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Improvement study for neutron shielding of the iodine-123 production line for the CV-28 Cyclotron

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1. Introduction

The Institute of Nuclear Engineering has a radiopharmaceutical production sector that manufactures ultrapure Iodine-123 for medical applications. Radiopharmaceuticals are radioisotopes linked to chemical compounds, with the aim of assisting in radiodiagnosis and disease monitoring, used in nuclear medicine. The 123I is one of the most relevant produced at Divisão de Radiofármacos (DIRA-IEN). The ultrapure 123I emits gamma radiation and has a half-life of about 13.2 hours, making its distribution appropriate for nuclear medicine throughout Brazil. The production of this radionuclide happens with the use of the CV 28 cyclotron particle accelerator. This is a variable energy multiparticle accelerator able of producing several radionuclides and used for various research but is currently set to accelerate proton only with a maximum energy of 24 MeV. The Institute chose to keep constant the operational parameters of the machine to meet the routine production of radiopharmaceuticals. The particles are generated and accelerated in the central region of the equipment and conducted through primary and secondary beam lines to the target. The main beam line is distributed, and the beam is driven to the external transport line, which contains the material to be irradiated, in this case Xenon-124 gas.
This target cave, where a xenon-124 gas target is positioned, reaches the dose level limits set out by the Brazilian National Nuclear Energy Commission Standard 3.01 [1] in its surrounding area. At the cave access door, it is necessary to use a protection block formed by lead and polyethylene doped with boron, however this shield does not meet the minimum exposure limits of the standard in some specific points of the facilities. For this reason, it is necessary to improve this shield and minimize the dose received by individuals occupationally exposed to the lowest possible level in the region of work. The purpose of the work is to determine the neutron flow around the production target of the 123I and in the external region of the cave, to determine materials for neutron shielding that allows the reconstruction of the protection block in order to reduce the dose rates at these points of the installation to the lowest possible level, satisfying the ALARA (As Low as Reasonably Achievable) principle and contributing to the installation's radiological protection program.

2. Methodology

The 24 MeV proton beam ($^{1}_{1}H$) is conducted to irradiate a ^{124}Xe target, with an electrical current of 20 μA. There is an instability by the proton excess in nuclides, and then a positron (+β) is emitted by the nucleus, or an Electron Capture (EC) occur with the emission of gamma radiation (^{123}I), following the reaction [2], see Eq. 1:

$$^{124}_{54}Xe\ (p, 2n)\ ^{123}_{55}Cs \ \rightarrow\ ^{123}_{54}Xe \ \rightarrow\ ^{123}_{55}I \tag{1}$$

During the irradiation procedure in the ^{123}I target chamber, the emission of fast neutrons and gamma radiation with high energies occur as a product of nuclear interactions, which justifies the need for efficient shielding. Using the MCNPX code [3] to implement computational models with different materials in different associations, it will allow estimating the ideal composition to minimize the doses generated by neutrons.

In the initial model a sphere was simulated with a radius of 100 cm and 10 cm thick, representing the iron door, two more spheres were added to coating, modifying in each simulation the composition of them according to the material to be analyzed: boron polyethylene, paraffin and water.

A point detector was positioned outside the sphere on the same axis of the source one meter away. In the code entry file, Tally F5 was used to estimate the flow in a point detector (neutron.cm-2) with the aid of the conversion coefficient recommended by Conversion Coefficients for use in Radiological Protection against External Radiation - ICRP publication nº. 74 [4], used to convert neutron flow into an equivalent ambient dose, making it possible to analyze which material presents the greatest reduction in the equivalent dose.

A survey of the neutron energy spectrum emitted by the system is being carried out using the neutron activation analysis technique. The leaf activation method consists of subjecting the material of choice to a neutron flow, these materials were properly positioned in the area of access to the cave during the irradiation procedure for the production of 123I. The neutron interacts with the sample and produces radioactive isotopes from the nuclei that compose it. Radioisotopes decays emitting radiation, which was measured by a HPGe Hyper pure Germanium detector. Four plates of different materials were used: gold-197, indium-115, nickel-58 and aluminum-27. The chosen leaves present different cross sections for the energy groups. Gold is used with activation shock section values referring to thermal and epithermal regions, the Indian refers to the epithermal and fast, and nickel and aluminum represent the fast neutron range, presenting different activation thresholds within this region.

3. Results and Discussion

Keeping the thickness of the hydrogenated material in the initial simulations always the same (46.5 cm) and modifying only the material to be studied, it was possible to observe that there was a reduction of the equivalent environmental dose in one of the results, Fig. 1.

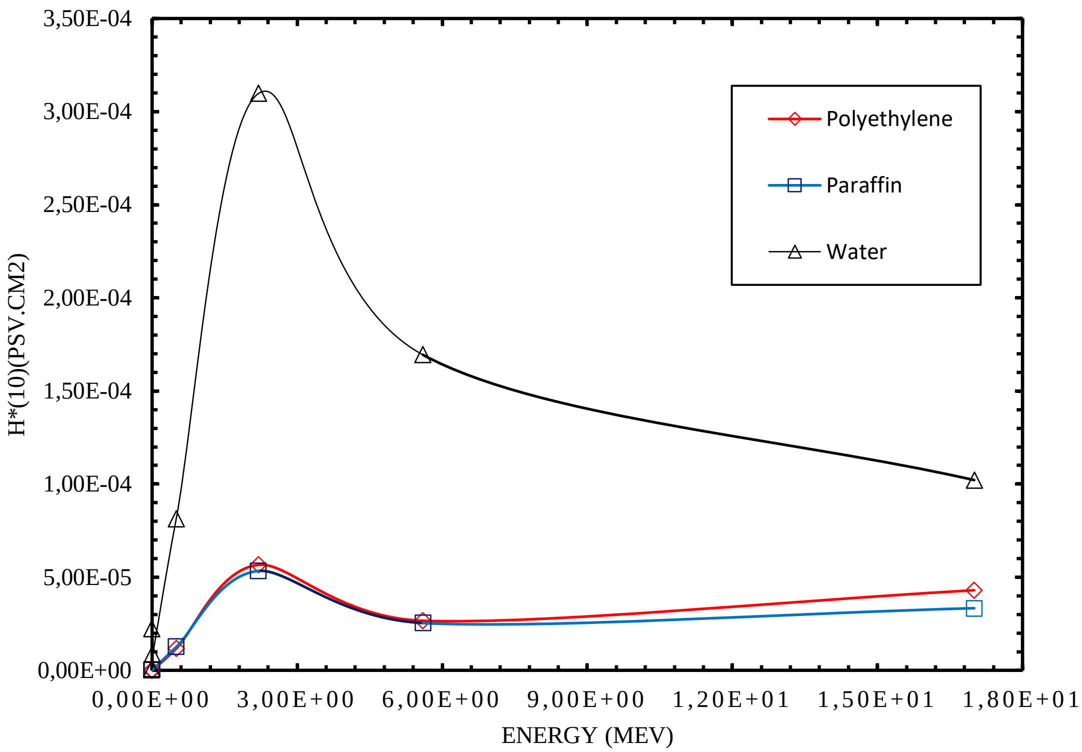

Figure 1: Spectra generated from results of the moderator materials.

Although dose levels in the thermal neutron range have similar behavior, it is possible to observe that paraffin showed a small dose reduction in the high energy neutrons range, when compared to boron polyethylene, which is the material of the current shielding, while water did not present satisfactory results. Fast neutrons suffer less interactions with matter, which makes it more difficult to shield, so it is necessary to thermally moderate the fast neutrons to later capture the thermal neutrons, for this, new combinations are being studied.

4. Conclusions

Monte Carlo simulations are still underway with the development of a more detailed mathematical model. Together the analysis of data from the experimental phase is being finalized and the neutron energy spectrum data emitted by the system is being finalized. The survey of the neutron energy spectrum will make it possible to determine and know the neutron flow in the region of access to the production cave of 123I, and thereby evaluate through mathematical models the materials for shielding. At the end of the study, with the new simulations completed it will be possible to determine if there is any configuration more efficient than the current one to perform the shielding.

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