



DEVELOPMENT OF IRRADIATION FACILITY FOR BNCT TREATMENT OF TUMORS AT THE IJS TRIGA RESEARCH REACTOR

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Boron Neutron Capture Therapy (BNCT) is a potentially powerful tool, offering the possibility of various forms of cancer treatment, and is believed to be one of the most promising methods of cancer radio therapy in the future. The method is particularly suitable for tumors which resist more conventional methods such as surgery, X-rays and chemotherapy.

The development of the efficient irradiation facility at IJS TRIGA research reactor, using Monte Carlo simulation code MCNP4A is presented in this contribution.

A simple spherical reactor, filled with homogenized fuel mixture proved to be quite useful for the purpose of optimizing the filters of the irradiation facility. But for the determination of the complete dose rate at the irradiation point, the activation of particular parts and materials of the facility, fast neutron leakage rate and evaluation of the influence of the collimator, we developed the complete Monte Carlo model with all the details about the reactor core, graphite reflector and particularly the irradiation channel.

Boron Neutron Capture Therapy (BNCT) je ena izmed najbolj obetavnih metod zdravljenja rakastih obolenj. Največjo prednost ima pri zdravljenju tumorjev, pri katerih je uporaba konvencionalnih metod zdravljenja, kot so kirurški posegi, uporaba rentgenskih žarkov in kemoterapija, manj uspešna.

V prispevku bo predstavljen razvoj obsevalne naprave (obsevalnega kanala) na raziskovalnem reaktorju TRIGA na IJS, z uporabo Monte Carlo simulacijskega programa MCNP4A.

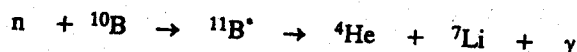
Enostavni krogelni reaktor, napolnjen s homogeno mešanico goriva se je izkazal za zelo koristnega pri optimiziranju sestave in dolžine filtrov v obsevalnem kanalu. Vendar smo za določitev dozne porazdelitve na obsevalnem mestu, aktivacije posameznih materialov, deleža pobega hitrih nevtronov ter vpliva kolimatorja razvili popolni Monte Carlo model, ki vključuje večino podrobnosti sredice, grafitnega reflektorja ter obsevalnega kanala.

1. Introduction

Boron Neutron Capture Therapy (BNCT) is radiation treatment that offers the potential of a highly selective effect of radiation - alpha particles - while sparing normal tissues. It brings together two components that when kept separate have only minor effects on normal cells. The first component is a stable isotope of boron that can be concentrated in tumor cells. The second is a beam of low-energy neutrons that produces short range radiation when absorbed or captured by boron nuclei. The combination of these two conditions at the site of the tumor releases intense radiation that destroys malignant tissues.

The central feature of effective BNCT is the selective delivery of a drug containing the stable, naturally occurring isotope boron-10 into the tumor. As the tumor is irradiated with the

low energy neutrons (epithermal neutrons become thermalized), there is a higher likelihood of the boron-10 nucleus absorbing them than that of the nucleus of the any other element normally present in tissues. The boron nucleus becomes unstable and immediately splits into two recoiling ionizing particles, an alpha particle (i.e. helium nucleus) and a lithium nucleus:



These products of the BNCT reaction ${}^{10}\text{B}(n, \alpha){}^7\text{Li}$ are of short range (alpha particles have a pathlength of about one cell diameter, which is around 10 μm) and are confined to the immediate vicinity of the boron containing compound which, hopefully, should be concentrated in the tumor.

A major appeal of BNCT is that lithium-7 nuclei and energetic alpha particles are produced by fission reaction following the neutron capture. These heavy particles which carry 2.79 MeV of energy, have very high ionization density. Another advantage of alpha particles is that they can affect dividing and nondividing tumor cells alike - tumors are known to have a large number of viable but inactive cells. Other forms of radiation treatment and chemotherapy tend to work best only on the cells that are dividing. These characteristics make BNCT a theoretically ideal system for the destruction of malignant cells.

2. Epithermal channel design

The efficient irradiation facility for boron neutron capture therapy (BNCT) consists of following main parts:

- appropriate (epi)-thermal filter which enhances thermal and epithermal neutron flux on one side and suppresses the fast neutrons on the other side,
- gamma filter which minimizes the amount of gamma radiation at the irradiation point (mostly hard gamma radiation with energies above 10 MeV is particularly considered),
- collimator which is used optional mostly for beam shaping when small irradiation areas are desired (brain tumors).

There are four operating epithermal neutron beams in the world at this moment: at the Brookhaven Medical Research Reactor of the BNL, USA, the Massachusetts Institute of Technology Reactor, USA, Musashi Institute of Technology, Japan and the reactor at Petten, Netherlands.

It has been reported that a satisfactory thermal/epithermal neutron beam could be designed at a TRIGA research reactors. These reactors are generally considered as quite safe, also for installing and operating in populated areas. Therefore decision has been made that similar epithermal beam could be developed at IJS TRIGA Mark-II research reactor, also. The Slovenian BNCT project was approved in January 1994, and is financed by the Ministry of the Science and Technology.

2.1. Model of spherical reactor with homogenized core

The first step of the development was the study of appropriate configuration of thermal/epithermal neutron and gamma filter using Monte Carlo simulation code MCNP4A. As emerging point for calculation we decided to use a simple spherical reactor, filled with homogenized mixture of materials contained in TRIGA fuel unit cell. Dimensions of the sphere were picked out in such manner that the bare sphere was approximately critical (k_{eff} was 1.011). Particular filter configurations were simply arranged as concentric spheres. We did not include the graphite reflector around the fuel sphere, because in the filter calculations we need the "proper reactor spectrum" and not the spectrum additionally moderated in the graphite reflector blanket. The radial channel, where the irradiation facility is going to be installed,

leads right through the graphite, so the assumption mentioned above is quite acceptable in the first approximation.

2.1.1. Selecting the best spectrum shifter material

It is obvious that this model did not have much in common with the real TRIGA reactor from the geometrical point of view, but rather for defining neutron source for the filter configuration optimization. It was quite suitable, because it provided very good statistics and reasonable computing times (estimated tally relative errors were approximately less than 1%, and computer times used varied around 350 minutes).

Because of simplicity of proposed configuration we were allowed to use criticality calculation which provide the most reliable reactor spectra calculation. Computing procedure for criticality calculation integrated in the MCNP4A (KCODE) is sophisticated and developed to that level, that we do not have to concern with complicated variance reduction technique and still obtain statistically acceptable results.

We performed an extensive number of parametric studies, and calculated the neutron and gamma spectra for different combinations of neutron epithermal and gamma filters. We altered the length of the scatterer and particular filter elements, and the density of the used alumina. With reference to the density of Al_2O_3 , which is approximately 1.5 g/cm^3 it has to be emphasized that this value could be raised up to the value of 3.9 g/cm^3 , if alumina sand is sintered. This density was found to be almost ideal for epithermal filter (shifter of neutrons from fast to epithermal energies - Fig. 1).

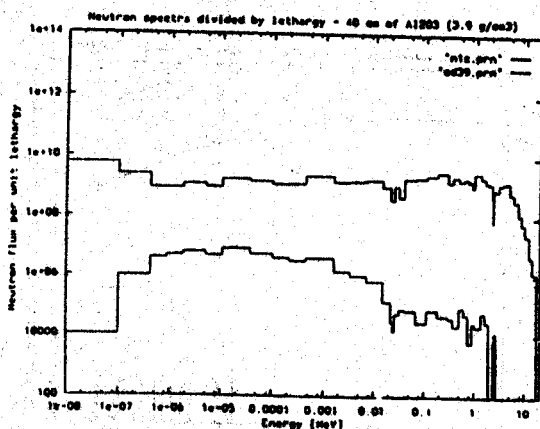


Fig. 1 Neutron spectra divided by lethargy: used 40 cm of alumina with density 3.9 g/cm^3 for epithermal filter

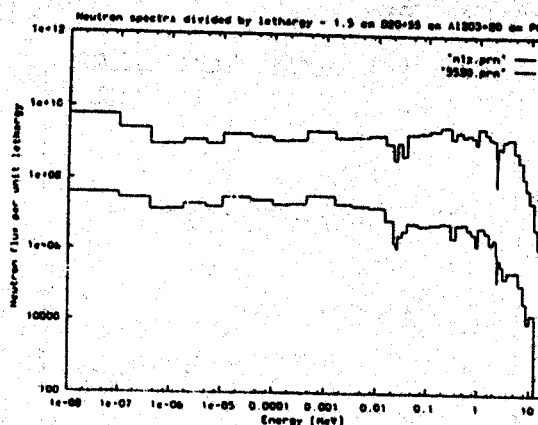


Fig. 2 Neutron spectra divided by lethargy: used combination of 1.5 cm of D_2O , 55 cm of alumina and 20 cm of Pb

But unfortunately, the procedure of sintering is quite expensive (special tools would have to be machined), so we preferably used an alternative of keeping the density of Al_2O_3 around 2.0 g/cm^3 (with pressing it into compact cylindrical tablet), and increasing the length of the spectrum shifter.

Considering technical and financial feasibility we decided to adopt two different combinations of epithermal column, described below (values in parenthesis present intervals within which we altered the length of elements):

- a thin layer of D_2O used as the scatterer at the beginning of the irradiation channel (0.5 - 2.0 cm),
- a rather thick layer of alumina (Al_2O_3) destined for the epithermal filter (40 - 60 cm),
- a layer of Bi or Pb used as a gamma filter (10 - 25 cm).

Calculated spectra for combination which showed out as the most appropriate (1.5 cm D₂O, 55 cm Al₂O₃, 20 cm Pb) are presented on Fig. 2.

The second composition was slightly different and was based on a model, developed for the proposed 300 kW slab TRIGA reactor in a Brookhaven National Laboratory [10]:

- a layer of Al (20 - 40 cm),
- a layer of alumina (Al₂O₃) (40 - 90 cm),
- a very thin layer of Cd, used to cut off the thermal neutrons (0.05 cm),
- a layer of Pb (gamma filter) (10 - 20 cm).

It is obvious that the main scope of our modeling work was to find the configuration with the highest level of shifting the neutrons from fast to epithermal energies and the minimum gamma ray portion. On the other hand, the requirement of as low as possible attenuation of neutron flux after traversing the filters, was also quite important.

After analyzing the results we are in a position to confirm, that the most appropriate configuration of the irradiation facility for the BNCT at our reactor would be composed of:

- 1.5 cm of D₂O,
- 40 cm of Al,
- 80 cm of alumina with density 2.0 g/cm³ (or 40 cm with density 3.9 g/cm³, if sintered),
- 0.05 cm of Cd,
- 15 cm of Pb.

Neutron and gamma spectra calculated using above configuration are presented on Fig. 3 and Fig. 4, respectively.

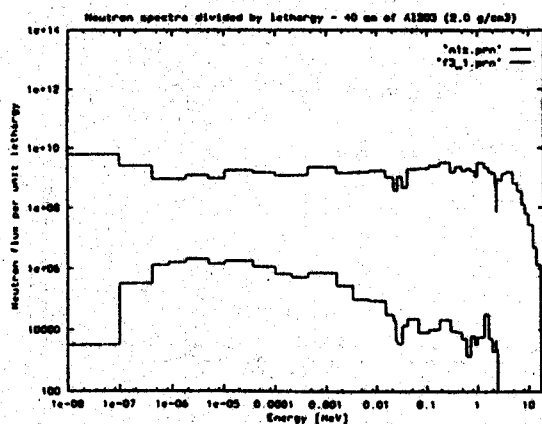


Fig. 3 Neutron spectra divided by lethargy: used 40 cm of alumina with density 2.0 g/cm³ for epithermal filter

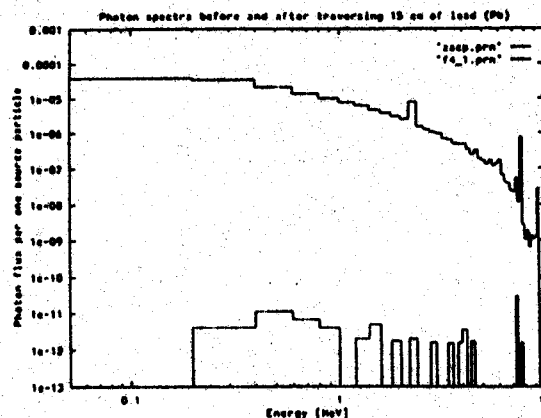


Fig. 4 Photon spectra before and after traversing gamma filter: 15 cm of lead (Pb)

This configuration gives very good results in minimizing the fast neutron flux, and shifting it to the lower, epithermal energies (between 1 eV and 10 keV), what was our intention. It also cuts off successfully the thermal neutrons and minimizes the gamma flux to almost zero. Comparing to "reference spectrum" on Fig. 1, we can conclude that this configuration is rather less successful in minimizing the fast neutron flux, and the attenuation is higher.

Effect of cutting-off thermal neutrons using additional layer of Cd (0.05 cm) is quite evident, when comparing Figures 2 and 3.

2.2. Source intensity normalization:

Spectra (neutron and gamma) that MCNP4A calculates during tallying are obtained in relative units - they are normalized to one source particle (neutron, photon). This is good enough for presenting the energy distribution, but for creating the real picture of flux intensities in particular tally energy bin, we have to multiply the ratio in every bin with correct multiplication factor (constant) which represents the average number of neutrons born in the core during one fission. Considering that the maximum operating power of LJS TRIGA reactor is 250 kW, and that 193 MeV of energy is released during each fission of ^{235}U and that approximately 2.53 new neutrons emerge from it, the average number of neutrons being born in the TRIGA reactor core is $1.874 \cdot 10^{16}$. We simply multiply the neutron flux densities in each energy bin with that constant, using one of MCNP input options. Now all the neutron spectra calculated are presented in appropriate flux units (neutrons/cm²sec).

It was not necessary to repeat the procedure with photon tallies, because we do not need the actual number of photons at the irradiation port, but only the energy distribution which has to be minimized.

2.3. Cylindrical model of the reactor core and dehomogenisation of the core

It is obvious that spherical reactor is only idealization of the real reactor core, which is cylindrical. So, transition of the spherical reactor to the cylindrical model had to be performed. First we modeled the core in shape of cylinder filled with homogenized mixture, exactly the same as in the case of sphere.

The next step was dehomogenisation of the core: it was divided into a five concentric cylinders B, C, D, E and F. Every ring is filled with homogenized mixture, according to the elements (fuel or control rods) positioned in it. Considering previous reports about the neutron spectra at the most outer radius of the core, it can be seen that the ring F is the one which affects the shape of the spectrum the most, so it's mixture has to be calculated very precisely. The comparison between the spectra calculated with homogenized and dehomogenized cylindrical core is presented on Fig. 5.

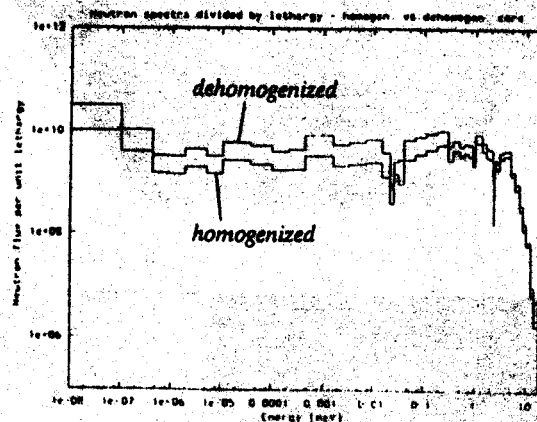


Fig. 5 Neutron spectra divided by lethargy: comparison between used homogenized and dehomogenized model of the reactor core

Surprisingly, the difference between the spectra is more or less negligible and almost in the interval of statistical uncertainty.

In spite of this quite good agreement of the spectra, we preferred to use dehomogenized core model in our further calculations.

3. The complex model of the TRIGA reactor with radial irradiation channel

As already mentioned, the spherical model was quite useful for the purpose of optimizing the filters of the irradiation facility. But, on the other hand, this model is inconvenient for determination of the complete dose rate at the irradiation point, the activation of particular parts and materials of the facility, fast neutron leakage rate, evaluation of the influence of the collimator and so on.

For this purpose we developed the complete Monte Carlo model, where we considered all the most important details about the reactor core, graphite reflector, thermal and thermalizing column and all irradiation channels engaged at TRIGA reactor. We focused a particular attention on the radial channel, which leads through the graphite reflector, right to the F ring of the core, and where the irradiation facility will be inserted. The model shown from the different perspectives and cutting planes is presented on Figures 6a and 6b.

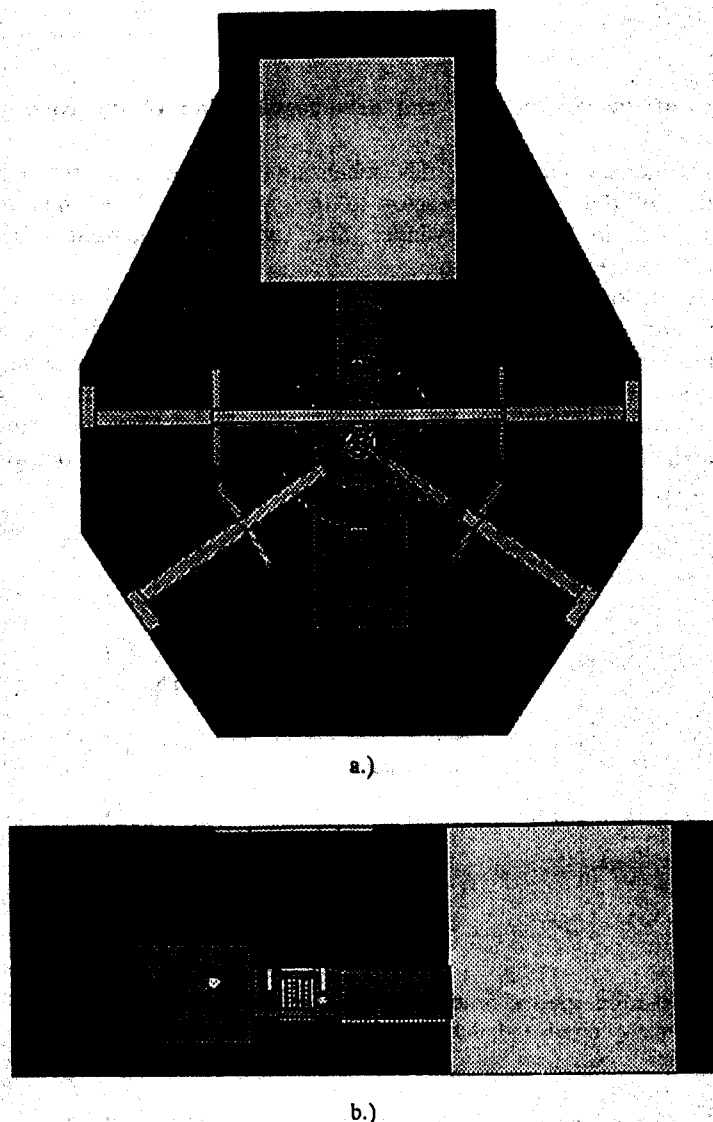


Fig. 6: a detailed model of IJS TRIGA research reactor presented from different perspectives:

- a.) view from the top, with tangential and radial channels,
- b.) view from the side

Considering technical feasibility and particularly safety requirements, we prepared the final shape of the epithermal part of the irradiation facility:

- 30 cm of Al and 40 cm of Al_2O_3 joined together in one piece,
- 30 cm of Al_2O_3 with 0.05 cm of Cd foil,

All the elements stated above are in the shape of cylinder, with the diameter of 136.2 mm. Alumina (Al_2O_3) was compressed to the final density of 2.3 g/cm³. Cylinders are inserted in the Al tubes with inner diameter of 139 mm and thickness of walls 2 mm. The gap between the alumina cylinder and the Al cylinder walls, which is 1.4 mm thin, is filled with Alurud PKB sand with density of 3.97 g/cm³. At front and back side of described element there is 24 mm plate of Al with the appropriate hole for pull-out equipment in the front, and additional Al pin in the back, which fills the hole when element is inserted in the channel, and provides for homogeneous traveling path (without air gaps) to the neutrons. All the elements are equipped with six small wheels made from stainless steel, to enable unrestrained transport through the channel.

The gamma filter which is 15 cm long and made of one piece of Pb, is made in exactly the same way as epithermal filters.

We adopted three small changes in combination proposed in Section 2.1.1:

- we left out 1.5 cm of D_2O .
- the cumulative length of Al_2O_3 is not 80 but 70 cm. We are able to elaborate pressed alumina cylinders with density of 2.3 g/cm³ and there is no need to keep the length 80 cm and we can decrease it on 70 cm. Shorter alumina filter provides lower attenuation of neutron flux.
- the length of Al filter is reduced to 30 cm, because the pins at the front and the back of the filter elements contribute additional 14.4 cm of Al.

Calculated neutron and gamma spectra for described epithermal column incorporated in detailed TRIGA reactor model, are presented on Figures 7 and 8, respectively.

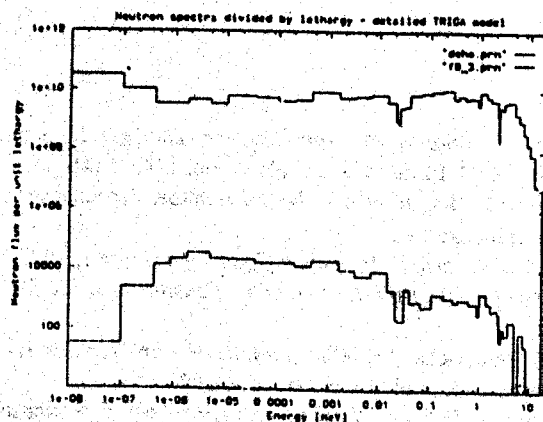


Fig. 7 Neutron spectra divided by lethargy obtained using detailed TRIGA reactor model, with 70 cm of alumina (density: 2.3 g/cm³) for the epithermal filter

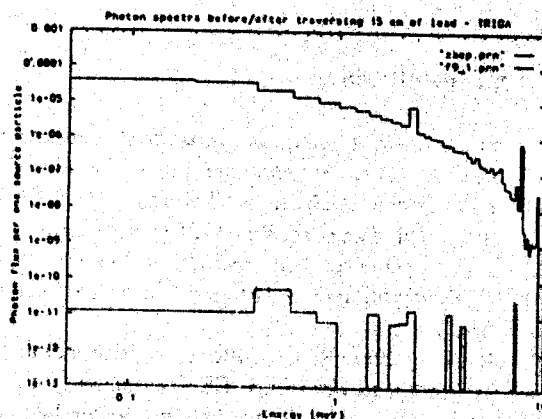


Fig. 8 Photon spectra calculated using detailed TRIGA reactor model, with 15 cm of lead for gamma filter

Analyzing the results we can determine, that this column provides very successful shifting of neutron spectra to the epithermal energies, and decreasing of fast neutrons (estimated tally relative error is 4.6%).

Unfortunately rather high attenuation is observed; almost five orders of magnitude. 15 cm of Pb is also quite efficient in minimizing the gamma flux at the irradiation point.

4. Conclusion

The first theoretical results presented in this contribution show that modeled irradiation epithermal facility at the LJS TRIGA reactor produces good epithermal neutron spectrum, with extensive decrease of fast neutrons, and minimized gamma irradiation. We hope that, when finished, it could present the efficient source of epithermal neutrons for malignant tumors treatment. But, unfortunately, the suitable shape of neutron and gamma spectra is not good enough for successful treatment - high enough level of neutron flux at the irradiation point is also one of the most important facts: higher the flux, shorter are the irradiation times to get the same effect in the tumor (it is also of crucial importance for the patient, because of demand for the lowest possible absorbed dose in the healthy tissue). Japanese experience shows that the "eye dose", "total body dose", and "except-head dose" should be less than 200, 100 and 50 RBE-cGy, respectively. The thermal neutron fluence at the tumor position (approx. 5 cm from the surface) should be over 2.5×10^{12} n/cm² in the "irradiation time" [1]. According to their results, the distance from the core to the irradiation point is a very important factor to design a neutron irradiation field for BNCT. We can get acceptable dose with only 1 hour irradiation by using a 100 kW reactor if we can get the irradiation port at the distance of 120 cm from the core side.

In our case the maximum power of TRIGA reactor is 250 kW, but on the other side the irradiation point is almost 360 cm from the core. So, unfortunately the irradiation times would probably be much longer than one hour, but the exact numbers will be available after first experiments.

Nevertheless, the irradiation channel can of course be used for "in vitro" studies: boronated malignant tissue will be inserted into the channel, where fluxes are much higher.

Since detailed geometry model of the reactor is already prepared, the focal point of our future work will be searching for suitable and efficient variance reduction techniques, necessary for successful calculations on geometrically demanding Monte Carlo models.

The applicability of developed TRIGA geometrical model would not be only for the purpose of BNCT, but also for the Prompt Neutron Gamma Activation Analysis (PNGAA), Proton Recoil Spectrometry and many other activities.

5. References

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