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Computation of Neutron Multiplicity Statistics using Deterministic Transport Methods

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Workshop on Models and Methods of Non-destructive Assay in Nuclear Safeguards and Non-proliferation

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Introduction

- Inverse transport methods can infer the configuration of an unknown radiation source from its measured radiation signatures

- Source features
  - Isotopic composition
  - Fissile mass and multiplication
  - Geometric arrangement of radiating and shielding materials

- Signatures
  - Gamma spectrometry
  - Neutron time-correlation and multiplicity counting

Technical Approach

- Compute radiation signatures using transport models
- Iteratively modify the model parameters to minimize the error between predicted and measured signatures

- This method is fairly mature using gamma spectral signatures
- However, the gamma spectrum primarily sensitive to the outer surface of the source
- Therefore, a solution based on gamma spectrometry alone is weakly constrained

- Neutron multiplicity signatures tend to be sensitive to the entire source volume
- A simultaneous solution based on gamma spectral and neutron multiplicity signatures is better constrained
- Sandia has developed a fast method to accurately compute neutron multiplicity signatures – it's based on the original work by Muñoz-Cobo, Perez, and Verdú

Objective and Application

Objective

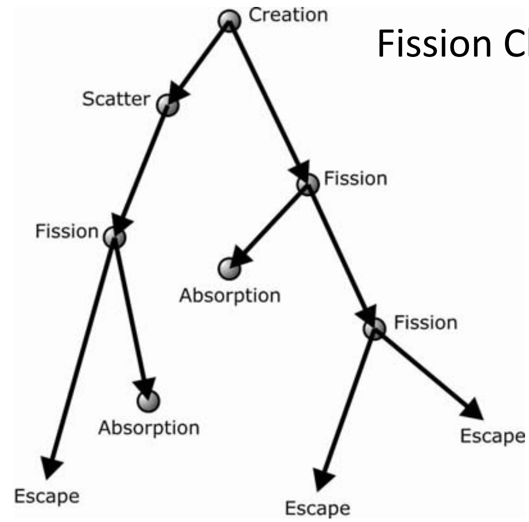
- Rapidly and accurately estimate the properties of bulk quantities of SNM from neutron multiplicity counting measurements
  - Neutron source strength
  - Neutron multiplication
  - Neutron lifetime

Application

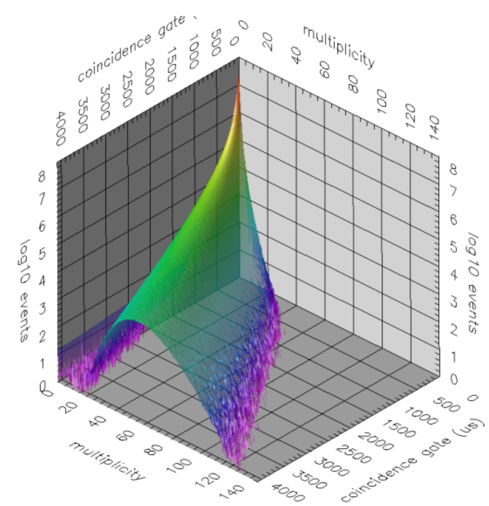
- Combine the analysis of neutron multiplicity and gamma spectrometry in the solution of inverse transport problems
- Useful for material accountability and safeguards measurements for nuclear nonproliferation

Fission Chain Reactions and Neutron Multiplicity Counting

- Fission chain reactions multiply the number of neutrons in a fissile transport medium
- Chain reaction characteristics:
  - Number of neutrons made during the chain reaction: neutron multiplication
  - Speed of the chain reaction evolution: neutron lifetime
- Neutron multiplicity measurements are sensitive to both characteristics
- Neutron multiplicity counting measures the frequency of neutron detection versus:
  - Counting time (a.k.a. coincidence gate width) – usually on order of microseconds
  - Number of coincident counts (a.k.a. multiplicity) – usually between 10’s and 100’s of coincident neutrons



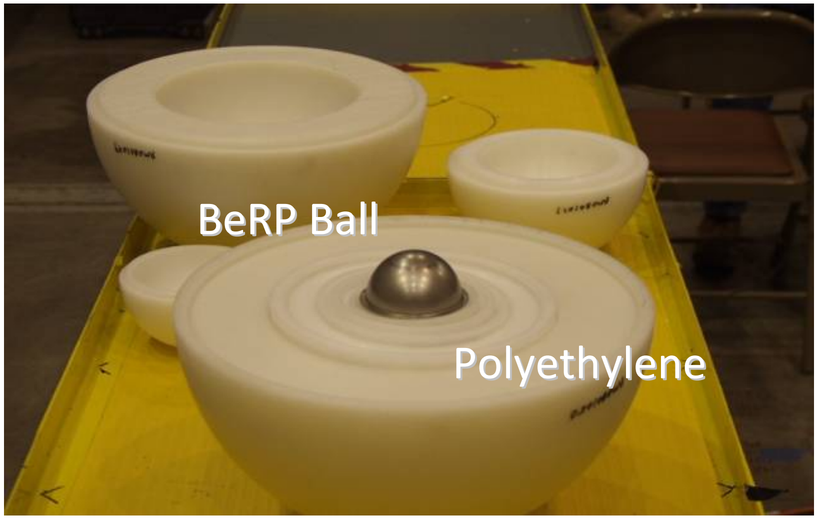
Fission Chain-Reaction

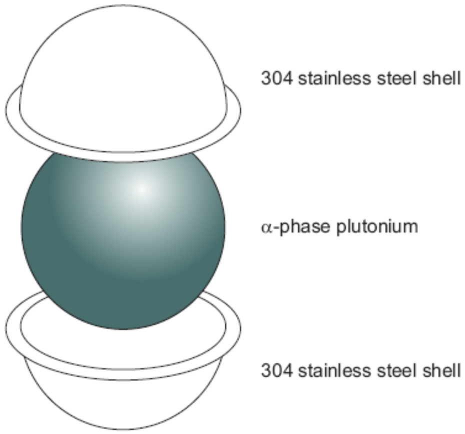


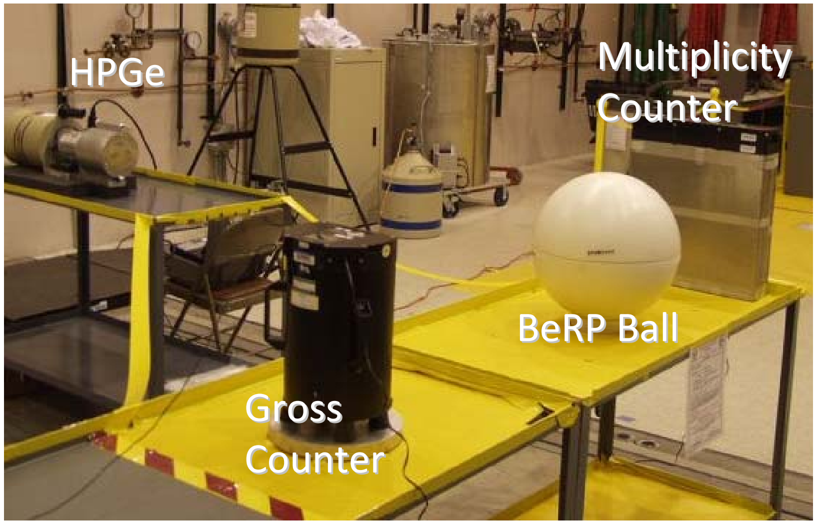
Neutron Multiplicity Distribution

Los Alamos BeRP Ball Experiments

- 4.4 kg of 94% Pu-239 α-phase plutonium metal
- Originally constructed for criticality safety experiments
- Reflected by high-density polyethylene shells 1.3 cm to 15.2 cm thick

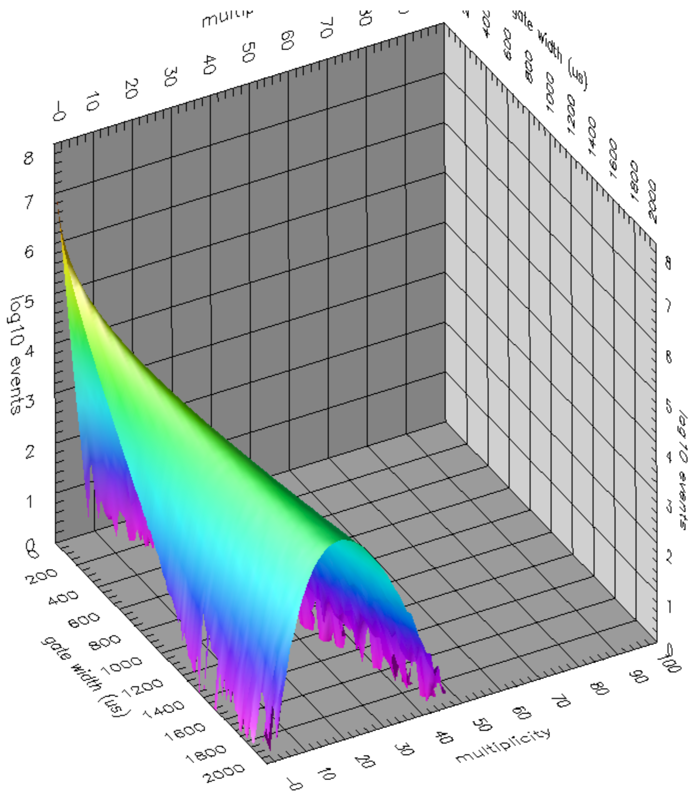






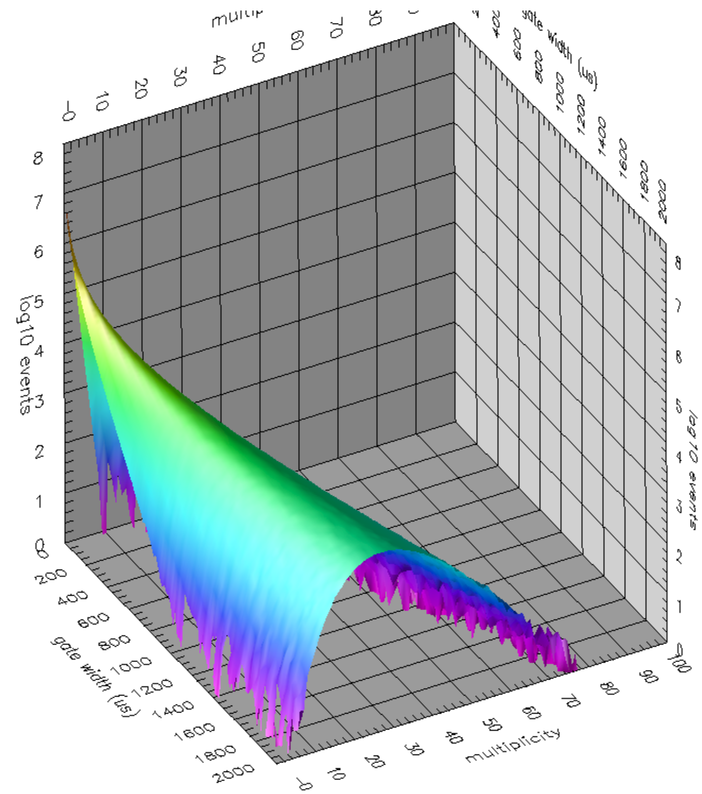
Neutron Multiplicity Distribution

Multiplication ~ 4.4
Generation Time ~ 3 ns



LANL BeRP Ball / Bare

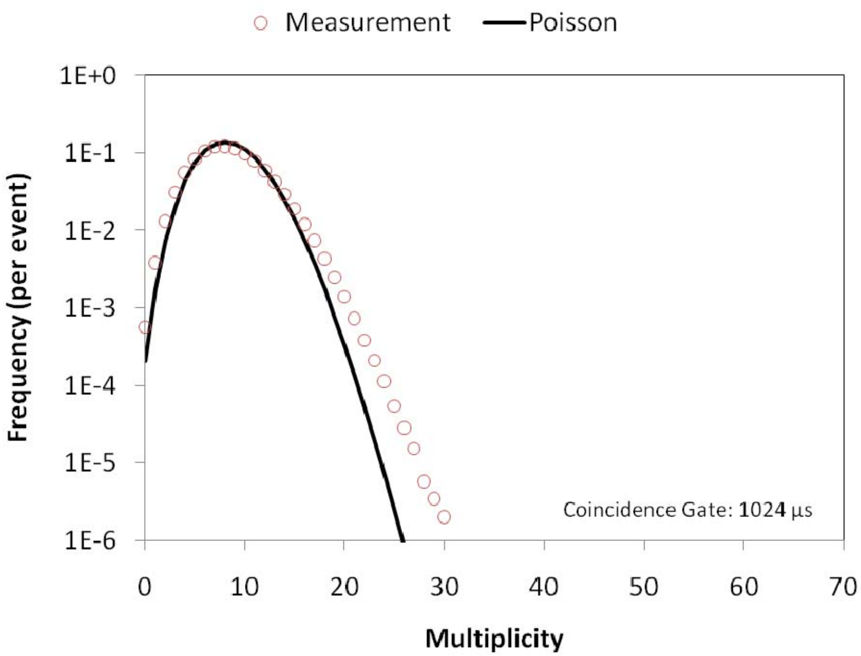
Multiplication ~ 16.3
Generation Time ~ 9 µs



LANL BeRP Ball / 7.6 cm Poly

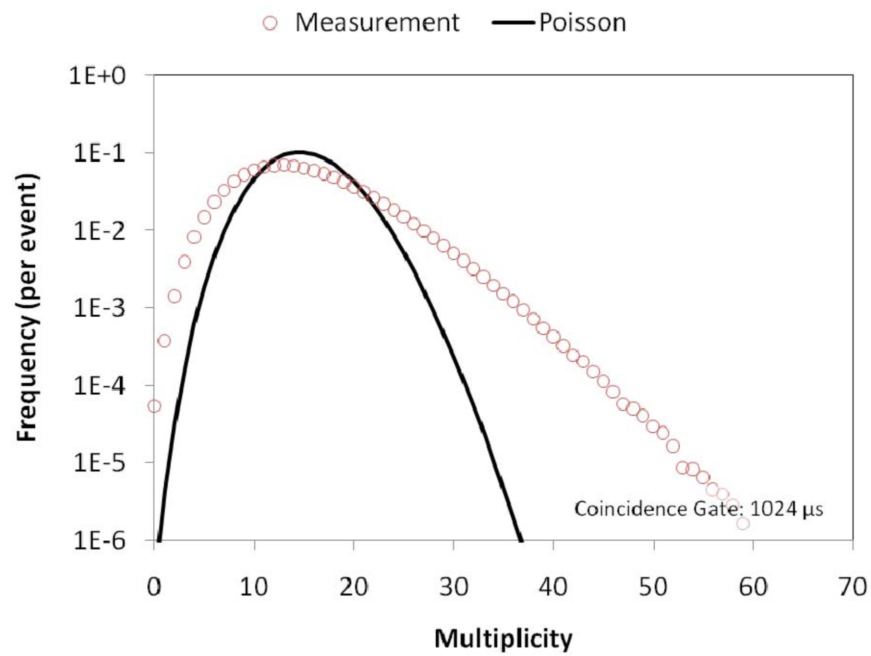
Multiplication Induces Excess Variance

Multiplication ~ 4.4



LANL BeRP Ball / Bare

Multiplication ~ 16.3



LANL BeRP Ball / 7.6 cm Poly

Feynman-Y Neutron Multiplicity Statistic

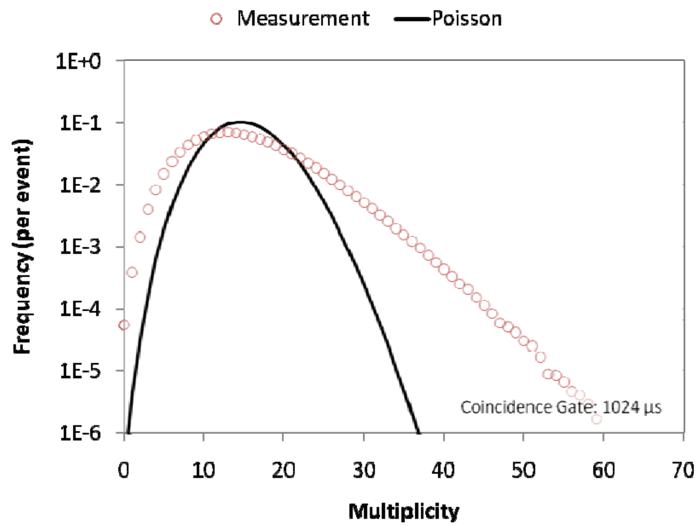
- Fission chain-reactions produce "excess variance"
- The Feynman-Y measures variance in excess of a Poisson distribution

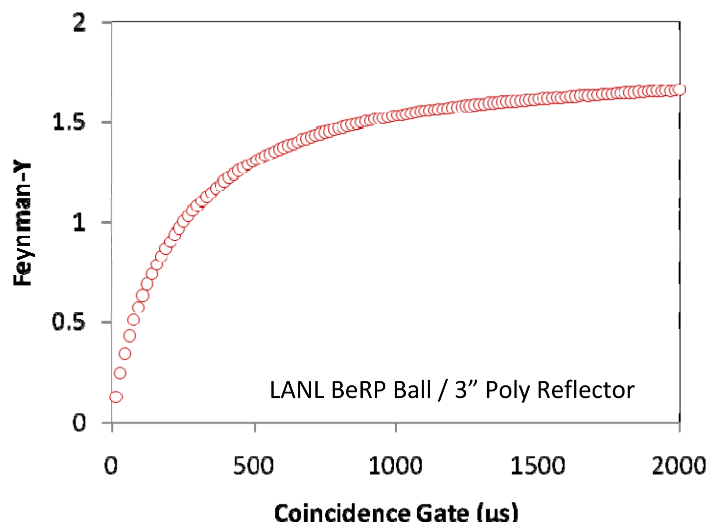
$$\frac{\sigma^2}{\mu} = 1 + Y$$

$$\sigma^2 : variance$$

$$\mu : mean$$

  - Y vanishes if the counting distribution is purely Poisson
  - Y increases with neutron multiplication
- Y is usually measured vs. coincidence gate width

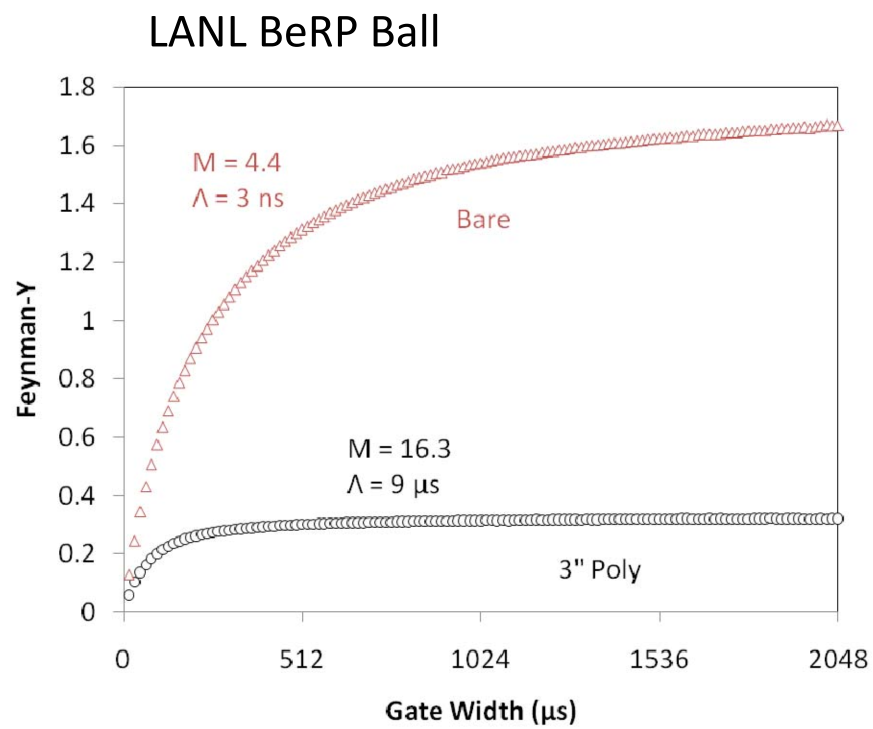




Effect of Multiplication and Generation Time

- Y is a measure of the second moment (the width) of the counting distribution
- Its asymptotic value tends to increase with square of neutron multiplication

- Y is a measure of the system's dynamic response
- Its shape vs. gate width tends to evolve more slowly with increasing neutron generation time



Models Used to Analyze Neutron Multiplicity Measurements

Point reactor models

- Very fast: the models are generally solvable in milliseconds to a few seconds
- Potentially inaccurate: systems where kinetics properties vary significantly across the transport medium cannot be modeled accurately

Monte Carlo models

- Very accurate: Monte Carlo models can accurately simulate transport medium heterogeneity and variations in kinetics properties
- Very slow: calculation of neutron multiplicity statistics can require many minutes to hours (and in some cases, days)

Deterministic models

- Accurate: discrete ordinates models can accurately model heterogeneous transport media with varying kinetics properties
- Fast: neutron multiplicity statistics can be computed in seconds from one-dimensional models

Computation of the Feynman-Y Using Deterministic Transport Calculations

- Feynman-Y exhibits two notional features
  - Asymptotic value
  - Shape dependent on coincidence gate width
- Asymptote
  - Computed from static forward and adjoint transport solution
  - Accounts for relative contribution of source and induced fission neutrons
  - The source term for the adjoint problem is the detection efficiency – the adjoint flux is a “weighting function” that represents importance to detection
- Shape
  - Computed from solution of the dynamic step response problem
  - The forward source term is instantaneously stepped
  - The leakage current is folded with detector cross section & impulse response
  - The detector response is integrated over gate width

See Muñoz-Cobo, Perez, and Verdú, Nuclear Science and Engineering #95, 83-105

Computation of the Feynman-Y Asymptote

- Excess variance comes from the source and induced fission

$$\frac{\sigma^2}{\mu} = 1 + Y \qquad\qquad \sigma^2 = \mu + {}_2S_0 + {}_2S$$

- Excess variance from source neutron production Q

$${}_2S_0 = \int d^3r \int dE\, \frac{\overline{\nu_0(\nu_0 - 1)}}{\overline{\nu_0}}\, Q(\vec{r},E)\, I_0^2(\vec{r}) \qquad I_0(\vec{r}) = \int dE'\, \frac{\chi_0(\vec{r},E')}{4\pi}\, \varphi^\dagger(\vec{r},E')$$

- Excess variance from fission neutron production $\nu\Sigma_f\varphi$

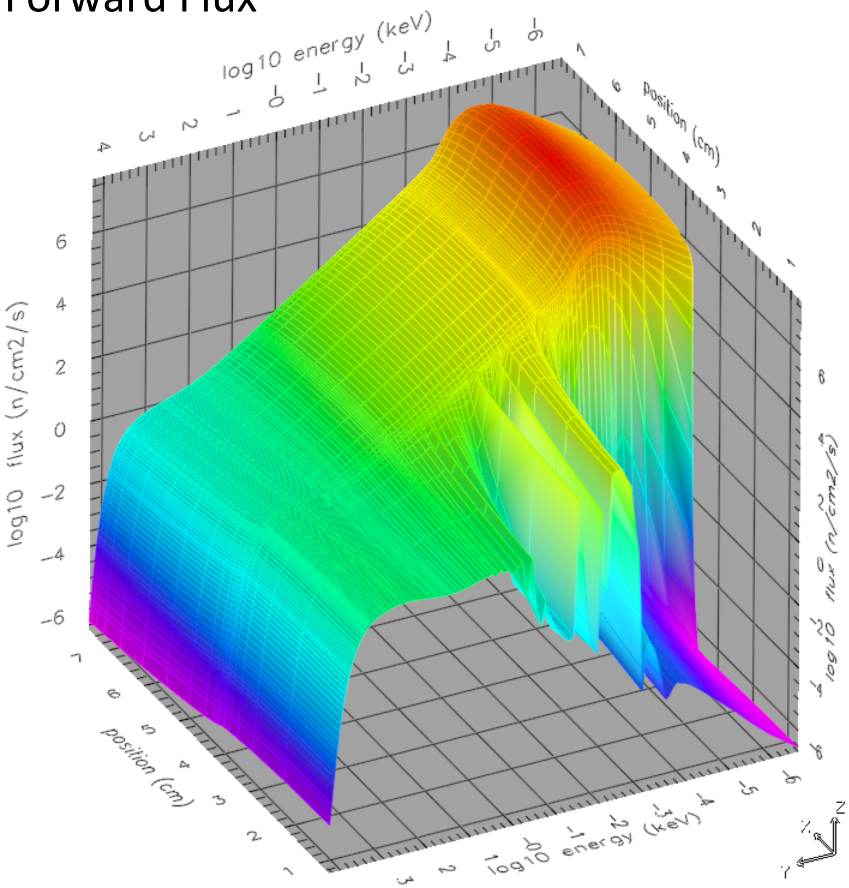
$${}_2S = \int d^3r \int dE\, \overline{\nu(\nu-1)}\,\Sigma_f(\vec{r},E)\,\varphi(\vec{r},E)\, I^2(\vec{r}) \qquad I(\vec{r}) = \int dE'\, \frac{\chi(\vec{r},E')}{4\pi}\, \varphi^\dagger(\vec{r},E)$$

- Note that the importances I_0 and I are weighted by adjoint flux $\varphi^\dagger$

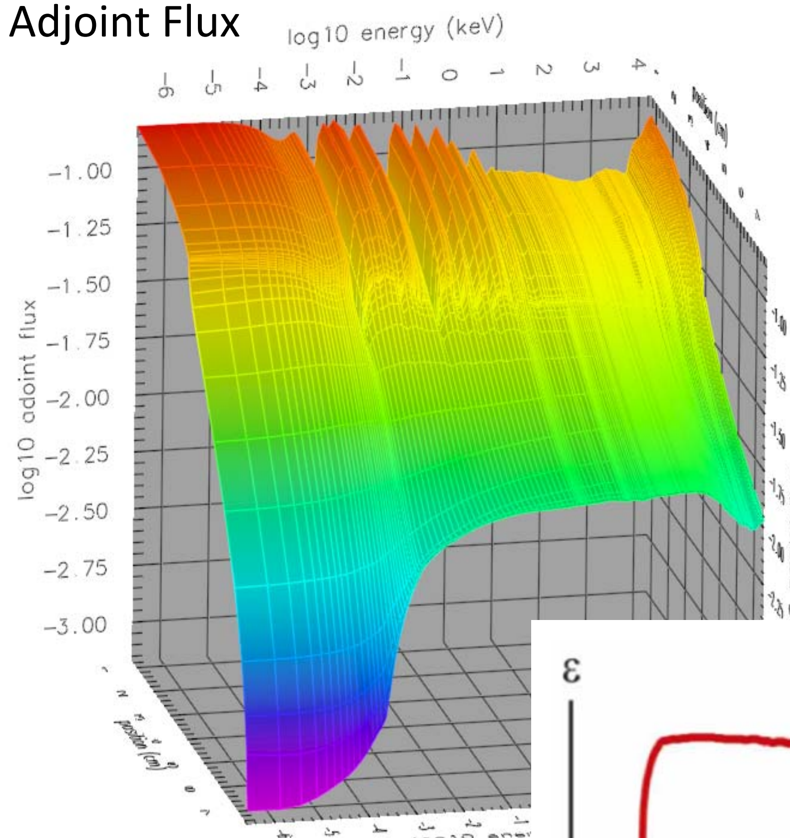
Static Forward and Adjoint Solutions

LANL BeRP Ball / 3.8 cm Poly

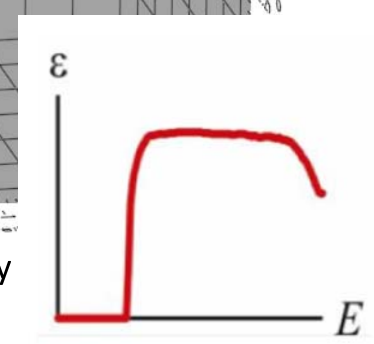
Forward Flux



Adjoint Flux



Adjoint source: detector efficiency on problem external boundary



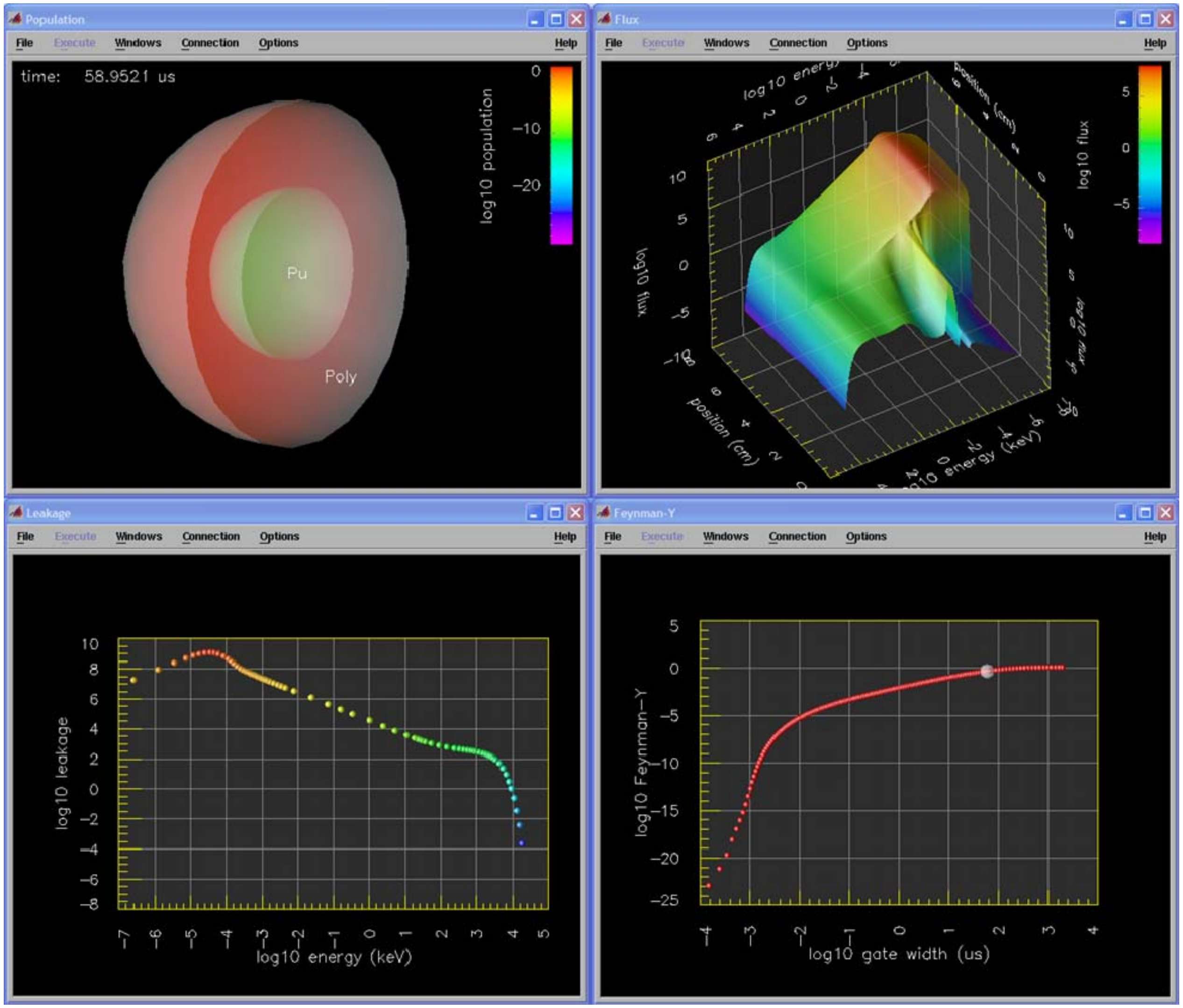
Computation of Feynman-Y Shape

- The Feynman-Y shape is computed from the solution of the forward dynamic step response problem

$$Y(T) \propto \frac{1}{T}\int_0^T dt \int_0^t dt' \, h(t-t')\Sigma_d(\vec{r},E)\,\varphi(\vec{r},E,t')$$

- The LANL transport solver PARTISN is used to compute the flux φ in response to an instantaneous step in the forward source term Q
- The time-dependent flux is convolved with the detector cross-section Σ_d and impulse response h
- The convolution is integrated over coincidence gate width T

Computation of Dynamic Response



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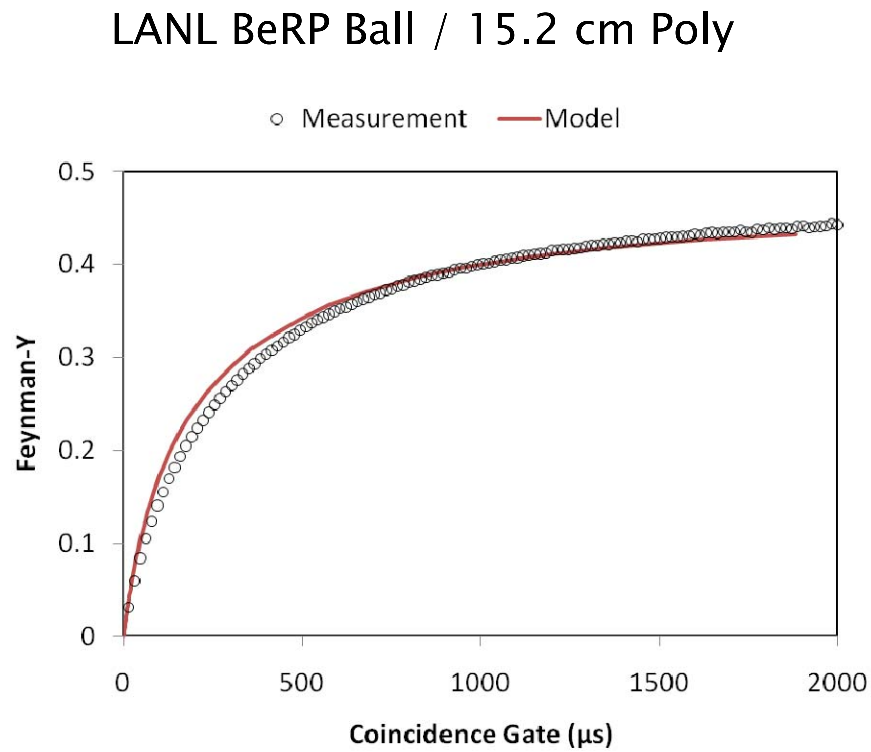
Test Results

- Neutron multiplicity calculations were tested against six measurements of LANL BeRP ball
  - Bare
  - Reflected by polyethylene shells 1.3, 2.5, 3.8, 7.6, and 15.2 cm thick
- The calculations match the measurements within the measurement uncertainty
- Computational times range between 1 and 15 seconds on a standard laptop
- The total computational time dominated by the dynamic response calculation, which is primarily dictated by the neutron lifetime



Conclusions

- It is possible to infer the configuration of an unknown radiation source from its radiation signatures

- Solutions based on multiple complementary signatures are generally better constrained

- Sandia is developing methods to solve for source configuration using gamma spectrometry and neutron multiplicity signatures

- Sandia developed a fast method to accurately compute the Feynman-Y neutron multiplicity counting statistic
- Our method is based on the original work by Muñoz-Cobo, Perez, and Verdú
- Its implementation uses the LANL-developed, time-dependent transport solver PARTISN

- Initial test results confirm the method's accuracy and potential speed

Ongoing Work

- We are working with the University of Florida to increase the speed of our neutron multiplicity calculations
  - Exploring alternative cross-section collapsing methods
  - Exploring different transport solver options using PENTRAN and TITAN
- We are working with the University of Michigan to benchmark the accuracy of our neutron multiplicity calculations
  - Validating MCNP-PoliMi against measurements of SNM
  - Using MCNP-PoliMi to validate deterministic transport calculations for cases where no measured data is available
- We are developing a systematic approach to combine neutron multiplicity and gamma spectrometry in nonlinear regression