Sudan University of Science and Technology

College of Graduate Studies

Monte Carlo Modeling of Fast Neutron Moderating System Suitable For Use in Boron Neutron Capture Therapy

محاكاة مونتي كارلو لنمذجة نظام لتهدئة النيوترونات السريعة المستخدمة في تقنية إلتقاط البورون نيوترون العلاجية

A thesis submitted in fulfillment for the requirements of M.Sc. Degree in Physics

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الآيـــــــــة

قال تعالى:

(إِنَّ فِي ذَٰلِكَ لَآيَةً ۖ وَمَا كَانَ أَكْثَرُهُمْ مُؤْمِنِينَ ﴿٨﴾)

صدق الله العظيم

الشعراء: ٨

Dedication

Persons we knew that they did not want to be named, these persons that bialy their life for our weal our parents.

for best persons in our life our friends.

For our spiritual father Dr/ Nassreldeen Abdelrazig Abdelbari Elsheikh

And Dr/Ahamed Elhasan Elfaki

Acknowledgment

First and above all, I praise Allah the almighty for providing me this opportunity and granting me the capability to do such work.

I want to express deep thanks for accepting me in M.Sc. student, the warm encouragement, deliberated guidance and critical comments that promoting our research skills.

Abstract

There is no one treatment for cancer and the search for ways to combat cancer have led to many different treatments, including surgery, chemotherapy, and radiation therapy. However, these treatments are not always effective, and in such cases new treatments must be developed. Boron neutron capture therapy (BNCT) is a treatment that has been proposed to combat brain tumor and Neck tumors, two tumors that are resistant to traditional cancer therapies. Boron neutron capture therapy (BNCT) is based on the nuclear reaction that occurs when boron-I0 is irradiated with low-energy thermal neutrons to yield alpha particles and recoiling lithium- 7 nuclei ($B^{10}+n_{th} \rightarrow [B^{11}] \rightarrow He^4_2+Li^7+2.31$ MeV). The potential of the spontaneously fissioning isotope ^{252}Cf, to provide epithermal neutrons for use in boron neutron capture therapy (BNCT) has been investigated using Monte Carlo simulations to design an assembly composed of located inside a gamma filter made of sphere of high density lead, 1cm inner radius and 3 cm outer radius, which in turn located inside a fast neutron moderator in a form of a cylinder of radius 13cm, the moderator is surrounded by the outer radius is 15cm and the space between the two cylinders is filled with graphite for reflection. A third cylinder is added with a diameter of 17cm and filled with borated material to work as staff shielding. The height of the containing outer cylinder is 80cm the pump was designed and tested and the neutron flux was found 3.4×10^9 n / $cm^2.s$ this result is in excellent coherence compared to the result of previous studies this is due to optimal selection of materials according to the required standards as explained by the simulation in chapter four.

المستخلص

لا يوجد علاج واحد للسرطان وقد أدى البحث عن طرق لمكافحة السرطان إلى العديد من العلاجات المختلفة بما في ذلك الجراحة والعلاج الكيميائي والعلاج الإشعاعي. ومع ذلك فإن هذه العلاجات ليست فعالة على الدوام ، وفي مثل هذه الحالات يجب تطوير علاجات جديدةتمثل العلاج عن طريق اسر النيوترونات بالبورن (BNCT) هو العلاج الذي تم اقتراحه لمكافحة اورام المخ الخبيثة و اورام الرقبة المتكررة ، لانهما تقاومان علاجات السرطان التقليدية. يعتمد علاج التقاط البورون النيوتروني (BNCT) على التفاعل النووي الذي يحدث عندما يتم إشعاع البورون10مع النيوترونات الحرارية المنخفضة الطاقة لإنتاج جسيمات ألفا ونوات ذرة الليثيوم 10 وفقا للمعادلة التالية $(B^{10} + {}_{nt}h \rightarrow [B^{11}] \rightarrow He^{4}_{2} + Li^{7} + 2.31\ MeV)$ وقد تم التحقيق في إمكانات نظير الانشطار العفوي لعنصر الكاليفورنيا $Cf^{52\ 2}$ لتوفير نيوترونات حرارية لاستخدامها في تقنية التقاط البورون نيوترون العلاجية (BNCT) تم استخدام محاكات مونت كارلو MCNP لتصميم مضخة للنيوترونات الحرارية و التي يتكون من الداخل مرشح لاشعة قاما مصنوع من اسطوانة من الرصاص عالي الكثافة نصف القطرها الداخلي1سم ونصف قطرها خارجي 3 سم ، والذي يقع بدوره داخل مهدئ للنيوترونات السريعة في شكل اسطوانة من دائرة نصف قطرها 13 سم ، يحيط المهدئ دائرة نصف قطرها الخارجي 15 سم وتمتلئ الفراغ بين الاسطوانتين بالجرافيت تعمل كعاكس للنيوترونات و تضاف أسطوانة ثالثة بقطر 17 سم مليئة بمواد معالجة بالبورون يعمل كدروع حماية للموظفين. يبلغ طول الاسطوانة الخارجية المحتوية على 80 سم صممت المضخة و تم اخبتارها و وجد ان التدفق النيوتروني عبارة عن $3.4\times10^{9}\ n/cm^{2}.s$ و هذه النتيجة متسخة بامتياز مع النتائج التي تم التوصل اليها في الدراسات السابقة و يرجع ذلك للاختيار الامثل للمواد وفقا للمعايير المطلوبة كما وضحتها نتائج المحاكاة في الفصل الرابع

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Abbreviations

BNCT	Boron neutron capture therapy
MCNB	Monte Carlo N-Particle Transport Code
EV	Electron Volt
CBE	conventional term Radiobiological Effectiveness
ABNS	Accelerator Based Neutron Source

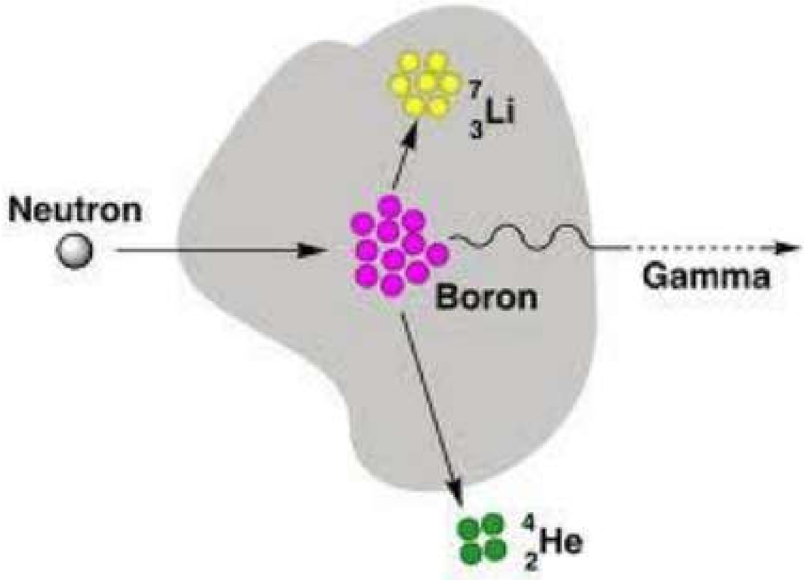
Chapter One
Introduction

1.1 Overview

Just 4 years after the discovery of neutron the concept of neutron capture theory was introduced in 1936 by G.L.Locher in Pennsylvania NCT is a promising method for cancer treatment where the conventional radiotherapy fail in some situation, the tumor has not well defined limit. Boron and Neutron constitute together a binary weapon against cancer therapy. BNCT is a technique, very innovative cancer therapy and an investigational form of radiation therapy. That is currently in the development stage for the treatment of gliblastoma mutiforme or anaplastic astrocytoma and all the form of human brain tumor. [1]

Clinical trial of BNCT was started in 1951 by Farr et al and it was improved by sweet et al But the result of the clinical trials disappointing due to inadequate boron compounds. So it is discontinued in 1961 and further clinical trials were performed in 1968 by H.Hatanaka in japan and the results shows some outstanding agreement .Then United States and some European countries restarted their clinical trials about BNCT after the interesting and outstanding outcome from the H. Hantanaka and co-works study. The International society for Neutron capture therapy was founded in 1984 after the encouraging out come from his clinical trials. Before start of clinical trials, the development and construction of BNCT facility were realized at LVR reactor of NRI for treatment of human brain gliomas Boron. [1]

Neutrons capture therapy based on two phenomena. First one, Non-radioactive boron Drag was delivered through injection to the tumor tissue and the next one, patient was irradiated with epithermal neutron until the normal tissue dose limit is reached. The advantage of the boron is it has large cross section (3840 barn) for slow thermal neutron. [1]



$$^{10}B + {}^{1}n \rightarrow {}^{7}Li + {}^{4}He$$

Fig 1.1: Nuclear reaction utilized in Boron neutron capture therapy.

When thermalized neutron captured by the non-radioactive boron atom, it decays through short range α-particle and recoiling Li-7 nucleus. These emitted charged particles can travel only a short distance (10µm) has a high linear energy transfer and an associated high relative biological effectiveness. This energy has an ability to locally destroy the cancer tissue that containing the boron drag without appreciably harming the healthy cell around the tumor cell. [1]

1.2 Research problem

The problem of research is to use Monte Carlo Modeling of fast neutrons moderating system suitable for use in Boron neutron capture therapy BNCT because all the sources of neutrons in nature are sources that produce fast neutrons and these fast neutrons are not compatible with BNCT technology. Neutrons are so quick to design a pump to slow the high energy and reduce the permissible amount of treatment.

1.3Objectives of the research

The objectives of this work were, using Monte Carlo simulations to model a thermal neutron pump suitable for the applications of BNCT and to examine the efficiency of the modeled neutron pump against different materials shielding using Monte Carlo transport code (MCNP5).

1.4Research Methodology

In this work the literature review about BNCT and its working principles was covered and Monte Carlo radiation transport code MCNP 5 is used to model a system that works as a thermal neutron pump suitable for BNCT. Monte Carlo Simulations will be carried out to examine the modeled thermal neutron pump against different material shields. The results will be presented and analyzed with respect to the output energy from thermal neutron pump.

1.6Previous studies

The potential of the thermal column of Tehran Research Reactor (TRR) to provide epithermal neutron beam for Boron Neutron Capture Therapy (BNCT) has been investigated using Monte Carlo simulation. A Beam Shaping Assembly (BSA) has been designed and optimized to meet BNCT neutron beam criteria recommended by International Atomic Energy Agency. The suggested BSA configuration in cylindrical geometry consists of 20 cm Al as a moderator, 35 cm Pb as a reflector, two 5 cm Bi slabs as gamma shield, and two 2 mm Cd sheets as thermal neutron filters. The results show that epithermal neutron flux at the exit of the BSA can be 0.65 × 109 n/cm2 s. In-phantom dose analysis indicates that the designed neutron beam can be used for treatment of deep-seated brain tumors in acceptable time[2] .
(YaserKasesazaHosseinKhalafi aFaezehRahmani)

This article involves two aims for BNCT. First case includes a beam shaping assembly estimation for a D–T neutron source to find epi-thermal neutrons which are the goal in the BNCT. Second issue is the percent depth dose calculation in the adult Snyder head phantom. Monte-Carlo simulations and verification of a suggested beam shaping assembly (including internal neutron multiplier, moderator, filter, external neutron multiplier, collimator, and reflector dimensions) for thermalizing a D–T neutron source as well as increasing neutron flux are carried out and our results are given herein. Finally, we have simulated its corresponding doses for treatment planning of a deeply-seated tumor.

Highlights

An assembly for the D–T neutron source including many regions is given herein. Dosimetry simulations in the Snyder head phantom for a deeply-seated tumor are carried out. Brief literatures conclusions on the recent BNCT studies are presented herein (F.FaghihiabS.Khalilia [3]) .

The slowing down of the fast neutrons, resulting in a thermal neutron distribution of a phantom, has been computed using a Monte Carlo model. This model, which includes a deep-seated tumour, was experimentally verified by measurements of the thermal neutron fluence rate in a phantom using neutron activation of gold foil. When non-boronated water phantoms were irradiated with a total dose of 1 Gy at a depth of 6 cm, the thermal fluencies at this depth were found to be $2*10^{10}$ cm-2. The absorbed dose in a tumor with 100 ppm 10B, at the same depth was enhanced by 15%.

(F.Poller, W.Sauerwein and J. Rassow[4])

The potential of the spontaneously fissioning isotope, ^{252}Cf, to provide epithermal neutrons for use in boron neutron capture therapy (BNCT) has been investigated using Monte Carlo simulation. The Monte Carlo code MCNP was used to design an assembly composed of a 26 cm long, 11 cm radius cylindrical D_2O moderator followed by a 64 cm long Al filter. Lithium filters are placed between the moderator and the filter and between the Al and the patient. A reflector surrounding the moderator/filter assembly is required in order to maintain adequate therapy flux at the patient position. An ellipsoidal phantom composed of skull- and brain-equivalent material was used to determine the dosimetric effect of this beam. It was found that both advantage depths and advantage ratios compare very favourably with reactor and accelerator epithermal neutron sources. The dose rate obtainable, on the other hand, is 4.1 RBEc Gy min-1, based on a very large (1.0 g) source of ^{252}Cf. This dose rate is two to five times lower than those provided by existing reactor beams and can be viewed as a drawback of using ^{252}Cf as a neutron source (J C Yanch, J K Kim and M J Wilson [5]).

The pathology of malignant brain tumors often precludes successful treatment by surgery and standard radiation therapy. Boron neutron-capture therapy consists of the selective loading of tumor with ^{10}B and subsequent irradiation with a thermal or epithermal neutron field. The neutron-capture reaction $^{10}B(n,\alpha)^{7}Li$ produces high-linear-energy-transfer-charged particles that deposit energy principally within the abnormal tissue that contains a high ^{10}B concentration.

Constraints on this therapy modality are imposed by radiation effects in normal tissue from thermal neutrons, neutron-induced gamma rays, fast-neutron and gamma-ray beam contaminants, and also from the $^{10}B(n,\alpha)7Li$ reactions in circulating blood. The ANDY general geometry Monte Carlo code is used to calculate the space-energy distribution of all pertinent components of the dose within a simple head phantom in an idealized therapy configuration at the Massachusetts Institute of Technology Research Reactor. The effects of 10B concentration, gamma-ray contamination of the therapy beam, thermal neutron beam aperture, and surgically formed re-entrant cavities are examined with respect to several clinical criteria for therapeutic efficacy.

It is found for the model considered that the maximum effective relaxation length for the thermal neutron flounce (Murray et al [6]).

This paper reviews the development of boron neutron capture therapy (BNCT) and describes the design and dosimetry of an intermediate energy neutron beam, developed at the Harwell Laboratory, principally for BNCT research. Boron neutron capture therapy is a technique for the treatment of gliomas (a fatal form of brain tumor). The technique involves preferentially attaching ^{10}B atoms to tumor cells and irradiating them with thermal neutrons. The thermal neutron capture products of ^{10}B are short range and highly damaging, so they kill the tumor cells, but healthy tissue is relatively undamaged. Early trials required extensive neurosurgery to expose the tumor to the thermal neutrons used and were unsuccessful. It is thought that intermediate-energy neutrons will overcome many of the problems encountered in the early trials, because they have greater penetration prior to thermalization, so that surgery will not be required. An intermediate-energy neutron beam has been developed at the Harwell Laboratory for research into BNCT. Neutrons from the core of a high-flux nuclear reactor are filtered with a combination of iron, aluminium and sulphur. Dosimetry measurements have been made to determine the neutron and gamma-ray characteristics of this beam, and to monitor them throughout the four cycles used for BNCT research. The beam is of high intensity (~2×l07neutrons cm−2s−1, equivalent to a neutron kerma rate in water of 205mGyh−1) and nearly monoenergetic (93% of the neutrons have energies ~24keV, corresponding to 79% of the neutron kerma rate) (K.G.Harrison et al [7])

A modified neutron production target assembly has been developed to provide improved performance of the proton–cyclotron-based neutron radiotherapy facility at the University of Washington for applications involving neutron capture enhanced fast-neutron therapy. The new target produces a neutron beam that yields essentially the same fast-neutron physical depth-dose distribution as is produced by the current UW clinical system, but that also has an increased fraction of BNCT enhancement relative to the total therapeutic dose. The modified target is composed of a 5-millimeter layer of beryllium, followed by a 2.5-millimeter layer of tungsten, with a water-cooled copper backing. Measurements of the free-field neutron spectrum of the beam produced by the new target were performed using activation foils with a direct spectral unfolding technique. Water phantom measurements were performed using a tissue-equivalent ion chamber to characterize the fast-neutron depth-dose curve and sodium activation in soda–lime glass beads to characterize the thermal-neutron flux (and thus the expected neutron capture dose enhancement) as a function of depth. The results of the various measurements were quite consistent with expectations based on the design calculations for the modified target. The spectrum of the neutron beam produced by the new target features an enhanced low-energy flux component relative to the spectrum of the beam produced by the standard UW target. However, it has essentially the same high-energy neutron flux, with a reduced flux component in the mid-range of the energy spectrum. As a result, the measured physical depth-dose curve in a large water phantom has the same shape compared to the case of the standard UW clinical beam, but approximately twice the level of BNCT enhancement per unit background neutron dose at depths of clinical interest. In-vivoclinical testing of BNCT-enhanced fast-neutron therapy for canine lung tumors using the new beam was recently initiated.

(Yale D. Harker, George E. Laramore [8]).

The $^{7}\mathrm{Li}(p,n)^{7}\mathrm{Be}$ reaction has been investigated as an accelerator-driven neutron source for proton energies between 2.1 and 2.6 MeV. Epithermal neutron beams shaped by three moderator

materials, $\mathrm{Al}/\mathrm{AlF_3}$, $^{7}\mathrm{LiF}$, and D_2O, have been analyzed and their usefulness for boron neutron capture therapy (BNCT) treatments evaluated. Radiation transport through the moderator assembly has been simulated with the Monte

Carlo N-particle code (MCNP). Fluence and dose distributions in a head phantom were calculated using BNCT treatment planning software. Depth-dose distributions and treatment times were studied as a function of proton beam energy and moderator thickness. It was found that an accelerator-based neutron source with Al/AlF_3 or ^{7}LiF as moderator material can produce depth-dose distributions superior to those calculated for a previously published neutron beam design for the Brookhaven Medical Research Reactor, achieving up to ~50% higher doses near the midline of the brain. For a single beam treatment, a proton beam current of 20 mA, and a ^{7}LiF moderator, the treatment time was estimated to be about 40 min. The tumor dose deposited at a depth of 8 cm was calculated to be about 21.Gy-Eq.
(D. L. Bleuel, R. J. Donahue, B. A. Ludewigt, J. Vujic[9]).
In general, in BNCT, a high neutron flux is necessary in order to achieve high sensitivities. In this regard, the obvious choices are facilities based on nuclear reactors or cyclotron or neutron generators. However, for reasons of cost and/or applications requiring mobility, flexibility, etc., an isotopic source such as ^{252}Cf provides a good alternative in this work, we would like to investigate the possibility of using Cf neutron source in a different geometry for providing thermal neutrons.

Chapter Two

The Working Principles of (BNCT)

2.1 Introduction

Boron Neutron Capture Therapy is a non-invasive therapeutic modality, developed for the treatment of malignant tumors, especially malignant tumors of the brain. BNCT was first proposed by Locher in 1936. In BNCT, a ^{10}B containing compound is introduced into a patient. The patient is then irradiated in an epithermal neutron field. Neutrons from this field thermalize as they pass through brain tissue resulting in a thermal neutron flux at and around a tumor site. The ^{10}B atoms, strongly absorb thermal neutrons in an (n, α) reaction and emit energetic α particles and ^{7}Li recoil nuclei that deposit their energy within approximately one cell diameter in the surrounding tissue. In successful BNCT, tumor cells are destroyed while the surrounding healthy tissue remains, to as large an extent as possible, unharmed. There are three requirements that must be met in order to accomplish this. The first requirement is that a sufficiently large amount of ^{10}B must be delivered to the tumor cells. The second requirement is that the ^{10}B must be preferentially absorbed by the tumor cells, as opposed to healthy tissue. In addition, ^{10}B concentrations in the blood must be minimized to prevent excessive damage to the blood vessel walls. The third requirement is that sufficiently large thermal neutron fluence must be delivered to the tumor site in a reasonable treatment time while respecting the tolerance of normal tissues to neutron dose. [10]

2.2 BNCT Physical Mechanisms

BNCT is a form of binary radiotherapy and therefore involves two key stages. The first is the preferential accumulation, in tumour cells, of an isotope with a suitable affinity for neutrons at certain energy. This must then be followed by an intense irradiation of these cells with neutrons at energy such that their probability for capture is maximized.

In BNCT, this means a preferential accumulation of boron, followed by submerging the tumour and the surrounding healthy brain tissue in a bath of thermal neutrons. These thermal neutrons may either be incident directly onto the tissue, for superficial tumors or intra-operative treatment, or may be produced locally by moderation of a higher energy neutron source which is incident on to the patient surface.

The interaction of thermal and epithermal neutrons is dominated by scattering. As a result, neutrons of these energies cannot be directed in a beam-like manner. In order to achieve the required thermal neutron fluence at the tumor, the whole head (in the case of brain tumors) is given a sub therapeutic radiation dose, which is therapeutically effective in the tumor cells because of the increased dose from the boron capture reaction.

However, for the currently available boron carrier compounds, the degree of selectivity in concentration between tumour and healthy tissues is still so low that careful treatment planning, combined with a detailed understanding of the effects of the neutron irradiation on the healthy brain, are essential prerequisites to a therapy program.

In describing the physical basis for BNCT, it is necessary to use a number of terms which will be familiar to clinicians and physicists alike. However, in the case of BNCT, these terms may not carry their normal inferences. For example, absorbed dose is related to energy deposited by the radiation field, and in clinical radiotherapy is directly correlated with tumour control probability.

In BNCT the major dose component, that from the boron capture reaction ^{10}B (n, a) ^{7}Li, is deposited non-uniformly on the microscopic scale. The ranges of the reaction products from this reaction are 5 and 9 mm for the ^{7}Li and a respectively which are comparable with typical cellular dimensions of around 10 mm .Hence, if the boron is located far from the radiosensitive structures with in cell nucleus, dose (i.e. energy) will be deposited during the therapy with negligible impact on tumour control. This is a central problem to the whole of BNCT and has been investigated by many authors (Gabel et al., 1987; Kalend et al., 1995).[11]
Boron Neutron Capture Therapy(BNCT) is based on the nuclear capture and fission reactions that occur when non-radioactive ^{10}B, which makes up approx. 20% of natural elemental boron, is irradiated with neutrons of the appropriate energy to yield high energy alpha particles ("stripped" down ^{4}He nuclei) and high energy lithium-7 (^{7}Li) nuclei. The nuclear reaction is:

$$^{10}B + n_{th} \rightarrow [^{11}B] \rightarrow \alpha + {}^{7}Li + 2.31\ MeV \quad (2\text{-}1)$$

Both the alpha particles and the lithium ions produce closely spaced ionizations in the immediate vicinity of the reaction, with a range of approximately 5 - 9 μm, or approximately the diameter of one cell. Their lethality is limited to boron containing cells. BNCT, therefore, can be regarded as both a biologically and a physically targeted type of radiation therapy.

The success of BNCT is dependent upon the selective delivery of sufficient amounts of ^{10}B to the tumor with only small amounts localized in the surrounding normal tissues.

Thus, normal tissues, if they have not taken up boron-10, can be spared from the nuclear capture and fission reactions. Normal tissue tolerance is determined by the nuclear capture reactions that occur with normal tissue hydrogen and nitrogen.[12]

In theory BNCT is a highly selective type of radiation therapy that can selectively target the tumor at the cellular level without causing radiation damage to the adjacent normal cells and tissues. Doses up to 60 - 70 Gy can be delivered to the tumor cells in one or two applications compared to 6 - 7 weeks for conventional external beam photon irradiation. However, the effectiveness of BNCT is dependent upon a relatively homogeneous distribution of ^{10}B within the tumor, and this is still one of the key stumbling blocks that have limited its success. [13]

2.3 Biological Effectiveness

There is therefore a need to characterize the biological effectiveness exhibited by the boron distribution associated with a particular compound. For this purpose the term Compound Biological Effectiveness (CBE) has been developed.

This is an empirical term which has utility in characterizing a treatment plan. It is analogous to the conventional term Radiobiological Effectiveness (RBE) since it is a simple ratio:

$$\mathrm{CBE} = \frac{\text{dose of photons required to give a certain surviving fraction}}{\begin{array}{c}\text{dose from the neutron capture reaction which}\\ \text{gives the same surviving fraction}\end{array}}$$

It has tended to be used as a multiplier for physical dose (just as RBE is used) in order to provide an overall characterization of the effectiveness of a particular treatment.

However, by agreement of all parties at the recent Seventh Symposium on Neutron capture Therapy, Zurich, September 1996, papers and patient treatment prescriptions in BNCT will carry full information on physical and biologically equivalent dose, as well as on the conversion factors (RBE and CBE) which have beenused to derive equivalent doses from absorbed dose measurements and calculations.[11]

2.4. Boron delivery agents

Since the tumor control is affected directly by the location of the boron atoms within a cell, it will be affected directly by the compound used to carry boron to the cell. Although there is not a lot of physics behind the boron compounds used in BNCT, because the principal dose in BNCT comes from the boron-10 capture reaction, it is important to have a good understanding of how these compounds are delivered and selectively taken up by tumors. The development of boron delivery agents for BNCT began approximately 50 years ago and is an ongoing and difficult task of high priority.

A number of pronated pharmaceuticals using boron-10, have been prepared for potential use in BNCT. The most important requirements for a successful boron delivery agent are:

i. Low systemic toxicity and normal tissue uptake with high tumor uptake and concomitantly high tumor: to brain (T:Br) and tumor: to blood(T:Bl) concentration ratios (> 3– 4:1).
ii. Tumor concentrations in the range of ~20 μg ^{10}B/g tumor.
iii. Rapid clearance from blood and normal tissues and persistence in tumor during BNCT.

The two compounds currently most important to BNCT are di-sodium undecahydro-mercapto-closo-dodecacarborate (BSH), and p-boron phenylalanine (BPA).

These two compounds represent two different approaches to delivering boron to tumors. BSH relies on passive diffusion from the blood into brain tumors.

Brain tumors disrupt the blood brain barrier (BBB), and as a result BSH is able to diffuse into cancerous cells but not into healthy areas of the brain where the blood brain barrier is still intact. [14, 15]

Studies have shown that BSH can be taken up selectively by tumors, typically with a tumor/blood ratio between1:1 and 2: 1.6 BPA, on the other hand, is actively taken up by cancerous cells. [15, 16]

New methods for enhancing the uptake of boron delivery agents have also been studied. These methods focus on increasing blood flow to the tumor and the permeability of the BBB at the time of the boron delivery agent's administration. This is typically accomplished with different drugs. Direct injection of the boron delivery agent into the tumor has also proved to be an effective way of increasing the boron concentration in cancerous cells. The concentration of boron in both healthy and tumor cells is extremely important to BNCT, and the ways in which boron is delivered and taken up must always be kept in mind when thinking of how dose will be delivered and how effective a BNCT treatment will be delivered and how effective a BNCT treatment will be.(See Figure 2.1). [14, 16]

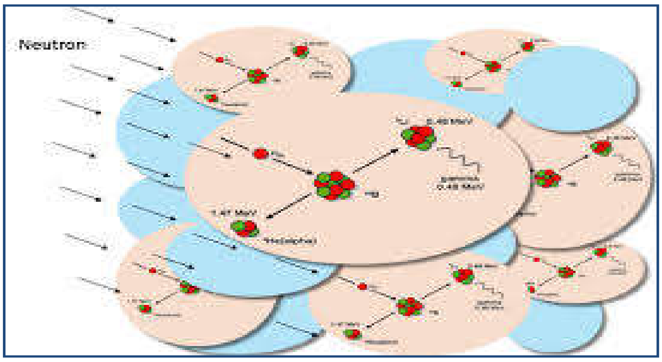


Fig2.1: Mechanism of BNCT binary treatment.

2.5 Treatment Planning

With an understanding of how ionizing radiation delivers a dose to cells and tissues, treatments for radiation therapy can be put together. Each treatment plan is unique, taking into account the specific geometry of the patient in order to find a way to deliver the dose necessary to kill the tumor, without delivering an excessive radiation dose to healthy tissues.

In order to do this, the radiation dose delivered to every point in the body for a given irradiation must be estimated and then optimized.Calculating these estimates is not an easy task, especially in the case of BNCT with all the different dose components described above.

To develop an accurate an effective treatment plan for BNCT, one must know the anatomy of the patient, the dimensions of the tumor, the concentration of boron in all the parts of the body to be irradiated, the components of the neutron beam, and how they will interact with the tissues present.[13]

For radiation therapy, Monte Carlo simulations take into account the characteristics of an incident radiation beam, such as the number of neutrons and photons, their energy spectra, and angular distributions, and the geometry of the target, given by data from MRI (magnetic resonance imaging), CT (computed tomography), and PET (positron emission tomography) images, and simulate the many possible trajectories of each incident particle, arriving at an average result. [16, 17]

The biggest challenge in accurately calculating dose for BNCT is the boron distribution in tissues. In the simplest treatment planning programs, the boron concentration in tumor cells is given a certain value and assumed to be uniform throughout the tumor, and boron concentration in healthy tissues is assumed to be zero .As mentioned above, boron concentration within the tumor is highly variable, between patients and within single patients between different boron administrations, and not all tumor cells take up the same amount of boron.Because the products of the boron capture reaction, which deliver the primary dose in BNCT, give up their energy in such a short distance, dose distributions on a microscopic scale are important to treatment planning.One must know where in the cell the energy from the $^{10}B(n,\alpha)^{7}Li$ reaction is deposited. [13, 16]

Treatment planning for BNCT is a continually evolving process, and as it continues to evolve, it makes BNCT an ever more attractive and effective treatment for tumors that have resisted traditional radiotherapy.

2.6 Neutron sources

Neutron beams for BNCT must have some very specific characteristics. They must be thermal or epithermal, depending on the depth of the tumor, and they must be uncontaminated by fast neutrons or gamma rays. They also must have an adequate fluence rate, the total number of neutrons passing through a given area.

This is dependent on the concentration of boron in the target, but is typically around 10^9 $cm^{-2}s^{-1}$, so that the dose can be delivered in a short time, around 30 minutes. Finally, the beam should be well collimated to avoid excessive dose to tissues outside the treatment area.

To date, all clinical BNCT trials have been carried out using reactor-based neutrons due to the high neutron flux required, while there is urgent need for hospital-based neutron sources. The development of accelerator based neutron sources (ABNSs), which could be safely installed at the hospital, has been of interest for almost three decades.To make BNCT more accessible and practical, the use of linear accelerators (linacs) and cyclotrons as sources of neutron beams has been investigated. Linacs and cyclotrons can produce neutrons by accelerating either protons or deuterons at lithium, beryllium, or tritium targets. Many different neutron-producing reactions could be exploited with an accelerator. The neutron-producing reactions are induced by accelerated protons, deuterons, or tritons targeting ^{7}Li,^{9}Be,^{13}C,^{12}C,^{2}H, or ^{3}H nuclei, via the reactions listed in(Table2.1).

Table2.1:Characteristics of charged particle reactions considered for accelerator-based BNCT.

Reaction	**Bombardingenergy (MeV)**	**Average neutronenergy (MeV)**	**Maximum neutron energy(MeV)**	**Neutron production rate (n $mA^{-1}s^{-1}$)**
$^{7}Li(p,n)^{7}Be$	2.5	0.55	0.79	9.1×10^{11}
$^{9}Be(p,n)^{9}B$	4.0	1.06	2.12	1.0×10^{12}
$^{9}Be(p,n)^{9}B$	30		28	1.9×10^{14}
$^{9}Be(d,n)^{10}B$	1.5	2.01	5.81	3.3×10^{11}
$^{13}C(d,n)^{14}N$	1.5	1.08	6.77	1.9×10^{11}
$^{2}H(d,n)^{3}He$	o.15	2.5	2.5	4.7×10^{8}
$^{3}H(d,n)^{3}He$	0.15	14.1	14.1	5.0×10^{10}

Neutrons are classified according to their energies as thermal (E_n<0.5 eV), epithermal (0.5 eV<E_n<10 eV) or fast (E_n>10 eV). Thermal neutrons are the most important for BNCT since they usually initiate the ^{10}B (n, α) ^{7Li} capture reaction. However, because they have a limited depth of penetration, epithermal neutrons, which lose energy and fall into the thermal range as they penetrate tissues, are now preferred for clinical therapy. [18, 13]

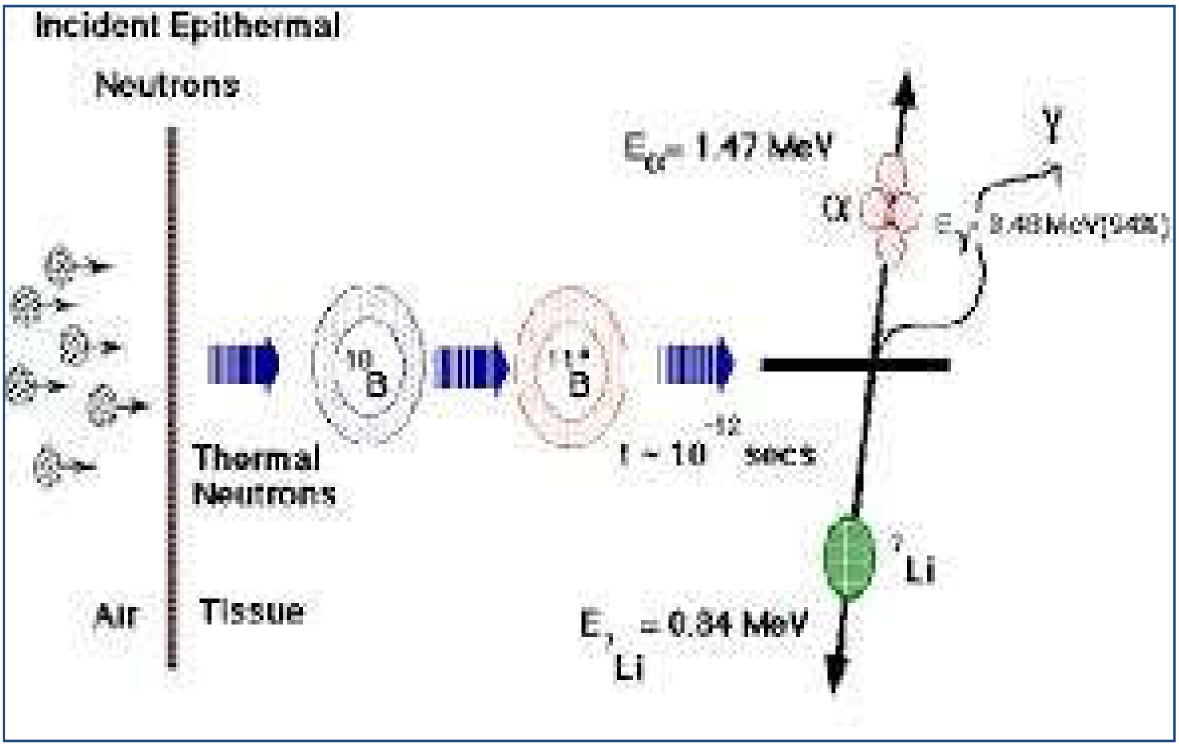


Fig2.2: Epithermal Neutrons moderate to thermal neutrons when penetrate tissue.

A common problem for all neutron sources in BNCT is that, in contrast to other beam therapies (e.g. proton, fast neutron, photon, or electron sources), considerable moderation of the source neutrons is usually required, since the efficient ways of producing neutrons usually yields neutrons of high energy (> 0.7 MeV). In this work, Monte Carlo calculations were carried out to model a geometry for a suitable thermal neutron pump applicable to BNCT.

The only way to slow these high energy neutrons down is through nuclear interactions, and thus during the moderation, neutron intensity is reduced by about four orders of magnitude. Consequently, the BNCT neutron source needs to be operated at considerably higher power than a proton therapy accelerator.

Once clinical trials have been initiated, it will be important to determine how these ABNS compare to BNCT that has been carried out in the past using nuclear reactors as the neutron source. [15]

2.7 Beam Monitoring and Control

Reliably and reproducibly administering the prescribed dose in external beam therapy requires a method for monitoring and integrating beam output while a radiation field is administered. In BNCT, most of the absorbed dose is derived from neutron interactions in tissue, and so a method is needed for monitoring neutrons transmitted by the beam. (See Figure.2.3)[16]

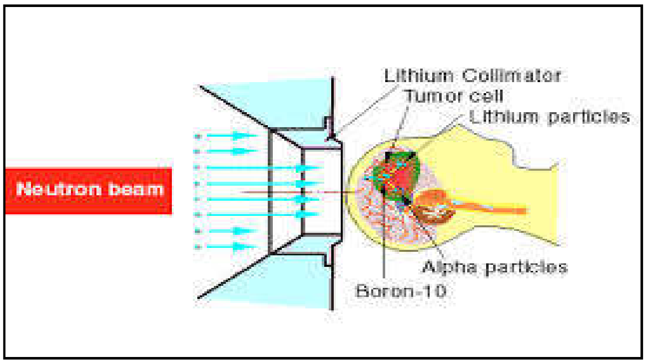


Fig2.3:A schematic diagram illustrating of intra operative BNCT.

Uranium-lined fission counters are used in many systems because they easily discriminate against the gamma rays inevitably contained in the beam and can be fabricated with the sensitivity necessary for sampling output in the epithermal energy range without significantly perturbing beam characteristics. Helium or boron gas- filled detectors are also a good choice for these reasons and have been used successfully at some clinical centers. [15, 19]

These detectors are inherently sensitive to thermal neutrons, and it is therefore common practice to use a thermal neutron-absorbing shroud such as cadmium to reduce this response, which may arise from neutrons that backscatter toward the detector in either the collimator or patient. Gamma rays emitted by the beam itself sometimes comprise a non-negligible portion of the absorbed dose in tissue, and beam monitoring systems may therefore contain either ionization chambers or Geiger-Müller detectors that are sensitive to gamma rays. [12]

In practice, since this dose component mostly derives from activation of beam line components, it is generally proportional to the neutron output of the beam, and the gamma-ray monitor is used only for informational purposes. [12]

The beam monitoring system is frequently equipped with a computer for displaying and archiving readings from the beam monitors throughout an irradiation.Activation foils are insensitive to gamma rays and for low neutron energies have an energy-dependent response very similar to that for neutron capture in boron.These measurements may also be augmented with small thermol uminescent dosimeters (TLDs) to measure the gamma-ray dose in situ. Although wires and TLDs may be affixed to the skin for monitoring external beam irradiations, this approach is most practical for intraoperative BNCT where detectors may be implanted near the tumor.

Relatively long irradiations are required for this technique to ensure that there is enough time to complete the necessary measurements and determine the appropriate stopping time. The precision and accuracy required for timing control in patient irradiations are inversely proportional to beam intensity. [15, 12]

2.8 Filtration

Whether neutron beams are produced by nuclear reactors, linear accelerators, or cyclotrons, they start with a broad energy range and are contaminated with other types of radiation, and therefore require filtration. Although these neutron beams can be produced in different ways, the same materials and techniques can be used to filter them. A good neutron filter should possess some basic characteristics: it should maintain a high neutron flux, significantly decrease the energy of fast neutrons without producing γ rays, filter out unwanted γ rays, and filter out neutrons with an energy lower than desired.The degree to which a filter does any of the above mentioned things to an incident neutron beam depends on its cross-section for these various processes. [14]

The cross-section is a way of describing the probability of an interaction taking place between two particles, and is a very important concept in both the filtration of neutron beams. Every type of interaction has a cross section, and cross sections are dependent on the particles involved and their energies. It should also be noted that two particles can often interact in a variety of ways, and each of these interactions will havea cross section. [14, 18]

The most promising materials to moderate fast neutrons while maintaining a high beam flounce and purity seem to be heavy water (D_2O) or a combination of heavy elements like uranium, moderate elements like manganese and copper, and lighter fluorine compounds. Heavy elements such as uranium slowdown fast neutrons and increase neutron flux by fission reactions, moderate elements like manganese further slowdown the neutron beam, and finally, the fluorides shape the neutron beam by brining all of the neutrons to a desired energy level. Fluorides shape the beam by preferentially scattering neutrons. [14, 20]

That is, neutrons with higher energies will continually be scattered until they are at an energy where the scattering cross section with fluorine is much lower, and they pass out of the filter at this energy. These filters are fairly good at eliminating high energy neutrons, but they still give a fairly broad spectrum of lower energy neutrons.

In addition, a good filter for BNCT must remove unwanted forms of radiation, such as neutrons with energies too low to be useful in treating deep seated tumors, and γ rays. The best materials to remove low energy neutrons have a large capture cross section for neutrons, but do not produce unwanted forms of radiation after they capture neutrons. ^{6}Li seems to be the most promising material to remove thermal neutrons. To remove γ rays contaminating the neutron beam lead is typically used because it has large cross sections for various photon interactions.

Finally, some material must be used to reflect scattered neutrons back into the beam to maintain a sufficient neutron flux to the patient. Lead and graphite are typically used because they have high scattering cross sections for neutrons, but maintain a sufficient flux of neutrons at the desired energy. [12]

Chapter Three
Monte Carlo Modeling

3.1 Introduction

One of the codes recommended for the BNCT calculations is the General Monte Carlo N-Particle Transport Code (MCNP) (X-5 Monte Carlo Team 2008). Monte Carlo N-Particle Transport Code (MCNP) was originally developed by the Monte Carlo Group, currently the Diagnostic Applications Group, (Group X-5) in the Applied Physics Division (X Division) at the Los Alamos National Laboratory.

MCNP is a general-purpose code that can be used for neutron, photon, electron,or coupled neutron/photon/electron transport, including the Capability to calculate Eigenvalues for criticalsystems.

The code treats an arbitrary three–dimensionalconfiguration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. [13, 17]

Point wise cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation (such as ENDF/B-VI) are accounted for. Thermal neutrons are described by both the free gas and S (α, β) models. For photons, the code accounts for incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorptionin pair production with local emission of annihilation radiation, and bremsstrahlung.

Continuous slowing-down model is used for electron transport that includes positrons, k x-rays, and Bremsstrahlung, but does not include external or self-induced fields.

Important standard features that make MCNP very versatile and easy to use include a powerful general source, criticality source, and surface source; both geometry and output tally plotters; a rich collection of variance reduction techniques; a flexible tally structure; and an extensive collection of cross -section data.MCNP contains numerous flexible tallies: surface current & flux, volume flux (track length), point or ring detectors, particle heating, fission heating, pulse height tally for energy or charge deposition, mesh tallies, and radiography tallies. [17]

3.2 The Input file

The input file developed to model the current proposed neutron source suitable for BNCT follows the requirements of MCNP code. It consists of data showing the geometry, the source specification, the materials used and the tallies. All input lines are limited to 80 columns. Alphabetic characters can be upper, lower, or mixed case. A $ (dollar sign) terminates data entry on a line. Anything on the line that follows the $ is interpreted as a comment. Blank lines are used as delimiters and as an optional terminator. Data entries are separated by one or more blanks.

The input file that contains the input information to describe the problem, and it has the following forms:

i. Cell cards.
ii. Surface cards.
iii. Material cards.
iv. Source Specification Cards.
v. Tally cards.

Cell and Surface Parameter Cards define values of cell parameters and used to specify relative cell importance in the sample problem. Material cards (Mn) are used to describe the material specification; the cards specify both the isotopic composition of the materials and the cross-section evaluations to be used in the cells. Complete specification of the geometry of the cell and surfaces boundary. Source Specification Cards are used to define starting particle for a point isotropic source. The tally cards are used to specify what we want to learn from the Monte Carlo calculation, cross sections, intensity, flux at a point, and energy.

3.3 Neutron source

Californium-252, an intense neutron emitting radioisotope, is a promising new source for radiography. A gram of ^{252}Cf emits 2.34 X 10^{12} neutrons per second by spontaneous fission.

^{124}Sb-Be is the only other isotopic source with a high neutron yield that has attracted interest for neutron radiography A comparison of the characteristics of ^{252}Cf and ^{124}Sb-Be sources with a yield of 5 x 10^{10} neutrons per second, which is adequate for radiography, is shown in Table 1.

Five thousand curies of ^{124}Sb are required for an emission rate of 5 x 10^{10} neutrons per second while only 11 curies of ^{252}Cf are needed. ^{124}Sb-Be inefficiently produces neutrons by the reaction of gamma decay of ^{124}Sb with beryllium. A well designed ^{124}Sb-Be source, such as described by Hennelly(3), will emit 10^7 neutrons per second per curie; in contrast, ^{252}Cf emits 4.4 x 10s neutrons per second per curie. The very small volume of the ^{252}Cf source provides essentially a point source of neutrons.

The comparatively large volume of the ^{124}Sb-Be source is due to the Large amount of beryllium required for an efficient source. The gamma dose rate from ^{124}Sb-Be is 5000 times greater than from ^{252}Cf. Some of the methods of detecting neutrons for radiography cannot be used in the presence of high levels of gamma radiation, such as is associated with ^{124}Sb-Be. Other detection methods become complicated and expensive because of the need for gamma discrimination or reduction. The high gamma radiation from ^{124}Sb-Be will require great attention to the protection of personnel during radiography. The magnitude of the gamma radiation is illustrated by this comparison. The dose rate a few inches from a typical 400 kVp industrial radiography generator is 12 R/minute; the gamma dose rate at two inches from a 5000 curie ^{124}Sb-Be source is 10^5 R/minute.

At a cost of $\$10^3$ per milligram, which is an order of magnitude estimate over the next ten years, the ^{252}Cf source offers the most favorable combination of initial investment plus replenishment costs. The initial costs of ^{252}Cf and ^{124}Sb-Be sources are approximately the same; however, the annual replenishment costs are much higher for the ^{124}Sb-Be source because of its shorter half-life.

Because a ^{252}Cf neutron source requires no target material, fabrication cost would be less than for the ^{124}Sb-Be Source.

The ^{252}Cf source requires no maintenance and can be transported easily to work areas or remote field locations. Also, the neutron emission from a ^{252}Cf source is constant over long periods of time, and it can be determined without supplementary measurements. [21]

Table 3.1: Characteristics of ^{252}Cf and ^{124}Sb-Be sources with yields of 5×10^{10} neutrons per second

Characteristics	**^{252}Cf**	**^{124}Sb – Be**
Curies	11	5000
Volume(cc)	< 1	200
Neutron energy (avg)	2.3 MeV	24 keV
Gamma Dose Rate (R/hr at 1 meter)	2.9	1.4×10^4
Half- life	2.6 years	60 days
Initial cost	20000 $	25000$
Annual replenishment cost	4000$	76000$

Isotopic neutron sources ^{241}Am-^{9}Be, with an activity of 5 Ci of ^{241}Am, with an emission of 1.1×10^7 n/s and ^{239}Pu-^{9}Be, with an activity of 33 Ci which emits 5.5×10^7 n/s [22]

In this work we can choosing ^{252}Cf source of neutron because ^{252}Cf neutron source requires no target material, fabrication cost would be less than other source of neutron and requires no maintenance and can be transported easily to work areas or remote field locations. Also, the neutron emission from a ^{252}Cf source is constant over long periods of time, and it can be determined without supplementary measurements.

3.3.1 Low-intensity neutron sources

Low-intensity neutron sources prepared from a mixture of an alpha-, Preparation and Uses of Low-Intensity Neutron Sources or high-energy gamma-emitting radioisotope and beryllium produce neutrons by the following reactions:

$$^{9}Be + \alpha \rightarrow n^{1} + {}^{12}C \quad (3\text{-}1)$$

$$^{9}Be + \gamma \rightarrow n^{1} + {}^{8}Be \quad (3\text{-}2)$$

Both reactions will take place in a Ra-Be source, since Ra^{226} decays by4.78-MeV α emission to short-lived daughters which emit gamma radiations with energy of greater than 1.63 MeV. Pu-Be and Po-Be sources produce neutrons only by the (α, n) reaction.

The Sb^{124} -Re source is apure (γ, n), or photo neutron, source. Each of these sources has several disadvantages which make them relatively undesirable. Sources made from Ra^{226} and Sb^{124} have a very high gamma level relative to neutron productivity. The Sb^{124} -Re source is a low specific activity of Ra^{226} and Pu^{240} necessitates physically large sources for comparable neutron output relative to the size of Po^{210} initiated sources.However, the short half-life of Po^{210} (138 d) requires frequent recalibration of the source.The same is true of sources made from 60-days Sb^{124}. [22]

Table 3.2: Comparison of The Characteristics of neutron Sources

Source Material	Radionuclide Half-life	Source Dimensions	Neutron Mission n/sc	Average Neutron Energy MeV	Gamma Intensity
^{210}Po-Be	138 d	0.5 × 0.5 in	5.7×10^{6}	4.5	< 100mr/hr
^{239}Pu-Be	24360 y	1.5 × 1.5 in	3.4×10^{6}	4.5	< 100mr/hr
^{241}Am-Be[a]	462 y	1 × 1 in	4.8×10^{6}	4.5	100 mr/hr
^{241}Am-B[a]	462 y	1.2 × 1.2 in	0.93×10^{6}	2.5	10mr/hr
^{226}Ra-Be[a]	1622 y	1 × 2 in	20.2×10^{6}	4	>100 R/hr
^{124}Sb-Be	60 d	1.6 × 1.6 in	0.4×10^{6}	0.035	> 100R/hr

3.4 Neutron Moderators

3.4.1 Light water

Water is an excellent neutron attenuator because of its large hydrogen content.It is not a good gamma ray attenuator because of its low electron density in a water shield the dose contributed by neutrons is largely due to fast neutrons.The 330mb thermal neutron capture cross-section of hydrogen gives only 2.2MeV gamma rays, so that water is relatively low in secondary gamma-ray production. When water is used as a shield material the problems of purification, containment, temperature, control, and corrosion must be addressed. [19].

3.4.2 Heavy water

Heavy water radiolysis is similar to that of light water, but it is necessary to recover the deuterium released in order to use it to generate heavy water for further use. Also, the reaction $^2D \rightarrow$(gamma, n) $\rightarrow ^1H$ depletes the deuterium. Isotopic pollution of heavy water results from atmospheric humidity and leakage or condensation of normal water, so that the surface should be covered with a pressurized inert dry gas. The irradiated heavy water has to be subjected to a scrubbing process to remove tritium. Tritium is generated in heavy water by the reaction $^2D \rightarrow$ (n, gamma) $\rightarrow ^3T$, and decays with the emission of beta particles of0.018MeV energy and with a 12.3y half-life [19].

3.4.3 Polyethylene

Is an industry name for a group of thermoplastics made up of ethylene polymers.Its high-strength, ductility, excellent chemical resistance, low water vapor permeability, low water absorption, and ease of processing contribute to it being the highest volume polymer in the world (Harper, 2000, p. 1.40). PE is used in creating bottles, house wares, toys, food containers, and garbage and grocery bags. The variation in the density of PE, such as very low-density PE (VLDPE), low-density PE (LDPE), linear low-density PE (LLDPE), and high-density PE (HDPE)[27].

3.5 Shielding Materials

3.5.1 Borated Polyethylene

5%Borated Polyethylene is typically used in neutron radiation shielding applications where low- and intermediate-levels of neutron flux are expected, but elevated temperatures are not. Its optimal boron content of 5% provides excellent attenuation of thermal neutrons, thus reducing the levels of capture-gamma radiation. It has high hydrogen content, making it an effective fast neutron radiation shield. 5% Borated Polyethylene is available in a wide range of form factors including slabs, bricks, cylinders, pellets, and other custom shapes, in this work we used it as a cylindrical shape[19]

3.5.2 Boron Carbide

Boron carbide is characterized by a unique combination of properties that make it a material of choice for a wide range of engineering applications. Boron carbide is used in refractory applications due to its high melting point and thermal stability; it is used as abrasive powders and coatings due to its extreme abrasion resistance; it excels in ballistic performance due to its high hardness and low density; and it is commonly used in nuclear applications as neutron radiation absorbent. [31]

3.5.3 Boric Acid

Boric acid is used as a product in the industry. It is also used as a raw material to produce boron compounds. Boric acid has triclinic crystal structure and its chemical composition is H_3BO_3 containing 56.3% B_2O_3.

In the nuclear industry natural boric acid (NBA) is used as soluble reactivity control agent. The dissolved boric acid is referred to as a soluble poison or chemical shim due to the high cross section for thermal neutron absorption (3840 barn) exhibited by ^{10}B isotope contained in the boric acid .[32,33]

3.6 Neutron Reflectors

3.6.1 Graphite

Besides its use as a moderator and reflector, graphite is used in thermal shields. It is a good shield material for slowing down fast neutrons and when mixed with boron carbide it is a very effective material for capturing neutrons. Boron-impregnated graphite has a very low residual radioactivity after irradiation. However, it is a very poor gamma-ray attenuator.

Itshigh temperature properties make it suitable forneutron shielding at high temperatures and high flux densities. Graphite is made from petroleum coke and coal tar pitch. Graphite logs in diameters up to 1.25m are manufactured by a hot extrusion process. It has an anisotropic structure so that its physical and mechanical properties are directional. Irradiation at temperatures below about 250°C results in the accumulation of strain energy (stored energy), which is released upon annealing at higher temperatures by a self-sustaining energy release during a temperature excursion[29].

3.6.2 Beryllium

As a structural material, beryllium (Be) is a light metal which has high tensile strength in comparison with aluminum. Be surface forms a thin oxidation film by interacting with air like aluminum, and it is highly resistant to corrosion in dry gases.

Beryllium's properties such as high thermal conductivity, good elevated-temperature mechanical properties for a light metal and high melting point make the material attractive for use in nuclear reactors. In particular, Be has been used as a reflector in a number of materials test reactors. The key nuclear properties of Be are its low atomic number, low atomic weight, low parasitic capture cross-section for thermal neutrons, readiness to part with one of its own neutrons (n, 2n), and good neutron elastic scattering characteristics.[29]

3.7 Moderation of Neutrons to Thermal Energy

The (n, α) reactions for boron are based on neutrons at thermal energies of 0.0253 eV. Most neutrons are not initially at thermal energies. Therefore, in order to slow down a neutron to thermal energy, the neutron must have numerous elastic scattering collisions in order to decrease its energy. The number, n, of elastic scattering collisions needed to slow a neutron from its initial energy, E_i to a lower energy, E_f, in a material of atomic weight A can be found by:[30]

$$n = \frac{1}{\xi} \ln\left(\frac{E_i}{E_f}\right) \quad (3\text{-}7\text{-}1)$$

$$\text{Where} \quad \xi = 1 + \frac{\alpha}{1-\alpha} \ln(\alpha) (3\text{-}7\text{-}2)$$

$$\text{Where} \quad \alpha = \frac{(A-1)^2}{(A+1)^2} (3\text{-}7\text{-}3)$$

The above equations show that materials with smaller A numbers are more effective at slowing down neutrons than materials with large A numbers as shown in Table 3.3.

Table 3.3: The average number, n, of elastic Scatters required to slow a neutron from 2.5 MeV to 0.025 eV

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Material	A			N
H	1	0	1	18.4
D	2	0.111	0.725	25.4
C	12	0.716	0.158	116.6
O	16	0.779	0.120	153.4

Therefore, water, heavy water, and high densityPolyethylenetypically used as moderating materials for neutrons. However, all of these materials, with the exception of heavy water which is often cost prohibitive, contain large quantities of hydrogen which might affect the detection system's ability to distinguish between hydrogen from the explosives material and hydrogen from the moderator.[30]

3.8 Neutron detectors

A neutron detector operates by detecting the signal generated when a neutron interacts with certain parts of the detector. Neutron detectors can be classified by how the detection process occurs. Because helium-3 has characteristics that made it effective for use in neutron detectors, it was considered a “gold-standard” for neutron detection. However, helium-3 is a rare material, and its production in the United States has been declining while its demand has been increasing, requiring helium-3 users to take action to reduce their consumption of the gas. [23]

3.8.1Using helium-3 in neutron detection applications

Beginning in the 1980s, when helium-3 became available to DOE's Isotope Development and Production for Research and Applications Program (Isotope Program) to sell, the characteristics that led. These detectors containing helium-3 to become, according to experts, the "gold standard" for neutron detection include:

- High neutron detection efficiency—the likelihood that a helium-3 neutron detector will absorb a neutron and produce a detection signal (Kouzes et al. 2009a);
- Good gamma radiation discrimination—the ability to minimize false positives by determining whether a signal is due to neutron radiation or gamma radiation (Kouzes et al. 2009)
- Nontoxicity—neutron detectors containing the nontoxic helium-3 gas do not pose a health hazard as a result of leaks of the gas (GAO 2011);and
- Low cost—before the shortage, helium-3 ranged in cost from about $40 to $85 per liter, so neutron detectors containing it were low or competitive in cost compared to alternatives. [23]

3.8.2Other Neutron Detectors

There are many types of other neutron detectors. Old type Boron (BF3) neutron detectors are hardly ever used anymore due to safety considerations (the BF3gas is highly toxic). They have been replaced by He-3 detectors. Neutron scintillators use a conversion plate made of a neutron absorbing material (mostly Gd2O3) that emits gammas upon neutron absorption. The gamma rays are then detected as any other photons would through the use of photomultipliers. Neutron scintillators are very sensitive to gamma ray background.

In this work we can choose He-3 because of high neutron detection efficiency, good gamma radiation discrimination and low cost. [25].

Table 3.4: Comparing a few characteristics for three types of neutron detectors. The B-10 and the He-3 types are proportional counters. The Li-6 type is a scintillator.

Detector type	**B-10(n,α)Li-7**	**He-3(n,p)T-3**	**Li-6(n, α)T-3**
Energy of Reaction	2.79 MeV	0.76 MeV	4.78 MeV
Charged particles energies	α =1.77 MeV Li =1.01 MeV	P= 0.57 MeV T=0.19 MeV	T=2.73 MeV α = 2.05 MeV
Particles range	α =3mm Li = 2mm	P = 30mm T =6mm	T = 0.04mm α =0.007mm
Emitted Gammas	0.48 MeV	None	None
Typical thickness	5mm	20mm	2mm
Atomic density	$.053*10^{20}$ cm^{-3}	$0.81*$ $10^{20}cm^{-3}$	$173*10^{20}cm^{-3}$
Absorption cross section at 5 A	10,67 Barn	14,83 Barn	2,62 Barn
Efficiency at A	3%	80 %	100%

3.9 Monte Carlo Modeling

In this work, MCNP code was used to simulate the performance of three moderator; high density polyethylene (HDPE), light water (H_2o) and heavy water (D_2O).The result in terms of moderated neutron flux are to be compared and will be used to select the optimal moderator. The performance of several borated materialssuch as; borated polyethylene (BPE), boric acid (BA) and boron carbide (BC) as shielding layers was explored using MCNP5.

The result in terms of neutron dose have to compare to determine the optimal structure for the proposed thermal neutron pump. Figure 3.1 shows sketched diagram for the proposed geometry for the thermal neutron pump while Figure 3.1 shows a snap shot for MCNP in action.

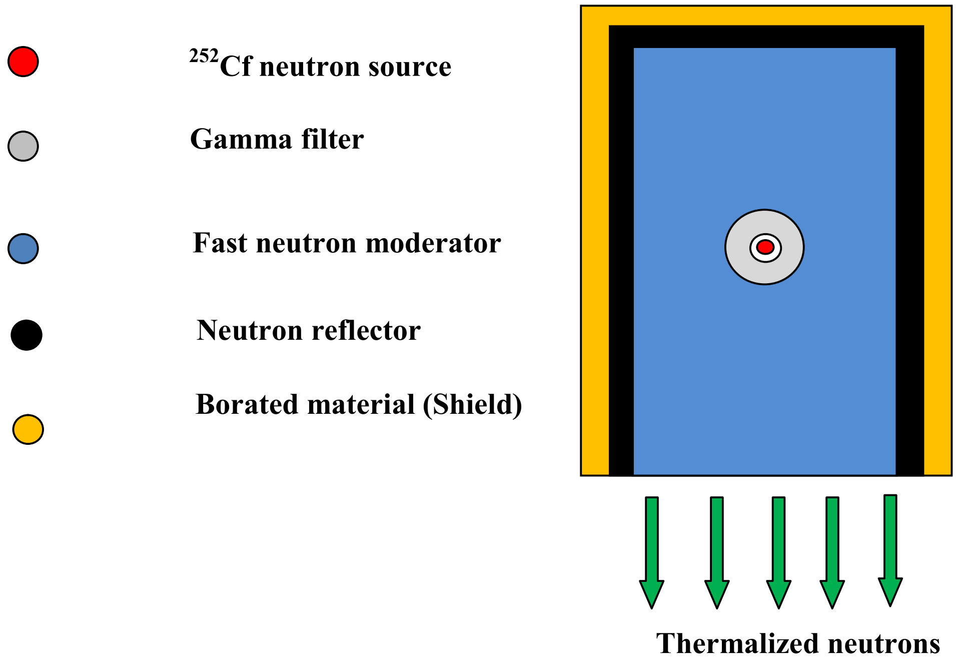

Fig. 3.1, sketched diagram for the propose geometry for the thermal neutron pump

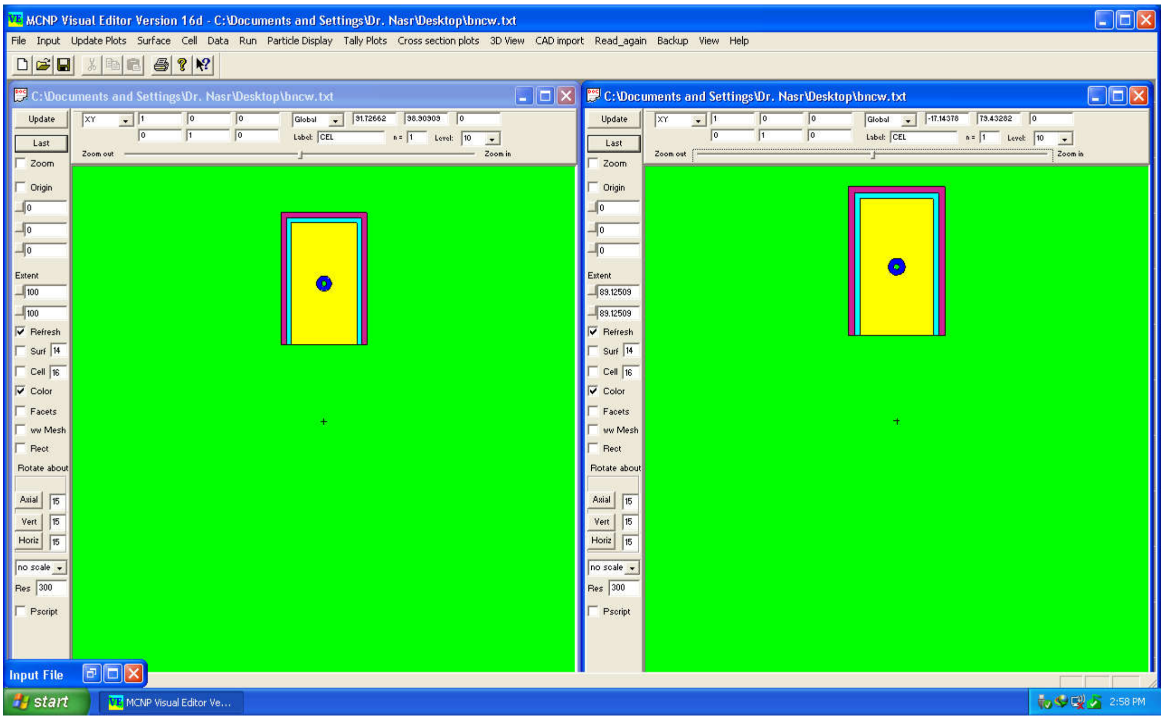

Fig 3.2, a snap shot for MCNP in action.

3.10 geometry description

The model consists of a point isotropic source with the neutron energy distribution characteristic of a fast neutron source, located inside a gamma filter made of sphere of high density lead, 1cm inner radius and 3 cm outer radius, which in turn located inside a fast neutron moderator in a form of a cylinder of radius 13cm, the moderator is surrounded by the outer radius is 15cm and the space between the two cylinders is filled with graphite for reflection. A third cylinder is added with a diameter of 17cm and filled with borated material to work as staff shielding. The height of the containing outer cylinder is 80cm.

3.11. Materials used in current study

Table 3.5 shows the composition and mass fractions of the materials used in the current MCNP simulations.

Table 3.5: Compositions of materials modeled in the MCNP simulations

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Mass fractions	Material-Density (g/cm^3)					
	HDPE 1.18	**BPE 0.955**	**H_2O 1**	**D_2O 1.11**	**C 2.23**	**Pb 2.7**
H	13.93	0.116	11.1			
C	83.29	0.0612			1	
N	0.20					
O	2.51	0.222	88.8	0.888101		
Na						
Ni						
Cu						
W						
I						
P/b		0.05				1
^{10}B						
^{11}B						
D				0.111898		

Chapter Four

Results and discussion

4.1 Introduction

This chapter is concerned on presenting the results of simulation for the geometry and discusses it in order to obtain the effective geometry of thermal neutron pump.

4.2 Moderator Material Modeled

The moderator materials which described previously in chapter (3) is examined using an MCNP input file to simulate their performance and the SigmaPlot software to analyze and plot the data. By using the specified input file which is attached in the appendix, the flux calculation was carried out in the thermal neutron range. The data were plotted for each moderator.

4.2.1 Heavy water (D_2O)

For heavy water, D_2O, as shown in figure 4.1, the results were plotted in terms of variations of thermal neutron flux with respect to neutron energy. The flux peaks at 0.025 eV with value 1.03×10^{-6}n.cm^{-2}.s^{-1}. Neutron flux 3.4×10^{9}n/s.cm^{2}

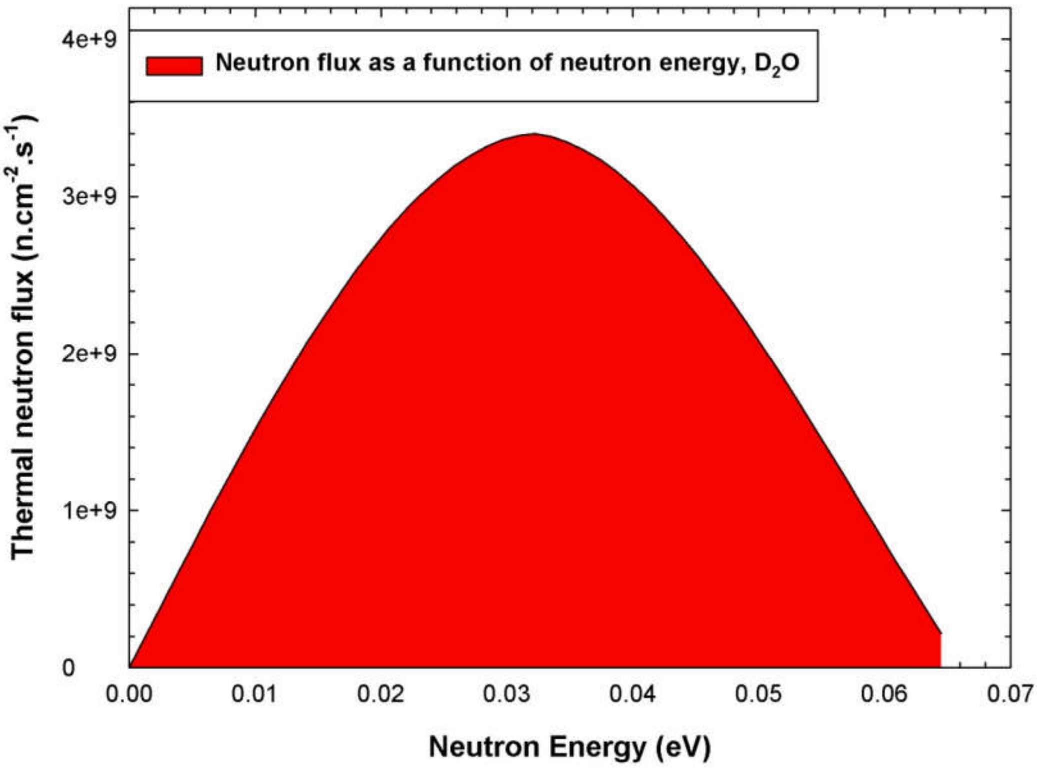

Fig 4.1: Neutron energy vs neutron flux for heavy water as a moderator

4.2.2 Light water (H_2O)

For light water, H_2O, as shown in figure 4.2, the results were plotted in terms of variations of thermal neutron flux with respect to neutron energy. The flux peaks at 0.025 eV with value 1.08×10^{-5}n.cm^{-2}.s^{-1}. Neutron flux 3.6×10^{8}n/s.cm^{2}

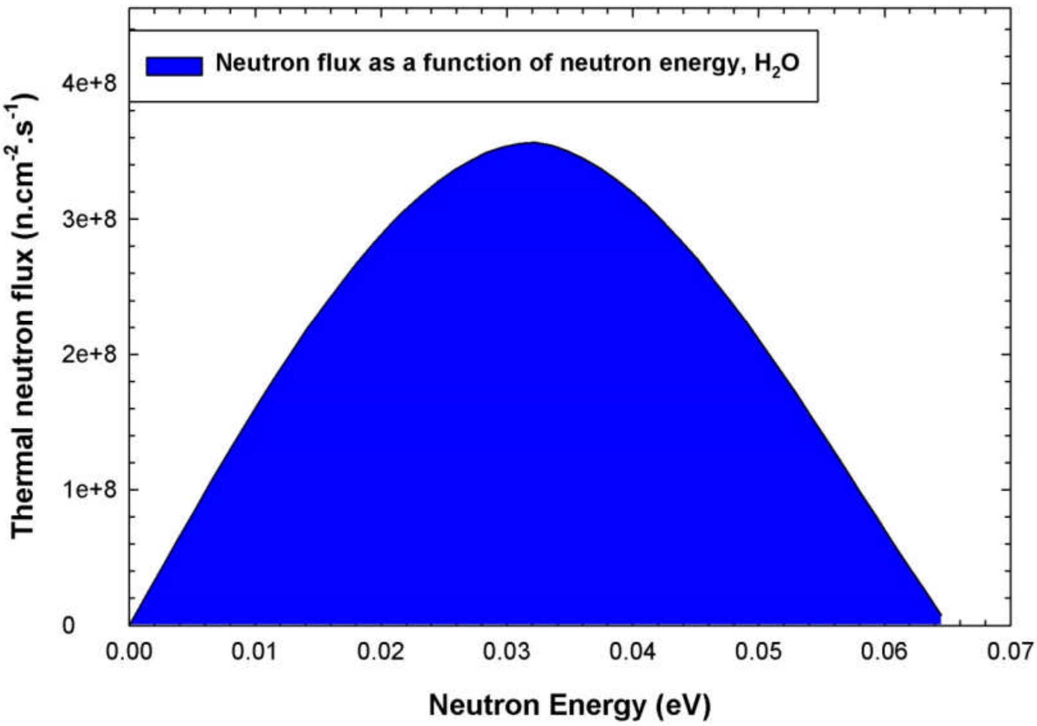

Fig 4.2: Neutron energy vs neutron flux for light water as a moderator

4.2.3 High Density Polyethylene (HDPE)

For high density polyethylene, HDPE, as shown in figure 4.3, the results were plotted in terms of variations of thermal neutron flux with respect to neutron energy. The flux peaks at 0.025 eV with value 8.21×10^{-6}n.cm^{-2}.s^{-1}. Neutron flux 2.7×10^{8} n/s.cm^{2}

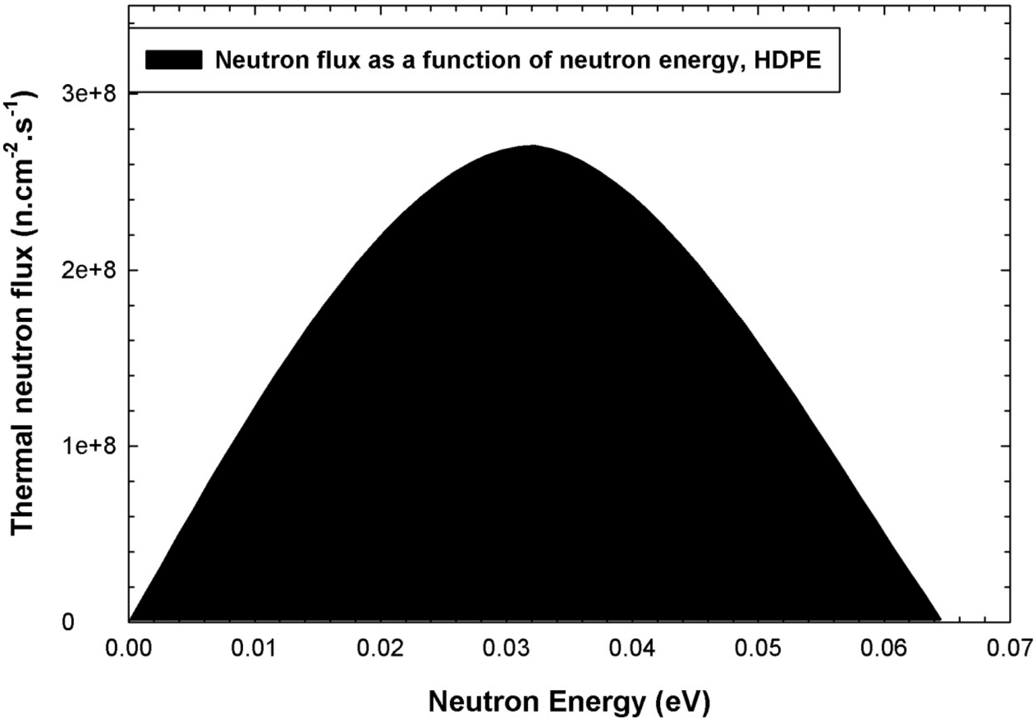

Fig 4.3: Neutron energy vs neutron flux for polyethylene as a moderator

4.2.4 Comparison

The three spectra for the moderators were compared and plotted in figure 4.4. As seen, the highest flux is obtained in case of D_2O, which exceeds that of H_2O by a factor of 9 and that of HDPE by a factor of 12. Based on these results D_2O was selected as the optimal moderator.

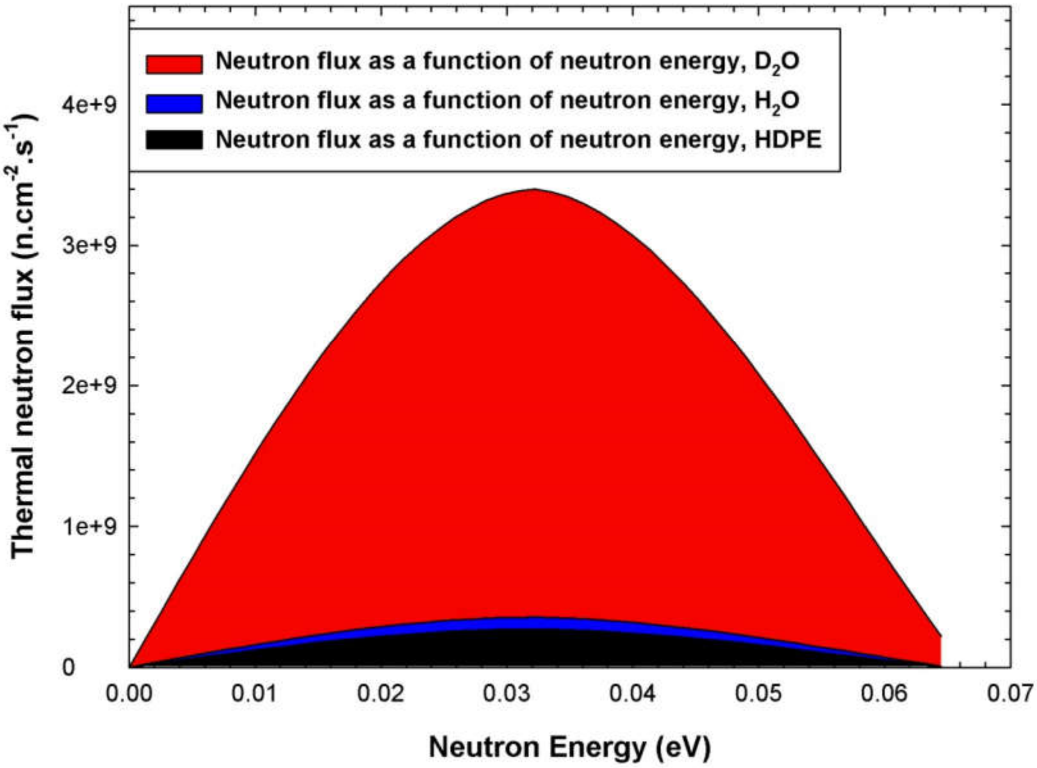

Fig 4.4: Neutron energy vs neutron flux for all moderators

4.3ReflectingMaterials

For beryllium , Be, and graphite , C , as shown in figure 4.5, the results were plotted in terms of variations of thermal neutron flux with respect to neutron energy. The flux peaks at 0.025 eV with value $1.17\times10^{-6}n.cm^{-2}.s^{-1}$, for beryllium and 1.03×10^{-6} $n.cm^{-2}.s^{-1}$, for graphite. Suggesting that an increasing in flux by about 13% in case of be. Beryllium was selected as the optimal reflector in our proposed system. The results were plotted in figure 4.5.

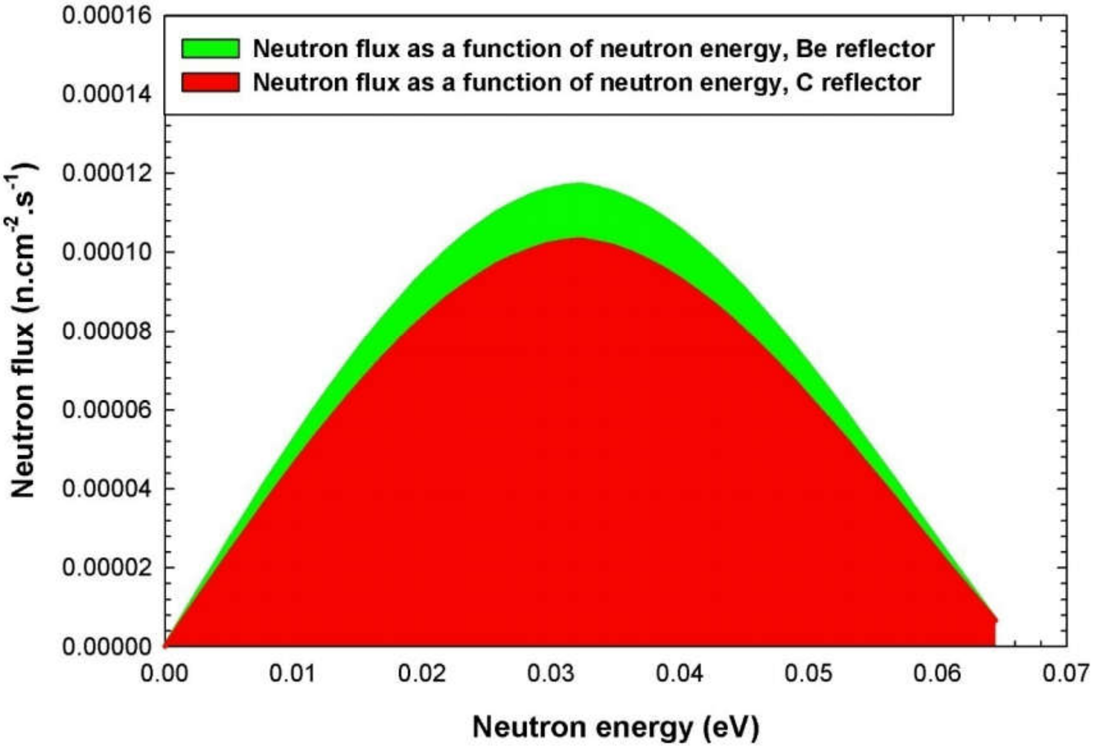

Fig 4.5: Neutron energy vs neutron flux for beryllium and graphite as a reflector

4.4 Shielding Material modeled

The effective dose was calculated by locating a ring detector of radius 100 cm around the shield while measuring the amount of doses for several shielding materials. The results are presented in the table 4.1. As shown, borated polyethylene shield provided the least effective dose and thus is selected as the optimal shielding material for the proposed system.

Table 4.1 The effective dose measurement and operating time estimation

Shield	Dose		Operating time	
	mSv/h	mSv/yr	Hours	Years
Borated Polyethylene	$4.5*10^{-3}$	8.64	4444.44	2.315
Boric Acid	$5.2*10^{-3}$	9.984	3846.2	2.003
Boron Carbide	$1.2*10^{-2}$	23.04	1666.67	0.868

Discussion

the results confirm the suitable of the proposed model as a thermal neutron pump ,achieved are in good accordance with the required standards in terms of sufficient neutron flow for use in BNCT which is estimated to be approximately $3.4\times10^{9}n.cm^{-2}.s^{-1}$this is due to the optimal selection of materials utilized in the simulation .

the following table presents the previous result compared to the current one confirm that what we have achieved is one of the best results due to the optimal selection of materials used in the design of the thermal neutron pump , as demonstrated by the simulation results and the following tables shows that

Table 4.2 the previous studies

Name	**neutron source**	**Gamma filter**	**Fast neutron moderator**	**Neutron reflector**	**material (Shield)**	**Neutron flux n/cm^2 s**
YaserKasesaz	^{252}Cf	Cd	Al	Pb	Bi	0.65×10^{9}
J C Yanch	^{252}Cf	Al	D_2O	Li	An	
C. A. Perks	^{252}Cf					2×10^{7}

Table 4.3 the current study

Name	neutron source	Gamma filter	Fast neutron moderator	Neutron reflector	material (Shield)	Neutron flux n/cm^2 s
Tahani	^{252}Cf	Pb	D_2O	C	HDPE	3.4×10^9

^{252}Cf neutron source requires no target material, fabrication cost would be less than other source of neutron and requires no maintenance and can be transported easily to work areas or remote field locations. Also, the neutron emission from a ^{252}Cf source is constant over long periods of time, and it can be determined without supplementary measurements.

Lead is the best material used for gamma ray filtration because it is characterized by a very large absorption coefficient of gamma ray , and its standard dimensions are used to prevent radiation about 1cm .

Graphite Besides its use as a moderator and reflector, graphite is used in thermal shields. It is a good shield material for slowing down fast neutrons and when mixed with boron carbide it is a very effective material for capturing neutrons. Boron-impregnated graphite has a very low residual radioactivity after irradiation. However, it is a very poor gamma-ray attenuator.

In this work, MCNP code was used to simulate the performance of three moderator; high density polyethylene (HDPE), light water (H_2o) and heavy water (D_2O).The result in terms of moderated neutron flux are to be compared and will be used to select the optimal moderator The three spectra for the moderators were compared, the highest flux is obtained in case of D_2O, which exceeds that of H_2O by a factor of 9 and that of HDPE by a factor of 12. Based on these results D_2O was selected as the optimal moderator.

The performance of several borated materials such as borated polyethylene (BPE), boric acid (BA) and boron carbide (BC) as shielding layers was explored using MCNP5.borated polyethylene shield provided the least effective dose and thus is selected as the optimal shielding material for the proposed system.

The result in terms of neutron dose has to compare to determine the optimal structure for the proposed thermal neutron pump

For beryllium, Be, and graphite,C, the results were plotted in terms of variations of thermal neutron flux with respect to neutron energy.. Suggesting that an increasing in flux by about 13% in case of Be. Beryllium was selected as the optimal reflector in our proposed system.

Finally we may say that an effective and reliable system for boron neutron capture therapy has been modeled.

Chapter Five
Conclusion and Recommendations

5.1 Conclusions

In this work, MNCP simulation was carried out to model a thermal neutron system suitable for boron neutron capture therapy. The results in comparison to the targeted previous studies:, clearly proofed that our proposed system provide sufficient thermal neutron flux$3.4\times10^{9}n.cm^{-2}.s^{-1}$. Furthermore the effective dose at 100cm away from the system was calculated to be (8.64mSv). Which suggest a working hours of 2.5years before it reaches the permissible dose 20mSv.Finally we may say that an effective and reliable system for boron neutron capture therapy has been modeled .

5.2 Recommendations

As recommendations, we may say that further experimental verification should be carried out to realize the results for the current MCNP simulations. Further efforts are needed to study the performance of other different neutron moderators, reflectors and shielding materials.

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