The neutron_hp neutron transport code.

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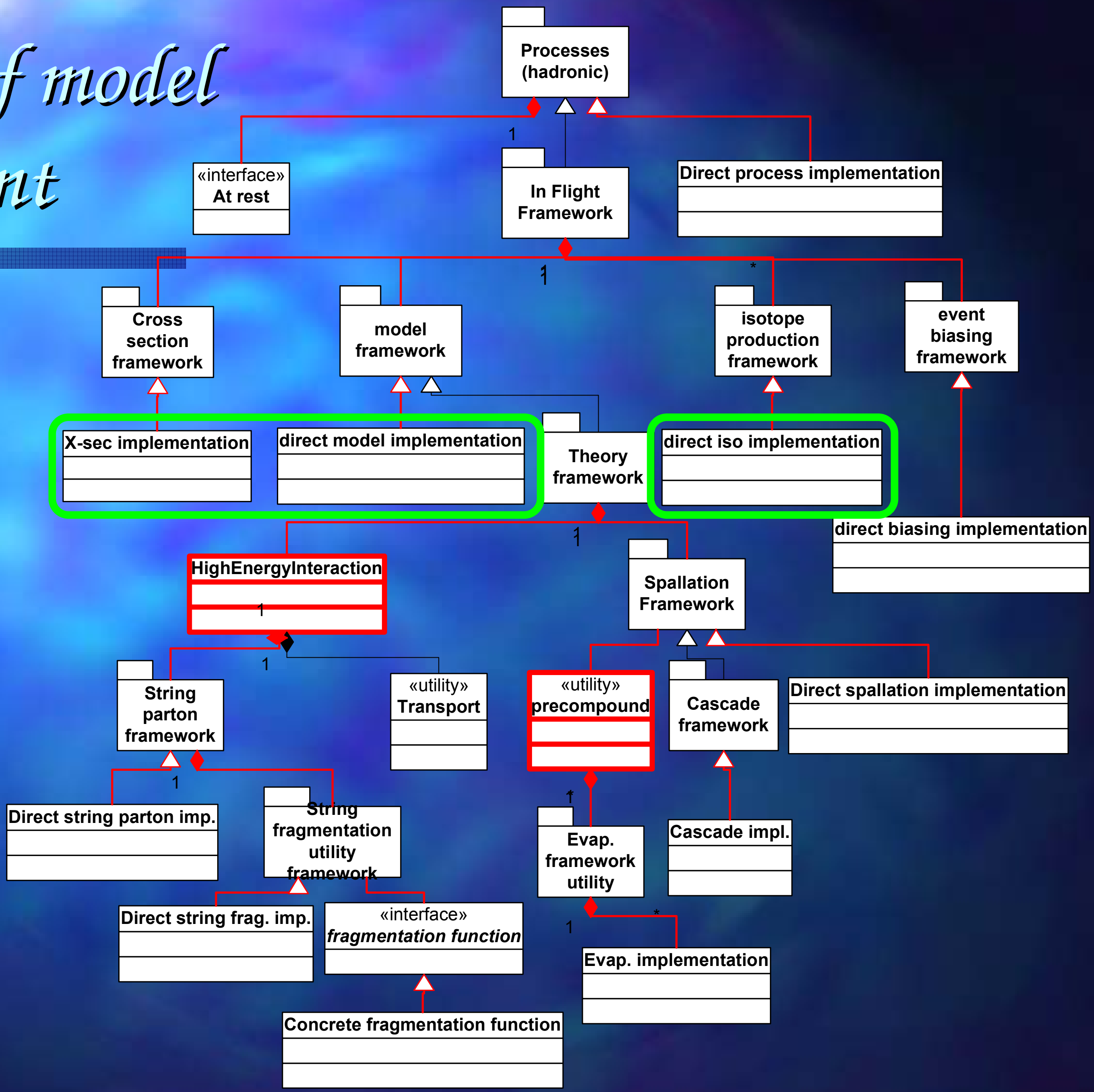
A sketch of model management



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Models for neutron interaction and thermalization.

- Neutron_hp sampling codes for ENDF/B-VI derived data formats are completely generic.
- Simulate the cross-sections and interactions of neutrons with kinetic energies below 20 MeV down to thermal energies .
- The upper limit is set only by the evaluated data libraries the code is based on.
- I consider elastic scattering, fission, capture and inelastic scattering as separate models

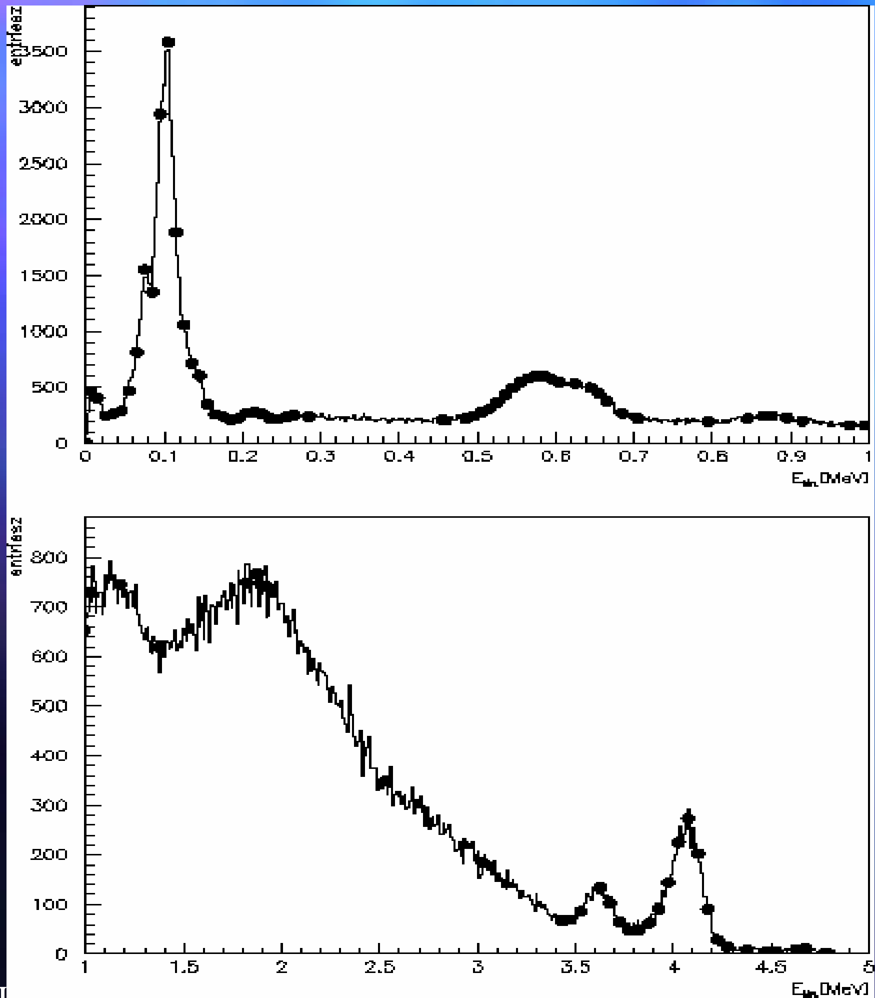
The neutron_hp transport models

- neutron_hp models and cross-sections:
  - Uses the unix file-system to ensure granular and transparent access/usage of data sets.
  - More than 10^11 events run.
  - Uses point-wise cross-sections
    - ➔ no artifacts due to multi-group structure.

Low energy neutron data: G4NDL0.2, 3.7

- Are granular selections of data from (alphabetic)
  - Brond 2.1
  - CENDL 2.2
  - EFF-3
  - ENDF/B (VI.0, VI.1, VI.5)
  - ENSDF
  - FENDL/E2.0
  - JEF 2.2
  - JENDL (3.1, 3.2, FF; 3.3 currently under study)
  - MENDL-2(P)
- Large parts of the initial (0.2) selection is guided by the FENDL-2
- G4NDL0.2 for non-thermal application

Example of data driven modeling: neutron capture, and isotope production



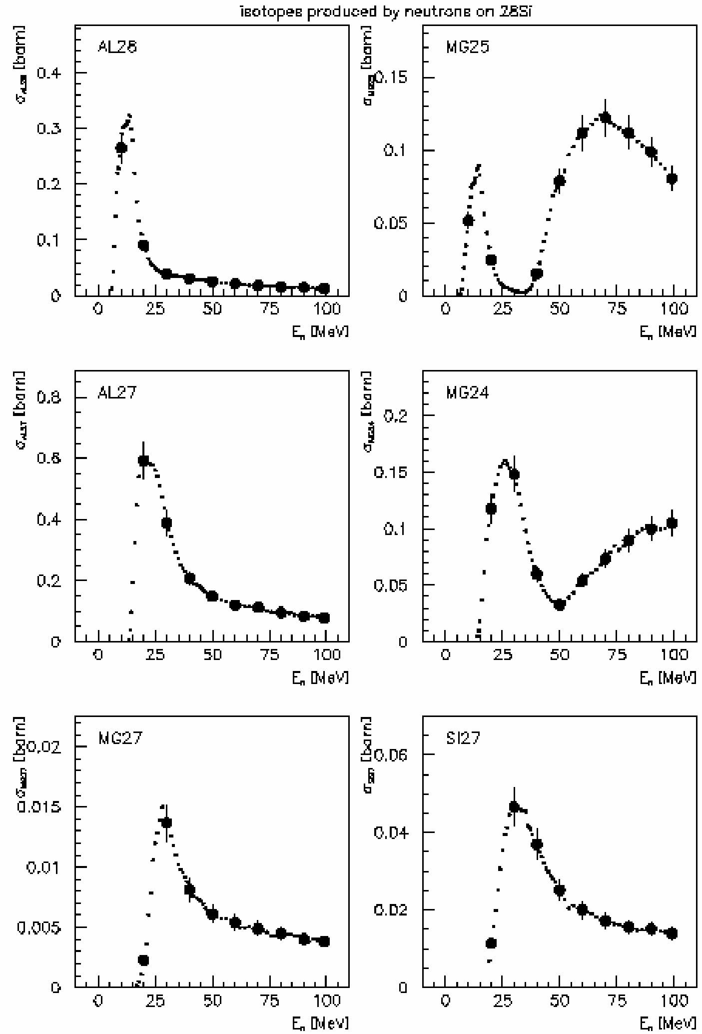


Figure 48: Isotope production cross-sections for neutron induced production of important isotopes as simulated using the isotope-production code in GEANT4. Large points are simulation results, small points are evaluated data from the MENDL2 data library.

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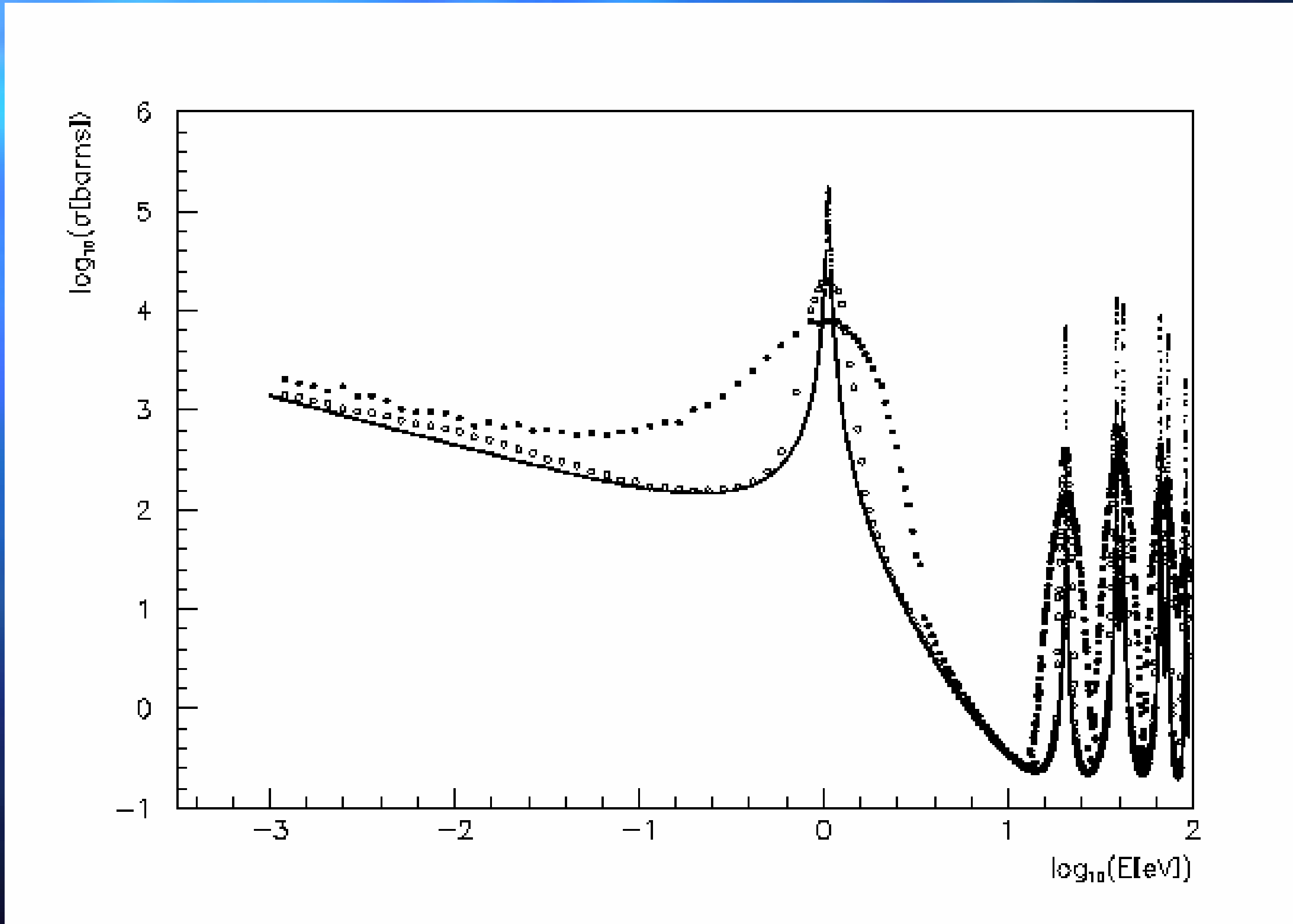
Doppler broadening

- Does exact doppler broadening on the fly, based on 0K data
  - ➔ No pre-formatting of data to fixed temperatures, and easy simulation of set-ups with mixed temperatures.
  - ➔ Adds the doppler bias to the nuclear momentum distribution
- Point one is to the best of my knowledge not possible with any other transport code.

Doppler broadening



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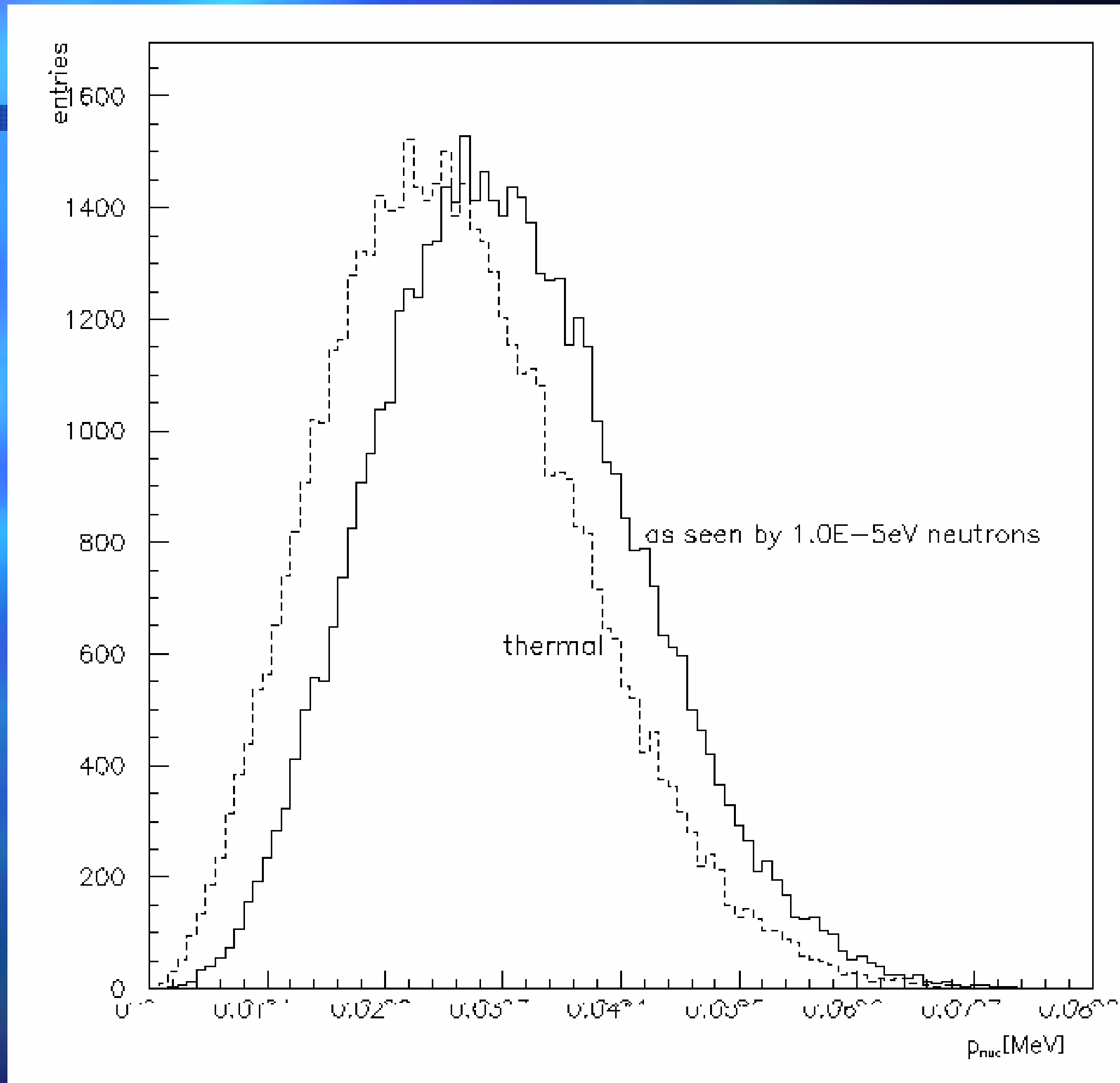
The doppler bias illustrated for Carbon



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Elastic scattering

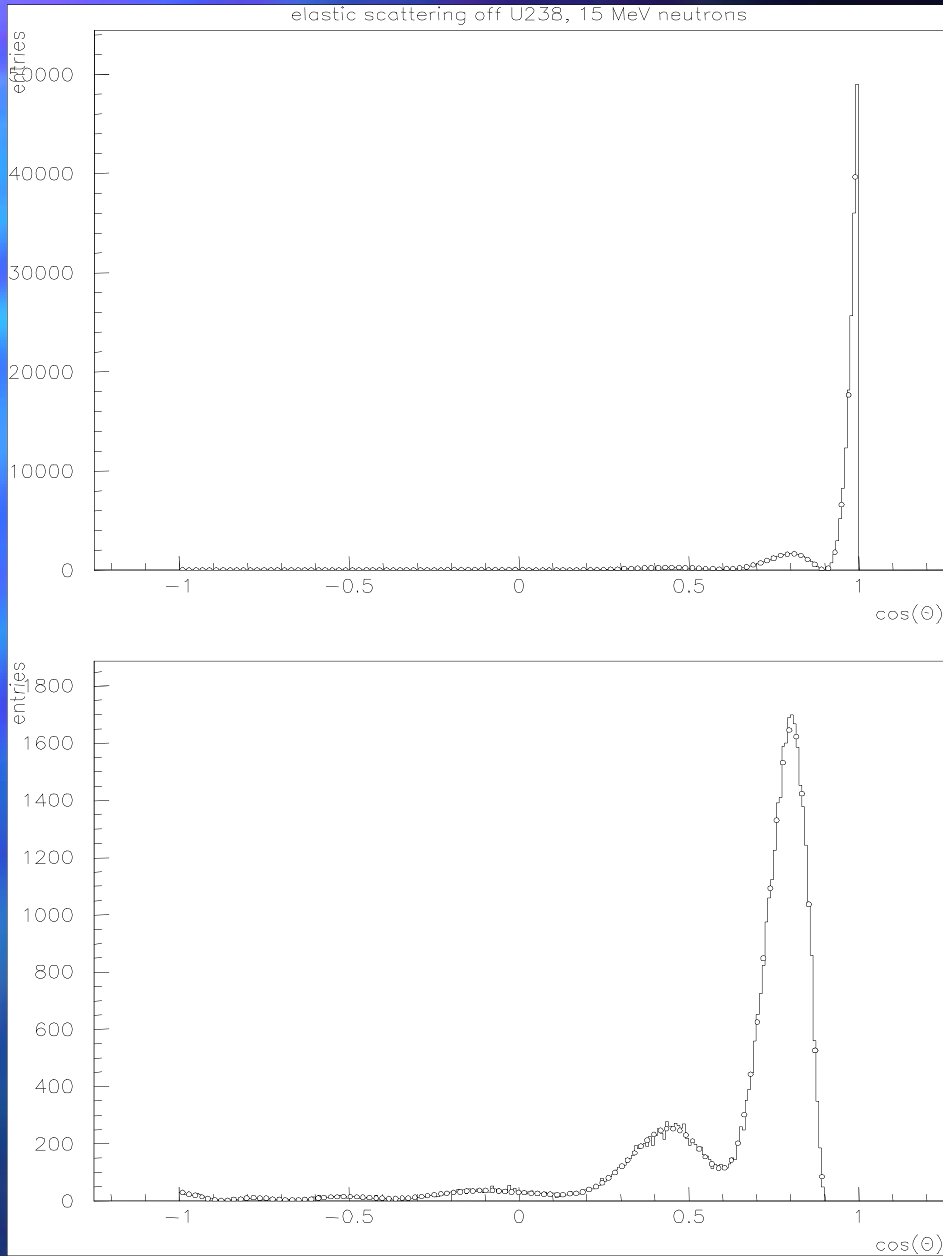
- Two representations of the differential cross section are supported
  - Tabulation as a function of the cosine of the scattering angle and incident neutron energy

$$\frac{\mathrm{d}\sigma}{\mathrm{d}\Omega} = \frac{\mathrm{d}\sigma}{\mathrm{d}\Omega}(\cos\theta, E_n)$$

  - Legendre polynomial expansion

$$\frac{2\pi}{\sigma(E)}\frac{\mathrm{d}\sigma}{\mathrm{d}\Omega}(\cos\theta, E_n) = \sum_{l=0}^{n_l}\frac{2l+1}{2}a_l(E)P_l(\cos(\theta))$$

Elastic scattering



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Radiative Capture

- Described using
  - Photon multiplicities or photon production cross sections.
  - Discrete and continuous contributions to the photon energy spectrum.
  - Photon angular distributions.

Radiative Capture (2)

- Multiplicity representations
  - Full transition probability array.
  - Or tabulation of the multiplicity for each discrete line and a continuum contribution
    - For the continuum contribution, we write the normalized emission probability as:

$$f(E \rightarrow E') = \sum_i p_i(E) g_i(E \rightarrow E')$$

- Cross section representations
  - Tabulation only.

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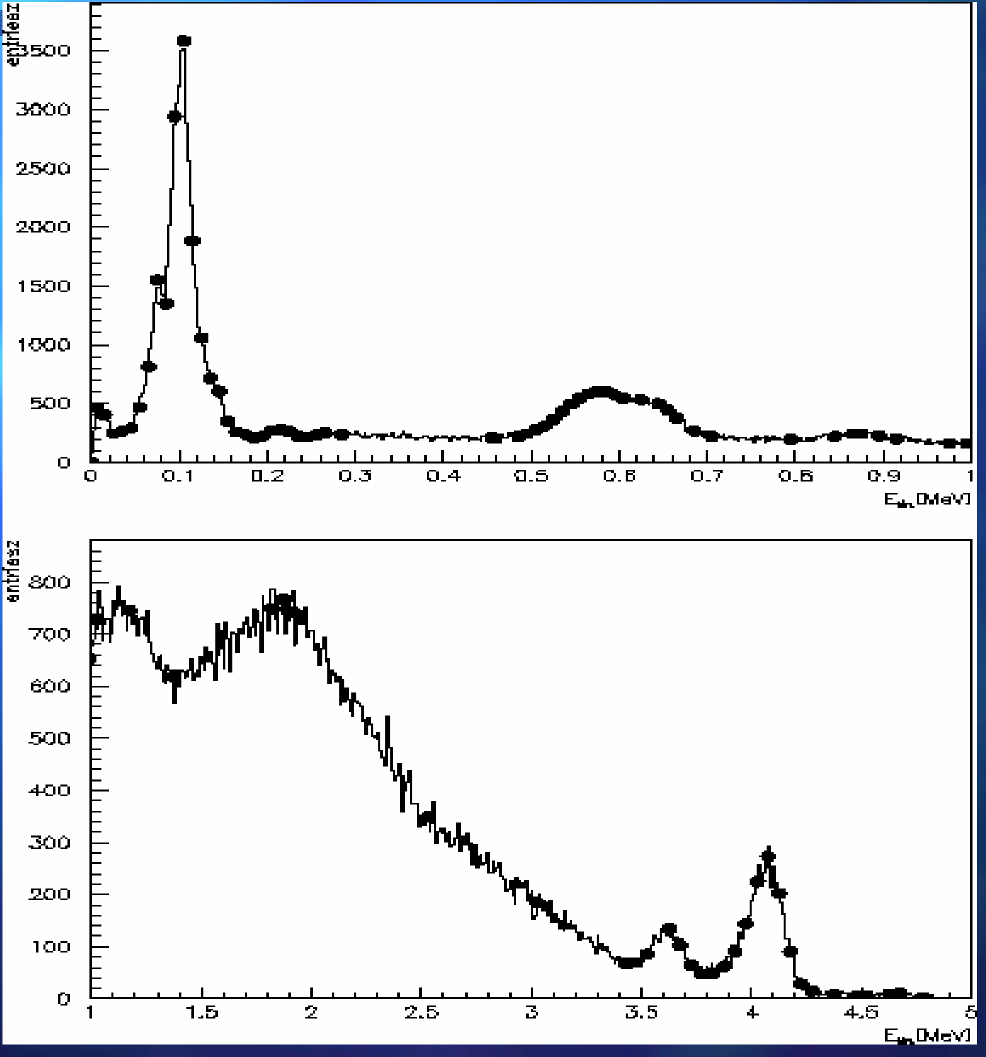
Low energy neutron capture



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Fission

- I include first, second, third, and fourth chance fission.
  - Neutron yields are tabulated as a function of incident and outgoing neutron energies
  - Angular distributions are either tabulated, or represented as a Legendre polynomial expansion.
    - If angular distributions are missing, isotropic distributions are assumed.

Fission (2)

- Six representations are available for neutron energy spectra
  - General evaporation spectrum

    $$f(E \rightarrow E') = f(E', \Theta(E))$$
  - Maxwell spectrum

    $$f(E \rightarrow E') \propto \sqrt{E'} e^{E'/\Theta(E)}$$
  - Evaporation spectrum

    $$f(E \rightarrow E') \propto E' e^{E'/\Theta(E)}$$
  - Watt spectrum

    $$f(E \rightarrow E') \propto e^{E'/a(E)} \sinh \sqrt{b(E)E'}$$

Fission (3)

- Madland Nix Spectrum

$$f(E \to E') \propto \frac{1}{2}\left[g\left(E',\langle K_l\rangle\right) + g\left(E',\langle K_h\rangle\right)\right]$$

- Where

$$g\left(E',\langle K\rangle\right) = \frac{1}{3\sqrt{\langle K\rangle\Theta}}\left[u_2^{3/2}E_1(u_2) - u_1^{3/2}E_1(u_1) + \gamma(3/2,u_2) - \gamma(3/2,u_1)\right]$$

- E_1 is the exponential integral, and

$$u_1\left(E',\langle K\rangle\right) = \frac{\left(\sqrt{E'} - \sqrt{\langle K\rangle}\right)^2}{\Theta}$$

$$u_2\left(E',\langle K\rangle\right) = \frac{\left(\sqrt{E'} + \sqrt{\langle K\rangle}\right)^2}{\Theta}$$

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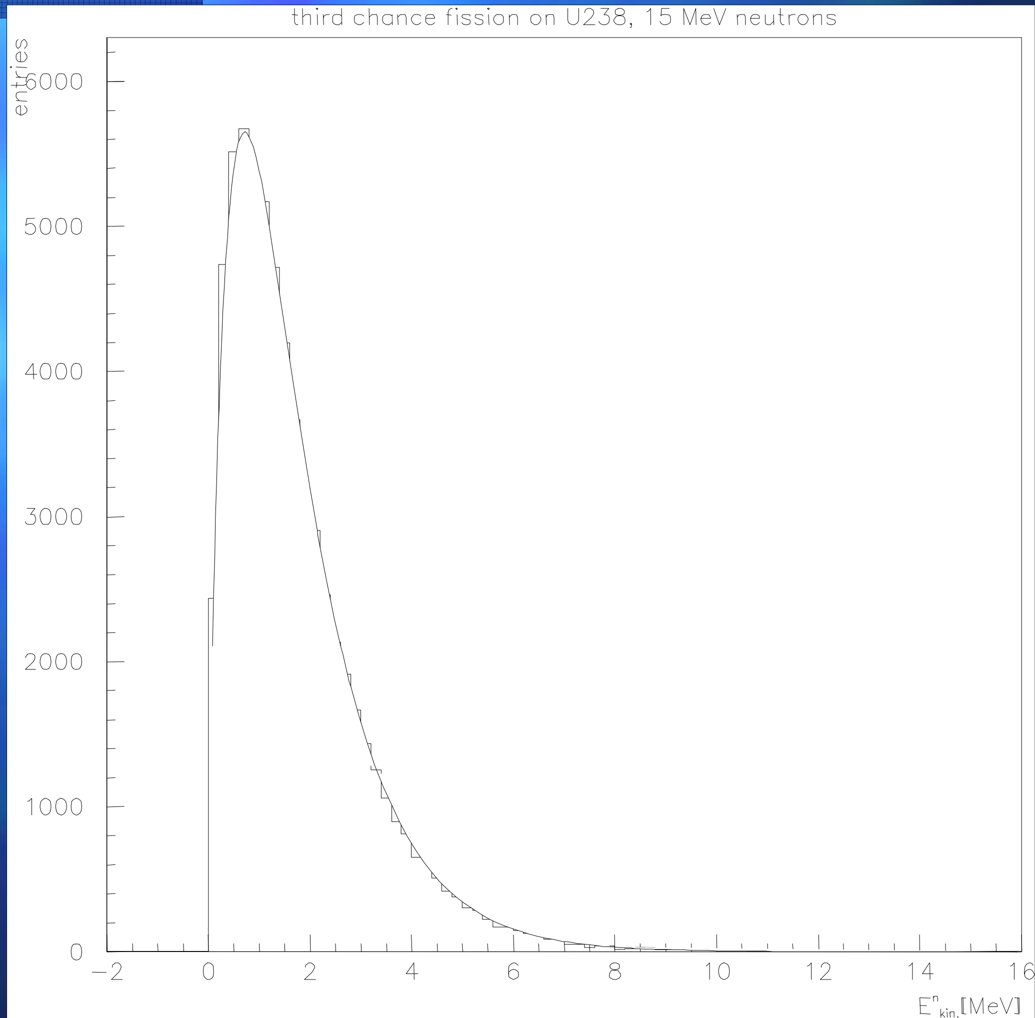
Fission simulation



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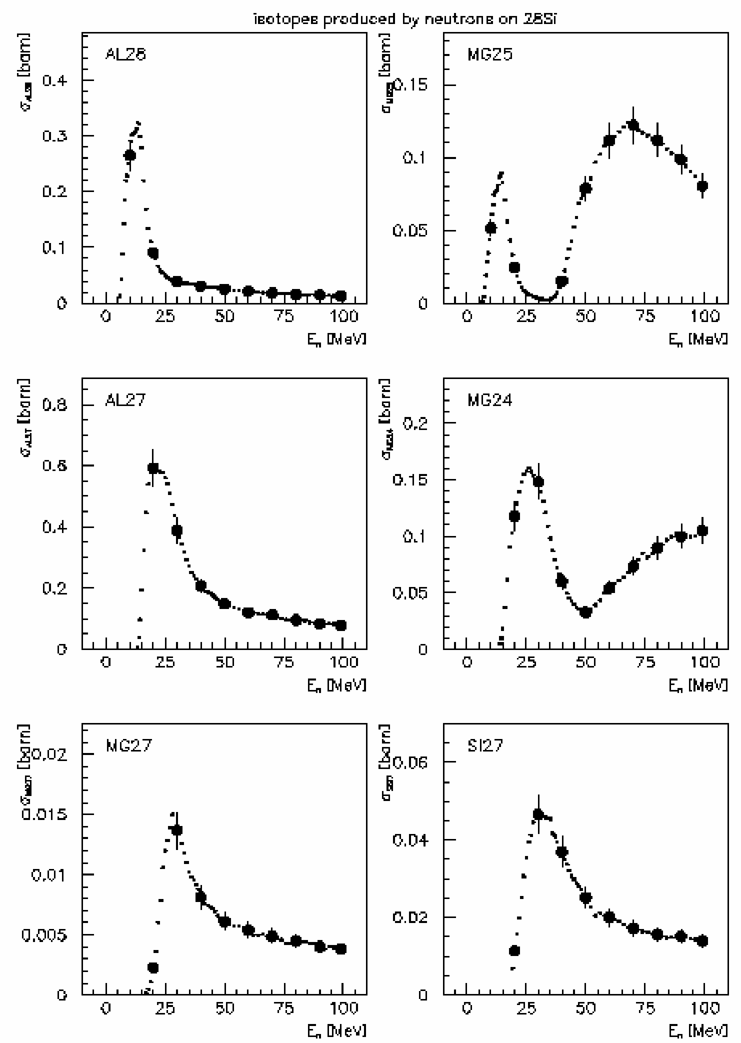
Inelastic scattering

- The following channels are currently included:

  n'(γs), np, nd, nt, n^{3}He, nα, nd2α, nt2α, n2p, n2α, npα, n3α, 2n, 2np, 2nd, 2nα, 2n2α, nX, 3n, 3np, 3nα, 4n, p, pd, pα, 2p, d, dα, d2α, dt, t, t2α, ^{3}He, α, 2α, and 3α.

- Photons that may be associated to the individual channels are described as in the case of capture.

Neutron induced isotope production



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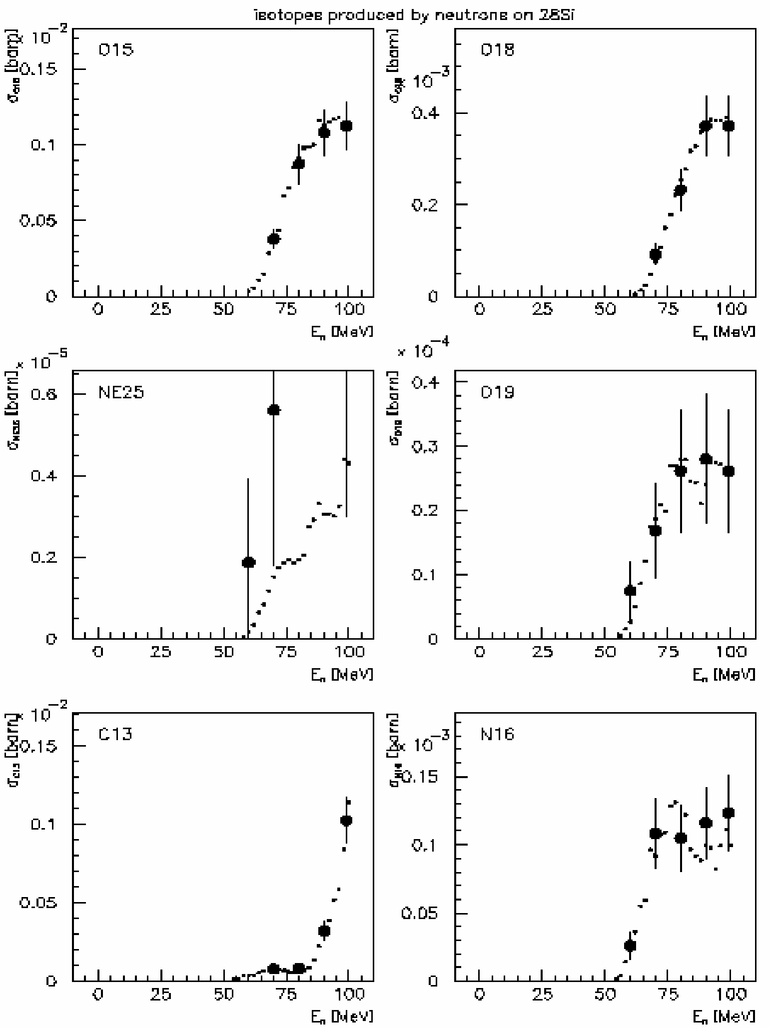
Isotope production



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Summary

- I have provided a neutron code suitable for a wide range of neutron transport problems.
- Future improvements will focus on details of the thermal scattering law, and the unresolved resonance region.