A methodology for automated radioactive waste characterization

André G. L. Otero, Julio T. Marumo, Ademar J. Potiens Jr.

Email: apotiens@ipen.br
Nuclear and Energy Research Institute, Radioactive Waste Management Department, Av. Prof. Lineu Prestes, 2242, ZIP Code: 05508-000 - São Paulo/SP - Brazil, Fone: +55 11 2810-5880

Introduction

The nuclear technology development must allow safe and secure waste management which proves from the steps of the nuclear fuel life cycle, research centers and industry, and radioisotopes applications in medicine. Proper nuclear waste characterization is complex due to the wide range of materials, processes, and applications. In this work, a methodology for final characterization was developed using deep neural networks.

Methods

The Monte Carlo Methos was applied to gamma spectra simulation in a scenario where the nuclear waste is stored in a 200 liters steel drum that can contain up to 10 different radionuclides. The simulation data was used to train and assess the performance of different deep neural network architectures. The VGG-19 architecture was chosen due to the best overall performance at the classification task, capable of identifying which radionuclides and which activity each radionuclide presented at the gamma spectra.

Results

Simulations were performed to generate a synthetic dataset containing 600 instances. This first step used several different parameters for initial multichannel energy (0.01 eV, 5 eV, 10 eV, 30 eV, 40 eV), number of simulated stories (1.0E+07, 1.0E+08, 1.0E+09), detector to drum distance (41 cm, 46 cm, 51 cm, 56 cm) and radionuclides (Am-241, Ba-133, Cd-109, Co-57, Co-60, Cs-137, Eu-152, Mn-54, Na-22, Pb-210), then a mixing procedure was applied to generate spectra with more than one radionuclide. Finally, the training dataset contains 10,575 instances, and the test dataset contains 825 instances. The final metrics for each task in the model are presented in the table below.

Metric	Training dataset	Test dataset
Accuracy	96.55%	96.01%
Mean Square Error	45.15	97.10
Accuracy with 95% threshold	69.59%	66.57%

Conclusions

The presented results show that the proposed methodology can be an important tool in the nuclear waste characterization process performed routinely by the IPEN`s Service of Radioactive Waste Management, allowing the decrease to occupational exposure to ionizing radiation.

Validation of Fricke xylene gel dosimetry through comparisons between MCNP and TOPAS simulations

Priscila Rodrigues[1,2], Ana Burin[1], Cristhian Talacimon[1], Ilca Medeiros[1], Júlio Pereira[1], Lara El Hajj Teodoro[1], Maria Rigo[1], Paulo Tavares[1], Anna Martins[2], Orlando Rodrigues Jr[1], Carla de Souza[1], Carlos Zeituni[1], Maria Rostelato[1]

Email: priscilasrodrigues@usp.br
[1]Instituto de Pesquisas Energéticas e Nucleares, Universidade de São Paulo (IPEN/USP), Avenida Professor Lineu Prestes, 2242, Cidade Universitária, São Paulo SP, CEP 05508-000, Brazil
[2]Programa de Educação Continuada em Economia e Gestão de Empresas (PECEGE/ ESALQ/USP), Rua Cezira Giovanoni Moretti, 580, Santa Rosa, Piracicaba SP, Brazil.

Introduction

The Monte Carlo (MC) technique has become a standard tool in medical physics, providing reliable numerical results by using large quantities of pseudo-random samplings and a physical description of phenomena. It is possible to simulate radiation transport, with special emphasis on transport involving particles and photons, for Fricke xylene gel (FXG) dosimetry. The objective of this work is to perform computational dosimetry using MCNP and TOPAS codes for the GammaCell 220 Irradiator, based on an experimental configuration that simulates an FXG dosimeter, in order to determine the absorbed dose in the environment.

Methods

The simulations of the GammaCell 220 cobalt-60 irradiator were performed using the radiation transport code MCNP Version 6.2 and TOPAS. The geometry of the irradiator was simulated as closely as possible to reproduce the experimental configuration with a FXG dosimeter with formaldehyde addition. For this work, the MCNP tally used is F6, which calculates the energy deposited in the cell, with units of energy per mass, which coincides with the definition of dose. And for the TOPAS code, volumetric scorers are used for dose or energy calculations.

Results

MC simulations were compared to FXG dosimetry experimental measurements, and both MCNP and TOPAS codes were able to predict radiation dose with similar accuracy, although they showed smaller deviations compared to the experimental values. However, FXG dosimetry has limitations that can affect measurement accuracy, despite being a well-established and widely used experimental method in radiation dosimetry. Figure 1 shows the simulated geometry in TOPAS.

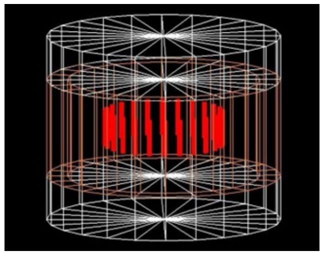

Figure 1: GammaCell 220 simulated geometry from TOPAS code.

Conclusions

Summarizing, Monte Carlo simulation with MCNP and TOPAS is a powerful tool in dosimetry but relies on accurate input parameters and models for precise modeling of energy absorption and distribution in tissue-equivalent materials.