Update on CIELO Related Nuclear Data Measurements at the Gaerttner LINAC Center at RPI

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2014 CSEWG/CIELO meeting, BNL, November 3-6, 2014

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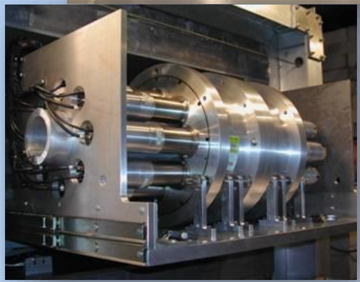
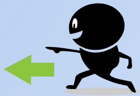
The Nuclear Data Program at the Gaerttner LINAC Center

- Driven by a 60 MeV pulsed electron LINAC ~**10^{12} n/s**
- **Neutron transmission**
  - Resonance region: 0.001 eV- 1000 keV,
  - High energy region: 0.4- 20 MeV
- **Neutron Capture**
  - Resonance region: 0.01-1000 eV
  - New detector array at 45m: 1 keV ~ 500 keV
- **Neutron Scattering**
  - High energy region: 0.4 MeV- 20 MeV
- **Prompt Fission neutron spectrum**
- **Lead Slowing Down Spectrometer**
  - Fission cross section and fission fragment spectroscopy.
  - (n,α), (n,p) and (n,γ) cross sections on small (radioactive) samples.

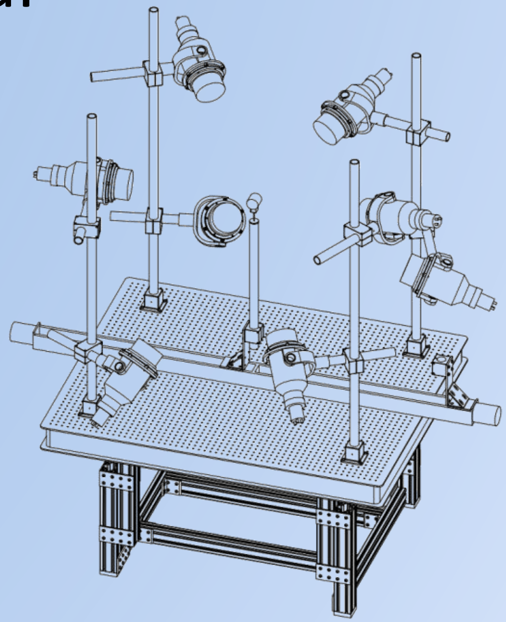


Summary of Related Activity

- ^{235}U fission and capture yield up to 3 keV.
  - Data was delivered to Leal (ORNL) for inclusion in the evaluation.
- ^{56}Fe - high resolution transmission, 0.5-20 MeV
  - Data was delivered to Leal (ORNL) for inclusion in the evaluation.
- Neutron scattering 0.5-20 MeV
  - Data on ^{238}U published.
  - Data on Fe finalized, publication in process. 
- Prompt fission neutron measurements
  - Data on ^{238}U in progress. 

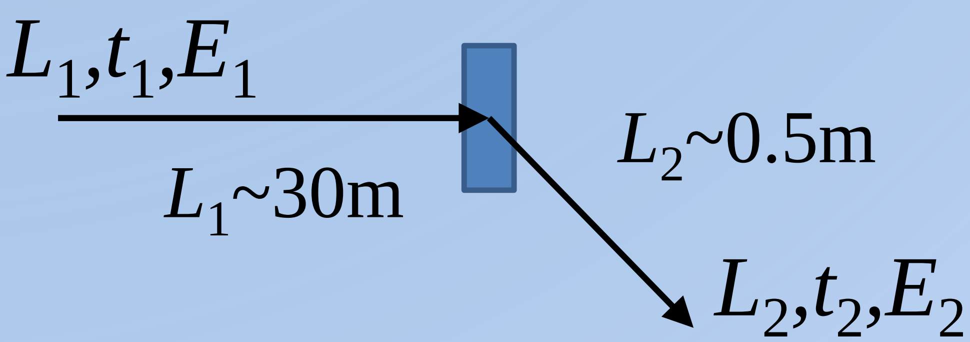
Neutron Scattering

- Provide accurate benchmark data for scattering cross sections and angular distributions in the energy range from 0.5 to 20 MeV
- Can be developed to provide differential elastic and inelastic scattering cross section measurements
- Design a flexible system: now also used for fission neutron spectra measurements



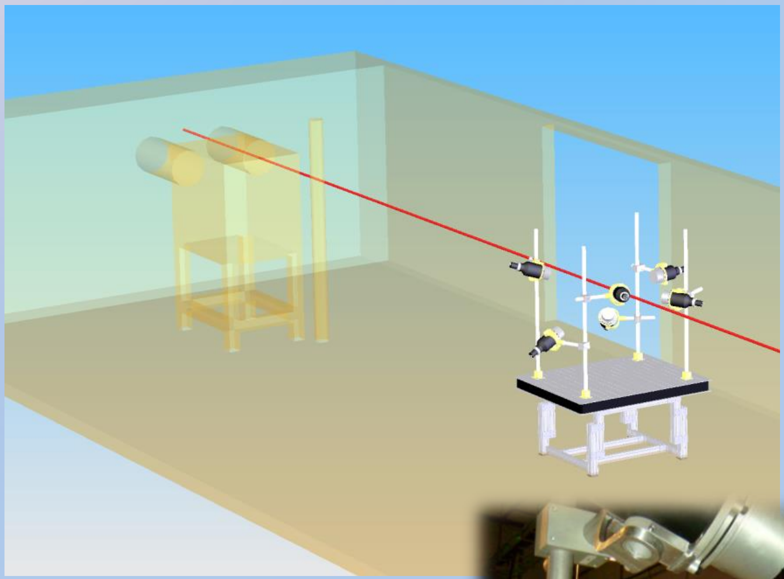
TOF Scattering Yield Measurement

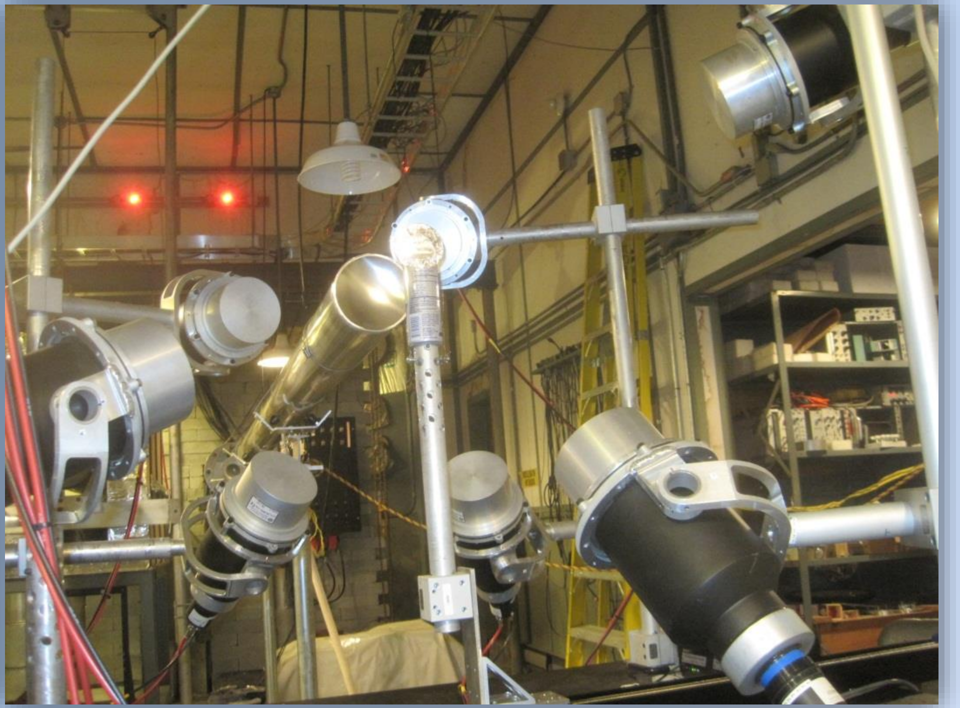
- Measure the total TOF $t=t_1+t_2$
- For all scattering events $E_2<E_1$
- In most cases the energy loss is small $E_1 \sim E_2$
- Since $t_1 >> t_2$ and $E_1 \sim E_2$ then for presentation the incident neutron energy E_1 is calculated using t and $L=L_1+L_2$



$$E(t) \approx m_n c^2 \cdot \left(\frac{1}{\sqrt{1-\left(\frac{L}{c \cdot t}\right)^2}} - 1 \right)$$

Scattering Detection System: Experimental Setup

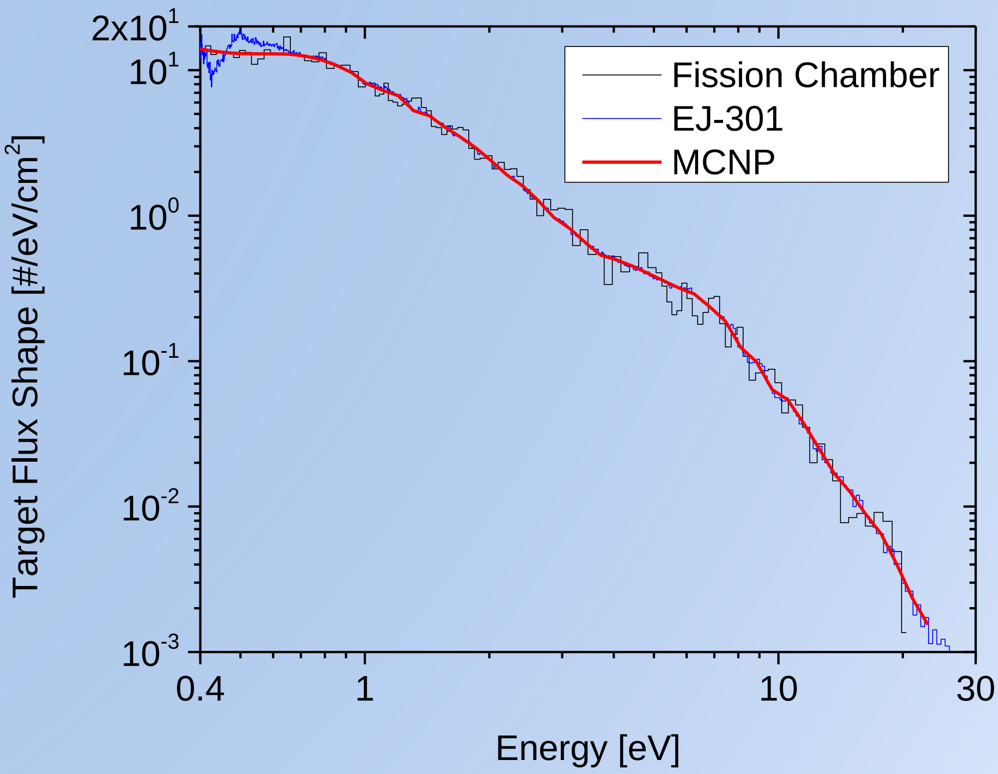




Low mass sample holder

Flux Shape Measurement

- Use a fission chamber with ~391 mg ^{235}U in the sample position
- Use ENDF/B-VII.1 fission cross section for ^{235}U
- Correct for transmission of all materials between the source and sample
- Compare to a similar measurement using EJ301 and SCINFUL calculated efficiency
- Combine the two data sets using fission for E< 1 MeV



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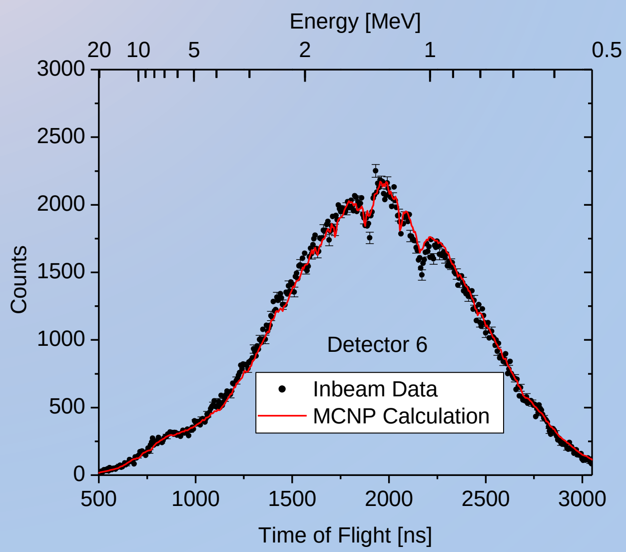
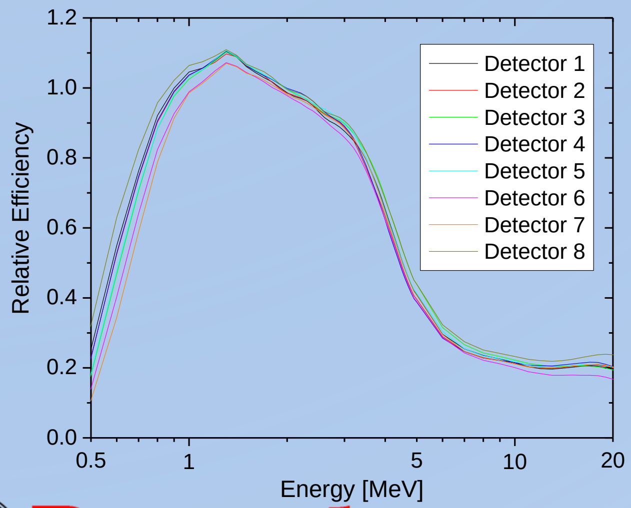
Efficiency as a Function of Energy





- Objective:
  - MCNP simulation of EJ301 response in the sample position must precisely agree with the measurement
- Methodology:
  - Use the measured flux as a source in MCNP simulation of the in-beam detector response
  - In MCNP set the detector efficiency η=1 (tally only the neutron flux shape)
  - Divide the measured response by the simulation results to get the efficiency $\eta(E)$ for each detector
  - During the experiment periodic gain calibrations are done to minimize gain shift.

Data Reduction

- Sum all files and dead time correct.
- The experimental count rate corrected for background and false neutrons:

$$Rn_i = Rn_i^s - fn_i^s - \frac{M^S}{M^O} \cdot \left(Rn_i^o - fn_i^o \right)$$

Rn_i^s, Rn_i^o - Sample and open neutron counts at TOF channel *i*

fn_i^s, fn_i^s - Sample and open false neutron counts for TOF channel *i*

M^S, M^O - Open and sample monitor counts for the run

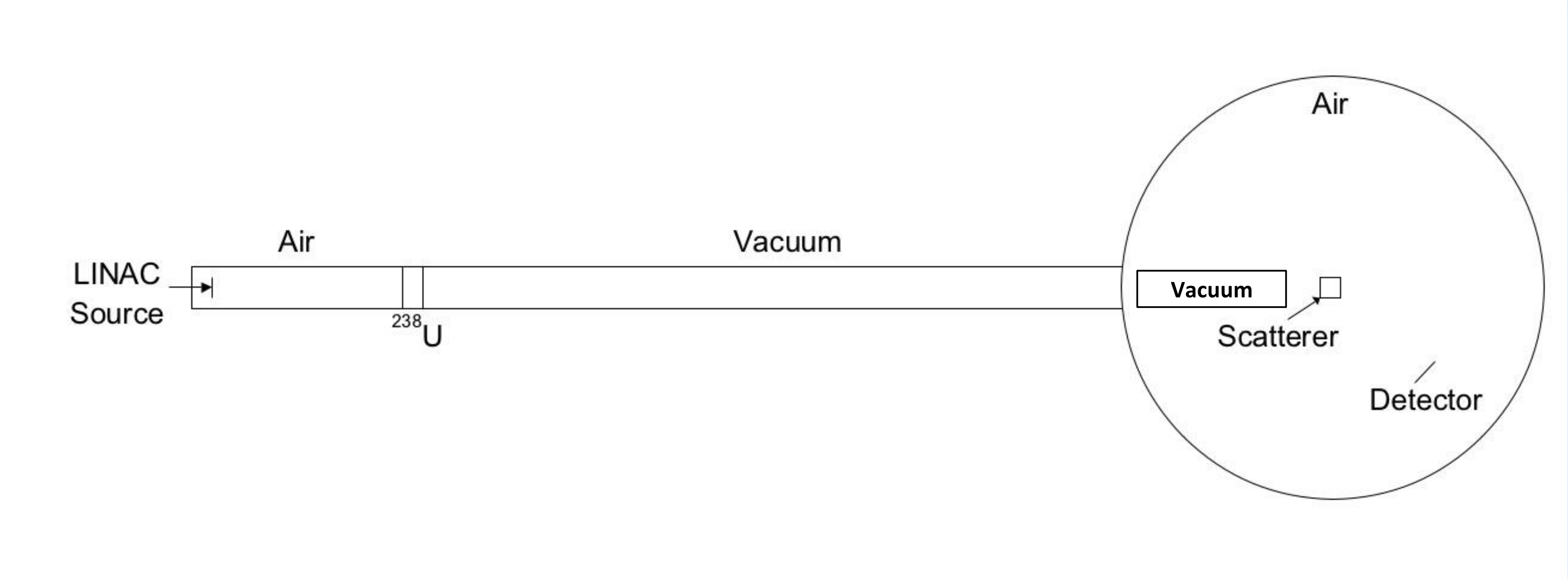
The false neutron correction: $$fn_i = \sum_{j=1}^{n_\gamma} f(A_j)$$ ~2% effect for ^{238}U

n_γ - Number of gammas in TOF channel *i*

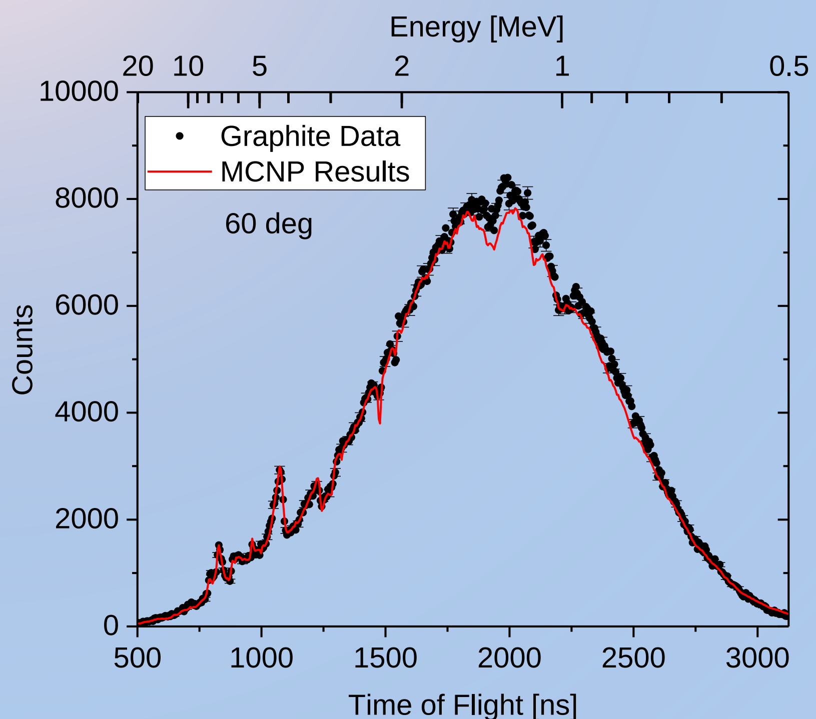
$f(A_j)$ - False neutron correction factor for pulse area A_j

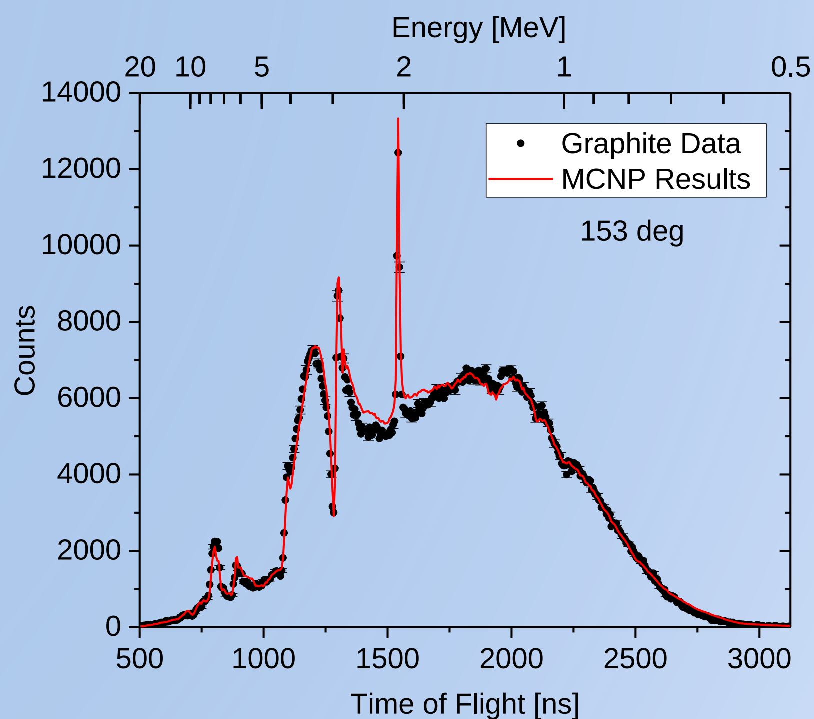
MCNP Simulation Geometry

- Use ASAP (As Simple As Possible) approach
- Use array of point detector tally F5 to model the EJ301 detector
  - Convolute the tally with the detector efficiency
- Include ¾” Depleted U filter in the simulation
- Include windows (Al)
- Include recent improvements of vacuum tube near the sample



Graphite Reference Results





- Differences between experimental data and MCNP calculations (ENDF/B-VII.1/JEFF-3.1) used to estimate systematic uncertainties
  - Systematic uncertainty ~ 2.6%

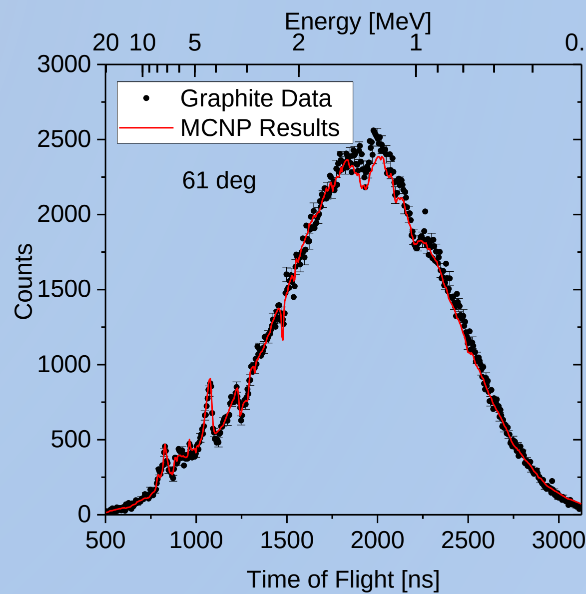
$$FOM_i = \frac{1}{n} \cdot \sum_{j=0.5MeV}^{n=20MeV} \frac{\left(C_{i,j} - MC_{i,j}\right)^2}{\varepsilon_{i,j}^2}$$

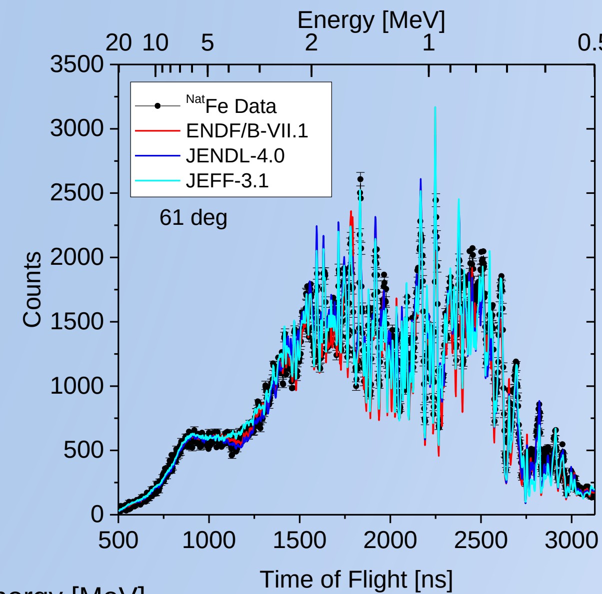
n – Total number of energy bins
C – Total neutron counts
MC – Normalized MCNP Results
i – Detector #
j – Energy bin
ε – uncertainty including experimental and simulation

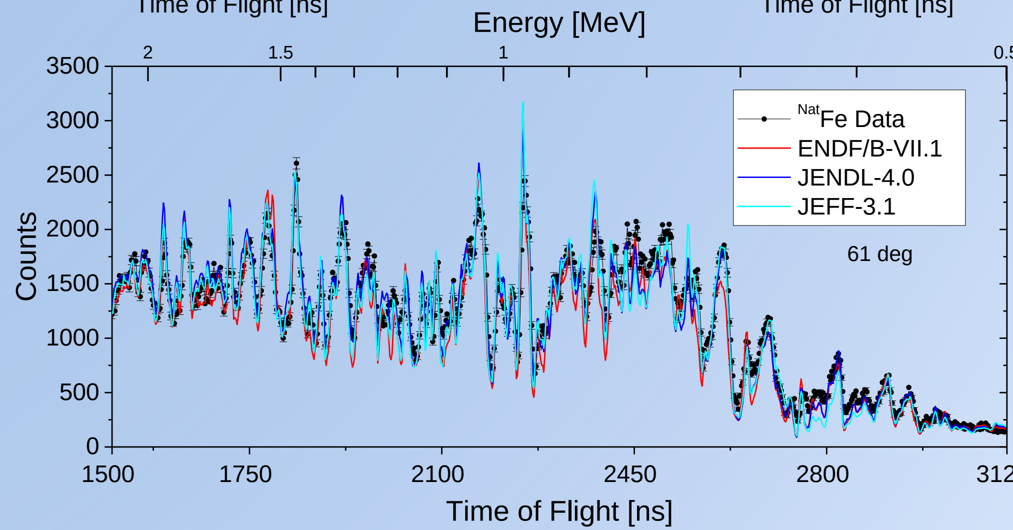
^{nat}Fe Scattering - 61°

Reference	FOM
Graphite	1.20

Library	FOM
ENDF/B-VII.1	13.67
JEFF-3.1	13.45
JENDL-4.0	10.97







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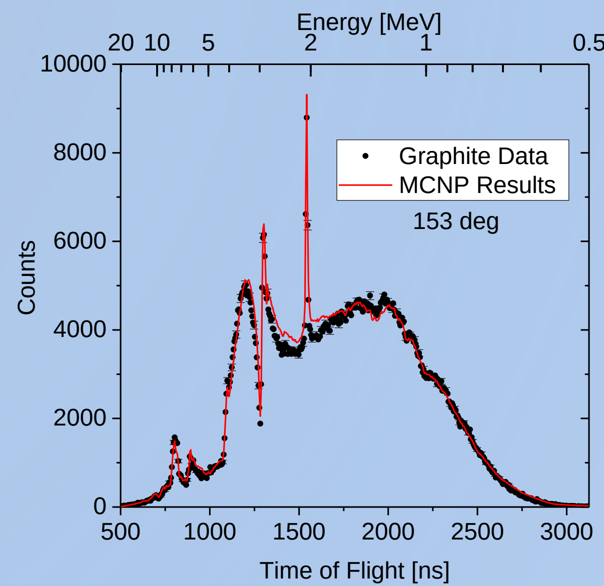
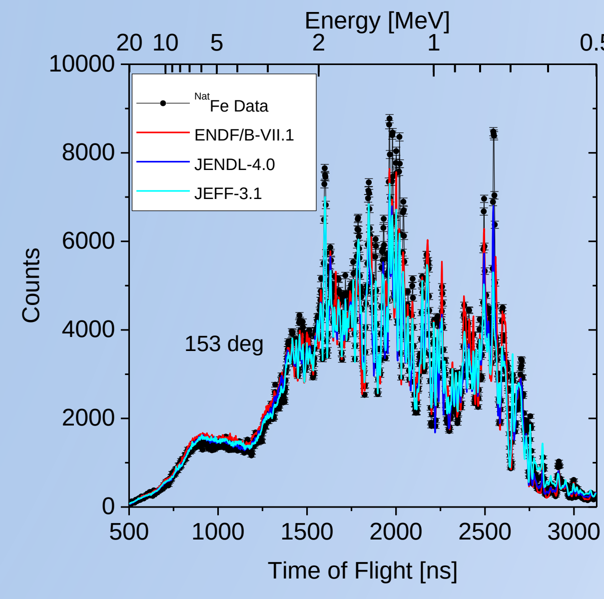
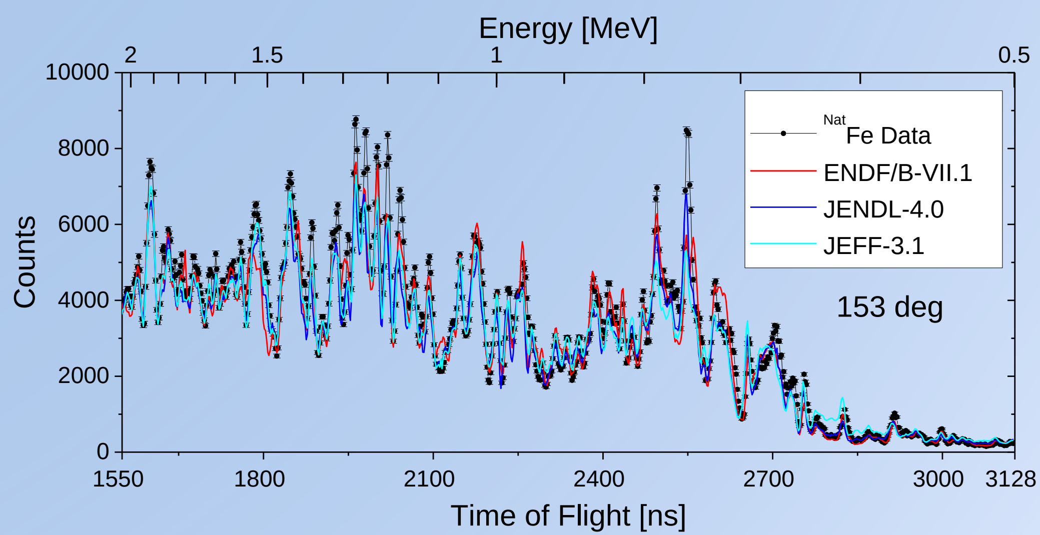
^{nat}Fe Scattering - 153°

Reference	FOM
Graphite	1.94

Library	FOM
ENDF/B-VII.1	14.42
JEFF-3.1	20.31
JENDL-4.0	14.68







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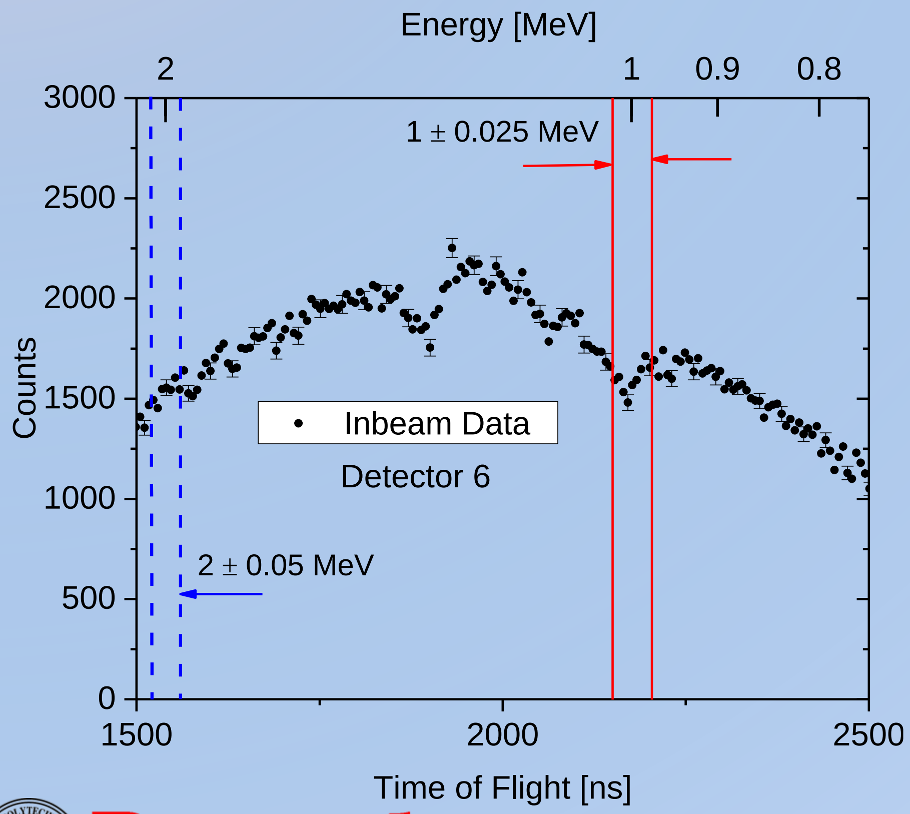
Observations for ^{nat}Fe

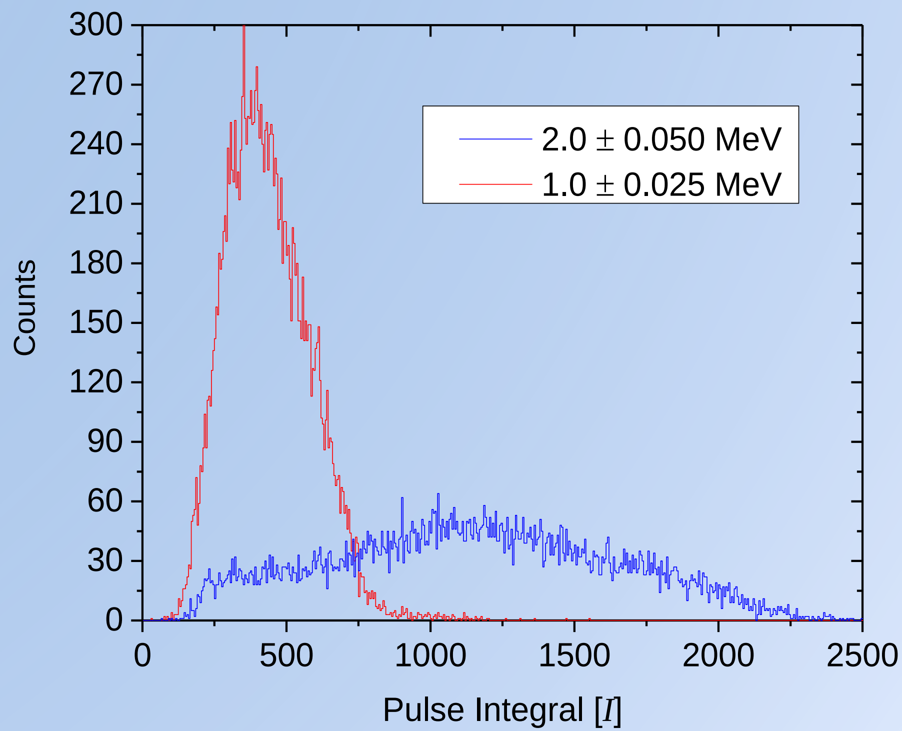
- The JENDL-4.0 evaluation had best overall agreement with experimental data from 0.5 to 20 MeV for all angles.
- Experimental data can be analyzed further to provide:
  - Inelastic to Elastic Scattering Ratios
  - Elastic (only) Scattering Contribution

Inelastic to Elastic scattering Ratio

EJ-301 Response Functions

- Detector in-beam measurements were used to develop response functions for energies $0.4 < E(t) < 2.0$ MeV





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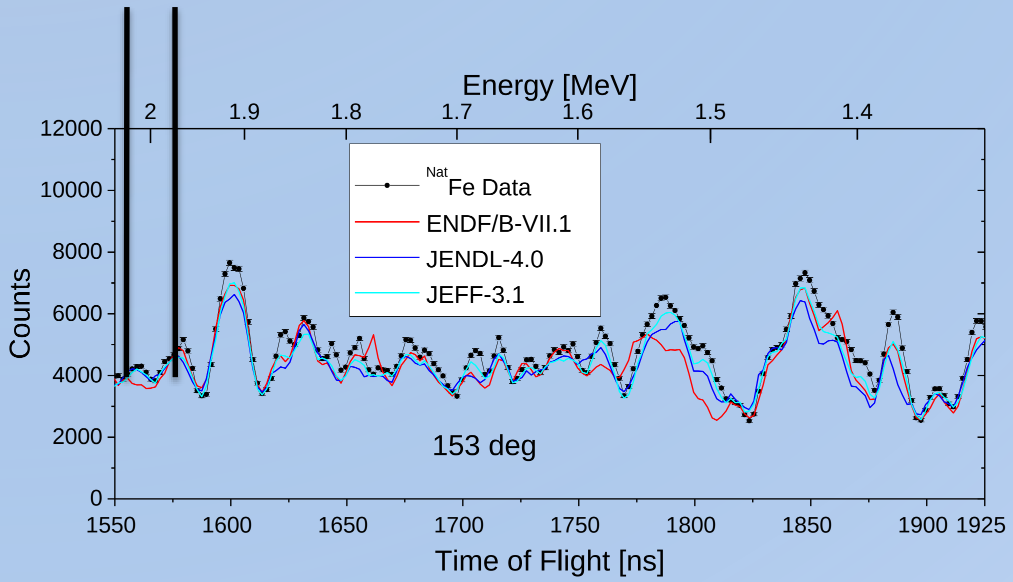
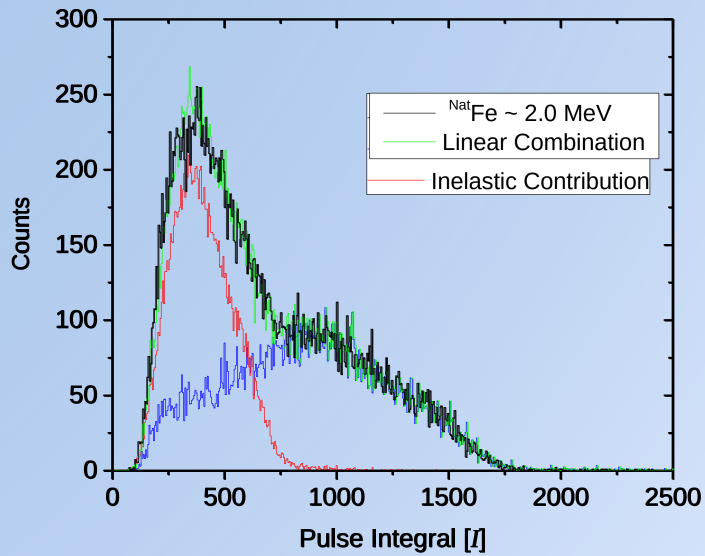
Inelastic to Elastic Ratio ^{nat}Fe

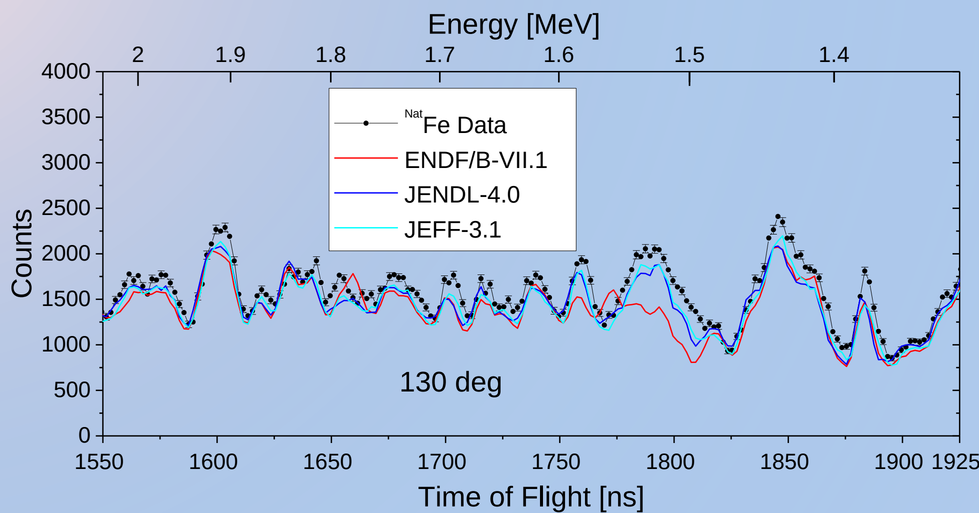
- Select an energy region (shown between the two black vertical)
- Fit in-beam response functions, $f_{el}(I)$ and $f_{inl}(I)$, to known levels

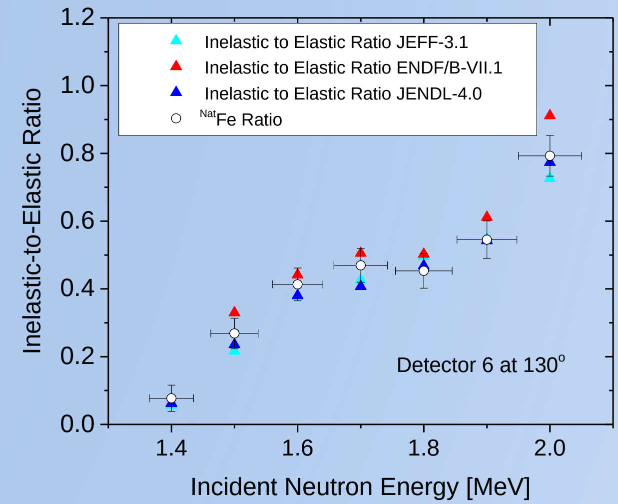
$$R(I) = A \cdot f_{el}(I) + (1 - A) \cdot f_{inl}(I) \qquad \text{Ratio} = \frac{(1 - A)}{A} \qquad A - \text{Fitted elastic fraction}$$

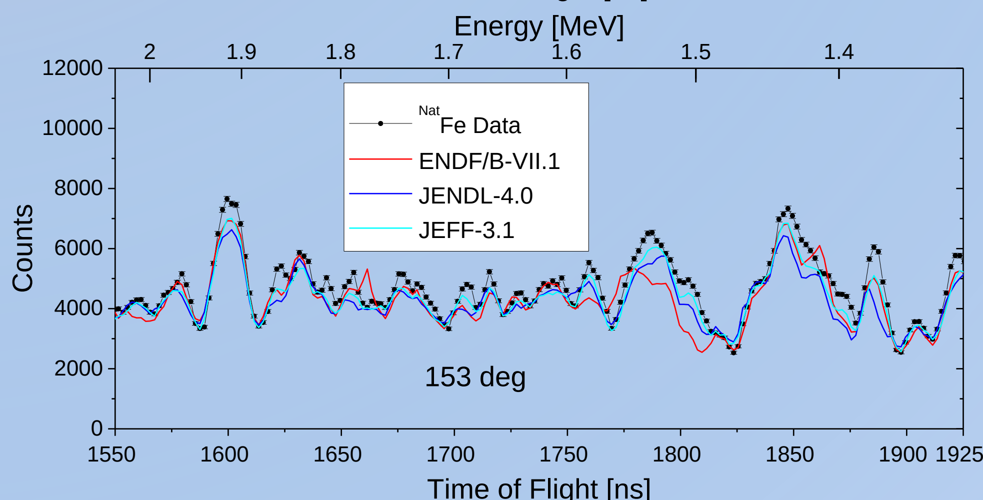


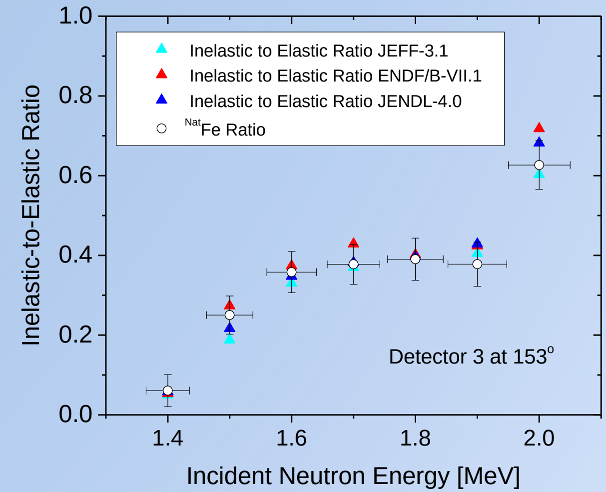


Inelastic to Elastic Ratio









- Multiple scattering effects included in MCNP simulations
- Statistical and systematic uncertainties included in analysis

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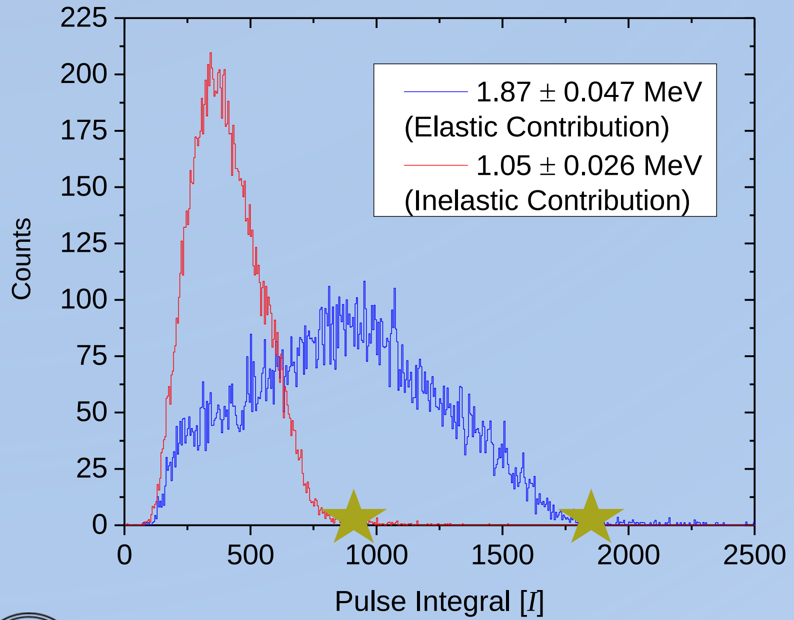
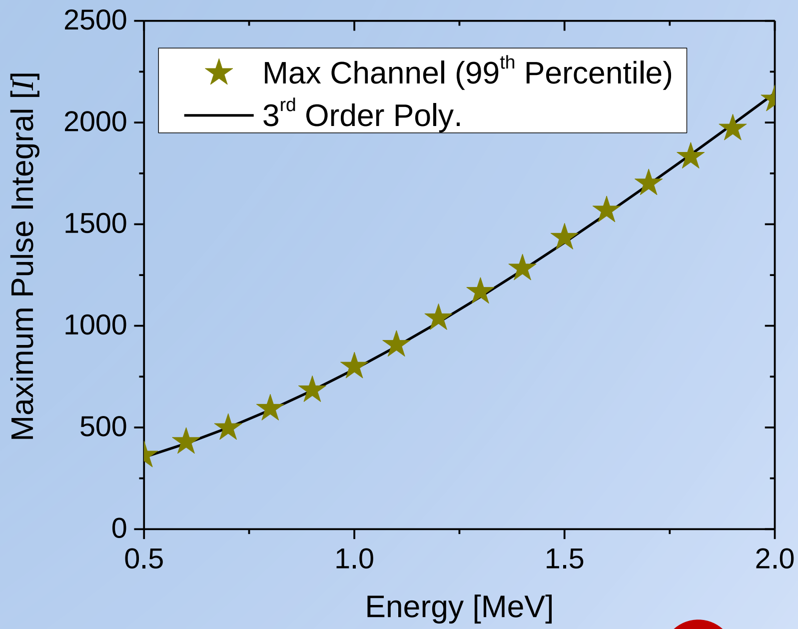
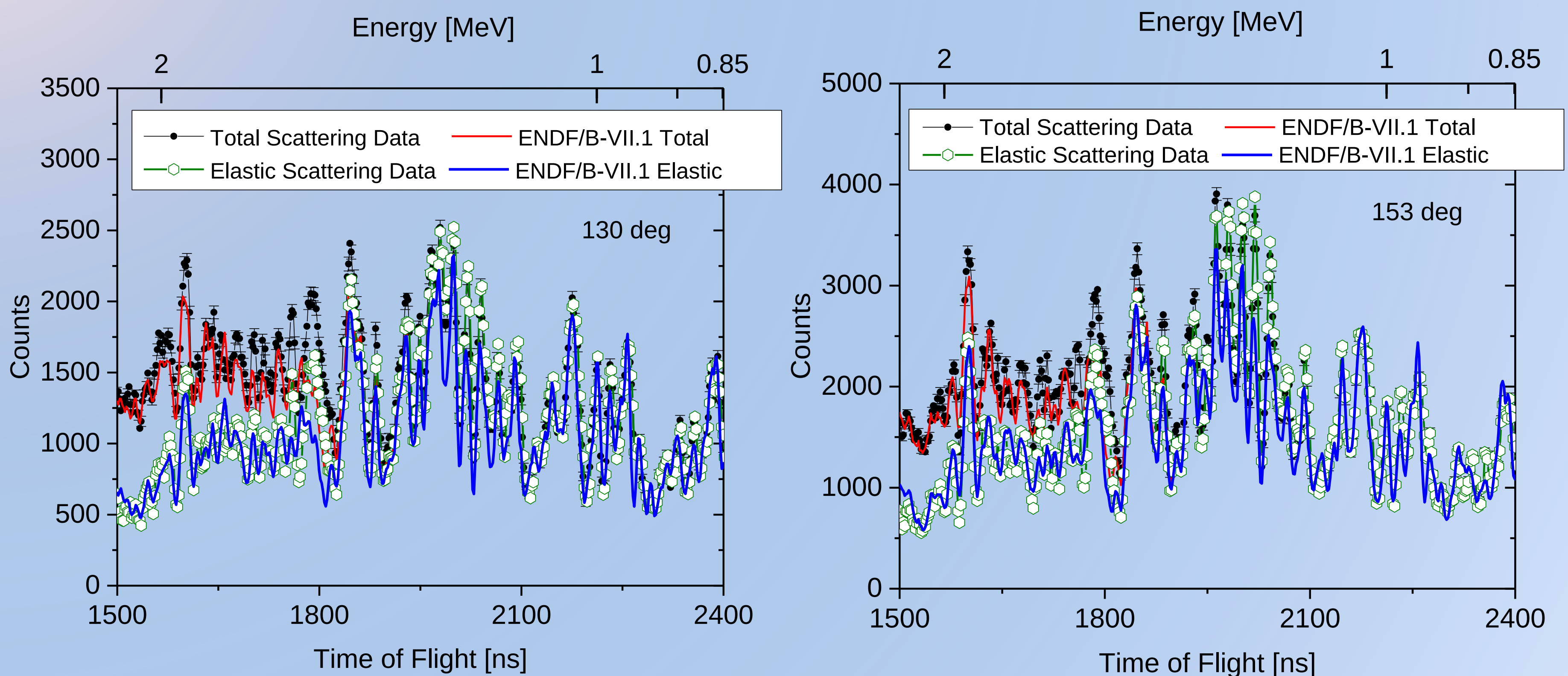
Elastic Scattering Contribution

- The response function fitted method does not easily extend to high energy resolution data and a new method was developed
- The goal is to isolate only the elastic scattering:
  - Cut pulses with integral less than the discrimination.
  - Correct for the elastic shape that was discriminated.
  - Method is insensitive to inelastic to elastic ratios.





Elastic Scattering vs MCNP simulation



- Elastic scattering can be measured from 0.5 to 2.0 MeV
- Only elastic scattering contribution measured and simulated
- Collaborating with ORNL to improve new ^{56}Fe evaluation

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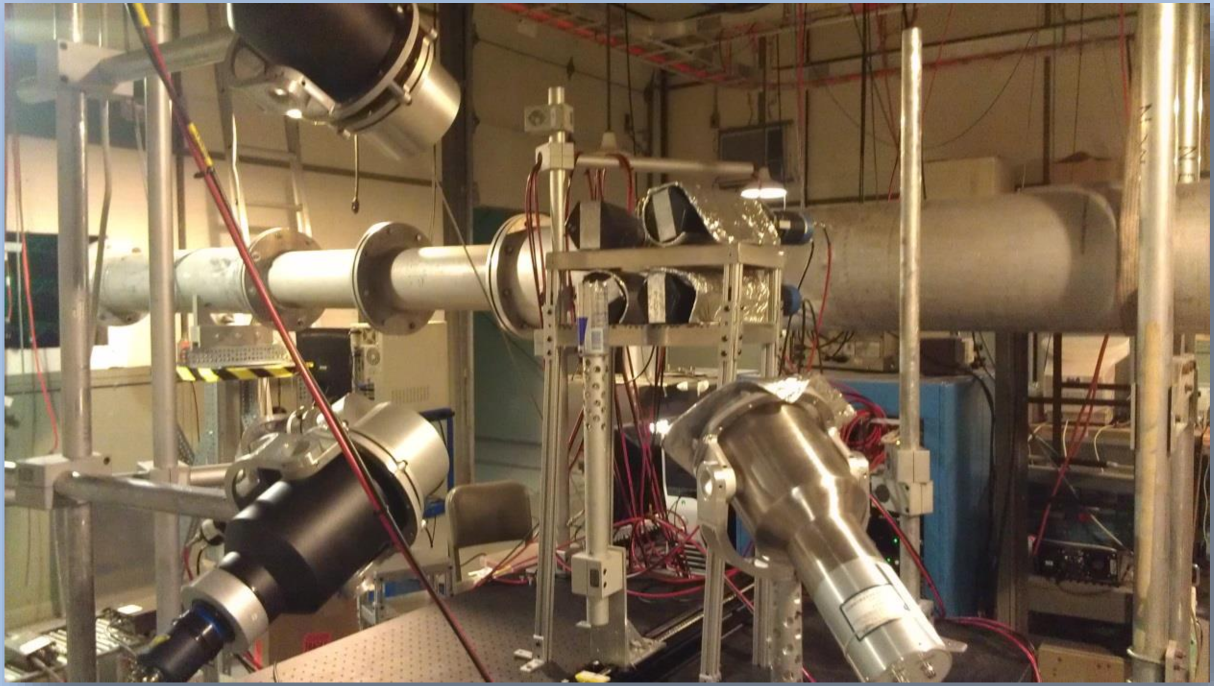
Prompt Fission Neutron Spectra



The Gamma Tagging Method

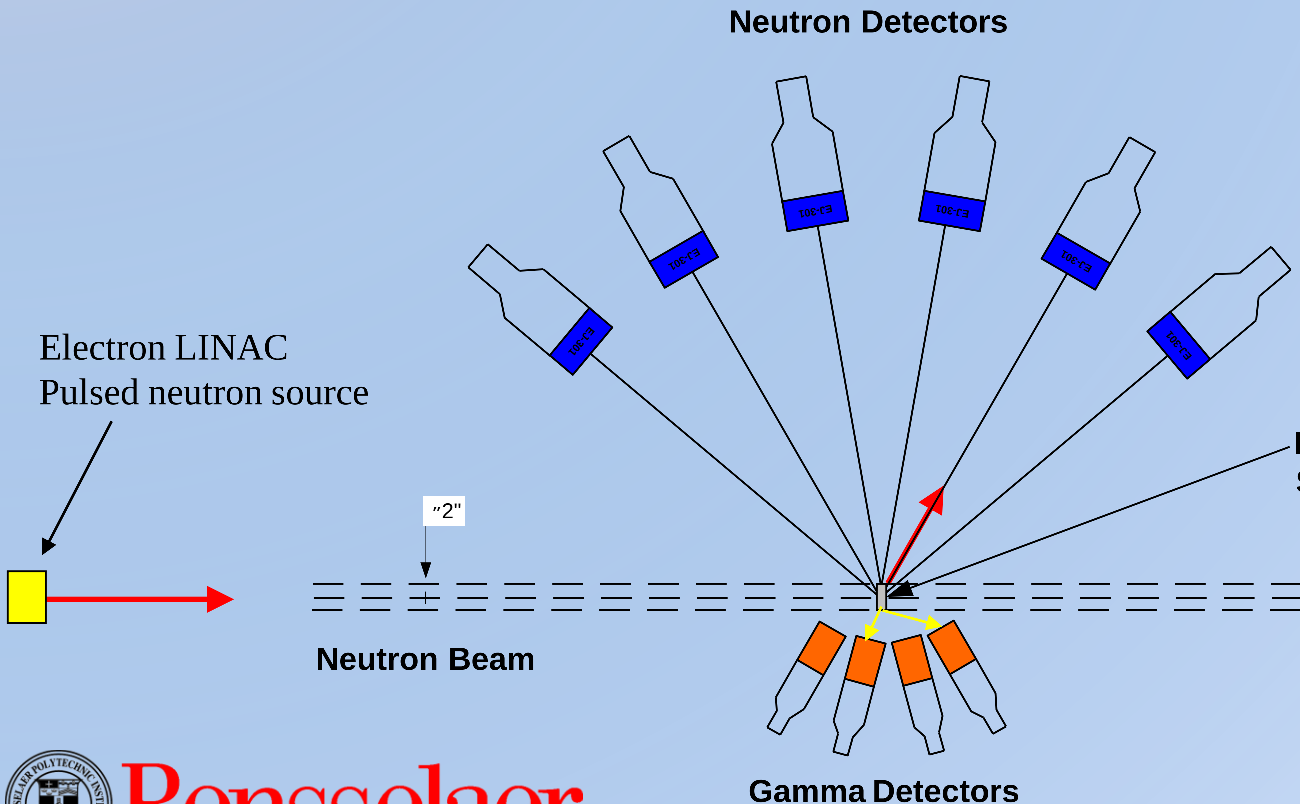
- Use the double TOF method
- Use a gamma tag for fission (instead of traditional fission chamber)
- Use a combination of Liquid Scintillators and Li-Glass neutron detectors

$$ToF = \frac{kL_1}{\sqrt{E_1}} + \frac{kL_2}{\sqrt{E_2}}$$

$$k \approx 72.3 \frac{\mu s \, eV^{1/2}}{m}$$

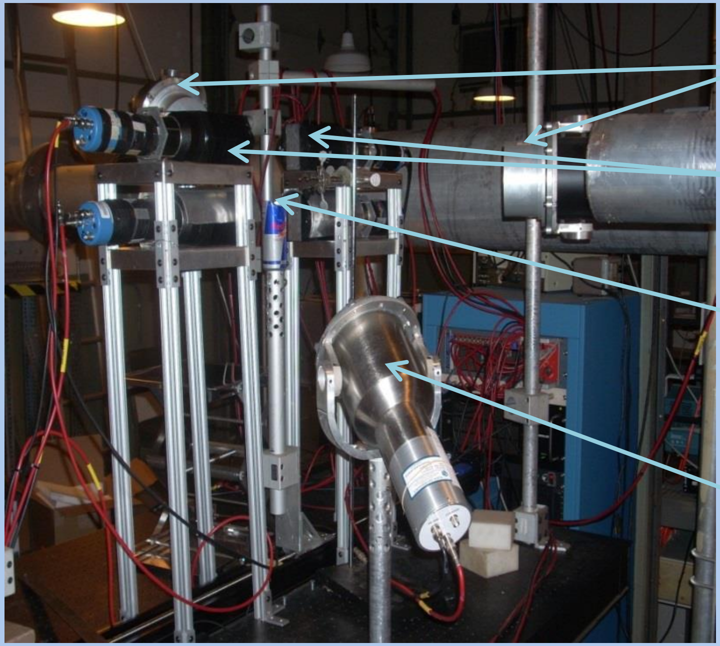
$L_{1,2}$ – flightpath distance

E_1, E_2 – incident, fission neutron energy



Experimental Setup

- Neutron Detectors
  - EJ-204 Plastic Scintillator
    - 0.5” x 5”
    - 47 cm away from center of sample
  - 2 EJ-301 Liquid Scintillators
    - 3” x 5”
    - 50 cm away from center of sample



- Gamma Detectors
  - 4 BaF_2 detectors on loan from ORNL
  - Hexagonal detectors 2” x 5”
  - 10 cm from center of sample
  - ¼” lead shield between detectors
    - Reducing scattering between detectors

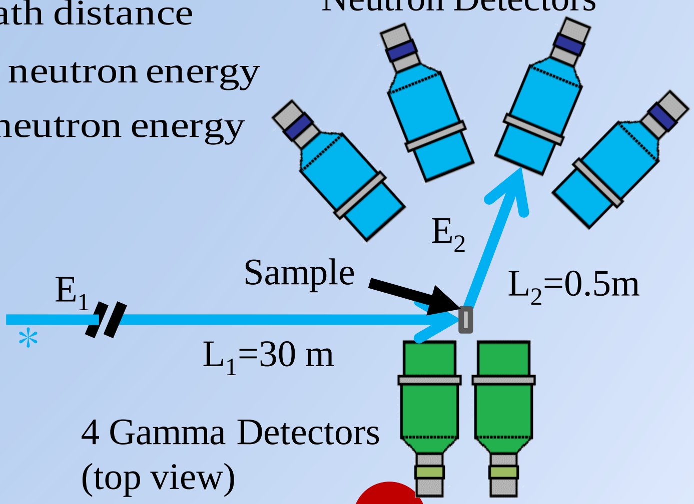
$$ToF = \frac{kL_1}{\sqrt{E_1}} + \frac{kL_2}{\sqrt{E_2}}$$

$$k \approx 72.3 \frac{\mu s\, eV^{1/2}}{m}$$

$L_{1,2}$ – flightpath distance

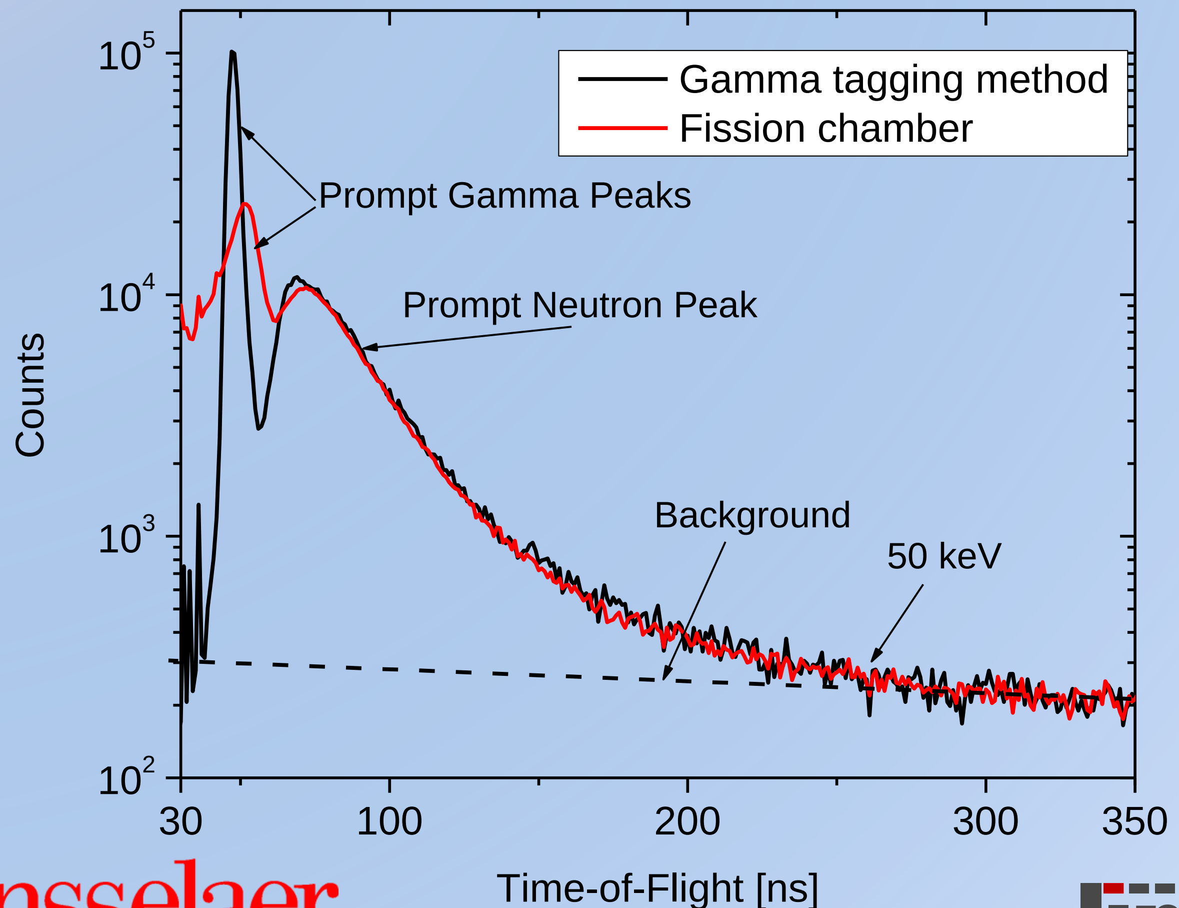
E_1 – incident neutron energy

E_2 – fission neutron energy

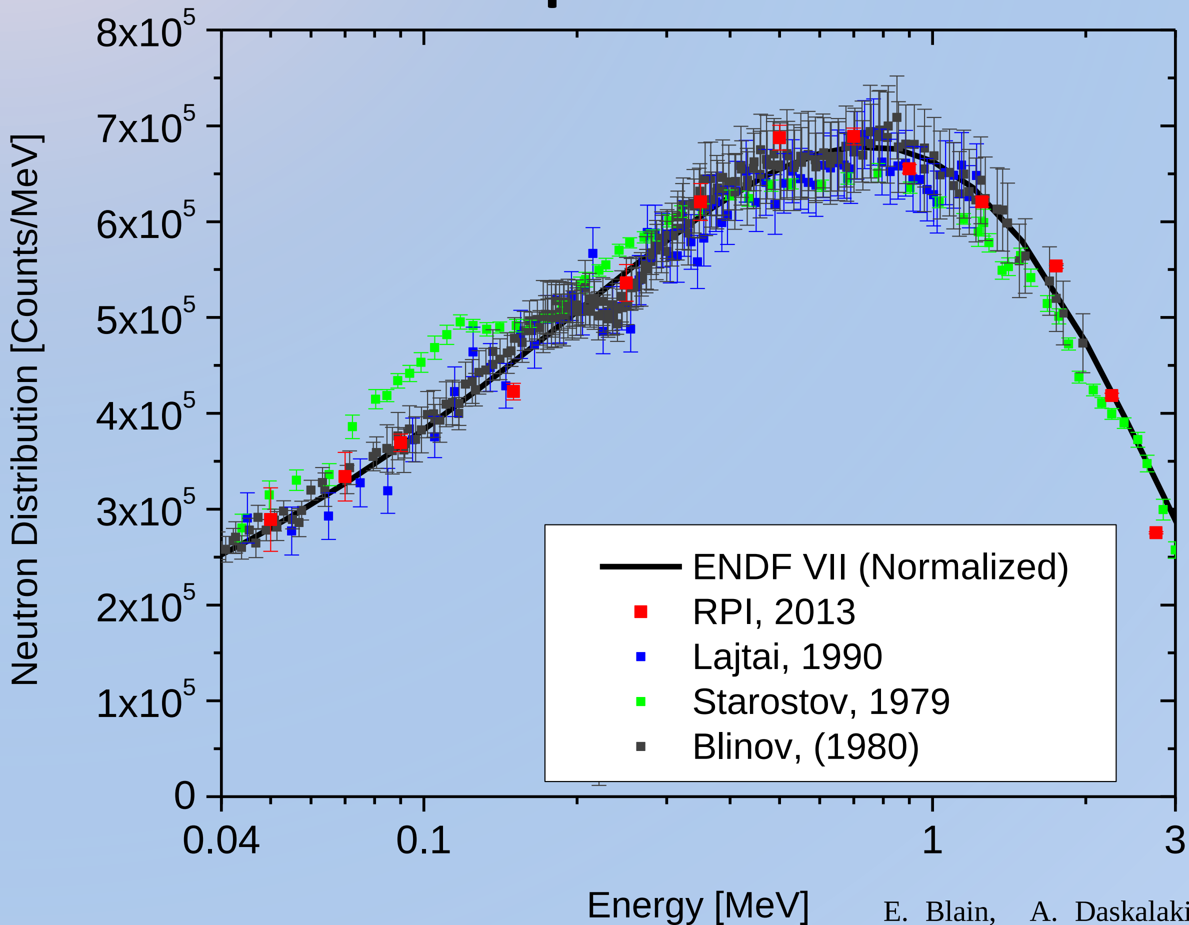


Gamma Tagging - EJ-204

- Gamma tagging method corrected for 30% detection efficiency compared to 83% detection efficiency with fission chamber



^{252}Cf Prompt Fission Neutron Spectrum Low Energy

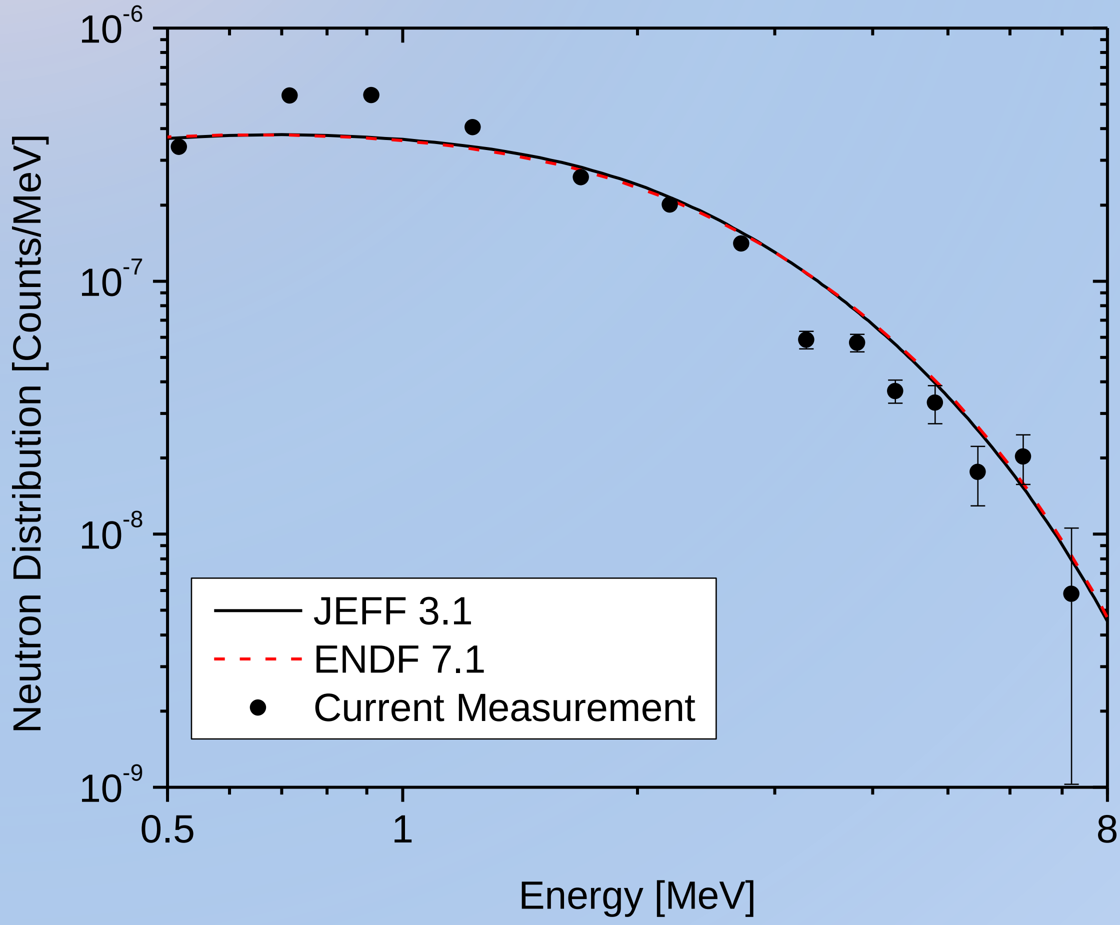


- Low energy data taken with 0.5” EJ-204 plastic scintillator
- RPI data show good agreement to Lajtai, Blinov data and ENDF evaluation
- Thin plastic detector allows for measurement down to 50 keV
- Gamma tagging method accurately reproduces PFNS for ^{252}Cf

E. Blain, A. Daskalakis, and Y. Danon, ”Measurement of Fission Neutron Spectrum and Multiplicity using a Gamma Tag Double Time-of-Flight Setup”, **invited talk**, International Conference on Nuclear Data for Science and Technology, New York, New York, March 4-8, 2013.

^{238}U Prompt Fission Neutron Spectrum High Energy

Preliminary Results



- Spectrum is normalized to ENDF/B-VII.1 at 1.2 MeV
- Spectrum is integrated over all incident time-of-flights
- Preliminary data show good agreement with current evaluations
- Increase near 1 MeV agrees with new data by Sardet et. al. (as presented in the FIESTA 2014 fission workshop at LANL)

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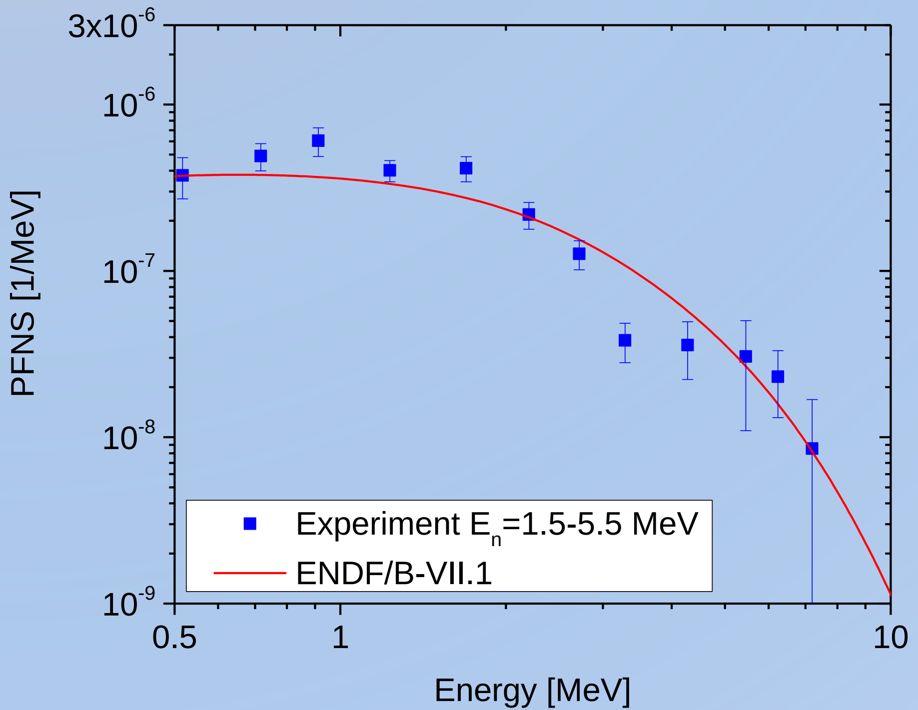
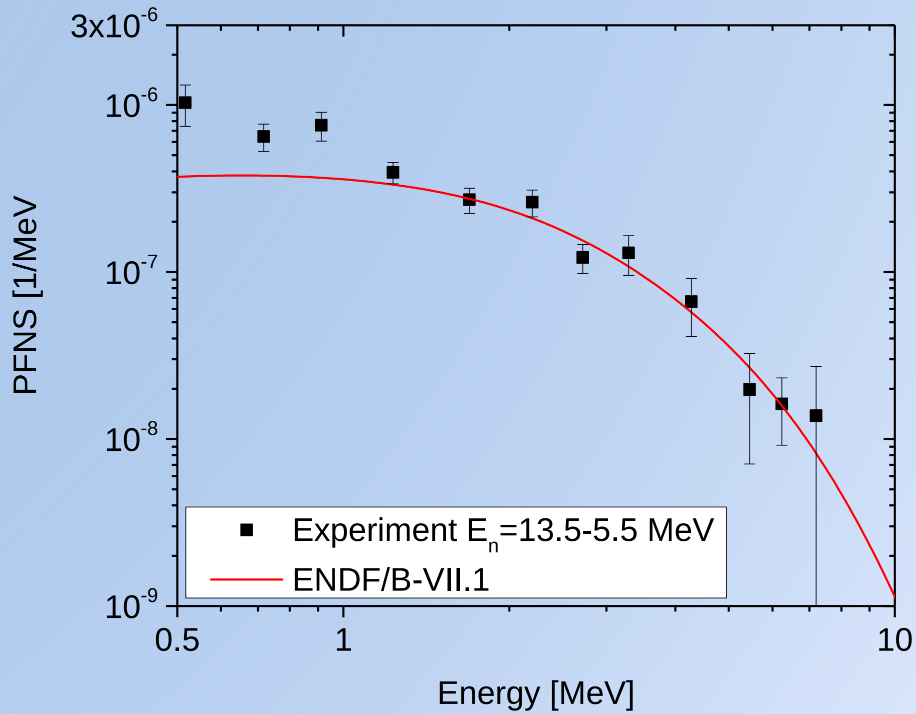
PFNS for two incident energy groups

- Data shown above and below the 2nd fission barrier
- Above the 2nd barrier – some indication of increased yield below 1 MeV





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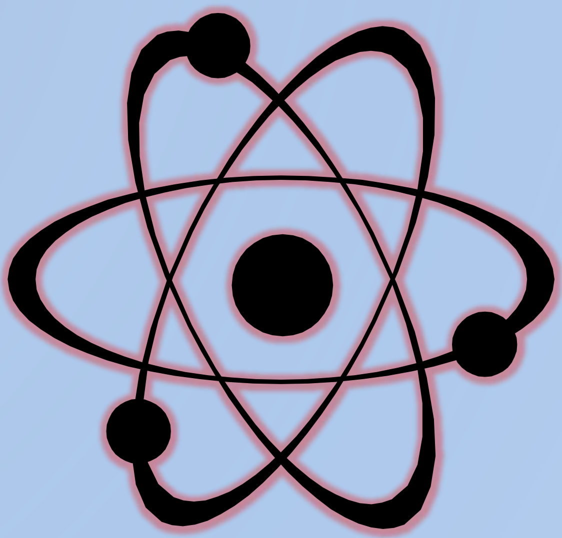
Summary

- Neutron scattering in the energy range from 0.5-20 was measured for Fe and ^{238}U at several scattering angles.
  - Data and MCNP simulation were used as benchmark for cross section and angular distribution evaluations.
  - Based on FOM the ^{238}U and ^{nat}Fe data is in best agreement with the JENDL-4 evaluations.
  - Inelastic to elastic scattering ratio were obtained for the 1st excited state and compared with evaluations. All evaluations agree with experimental data within 1-2 times the experimental uncertainty.
- Prompt fission yields were measured using the gamma tag method
  - ^{252}Cf measured fission neutron spectrum below 1 MeV is in good agreement with evaluations
  - Experimental results for ^{238}U provide some information about the change in the PFNS as a function of energy.



Thank You