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

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2013 NEMEA-7/CIELO Workshop, 5-8 November 2013, Geel, Belgium
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RPI Nuclear Data Group
The Nuclear Data Program at the Gaerttner LINAC Center

- Driven by a 60 MeV pulsed electron LINAC $\sim 10^{12}$ 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.
  - $(n,\alpha)$, $(n,p)$ and $(n,\gamma)$ 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 \( t_1 \gg t_2 \) and \( E_1 \sim E_2 \) then for presentation the incident neutron energy \( E_1 \) is calculated using \( t \) and \( L = L_1 + L_2 \)

\[
L_1, t_1, E_1 \\
L_1 \sim 30m \\
\]

\[
L_2 \sim 0.5m \\
E(t) \approx m_n c^2 \cdot \left( \frac{1}{\sqrt{1 - (\frac{L}{c \cdot t})^2}} - 1 \right)
\]
First Order Approximation of the Scattering Yield

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

Detector Efficiency

Probability to Interact

Incident Neutron Flux

Probability to Scatter in direction \(\phi\)

In this approximation - multiple scattering is ignored
Experimental Setup Overview

- Air
- Isotropic Neutrons
  - Tantalum Target
  - Target Room
  - Electron Beam (50 MeV) 5nsec pulse at 400 Hz
  - 6 m
  - Vacuum
  - West Drift Tube Collimation System
  - Concrete
  - ¾” U238
  - 24 m
  - 25m Station Wall
  - Scattering Experiment Setup
  - EJ301 Detector
  - Scattering Sample
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.

![Graph showing pulses and energy distribution](image-url)
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 $\eta=1$ (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:

\[ Rn_i = Rn_i^s - fn_i^s - \frac{M^S}{M^O} \cdot (Rn_i^o - fn_i^o) \]

- \( Rn_i^s, Rn_i^o \) - Sample and open neutron counts at TOF channel \( i \)
- \( fn_i^s, fn_i^o \) - 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:

\[ fn_i = \sum_{j=1}^{n_\gamma} f(A_j) \]

- \( n_\gamma \) - 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 considered 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)
238U Scattering - Forward Angles

<table>
<thead>
<tr>
<th>Library</th>
<th>$\chi^2$</th>
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<tbody>
<tr>
<td>ENDF/B-VII.1</td>
<td>4.4</td>
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<tr>
<td>ENDF/B-VI.8</td>
<td>2.7</td>
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<tr>
<td>JENDL-4.0</td>
<td>2.5</td>
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<tr>
<td>JEFF-3.1</td>
<td>3.3</td>
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<tr>
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<td>JENDL-4.0</td>
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<td>JEFF-3.1</td>
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$^{238}$U Scattering – Back Angles

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<tr>
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<td>JEFF-3.1</td>
<td>4.7</td>
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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 the lowest $\chi^2$
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.

![Graph showing scattering results](image)
High Energy $^{56}$Fe Transmission

Experimental Setup

Detector

$^{56}$Fe Samples
56Fe 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 56Fe.
- 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 Fe₂O₃ (need to correct for O₃)
- Above 10 MeV the data has low errors and is in good agreement with both ENDF/B-VII.0 and JEFF 3.1
Simultaneous Measurement of $^{235}\text{U}$ Fission and Capture
Measurements of $^{235}$U Capture & Fission Yields

- **Thermal** measurement with enriched $^{235}$U sample
- 16 Segment Multiplicity Detector with 4 $E_\gamma$ 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

\[ Y_f^{\text{ENDF}} = k_1 \cdot Y_f \]  
Solve for $k_1 \approx 0.0253$ eV

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

\[ Y_{1,\gamma}^{\text{ENDF}} = k_2 \cdot Y_{1,\gamma} - k_3 \cdot k_1 \cdot Y_{1,f} \]  
\[ Y_{2,\gamma}^{\text{ENDF}} = k_2 \cdot Y_{2,\gamma} - k_3 \cdot k_1 \cdot Y_{2,f} \]

- Solve the two equations for $k_2$ and $k_3$


**235\textsuperscript{U} Capture & Fission Yield Data - Epithermal Measurement**

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

  ➢ Normalize experimental fission yield to resonance

  \[
  Y_f^{ENDF} = k_1 \cdot Y_f
  \]

  Solve for \( k_1 \) @ 19.3 eV res \( \left( \frac{\Gamma_f}{\Gamma} = 0.63 \right) \)

  ➢ Use two equations for predominantly capture and fission resonances

  @ 11.7 eV res \( \left( \frac{\Gamma_r}{\Gamma} = 0.86 \right) \) @ 19.3 eV res \( \left( \frac{\Gamma_f}{\Gamma} = 0.63 \right) \)

  \[
  Y_1^{ENDF} = k_2 \cdot Y_1 - k_3 \cdot k_1 \cdot Y_1_f
  \]

  \[
  Y_2^{ENDF} = k_2 \cdot Y_2 - k_3 \cdot k_1 \cdot Y_2_f
  \]

  ➢ 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$_4$C liner, and subsequently was captured in the NaI(Tl), and deposited a total $\gamma$ energy exceeding 2 MeV
- The false capture fraction was first studied with MCNP-Polimi simulations which are sensitive to $^{127}$I (n,$\gamma$) 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)

![Neutron Beam from Target]

![Scattered neutron from sample]

![False Capture Fraction vs Neutron Energy [eV]]
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, $f_c Y_s$

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

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

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

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

- Solve for the capture yield:

$$Y_\gamma = \frac{k_2 Y_\gamma^{exp} - (k_3 - f_{c_i}) k_1 Y_f^{exp} - f_{c_i} Y_t}{1 - f_{c_i}}$$
235U Resonance region Data

**Fission**
- RPI Experiment
- SAMMY (ENDF/B-VII.1)

**Capture**
- RPI Experiment
- SAMMY (ENDF/B-VII.1)
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

\[ \text{ToF} = \frac{kL_1}{\sqrt{E_1}} + \frac{kL_2}{\sqrt{E_2}} \]
\[ k \approx 72.3 \frac{\mu s \text{ eV}^{1/2}}{m} \]

\( L_{1,2} \) – 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
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

\[ \text{ToF} = \frac{kL_1}{\sqrt{E_1}} + \frac{kL_2}{\sqrt{E_2}} \]

\[ k \approx 72.3 \frac{\mu s \ eV^{1/2}}{m} \]

$L_{1,2}$ – 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
The Gaertner LINAC Center

$^{252}$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

<table>
<thead>
<tr>
<th>Neutron Distribution [Counts/MeV]</th>
<th>Energy [MeV]</th>
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<tbody>
<tr>
<td>$^{252}$Cf ENDF B-VII (Normalized)</td>
<td></td>
</tr>
<tr>
<td>RPI, 2013</td>
<td></td>
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<tr>
<td>Starostov, 1979</td>
<td></td>
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<tr>
<td>Lajtai, 1990</td>
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<td>Maerten, 1990</td>
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</table>
$^{252}$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

\(^{238}\text{U}\) 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

![Neutron Distribution Graph]

- **JEFF 3.1**
- **ENDF 7.1**
- **Current Measurement**
Summary

• Neutron scattering in the energy range from 0.5-20 was measured for \(^{56}\text{Fe}\) and \(^{238}\text{U}\) at several scattering angles.
  – Data is used with MCNP as benchmark for evaluations.
  – Based on \(\chi^2\) the \(^{238}\text{U}\) data is in best agreement with the JENDL-4 evaluations.
• High energy transmission experiment of \(^{56}\text{Fe}\) provides additional data above 4 MeV
  – Uses metallic sample.
  – In good agreement with current evaluations.
• Capture and fission yields were measured for \(^{235}\text{U}\)
  – 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
  – \(^{252}\text{Cf}\) measured fission neutron spectrum below 1 MeV is in good agreement with evaluations
  – Experimental results for \(^{238}\text{U}\) are preliminary.
Thank You