MCNP6 Simulations of Active Neutron Interrogation of Fissile Samples using a Deuterium-Deuterium Neutron Generator

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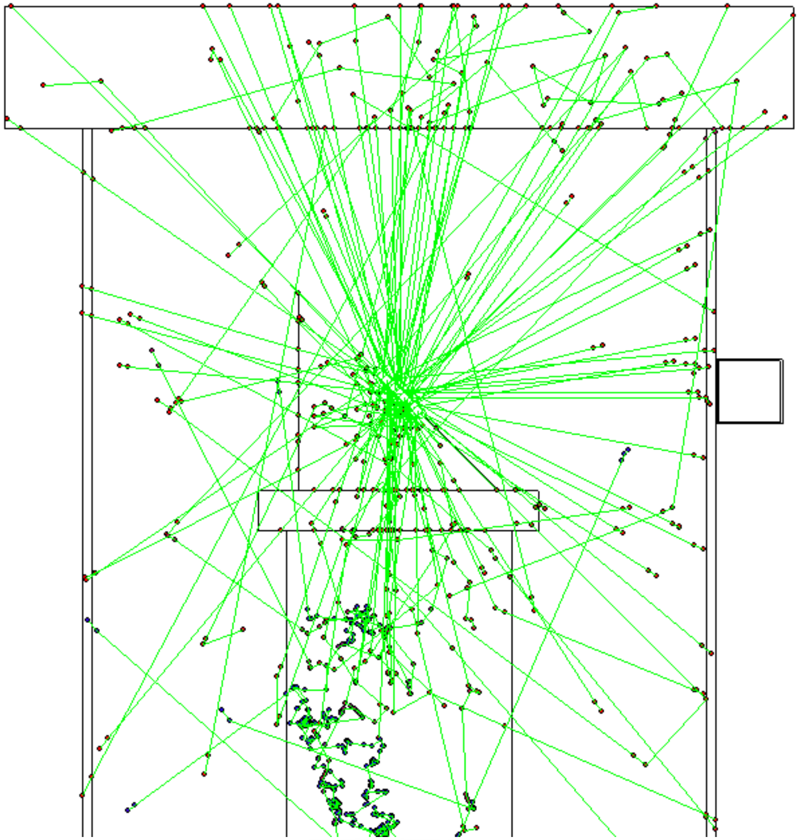
Outline

- Introduction to Active Neutron Interrogation using DD Neutron Generator
- MCNP6 Simulation Strategies for Delayed Neutron Emission
- Delayed Neutron Yield from Active Neutron Interrogation Irradiations using DD Neutron Generator
- Conclusions and Future Work

Introduction to Active Neutron Interrogation using DD Neutron Generator

Purpose of Study



- At Canadian Nuclear Laboratories (CNL), a Deuterium-Deuterium (DD) neutron generator has recently been installed in the Health Physics Neutron Generator (HPNG) facility

- Monte Carlo N-Particle version 6 (MCNP6) is used to simulate active neutron interrogation of a variety of fissile materials using the DD neutron generator and determines the *approximate* **minimum** mass of fissile materials that can be interrogated by the generator

Active Neutron Interrogation

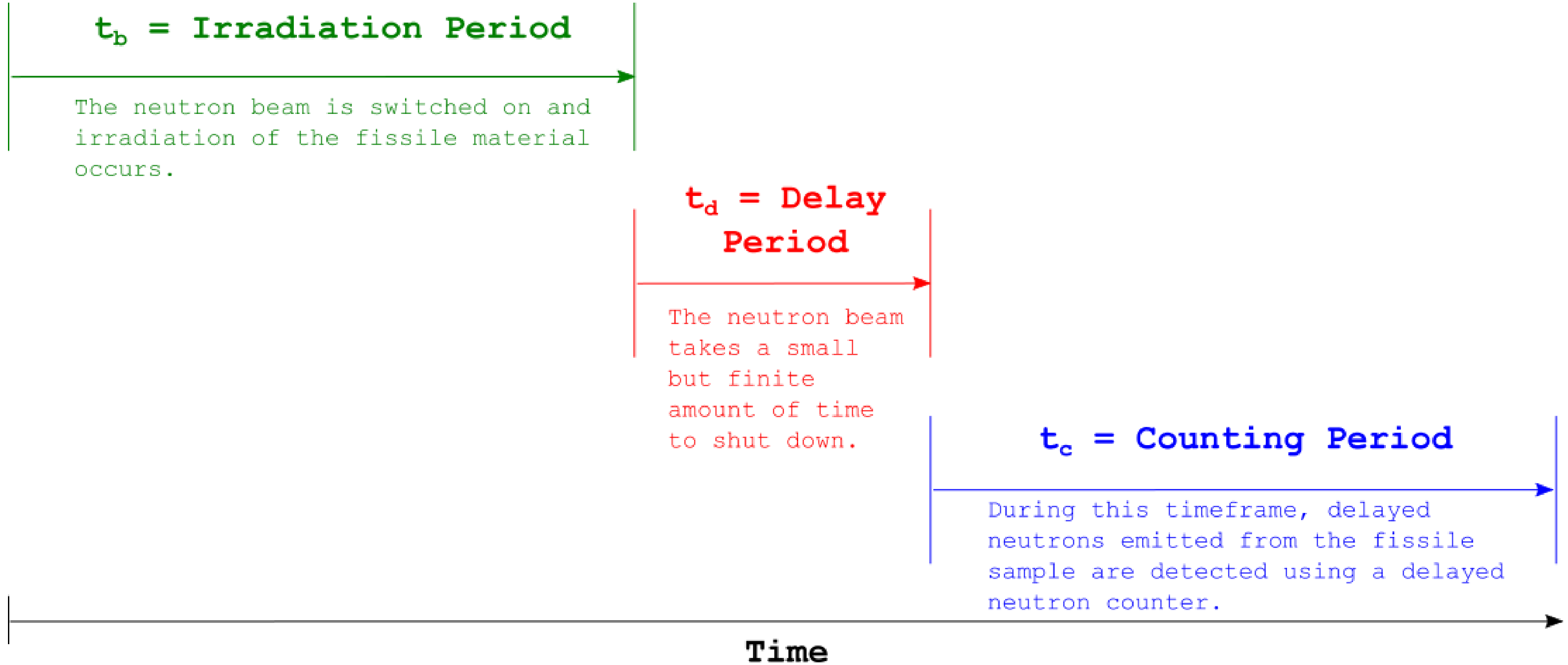
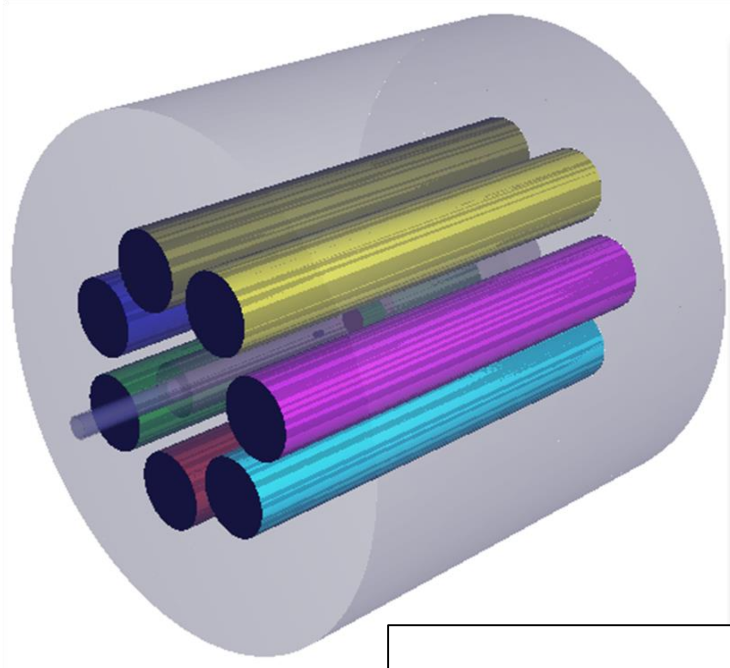


Figure 1
Time Profile for Active Neutron Interrogation Irradiations (Amiel, 1962 [1])



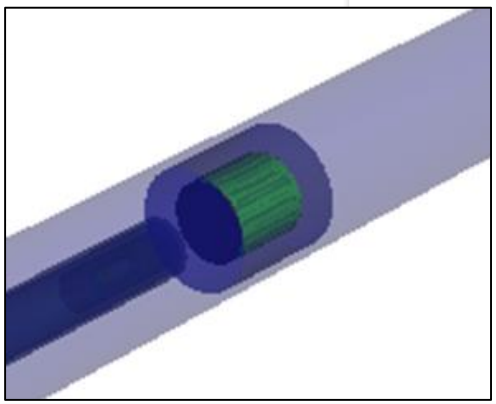
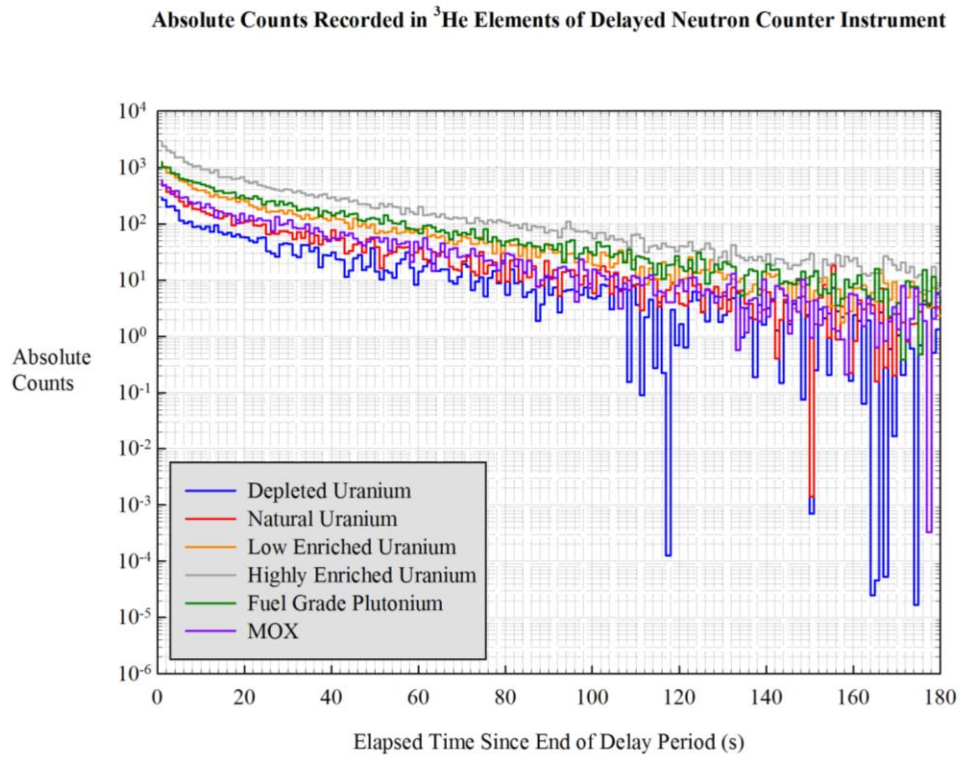


Figure 2
Interrogation of Fissile Sample using ^{252}Cf Embedded Inside a Delayed Neutron Counter

Inset: Positioning of ^{252}Cf Neutron Source with respect to Fissile Sample



Counting profile *above* background levels indicates the presence of fissile material in interrogated sample

Figure 3
Time-Dependent Counts Recorded by ^{3}He Detectors in Delayed Neutron Counter

Deuterium-Deuterium Neutron Generator

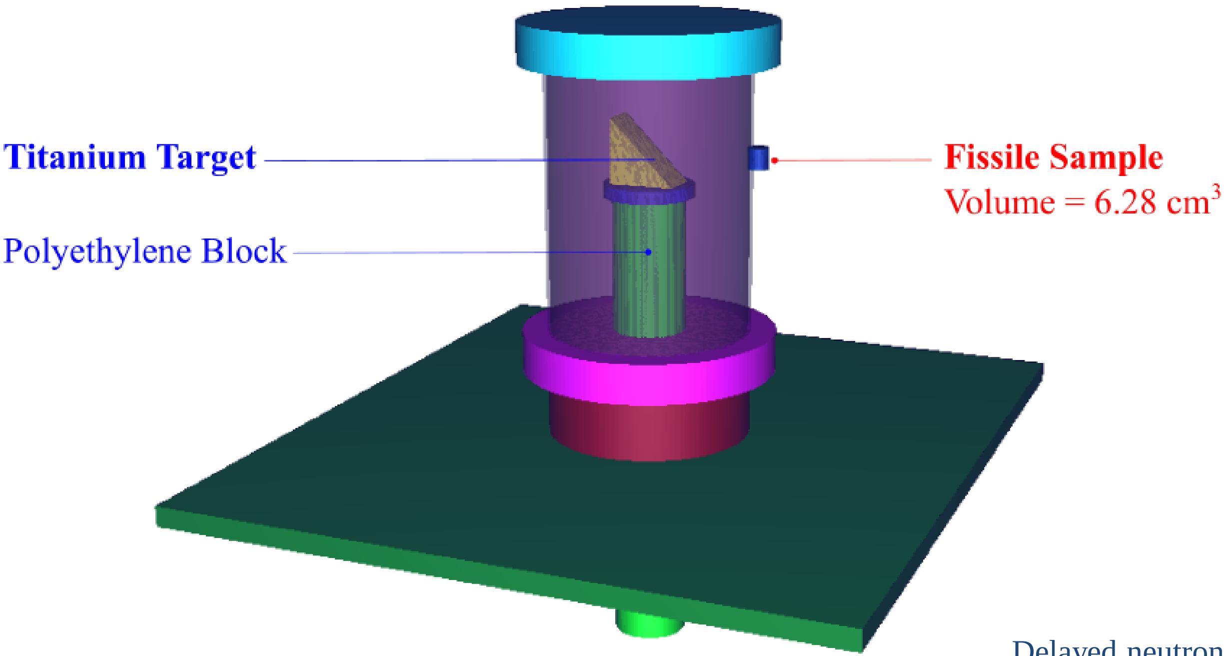


Figure 4
3-D Rendering of Deuterium-Deuterium Neutron Generator from Adelphi Technology, Inc. (Chen, 2016 [2])

A **100 kV** Deuterium ion beam is incident on titanium target, causing D(d,**n**)^{3}He reaction

Nominal Neutron Emission Rate = $\mathbf{10^{9}\ n\ s^{-1}}$

Irradiation Time (t_b) = 60 s
Delay Period (t_d) = 10^{-3} s
Counting Period (t_c) = 180 s

Delayed neutron emission from fissile samples are tabulated.

This is a theoretical scenario that is used to establish the **minimum detectable mass** of fissile isotopes in an interrogated sample.

MCNP6 is needed to simulate the irradiation of fissile samples using an ***anisotropic*** neutron source.

Simulation of Time-Dependent Delayed Neutron Transport in MCNP6

Part 1

```
SDEF ... TME=D4
SI4 H 0 60E8
SP4 D 0 1
```

A *uniform* number of neutrons will be *continuously* emitted from the source from 0 s to 60 s

0 s = Neutron Source Swithced **On**
60 s = Neutron Source Swithcd **Off**

Part 2

```
ACT DN=LIBRARY DNBIAS=10
```

Model delayed neutron production using **data libraries** and produce **10** delayed neutrons per delayed neutron emission event (this *reduces* statistical uncertainty of tallies)

Part 3

```
CUT:N 300E8 J J J J
```

Begin neutron trasnport at **0 s**
Terminate neutron transport at **300 s**

Part 4

```
F1:N (81.1 81.2 81.3)
E1 ...
T1 60.001E8 240.001E8
```

Quantify the number of delayed neutrons emitted from a sample as a function of time using time binning

Figure 5
Cards in Input File to Model Time-Dependent Delayed Neutron Transport in MCNP6 (Andrews, 2015 [3])

Fissile Materials

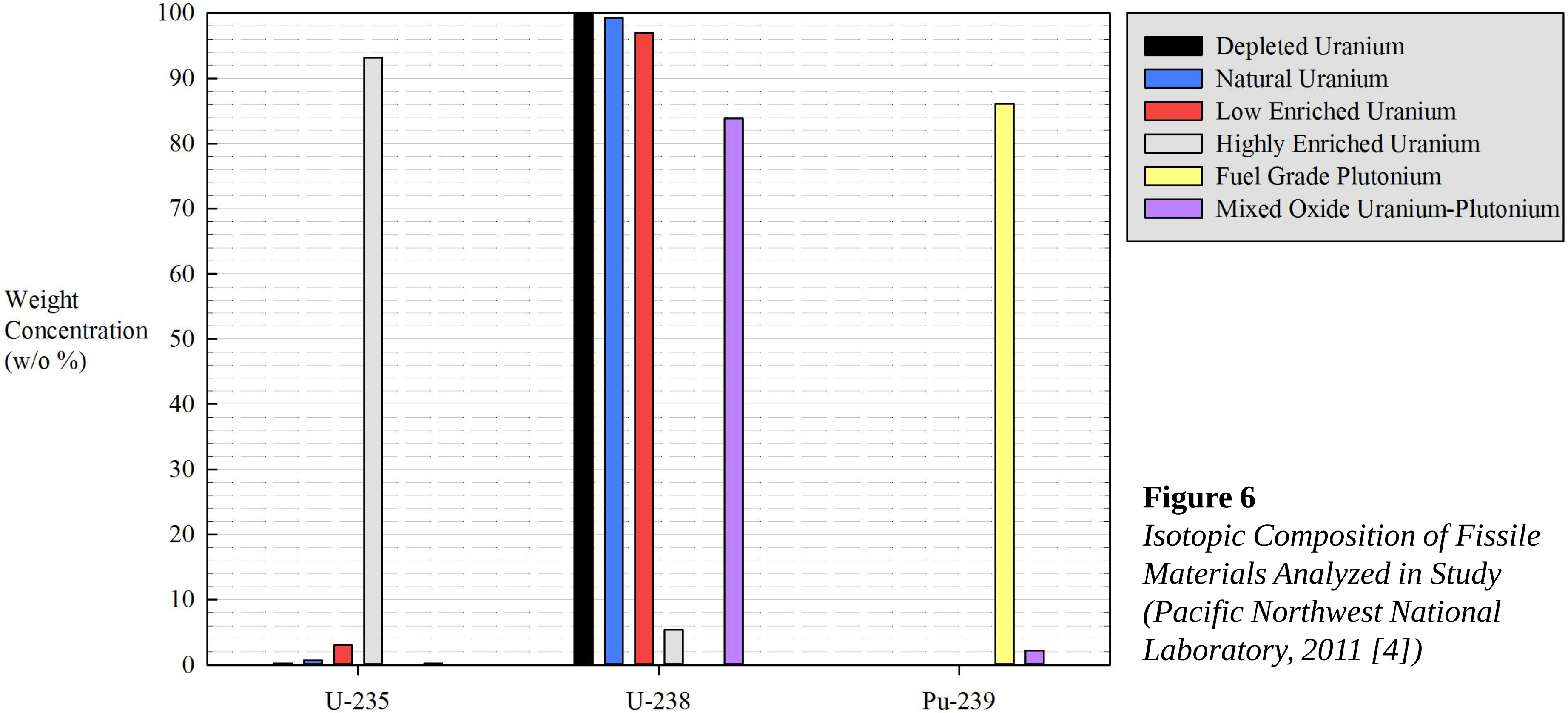
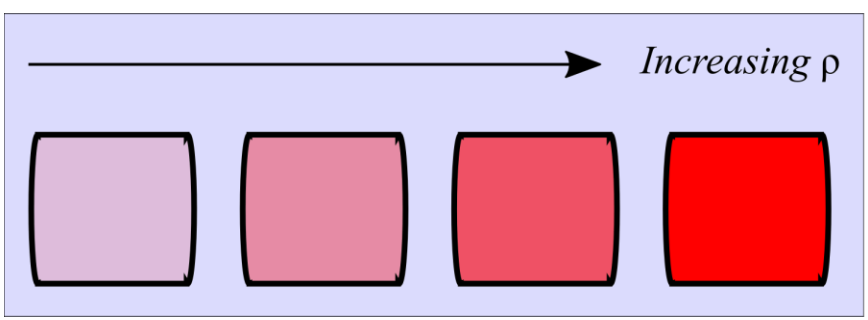


Figure 6
Isotopic Composition of Fissile Materials Analyzed in Study (Pacific Northwest National Laboratory, 2011 [4])



Density of each fissile material was varied to reflect their varying concentration in intercepted articles

Figure 7
Fissile Sample Density Variation

Neutron Kinetic Energy Spectrum and ^{235}U Fission Reaction Data

E_{ave} = **1.80 MeV**
φ_{tot} = **6.38 × 10^5 n cm^{-2} s^{-1}**

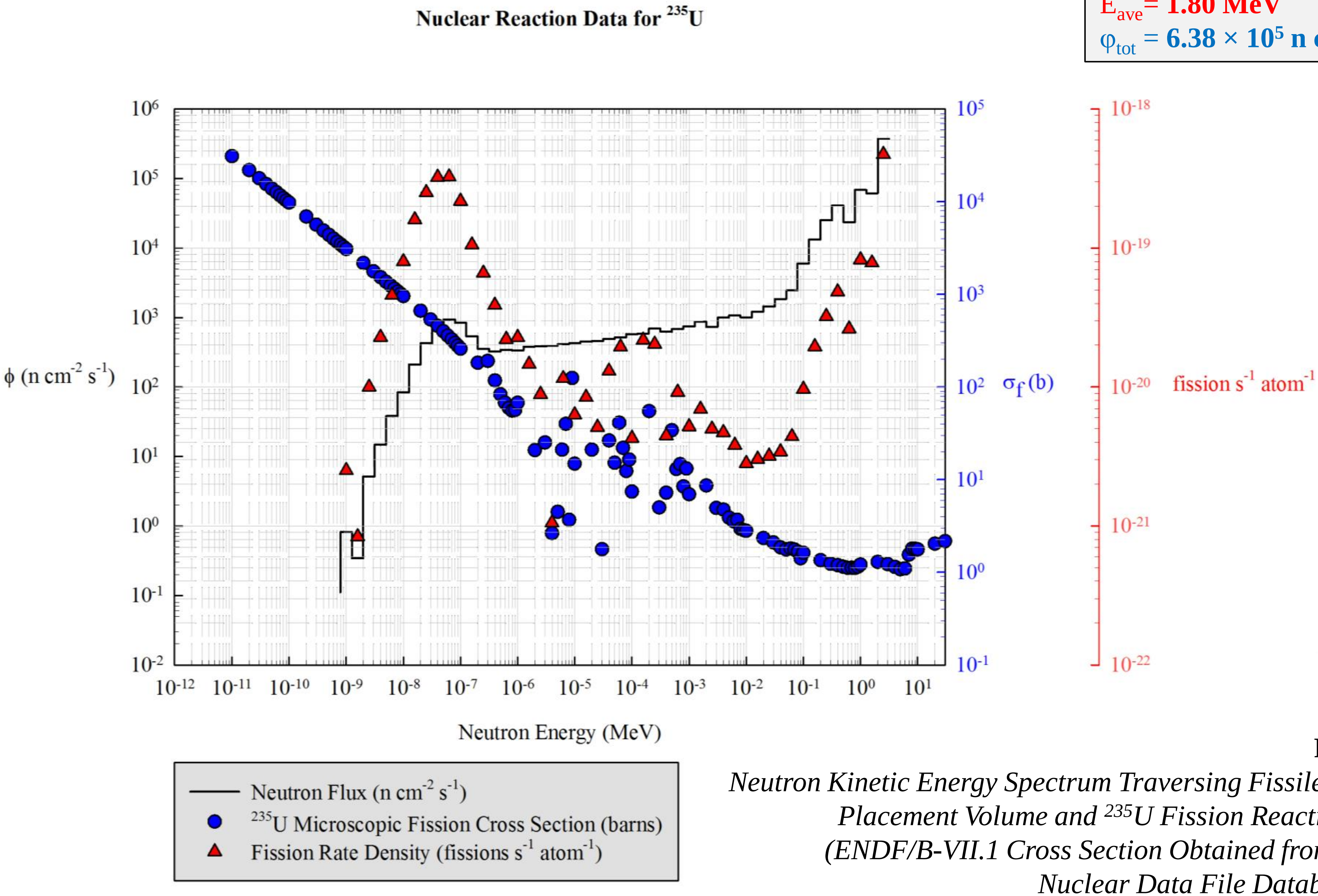


Figure 8

Neutron Kinetic Energy Spectrum Traversing Fissile Sample Placement Volume and ^{235}U Fission Reaction Data (ENDF/B-VII.1 Cross Section Obtained from ENDF Nuclear Data File Database [5])

Neutron Kinetic Energy Spectrum and ^{239}Pu Fission Reaction Data

$E_{ave} = 1.80$ MeV
$\varphi_{tot} = 6.38 \times 10^5$ n cm^{-2} s^{-1}

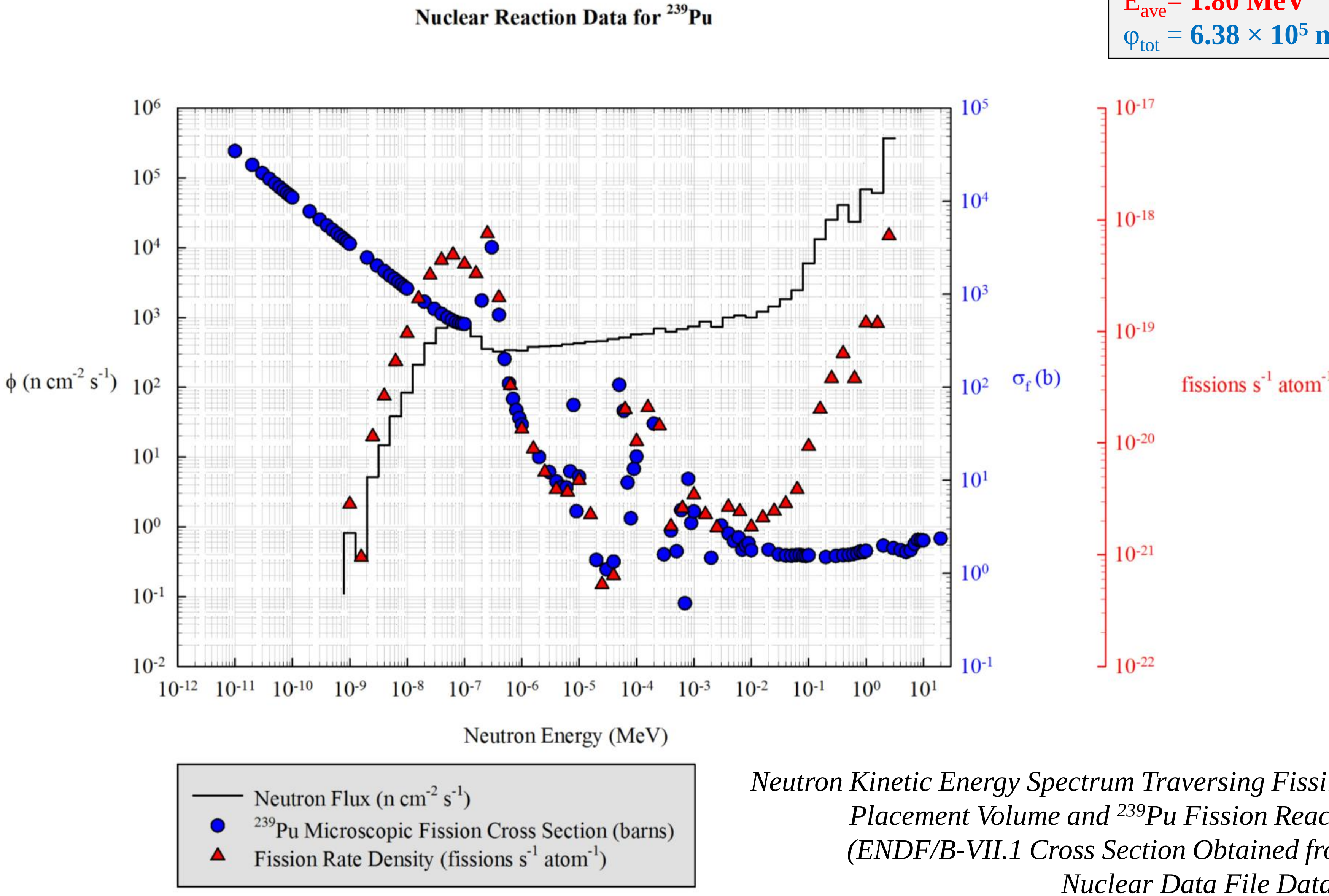
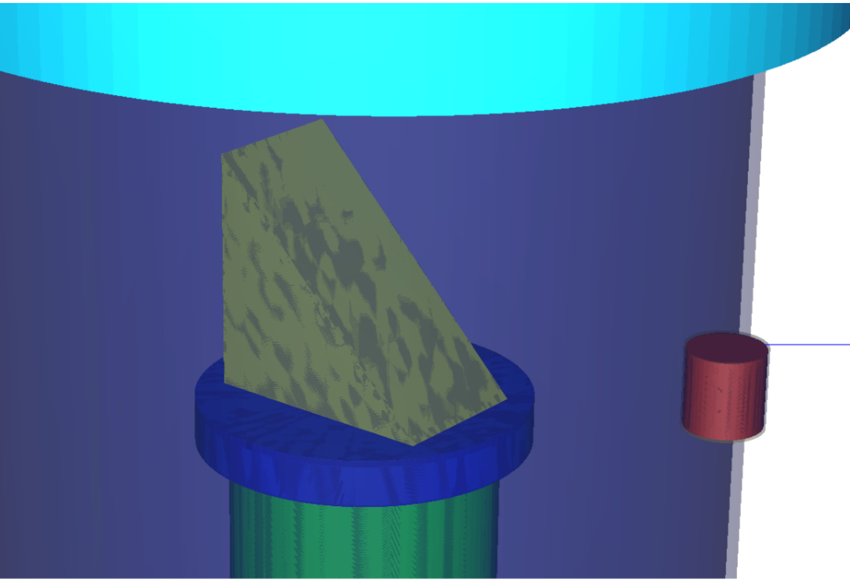


Figure 9

Neutron Kinetic Energy Spectrum Traversing Fissile Sample Placement Volume and ^{239}Pu Fission Reaction Data (ENDF/B-VII.1 Cross Section Obtained from ENDF Nuclear Data File Database [5])

Delayed Neutron Yield from Active Neutron Interrogation Irradiations using DD Neutron Generator

Objective: Determine, in *absolute* terms, the total number of delayed neutrons emitted from a fissile sample, as a function of mass, during the counting period



Number of Delayed Neutrons Emitted from Fissile Sample

```
F11:N (81.1 81.2 81.3)
FT11 TAG 3
FU11 0008092234.99999
0008092235.99999 0008092236.99999
0008092238.99999
T11 60E8 60.001E8 240.001E8
```

Surfaces of bounding cylinder surrounding fissile sample

Delayed neutrons (ID: 99999) emitted from **all** fissile isotopes in sample

Irradiation Period
Number of emitted delayed neutrons from **0 s** to **60 s**

Delay Period
Number of emitted delayed neutrons from **60 s** to **60.001 s**

Counting Period
Number of emitted delayed neutrons from **60.001 s** to **240.001 s**

These delayed neutrons are eligible to be counted by the delayed neutron counter as the counter is active during the counting period

!

The *larger* this number, the *more* certain we are that fissile material exists in intercepted sample

Figure 10
MCNP6 Strategy to Calculate Cumulative Number of Delayed Neutrons Emitted from Fissile Sample During Counting Period

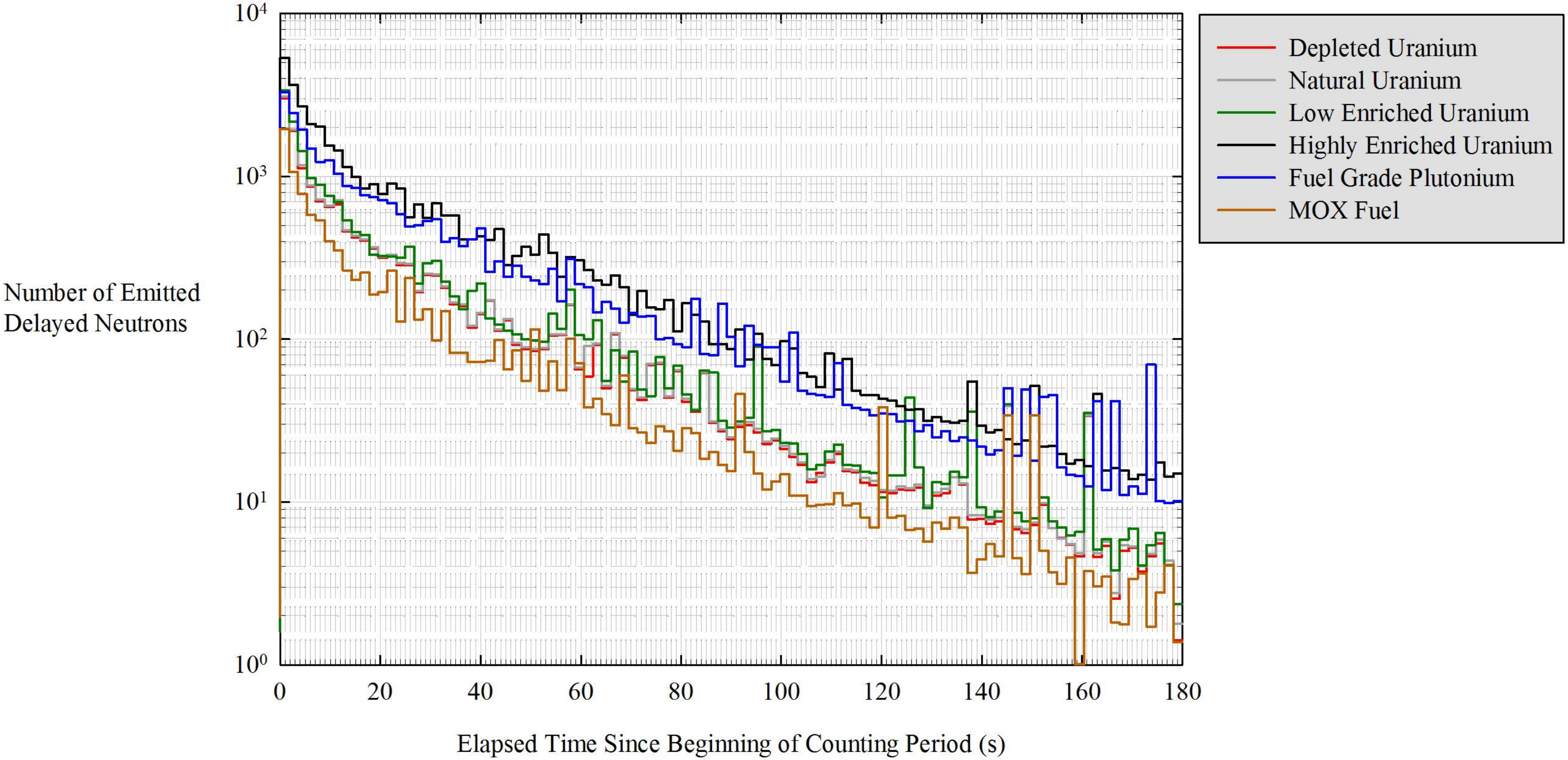


Figure 11

Time-Dependent Instantaneous Delayed Neutron Emission from Fissile Samples (at nominal density) Interrogated by DD Neutron Generator

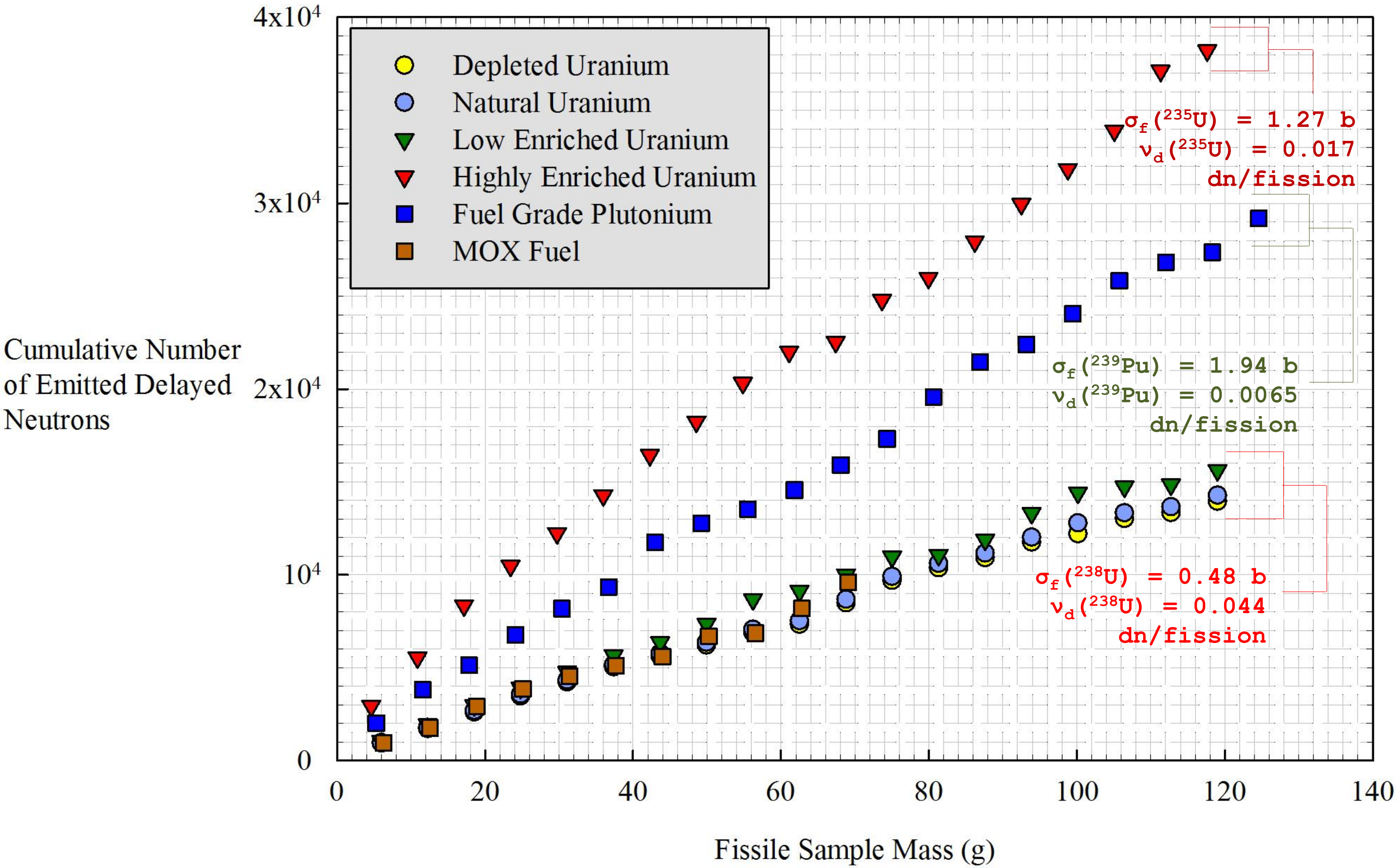
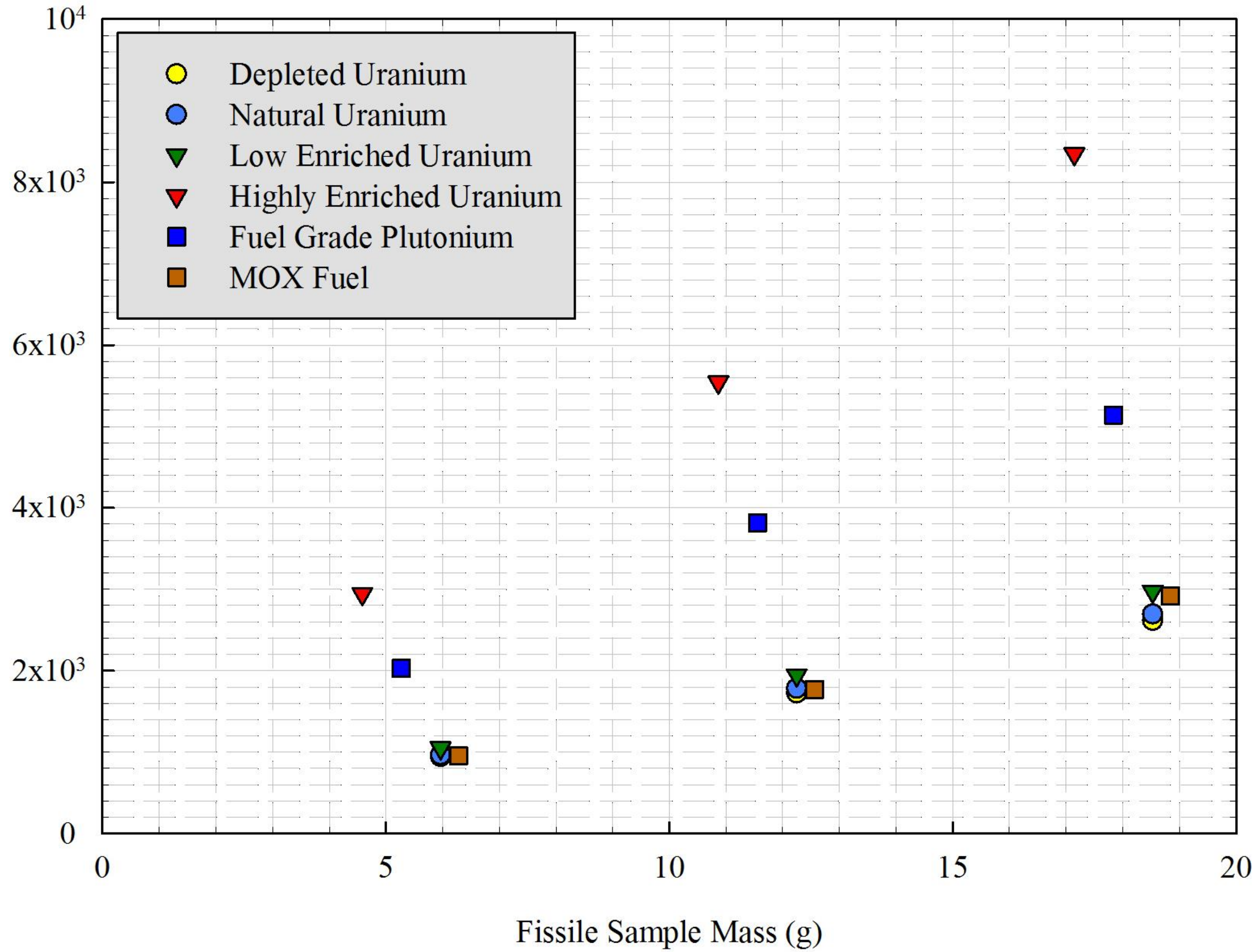


Figure 12

Cumulative Number of Delayed Neutrons Emitted from Fissile Sample Up to 180 Seconds After DD Neutron Generator Termination (ENDF/B-VII.1 Cross Section Obtained from ENDF Nuclear Data File Database [5] and ENDF/B-VII.1 Delayed Neutron Yield Obtained from PENDL [6])

Cumulative Number of Emitted Delayed Neutrons



Considerable delayed neutron yield is achieved at *low* fissile sample mass

Figure 13

Cumulative Number of Delayed Neutrons Emitted from Fissile Sample Up to 180 Seconds After DD Neutron Generator Termination for Fissile Sample Mass Ranging from 0 g to 20 g

Minimum Fissile Material Mass for DD Interrogation Irradiations

Minimum Mass = mass of fissile material that causes count rate in delayed neutron counter to be *just* beyond background count rate

$$\text{Average Count Rate} = \left(\frac{\begin{array}{c}\text{Cumulative Absolute Number of Delayed Neutrons} \\ \text{Emitted from Fissile Sample,} \\ \text{at } \textbf{lowest} \text{ density, over Counting Period}\end{array}}{\text{Length of Counting Period (180s)}} \right) \times \left(\begin{array}{c}\text{Overall Detection} \\ \text{Efficiency of Delayed} \\ \text{Neutron Counter}\end{array} \right)$$

For **each** fissile material, the **minimum** count rate that can be registered occurs when the material is at its **lowest** density

Typically around **20%**

Minimum Fissile Material Mass for DD Interrogation Irradiations

Fissile Material	Mass (g)	Average Count Rate (cps)
Depleted Uranium	5.97	1.05
Natural Uranium	5.97	1.08
Low Enriched Uranium	5.97	1.17
Highly Enriched Uranium	4.58	3.27
Fuel Grade Plutonium	5.28	2.25
MOX	6.28	1.06

Table 1
Average Count Rates Recorded in Delayed Neutron Counter from Fissile Samples Interrogated by DD Neutron Generator

The *typical* background count rate for neutron detectors in the HPNG is **< 1 cps**.

This table suggests that the **minimum** mass of fissile material in an intercepted article must be on the order of **a few grams** in order to produce count rates differentiable from background.

Conclusions and Future Work

- This study used MCNP6 to explore the use of the DD neutron generator at CNL to perform active neutron interrogation of fissile samples

- It is determined that the **minimum mass of fissile material** in an intercepted article, interrogated by the DD neutron generator, needed to produce count rates in a delayed neutron counter that is differentiable from background must be on the order of **a few grams**

- Experimental active neutron interrogation irradiations using the DD neutron generator will take place in the near future

- **This study is funded by the AECL Federal Nuclear Science and Technology Work Plan**

References

[1] Amiel, S. (1962). Analytical Applications of Delayed Neutron Emission in Fissionable Elements. *Analytical Chemistry*, *34*(13), 1683 – 1692.

[2] Chen, A. (2016). Personal Communication. Redwood City, California: Adelphi Technology, Inc.

[3] Andrews, M. (2015). *Delayed Neutron & Gamma Measurements of Special Nuclear Materials, their Monte Carlo Simulations, and Applications – Doctoral Thesis*. Kingston, Ontario: Royal Military College of Canada.

[4] Pacific Northwest National Laboratory. (2011). *Compendium of Material Composition Data for Radiation Transport Modeling*. Richland, Washington: PNNL.

[5] International Atomic Energy Agency. (2017). *Evaluated Nuclear Data File (ENDF) Database*. Retrieved July 2017, from https://www-nds.iaea.org/exfor/endf.htm

[6] Japan Charged-Particle Nuclear Reaction Data Group. (2014). *PENDL: Plotter for Evaluated Nuclear Data Libraries*. Retrieved July 2017, from http://www.jcprg.org/endf/

Appendix:
Analytical Count Rate Estimates

$$N_{dn} = \underbrace{\sum_{j=1}^{k} \underbrace{\sum_{i=1}^{n} \varphi\sigma_f^{(j)} \left(\frac{N_A w^{(j)} m}{M_M^{(j)}} \right) \left(\frac{a_i}{\lambda_i} \right) \left(1 - e^{-\lambda_i t_b} \right) \left(e^{-\lambda_i t_d} \right) \left(e^{-\lambda_i t_{c,1}} - e^{-\lambda_i t_{c,2}} \right)}_{\text{Number of delayed neutrons emitted from decay of all precusor groups produced from fission with } j^{th} \text{ fissile isotope in sample}}}_{\text{Total number of delayed neutrons emitted due to fission with all fissile isotopes in sample}}$$

Adapted from Amiel, 1962 [1]

$$\overline{\frac{dC}{dt}} = \left(\frac{N_{dn}}{t_c} \right) \times \varepsilon$$

Average Count Rate (cps) → $\overline{\frac{dC}{dt}}$

Cumulative Number of Delayed Neutrons Emitted from Fissile Sample during Counting Period → N_{dn}

Overall Detection Efficiency of Delayed Neutron Counter (20%) → ε

Length of Counting Period (180 s) → t_c

Fissile Material	MCNP6 Average Count Rate (cps)	Analytical Average Count Rate (cps)
Depleted Uranium	1.05	1.28
Natural Uranium	1.08	1.29
Low Enriched Uranium	1.17	1.31
Highly Enriched Uranium	3.27	1.87
Fuel Grade Plutonium	2.25	1.35
MOX	1.06	1.20

Table 2 *Comparison of Average Count Rates Predicted from MCNP6 and Analytical Technique for each Fissile Material*