Use of the Monte Carlo code MCNP-PoliMi for the simulation of correlation measurements

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Overview

Part 1: Description of the Monte Carlo code MCNP-PoliMi.

Part 2: Validation of the Monte Carlo code MCNP-PoliMi.

Part 3: Running MCNP-PoliMi.

The code was developed by Enrico Padovani and Sara Pozzi at the Polytechnic of Milan, with the aim of aiding in the design and analysis of nuclear safeguards experiments based on the measurement of correlation functions of the signals from detectors.

Part 1: Description of the Monte Carlo code MCNP-PoliMi

1. Introductory remarks: correlation measurements, MCNP simulation

2. Main features of MCNP-PoliMi
   - Physics of particle transport
   - Physics of detection

3. MCNP-PoliMi output post-processing

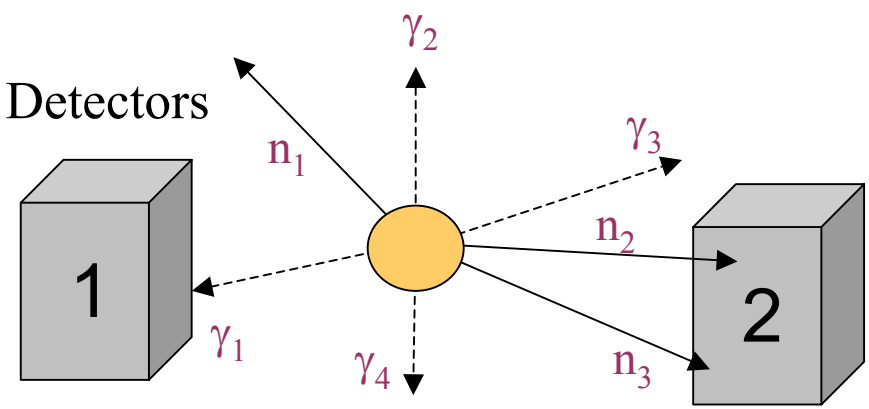
4. Conclusions

Correlation Measurements

Nuclear Materials Identification System

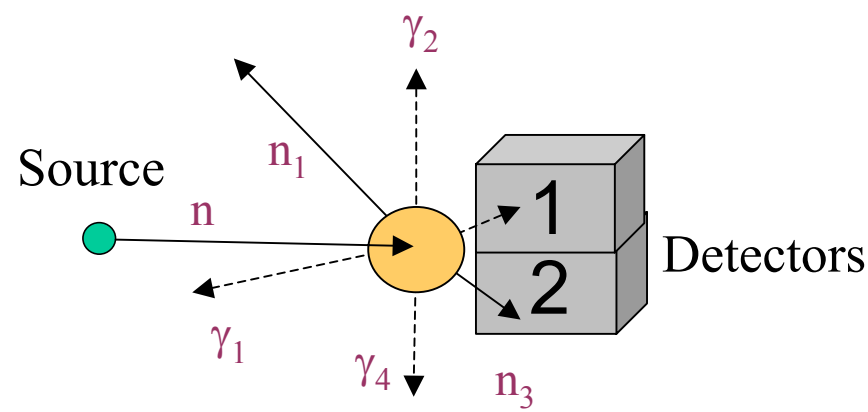
- Scintillation detectors, sensitive to fast neutrons and gamma rays
- Measurement of correlation functions with a time resolution of 1 ns
- External source for active measurements

Passive measurements



Fissile sample

Active measurements



Fissile sample

MCNP simulation

In standard MCNP, at each neutron-nucleus interaction:

- first secondary gamma production is performed
- then the kind of neutron interaction is independently sampled

E.g.: a neutron elastic collision may produce a fission gamma ray in addition to the scattered neutron (it follows that energy is not conserved in the single interaction)

- in fission, neutron multiplicity is restricted to integer less than and greater than $\bar{\nu}$

Mean values are correct

MCNP simulation (cont.)

The simulation of higher order moments (correlations) cannot be based on mean values.
In fact, MCNP is not intended for the simulation of higher order moments.

Main features of MCNP-PoliMi

The modifications performed to MCNP can be grouped into two subsets:

- Physics of particle transport
- Physics of detection

Main Features of MCNP-PoliMi
Physics of particle transport

1. Neutron interaction and secondary gamma production are correctly linked

2. Neutron and gamma fission multiplicity distributions have been implemented

Improved simulation of time-correlation measurements
Analog simulation

Physics of particle transport
Program flow

MCNP

Secondary gamma generation → Neutron collision type

MCNP-PoliMi

Neutron collision type → Secondary gamma generation

Physics of particle transport
Secondary gamma generation

MCNP-PoliMi: secondary gamma generation is based on the principle of conservation of energy in each neutron interaction and relies on the data present in the ENDF-based MCNP libraries. Approximations were made as necessary.

Physics of particle transport
Secondary gamma generation (cont.)

Elastic scattering

Secondary gamma ray production: no gamma ray is generated.

Energy deposited locally: $E_{recoil} = E_{in} - E_{out}$

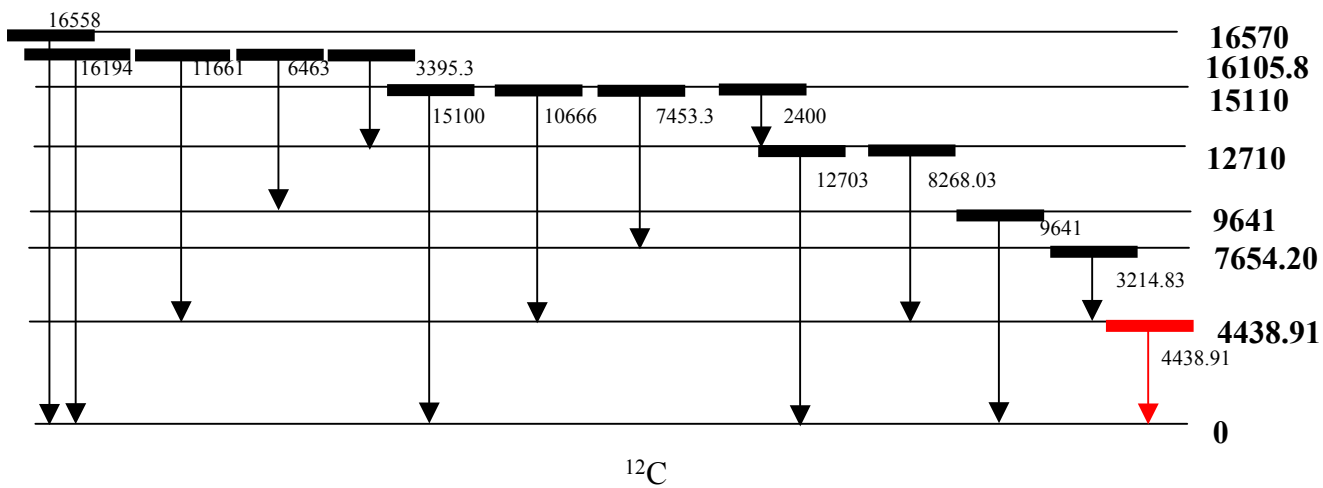
Inelastic scattering

Secondary gamma ray production: $E_g = |Q|$

Energy deposited locally: $E_{recoil} = \dfrac{E_{in} + E_{out} - 2\mu\sqrt{E_{in}E_{out}}}{A}$

where E_{in} = energy of the incoming neutron; E_{out} = energy of the outgoing neutron; A = mass of the target nucleus; Q = Q-value of the reaction; μ= cosine of the scattering angle.

Inelastic scattering on light nuclei: levels and gammas of the C-12 nucleus



Only the first excited level (shown in red) is described by MCNP ENDF-based 60C libraries. In all other cases, no gammas are generated.

Inelastic scattering on heavy nuclei

Many levels are present, and photon production is generally described in the ENDF-based MCNP libraries by a continuous distribution. The algorithm samples from the relevant distributions until the sum of the photon energies equals E_g and energy is conserved in the interaction. The adopted procedure slightly alters the cumulative distribution of photon energies.

Physics of particle transport
Secondary gamma generation (cont.)

Capture reactions

Secondary gamma ray production: $E_g = \frac{A}{(A+1)} E_{in} + Q$

Energy deposited locally: $E_{recoil} = \frac{E_{in}}{(A+1)}$

where E_{in} = energy of the incoming neutron; E_{out} = energy of the outgoing neutron; A = mass of the target nucleus; Q = Q-value of the reaction

Physics of particle transport
Fission

Fission (spontaneous and induced)

A dual particle source is generated emitting neutrons and gamma rays with multiplicity and energy distributions given by Valentine, *Annals of Nuclear Energy* 28, p. 191-201.

A delay in the emission of gamma rays from fission can be optionally included in the simulation.

Physics of particle transport
Sources

Source options

Single-particle sources

- Any standard MCNP source

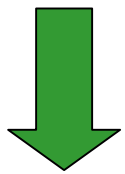
Multiple-particle sources (neutron and gamma)

- Cf-252, U-238, Pu-240, Pu-242, Cm-242, Cm-244 (spontaneous fission)
- Am/Be (alpha-n)

Main Features of MCNP-PoliMi

Physics of detection

Detailed information on the interactions of neutrons and gamma rays inside user-defined cells (typically detectors) is printed to file, to be analyzed with a post-processing code

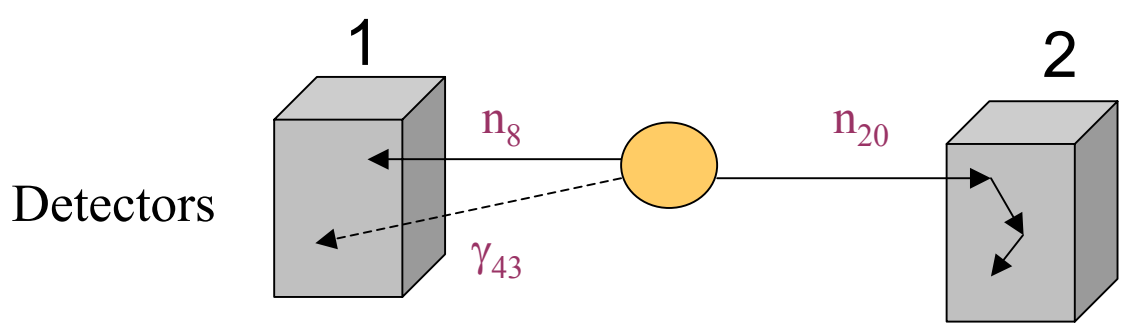


Improved simulation of detector response

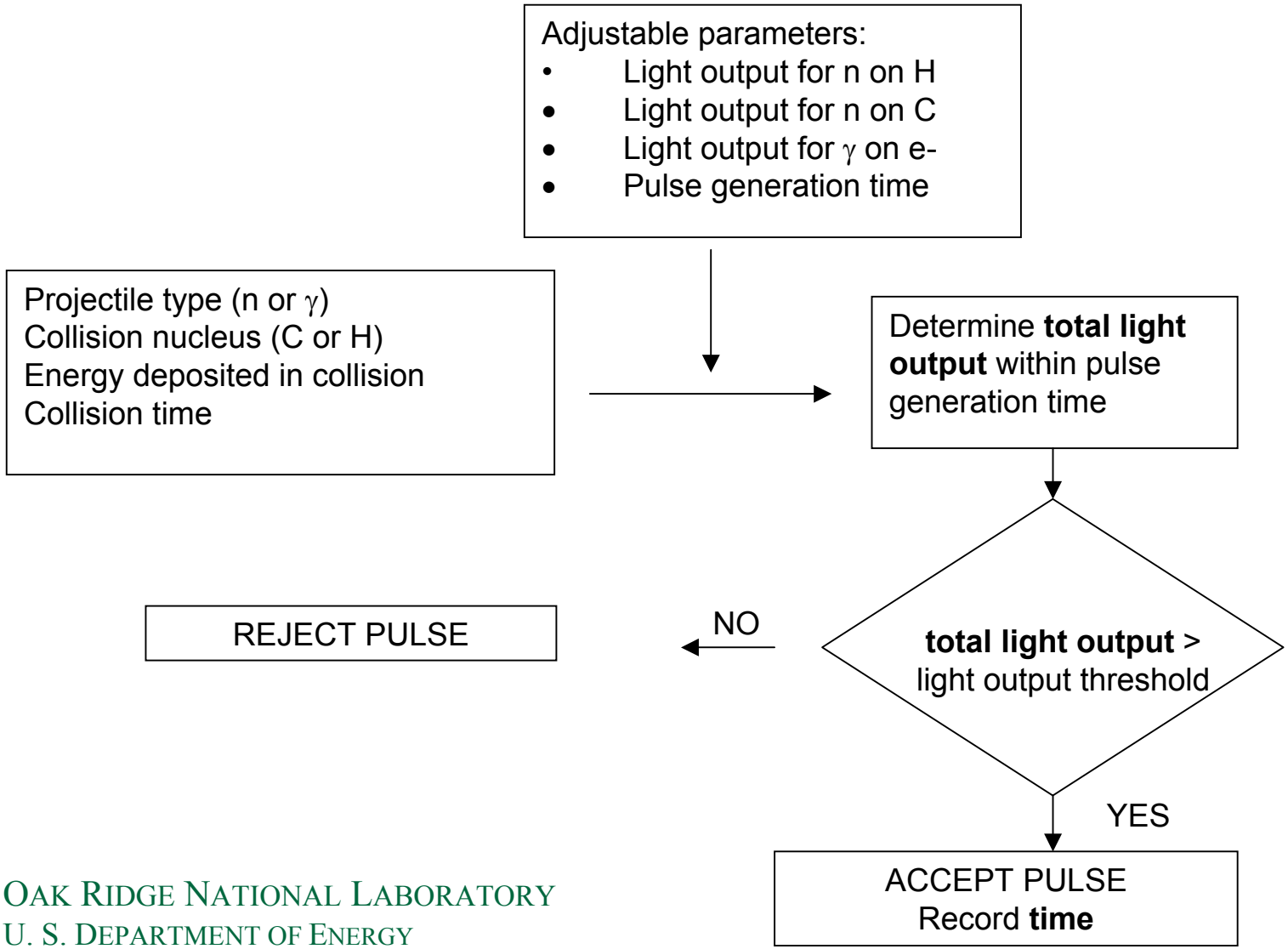
MCNP-PoliMi Collision output file (excerpt)

History number	Particle number	Projectile type*	Interaction type$^{\Delta}$	Target nucleus•	Cell number of collision event	Energy deposited in collision (MeV)	Time (shakes)	Collision position (X)	(Y)	(Z)	Particle weight	Generation number	Number scatterings
...													
29	20	1	-99	1001	2	0.3105	2.743	-25.64	14.37	8.7	1	2	2
29	20	1	-99	1001	2	0.1894	2.846	-26.01	-14.8	9.48	1	2	3
29	20	1	-99	1001	2	0.1617	3.045	-27.27	15.07	10.1	1	2	4
29	43	2	1	6	1	0.7734	6.514	31.83	7.07	11.2	1	2	1
29	8	1	-99	1001	1	0.3731	3.965	27.84	18.82	-0.6	1	0	1
...													

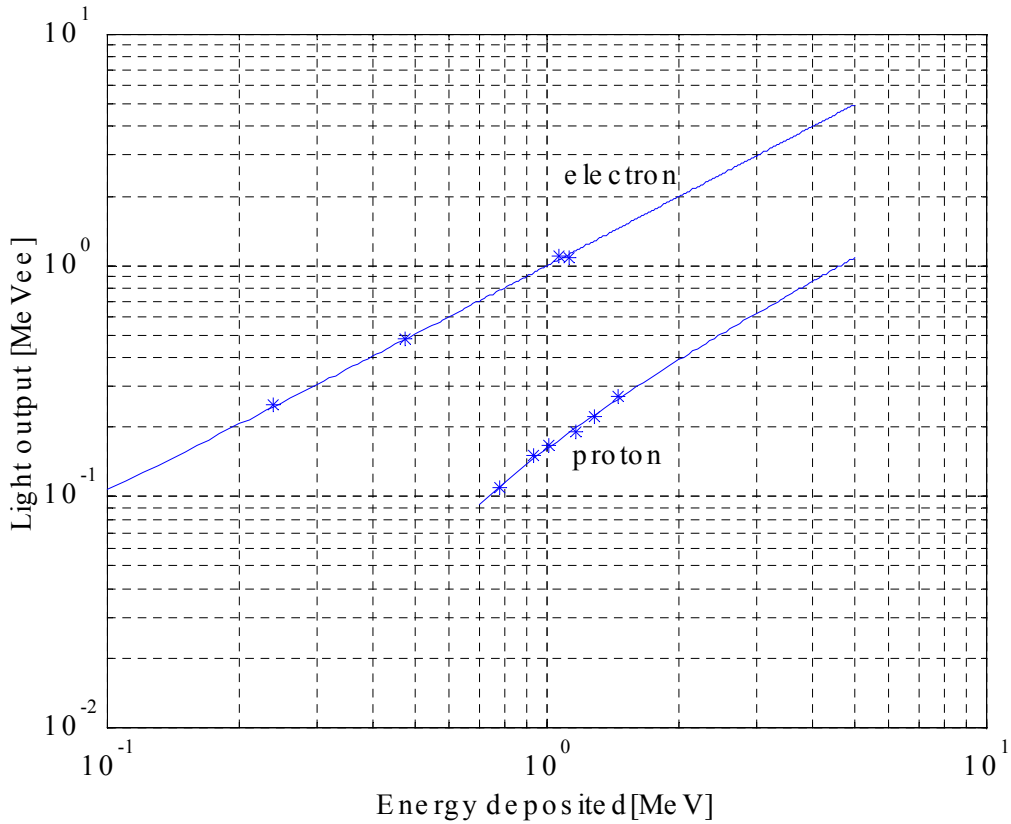
* 1 = neutron; 2 = photon
Δ -99 = elastic scattering; 1 = Compton scattering
• 1001 = hydrogen; 6 = carbon



Post-processing code for MCNP-PoliMi collision output file: scintillation detectors



Post-processing code for MCNP-PoliMi collision output file: scintillation detectors (cont.)



Experimental calibration of scintillator to determine the light output for neutron interactions on H and the light output for γ interactions on e-.

Post-processing code for MCNP-PoliMi collision output file: scintillation detectors (cont.)

electron light output = 0.008 + 0.99 * Ee

proton light output = -0.0685 + 0.23 * Ep (Ep>0.30 MeV)

Part 1: Conclusions

The MCNP-PoliMi code has been developed at the Polytechnic of Milan, Italy, by suitably modifying the standard MCNP code.

Major modifications were introduced to the physics of particle transport. In particular, the correlation of neutron interaction type and secondary gamma ray production, and the emission of neutrons and gamma rays in spontaneous and induced fission.

Part 1: Conclusions (cont.)

The code output consists in detailed information of the collisions occurring inside detector cells.

The post-processing code can be tailored to model different detectors and to convey information on the particles detected. For example, neutrons can be distinguished from photons and particles from induced fission can be distinguished from source particles.

Part 2: Validation of the Monte Carlo code MCNP-PoliMi

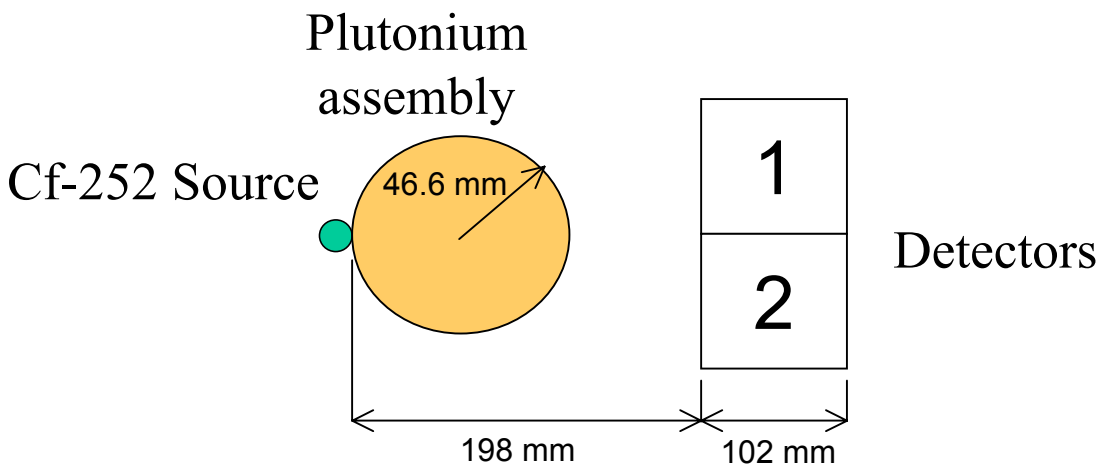
Validation of MCNP-PoliMi for correlation measurements

NMIS experiments with fissile samples:

Plutonium metal shells (VNIIEF)
Uranium metal castings (Y-12)

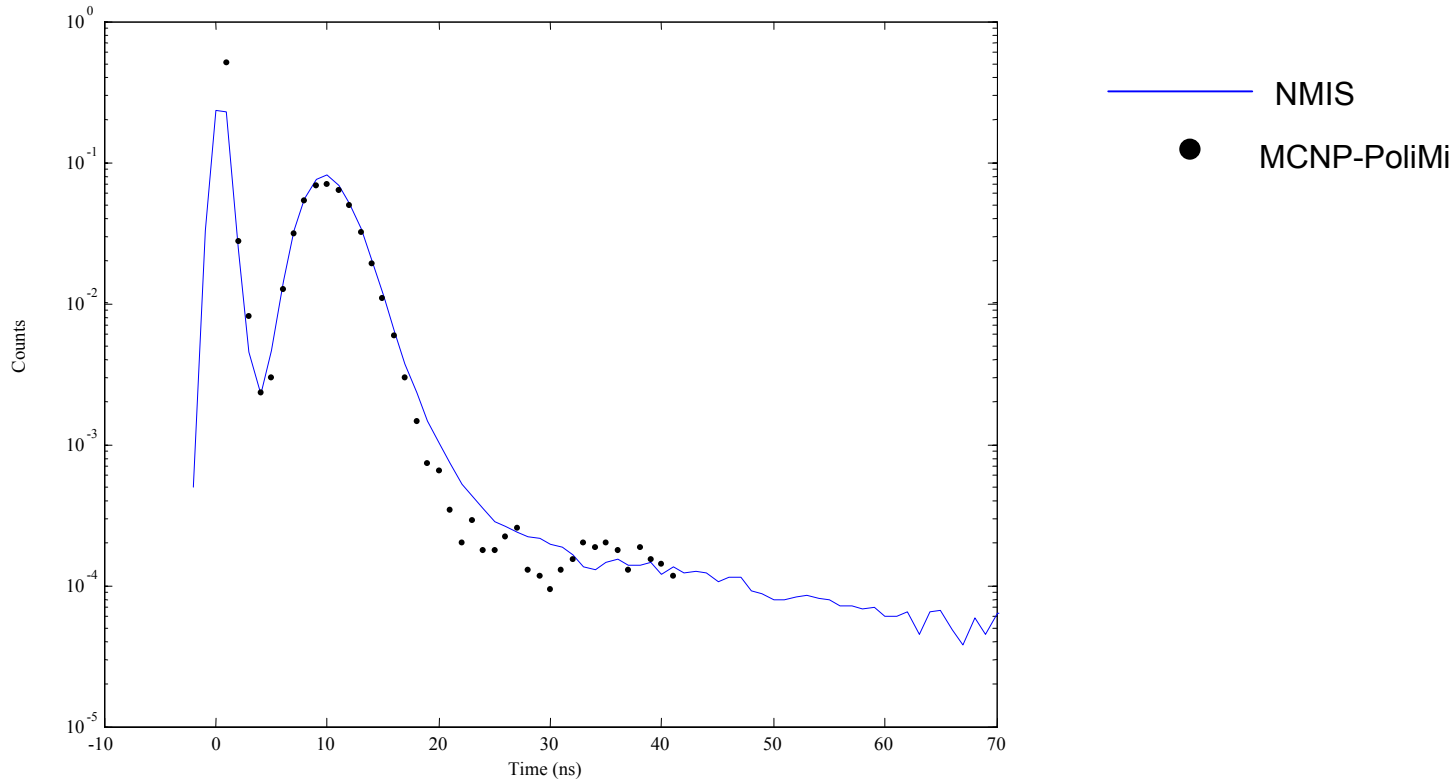
NMIS measurement of plutonium metal shell

Delta-phase, nickel-plated, 98% ^{239}Pu
Total mass ~4.4 kg

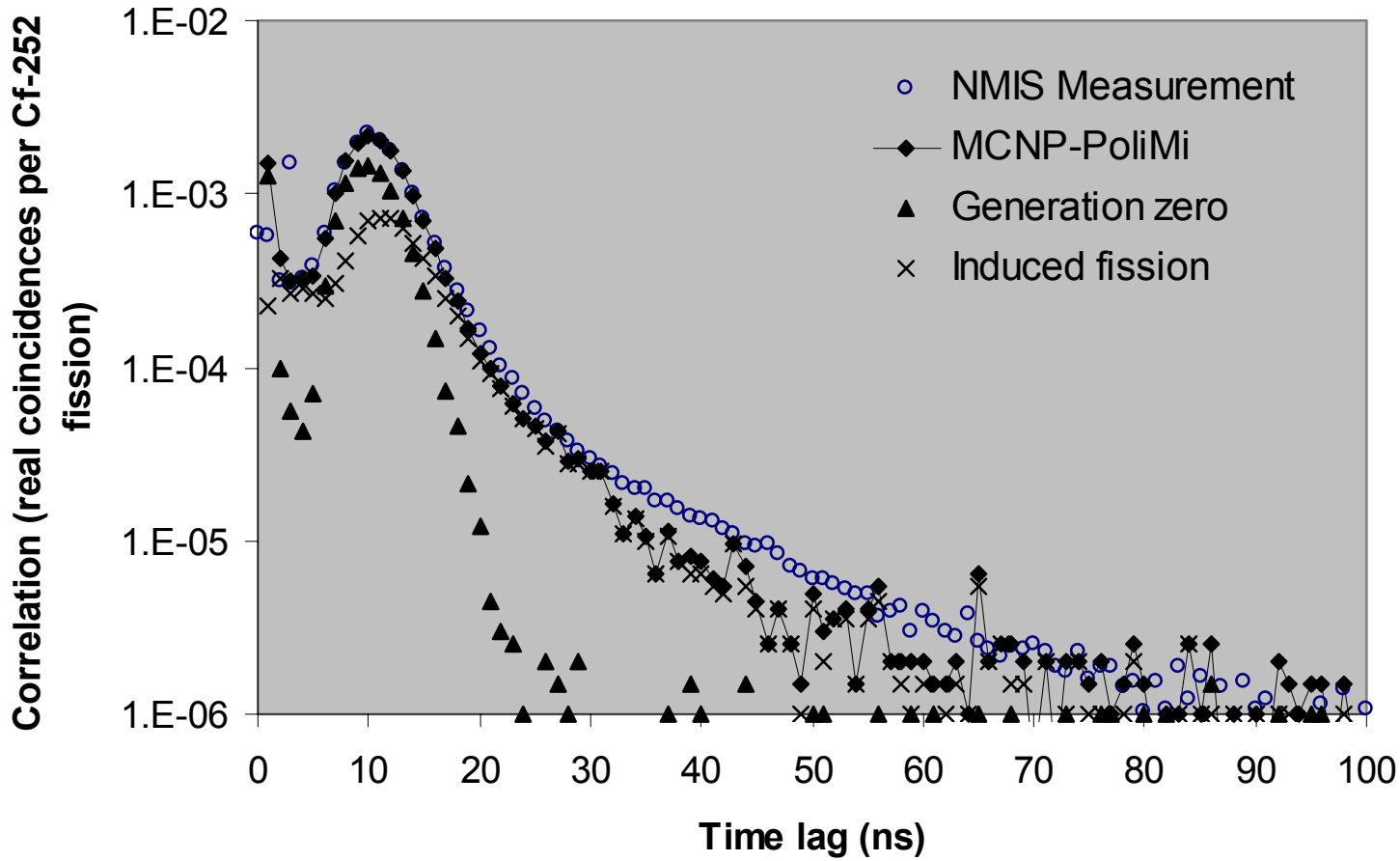


Measurements performed at the Russian Federal Nuclear Center,
All-Russia Scientific Research Institute of Experimental Physics in Sarov.

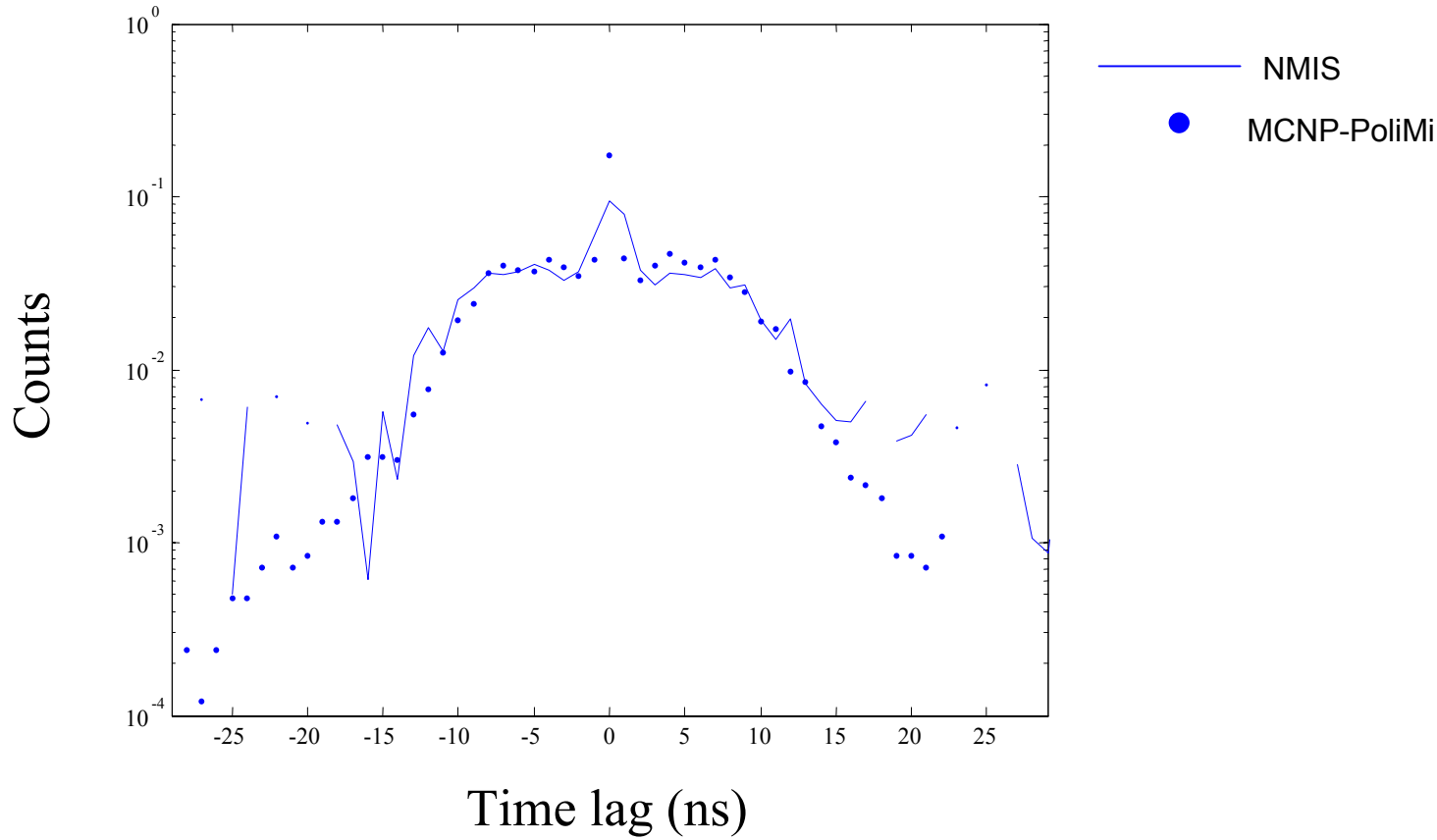
Comparison of NMIS measurement and simulation for time-of-flight in air



Comparison of NMIS measurement and MCNP-PoliMi simulation of plutonium metal shell: source-detector correlation



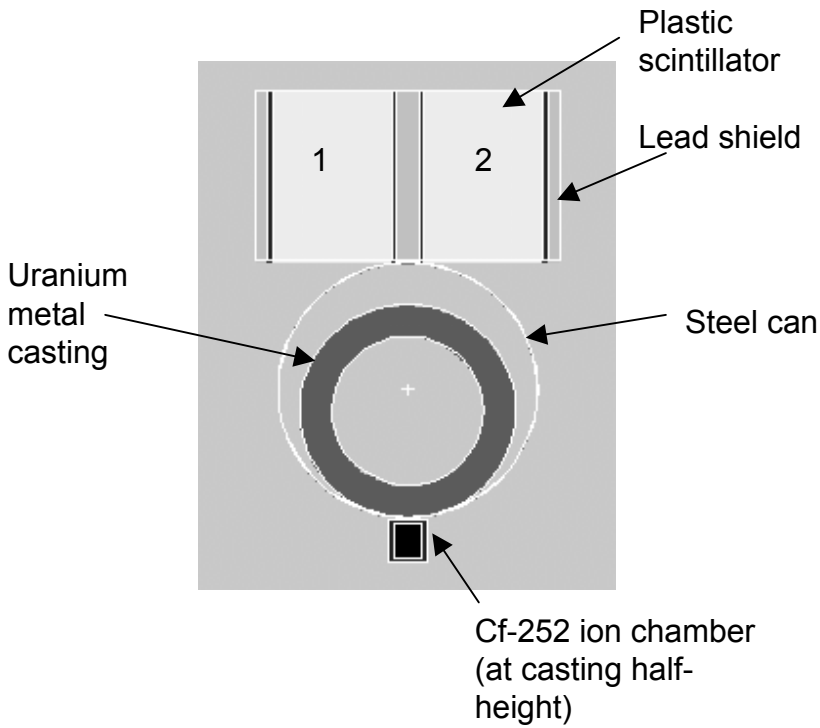
Comparison of NMIS measurement and MCNP-PoliMi simulation of plutonium metal shell: detector-detector correlation



NMIS measurement of uranium casting

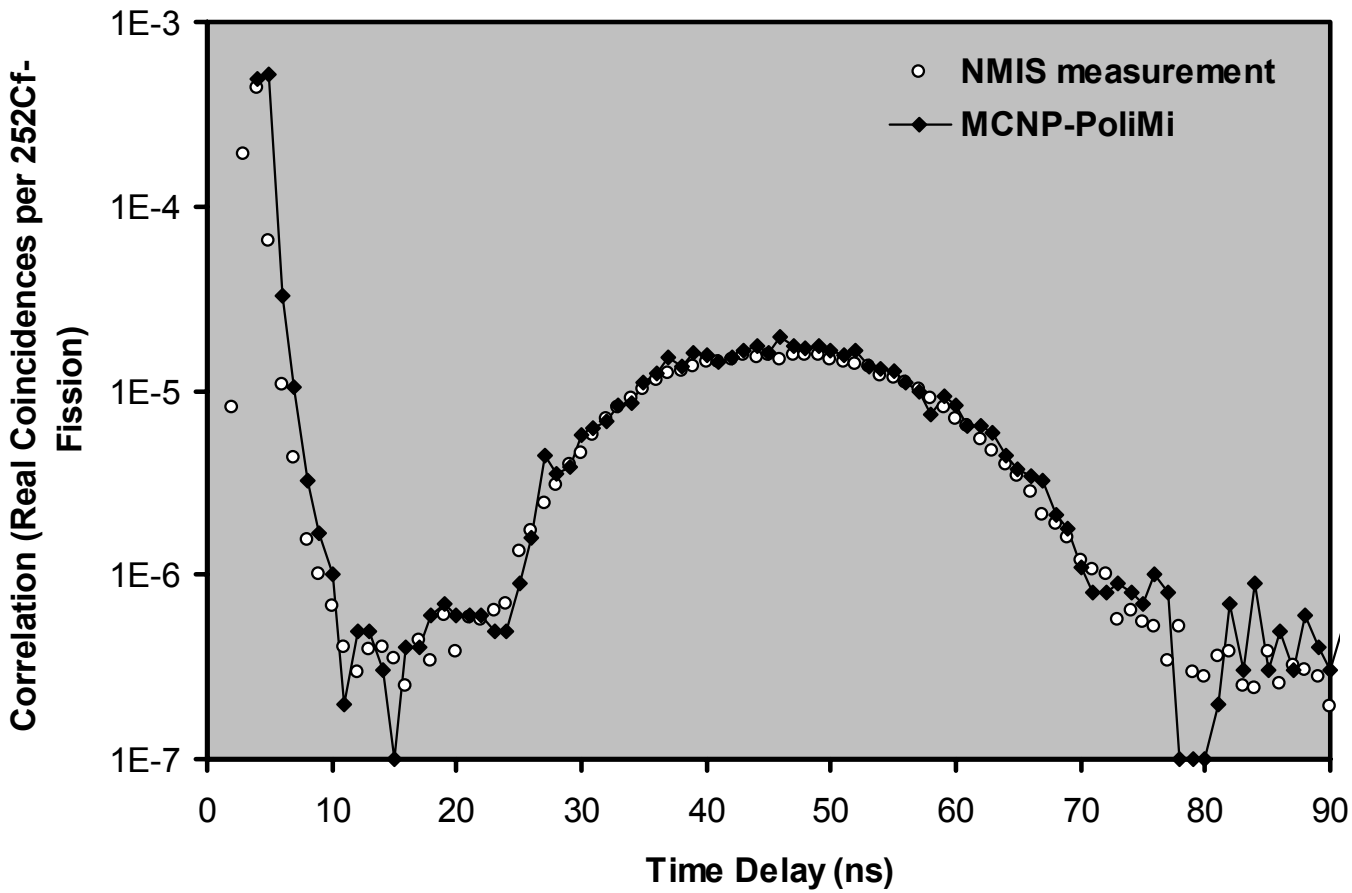
Total mass ~18.6 kg

93.15% ^{235}U

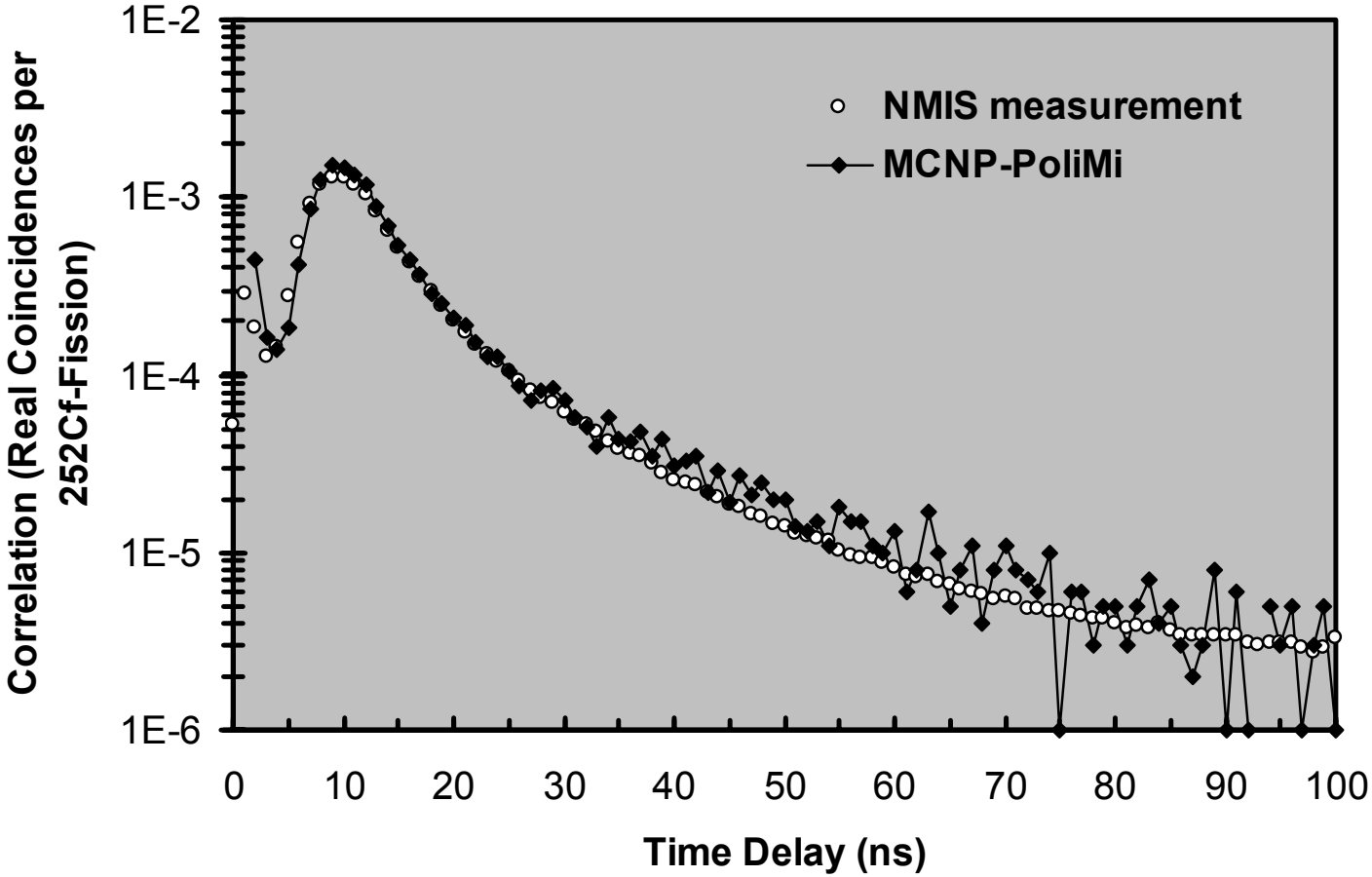


Measurements performed at the Oak Ridge Y-12 plant.

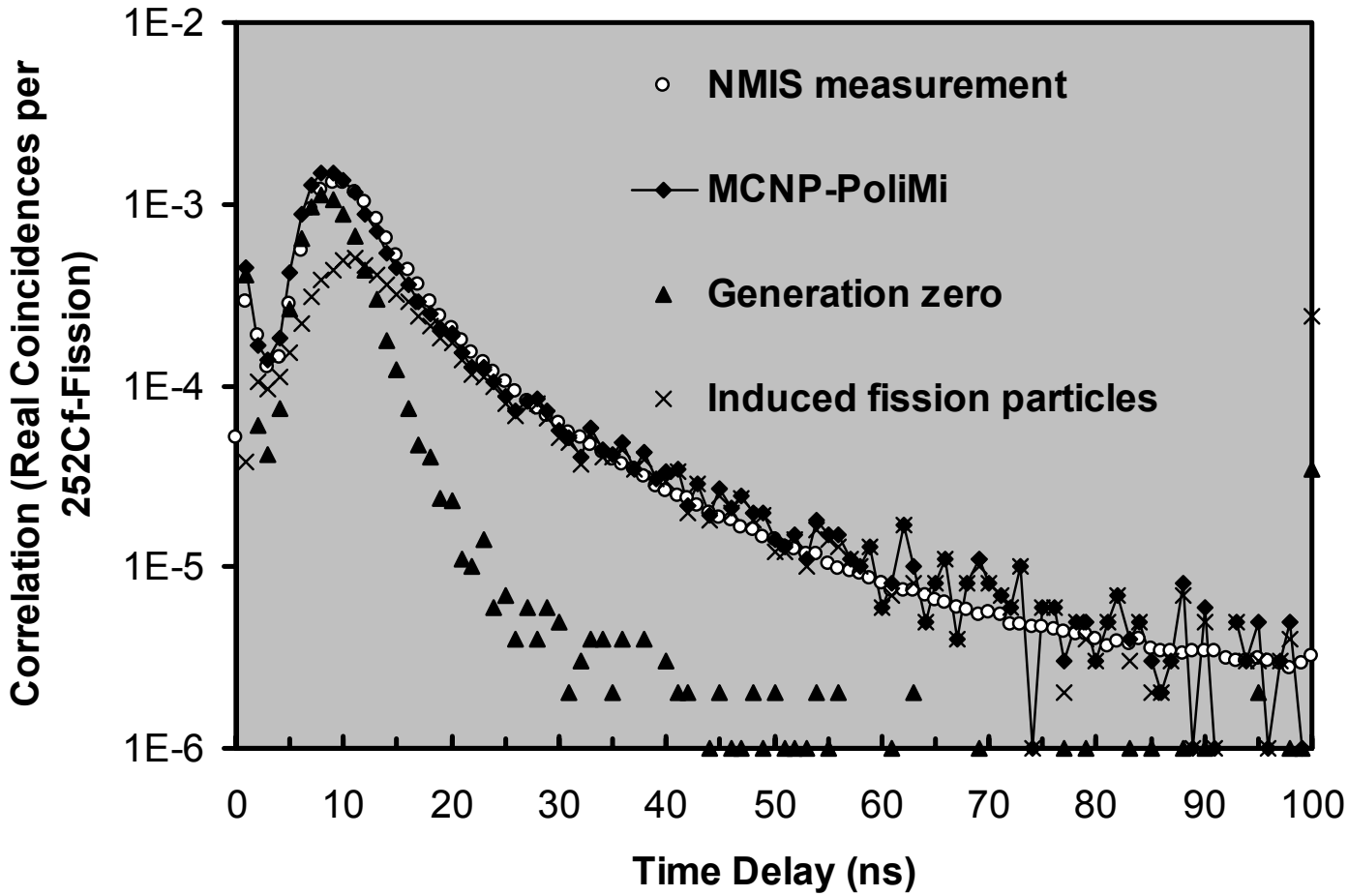
Comparison of NMIS measurement and MCNP-PoliMi simulation: time-of-flight in air



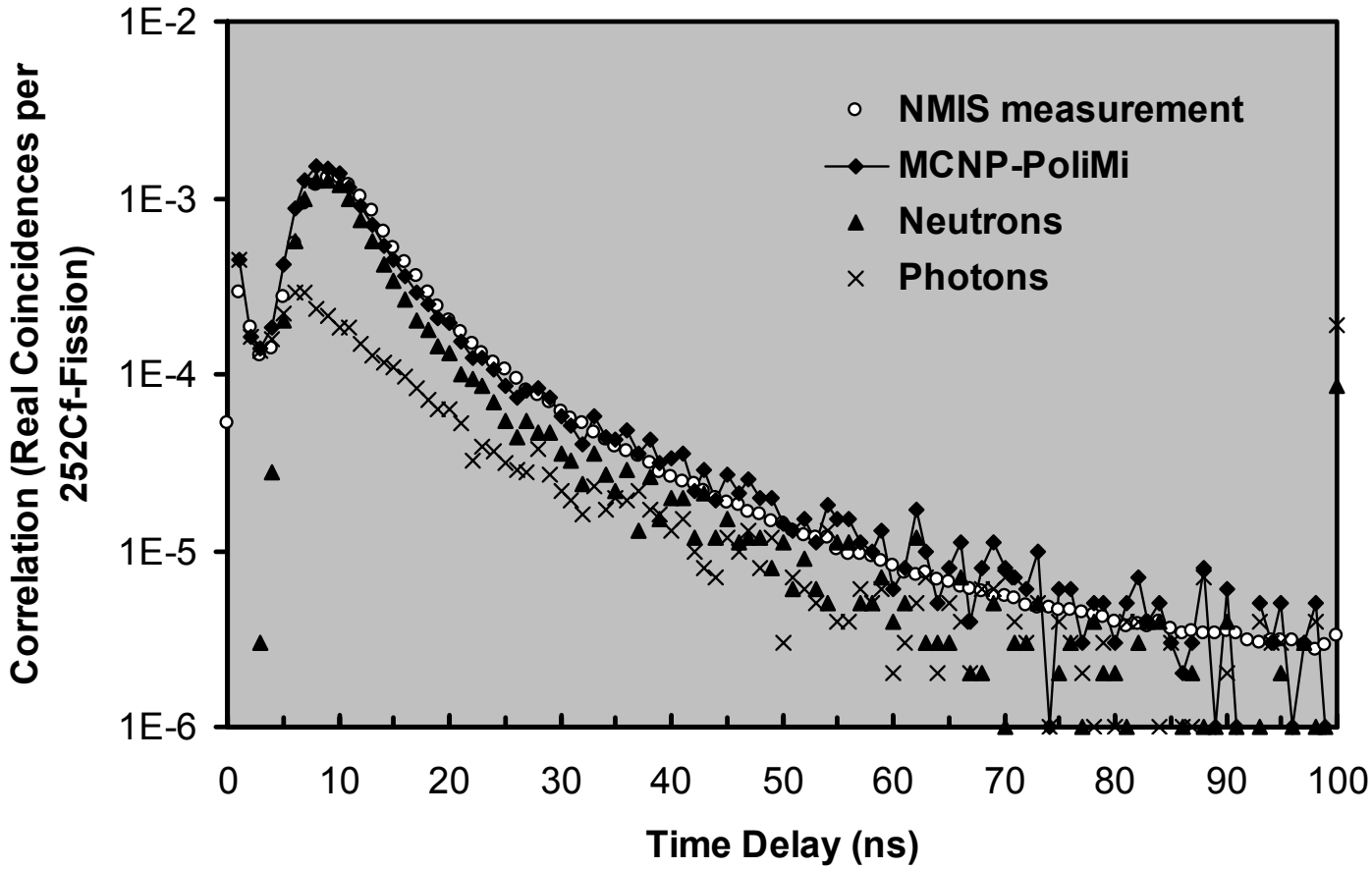
Comparison of NMIS measurement and MCNP-PoliMi simulation of uranium casting: source-detector correlation (1)



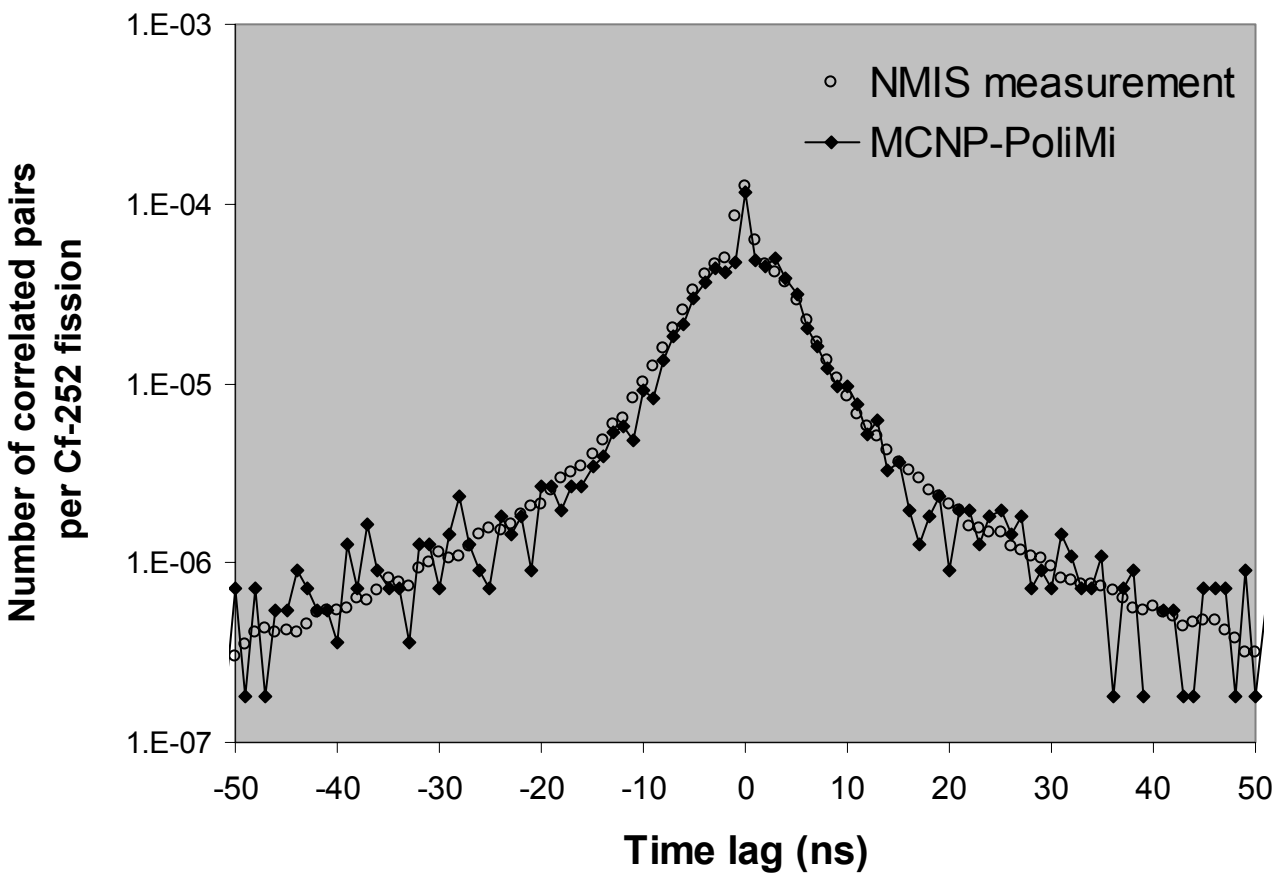
Comparison of NMIS measurement and MCNP-PoliMi simulation of uranium casting: source-detector correlation (2)



Comparison of NMIS measurement and MCNP-PoliMi simulation of uranium casting: source-detector correlation (3)



Comparison of NMIS measurement and MCNP-PoliMi simulation of uranium casting: between-detectors correlation



Comparison of NMIS measurement and MCNP-PoliMi simulation: errors

Plutonium passive

Plutonium assembly	Gamma peak (% error)	Neutron peak (% error)
Pu4	+2.9	-0.74
Pu7	+6.1	-2.2
RMS	4.77	1.64

Plutonium active

System ID	First mode (% error)	Second mode (% error)
Pu3	+2.5	+0.9
Pu4	+9.2	+0.8
Pu7	+5.9	-3.9
RMS	6.47	2.36

Uranium active

Cross-correlation	First mode (% error)	Second mode (% error)
Source – detector 1	+0.5	+6.8

Uranium active

Cross-correlation	Total area (% error)
Detector 1 – detector 2	-6.3

Part 2: Conclusions

A number of simulations having the aim of validating the MCNP-PoliMi code showed good agreement between the MCNP-PoliMi simulations and the NMIS measurements on uranium and plutonium metal.

The error was below 5 %, in most cases.

The post-processing code can be used to subdivide the measured signature into its neutron/photon components and source particles/induced fission components.

Part 3: Running the Monte Carlo code MCNP-PoliMi

Extra input cards

Running MCNP-PoliMi does not require an extra input file.

The standard MCNP input file is used for the description of

- Geometry
- Materials
- Standard MCNP sources

Extra input cards (cont.)

The extra input cards are:

IDUM

- extra sources
- neutron multiplicity from induced fission
- secondary gamma ray emission (correlation with neutron events)
- time delay in gamma emission following fission
- cells for which collision information printout is required

Extra input cards (cont.)

RDUM

Neutron and photon energy release threshold for printout in the collision information data. Limits the size of the output file.

Part 3: Conclusions

The MCNP-PoliMi code executable is available for users upon request: pozzisa@ornl.gov, enrico.padovani@polimi.it

Citation:

E. Padovani and S.A. Pozzi, "MCNP-PoliMi ver. 1.0. User's Manual", CESNEF-021125 Library of Nucl. Eng. Dept., Polytechnic of Milan, Italy, Nov. 2002.