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SHIELDING DESIGN FOR A RESEARCH FACILITY IN THE IEA-R1 REACTOR

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ABSTRACT

This work presents the final results of a set of simulations run to achieve the most efficient shielding design within initial premise: low cost piecewise materials so to attain to the disposed budget and to allow future improvements in and around the facility. Simulations were performed by DOT 3.5 transport code and ISODOSE code was used convert radiation fluxes into dose. As long as the design was defined, MCNP code was employed to evaluate dose miscount derived by geometric modeling constrains which are inherent to DOT 3.5.

1. INTRODUCTION

A facility for research in Boron Neutron Capture Therapy – BNCT – in the IEA-R1 Reactor[1] is under construction in IPEN-CNEN/SP and its execution is finishing. This experimental facility, showed in a schematic view in figure 1, is comprised of a 20.32 cm beam port – Beam Hole #3 (BH#3) – of the IEA-R1 Reactor. Moderators and filters, sample holder and shielding will be placed inside BH#3 and, externally, in the experimental room, a biological shielding will be built to provide a proper isolation from one facility to another as well to fulfill the radiation protection requirements to allow people circulation in the area.

As the sample might be taken out from the BH#3 with the reactor under operation, causing the removal of the internal lead shielding from BH#3, a strong neutron and gamma fields will be generated inside the biological shielding. The purpose of this work is to design this biological shielding with a so high attenuation factor – dose rate inside/outside – as to satisfy the radiological protection rules.

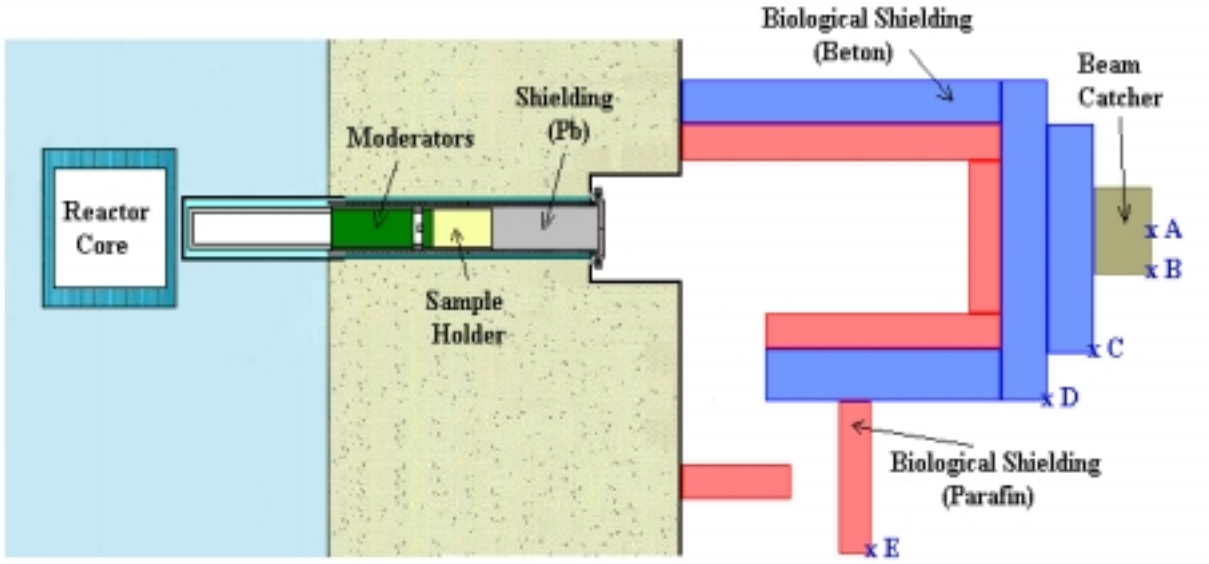

Figure 1. Schematic view of the BNCT facility which is under construction in BH#3 of IEA-R1 Reactor in IPEN.

2. DESCRIPTION OF THE ACTUAL WORK

The prime condition for this shielding design relied on the use of low cost materials and on an easy to build and rebuild structures; these parameters conduced to the use of paraffin boxes and concrete blocks with respective masses of 25 kg and 34 kg. An old beam catcher will also be used to reinforce the shielding. It comprises in a box of leaded walls with borate paraffin inside it.

The biological shielding will be consisted of a set of walls and roof made from paraffin boxes and concrete blocks with the beam catcher standing along the BH#3 axis outside the shielding.

A computational model of the biological shielding was used and a set of simulations were performed for many different thickness of the walls and the roof, aiming to obtain neutron and gamma ray dose rates outside this shielding according to the radiation protection standards. Calculations were done using the discrete ordinates transport code DOT 3.5[2] and the multigroup cross sections data generated by NJOY/AMPX II system. The computational program ISODOSE converted the results from DOT neutron and gamma ray fluxes to dose.

Besides these calculations, this biological shielding has been modeled using the Monte Carlo transport code MCNP 4B[3] with the purpose to compare the results from the application of the two methodologies.

3. SHIELDING RESULTS

The thickness of the roof and walls of the biological shielding were determined from the calculation results which attained, outside the shielding, dose rates that fulfilled the radiation protection standards, as are presented in table I for some representative points around the shielding. Roof and walls will be consisted of a 30 cm of paraffin layer followed by a 38 cm of concrete one except for the front wall that will be thicker – 76 cm of concrete. The gamma ray, neutron and gamma-neutron attenuation factor obtained thereafter are high for the frontal wall: 15300, 600 and 3000; and for the roof and the remaining walls are: 5100, 350 and 1600.

Table 1: Dose rate calculated for several positions around the outer side of the shielding. Positions are identified in figure 1.

Position	Dose (μSv/h)		
	gamma	neutron	gamma - neutron
A	9.5	1.0	10.5
B	51.6	8.4	60.0
C	47.9	8.8	56.7
D	65.1	35.5	100.6
E	77.5	36.0	113.5

4. MCNP

As long as the biological shielding design had been determined by DOT – ISODOSE coupled calculations, a geometrical sensitivity analysis was initiated using MCNP. This analysis was motivated by the fact that DOT is a two dimensional radiation transport code and an exactly geometrical description can only be attained by those configurations which presents cylindrical symmetry. Therefore most of the shielding configuration evaluations must be conducted in sets of calculations and even so dose distribution is inferred in the spaces between the planes contemplated in calculations.

As MCNP relies one of its major potentials on geometrical flexibility to reproduce practically any problem, it has been used to study the dose miscounts present in DOT calculations derived from its restricted geometrical characterization.

To accomplish the present study, simulations were performed for two distinct shielding design: one which represents the real final proposed shielding and another which describes, in a cylindrical symmetry, the shielding with the lateral door, in a similar geometry described by DOT. For both cases, dose estimation was accomplished through a two step simulation to allow the description of neutron and photon sources. These samples were defined in a plane perpendicular to the beam-hole, next to its external aperture. This was done to avoid a time consuming stage represented by the radiation transport from the reactor kernel to this plane. Radiation fluxes calculate by DOT have been used as input data to represent the

sources. As the beam-hole and the facility presents a cylindrical symmetry all the way up to this plane there seems to be no incongruity.

In a prime step, results from DOT-ISODOSE have been compared with a similar case run by MCNP. Figures 2 and 3 show the neutron dose distribution profiles for parallel axes displaced along different radial distances. Results shown were obtained respectively by DOT-ISODOSE and MCNP. Dose distribution was not evaluated inside shielding materials and, therefore, is not shown in figure 3.

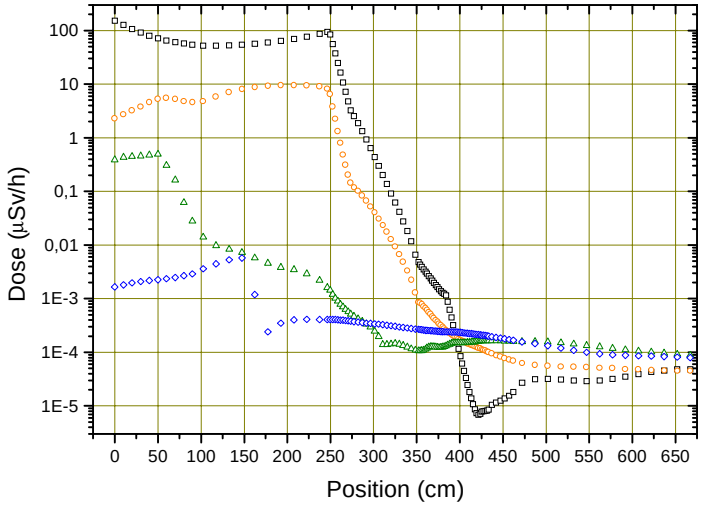

Figure 2. Neutron dose distribution profiles obtained by DOT-ISODOSE calculations. Black square: central axis; orange circles: axis traversing the front wall; green triangles: axis traversing the lateral wall, blue diamonds: outer axis (apart from shielding).

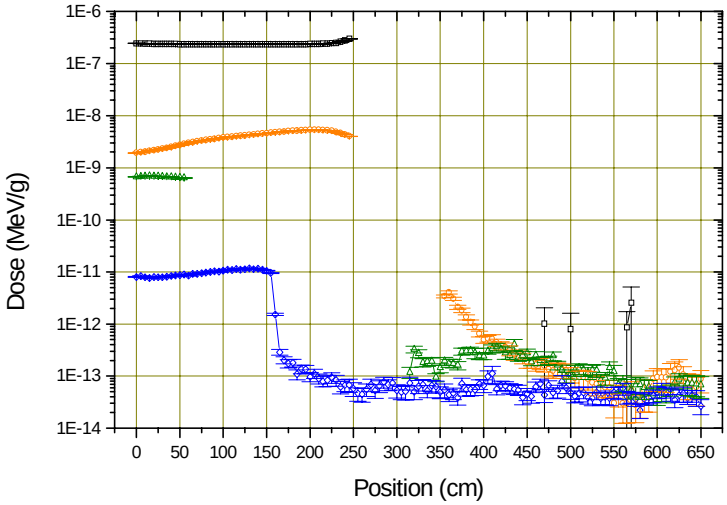

Figure 3. Neutron dose distribution profiles obtained by MNCP simulations**.** Black square: central axis; orange circles: axis traversing the front wall; green triangles: axis traversing the lateral wall, blue diamonds: outer axis (apart from shielding).

Figures 4 and 5 show the results obtained by MCNP simulations run for distinct description geometrical configuration modeling.

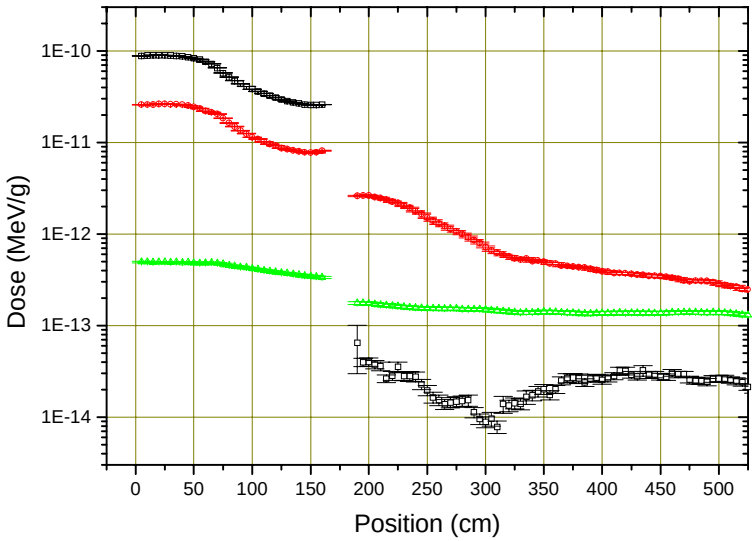

Figure 4. Dose distribution profiles obtained by MNCP simulations along a single axis in a cylindrical symmetrical configuration. Black squares correspond to neutron dose; red circles: secondary gammas (n,g); green triangles: sample gammas.

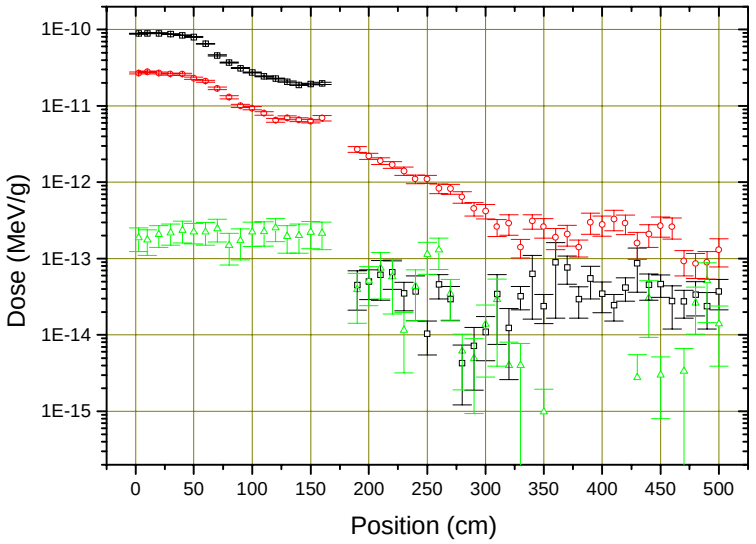

Figure 5. Dose distribution profiles obtained by MNCP simulations along a single axis in a real modeled configuration. Black squares correspond to neutron dose; red circles: secondary gammas (n,g); green triangles: sample gammas.

The profiles presented in figures 4 and 5 are correspondent to an axis parallel to the beam hole axis but placed 165 cm away. It corresponds to a line, which traverses the paraffin wall that works as the entrance shielding. Dose profiles have been presented according to tallies output from MCNP, however it express in a clearer way which radiation component has its evaluation more affect by the different geometrical consideration. It can be seen that in a cylindrical shielding configuration, gamma dose seems to overestimate when compared with the results from real shielding modeling simulation.

CONCLUSIONS

DOT – ISODOSE coupled calculations is still the most efficient and feasible procedure to be employed in shielding design studies as it presents reliable results and faster results compared with MCNP. It should be pointed out that although MCNP presents a more flexible and powerful tools to accomplish radiation transport evaluations, it is time-consuming code, which becomes even an eager time consuming procedure when evaluating shielding design. The reason for this is that confidence in results from MCNP is related with their uncertainties which are proportional to the number of particles that reaches the tallied position. However in shielding design evaluations this number is intended to be reduced.

A shielding design project has been conceived by DOT-ISODOSE calculations attaining the proposed goals: low cost materials in a flexible structure. MCNP has shown the adequacy of DOT calculations and has been presented as toll for dose evaluation refinement.

ACKNOWLEDGEMENTS

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[2] "DOT 3.5 – Two Dimensional Discrete Ordinates Radiation Transport Code", CCC-276. Oak Ridge National Laboratory, USA.

[3] "MCNP – 4A, Monte Carlo N-Particle Transport Code System", CCC-200, Oak Ridge National Laboratory, 1996.