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ORIGINAL PAPER

Numerical optimisation of the fission-converter and the filter/moderator arrangement for the Boron Neutron Capture Therapy (BNCT)

Grzegorz Tracz, Ludwik Dąbkowski, Dominik Dworak, Krzysztof Pytel, Urszula Woźnicka

Abstract The paper presents results of the numerical modelling of the fission-converter-based epithermal neutron source designed for a BNCT (Boron Neutron Capture Therapy) facility to be located at the Polish research nuclear reactor MARIA at Świerk. The unique design of the fission converter has been proposed due to a specific geometrical surrounding of the reactor. The filter/moderator arrangement has been optimised in order to moderate fission neutrons to epithermal energies and to get rid of both fast neutrons and photons from the therapeutic beam. The selected filter/moderator set-up ensures both the high epithermal neutron flux and the suitably low level of beam contamination. The elimination of photons originated in the reactor core is an exceptional advantage of the proposed design. It brings one order of magnitude lower gamma radiation dose than the permissible dose in such a type of therapeutic facility is required. The MCNP and FLUKA codes have been used for the computations.

Key words BNCT • fission converter • FLUKA • MCNP • Monte Carlo • neutrons

G. Tracz✉, D. Dworak, U. Woźnicka
Department of Environmental and Radiation Transport Physics,
The Henryk Niewodniczański Institute of Nuclear Physics,
152 Radzikowskiego Str., 31-342 Kraków, Poland,
Tel.: +48 12/ 662 83 25, Fax: +48 12/ 662 84 58,
e-mail: Grzegorz.Tracz@ifj.edu.pl

L. Dąbkowski, K. Pytel
Reactor Analysis and Technology Department,
Institute of Atomic Energy,
05-400 Otwock-Świerk, Poland

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Introduction

Boron Neutron Capture Therapy (BNCT) [4, 5] is a form of a radiation therapy used for treatment of some kinds of cancer (mainly glioblastoma multiforme) which, due to their character, are hard to be extracted with surgical methods. The BNCT method involves the delivery of a capture compound – the ^{10}B isotope is usually used – which preferentially concentrates in the cancer tissue, followed by the irradiation of the patient with epithermal neutrons. The incident neutrons are moderated in patient's body to thermal energies and cause the $^{10}B(n,\alpha)^{7}Li$ reaction. The reaction products, i.e. ^{7}Li and α particles, deposit their energies on the reaction spot – typically within $\sim$10 μm of the reaction origin. When concentration of the ^{10}B isotope in the cancer tissue is considerably larger than in the healthy tissue, a proportionally higher radiation dose is delivered to the tumour and makes the method effective. During the therapy, an adequate number of epithermal neutrons incident patient's skin in a suitable time should be provided. One estimates [4] that the time of irradiation does not exceed 10÷15 min when the epithermal neutron flux is at least 10^{10} n/cm^2·s at the therapy position.

Currently, clinical trials of the BNCT method utilise nuclear reactors as the neutron source. The MARIA reactor is to be used for the planned Polish BNCT facility. Because intensity of neutron flux at the outlet of the reactor duct does not meet the therapy requirements (mainly thermal energies) it is essential to use a neutron converter, which contains a fissionable material. Fast neutrons from the ^{235}U fission will be slowed down using a filter/moderator set-up. Photon and fast neutron doses should be suitably reduced with some filters not to exceed 10% epithermal neutron dose.

Typical constructions of the BNCT neutron sources are located along the reactor duct. That location of the BNCT set-up simply ensures spatial homogeneity of the therapeutic neutron beam at the irradiation plane. The technical limitations of the MARIA reactor surrounding force to find untypical solution. The neutron source has to be placed at a specific angle in respect to the selected reactor duct. It complicates much the structure of the whole irradiation set-up. An adaptation of the solutions applied in the world-known BNCT facilities is not possible. Due to the foregoing, the neutron beam is expected to be geometrically heterogeneous at the inlet of the filter/moderator system. The homogenisation of the therapeutic beam may cause losses of the intensity of the epithermal neutron flux.

Two new unique solutions have been proposed for the bent geometry [6]. The homogenisation of the neutron flux can be realised either by the irregular arrangement of the fuel elements of the uranium converter (Variant I) or by the disturbance of the primary reactor beam inside a scattering block of variable density (Variant II). Both of variants should create a homogeneous neutron flux on the input to the filter/moderator system.

A typical fission converter (Variant R) situated on the axis of the primary neutron beam has been considered, too. Variant R presents the Reference for the proposed solutions. The advantages and disadvantages of the projects are presented in relation to Variant R.

A design of the irregular space distribution of the fuel plates (Variant I) and/or of the density of the scattering block (Variant II) has been done according to the consideration given in [6]. A scheme of the analysis is shown in Fig. 1. The primary neutron beam from the reactor hits the block and a secondary neutron beam arises. The drawn block presents a fissile material (converter) or a neutron scatterer. Using a simple calculation model one can estimate the density distribution along the length of the block assuring homogeneous distribution of secondary neutron sources.

The density of secondary neutron sources $\varphi_s(x)$ in a given point inside the block is proportional to the density of the primary beam and to the fission or scattering macroscopic cross section $\rho(x)\Sigma_r$, where $\rho(x)$ is the relative material density of the block (with boundary conditions: $\rho(0) = 0$ and $\rho(L) = 1$):

$$\varphi_s(x) \sim \frac{\rho(x)\Sigma_r}{(d-L+x)^2}\exp\left(-\int_0^x \rho(x)\Sigma_{\text{tot}}\,dx\right) \tag{1}$$

where Σ_{tot} is the total macroscopic cross section of the

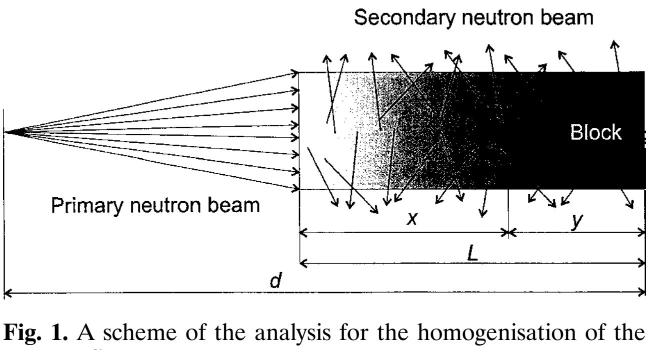

Fig. 1. A scheme of the analysis for the homogenisation of the neutron flux.

material of the block with the relative density $\rho = 1$, dimensions L, d and x are defined in Fig. 1. The even distribution of the secondary neutron source is obtained for such a material density distribution $\rho(x)$ for which the function $\varphi_s(x)$ is constant. The following distribution fulfils that condition:

$$\frac{d\rho}{dx} = \frac{2\rho}{d-L+x} + \Sigma_{\text{tot}}\,\rho^2 \tag{2}$$

which gives:

$$\rho(x) \equiv \rho(L-y) = \frac{(d-y)^2}{d^2 + \Sigma_{\text{tot}}\,y\left(d^2 - yd + \frac{1}{3}y^2\right)} \tag{3}$$

for the material density boundary conditions mentioned above.

The formula (3) has been used to calculate the spatial distribution of the layers of fuel plates for Variant I of the converter and to calculate the density distribution of the scattering block for Variant II. The variable density of the scatterer can be realised, for example, by an arrangement of the consecutive layers of polyethylene foils of given thickness.

The numerical calculations of the neutron fluxes, photons and radiation doses have been done to optimise Variants I and II and then to compare the results to Variant R. First of all the calculations for Variant R have been carried out with two Monte Carlo codes: MCNP [1] and FLUKA [2]. Obtained results are almost the same and validated the foregoing programs and libraries of nuclear cross sections [8]. All results shown below in the paper have been obtained using only the MCNP code.

Fission converter

Variant I with "the bent secondary beam" is presented in Fig. 2. The fission converter is placed downstream the reactor duct outlet while the filter/moderator system is situated obliquely out of reach of the primary beam. The rows of fuel plates are arranged irregularly – the further from the reactor duct outlet the more dense, according to the rule described in the previous chapter. Such an arrangement has been used to ensure homogeneity of fast neutrons, which come at the inlet surface of the filter/moderator system (the surface A–B in Fig. 2). The advantage of the foregoing Variant should be an almost complete use of neutrons from the original beam in the conversion process and minimisation of the photon dose from the reactor core. The disadvantage is a small solid angle in which fast neutrons originated in the fission converter are utilised in the filter/moderator system.

Variant II with "the deviated secondary beam" and the scattering block is presented in Fig. 3. The neutron beam from the reactor core hits the suitably arranged scattering block and next gets into the fission converter located outside of the original beam. The density of the scattering material has been adjusted in a similar way as for Variant I in order to achieve homogeneity of fission neutron source. The filter/moderator system is placed aslant to the duct axis sticks directly to the fission converter. The advantage of this set-up is almost a full removal of photons emerging

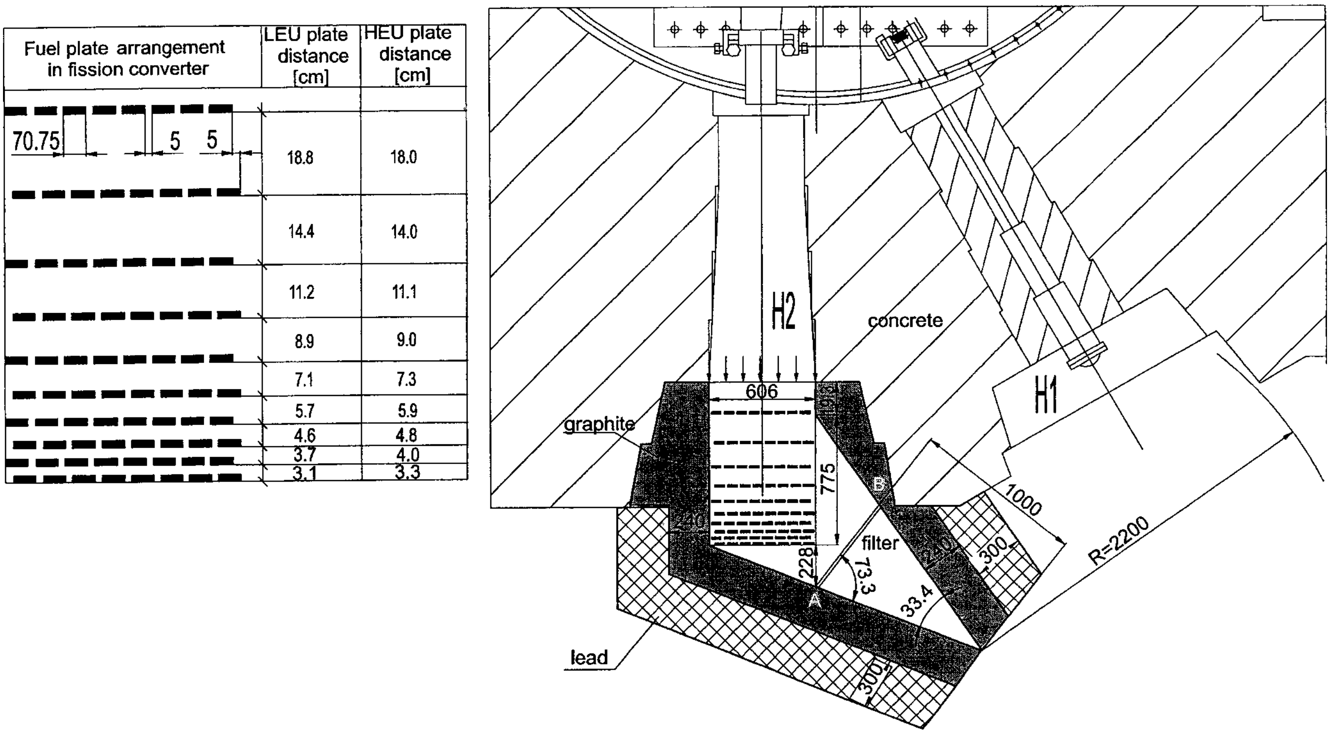

Fig. 2. Fission converter location (Variant I) with reference to the horizontal duct H2 at MARIA reactor along with the arrangement of fuel plates.

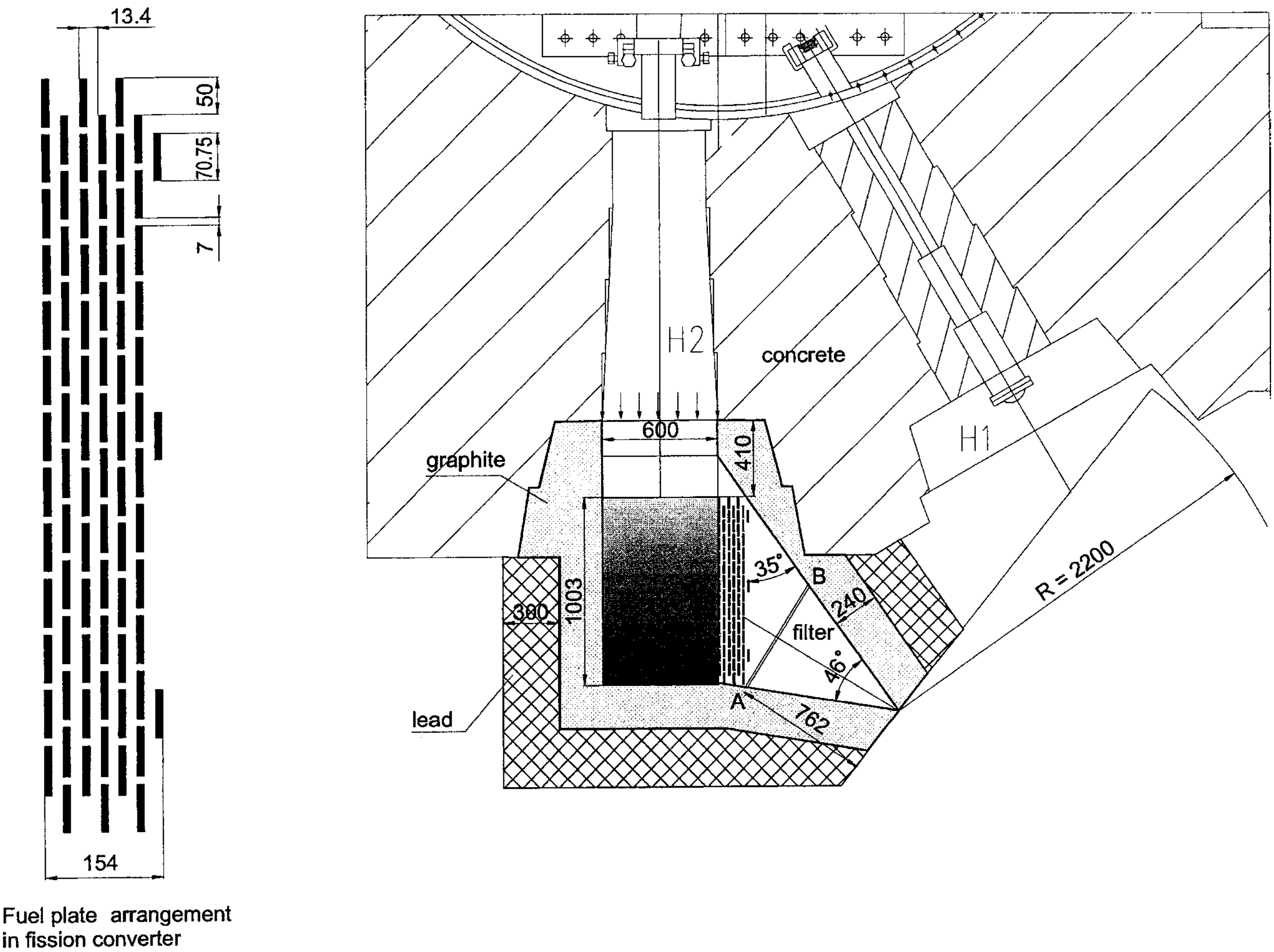

Fig. 3. Fission converter location (Variant II) with reference to the horizontal duct H2 at MARIA reactor along with the arrangement of fuel plates.

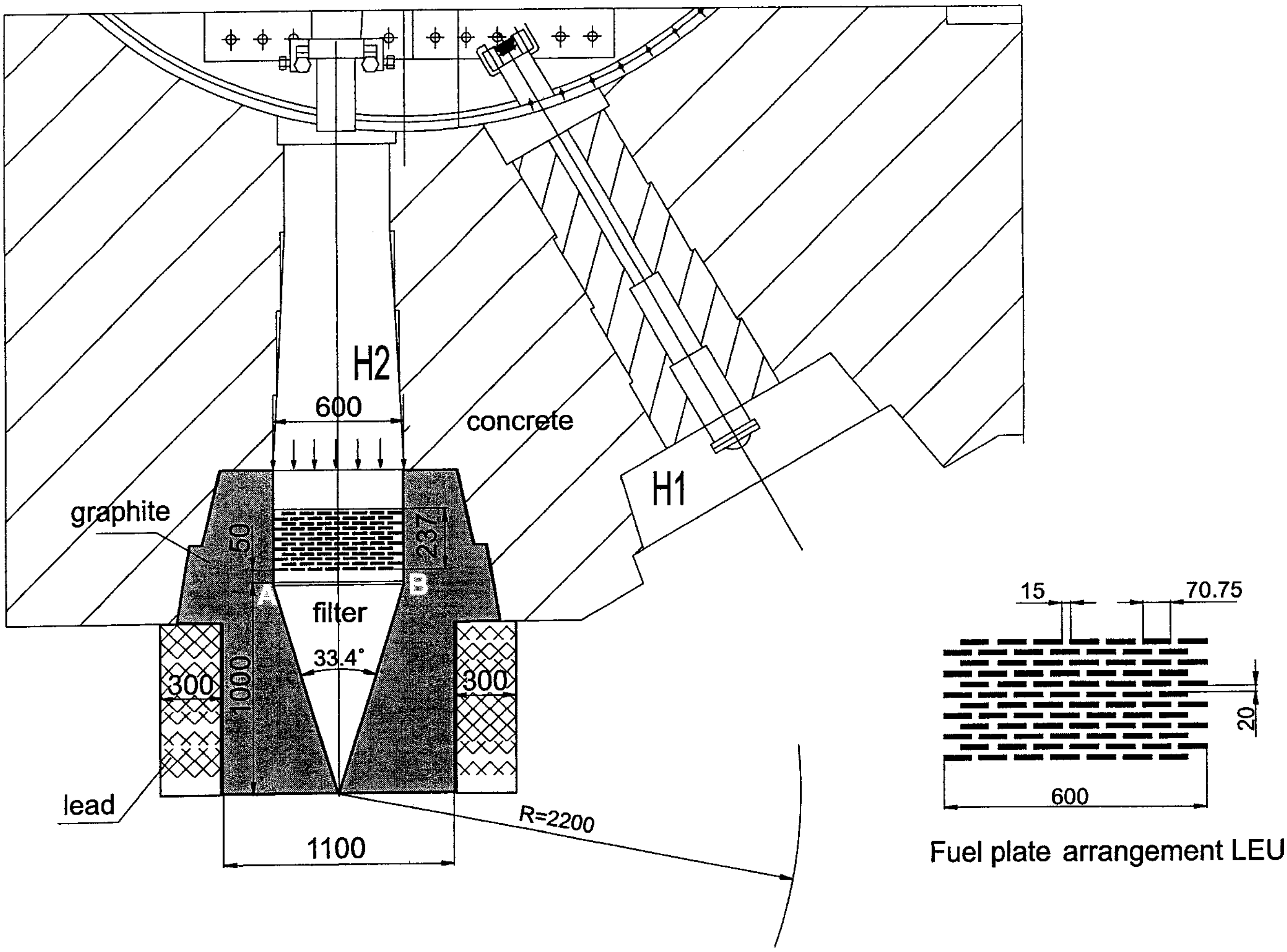

Fig. 4. Fission converter location (Variant R) with reference to the horizontal duct H2 at MARIA reactor along with the arrangement of fuel plates.

from the reactor duct. Also production of heat in the fission converter is the lowest. The disadvantage is an incomplete use of the primary neutron beam in the converter.

Various moderators can be used as the scattering block. Polyethylene, plexiglas and graphite of variable density have been examined. Graphite turned out to be the best material [8] and the examples shown in the paper are calculated only for that scatterer.

Variant R, where the fission converter is perpendicular to the original beam, is shown in Fig. 4. The filter/moderator system as well as the fission converter is situated along the primary neutron beam axis. The advantage of this arrangement is the most effective usage both thermal neutrons from the reactor core and fission neutrons. The disadvantage is a significant intensity of photons emerging directly from the reactor duct.

Two kinds of fuel plates (HEU and LEU) are available for sale. The single HEU plate contains 10.05 g of ^{235}U while the single LEU plate 12.34 g of ^{235}U. The converter efficiency has been examined for both types of plates on the example of Variant R. Calculations showed that there is practically no difference in efficiency of the fission neutron production for both types of the fuel plates [8]. The LEU plates have been selected for further considerations.

As the criterion of the converter efficiency a parameter called factor φ^{60} has been used, defined as the number of fast neutrons (of energy above 10 keV) per one source neutron crossing a circle of 30 cm in radius situated at the inlet of the filter/moderator (A–B surface in Figs. 2–4).

In each variant of the fission, the lateral graphite reflector surrounds converter. The thickness of the reflector should be optimised from the point of view of maximum value of factor φ^{60}. The optimal thickness of the graphite reflector appeared to be different for each variant. The optimal thickness have been calculated for each case and the following sizes have been selected: 36 cm, 60 cm and 24 cm in Variant I, II and R, respectively. Further enlargement of the reflector thickness does not provide a meaningful neutron current increase. Comparisons between variants have been done for the given optimal thickness of the graphite reflector. Additionally, the graphite reflector of the same dimensions in each case (Fig. 5), downstream the fission converter has been modelled in order to provide the same albedo for all variants.

In Variant R, photons originated in the reactor core cause a larger photon current at the A–B surface in comparison with Variant I and II. Therefore, in Variant R the bismuth layer (photon absorber) has been placed between the reactor source and the fission converter. The thickness of the foregoing layer should be chosen to obtain the photon current from the reactor core $\varphi_\gamma(\gamma)$ in Variant R comparable with the analogous photon current in

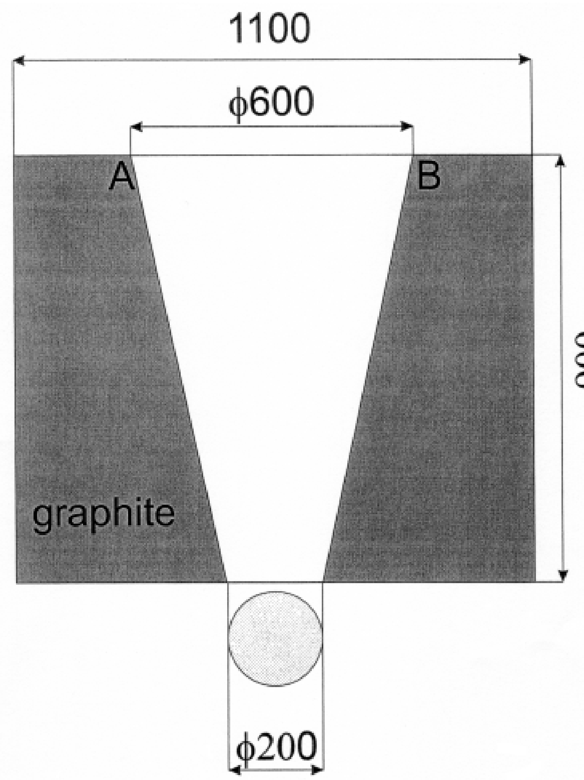

Fig. 5. A graphite shield of the filter.

Variant I and II (without bismuth). The results are presented in Fig. 6. The photon current $\varphi_\gamma(\gamma)$ for Variant R has been calculated as a function of the thickness of the bismuth layer. The same current has been also calculated for Variants I and II. It emerges that the photon current from the reactor core in Variant R is comparable with the photon currents in Variants I and II when the bismuth thickness is about 6 cm. The factor φ^{60} has been also calculated for Variant R as a function of bismuth thickness (the dashed line in Fig. 6). Only a small decrease of the neutron efficiency caused by this material is observed.

These modelled variants can be compared with each other. The φ^{60} factor, the total photon current φ_γ(tot) (the sum of the photon current from reactor core $\varphi_\gamma(\gamma)$ and the photon current originated from neutron interactions φ_γ(n)) have been calculated and collected in Table 1. With regard to epithermal neutrons, Variant R is the most efficient even with the bismuth layer added. On the other hand, the photon current in this Variant is largest but the difference is not outstanding.

Table 1. Comparison of neutron and photon efficiencies on the A–B surface for various fission converter variants.

	φ^{60}	φ_γ(n)	$\varphi_\gamma(\gamma)$	φ_γ(tot)
Variant I (lateral shield 36 cm)	0.229	0.402	0.079	0.480
Variant II (lateral shield 60 cm)	0.217	0.387	0.038	0.425
Variant R (lateral shield 24 cm, bismuth 6 cm)	0.301	0.498	0.059	0.557

The neutron flux homogeneity has been investigated for Variants I and II. The surface A–B of emerging neutrons from the converter has been divided in six rectangles 10 × 60 cm and numbered from 1 to 6, where the surface 1 is taken at point B (cf. Fig. 2). The relative values of the φ^{60} factors have been calculated for each rectangle. The results are shown in Fig. 7 for Variants I and II (with the graphite scatterer). About 15% distortion of the space distribution is observed. It seems that the boundary conditions for the density distribution (Eq. (3)) are not realistic. The density at the point $x = 0$ has to be higher than zero and should be optimised. Further modifications of the density distribution can be done numerically.

Thus, Variant I is promising for the next step of optimisation of the entire source set-up even though the neutron efficiency given by the factor φ^{60} seems to be too low. The main reason of this result is a small total amount of the ^{235}U – here only about 1000 g. Variant IA has been recalculated replacing the 80 LEU fuel plates with 780 EK-10 uranium rods (6275 g of ^{235}U). Variant IA-1 (Fig. 8) achieves the φ^{60} factor equal to 0.307 which is about 34% higher than in previous Variant I. In comparison to the six-time grow of uranium mass it seems discontented. It is

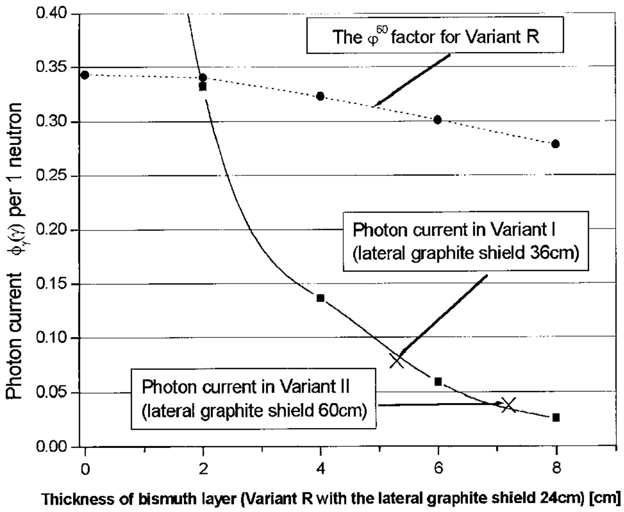

Fig. 6. Photon current in a function of the thickness of bismuth layer in Variant R. Variants I and II are comparable to Variant R with 6 cm of bismuth. Dashed line: φ^{60} factor for Variant R.

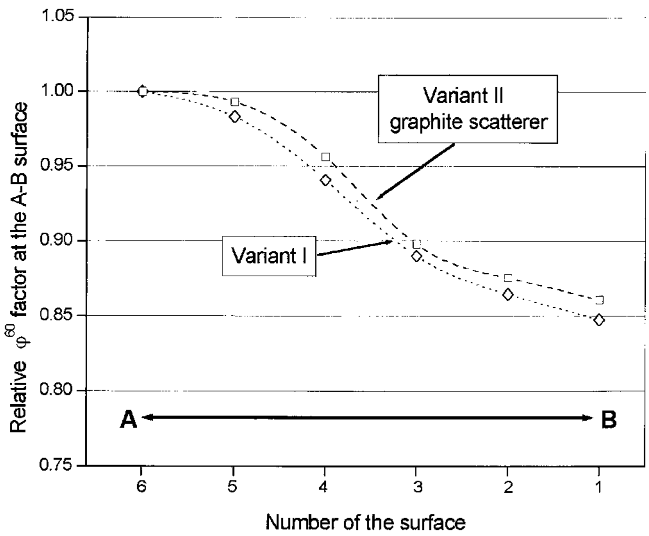

Fig. 7. Distribution of the relative φ^{60} factor along the A–B surface for Variant I and II.

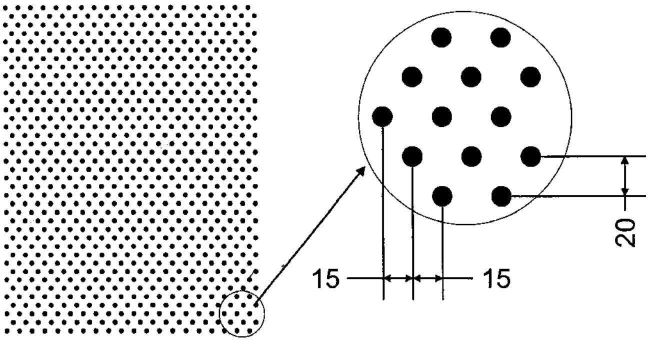

Fig. 8. Arrangement of 780 fuel rods EK-10 for Variant IA-1. Each rod contains 8.045 g of ^{235}U the rods being arranged in 39 rows with 20 rods in every row, which correspond to 6275 g of ^{235}U.

supposed that the geometrical arrangement of rods can be important for the neutron efficiency. The arrangement of the rods in the calculated Variant IA-1 forms empty channels, which cause loses in the neutron production. Another arrangement of 780 rods has been examined to check this assumption (Variant IA-2, Fig. 9). The φ^{60} factor grows to the value 0.332 i.e. about 8% more in comparison to Variant IA-1.

A further optimisation of the neutron efficiency of the converter could be done by the changes of the geometrical arrangement or by increasing of the ^{235}U mass. It has been checked [7] that the latter is unfavourable because the effects concerning the self-absorption are significant and it is impossible to enlarge the fission converter efficiency just by increasing the mass of the fissionable material.

The spatial distribution of the fast neutrons at the A–B surface for Variant IA-2 is more heterogeneous than analogous distribution for Variant I, what was expected. It is shown later in the paper that the filter/moderator system which has been designed for the converter IA-2 smoothes the spatial neutron distribution. The distribution of the relative fast and thermal neutron fluxes at the inlet A–B and the outlet C–D of the filter/moderator system is shown in Fig. 10. The calculations have been done in the same way as for Variant I (Fig. 7).

Because the epithermal neutrons do not involve ^{235}U fission they could be thermalized in a layer of water surrounding a fuel rod. On the other hand, the thermal neutrons can be absorbed in this layer. In order to check which of the foregoing effects predominates, the arrange-

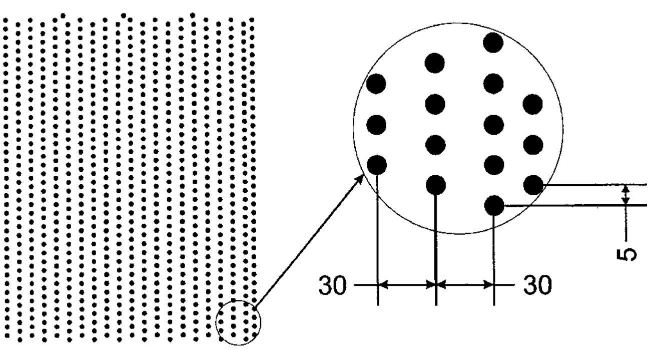

Fig. 9. Arrangement of 780 fuel rods EK-10 for Variant IA-2. The rods are placed in 37 rows – 10 and 11 rods alternately, the last row contains only three uranium rods.

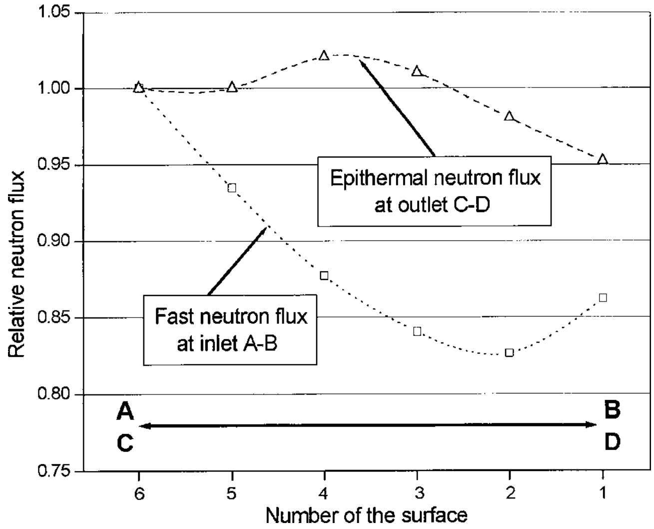

Fig. 10. Distribution of the relative fast and thermal neutron fluxes at the inlet A–B and the outlet C–D of the filter/moderator system. Variant IA-2 of the converter, Configuration 4j of the neutron filter/moderator.

ments with 1 mm and 2 mm layers of H_2O and D_2O enclosing EK-10 rods have been examined. In each case lower values of the φ^{60} factor have been obtained. The example of calculations for variant IA-1 is presented in Table 2. The results show that disadvantages associated with the absorption of the thermal neutrons exceed a benefit from thermalization of the epithermal neutrons.

The concentration of 780 rods can cause problem with the heat piping. The influence of a liquid coolant on the φ^{60} factor in the foregoing configuration has been examined [7]. Light or heavy water has been placed between the rods or in the gap between rows. The obtained results show (Table 3) that from the neutron current optimisation point of view the heat from the fission converter should be piped away using air since both heavy and light water significantly decrease the number of neutrons at the filter/moderator inlet.

Table 2. The φ^{60} factor in Variant IA-1 (rods enclosed with moderator).

Configuration	φ^{60}
IA-1 (no moderator)	0.307
IA-1 1 mm light water	0.206
IA-1 1 mm heavy water	0.265
IA-1 2 mm light water	0.162
IA-1 2 mm heavy water	0.211

Table 3. The φ^{60} factor for various liquid coolant arrangements in Variant IA-1.

Configuration	φ^{60}
IA-1 without moderator	0.307
IA-1 light water "in rows"	0.146
IA-1 light water "between rows"	0.060
IA-1 heavy water "in rows"	0.135
IA-1 heavy water "between rows"	0.131

The total mass of the fission material used in the proposed converter is not hazardous from the point of view of criticality. Criticality calculations of Variant IA-2 configuration show that the k_{eff} is 0.41. When the fission converter is filled with light water, k_{eff} increases to 0.94.

Epithermal neutron source based on Variant IA-2 of the fission converter

Neutron filter/moderator

The preliminary optimisation of the filter/moderator system has been modelled and basic output data of the epithermal neutron flux and radiation doses have been calculated for the fission converter variant with the "bent derivative beam". Variant IA-2 has been selected for the examination because the application of EK-10 rods for the converter construction seems to be more reliable when the BNCT neutron source at the MARIA reactor is built. Aside from typical calculations which have to be done for such a kind of facility, the calculations of the spatial distribution of the epithermal neutron distribution at the outlet of the system have been done.

The neutron filter/moderator system should consist of materials of high scattering cross section in the high-energy range (fast neutrons) and a low scattering cross section for epithermal neutrons. The useful neutron energy range for BNCT purposes is from 1 eV to 10 keV (sometimes 40 keV is assumed). Hence, in the present paper neutrons of energy above 10 keV are called fast neutrons and neutrons of energies between 1 eV and 10 keV are called epithermal neutrons. A selection of materials in the neutron filter/moderator modelling is based on the experience of the Massachusetts Institute of Technology (USA) [4]. Therefore, the use of aluminium, Al_2O_3, AlF_3, graphite, Teflon (polytetrafluoroethylene; -[-CF_2-CF_2-]n-), titanium and fluental [3] – which consists of 29% Al, 70% AlF_3 and 1% LiF – is considered.

The epithermal neutron flux at the patient's head position is the major assessment criterion of the fission converter but both photon and fast neutron doses must be also taken into account. Photons and fast neutrons are considered as a beam contamination since they do not differentiate between healthy and cancer tissues. As a result of, inherent for the BNCT method, epithermal neutron interaction both with hydrogen and nitrogen ($^1H(n,\gamma)^2H$ and $^{14}N(n,p)^{14}C$ reactions), irreducible dose arises in brain tissue, roughly estimated to be about 2×10^{-10} cGy·cm^2/n [4]. Hence, it is assumed that specific dose both for the impinging photons (D_γ/ϕ_{epi}) and the fast neutrons (D_{fn}/ϕ_{epi}) should be lower than 2×10^{-11} cGy·cm^2/n – one order of magnitude below the harmful dose being inherent for the BNCT therapy.

In order to remove the thermal neutrons from the beam, a cadmium layer of 0.04 cm is placed downstream the neutron filter/moderator. An 8 cm thick bismuth layer follows the cadmium layer and serves as a photon filter. Later, the material of the photon filter (bismuth or lead) and its optimal thickness are selected. Examples of calculations are presented later in the paper (Table 6). The beam is collimated by a graphite cone of 20 cm length. In order to increase the neutron flux the filter/moderator is enclosed by a lead reflector of 10 cm thickness (Fig. 11).

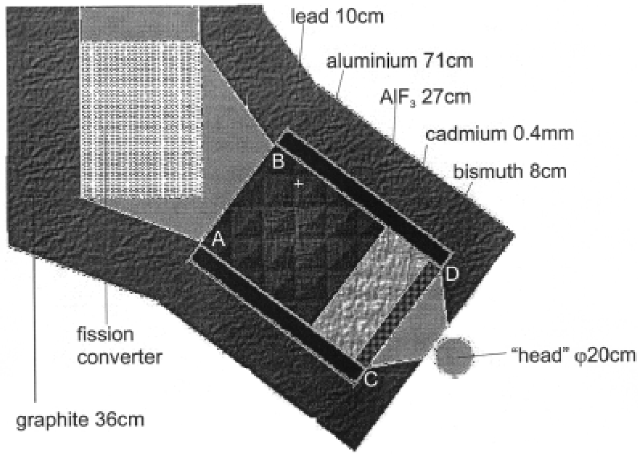

Fig. 11. Calculating model of the fission converter (Variant 1A-2) and filter/moderator arrangement (Table 4, Configuration 5).

The ultimate measure of the beam performance is the neutron flux and the specific doses in a phantom of a head. However, in order to simplify computer simulations at the stage of the filter optimisation, both the fluxes and the specific doses are estimated "in-air". Such procedure is used by other authors [4] to speed up the Monte Carlo calculations. In the present study the patient's head is simulated by a ball of 10 cm radius, while its centre is 12 cm downstream the collimator outlet plane. Results of computations for different combinations of materials and their thickness are presented in Table 4. The optimal filter/moderator system should fulfil the maximum value for the epithermal neutron flux ϕ_{epi} with the minimum values of relative fast neutron and gamma doses D_{fn}/ϕ_{epi} and D_γ/ϕ_{epi}.

The fast neutron specific dose in all cases is at least one order of magnitude lower than the required value 2×10^{-11} cGy·cm^2/n. The photon specific dose, except Configurations 2 and 10 (Table 4), does not exceed the recommended value and comes almost solely from photons produced in neutron interactions. The specific dose concerning photons emerging from the reactor duct in most cases is $\sim 10^{-16}$ cGy·cm^2/n, i.e. it is five orders of magnitude smaller than in the case when only neutrons get out of the reactor core. The foregoing value varies from 5.6×10^{-17} cGy·cm^2/n in Configuration 9 to 8.3×10^{-16} cGy·cm^2/n in Configuration 10.

The largest epithermal neutron fluxes are obtained in Configurations 4, 6 and 8. It is essential that the beam is pure enough according to the criteria assumed in the paper.

Since the maximal epithermal neutron flux has been obtained for the filter/moderator constructed with 71 cm of aluminium and 17 cm of AlF_3, some configurations of another aluminium and AlF_3 layers thickness have been investigated. Results are collected in Table 5.

The largest epithermal neutron flux has been obtained in Configuration 4i and 4h (Table 5) but in both cases the fast neutron specific dose exceeds the acceptable value. Therefore, Configuration 4j (21 cm Al and 17 cm AlF_3) seems the most favourable since in this configuration the beam is sufficiently free of contamination. This filter/moderator is selected to further calculations.

Table 4. Epithermal neutron flux and specific doses "in-air" at the outlet of the different filter/moderator arrangements in Variant IA-2 of the converter.

No.	Filter/moderator	ϕ_{epi} (10^{-6} n/cm^2) per source neutron	$D\gamma/\phi_{epi}$ (10^{-11} cGy·cm^2/n)	D_{fn}/ϕ_{epi} (10^{-11} cGy·cm^2/n)
1	66 cm fluental	1.58	1.24	0.17
2	68 cm AlF_3 – 2 cm Ti	0.98	3.96	0.15
3	68 cm fluental – 2 cm Ti	1.18	1.16	0.15
4	71 cm Al – 17 cm AlF_3	2.08	1.23	0.10
5	71 cm Al – 17 cm AlF_3	1.15	1.24	0.05
6	80 cm Al – 11 cm Teflon	1.95	1.25	0.14
7	80 cm Al – 17 cm Teflon	1.25	1.41	0.07
8	83 cm Al – 11 cm Al_2O_3	1.81	1.24	0.13
9	83 cm Al – 17 cm Al_2O_3	1.19	1.23	0.07
10	96 cm Al – 12 cm graphite	0.52	2.28	0.05

Table 5. Epithermal neutron flux and specific doses "in-air" for different modifications of Configuration 4 from Table 4.

No.	Filter/moderator	ϕ_{epi} (10^{-6} n/cm^2) per source neutron	$D\gamma/\phi_{epi}$ (10^{-11} cGy·cm^2/n)	D_{fn}/ϕ_{epi} (10^{-11} cGy·cm^2/n)
4	71 cm Al – 17 cm AlF_3	2.08	1.23	0.10
4a	71 cm Al – 12 cm AlF_3	2.68	1.18	0.18
4b	71 cm Al – 7 cm AlF_3	3.36	1.37	0.32
4c	61 cm Al – 17 cm AlF_3	2.76	1.60	0.18
4d	51 cm Al – 17 cm AlF_3	3.69	1.19	0.29
4e	51 cm Al – 7 cm AlF_3	5.63	1.45	0.92
4f	41 cm Al – 17 cm AlF_3	4.86	1.32	0.54
4g	41 cm Al – 7 cm AlF_3	7.19	1.43	1.52
4h	31 cm Al – 17 cm AlF_3	6.44	1.55	0.90
4i	31 cm Al – 7 cm AlF_3	9.17	1.82	2.66
4j	21 cm Al – 17 cm AlF_3	8.50	1.86	1.64
4h	21 cm Al – 7 cm AlF_3	11.60	2.05	4.60

Photon filter

A bismuth filter of 8 cm thickness has been used in the optimisation of the neutron filter/moderator to reduce the specific photon dose. Since both the filter material and its thickness have been chosen in an arbitrary way, it is necessary to check whether these are optimal parameters. A material used for the photon filter should have a large mass number and simultaneously the epithermal neutron flux must not be reduced significantly. Lead and bismuth are known to meet such requirements. Computer calculations for Bi and Pb layers of thickness from 0 to 14 cm with 2 cm increment have been carried out. Table 6 presents the beam characteristics "in-air".

Table 6. Epithermal neutron flux and specific doses "in-air" calculated for the optimisation of the photon filter.

Thickness (cm)	ϕ_{epi} (10^{-6} n/cm^2) per source neutron		$D\gamma/\phi_{epi}$ (10^{-11} cGy·cm^2/n)		D_{fn}/ϕ_{epi} (10^{-11} cGy·cm^2/n)	
	Bi	Pb	Bi	Pb	Bi	Pb
0	11.93	11.93	16.91	16.91	2.09	2.09
2	10.97	10.97	8.35	8.20	1.88	1.91
4	10.09	9.77	4.20	3.74	1.84	1.80
6	9.29	8.87	2.60	2.31	1.73	1.62
8	8.50	8.09	1.80	1.80	1.68	1.55
10	7.80	7.39	1.17	1.45	1.62	1.48
12	7.22	6.74	1.02	1.24	1.48	1.42
14	6.61	6.09	0.88	1.20	1.46	1.30

It emerges that the most primary assumption (8 cm bismuth filter) is the best choice, since the epithermal neutron flux reaches the maximum, while the beam contamination do not exceed the permitted values both for fast neutrons and photons. Because lead is easy available it may be considered to be used, though the epithermal neutron flux decreases of about 5%.

Since calculations of the specific photon dose associated with photons that origin in the reactor core are time-consuming and this dose almost does not affect the total specific photon dose, the optimisation of the photon filter is carried out only for the neutron source. However, in order to be sure, the specific photon dose due to photons from the reactor core has been calculated for the 8 cm bismuth filter. The obtained value is 5.5×10^{-16} cGy·cm^2/n – thus, its contribution is slight.

Neutron flux homogeneity

For Configuration 4j considered as the optimal one (21 cm Al – 17 cm AlF_3 – 8 cm Bi) the homogeneity of the neutron flux has been investigated both at the filter/moderator inlet (A–B surface in Fig. 10) and at the outlet of the photon filter (C–D surface in Fig. 10). The calculations have been done in the same manner as for Variant I and II. At the inlet of the filter/moderator fast neutrons have been taken into considerations and epithermal neutrons in case of the photon filter outlet. The relative neutron fluxes are shown in Fig. 10.

Although the fast neutron flux at the filter/moderator inlet is noticeably inhomogeneous, the flux of epithermal neutrons emerging from the photon filter is roughly homogeneous as a result of several scatterings while slowing down. Thus, the "bent" arrangement of the filter/moderator set does not make any harm with regard to the therapeutic beam homogeneity.

Conclusions

The epithermal neutron source at the research reactor MARIA at Świerk, based on the fission converter concept has been a subject of neutronic and gamma numerical studies. The unique design of the fission converter has been proposed for the specific geometrical conditions of surrounding of the reactor. Typical designs of epithermal neutron sources which are applied in the world-known medical irradiation facilities are not useful for that non-standard problem. Several different converter arrangements with various fuel types, converter – incident beam geometries, moderating/scattering materials and compositions, reflector thickness, filter/moderator arrangements etc. have been considered. The possibility to obtain the epithermal neutron beam of the required efficiency, spatial homogeneity and low contamination have been the main questions which have to be solved. The numerical calculations revealed that it is possible to realise the epithermal neutron source with acceptable parameters in such a difficult non-symmetrical geometry of the reactor surrounding. The exceptional advantage is the elimination of photons (that origin in the reactor core) in comparison to typical solutions of fission converters. This means that the gamma radiation dose is significantly diminished.

At the current status of converter optimisation process the available "in-air" beam intensity reaches $\varphi_{epi} = 0.3 \times 10^9$ n/(cm^2·s), assuming incident thermal neutron flux density of 10^{10} n/(cm^2·s). Both fast neutron and photon contaminations are kept below assumed limits i.e. 2×10^{-11} cGy·cm^2/n. The experience gained during optimisation performed up to now indicates, that a further increase of the epithermal beam intensity is still achievable and the possibility to build the BNCT facility at the MARIA reactor is proved.

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