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Neutron flux assessment of a neutron irradiation facility based on inertial electrostatic confinement fusion

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HIGHLIGHTS

- IEC fusion generators were included in a design of irradiation facility for BNCT.
- Neutron flux distribution calculations were performed for the IEC based facility.
- IEC generators can be combined to produce irradiations with thermal fluxes around 10^9 n cm^{-2} s^{-1}.

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ABSTRACT

Neutron generators based on inertial electrostatic confinement fusion were considered for the design of a neutron irradiation facility for explanted organ Boron Neutron Capture Therapy (BNCT) that could be installed in a health care center as well as in research areas. The chosen facility configuration is "irradiation chamber", a $\sim 20 \times 20 \times 40$ cm^3 cavity near or in the center of the facility geometry where samples to be irradiated can be placed. Neutron flux calculations were performed to study different manners for improving scattering processes and, consequently, optimize neutron flux in the irradiation position. Flux distributions were assessed through numerical simulations of several models implemented in MCNP5 particle transport code. Simulation results provided a wide spectrum of combinations of net fluxes and energy spectrum distributions. Among them one can find a group that can provide thermal neutron fluxes per unit of production rate in a range from $4.1 \cdot 10^{-4}$ cm^{-2} to $1.6 \cdot 10^{-3}$ cm^{-2} with epithermal-to-thermal ratios between 0.3% and 13% and fast-to-thermal ratios between 0.01% to 8%. Neutron generators could be built to provide more than 10^{10} n s^{-1} and, consequently, with an arrangement of several generators appropriate enough neutron fluxes could be obtained that would be useful for several BNCT-related irradiations and, eventually, for clinical practice.

1. Introduction

Neutron production from fusion reactions has been an important concern since the beginnings of fusion science and technology. Fusion field of study, mostly devoted to energy production, has been focused in neutrons as undesired by-products of fusion combustion of deuterium–deuterium (DD) and deuterium–tritium (DT) fuels. However, there have also been researches and developments that thought of fusion as a safe manner for production of neutrons for different applications and gave birth to a broad family of compact fusion-based neutron generators. These are small fusion reactors based on confinement principles different from those of reactors designed for energy production and can be built on a lab-bench scale (Miley and Murali, 2014; Sved 2003). The combination of size, relative ease of construction and operation, and safety levels have made of them a tempting possibility for their utilization in medical environments where neutrons are required and, specially, for Boron Neutron Capture Therapy (BNCT).

The main limitation for their application to BNCT has been the small levels of obtainable fluxes. Nevertheless, recent developments have brought successively increasing performances and promise to continue the production rate growth. One of the most promising devices are those based on inertial electrostatic confinement (IEC) fusion, which have been showed to be simple, reliable, durable, and one of the most compact ones (Miley and Murali, 2014; Sved, 2003). Accordingly, a facility considering this type of generators was conceptually designed which could be installed in a health care center as well as in research areas (Sztejnberg Gonçalves-Carralves, 2012). The design considers operation with DD or DT fuels, which would generate 2.45 or 14 MeV

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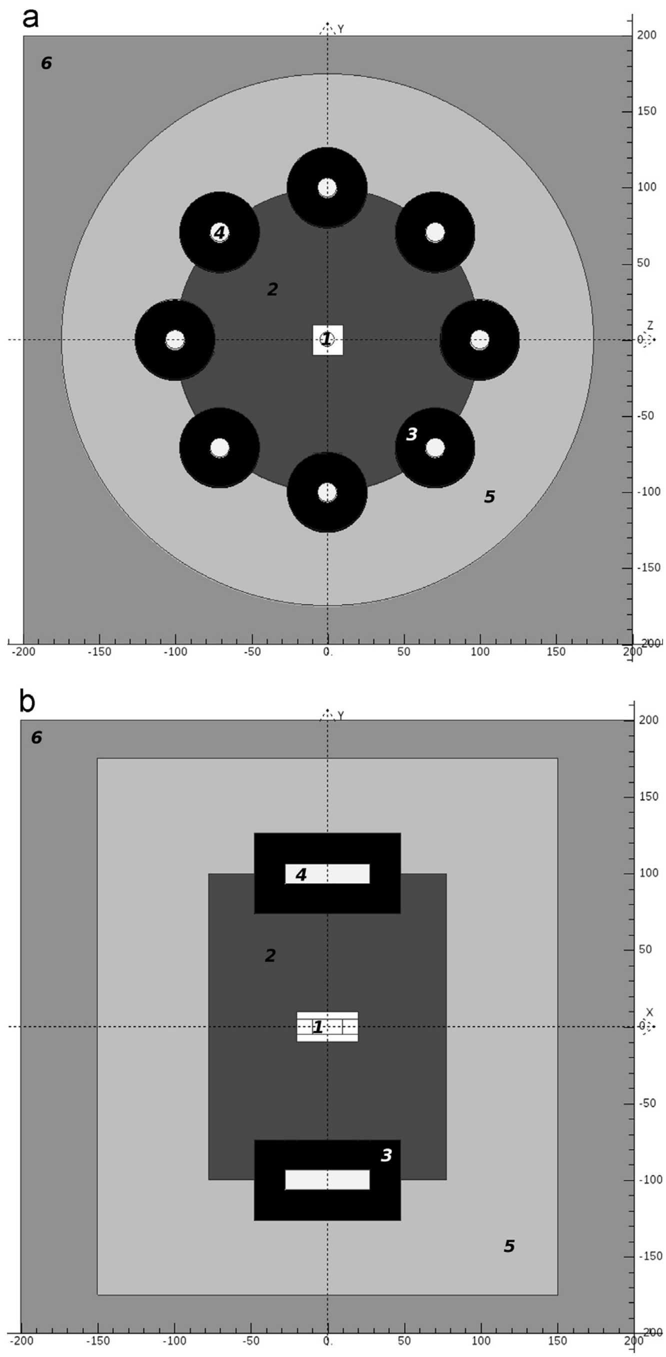

Fig. 1. Visualization of an MCNP model of the fusion-based neutron irradiation facility: (a) transverse and (b) longitudinal views at the center. Regions: 1 irradiation cavity; 2 moderator, 3 converter, 4 neutron generator, 5 internal reflector, and 6 external reflector (and/or shielding). (Scale unit: cm).

neutrons, respectively. As shown in Fig. 1, the chosen facility configuration is an "irradiation chamber" in a thermal column like facility, which could be an alternative to the already existing fission reactor based thermal columns for BNCT (Bortolussi and Altieri, 2007; Miller et al., 2009). The facility was named CINBF, the Spanish acronym for fusion-based neutron irradiation chamber ("channel" or "cavity" would work as well). It is based on a $\sim 20\times 20\times 40\ \mathrm{cm}^3$ air cavity near or in the center of the facility geometry where samples to be irradiated can be placed. The rest of the facility contains neutron generators and structures to give neutrons the appropriate spectral distribution and provide an appropriate shielding. A numerical dosimetry feasibility study was performed for explanted organ BNCT, which demonstrated the applicability of the design (Sztejnberg et al., 2012). This study is an extended neutron flux distribution analysis that considers several variations of the previous design. It aims at finding the path to neutron flux optimization in order to match BNCT requirements where thermal neutron flux must be maximized and fast-to-thermal flux ratio minimized.

2. Materials and methods

The presented irradiation facility consists of five main components or regions, as can be seen in Fig. 1: a set of neutron generators, converters, a reflector region, a moderator region, and the irradiation channel or cavity. For this case, cylindrical neutron generators were chosen. Converters are cylindrical shells around the generators with the purpose of obtaining more neutrons at lower energies from the DT-generated neutrons, utilizing (*n,2n*) nuclear reactions. The reflector region has the goal to scatter neutrons back to the center of the facility as much as possible. This part of the facility is also the first stage of shielding towards the external world. The moderator region must reduce fast neutron population, ideally to zero. This does not directly apply to epithermal neutrons that, depending on the case, could be useful for treating deep sited malignancies in large organs. The irradiation cavity is strictly what its name indicates, a hollow space where sample can be inserted.

Neutron flux distributions were assessed through numerical simulations of several models implemented in MCNP5 particle transport code (X-5 Monte Carlo Team, 2005). The calculations were primarily designed to study different manners of balancing scattering, capture, and (*n,2n*)-reaction processes in order to optimize neutron flux in the irradiation position. Typical materials for neutron moderation and reflection were considered according to their neutronic properties such as reaction cross-sections and neutron slowing down parameters. Table 1 shows scattering, capture and (*n,2n*) reaction cross sections for different energy regions. Large scattering and (*n,2n*) cross sections are desired for slowing down and multiplication, respectively, while small (ideally null) capture cross sections are necessary to allow the neutrons travel between the generators and the irradiation cavity. Table 2 shows, according to elastic scattering theory (Duderstadt and Hamilton, 1976), the energy loss parameters α and $f_{\Delta E}$. The first one tells what is the minimum energy that an outcoming neutron can have after a collision and the second one tells what is the average energy loss per collision. For slowing down purposes, small α and large $f_{\Delta E}$ are desired.

This study focuses on different utilization of graphite and heavy water due to the combination of their characteristic small capture (at most $2.7\cdot 10^{-4}$ and $3.8\cdot 10^{-5}\ \mathrm{cm}^{-1}$, respectively) and large thermalizing efficiencies (for C, α is 72% and $f_{\Delta E}$ is 14% and, for D_2O, both are 33%, scattering cross sections larger than $0.089\ \mathrm{cm}^{-1}$). Bismuth is also in the focus of this study due to the fact it has a large secondary additional neutron production (multiplication) cross section ($0.062\ \mathrm{cm}^{-1}$) as well as heavy water does ($0.012\ \mathrm{cm}^{-1}$), compared to other materials. Bismuth also has a relatively low thermal neutron capture cross section ($0.26\ \mathrm{cm}^{-1}$). Small capture cross section materials are chosen for allowing larger neutron fluxes and also for helping reduce activation levels. Accordingly, D_2O was chosen as first option for moderator, Bi and/or D_2O for converter, C and/or D_2O for reflector. Converter thickness was studied between 0 and 30 cm.

Source-to-irradiation position distance (*s2ip*) was varied between 33 and 100 cm. Reflector dimensions ranged from 1.5 to 8 m. Neutrons from DT (14 MeV) and DD (2.45 MeV) fusion reactions were considered to specify the sources for different models. *SDEF* was utilized for defining straight line sources in the axis of each of the generator regions. Isotropic angular distribution were

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