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

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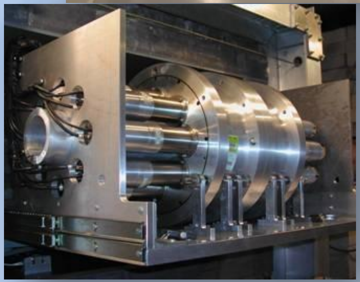
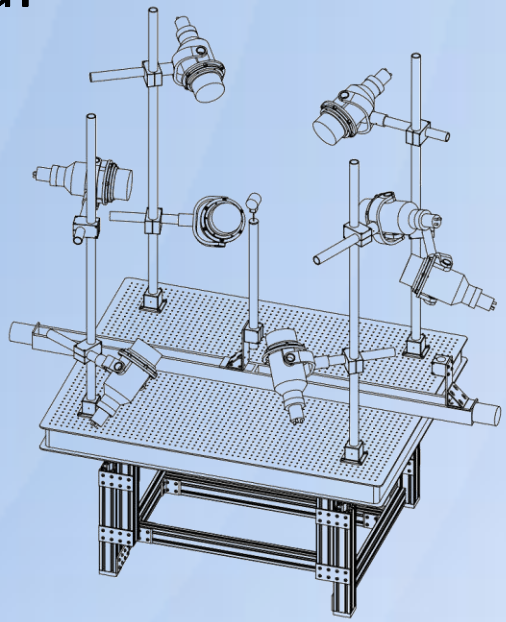
The Nuclear Data Program at the Gaerttner LINAC Center

- Driven by a 60 MeV pulsed electron LINAC ~$\mathbf{10^{12}}$ **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.



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

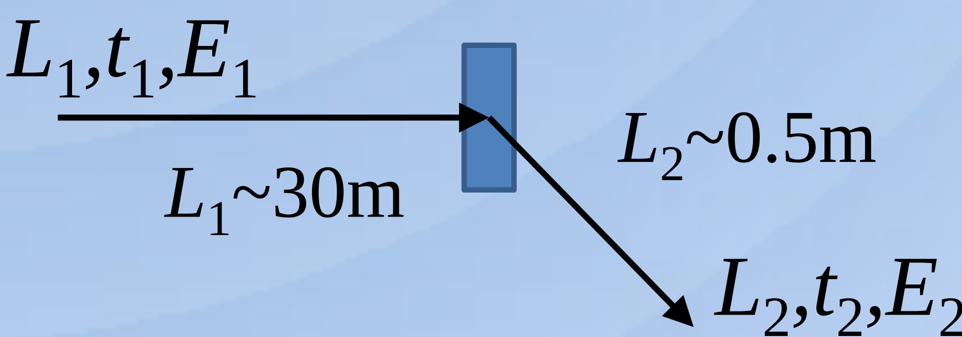


Methodology

- Measure the scattering yield at several angles around the sample.
  - Use TOF to measure the neutron incident energy
  - Use detectors that are insensitive to (capture/inelastic/background) gamma
- Compare the measurements to detailed simulations of the system with different cross section libraries
  - Characterize the incident neutron flux
  - Characterize the neutron detection efficiency
- Identify energy/angle regions where improvement is needed.

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 $L=L_1+L_2$



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

First Order Approximation of the Scattering Yield

Detector Efficiency

Probability to Interact

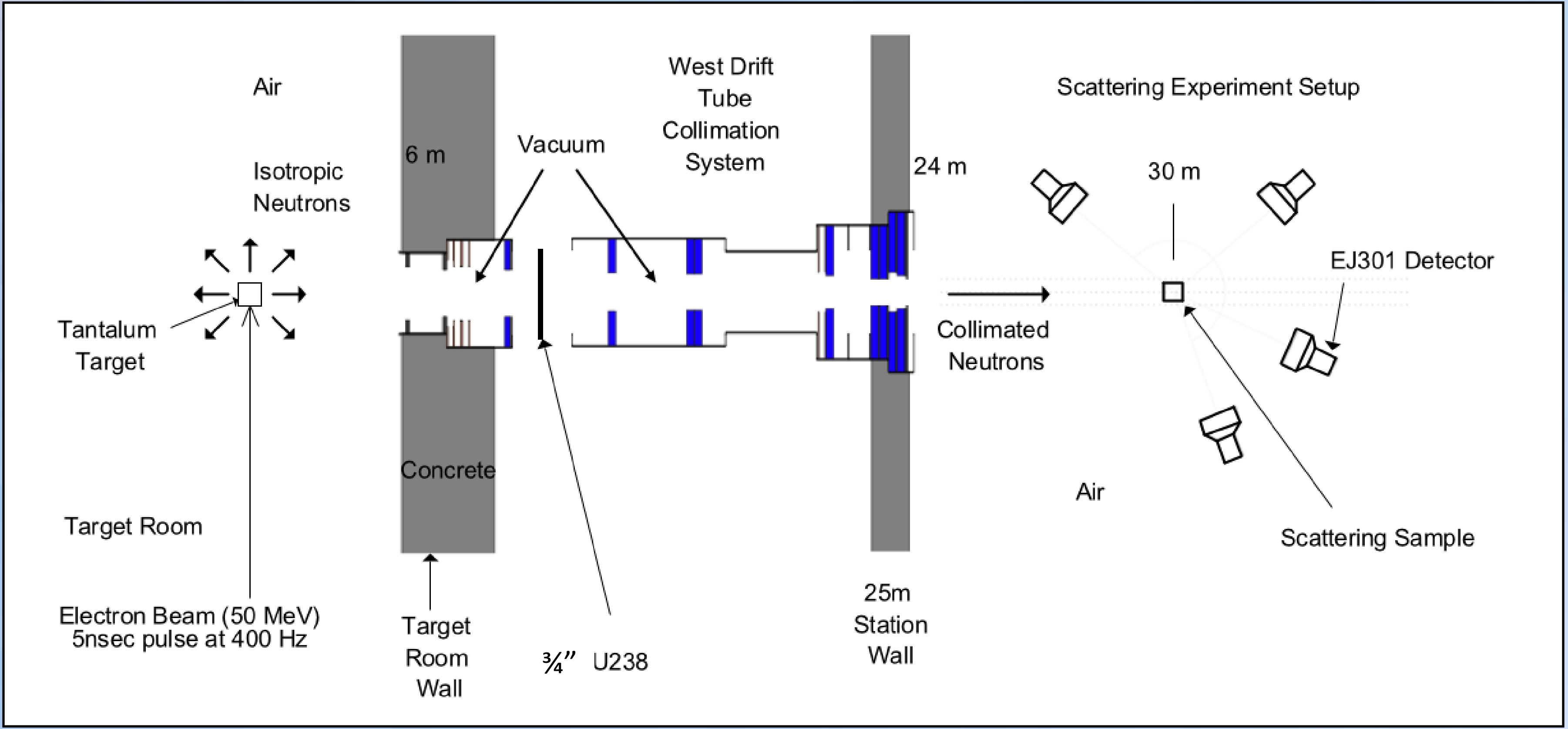
$$Y(E,\phi) \propto \eta(E')\Phi(E)\left(1-e^{\Sigma_T(E)L}\right)\frac{f(E,\phi)}{2\pi}$$

Incident Neutron Flux

Probability to Scatter in direction ϕ

In this approximation - multiple scattering is ignored

Experimental Setup Overview



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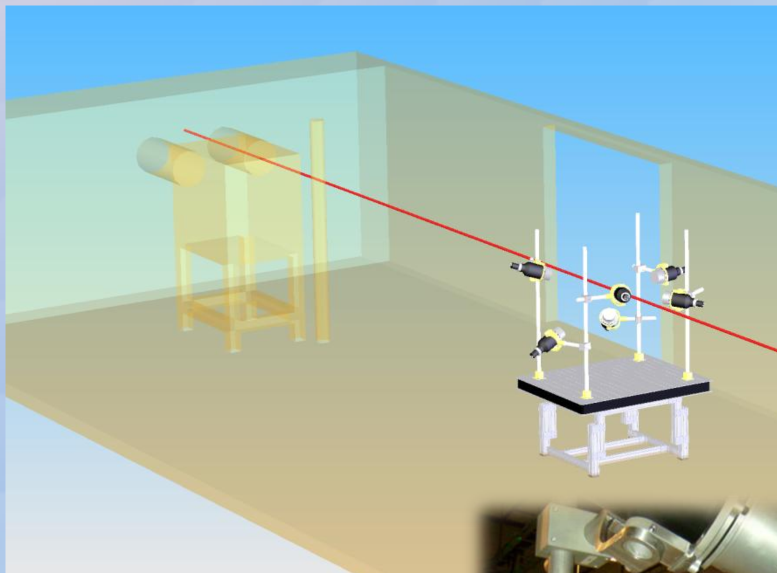
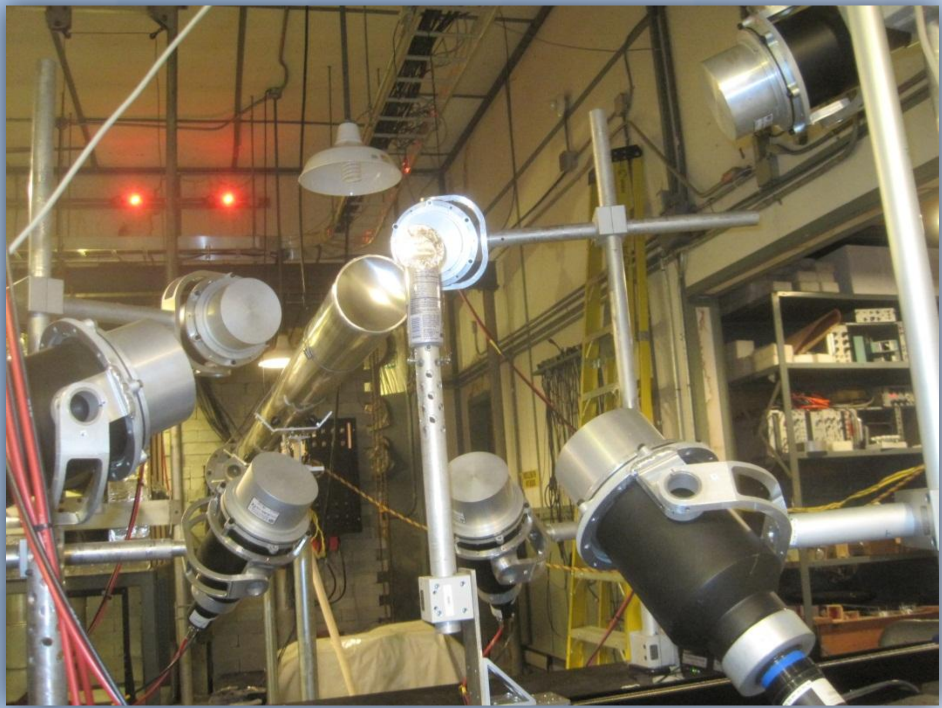
Scattering Detection System: Experimental Setup

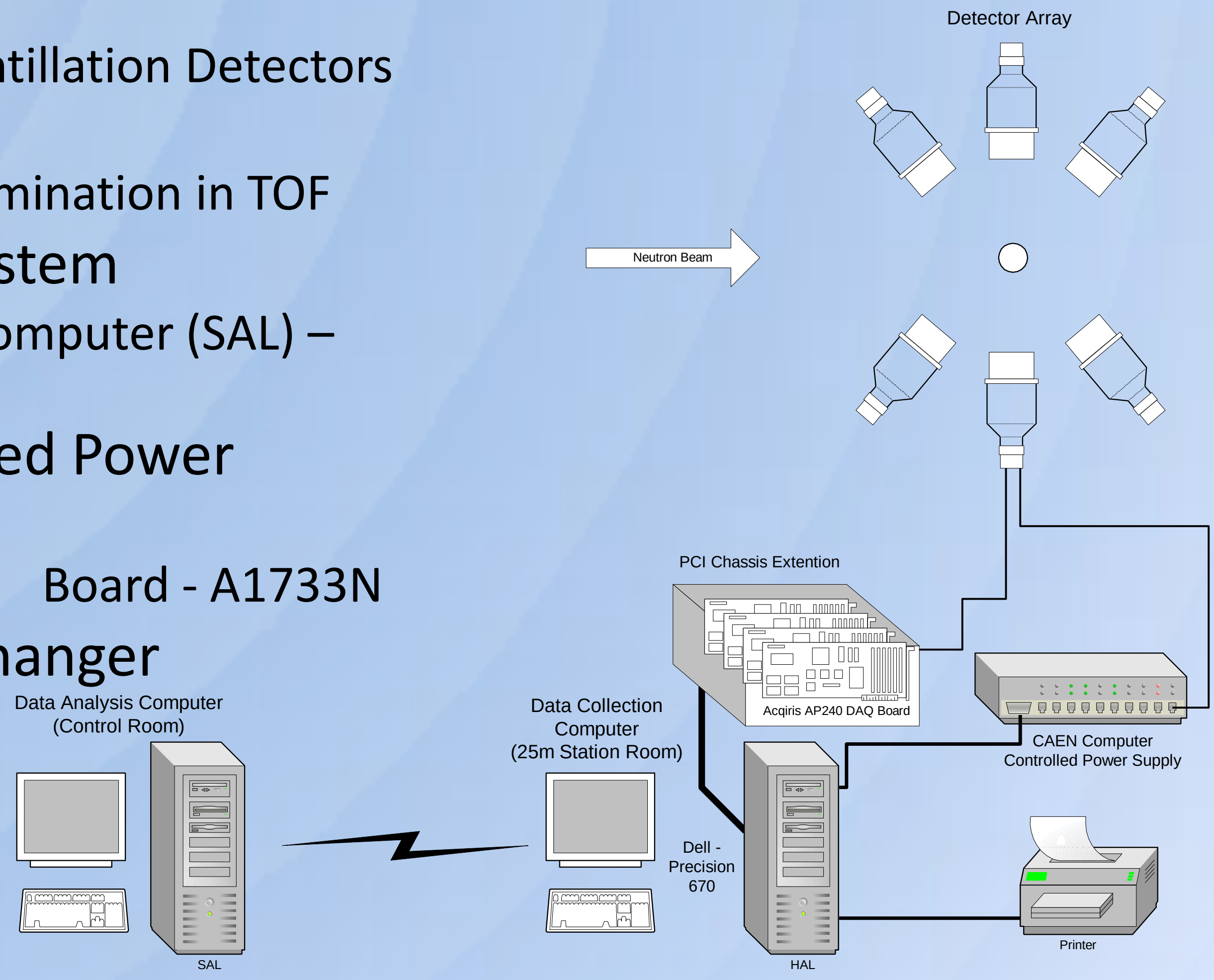




Low mass sample holder

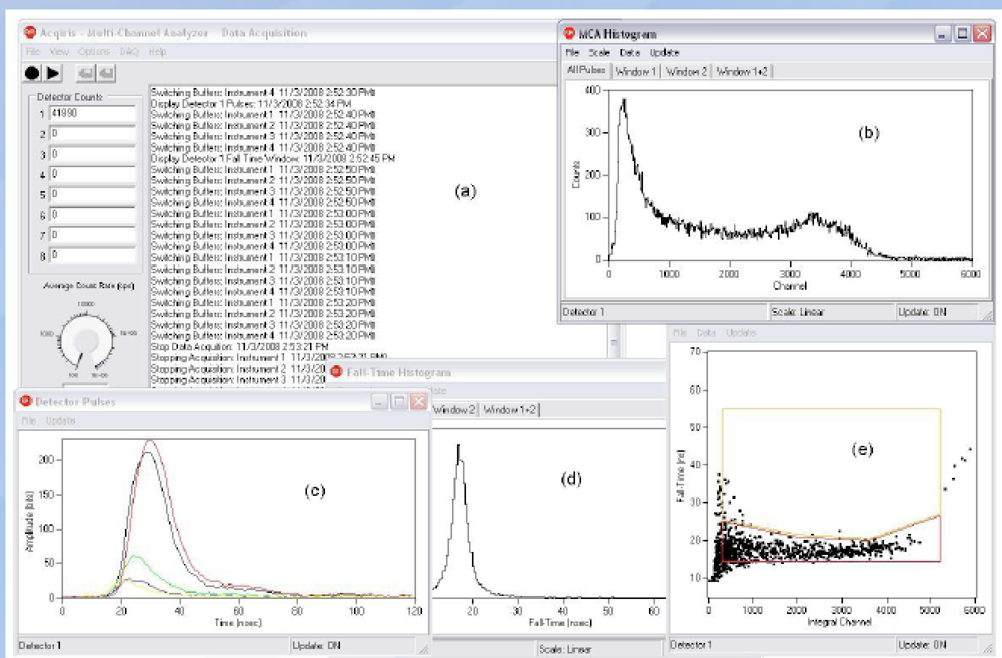
Scattering Detection System: Experimental Setup

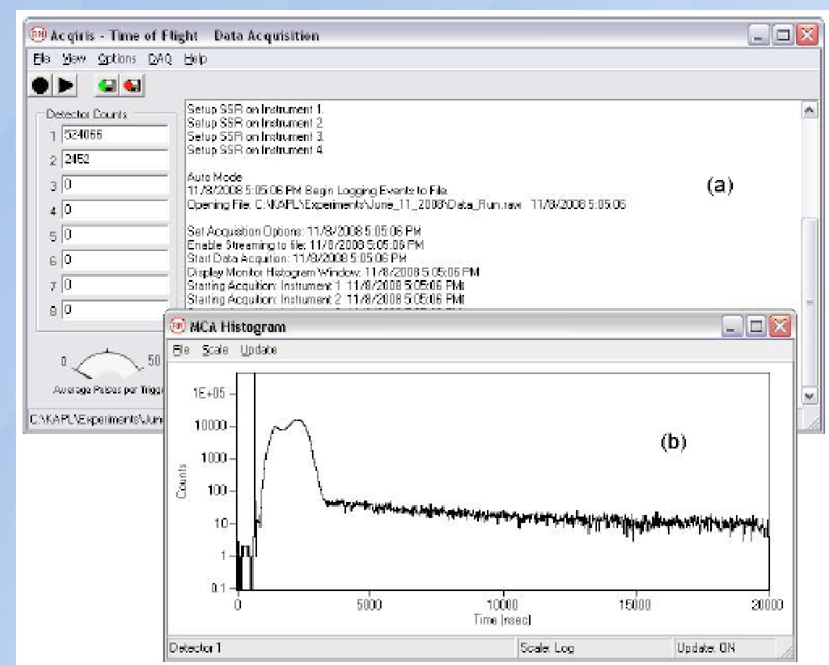
- Detector Array
  - 8 EJ301 Liquid Scintillation Detectors
  - 8 A/D channels
  - Pulse Shape discrimination in TOF
- Data Processing System
  - Data Processing Computer (SAL) – Control Room
- Computer Controlled Power Supply
  - Chassis - SY 3527 Board - A1733N
- Sample Holder / Changer



DAQ system

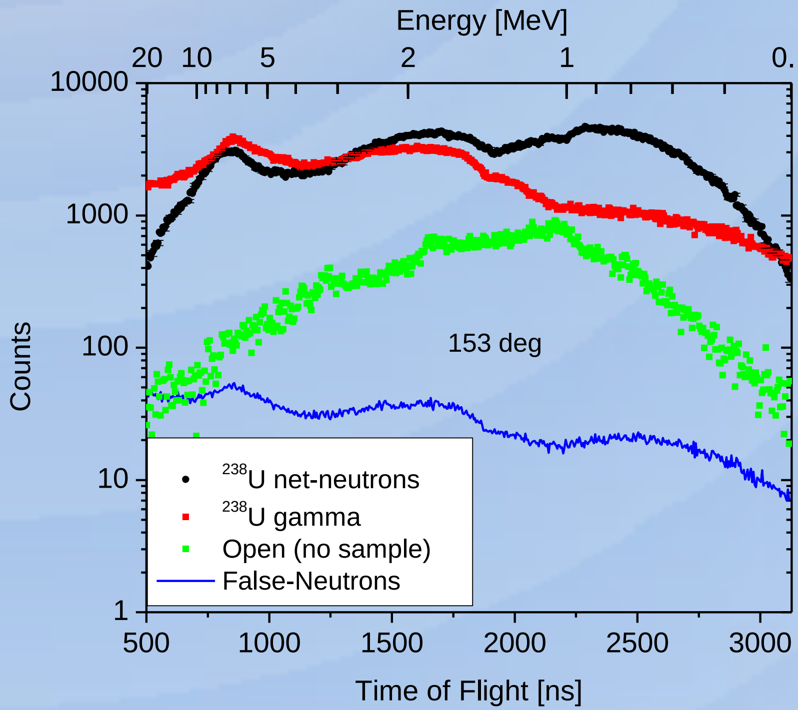
- All DAQ is automated using script based software running under Windows
- Alternate between sample, graphite and open (background) measurements
- Each position is measured for about 10 min
- Fission chamber monitors are used to normalize beam intensity fluctuations.
- Detector/system gain is periodically aligned using ^{22}Na

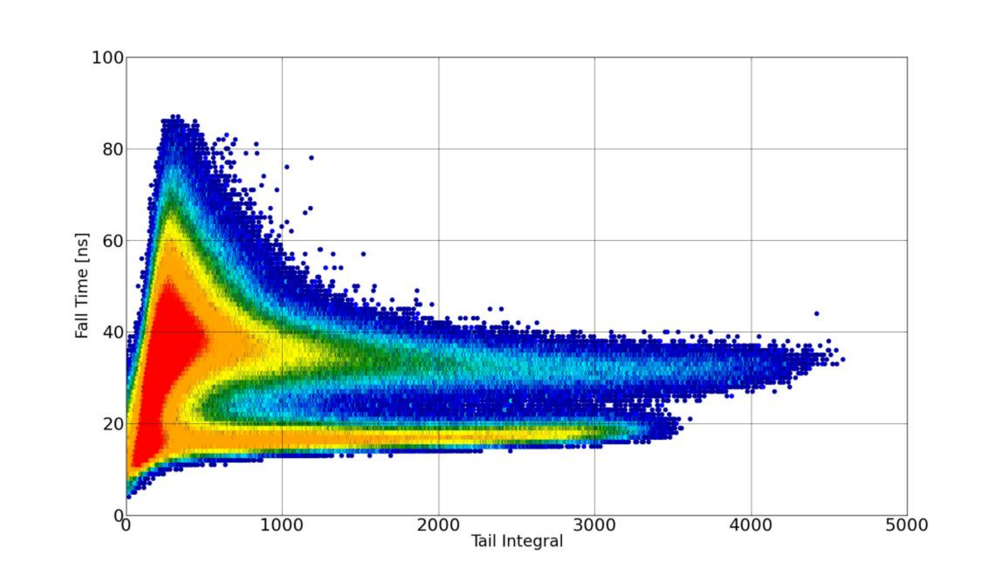




Neutron Gamma Separation with Pulse Shape Analysis

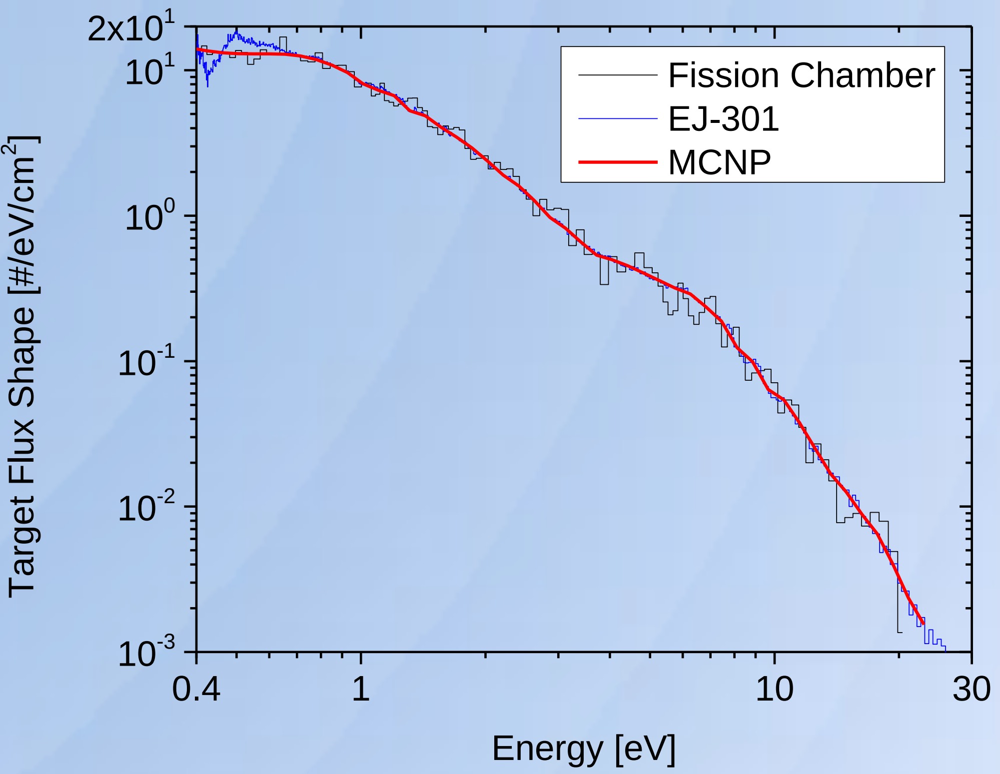
- Digitize 120 ns to get all the event-generated detector pulses
- Use pulse shape classification to discriminate gammas (~1-2% of gammas are recorded as neutrons)
- Developed a pulse rejection method to eliminate false neutrons.



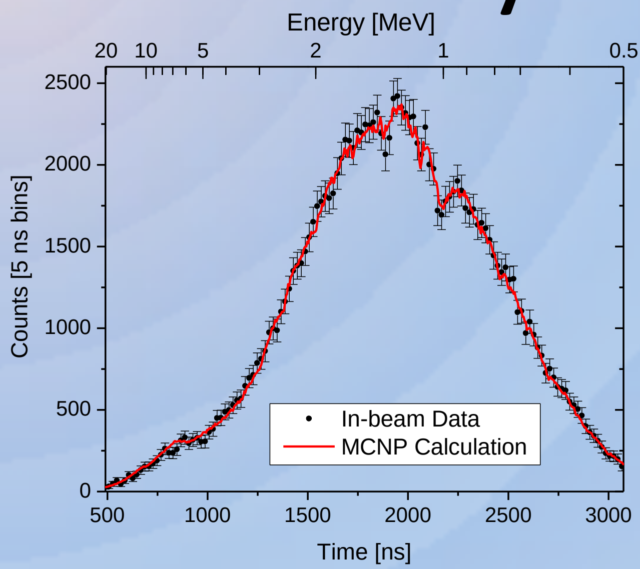


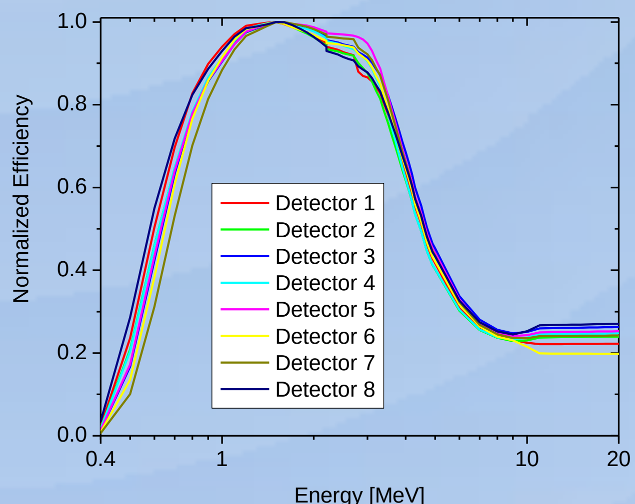
Flux Shape Measurement

- Use a fission chamber with ~391 mg ^{235}U in the sample position
- Use ENDF/B 7.1 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



Efficiency as a Function of Energy

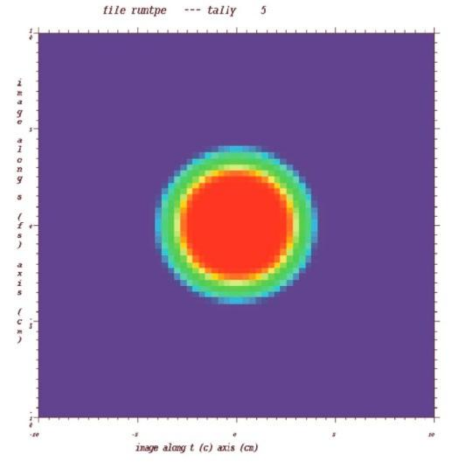


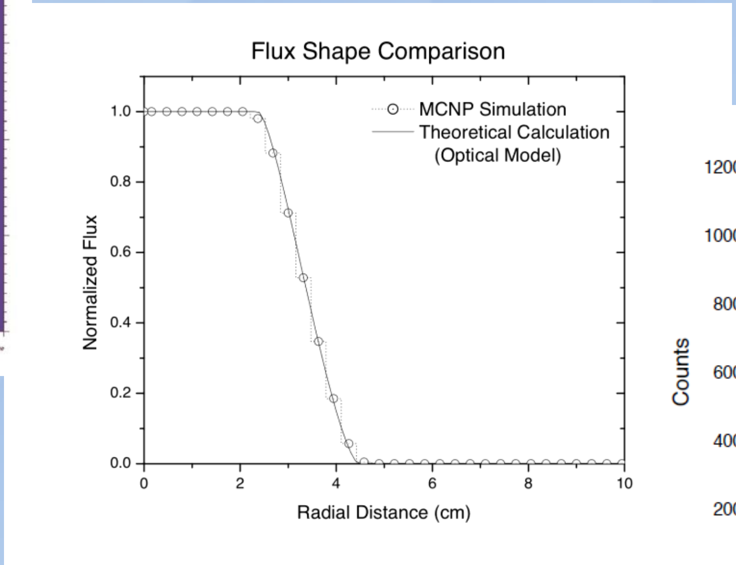


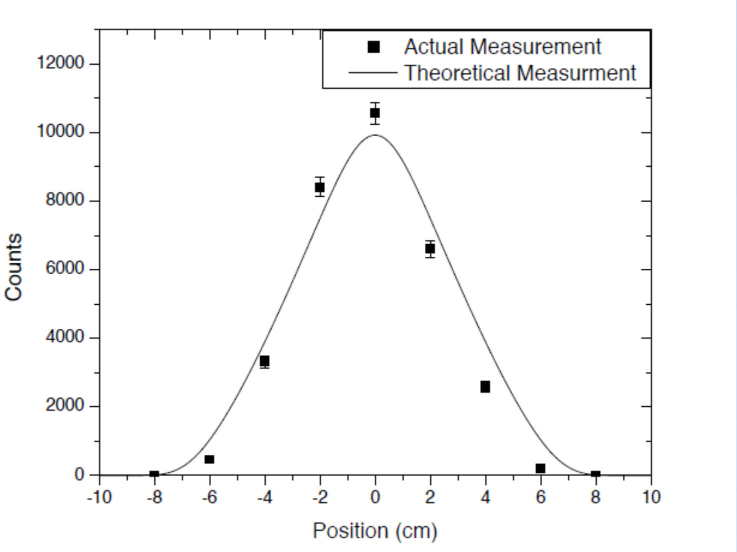
- Objective:
  - MCNP simulation of EJ301 response in the sample position must precisely agree with the measurement
- Methodology:
  - Use the measured flux as a source in MCNP simulation of the in-beam detector response
  - In MCNP set the detector efficiency η=1 (tally only the neutron flux shape)
  - 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.

Neutron Beam Collimation

- Characterize the collimation system
  - Ensure beam diameter agrees with sample diameter of 7.62 cm
  - Verify measurements and calculations agree







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

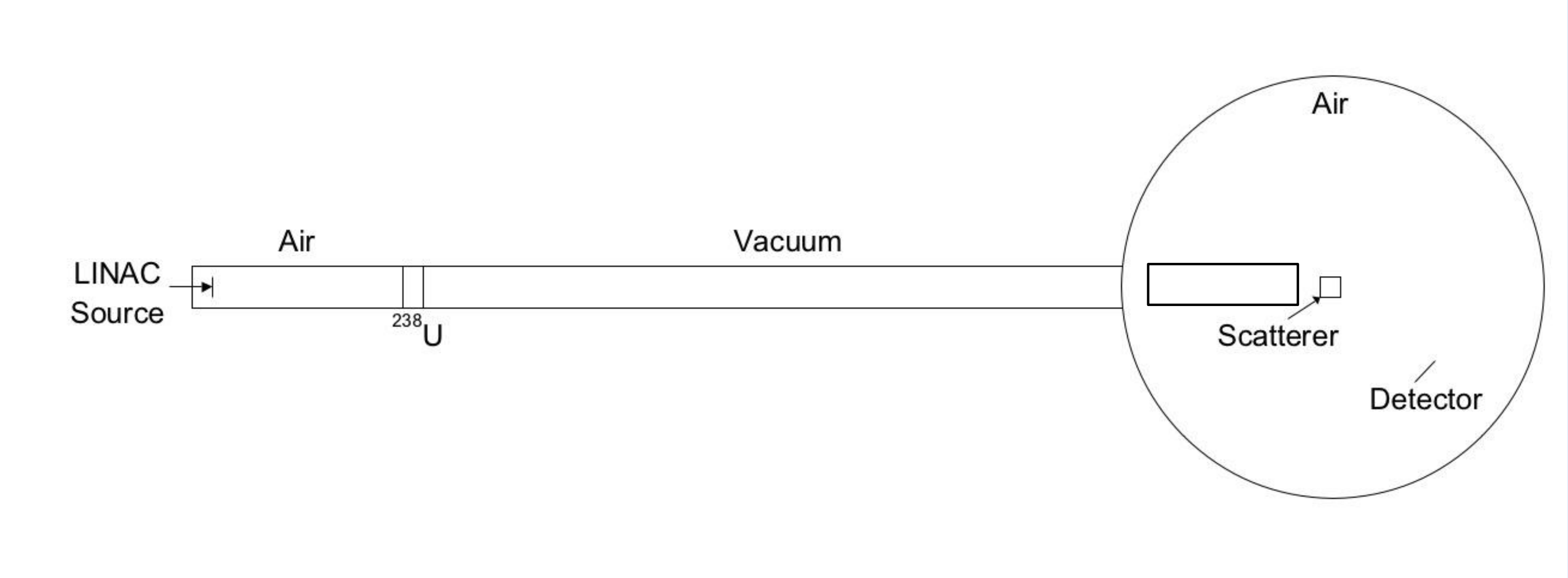
The false neutron correction: $$fn_i = \sum_{j=1}^{n_\gamma} f(A_j)$$

n_γ - Number of gammas in TOF channel *i*

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

MCNP Simulation Geometry

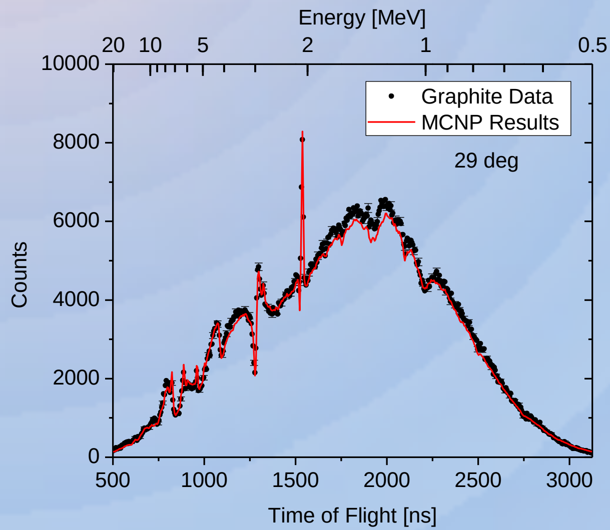
- Use ASAP (As Simple As Possible) approach
- Use array of point detector tally F5 to model the EJ301 detector
  - Convolute the tally with the detector efficiency
- Include ¾" Depleted U filter in the simulation
- Include windows (Al)
- Include recent improvements of vacuum tube near the sample

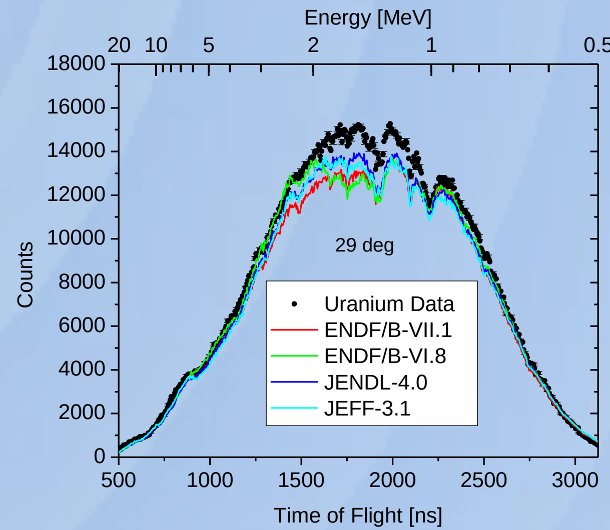


Data Analysis

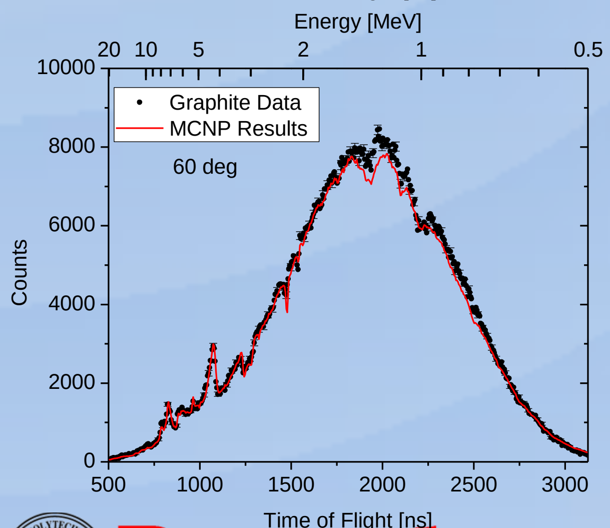
- Compare the shape (as a function of TOF) between the measurements and simulations
- Use graphite as a reference in all measurements
  - Differences between the measurement and simulation of graphite are considers systematic errors
- Measurements of Be, Mo and Zr
  - The efficiency was derived from SCINFUL calculations
  - Neutron flux shape based on a fit to in-beam measurements with EJ-301 and Li-Glass
  - Used **individual detector normalization** of the simulation to the experimental data based on graphite measurement
- For ^{238}U and ^{56}Fe experiment
  - Flux was derived from ^{235}U and EJ301 in-beam measurements
  - The efficiency was adjusted to match the MCNP calculations to the in-beam measured data
  - Use **one normalization** factor for all detectors (global normalization)

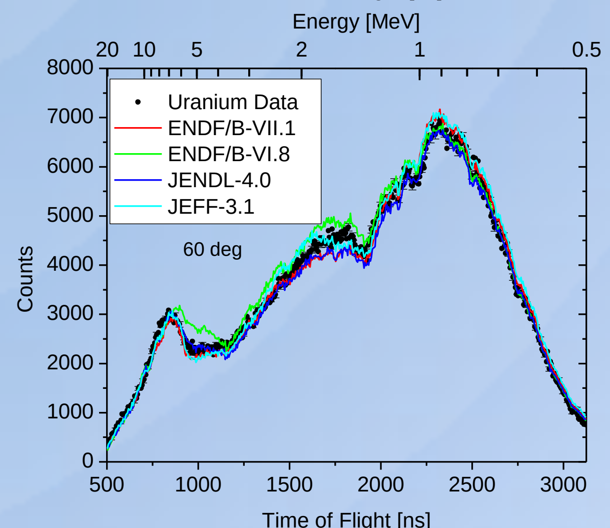
^{238}U Scattering - Forward Angles





Library	χ^2
ENDF/B-VII.1	4.4
ENDF/B-VI.8	2.7
JENDL-4.0	2.5
JEFF-3.1	3.3





Library	χ^2
ENDF/B-VII.1	1.8
ENDF/B-VI.8	3.7
JENDL-4.0	1.3
JEFF-3.1	2.2

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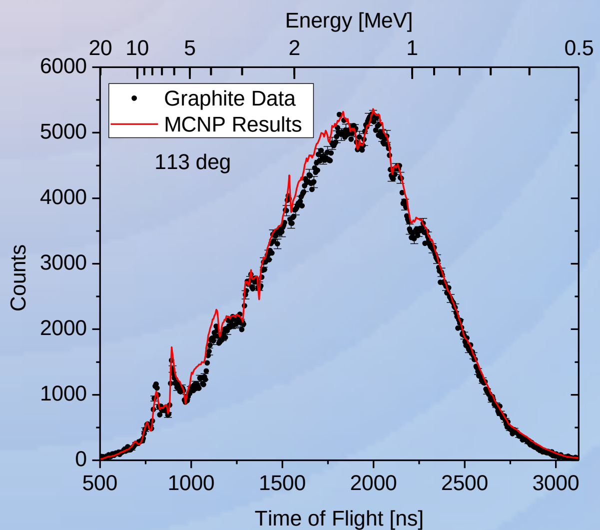
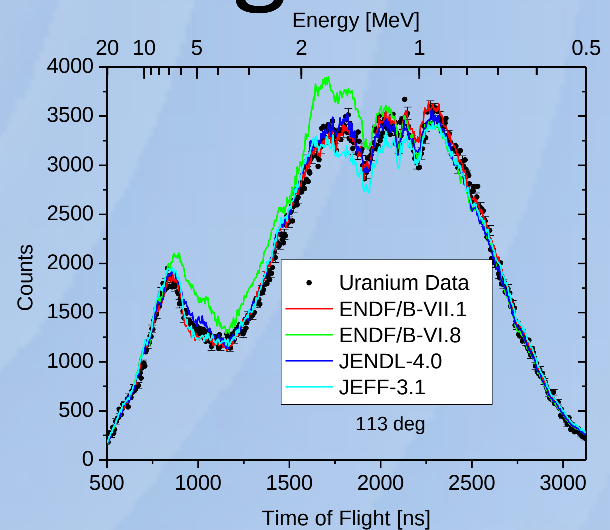
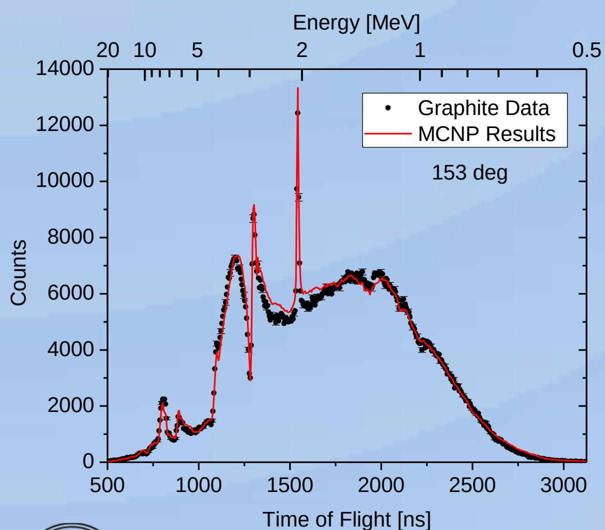
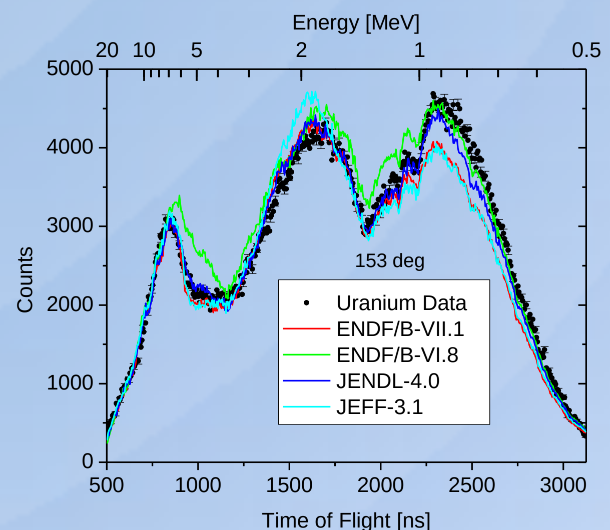
^{238}U Scattering – Back Angles





Library	χ^2
ENDF/B-VII.1	0.6
ENDF/B-VI.8	4.0
JENDL-4.0	0.8
JEFF-3.1	1.4





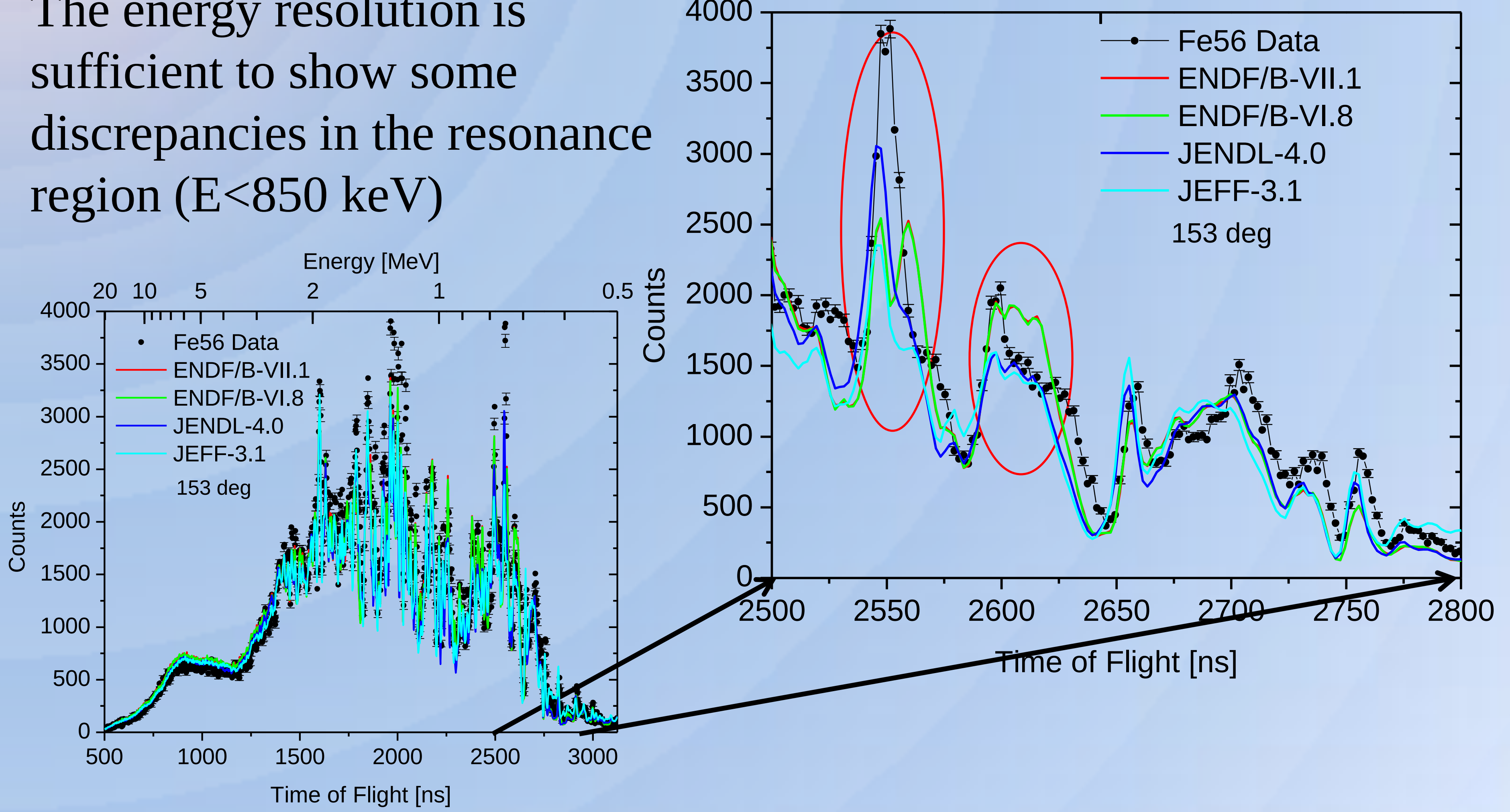
Library	χ^2
ENDF/B-VII.1	3.7
ENDF/B-VI.8	4.6
JENDL-4.0	1.2
JEFF-3.1	4.7

Observations for ^{238}U

- Differences between the evaluations are visible and exceed the experimental errors.
- To get better agreement the evaluations need to be adjusted mostly at back angles.
- Overall JENDL 4.0 has has the lowest χ^2

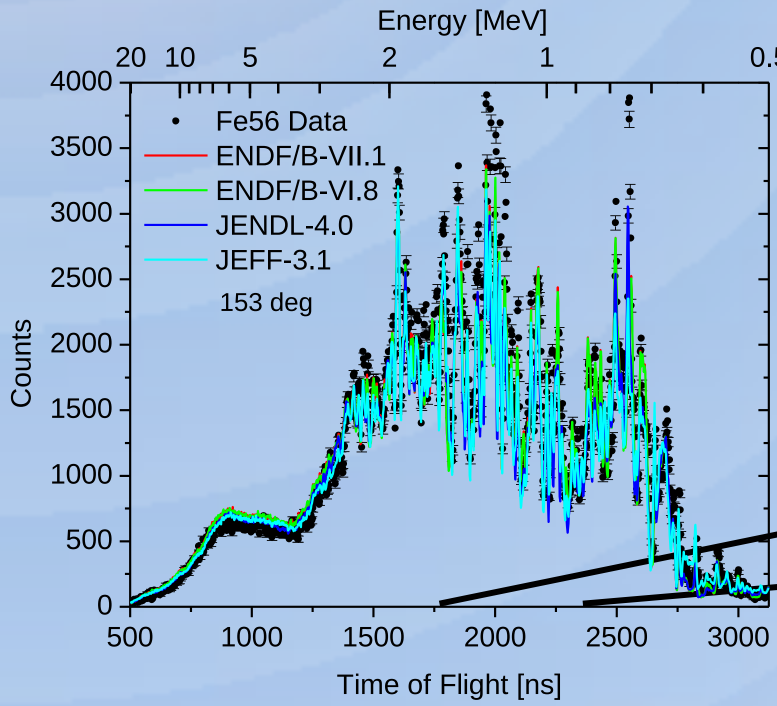
^{56}Fe Scattering Measurement – Results 153°

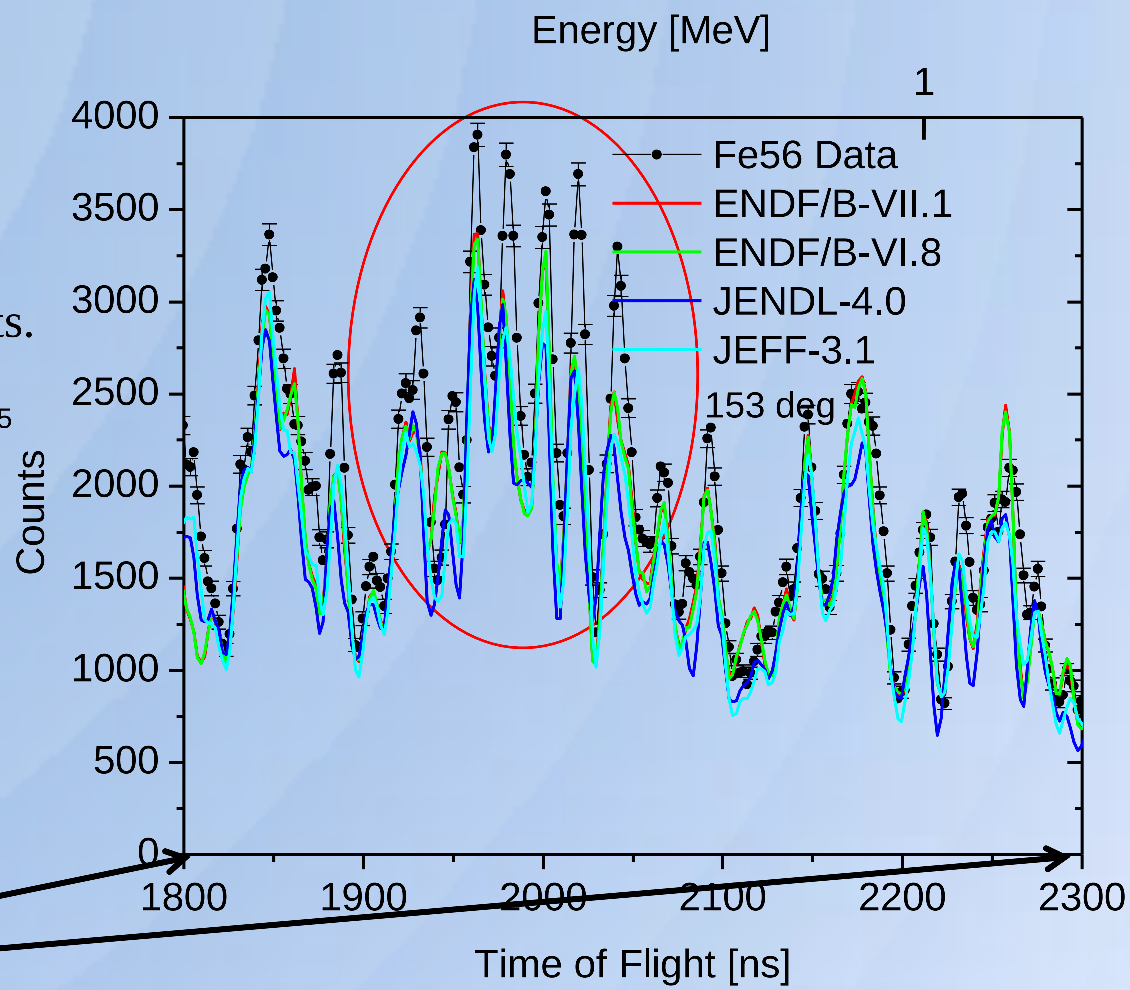
The energy resolution is sufficient to show some discrepancies in the resonance region (E<850 keV)



Fe-56 Scattering Measurement – Results 153°

- Above the first inelastic state (E>847 keV) there are some differences with the evaluations
- We are exploring the possibility to extract double differential cross section data from these experiments.





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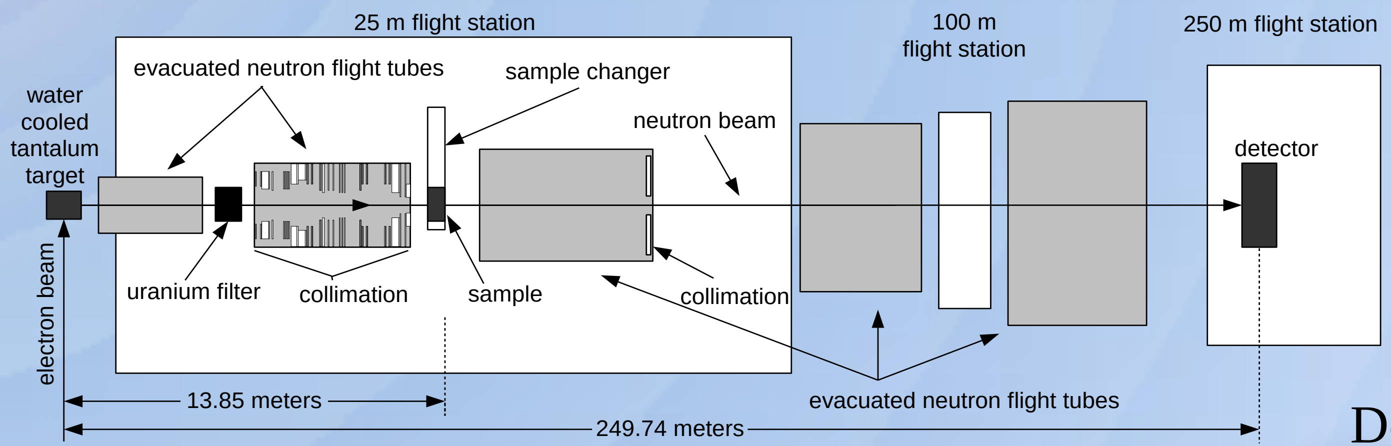
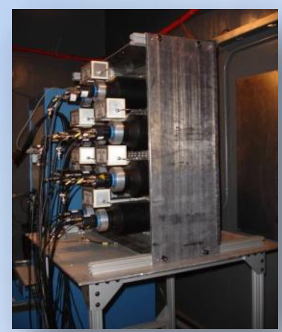
High Energy ^{56}Fe Transmission

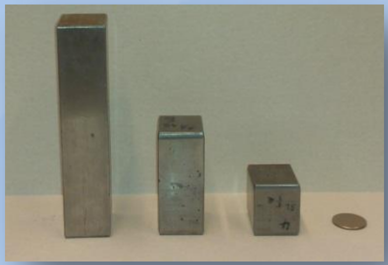
Experimental Setup



Detector

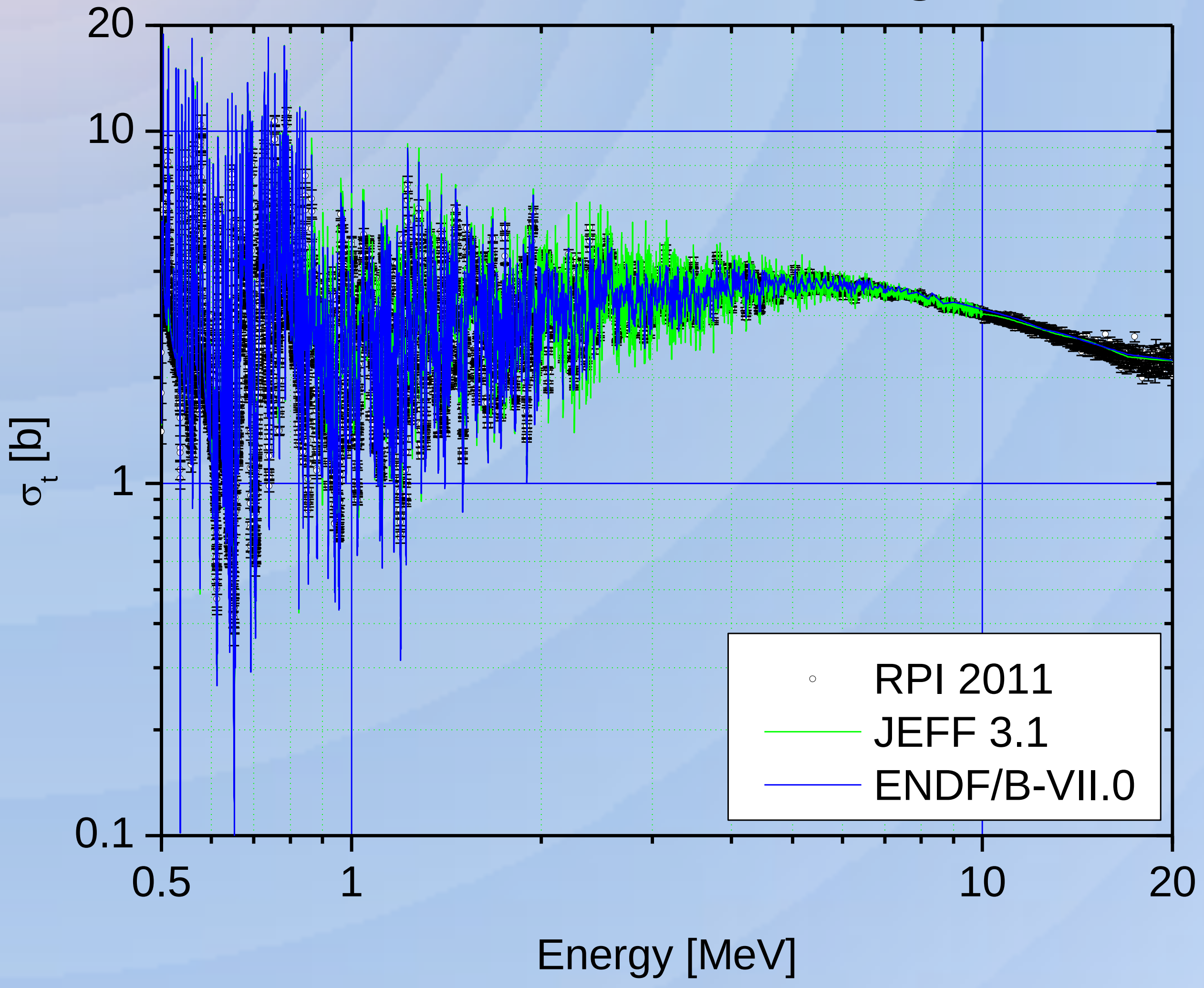


^{56}Fe Samples



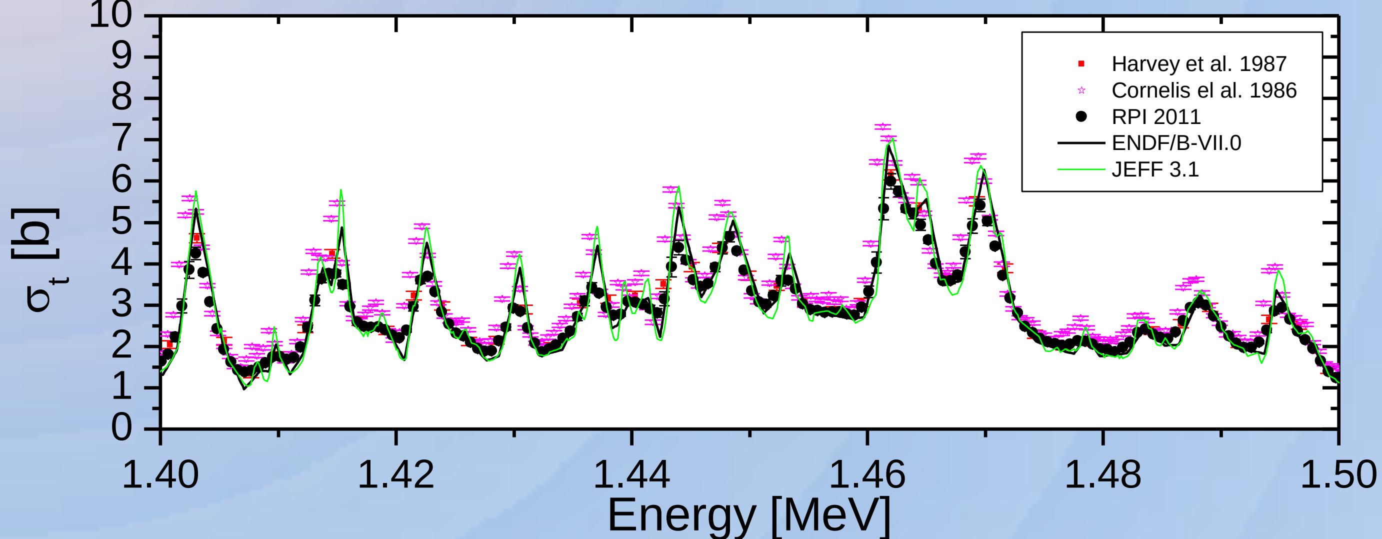
^{56}Fe Total Cross Section Measurements (NCSP)

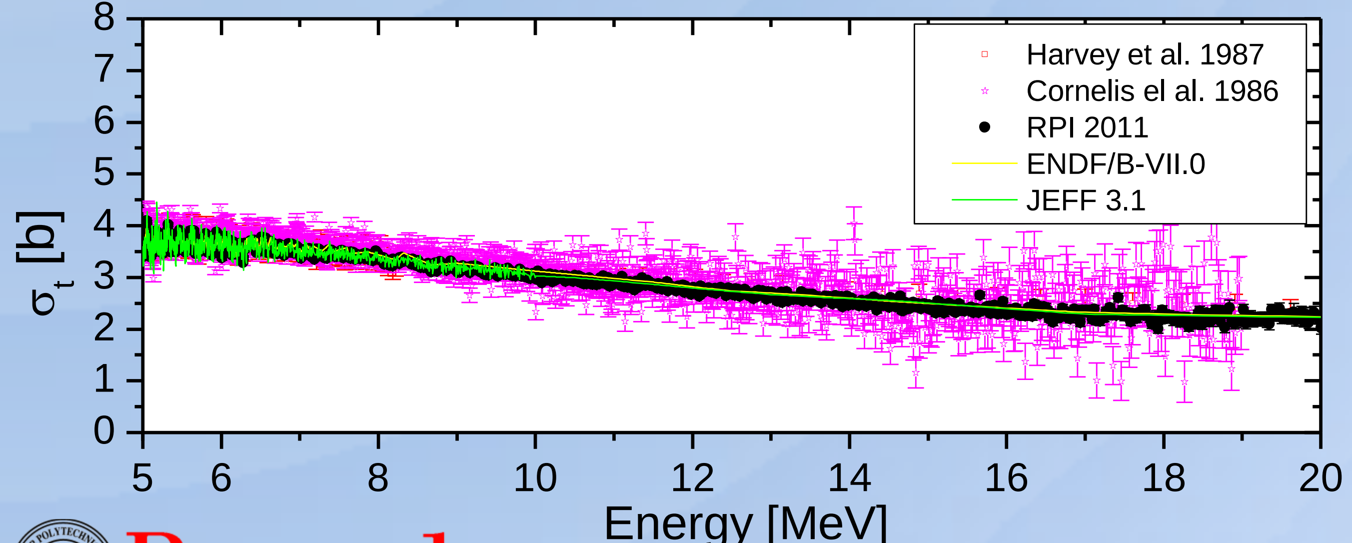
250m Flight Path



- Measured at 250m flight station with 8ns pulse width.
- Two sample thicknesses were used 0.271767 a/barn (3.22 cm) and 0.649742 a/barn (7.69 cm)
- Sample is 99.87% metallic ^{56}Fe
- Can help extend the resolved resonance region above 892 keV
- Only two other data sets available on EXFOR above 900 keV (Harvey et al. and Cornelis et al.)
- The JEFF 3.1 evaluation follows the Cornelis et al. data

^{56}Fe Total Cross Section Measurements





- New data has good energy resolution but lower than Cornelis et al.
- The Cornelis et al. data is based on an oxide sample Fe_2O_3 (need to correct for O_3)
- Above 10 MeV the data has low errors and is in good agreement with both ENDF/B-VII.0 and JEFF 3.1

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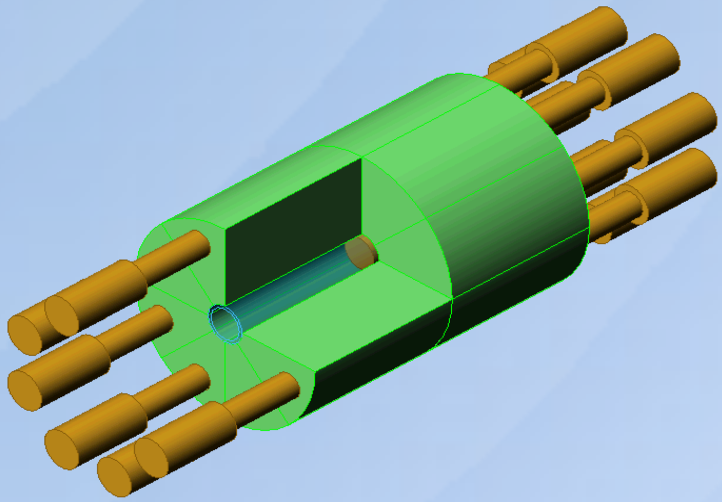
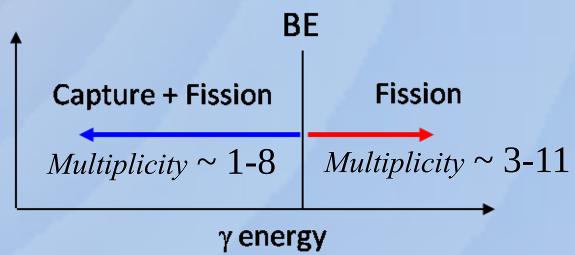
Simultaneous Measurement of ^{235}U Fission and Capture



Measurements of ^{235}U Capture & Fission Yields

- **Thermal** measurement with enriched ^{235}U sample
- 16 Segment Multiplicity Detector with 4 E_γ groups
- Good agreement with SAMMY calculations
- Extracting Capture Yield from data with mixture of capture and fission events



Normalization

➢ Normalize experimental fission yield to thermal point

$$Y_f^{ENDF} = k_1 \cdot Y_f$$ Solve for k_1 @ 0.0253 eV

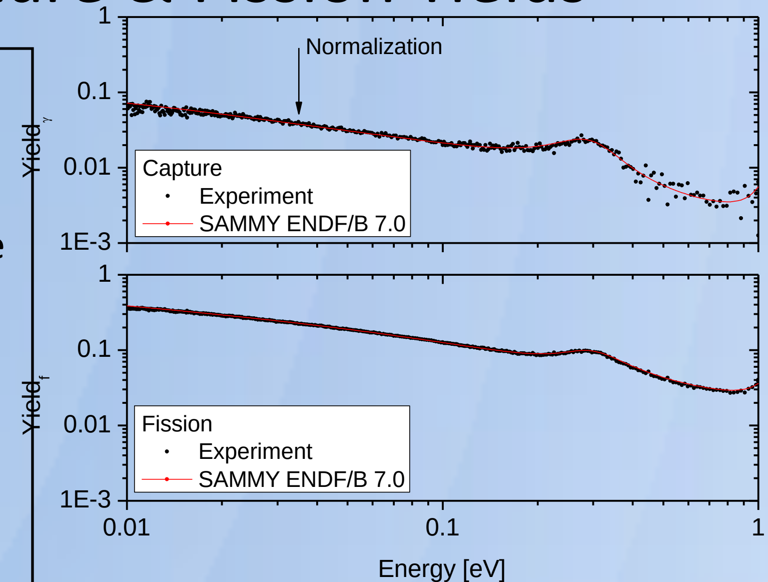
➢ Use two equations for a predominantly capture resonance and predominantly fission region (thermal)

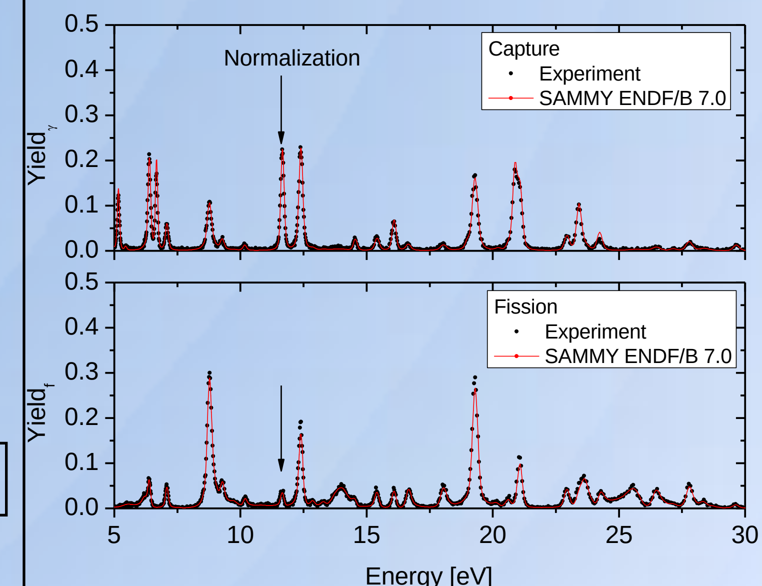
@ 11.7 eV res $\left(\frac{\Gamma_\gamma}{\Gamma} = 0.86\right)$ @ 0.0253 eV

$$Y1_\gamma^{ENDF} = k_2 \cdot Y1_\gamma - k_3 \cdot k_1 \cdot Y1_f$$

$$Y2_\gamma^{ENDF} = k_2 \cdot Y2_\gamma - k_3 \cdot k_1 \cdot Y2_f$$

➢ Solve the two equations for k_2 and k_3





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^{235}U Capture & Fission Yield Data - Epithermal Measurement

- Challenges:
  - Normalization
  - False capture due to neutron scattering

➢ Normalize experimental fission yield to resonance

$$\boxed{Y_f^{ENDF} = k_1 \cdot Y_f} \qquad \text{Solve for } k_1 \text{ @ 19.3 eV res } \left(\frac{\Gamma_f}{\Gamma} = 0.63\right)$$

➢ Use two equations for predominantly capture and fission resonances

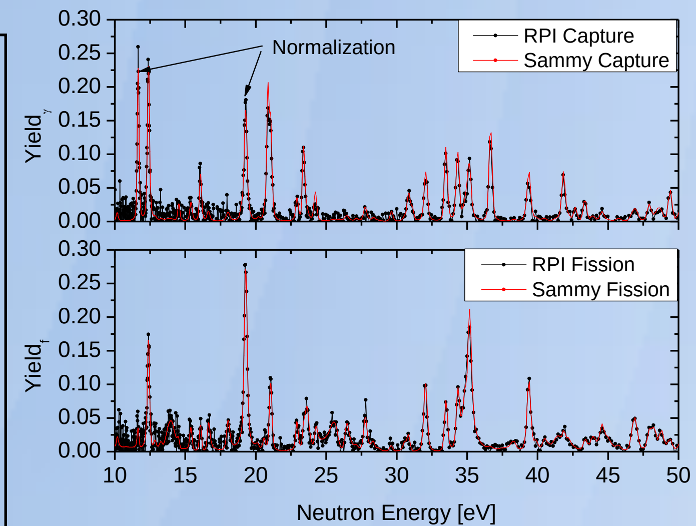
$$\text{@ 11.7 eV res } \left(\frac{\Gamma_\gamma}{\Gamma} = 0.86\right) \qquad \text{@ 19.3 eV res } \left(\frac{\Gamma_f}{\Gamma} = 0.63\right)$$

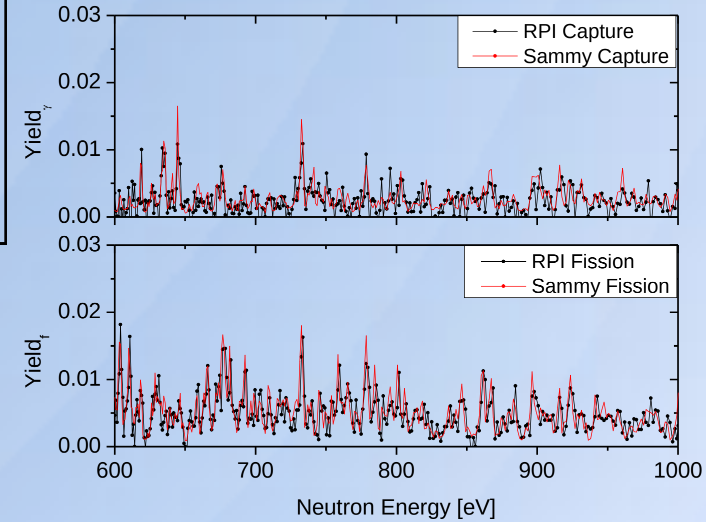
$$\boxed{Y1_\gamma^{ENDF} = k_2 \cdot Y1_\gamma - k_3 \cdot k_1 \cdot Y1_f} \qquad \boxed{Y2_\gamma^{ENDF} = k_2 \cdot Y2_\gamma - k_3 \cdot k_1 \cdot Y2_f}$$

➢ Solve the two equations for k_2 and k_3

► Need 2 resonances with known parameters ◄

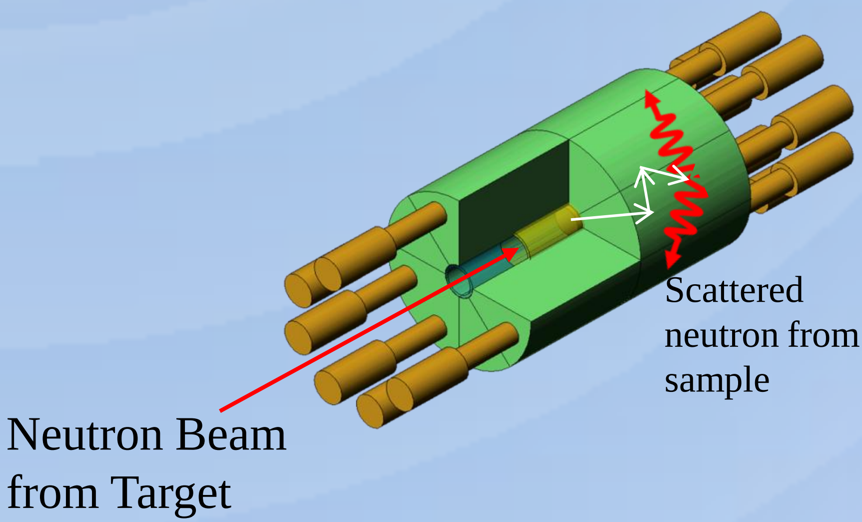
Provides data to address WPEC subgroup 29 report
"Uranium-235 Capture Cross-section in the keV to MeV Energy Region"

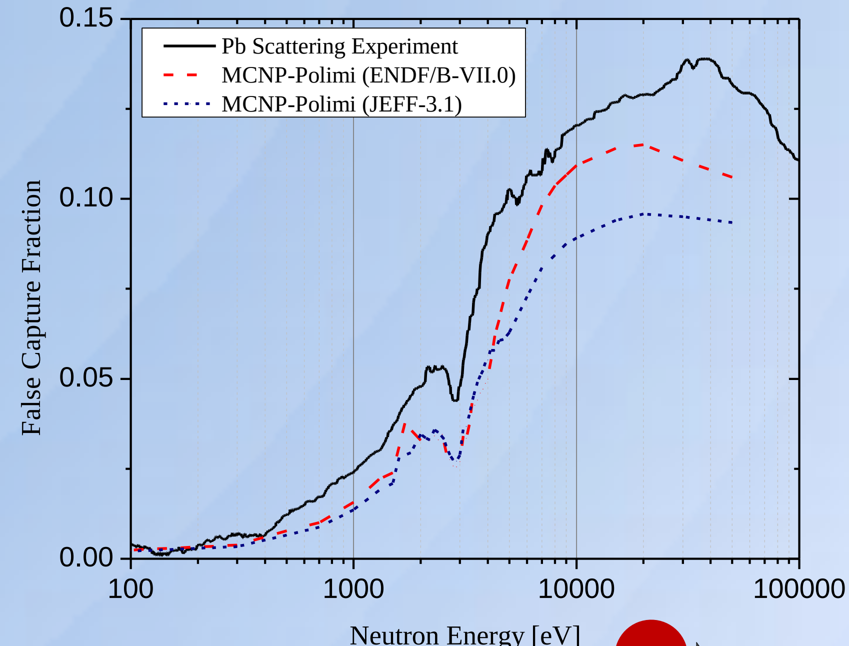




Analysis Method – False Capture

- Neutron scattering from the sample is termed "false capture" impacts the capture-fission group **above 600 eV**
- A fraction of the scattered neutrons penetrated the $^{10}B_4C$ liner, and subsequently was captured in the NaI(Tl), and deposited a total γ energy exceeding 2 MeV
- The false capture fraction was first studied with MCNP-Polimi simulations which are sensitive to ^{127}I (n,γ) and then measured with a Pb scattering experiment.
- The fraction in the experimental results is likely greater due to capture of scattered neutrons in detector materials not fully modeled (ie PMTs)





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Analysis Method – Capture Normalization Epithermal Experiment

- The analysis method for the epithermal experiment is generally the same as for the thermal experiment except for the treatment of an additional background
- Energies used for normalization: E_1 = 11.7 eV and E_2 = 19.0 eV
- The capture yield expression now includes contributions from false capture, $f_c Y_s$

$$Y_\gamma = k_2 Y_{\gamma f}^{\exp} - k_3 k_1 Y_f^{\exp} - f_c Y_s$$

- Since the scattering yield is the least known, it is replaced by the total yield, Y_t, minus the capture and fission yields

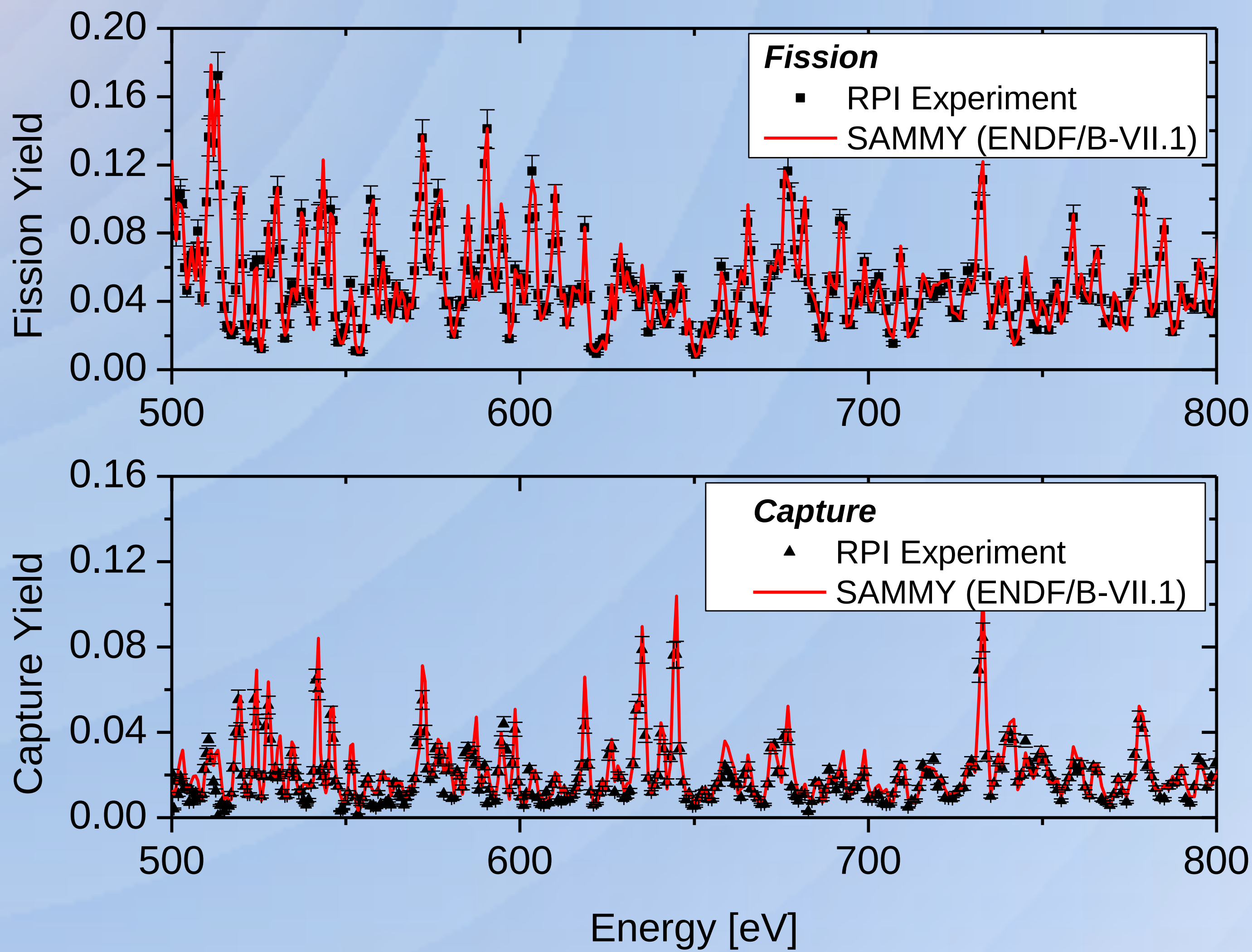
$$Y_s = Y_t - Y_\gamma - Y_f$$

$$Y_\gamma = k_2 Y_{\gamma f}^{\exp} - k_3 k_1 Y_f^{\exp} - f_c (Y_t - Y_\gamma - Y_f)$$

- Solve for the capture yield:

$$Y_{\gamma_i} = \frac{k_2 Y_{\gamma f_i}^{\exp} - (k_3 - f_{c_i}) k_1 Y_{f_i}^{\exp} - f_{c_i} Y_{t_i}}{1 - f_{c_i}}$$

^{235}U Resonance region Data



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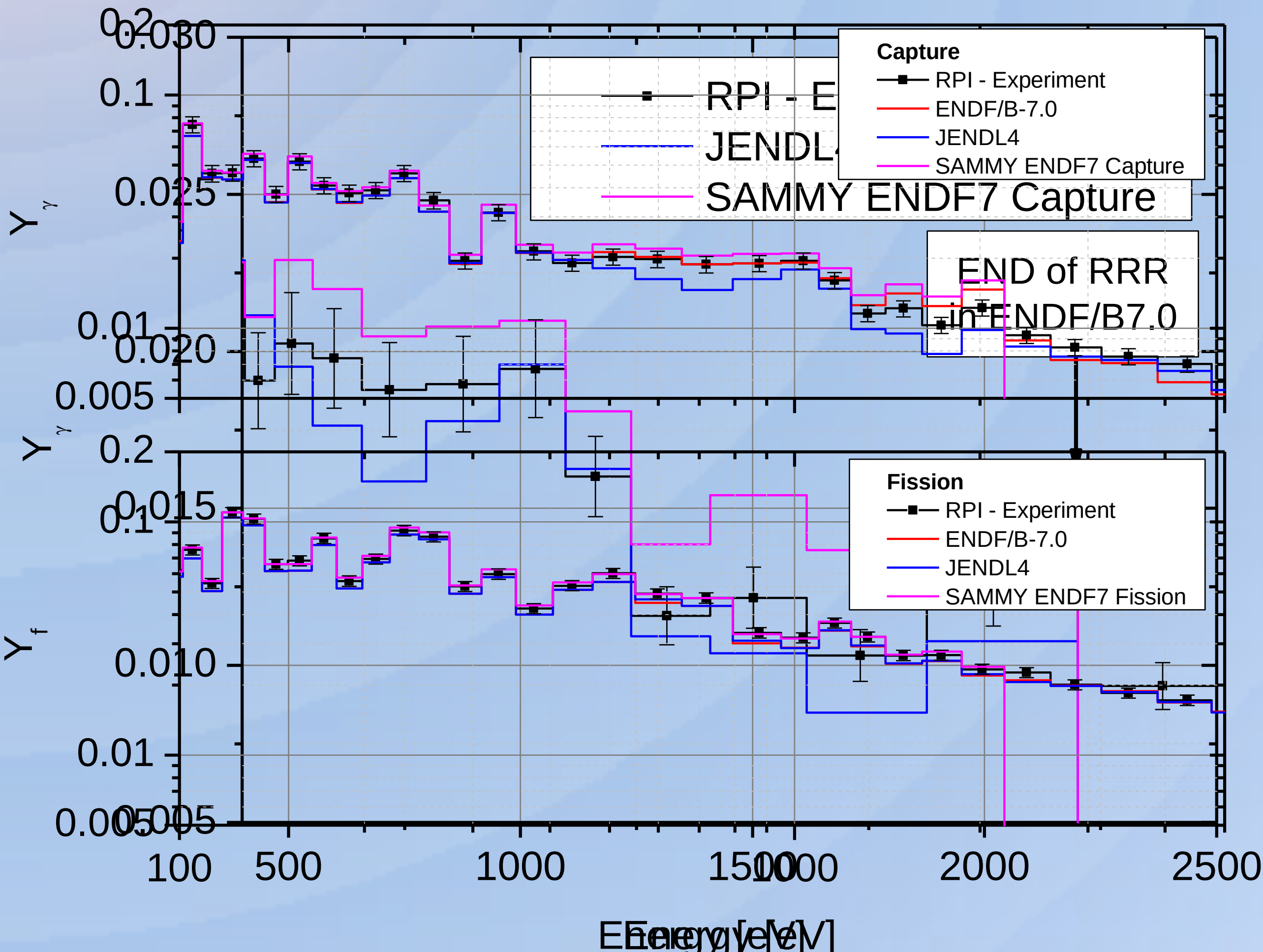
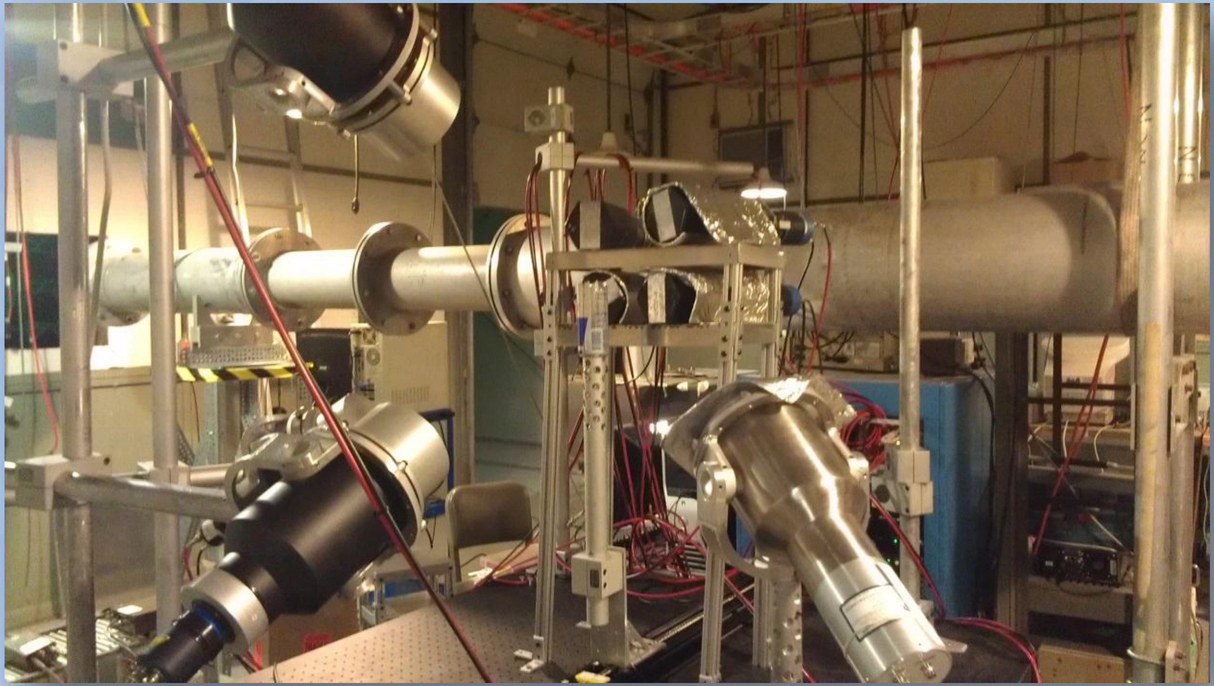
Comparing ^{235}U Fission and Capture with Evaluations



- Fission is in excellent agreement with evaluations
- Capture data has up to 8% multiple scattering that must be taken into account during the analysis
- Capture error is about 8%
- **0.4-1 keV capture data is closer to ENDF/B-7.0**
- **1-2 keV ENDF/B7.0 too high JENDL 4.0 too low.**
- **E>1 keV data is slightly higher than evaluations but within errors.**

Prompt Fission Neutron Spectra



Fission Spectrum Measurement

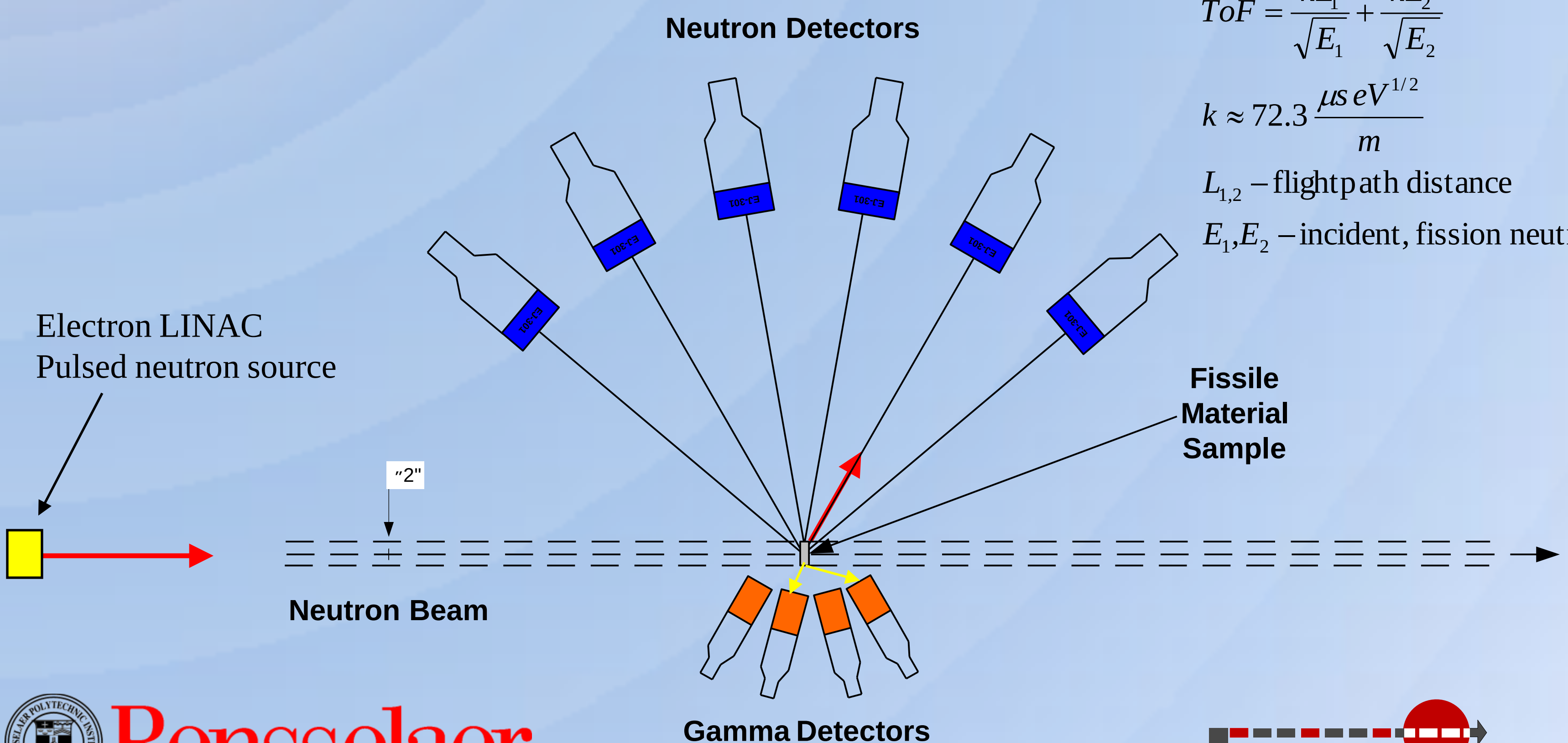
- Use the double TOF method
- Use a gamma tag for fission (instead of traditional fission chamber)
- Use a combination of Liquid Scintillators and Li-Glass neutron detectors

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

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

$L_{1,2}$ – flightpath distance

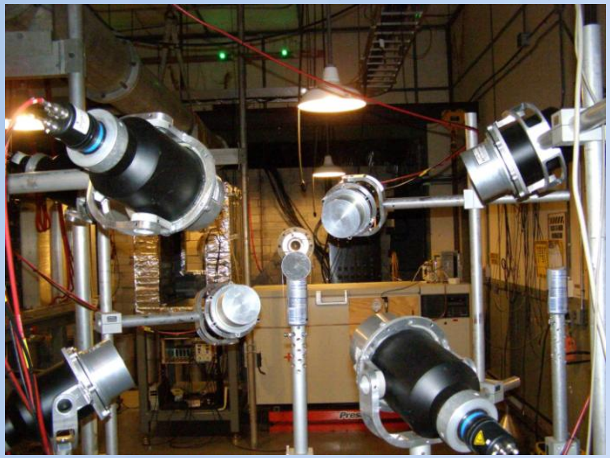
E_1, E_2 – incident, fission neutron energy



Gamma Tagging

- Advantages
  - Eliminated the need to construct a complicated multiplate fission chamber
  - Simpler sample preparation
  - Can use relatively large samples
  - Can increase the detected fission rate
- Disadvantages
  - False fission detection due to:
    - Random coincidence for radioactive decay
    - Neutron interactions with the gamma detector
    - Beam related:
      - Gamma capture
      - Inelastic Scattering
      - Increased background

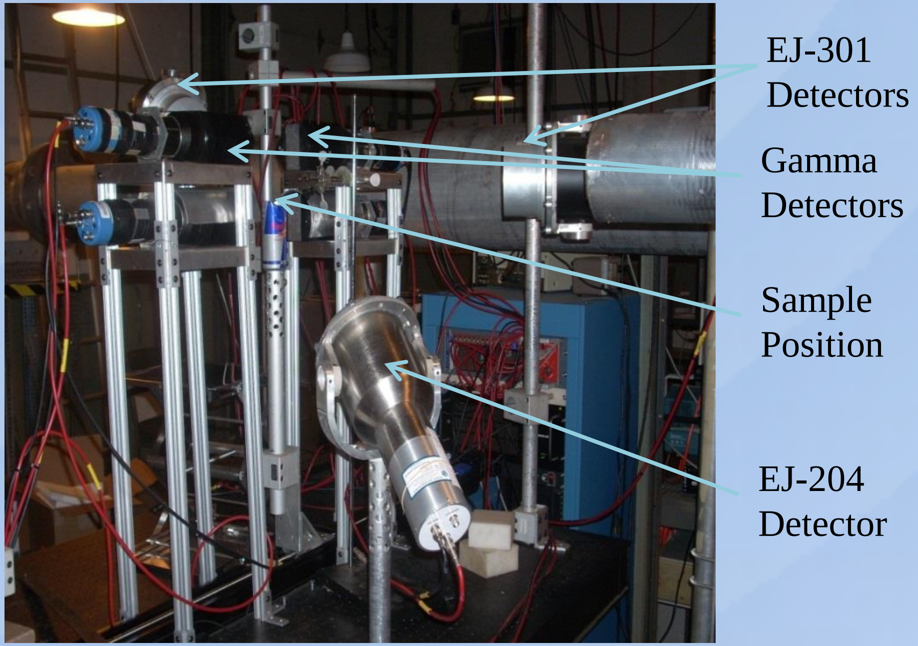
Fast neutrons scattering detector array



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



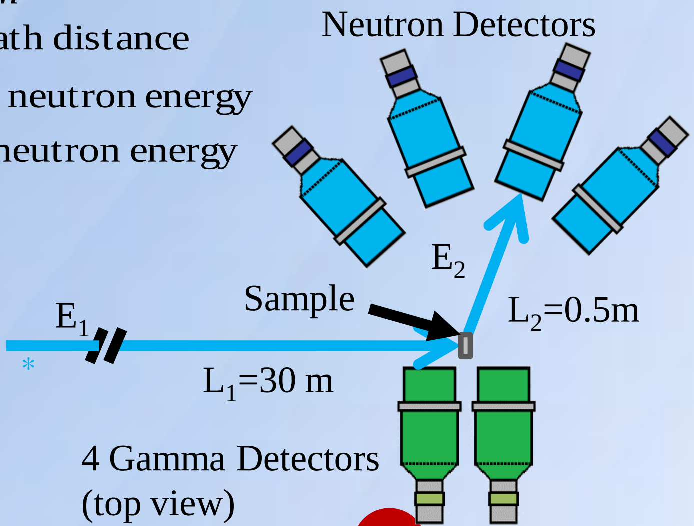
$$ToF = \frac{kL_1}{\sqrt{E_1}} + \frac{kL_2}{\sqrt{E_2}}$$

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

$L_{1,2}$ – flightpath distance

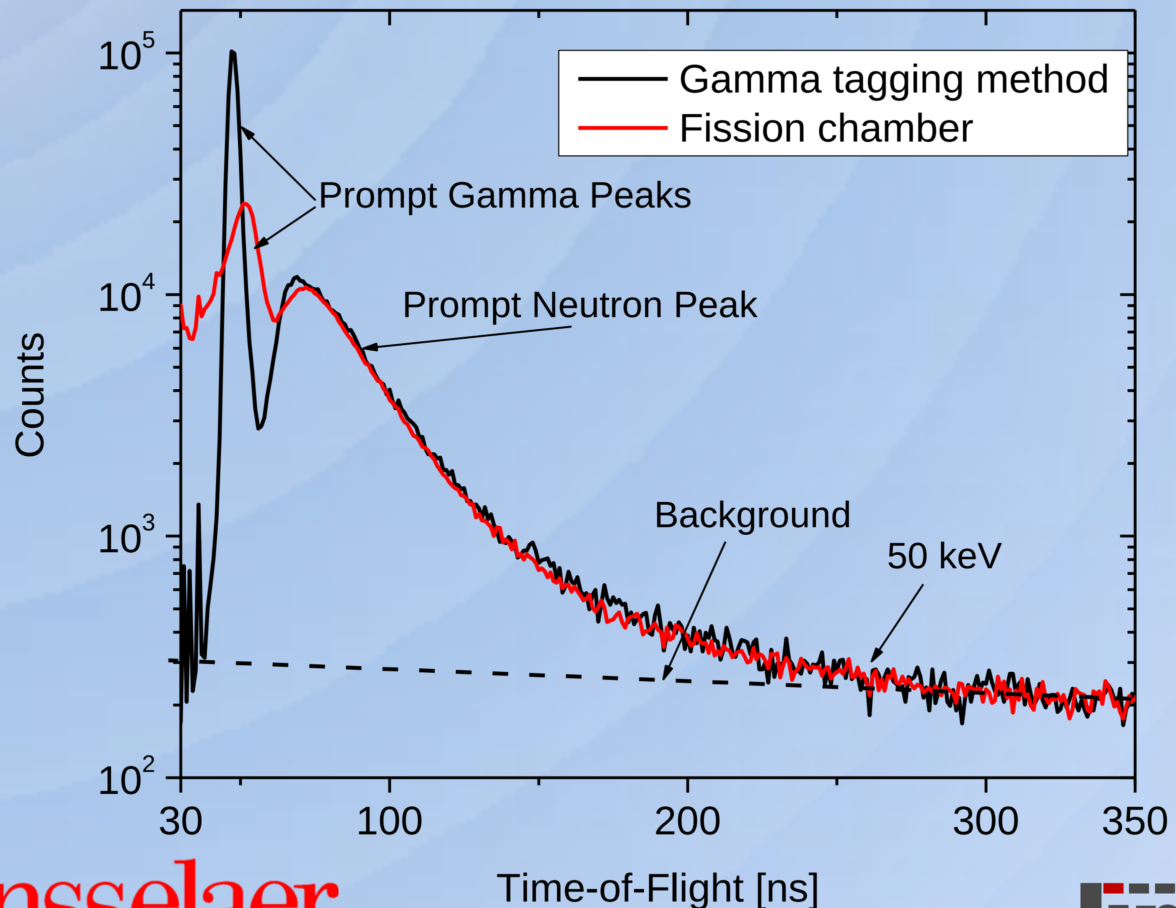
E_1 – incident neutron energy

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



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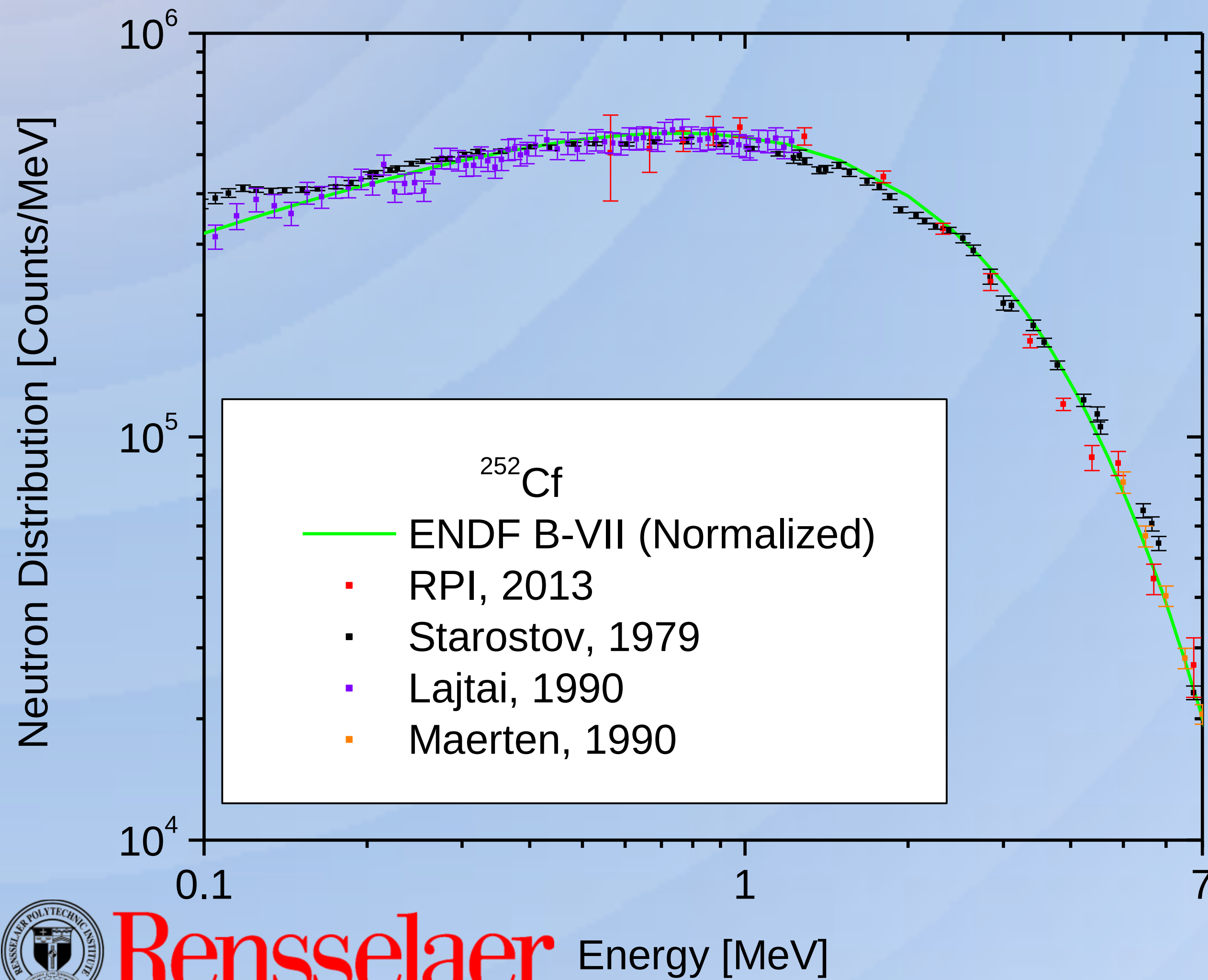
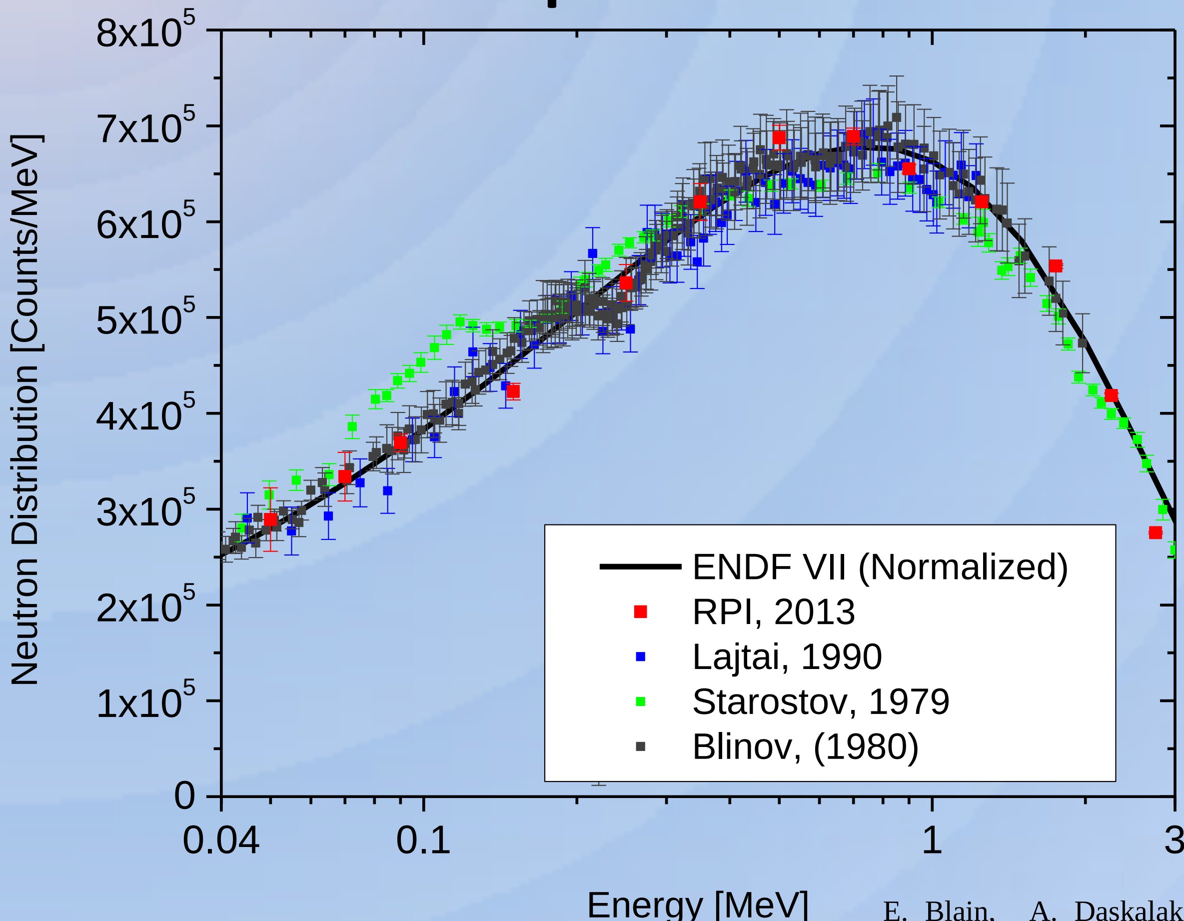
^{252}Cf Prompt Fission Neutron Spectrum High Energy



- High Energy spectrum taken with EJ-301 liquid scintillator
- The gamma tagging method shows good agreement to ENDF/B-VII in the energy range from 0.6 MeV to 7 MeV

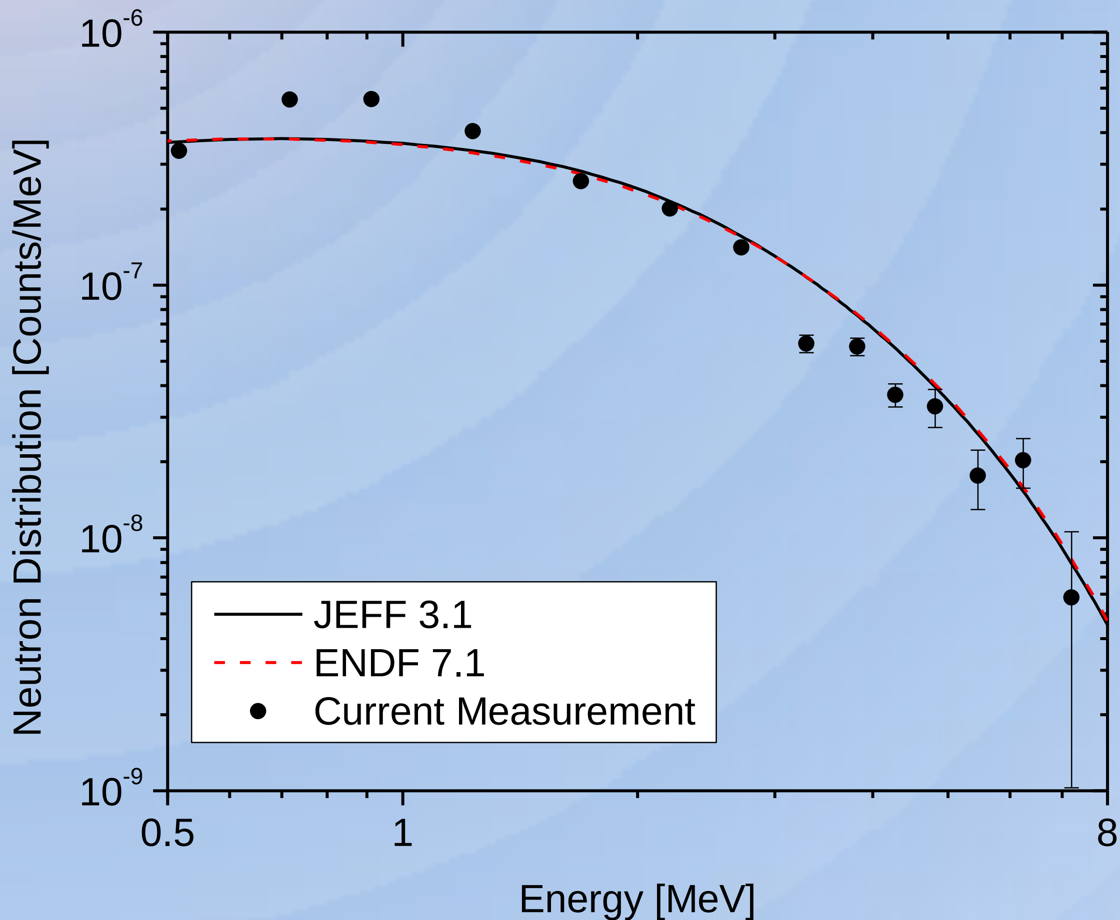
^{252}Cf Prompt Fission Neutron Spectrum Low Energy



- 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.

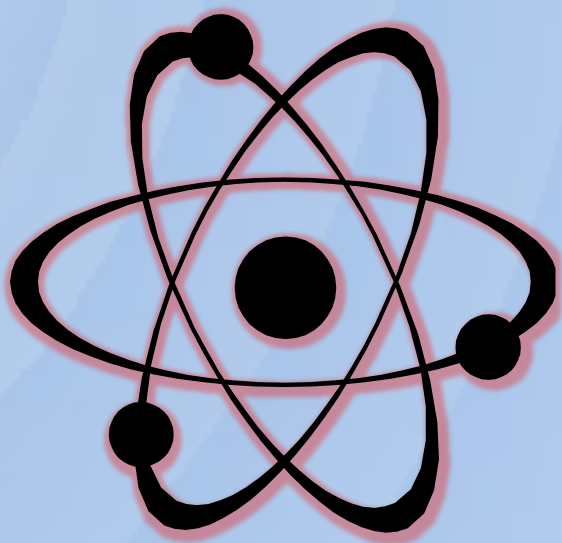
^{238}U Prompt Fission Neutron Spectrum High Energy
Preliminary Results



- Spectrum is normalized to ENDF using detector 3 at 0.9 MeV
- Spectrum is integrated over all incident time-of-flights
- Preliminary data show good agreement with current evaluations

Summary

- Neutron scattering in the energy range from 0.5-20 was measured for ^{56}Fe and ^{238}U at several scattering angles.
  - Data is used with MCNP as benchmark for evaluations.
  - Based on χ^2 the ^{238}U data is in best agreement with the JENDL-4 evaluations.
- High energy transmission experiment of ^{56}Fe provides additional data above 4 MeV
  - Uses metallic sample.
  - In good agreement with current evaluations.
- Capture and fission yields were measured for ^{235}U
  - The experimental data support a lower capture cross section above 500 eV
  - The average cross section is closer to the JENDL-4 evaluation
- 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 are preliminary.



Thank You