and CARBON (predefined)

Angular distribution

- Scattering transfer probability between groups is calculated (when the library is produced) by a Legendre polynomial expansion of order $N=5$

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

$\mu = \bar{\Omega} \cdot \bar{\Omega}' = \text{scattering angle}$

$N = \text{chosen Legendre order of anisotropy}$

- Result of this P5 expansion is a set of **3 discrete polar angle cosines and 3 corresponding probabilities**, i.e. for a given transfer $g \rightarrow g'$ only three values are possible for the polar angle
- The azimuthal angle is sampled from uniform distribution and can have any value between 0 and 2π

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 discrete 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 (cf. manual chapter output)

Materials with molecular binding

- 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 it can change the energy and spatial distributions
- Example: CH_2 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 card 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

Advanced low energy neutron features:

- The LOW-DOWN card biases the downscattering probability during low-energy neutron transport on a region by region basis
 - It can be useful for very particular problems
 - If not used properly it can lead to big errors
 - Only recommended for experts

The logo for the FLUKA Monte Carlo simulation code. It features the word "FLUKA" in a bold, yellow, 3D-style sans-serif font. The letters are slightly offset from each other, giving a sense of depth. The logo is set against a light green circular background that has a subtle gradient and a soft shadow beneath it.

Summary

- Most of the simulations in FLUKA use the low energy neutrons, implicitly via the DEFAULT card
- Low neutron transport in a material is enabled by a LOW-MAT card, only needed if the material is not a predefined material, or temperature or self-shielding different from the defaults ones have to be selected
- For materials at 87K a LOW-MAT card has to be given always
- Use self shielded materials properly for “bulky” and “pure” (in isotopic composition) materials
- To set transport cuts for neutrons be aware that there are 2 cards: LOW-BIAS and PART-THR. Which to use depend on the energy of your cut.
- Don't forget to give a LOW-MAT card for each MATERIAL (not needed for default materials), don't give a LOW-MAT card for compounds