Title: Development of MCNP Training Modules for International Safeguards

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Development of MCNP Training Modules for International Safeguards

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Motivation

- Monte Carlo N-Particle (MCNP) software is a vital tool for international safeguards
- No training modules exist specifically for international safeguards applications nor do any existing training modules address the complete set of specific needs of safeguards practitioners
- Training modules, suitable for virtual and in-person delivery, are being developed to fill this gap

Overview of Units

Unit 1: MCNP Basics for International Safeguards

- Cohesive set of modules to cover basic MCNP material with a focus on building neutron detectors
  - Geometry
  - Common material definitions
  - Fixed source definitions
  - Capture tallies
- Approximately 1 week in length

Unit 2: Advanced Topics in MCNP for International Safeguards

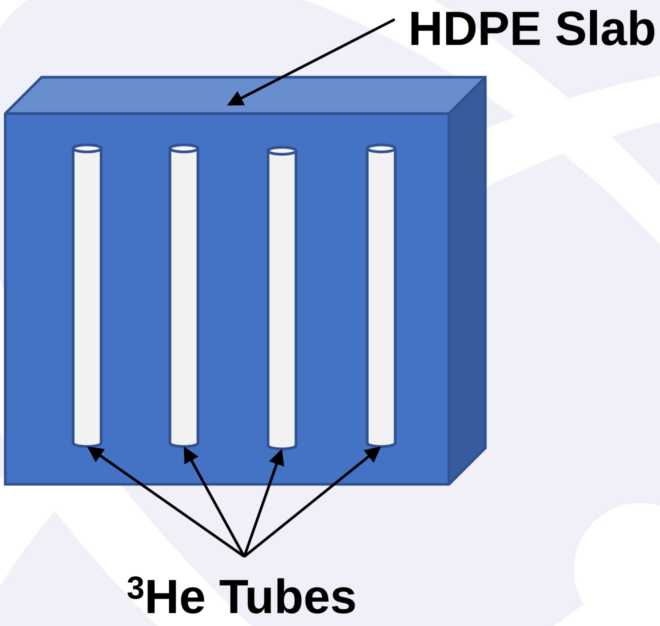
- Mix-and-match modules to cater to the audience
  - Burn-up simulations
  - MCNPTools & PTRAC
  - Gamma detectors
  - Principles of benchmarking
- Approximately 1 week in length with the option to add individual modules to the basic course as time permits

Unit 1: MCNP Basics for International Safeguards

- Geared towards intermediate MCNP users
- Examples focus on international safeguards applications
- Modules include:
  - Basic & advanced geometry concepts
  - Fixed source definitions
  - Tallies and reading output files
  - Nondestructive assay (NDA) system optimization

Basic & Advanced Geometry Modules

- Set-up basic neutron detector:
  - High-density polyethylene (HDPE)
  - ^{3}He tubes
  - Point source
- Provide basic material definitions
- Spent fuel assembly for advanced geometry concepts, such as lattices, universes, and fill



Fixed Source Definitions

- Cylindrical sources containing fissioning radionuclides are most commonly used for international safeguards
  - Example: PuO_2 and UO_2
- Demonstrate how to use source distributions for:
  - Sampling radiation origins throughout the volume
  - Energy distributions
  - Defining source with spontaneous fission and (alpha, n) neutrons
- Choosing the correct fission model
  - FMULT card
  - Importance of PAR=SF for coincidence and multiplicity counter modeling

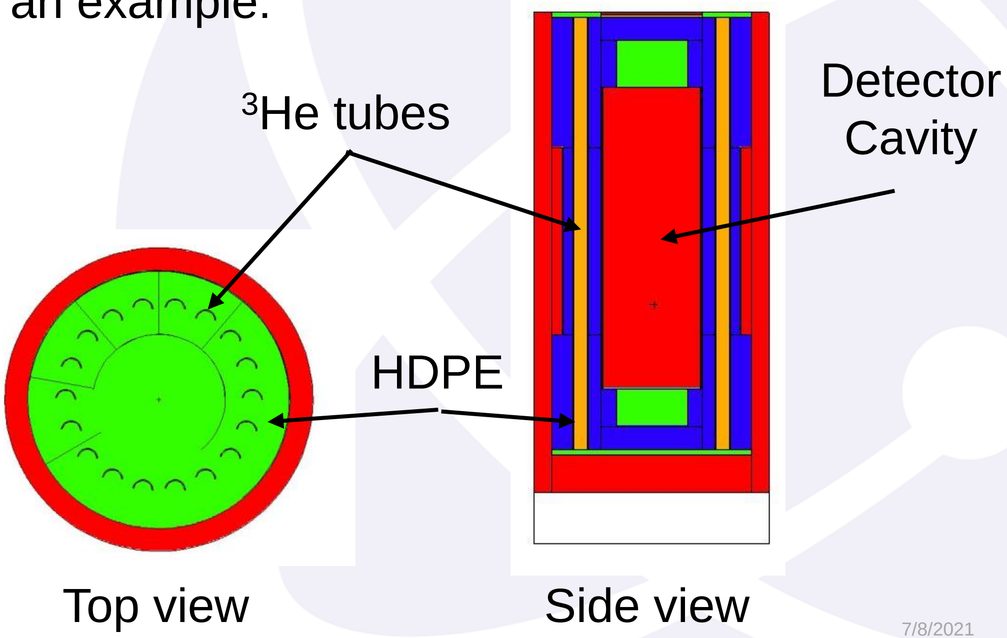
Tallies and Output Files

- Tallies:
  - F4 versus F8 tally
  - Tally multipliers for ^3He
  - Coincidence and multiplicity counting using the CAP function

- Read output files and convert data into Singles, Doubles, and Triples

NDA System Optimization Exercise

- Apply knowledge from entire course to optimize NDA system as a class
- Parameters to consider and how to optimize them using the high level neutron coincidence counter (HLNCC) as an example:
  - HDPE thickness
  - Diameter of ^{3}He tube rings
  - Number of ^{3}He tubes and rings
  - Response to different sources
    - Ex: ^{252}Cf versus PuO_2



Unit 2: Advanced Topics in MCNP for International Safeguards

- Gamma detectors
- Burn-up simulations
- SOURCES-4C & ISC
- MCNPTools/PTRAC for Safeguards
- Principles of benchmarking

Summary

- An MCNP course designed for safeguards practitioners is under development
- The first unit covers basic MCNP topics with relevant safeguards examples at an intermediate level
- The second unit covers advanced topics for international safeguards
- We hope to hold this course internally and domestically in the next FY

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