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Monte Carlo simulation-assisted project of a thermalization neutron system for neutrongraphy from ^{241}Am-Be sources: Progress report

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1.Introduction

Neutrongraphy can be defined as a set of techniques that use the neutron as penetrating radiation to investigate the internal structure of an object. This technique began to be developed with greater quality in 1956, with Thewlis and Derbishire, in Harwell, Great Britain, who obtained high quality images from a nuclear reactor, where these researchers indicated possible applications and advantages of neutrongraphy [1]. Since then, there has been a great development of this technique, mainly in relation to imaging with applications in the detection of explosive material, narcotics, research in the biological area and inspections in nuclear fuel cells, among others [2].

Neutrongraphy complements the results of a conventional radiography, which is done with X- or gamma-radiation, because neutrons interact with the nuclei of atoms, and not with their electrosphere. Neutrons have good penetration into heavy materials, such as lead and iron, among others, and have good sensitivity to hydrogen, so neutrongraphy is indicated for tests and inspections on parts with hydrogenated materials. The vast majority of neutrongraphy works has been done using collimated beams of thermal neutrons (Maxwellian energy spectrum below 0.55 eV).

Nuclear reactors, particle accelerators and radioisotope sources are the feasible generators for neutrongraphy. Only radioisotope sources allow portable and transportable arrangements. The current engineering challenge for neutrongraphy is to produce a thermal neutron beam suitable for obtaining the neutrongraphic image from portable sources, enabling the use of this technique on a large scale off the site.

The results of sizing and architecture of a compact device for generating a thermal neutron beam from ^{241}AmBe isotopic sources through Monte Carlo simulations of fast- and thermal-neutron fluxes and gamma-radiations are presented in this work. The optimal dimensions and materials are obtained by maximizing the thermal neutron fluence and minimizing gamma dose in the extracted beam, which meet neutrongraphy requirements.

2.Methodology

Traditional thermal neutron generating systems primarily use graphite as a moderating and thermalizing material [2, 3], which gives them larger dimensions than systems that would be built mainly with hydrogenated materials. Due to this reason, even despite the larger neutron absorption cross section of hydrogen then that of carbon, we are purposing to use hydrogenated materials as moderating- and thermalizing-materials instead of graphite in this work, trying to get a smaller device.
In the design and construction of the device object of this study, the following materials will be used: high density polyethylene (HDPE), graphite, lead, cadmium and borated polyethylene (BPE), the latter for external shielding for radiological protection purposes and to reduce environment radiation background.
HDPE is used in the central core of the device as a neutron moderating and thermalizing material because it has a large fraction of hydrogen atoms in its composition, enabling a more compact arrangement. Graphite is used as a reflective material around the HDPE core, returning to this core part of the neutrons that would escape from the arrangement, in addition contributing to the moderation and thermalization process, as it is a material that also moderates fast neutrons, although with smaller efficiency than HDPE, but with the advantage of having a smaller neutron absorption cross-section than hydrogen. Lead is used as a shield for the X- and gamma-radiation emitted by the ^{241}Am-Be sources and by the neutron capture gamma-rays in the arrangement materials. Lead also helps to minimize the leakage of neutrons from the system, acting as a secondary reflector. BPE is used for shielding thermal neutrons that escape from the device, due to the high boron absorption cross section for thermal neutrons.
PHITS (Particles and Heavy Ions Transport System) code was used for the simulations. PHITS is a Monte Carlo general purpose code for particle transport simulation, that was developed and is maintained in an international collaboration between JAEA, RIST, KEK and others research institutes. It can handle the transport of all particles in a broad spectrum of energy, using multiple nuclear reaction models and multiple nuclear data libraries [4].
As a starting point for the device configuration, the geometry of an old arrangement previously assembled in an empirical way in the Laboratory of Ionizing Radiation of the Institute of Advanced Studies (LRI/IEAV) was used, where some preliminary tests were carried out with eight 100 mCi ^{241}Am-Be sources. The semi-disassembled old arrangement and the scheme of its deimensions and materials, that was used as start-up design for the optimization of dimensions e selection of materials by Monte Carlo simulation, are shown in figure 1.

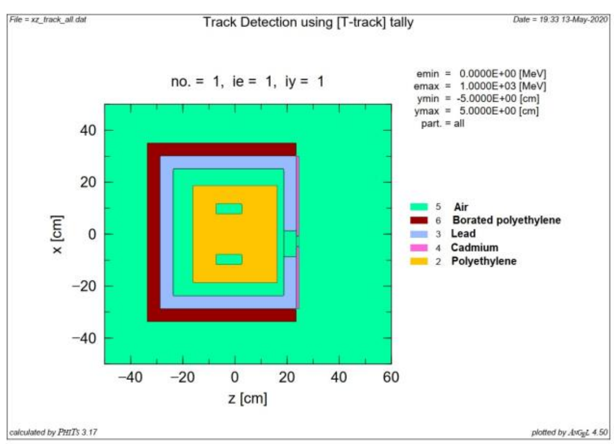

Figure1: Semi-disassembled old arrangement (at left) and its Initial dimensions for simulations (at right)

The fast neutron emission spectrum of ^{241}Am-Be in tabular form was taken from the ISO 8529-1 standard [5]. Several variations in the dimensions of the HDPE moderator block and the positioning of the ^{241}AmBe sources were simulated, maximizing the thermal neutron flux at the exit of the beam extraction channel, as well as minimizing the fast flux and gamma-radiation leakage.

3. Results

After several steps of simulating variations in the device dimensions, optimizing the beam parameters at the extraction channel output, the following results were obtained:

a) Thermal neutron fluence: $2{,}8675 \times 10^{-4}$ cm^{-2} by fast neutron emitted by all ^{241}Am-Be sources;

b) Fast neutron fluence: $3{,}5447 \times 10^{-4}$ cm^{-2} by fast neutron emitted by all ^{241}Am-Be sources;

c) Thermal to fast neutron fluences ratio: 80,9 %; and

d) Thermal neutron fluence to gamma-dose ratio: $3{,}94 \times 10^{5}$n.cm^{-2}.mrem^{-1}.

The final device dimensions for its mechanical manufacture are presented in table 1 and 3-D picture of designed neutron thermalization device is shown in figure 2.

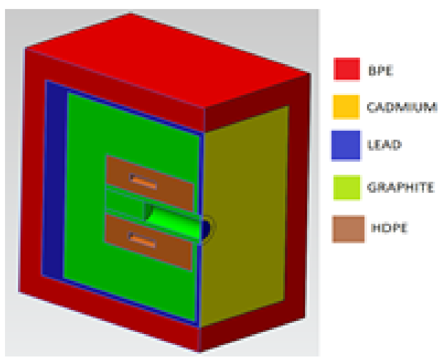

Figure 2: 3-D designed neutron thermalization device

Table1: Main dimensions of the designed device.

Structure	Dimension (cm)
HDPE core (X × Y × Z)	18,0 × 18,0 × 20,0
Distance from the center of the sources to the front face of the HDPE block	12,4
Outside diameter of the beam extraction channel	4,0
Thickness of the inner graphite tube wall of the beam extraction channel	1,0
Graphite corner	4,0
Distance from the center of the sources to the center of the beam extraction channel	6,0
Thickness of graphite reflector (side and rear)	10,0
Thickness of graphite reflector (front)	1,0
Length of the graphite cylinder inside the beam extraction channel	8,8
Lead shieldthickness	1,0
Shielding thickness of boron polyethylene (BPE)	5,0
Cadmium plate thickness	0,05

4. Conclusions

Thermal neutron beam generators with isotopic sources have the advantage of being transportable for carrying out radiographic inspections with neutrons in places where the object to be examined is installed (off site neutrongraphy).

The Monte Carlo computer simulation using the PHITS 3.17 program proved to be very useful and suitable for sizing the thermal neutron device employing small isotopic ^{241}Am-Be fast neutron sources. This tool allowed quick variations simulations in the device components dimensions in order to optimize the parameters that qualify its performance: maximum thermal neutron fluence at the extraction channel output, higher ratio between thermal and fast fluences and higher ratio of thermal neutron fluence to gamma dose rate. This simulation tool provided savings in time and materials, which would be needed in prototypes manufacturing and perform development tests.

The device beam parameters with the optimized dimensions in this work with the aid of the Monte Carlo method will be experimentally verified by means of a concept demonstrator built according to the mechanical design presented here, using low activity ^{241}Am-Be isotopic sources. The final performance of the system in terms of neutron fluence will be determined by the activity of the ^{241}AmBe sources that come to be employed.

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