Introduction to MCNPX-PoliMi

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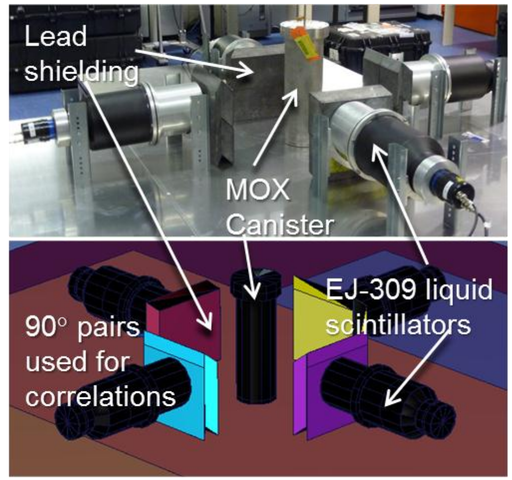
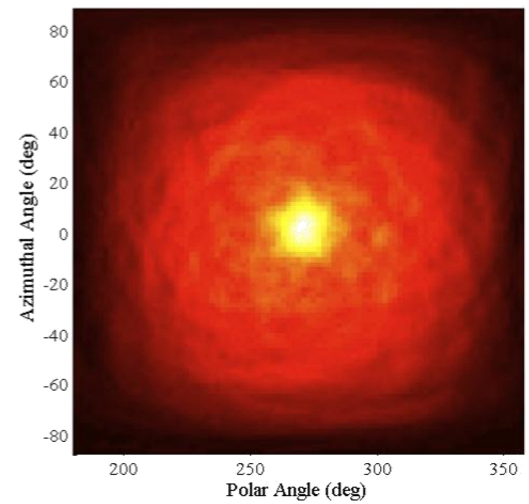
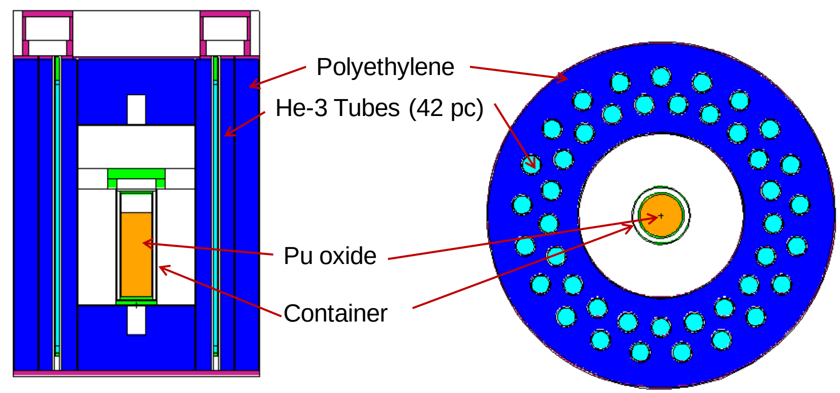
Motivation

- Complex measurement systems are needed to detect nuclear material among complex backgrounds





- Accurate simulations are needed to design and analyze data from these measurement systems



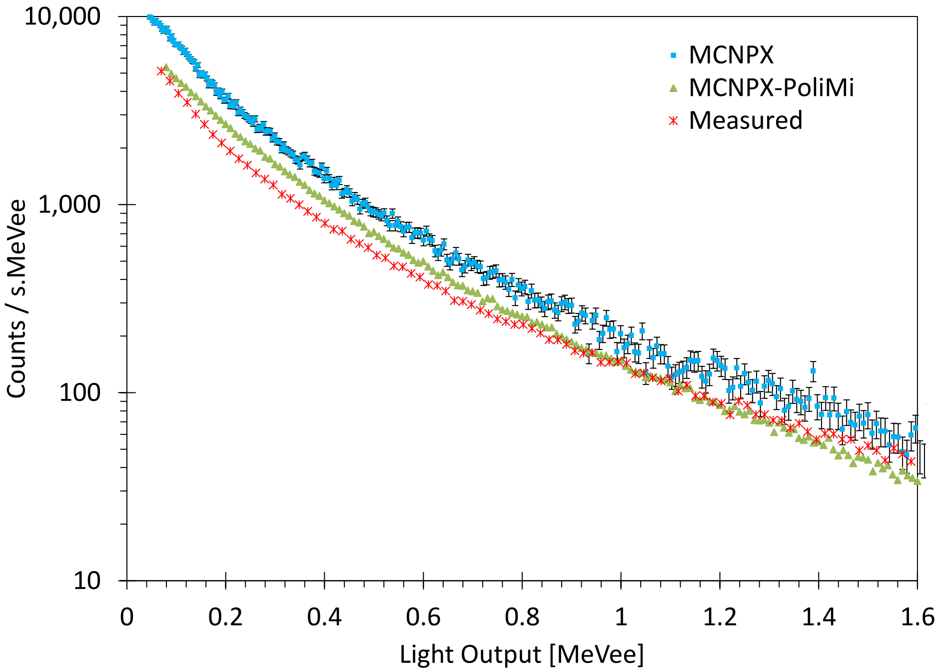
MCNPX-PoliMi Introduction

History

- Version 1.0 released through RSICC (2004)
  - Developed to model event-by-event and correlation physics
  - Used in approximately 50 institutions
- Version 1.2.4 (2008)
  - Integration with a modified version of MCNPX to model photonuclear physics reactions
- Version 1.3.6 (2010)
  - Conversion to MCNPX v.2.6.0
- Version 1.4.2 – Version 1.4.9 (2010 – present)
  - Several small changes to improve performance and usability
    - Corrected anisotropy treatment
    - Allowed for cadmium gamma cascade
    - Corrected induced fission neutron energy spectrum

Motivation
Simulating Detector Response

- MCNPX-PoliMi performs the particle transport and provides tallies and event-by-event information
- This detailed information allow subtle detector effects to be explicitly treated
  - Varying light output on hydrogen or carbon
  - Non-linearity of light output response
  - Pulse generation time
  - Dead time



Pulse height comparison of a ~330000 n/s ^{252}Cf source measured with EJ-309 and polyethylene shielding

MCNPX result obtained using a modified neutron pulse height tally.
Measurement performed by Shikha Prasad

MCNPX-PoliMi Introduction
Unique Features

- MCNPX-PoliMi was developed to simulate correlation measurements with neutrons and gamma rays
- Unique features:

1. Physics of particle transport (MCNPX-PoliMi code)
   - Prompt neutrons and gamma rays associated with **each event** are modeled explicitly; neutron and photon-induced **fission multiplicity** distributions have been implemented

Improved simulation of correlation and multiplicity distributions

2. Physics of detection (Detector Response Module)
   - **Each collision** in the detector is treated individually

Improved simulation of detector response

MCNPX-PoliMi Introduction

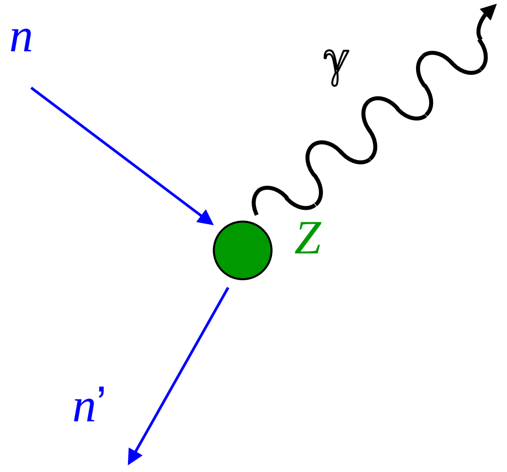
Secondary Gamma-Ray Physics

MCNPX

1. Incident neutron interacts
2. Secondary gamma rays produced
3. Neutron interaction type determined

MCNPX-PoliMi

1. Incident neutron interacts
2. Neutron interaction type determined
3. Secondary gamma rays produced



MCNPX-PoliMi Introduction

Example: 15-MeV Neutrons on ^{12}C

	Elastic		(n,n')		Absorption		Total	
	MCNP	physics	MCNP	physics	MCNP	physics	MCNP	physics
Inelastic photons	146,223	0.	77,317	236,550.1	13,837	0.	237,377	236,550.1
Capture photons	164	0.	99	0.	14	304.7	277	304.7
No photons	1,096,489	1,242,561.8	583,434	424,700.1	103,875	117,335.3	1,783,798	1,784,597.2
Collisions	1,242,876	1,242,561.8	660,850	661,250.2	117,726	117,640.0	2,021,452	2,021,452.0

MCNPX-PoliMi Source Capabilities
Overview

- Many neutron sources are used in various applications
  - Spontaneous fission (^{252}Cf, ^{240}Pu)
  - Alpha-neutron reactions (AmBe, AmLi, PuBe)
  - Neutron generators (DD, DT)
- The energy spectrum of these source can be quite complex
- MCNPX-PoliMi contains energy and number distributions for several important sources

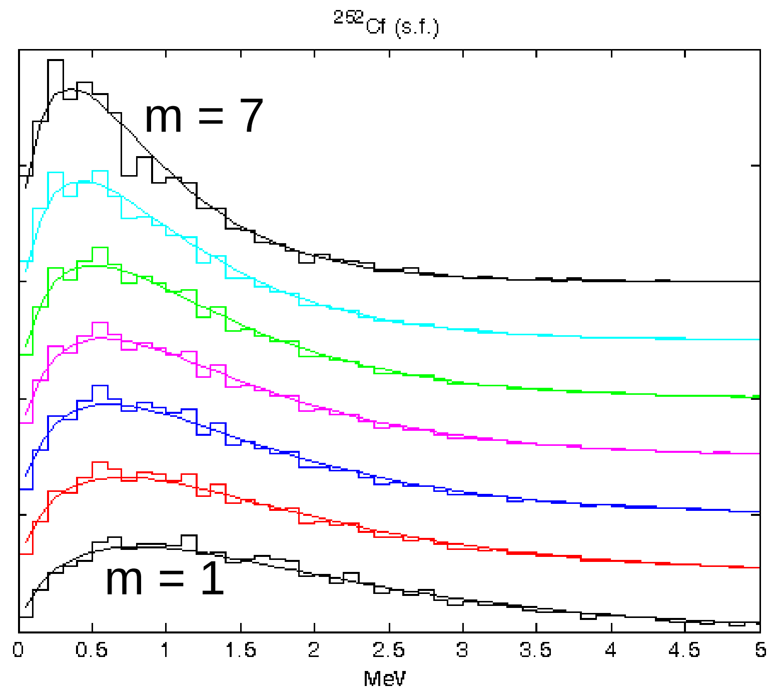
MCNPX-PoliMi Source Capabilities
Spontaneous Fission Sources

- MCNPX-PoliMi contains spontaneous fission source definitions for several common isotopes: ^{252}Cf, ^{238}U, ^{240}Pu, ^{242}Pu, ^{242}Cm, ^{244}Cm
- Each source history emits multiple neutrons and gamma rays
  - The energy and multiplicity distributions are unique for each source isotope
  - The energy of each neutron is *correlated* with multiplicity
- Anisotropic angular treatments is also available
  - Isotropic treatment is available for debugging

MCNPX-PoliMi Source Capabilities

Spontaneous Fission Energy Distributions

- The energy distributions of spontaneous-fission neutrons and gamma rays are independently sampled
- The data contained in MCNPX-PoliMi are based on Gamma-distribution fits to relevant sets of measured data
    - S. Lemaire, P. Talou, T. Kawano, M.B. Chadwick and D. G. Madland. *Monte-Carlo approach to sequential neutron emission from fission fragments.* Phys. Rev. C 72, 024601 (2005)



The neutron spectra shift based on the number of emitted neutrons

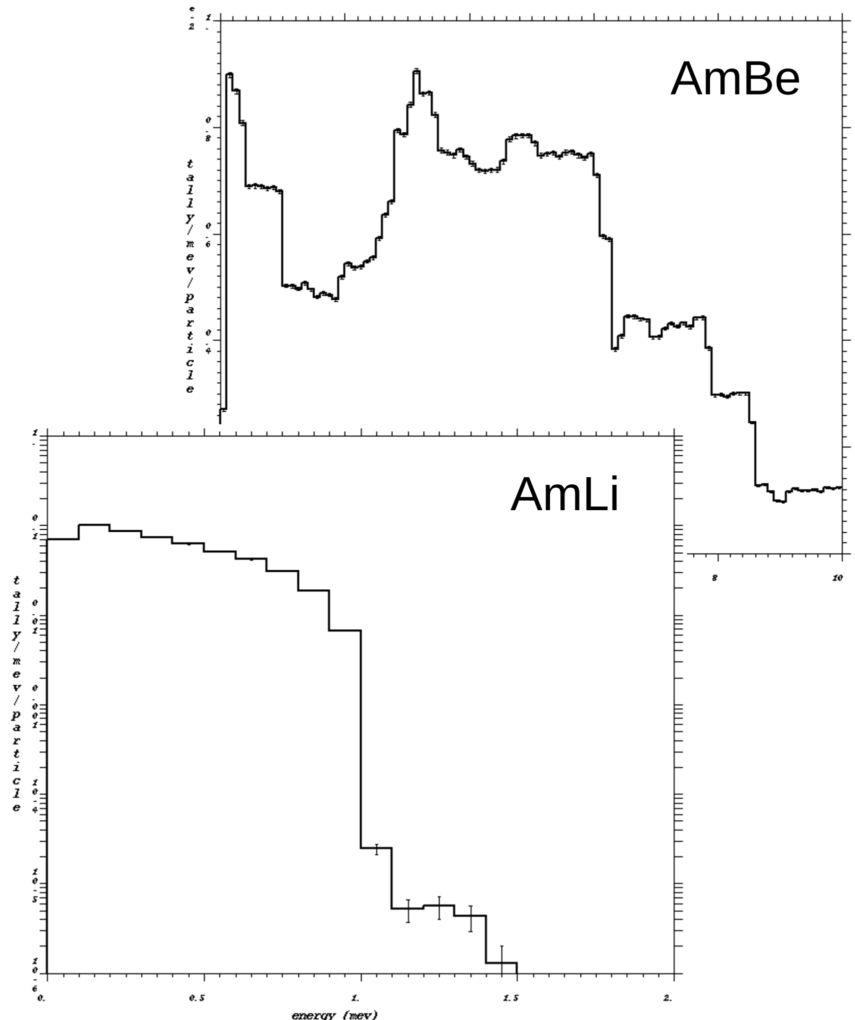
MCNPX-PoliMi Source Capabilities
Spontaneous Fission Angular Distributions

- Anisotropic angular emission of neutrons is available, and recommended for each spontaneous fission source
- The distribution was taken from a program by T. Valentine:
  - T. Valentine, “MCNP-DSP Users Manual,” *ORNL/TM-13334,R2* ***(2001).***
- The direction of each particle is sampled independently from any other parameters
- A completely isotropic distribution is available for debugging purposes

MCNPX-PoliMi Source Capabilities

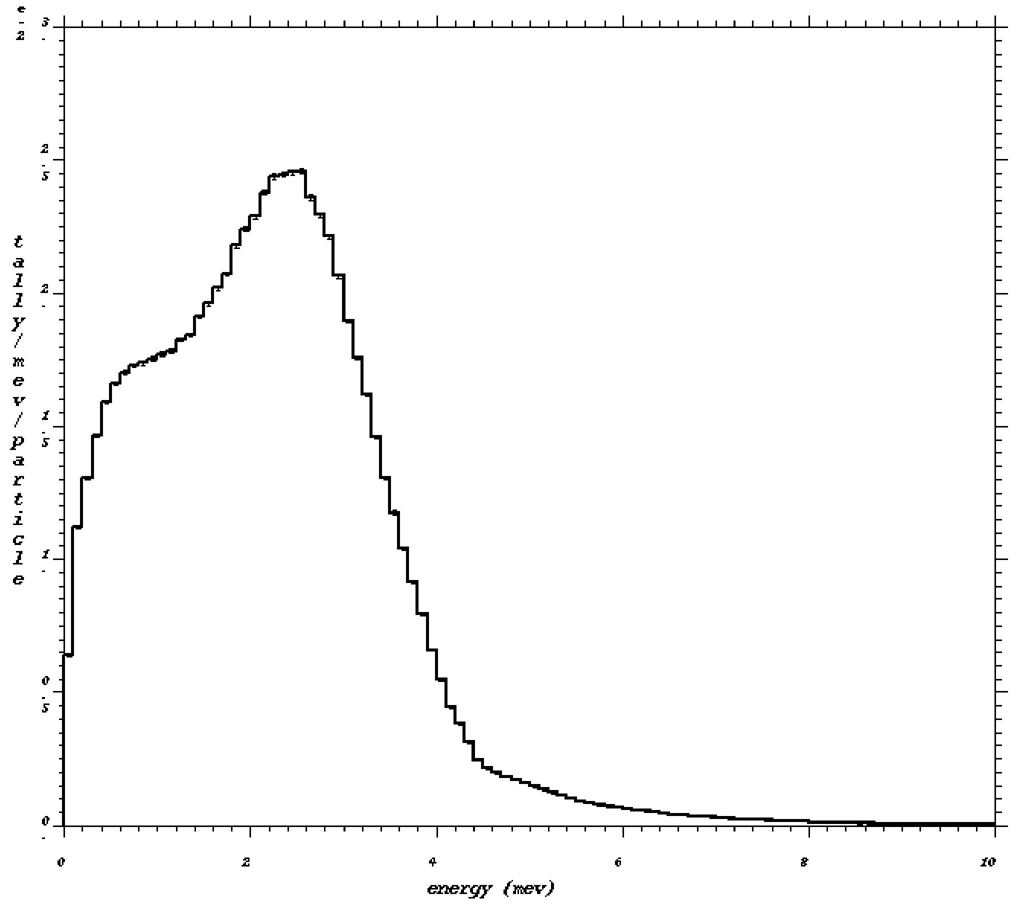
Alpha-n Sources

- Available sources: $^{238}PuO_2$, $^{239}PuO_2$, $^{240}PuO_2$, $^{241}AmO_2$, and AmLi, AmBe
- Many gamma rays are emitted from α-decays that do not produce a neutron
- MCNPX-PoliMi has the ability to simulate all gamma rays or only those gamma rays that are correlated with neutrons
  - This treatment is recommended unless the user has explicit interest in uncorrelated gamma-ray background



MCNPX-PoliMi Source Capabilities
Mixed Source (^{240}Pu(sf) and $PuO_2(\alpha, n)$)

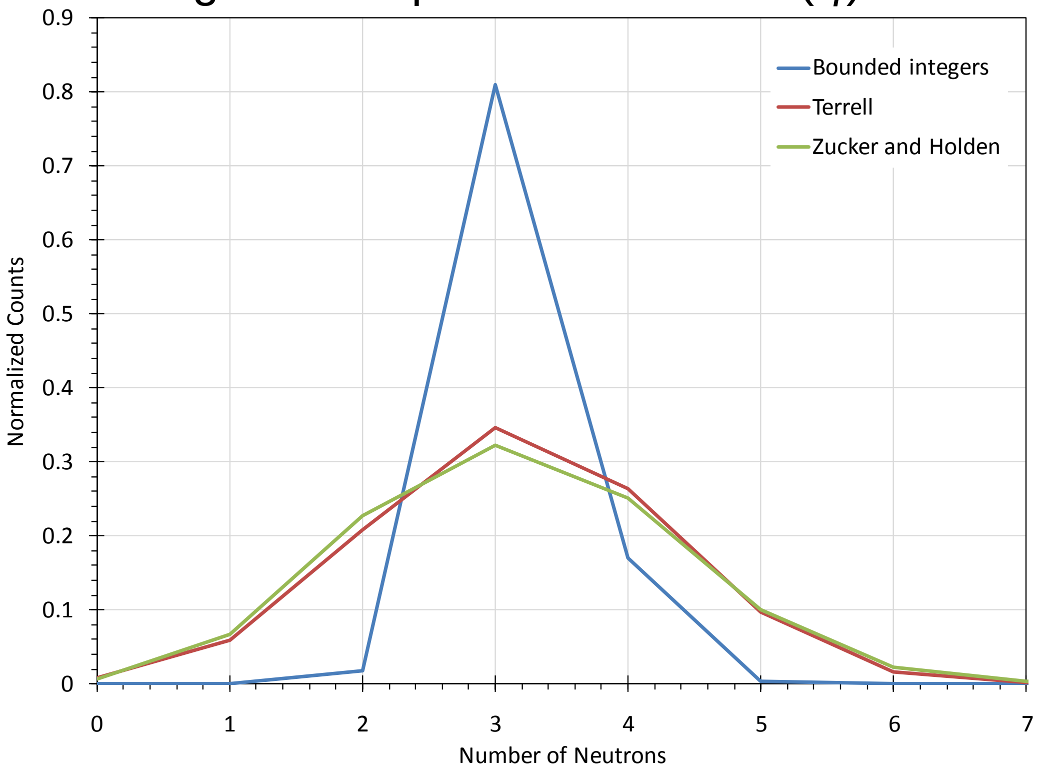
- MCNPX-PoliMi has the ability to combine multiple sources into a single distribution
- This capability has been used in the past to effectively model MOX fuel samples



MCNPX-PoliMi Capabilities
Neutron-Induced Fission Multiplicity

- Neutron multiplicity distributions are available from the following measured data sets:
  - Terrell (^{235}U, ^{238}U, ^{239}Pu)
  - Zucker and Holden (^{235}U, ^{238}U, ^{239}Pu)
  - Gwin et al. (^{235}U only), otherwise Zucker and Holden

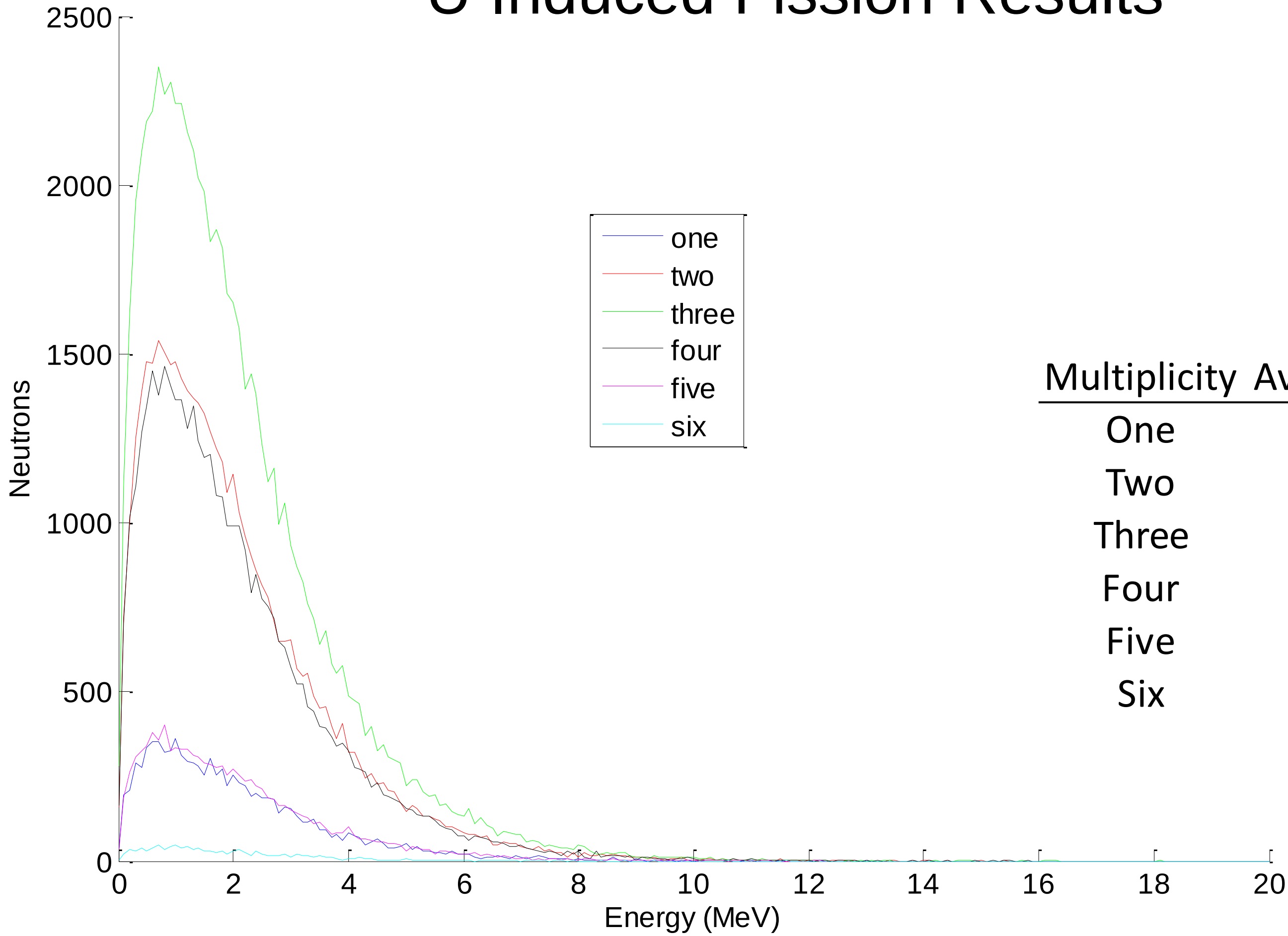
4.5-kg WGPu sphere with a $^{240}Pu(sf)$ source



MCNPX-PoliMi Capabilities

Fission-Neutron Energy Distributions

^{235}U Induced Fission Results



Multiplicity	Average Energy (MeV)
One	2.051
Two	2.050
Three	2.048
Four	2.044
Five	2.041
Six	1.986

Physics Options

Delayed γ-Ray Generation Models

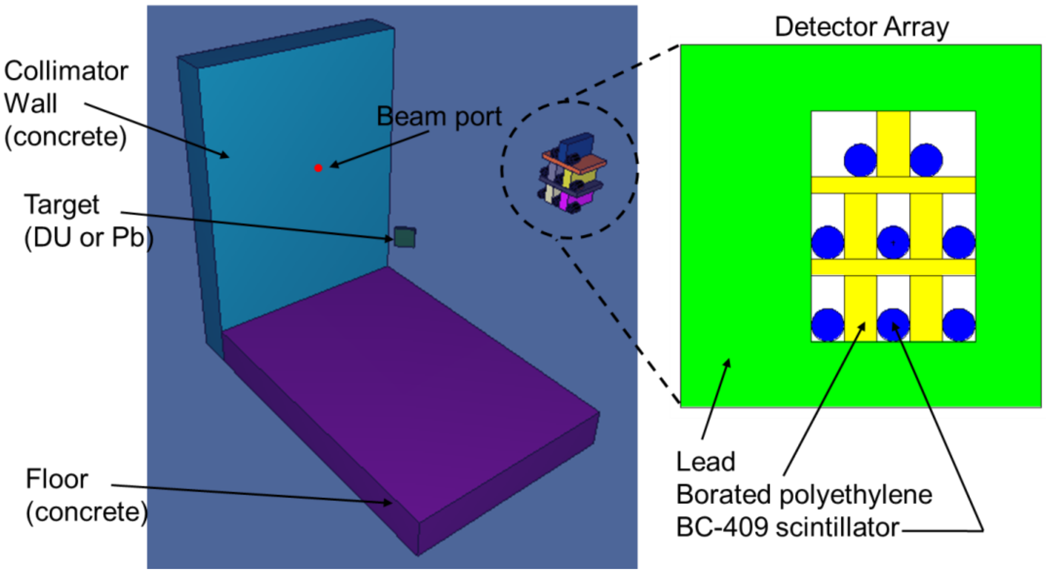
- The number of gamma-rays produced in a fission event ranges from 0 to 23
- Photon energies are sampled independently from appropriate energy distributions
- Approximately 18% of fission gamma rays are emitted at times greater than 1 ns after fission
  - This time distribution can be sampled using the model reported by Vandenbosch and Huizenga, *Nuclear Fission*, 1973

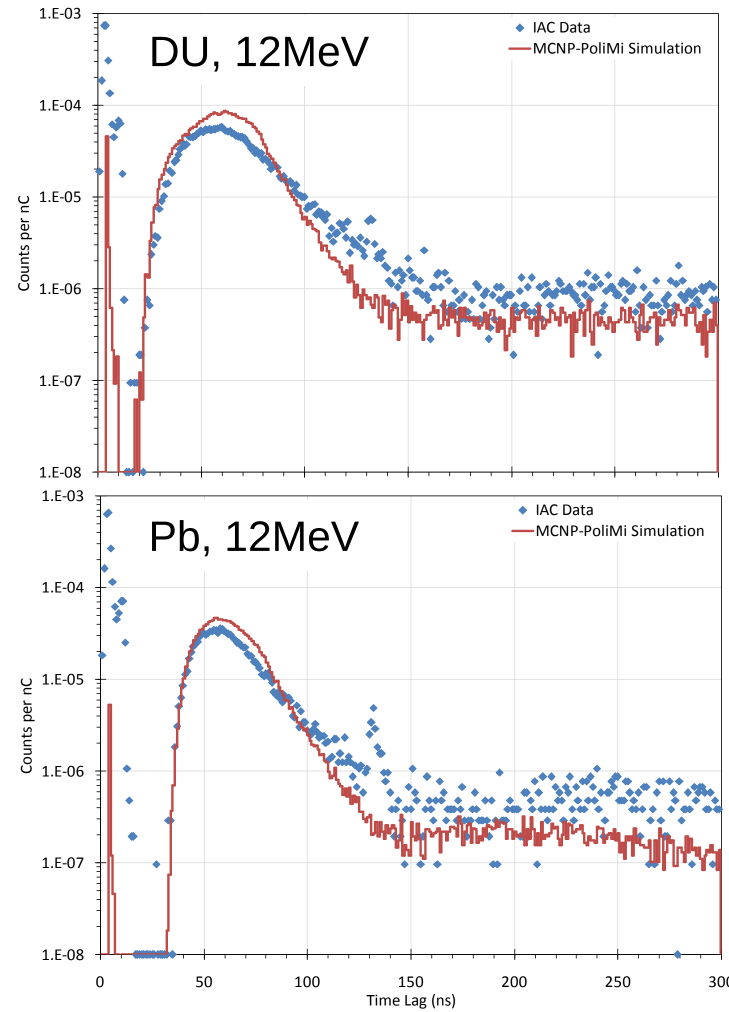
MCNPX-PoliMi Capabilities
Photofission Options

- Previous versions of MCNP-PoliMi required a modified version of MCNPX to transport the source photons to the target
- Upon interaction, a log file is written containing the location and type of interaction
  - For (γ, xn) reactions, all products are also written to the file
  - For fission, MCNP-PoliMi uses enhanced multiplicity distributions to sample all of the products
- MCNPX-PoliMi provides the ability to perform photonuclear calculations in a single simulation
  - The log file can be used as a variance reduction technique

MCNPX-PoliMi Capabilities
Photofission Options

- LINAC-based experiments have been performed at the Idaho Accelerator Center





MCNPX-PoliMi Capabilities

Data Printout File Excerpt

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
History	Particle Number	Particle type	Interaction	ZAID	Cell	Energy Released (MeV)	Time (Shakes)	X-Position	Y-Position	Z-Position	Weight	Generation Number	Number of Scatters	Code	Energy Prior to Collision (MeV)
105	7	2	1	6	6	0.1695	0.13	7.31	2.49	-1.39	1	0	0	0	0.3378
129	7	1	-99	1001	6	3.5291	1.11	1.12	3.4	-4.63	1	0	0	0	4.237
129	7	1	-99	6000	6	0.1897	1.27	1.63	4.52	-3.28	1	0	1	0	0.7079
136	4	2	1	6	6	0.0942	0.09	0.53	1.03	-5.72	1	0	0	0	0.2343
141	7	1	-99	1001	6	0.1534	3.49	1.31	0.57	-2.12	1	0	0	0	0.3977
141	7	1	-99	6000	6	0.0503	3.81	2.95	-0.14	-3.42	1	0	1	0	0.2443
141	7	1	-99	1001	6	0.1350	3.84	2.86	0	-3.5	1	0	2	0	0.194

Particle Type
- 1 = Neutron
- 2 = Photon

Interaction
- 1 = Compton Scattering
- -99 = Elastic Scattering

ZAID (Material Identification Number)
- 6 and 6000: Carbon (Natural)
- 1 and 1001: Hydrogen

MCNPX-PoliMi Data File

History and Particle Numbering

- The first two columns in the data file contain the MCNPX numbering of each particle undergoing a collision
- <u>Column 1:</u> history number (NPS)
  - Tracks the number of source histories initiated
  - In MCNPX-PoliMi, each source history can consist of multiple particles
- <u>Column 2:</u> particle number (NPAR)
  - Tracks the number of particles within each source history
  - Incremented each time a new particle, of any kind, is created and restarts at 1 with each new source history
  - Particles in multi-particle sources are uniquely numbered

MCNPX-PoliMi Data File

Interaction Data

- The next four columns contain interaction information
- Column 3: particle type (IPT)
  - Type of particle undergoing the collision: neutron (1), photon (2), or electron (3)
- Column 4: reaction (NTYN)
- Column 5: nucleus (ZAID)
  - The ZAID of the target nucleus
  - The material number if IPT=3
- Column 6: cell number (NCL)
  - The MCNPX-PoliMi cell number in which the collision occurs

NTYN	Interaction type
	Neutrons
0	Capture
-99	Elastic scattering
99	S(α,β) collisions
±1	Inelastic scattering
19	Fission
±X	(*n*, *Xn*)
	Photons
1	Compton scattering
2	Coherent scattering
3	Capture with fluorescence
4	Capture with double fluorescence
5	Pair production

MCNPX-PoliMi Data File

Collision Data

- The next five columns contain interaction data from the collision
- Column 7: energy deposited (EnReCo)
  - Energy deposited in the collision in MeV
  - In the case of fission printed number is the *Q*-value
- Column 8: time of collision (TME)
  - Absolute time if the collision in shakes (1 sh = 10 ns)
- Columns 9-11: x, y and z coordinates (XXX,YYY,ZZZ)
  - Location of the collision in the absolute coordinates of the MCNPX-PoliMi geometry in cm

MCNPX-PoliMi Data File
Particle Weight

- Column 12 contains the weight (WGT) of the particle undergoing the collision
- In order to use current detector post-processing algorithms WGT should always be 1.0
- Particle transport must be completely analog
  - This means that variance-reduction techniques cannot be applied and neutron implicit capture must be off
- Development is underway to relax this restriction

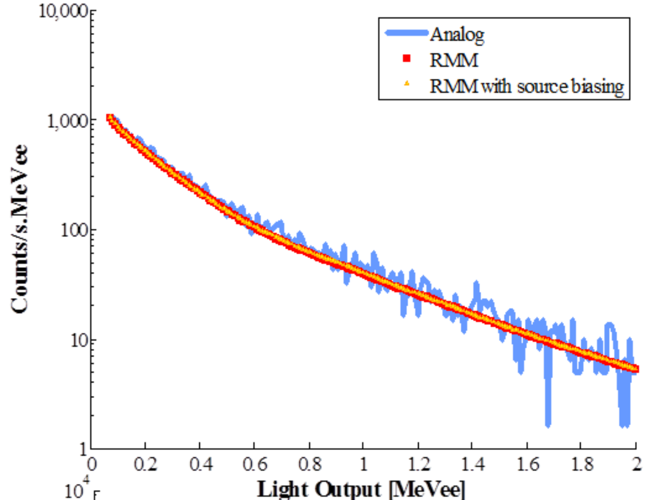
Non-Analog MCNPX-PoliMi

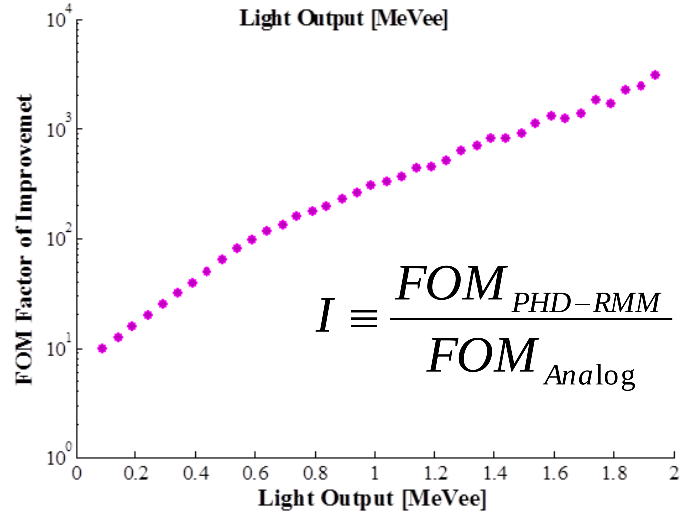
Preliminary Results

- An incident neutron current tally may be combined with a response matrix to calculate detector response

$$N(L) = \sum_{n=1}^{N} F1_{MCNP}(E_n) \cdot R(E_n, L)$$

- Traditional variance reduction techniques can be used to calculate neutron current tally

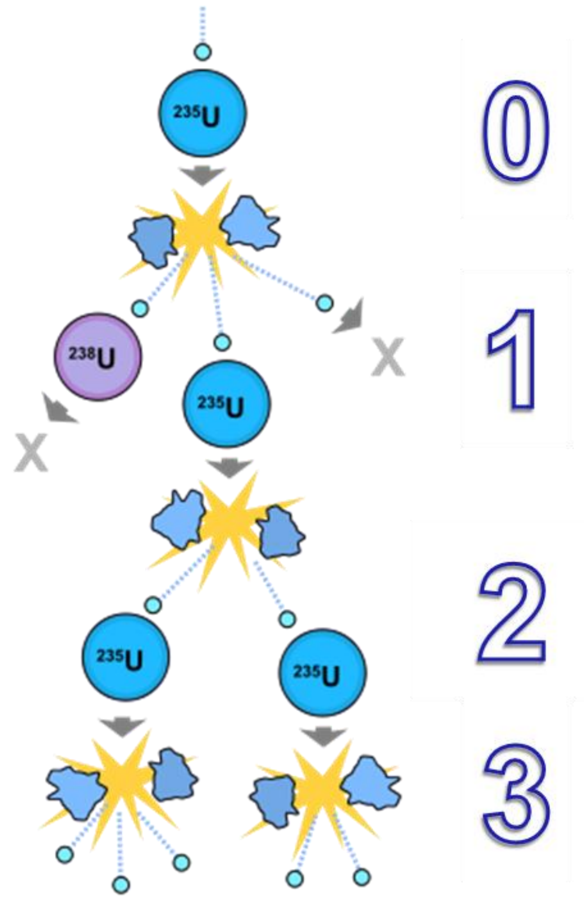




MCNPX-PoliMi Data File

Particle Generation Numbers

- <u>Column 13</u> contains the generation number (NGEN) of the particle undergoing the collision
- Source particles have an assigned generation number of zero
- The generation number is incremented at each fission event



MCNPX-PoliMi Data File
Incident Particle Data

- The last three columns contain information about the incident particle
- <u>Column 14:</u> number of scatterings (NSCA)
  - The total number of scatterings (elastic or Compton) the incident particle has had since birth
- <u>Column 15:</u> incident particle code (NCODE)
  - Code describing the reaction that produced the incident particle (if any)
- <u>Column 16:</u> incident particle energy (ERG)

MCNPX-PoliMi Data Printout Options

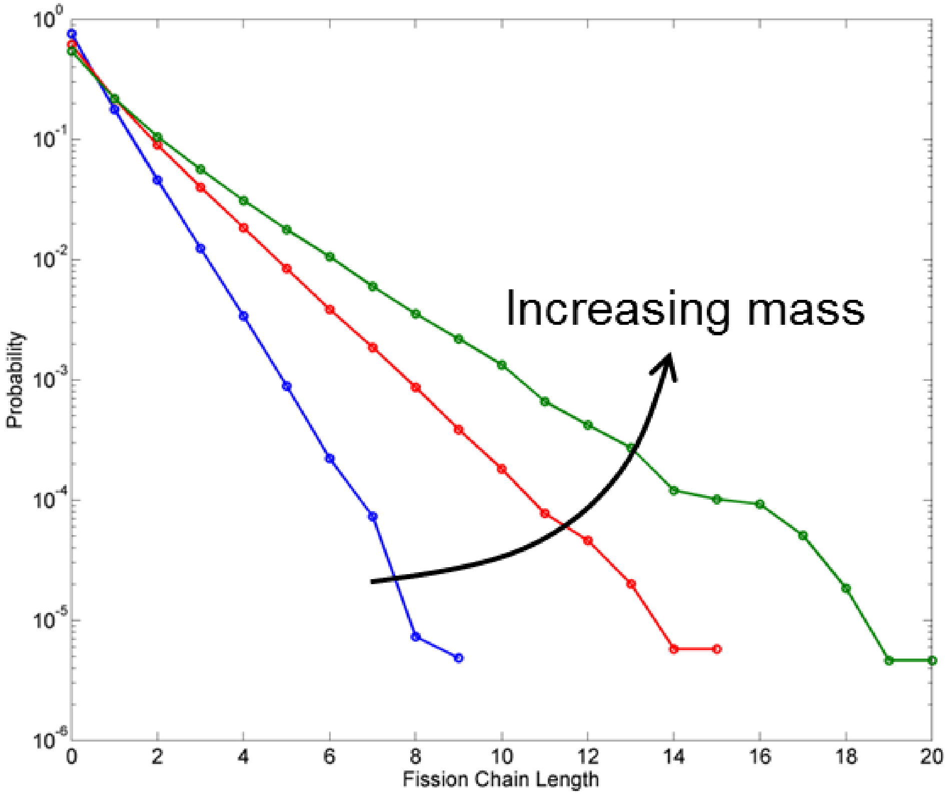
Multiplicity Data File

- The fission multiplicities of neutrons and gamma rays can be printed for each fission event
- A four-column data file is printed:
  1. History number
  2. Generation number
  3. Number of neutrons
  4. Number of photons

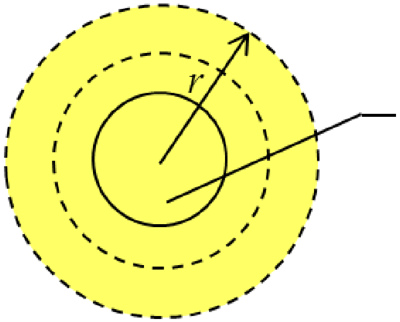
NPS	Gen Num	N	P
1	0	1	3
2	0	3	6
2	1	3	8
2	2	1	4
2	2	2	12
2	3	1	9
2	1	3	5
3	0	0	6
4	0	1	10
5	0	3	4
5	1	3	7
5	2	2	9
5	2	4	11

MCNPX-PoliMi Data Printout Options
Fission Generation Analysis

- In MCNPX-PoliMi, data may be printed from cells other than detectors
- This allows for detailed physics analysis in any cell of the simulation



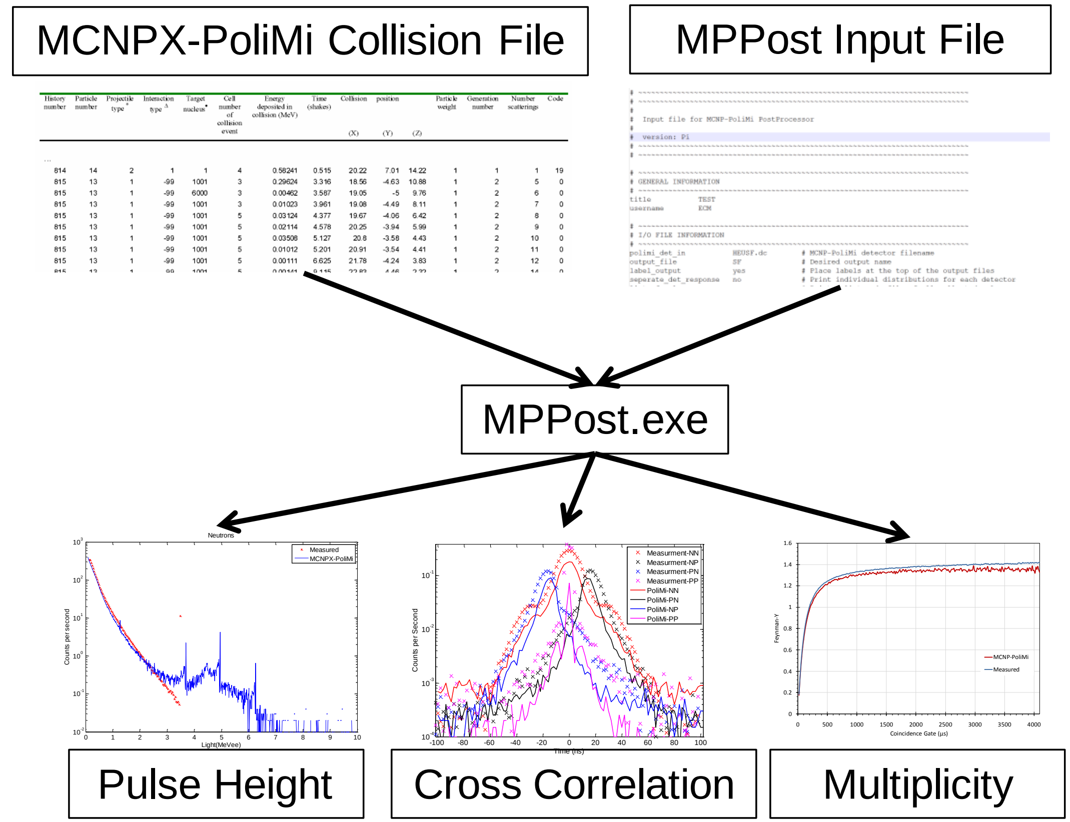
HEU metal sphere (ρ = 18.75 g/cm^3) of increasing mass: 1, 5, and 10 kg



MPPost

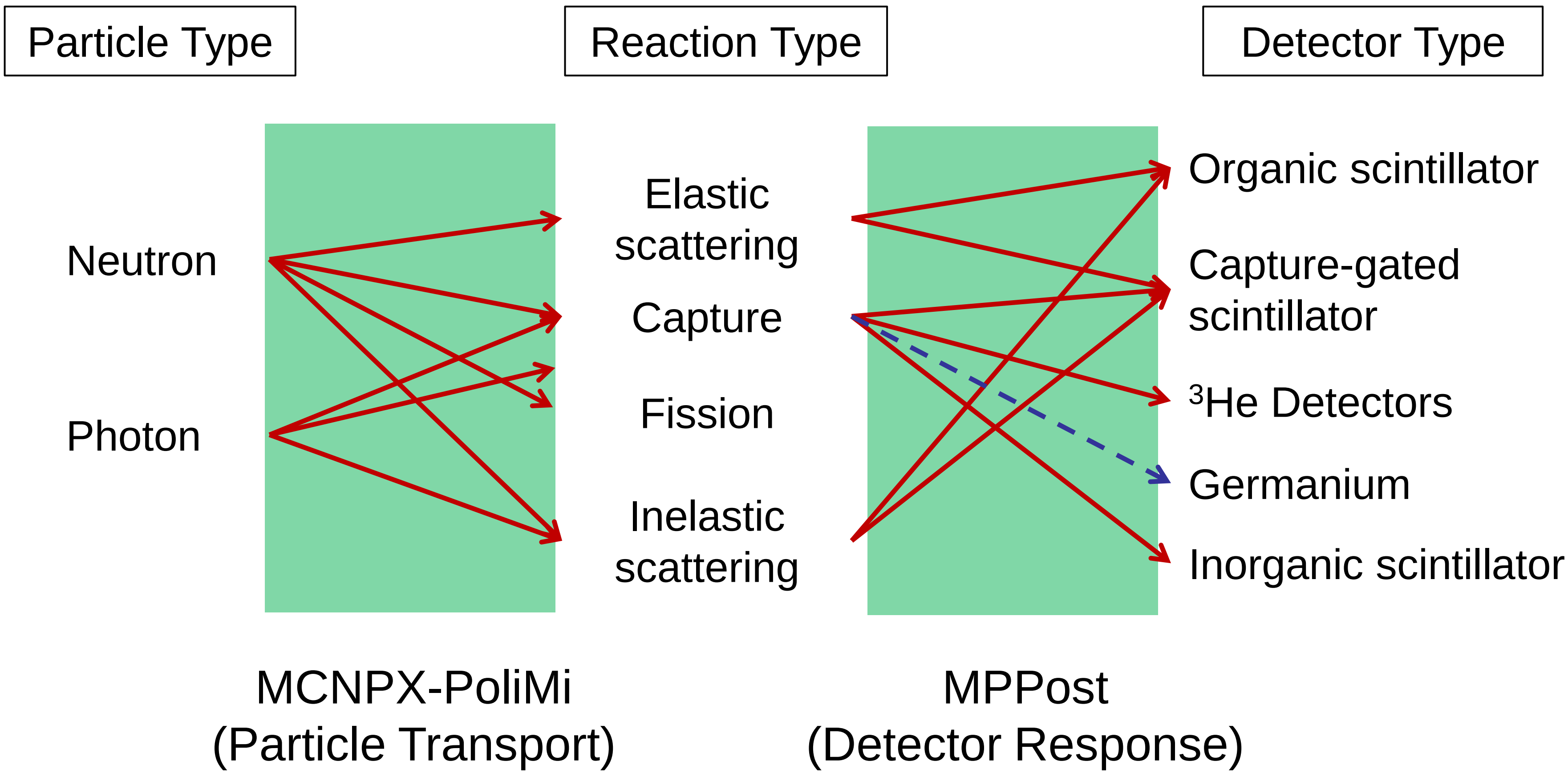
A MCNPX-PoliMi Post-Processor

- Is a Fortran program that has been developed to simulate detector response
- Using the collision file created by MCNPX-PoliMi and user-specified inputs, a wide variety of detector responses can be calculated



MCNP-PoliMi Post-Processor

Existing and Proposed Capabilities

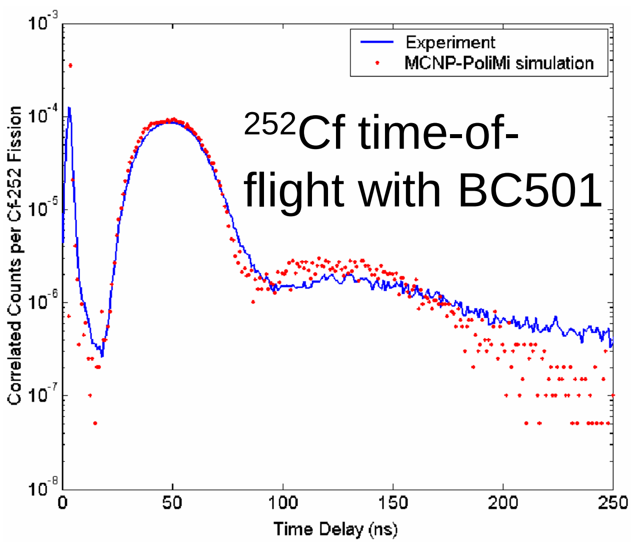


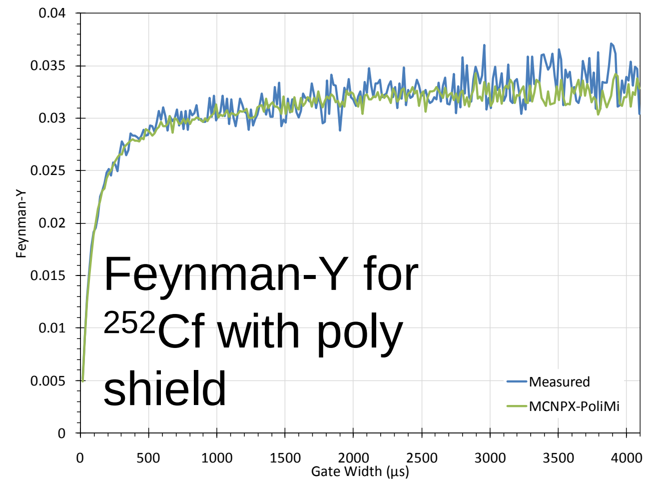
MPPost
Basic Capabilities

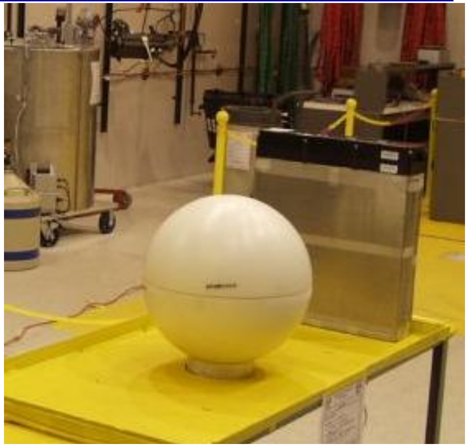
- Simulates the detector response for several common detector types
  - ^{3}He detectors
  - Organic scintillators
  - Inorganic scintillators
- Provides results that can be compared to measured data
  - Pulse height distributions
  - Time-of-flight
  - Cross correlations
  - Multiplicity

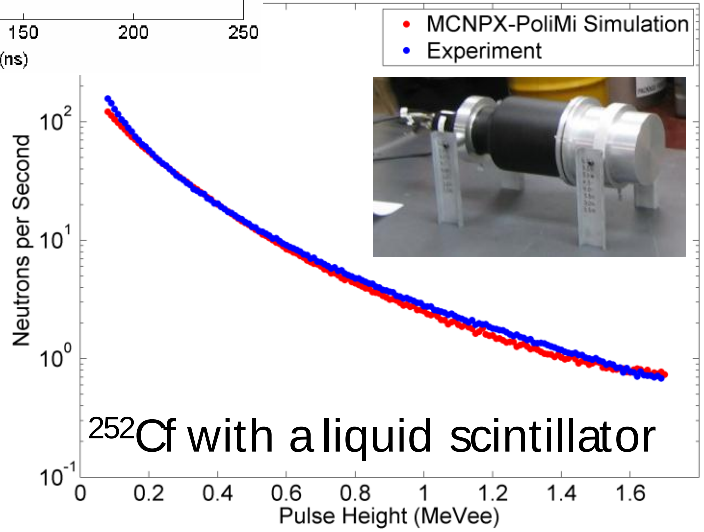
MCNPX-PoliMi Postprocessing

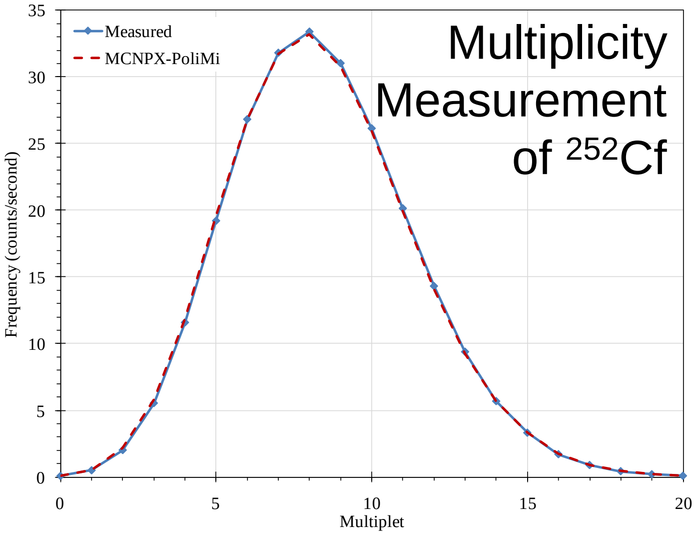
Validation with Measured Data





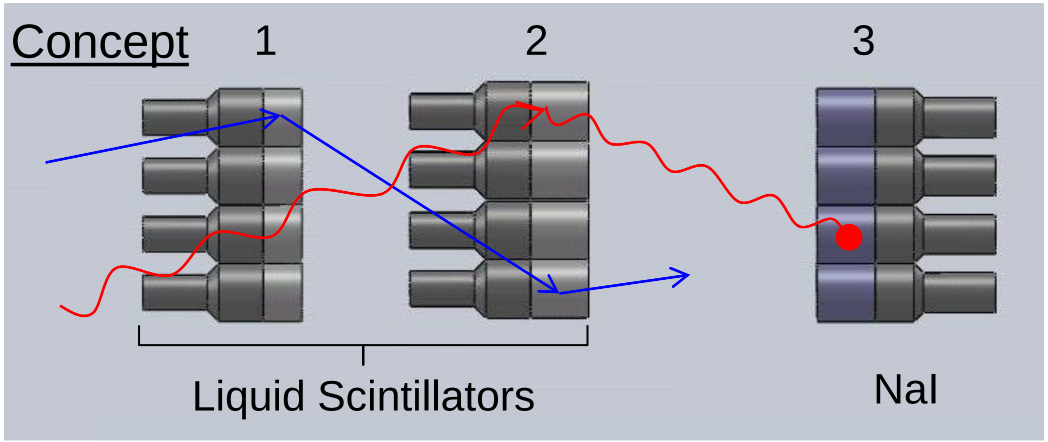






MCNPX-PoliMi Applications

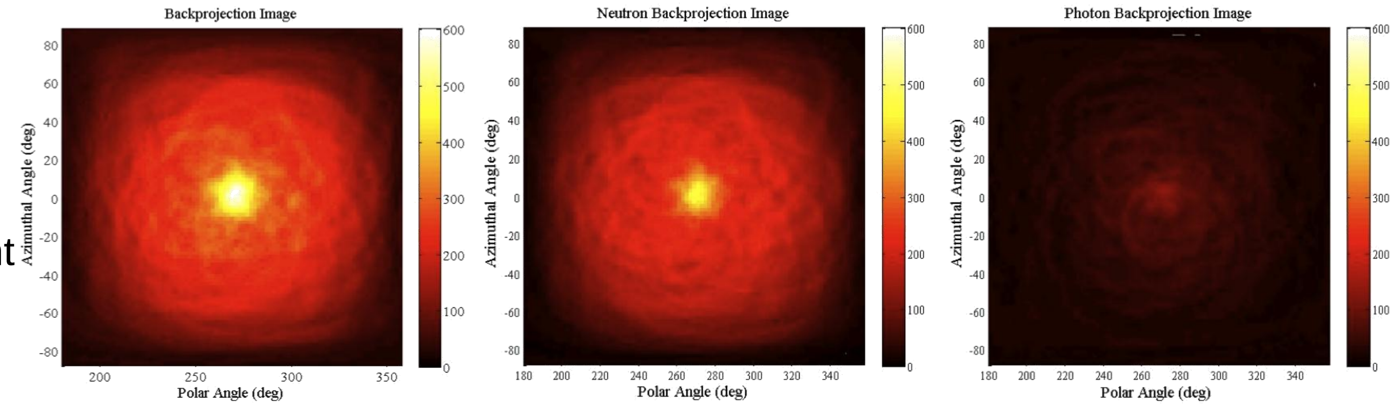
Measurement System Design



- Provides the ability to detect fission sources at a distance standoff
- Neutron and gamma-ray detection provides robustness to source shielding
- The system is fully scalable

Sample reconstructed image - simulated

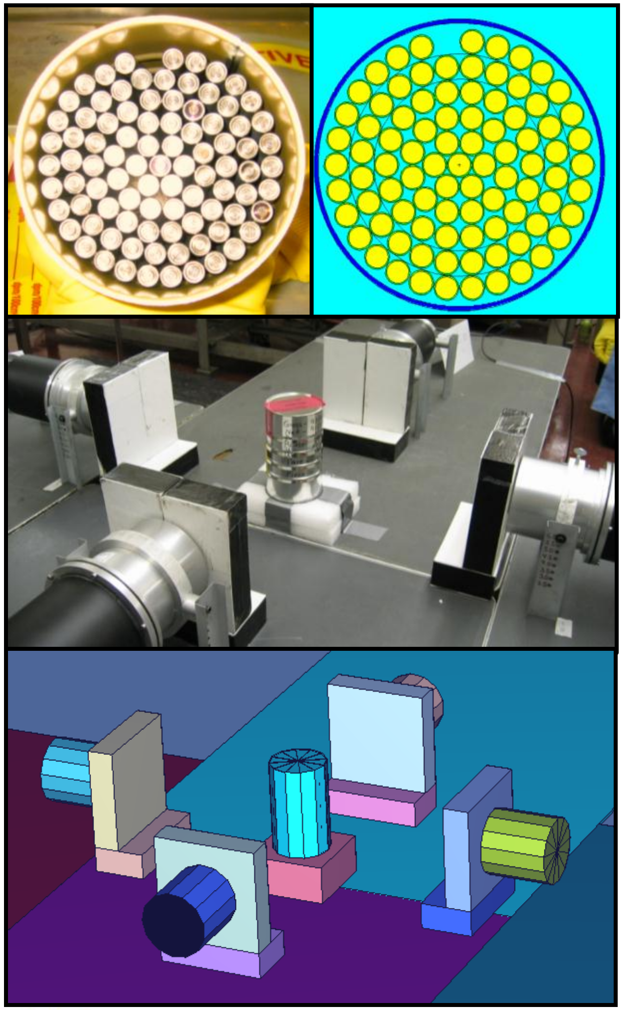
- A ^{252}Cf source emitting 300,000 neutrons per sec was shielded with 5 cm of lead
- A 20-minute measurement was simulated with the source 2.5 m away from the system



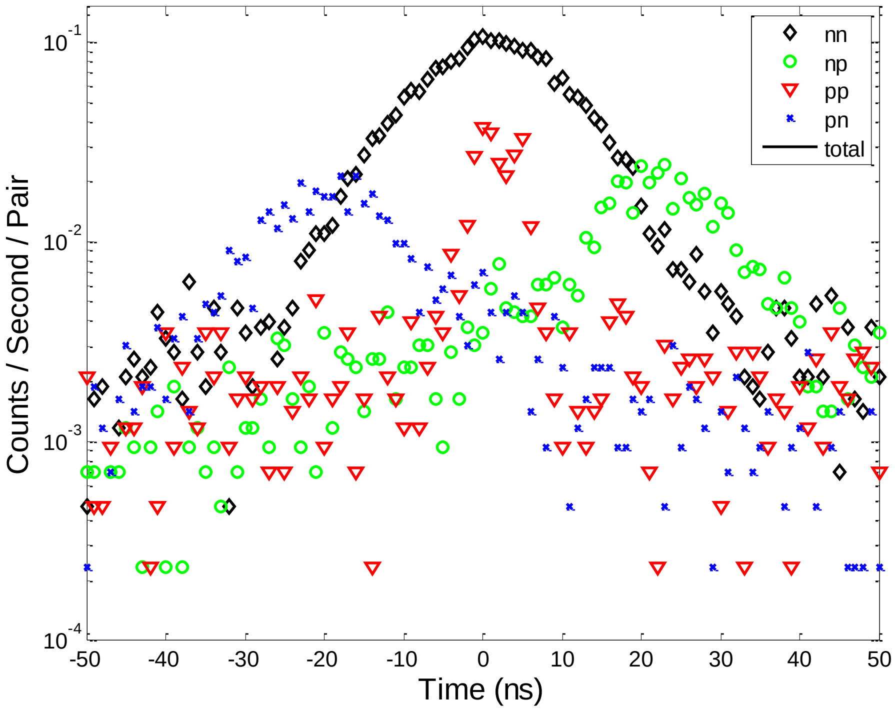
Provisional patent application #2115-004976_US_PS1, S.A. Pozzi et al., 4/6/11

MCNPX-PoliMi Applications

Measurement Planning



- Passive measurements can be used to measure advanced fuel types such as MOX
- Neutron-neutron (n, n) correlations provide information on fission events in the material



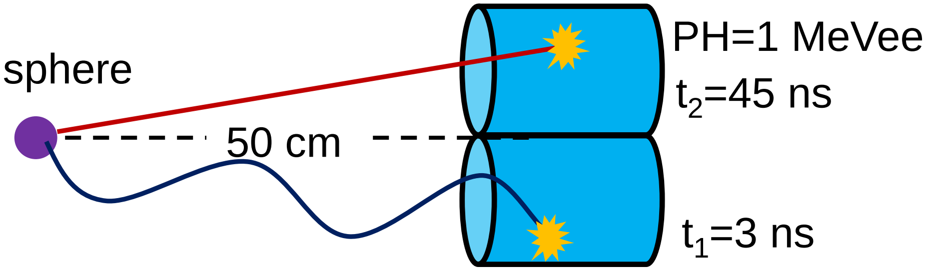
MCNPX-PoliMi Applications
Advanced Data Analysis

- Correlated (γ,n) events can give you information about the multiplication of a source
  - Events with high light outputs are not expected at large times
- For a given time the max pulse height for the neutron can be determined:

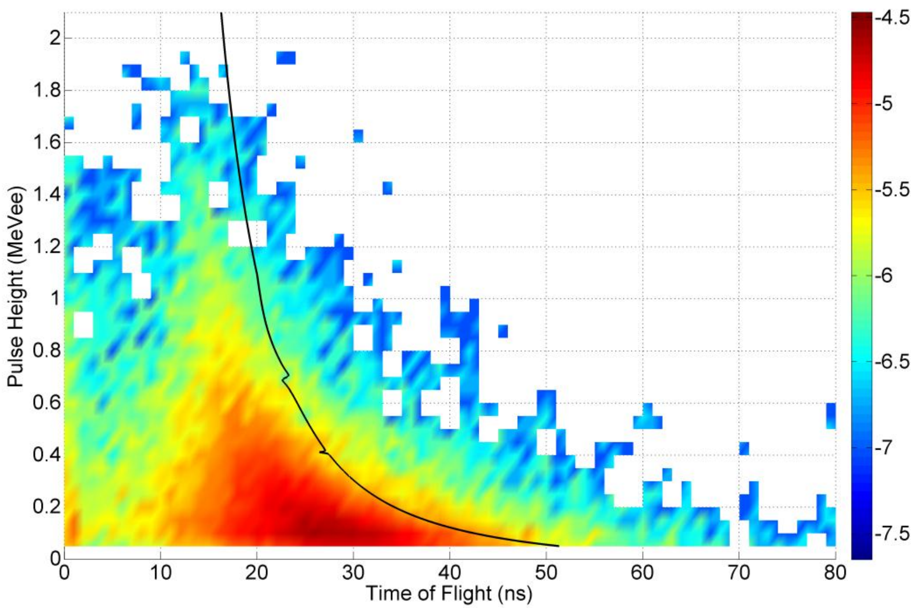
$$E_n = \frac{1}{2} m_n \left(\frac{d}{t}\right)^2$$

$$PH = aE_d^{\ 2} + bE_d + c$$

E_n = neutron energy
E_d = neutron energy deposited
d = source-detector distance
m_n = neutron mass
t = neutron arrival time
PH = light output



TCPH measurement setup



Simulated TCPH for 4.5-kg Pu Metal

Summary

- The MCNPX-PoliMi code provides much-needed to accurately simulate advanced nuclear material detection systems
  - Special attention is paid to the physics of fission
  - Many sources common in nuclear safeguards and nonproliferation are incorporated into the code
  - Event-by-event physics allow for detailed simulation of detector response
- Release of the code through RSICC is underway

MCNPX-PoliMi Training Workshop

- The first MCNPX-PoliMi training workshop was held August 23rd and 24th, 2011 on the UM campus
- The workshop introduced new users to the capabilities of the MCNPX-PoliMi code and acquainted experienced users with new features
    - MCNPX-PoliMi source capabilities
    - Detector-response calculations
    - Simulations of time-of-flight and cross-correlation distributions
    - Simulations of multiplicity distributions
- Another workshop is planned for July 25th and 26th, 2012

Detection for Nuclear Nonproliferation Group

Newly established group at the University of Michigan

Group Leader: Sara Pozzi

Group Members

- Marek Flaska, Assistant Research Scientist
- Shaun Clarke, Assistant Research Scientist
- Andreas Enqvist, Postdoctoral Research Fellow
- Eric Miller, Ph.D. candidate
- Jennifer Dolan, Ph.D. candidate
- Shikha Prasad, Ph.D. candidate
- Christopher Lawrence, Ph.D. candidate
- Alexis Poitrasson-Riviere, Ph.D. Student
- Mark Norsworthy, Ph.D. Student
- Alexis Kaplan, Ph.D. Student
- Marc Paff, Ph.D. Student
- Kyle Polack, Ph.D. Student
- Mark Bourne , Ph.D. Student
- Kiyotaka Ide, Graduate Student
- Brian Wieger, Masters student
- Kyle Weinfurther, Masters student
- Michael Hamel, Masters student
- Randy Schiffer, Masters student
- Bill Walsh, Graduate Student
- 6 Undergraduate Students

Collaborators – National

- Vladimir Protopopescu, Oak Ridge National Laboratory
- Alan Hunt, Idaho Accelerator Center
- Donald Umstadter, University of Nebraska
- Peter Vanier, Brookhaven National Laboratory
- John Mattingly, North Carolina State University
- Andrey Gueorgueiv, RMD Technologies
- Larry Reees, Bart Czirr, Brigham Young University
- Peter Vanier, Brookhaven National Laboratory
- David Chichester, Idaho National Laboratory
- William Bertozzi, Timothy Antaya, Massachusetts Institute of Technology
- Brandon Blackburn, Raytheon
- Pete Marleau, Dean Mitchell, Sandia National Laboratory
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