

بسم الله الرحمن الرحيم

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Monte Carlo Modeling of Effective Shielding Systems for Fast Neutrons

إستخدام محاكاة مونتى كارلو لتصميم نظام تدريع فعال للوقاية من النيوترونات السريعة

Thesis submitted to the College of Graduate Studies in fulfillment of the requirements of the degree of M.Sc. in Physics

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الآية

بسم الله الرحمن الرحيم

قال تعالى:

شَهِدَ اللَّهُ أَنَّهُ لَا إِلَهَ إِلَّا هُوَ وَالْمَلَائِكَةُ وَأُولُو
الْعِلْمِ قَائِمًا بِالنَّقِطِ لَا إِلَهَ إِلَّا هُوَ الْعَزِيزُ
الْحَكِيمُ

صدق الله العظيم

سورة ال عمران الآية 18

DEDICATION

This thesis is dedicated to
My great mother and my dear
father
My lovely... Brother and sister
And To all my teachers and
friends

ACKNOWLEDGEMENTS

Firstly thank Allah Almighty and I would like to express my thanks and gratitude to my supervisor Dr. Nassreldeen Abdelrazig Abdelbari Elsheikh. Last, but not least I wish to extend my thanks to Sudan University of science and technology, department of physics.

ABSTRACT

In this thesis, we have designed a shielding system consisting of dual layers to shield fast neutrons. Monte Carlo simulation code (MCNP5) was used to simulate the spectra of ^{252}Cf -neutron source and to verify the performance of several moderating materials and absorbers as well. The performance of high density polyethylene (HDPE) as moderating material was examined against different thermal neutron absorbers such as; borated polyethylene (BPE), Boric acid (H_3BO_3) and Boron carbide (B_4C).

The optimal configuration was found to be (20cm HDPE +5cm B_4C) since it has reduced the source energy by 88% and the intensity by 99%. To confirm the stability of the optimal configuration, its performance has been examined against two moderating materials such as; heavy water (D_2O) and Beryllium (Be). It is found that the configuration (20cm HDPE +5cm B_4C) is still dominant and has reduced neutron intensity with respect to configuration (20cm D_2O +5cm B_4C) by approximated 33% and with respect to configuration (20cm Be+5cm B_4C) by approximated 18%.

مستخلص

في هذه الاطروحة قمنا بإستخدام برنامج محاكاة مونتى كارلو لتصميم نظام تدريع ثنائى الطبقات للوقاية من النيوترونات السريعة. تم إستخدام البرنامج لمحاكاة طيف مصدر النيوترونات كاليفورنيوم-252 ثم لدراسة فعالية عدد من مواد توهين النيوترونات السريعة وكذلك عدد من المواد التي تتميز بمساحة مقطع امتصاص عاليه للنيوترونات الحرارية.

تمت دراسة فعالية البولى إثلين على الكثافة كمادة موهنه للنيوترونات السريعة كطبقة اولى في وجود ثلاث مواد مختلفة ذات مساحة مقطع امتصاص عالي للنيوترونات الحرارية مثل: البوريتد بولى اثلين ،البوريك اسيد والبورون كربايد.

اثبتت الدراسة ان افضل نموذج للتدريع هو (البولى اثلين 20سم +البورون كربايد 5سم) حيث قلل طاقة نيوترونات المصدر بنسبة 88% وكثافة النيوترونات بنسبة 99%

للتأكد ثبات أداء نموذج التدريع المقترح، قمنا باختبار أداءه في وجود مواد توهينيه اخري مثل الماء الثقيل والبرليوم. أثبتت الدراسة فعالية وافضلية التصميم (البولى اثلين 20سم +البورون كربايد 5سم)، حيث قلل شدة النيوترونات بنسبة 33% تقريبا مقارنة بالتصميم (الماء الثقيل 20سم +البورون كربايد 5سم)، وبنسبة 18% تقريبا مقارنة بالتصميم (البرليوم 20سم +البورون كربايد 5سم).

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CHAPTER ONE

INTRODUCTION

CHAPTER ONE

INTRODUCTION

1.1 Introduction

Day by day radiation protection becomes more and more important topic to be investigated in nuclear science, [1]. Protection from radiation depends on three basic methods used to reduce the external radiation hazards; time, distance, and shielding. Good radiation protection practices require optimization of these fundamental techniques, [2].

Radiation shielding materials are important infrastructure components in nuclear facilities such as nuclear reactors and particle accelerators, especially in mobile nuclear devices and manned spacecrafts. With respect to such facilities, the limits of space and requirements of portability, the weight and volume of radiation shielding system are restricted. Thus radiation shielding materials must be of light-weight and high effective shielding properties, [3].

Usually shielding materials have to service for long time in quite inclement environment with high temperature, corrosion, and so on. Furthermore, shielding materials could be damaged by constant radiation due to interaction with each other, [4] Therefore radiation shielding materials must have multiple endurances in many environments including radiation effects, heat, etc. Certainly, the cost of material is also an important requirement to be considered in the research. And then efforts must be made to reduce the exposure of employees in nuclear facilities on basis of the as-low-as-reasonably-achievable (ALARA) principle,[5].

1.2 Statement of the problem

Radiation protection systems against fast neutrons are essential to reduce the dose to the radiological workers and to common individuals as well. This research investigates the possibility of designing a protection system for fast neutrons using Monte Carlo computer code version 5 (MCNP5). The MCNP5 is used to model the geometry of the proposed system and examine its performance in reducing neutron energy and intensity as well.

1.3 Objectives of the research

The objective of the research is to develop a dual-layer shielding system for fast neutrons. This is done using MCNP5 to model the proposed system, examine the performance of different fast neutron moderating materials and thermal neutron absorbers. The MCNP code is used to simulate the performance of the proposed system and investigate the optimal configuration and operating conditions.

1.4 Hypothesis of research

The hypothesis is based on the fact that hydrogenous materials such as polyethylene are effective moderators due to the presence of hydrogen nuclei. The elastic scattering reaction is the dominant mechanism by which fast neutrons lose energy in low Z materials such as hydrogen. On the other hand, borated materials such as, borated polyethylene, boric acid and boron carbide are effective thermal neutron absorbers. This is because of the high absorption cross section of ^{10}B for thermal neutrons. Consequently, it is reasonable to develop a shield consisting of two layers, one work as fast neutron moderator to thermalize fast neutrons, while the second layer works as thermal neutron absorber. We believe that such shielding systems are promising and effective.

1.5 Significance of the research

The significance of this thesis lies on developing an effective shielding system for fast neutrons consisting of dual layers. Furthermore, the current work provides deeper view behind using hydrogenous materials as moderators and borated materials as thermal neutron absorbers. The proposed shielding system may be applied in Nuclear power plants, particle accelerators, Medical facilities and industrial research applications.

1.6 Limitations of the study

The research involved in this work is on simulation scale using MCNP5 code. Further experimental verification is required to examine the performance of the proposed shielding system on laboratory scale.

CHAPTER TWO

NEUTRON SHIELDING

CHAPTER TWO

NEUTRON SHEILDING

2.1 Introduction

Radiation is energy transmitted through space in the form of electromagnetic waves or particles. Radiation is usually classified into two categories: ionizing and non-ionizing radiation, [6]. While crossing a medium, radiation undergoes various interactions depositing energy in the medium. This deposition of energy is characterized as radiation dose, and if it occurs in living tissues of Individuals, the endpoint effects will be biological changes most of which are undesirable.

The mechanisms by which different radiations interact in an absorbing medium are fundamental to describing the amount of deposited energy, how the characteristics of shields or other absorbers modify and affect radiation exposure and dose, and the design of detectors to measure the various types of radiation based on their respective interaction principles, [7].

2.2 Radiation Dose and Units

In all fields where there is need for quantitative measurements, it is necessary to have understandable and precise quantities and units. Practically all countries use the SI system (from French: The International System dU' nites), [8]. In the field of radiation dosimetry and radiation protection, two other international organizations are active in relation to quantities and units.

The International Commission on Radiation Units and Measurements (ICRU) which is mainly working with the physical aspects of dosimetry and the International Commission on Radiological Protection (ICRP) which mainly works

with assessments and quantification of the biological effects of radiation and provides recommendations and guidance on all aspects of radiation protection against ionizing radiation, [9] The measured radiation to one of the following units:

2.2.1 Absorbed dose (D)

Absorbed dose describes the amount of radiation absorbed by an object or person (that is, the amount of energy that radioactive sources deposit in materials through which they pass), [10]. Given the unit Gray (Gy), [9].

$$D = \frac{1}{m} \int E dV \quad (2.1)$$

2.2.2 Exposure dose

Exposure describes the amount of radiation traveling through the air. Many radiation monitors measure exposure. The units for exposure are the roentgen (R) and coulomb/kilogram (C/kg), [10].

2.2.3 Equivalent dose (H)

When living matter absorbs radiation, the radiation can produce a biological effect. Since different types of ionizing radiation vary in how they interact with biological materials, absorbed doses of equal value do not necessarily have equal biological effects. For example, 1 Gy of alpha radiation is more harmful to tissue than 1 Gy of beta radiation. This is because an alpha particle is more heavily charged and deposits its energy much more densely along its path, [10]. Radiation weighting factor (w_R) is used to equate different types of radiation with different levels of biological effectiveness. The Unit 1 Sv = ($w_R \times Gy$), [9].

$$H_T = \sum_R W_R D_{T,R} \quad (2.2)$$

2.2.4 Effective dose (E)

Effective dose E Sum of all equivalent doses weighted with the weighting factor w_T for tissue T the Unit (Sv), [10].

$$E = \sum_T W_T H_T \quad (2.3)$$

2.3 Ionization radiation

Ionizing radiation is radiation with enough energy so that during an interaction with an atom, it can remove tightly bound electrons from the orbit of an atom, causing the atom to become charged or ionized, Ionizing radiation is part of the nature and of human activities in medicine, research, industry, energy production and military application, [11].

2.4 Types of ionizing radiation

- i. **Directly ionizing radiation** Composed of high-energy charged particles which ionize atoms of the material due to Coulomb interaction with their electrons, e.g., electrons and positrons (beta radiation), protons, (alpha radiation) and nuclear fragments.
- ii. **Indirectly ionizing radiation** Composed of neutral particles which ionize atoms due to interactions of those particles with matter e.g. high energy EM waves (Ultraviolet, X-ray Gamma radiation) and neutrons of any energy.

2.5 Biological effect of ionizing radiation

The fact that ionizing radiation produces biological damage has been known for many years. The first case of human injury was reported in the literature just a few months following Roentgen's original paper in 1895 announcing the discovery of x-rays. As early as 1902 the first case of x-ray induced cancer was reported in the literature, [12]. Ionizing radiation can cause damage to living cells and tissues. Its effects depend on the intensity of the radiation, exposure time and the kind of body cells affected, [13]. Deterministic effects and stochastic effect: From the biological effects of radiation on human body radiation effects are generally divided into two categories:

- i. **Deterministic effects:** Based on a large number of experiments involving animals and other researches, it was discovered that severity of certain effects on humans depends on a certain level the (threshold) below which the effect will be absent.
- ii. **Stochastic effects:** The severity of stochastic effects is independent of the absorbed dose. Under certain exposure conditions, the effects may or may not occur, Figure (2.1) presents the Steps of the biological action of the radiation.

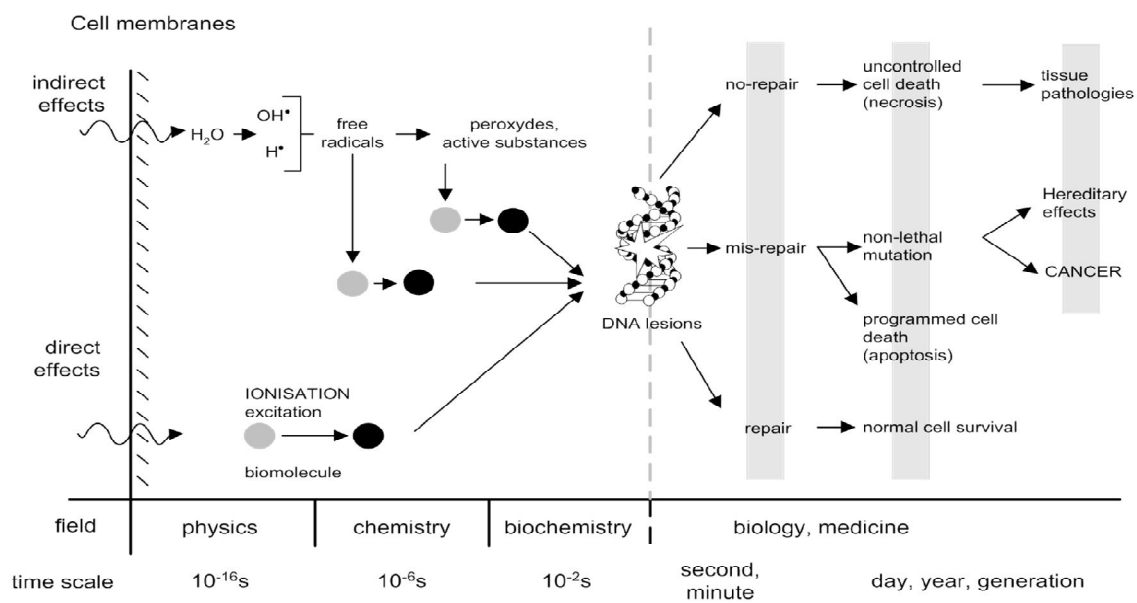


Fig (2.1) Steps of the biological action of the ionizing radiation

2.6 Radiation protection of ionizing radiation

Sources of radiation are features of the environment. Radiation and radioactive substances have many beneficial applications, ranging from power generation to uses in medicine, industry and agriculture. The radiation risks to workers and the public and to the environment that may arise from these applications have to be assessed and, if necessary controlled.

Radiation protection, sometimes known as radiological protection, is the science and practice of protecting people and the environment from the harmful effects of ionizing radiation,[8] Fundamental to radiation protection is the reduction of expected dose and the measurement of human dose uptake. For radiation protection and dosimetry assessment the International Committee on Radiation Protection (ICRP) and International Commission on Radiation Units and Measurements (ICRU) have published recommendations and data which can be used to calculate the biological effects on the human body, and set regulatory and guidance limit. IAEA definition of radiation protection the protection of people from the effects of ionizing radiation, and the means for achieving this, [14]

- i. Radiation Protection Training
- ii. Assessment of radiological risks at work places
- iii. Area monitoring
- iv. Individual monitoring of personnel
- v. Control and characterization of radioactive material and waste
- vi. Management of radioactive sources and waste
- vii. Assessment of radiological risks related to new projects

Radiation protection can be divided into occupational radiation protection, which is the protection of workers, medical radiation protection, which is the protection of

patients, and public radiation protection, which is protection of individual members of the public, and of the population as a whole. The types of exposure, as well as government regulations and legal exposure limits are different for each of these groups, so they must be considered separately, [8].

2.6.1 General principles of Radiation protection

- i. **Justification of practice** No practice should be authorized unless the practice produces sufficient benefit to the exposed individuals to offset the radiation harm that it might cause.
- ii. **Optimization of protection** In relation to exposures from a source within a practice, protection and safety shall be optimized in order that the magnitude of individual doses kept as low as reasonably achievable.
- iii. **Dose limitation** The normal exposure of individuals shall be restricted so that neither the total equivalent dose to relevant organs or tissues exceeds any relevant dose limit specified (20 mSv for workers).

2.6.2 ALARA principle

ALARA (As Low As Reasonably Achievable) is a safety principle designed to minimize radiation doses and releases of radioactive materials. More than merely best practice, ALARA is predicated on legal dose limits for regulatory compliance, and is a requirement for all radiation safety programs, [15]. The basis Current radiation safety philosophy is based on the conservative assumption that radiation dose and its biological effects on living tissues are modeled by a relationship known as the (Linear Hypothesis). The assertion is that every radiation dose of any magnitude can produce some level of detrimental effects which may be manifested as an increased risk of genetic mutations and cancer. Thus, the NCSU radiation safety program attempts to lower doses received by radiation workers by utilizing practical, cost effective measures, [16]. An effective ALARA program requires a commitment from all relevant staff in your clinic or hospital: veterinarians and vet techs as well as any other personnel who work in proximity to your radiology equipment. To maintain doses As Low as Reasonably Achievable, make sure staff follows these three major safety principles, [15].

i. Time

The amount of radiation an individual accumulates will depend on how long the individual stays in the radiation field because.

$$\text{Dose} = \text{Dose Rate} \times \text{Time} \quad (2.4)$$

Therefore to limit a person's dose one can restrict the time spent in the area. How long a person can stay in an area without exceeding a prescribed limit is called the (stay time) and is calculated from the simple relationship

$$\text{Stay Time} = \text{Exposure Limit/Dose Rate} \quad (2.5)$$

ii. Distance

The amount of radiation an individual receives will also depend on how close the person is to the source, [17]. The Inverse Square Law - Point sources of x-ray and gamma radiation follow the inverse square law which states that the intensity of the radiation (I) decreases in proportion to the inverse of the distance from the source (d) squared

$$I \propto \frac{1}{d^2} \quad (2.6)$$

This can be rewritten

$$I = K \frac{1}{d^2} \quad (2.7)$$

Where: K is a constant of unknown value

iii. Shielding

Radiation shielding of personnel and the public at the work place means placing a suitable material between the radioactive source and the personnel or public. The radiation is attenuated and the effect may be completely eliminated or reduced to an acceptable level. Some materials are more effective than others in shielding particular type of radiation, the type and amount of shielding material needed for shielding will vary with the type and quantity of radioactive material being shielded, [18].

When reducing the time or increasing the distance may not be possible one can choose shielding material to reduce the external radiation hazard. The proper material to use depends on the type of radiation and its energy, [17].

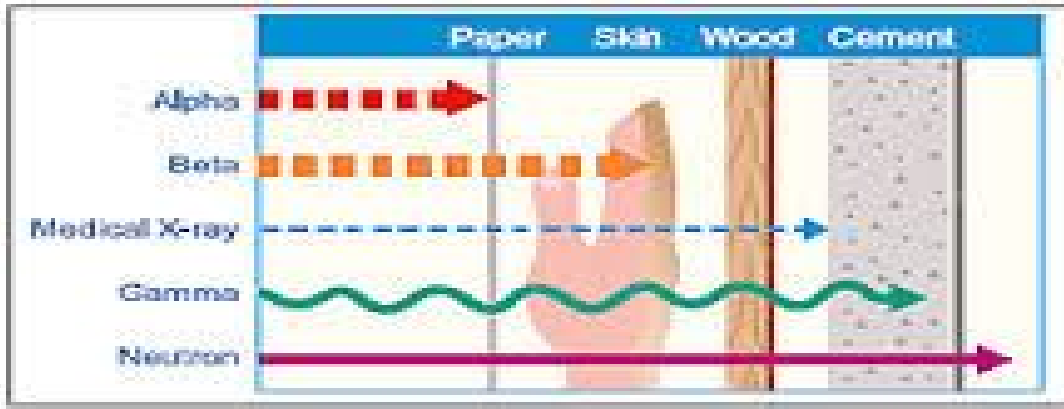


Fig (2.2) penetrating power of different kinds of ionizing radiation

Alpha particles are easily shielded. A thin piece of paper or several cm of air is usually sufficient to stop them. Thus, alpha particles present no external radiation hazard. Beta particles are more penetrating than alpha particles. Beta shields are usually made of aluminum, brass, plastic, or other materials of low atomic number to reduce the production of bremsstrahlung radiation see figure (2.2) Monoenergetic x-ray or gamma rays collimated into a narrow beam are attenuated exponentially through a shield according to the following equation:

$$I = I_0 e^{-\mu x} \quad (2.8)$$

Where: I = is the intensity outside of a shield of thickness x

I_0 = is the unshielded intensity

μ = is the linear attenuation coefficient of the shielding material

x = is the thickness of shielding material.

The linear attenuation coefficient is the sum of the probabilities of interaction per unit path length by each of the three scattering and absorption processes photoelectric effect, Compton effect, and pair production. Note that μ has dimensions of inverse length (1/cm). The reciprocal of μ is defined as the mean free path, which is the average distance the photon travels in an absorber before an interaction takes place. Because linear attenuation coefficients are proportional to the absorber density, which usually does not have a unique value but depends somewhat on the physical state of the material it is customary to use the mass attenuation coefficient which removes density dependence, [17].

$$\mu_m = \frac{\mu}{\rho} \quad (2.9)$$

Where: μ = Mass attenuation coefficient
 ρ = density (g/cm³)

Using the mass attenuation coefficient instead of the linear attenuation coefficient, the attenuation equation can be rewritten:

$$I = I_0 e^{-\mu_m \rho x} \quad (1.10)$$

2.6.3 Half Value Layer

The half value layer (HVL) is the thickness of a shielding material required to reduce the intensity of radiation at a point to one half of its original intensity, [17]. It can be calculated by setting $I = \frac{1}{2} I_0$ and solving the attenuation equation for x :

$$0.5 = e^{-\mu x_{1/2}} \quad (2.11)$$

$$x_{1/2} = \frac{\ln 0.5}{\mu} = HVL \quad (2.12)$$

When the HVL is known rather than μ the total attenuation from n half value layers can be calculated by using the following equation:

$$I = \frac{I_0}{2^n} \quad (2.13)$$

2.7 The Neutron

Neutron neutral elementary particle with a rest mass of $1.674\,92716 \times 10^{-27}$ kilogram and spin $\frac{1}{2}$ and no charge particle classified as a baryon in the nucleus of an atom it is stable but when free it decays, [19]. The neutron is composed of three quarks one up quark and two down quarks, [20]. The discovery of the neutron in physics that occurred in the first half of the 20th century. Discovered by James Chadwick in 1932, [8] Neutron radiation can be produced from nuclear fission, which occurs only for certain nuclear substances with a high atomic number, such as uranium and plutonium. Except for several fission fragments with very short half-lives, and californium-252 which undergoes spontaneous fission, there are no other radioisotopes that emit neutrons. Other neutron sources depend on nuclear reactions for the emission of neutrons, [6]. Neutrons are able to penetrate tissues and organs of the human body when the radiation source is outside the body.

2.7.1 Neutron source

Free neutrons are unstable, although they have the longest half-life of any unstable sub-atomic particle by several orders of magnitude. Their half-life is still only about 10 minutes however so they can be obtained only from sources that produce them freshly. Various neutron generators depend on irradiation of a target material, [7]. Neutron is usually classified into two categories Natural neutron background: small natural background flux of free neutrons exists everywhere on Earth. In the atmosphere and deep into the ocean, the (neutron background) is caused by muons produced by cosmic ray interaction with the atmosphere. Sources of neutrons for research: these include certain types of radioactive decay (spontaneous fission and neutron emission), and from certain nuclear reactions.

Convenient nuclear reactions include table down reactions such as natural alpha and gamma bombardment of certain nuclides, often beryllium or deuterium, and induced nuclear fission, such as occurs in nuclear reactors. In addition, high-energy nuclear reactions (such as occur in cosmic radiation showers or accelerator collisions) also produce neutrons from disintegration of target nuclei. Small (tabletop) particle accelerators optimized to produce free neutrons in this way, are called neutron generators, [8].

Table (2.1) Neutron sources by production mode and energy, [7]

source	reaction	Energy range	Average energy (MeV)
---------------	-----------------	---------------------	---------------------------------

Sb ₁₂₄ – Be	(γ ,n)	[a]	0.024
Y ₈₈ – Be	(γ ,n)	[a]	0.16
Na ₂₄ – D ₂ O	(γ ,n)	[a]	0.22
Y ₈₈ – Be	(γ ,n)	[a]	0.31
Y ₈₈ – Be	(γ ,n)	[a]	0.83
Fission	(n,n)	0–8 MeV	2
H ₂ –H ₂ (D-D)	(d,n)	[a]	3.27
Ra ₂₂₆ – Be	(α ,n)	0–8 MeV	5
Pu ₂₃₉ – Be	(α ,n)	0–8 MeV	4.5
Cf ₂₅₂	Spontaneous fission	0–10 MeV	2.3
H ₂ –H ₃ (D-T)	(d,n)	[a]	14.1

[a] Essentially monoenergetic depending on self-absorption in the source

The deuterium–tritium (D-T) generator is a straight forward source of high-energy neutrons in which a tritium target is bombarded by deuterons that have been accelerated to about 200 keV. The reaction is exoergic and releases 17.6 MeV, of which 14.1 MeV is given to the ejected neutron. The neutrons so produced are monoenergetic and are ejected isotropically which causes the flux to fall off as the square of the distance.

Cyclotron-produced neutrons are of high energy and are produced by accelerating high-energy deuterons onto a beryllium target. Neutrons released in the ${}^9\text{Be}(\text{d},\text{n}){}^{10}\text{B}$ reaction are peaked in the direction of the deuteron beam and, depending on the number of deuterons in the incident beam, can produce intense focused beams of neutrons. The neutrons are not, however, monoenergetic but are distributed around a peak energy.

Photoneutron sources yield monoenergetic neutrons which can be quite useful for many research purposes, calibration of instruments, and neutron dosimetry. Alpha-neutron sources consist of radium, polonium, and plutonium intermixed with beryllium which, because of the Q-value of the reaction yields specific-energy neutrons; however, self-absorption of the alpha particles in the source causes the neutrons to be distributed about an average value that is usually several MeV see table (2.1) Such sources are always a compromise between a desired source lifetime and the amount of alpha-emitting material that must be mixed with beryllium to obtain the desired source strength and energy range.

Spontaneous fission neutron sources are also available with ^{252}Cf being the most common and practical source because of its yield of 2.3×10^{12} n/s per g (4.3×10^9 per Ci). The neutrons are emitted with a spectrum of energies from thermal up to several MeV with an average energy of about 2.3 MeV.

The half-life of ^{252}Cf is 2.638 y and transformation occurs by fission about 3% of the time and by alpha particle emission about 97% of the time. Most neutron sources produce energetic neutrons see Table (2.2) that are termed “fast” neutrons because of their velocities at these energies. Although neutrons are born fast they quickly undergo various interactions with media and their energy is degraded. Neutrons are best classified in terms of their energies into the following broad groups:

- i. Cold ($T < 20^\circ \text{M}$) < 0.0253 eV.
- ii. Thermal: 0.0253 eV.
- iii. Epithermal: 0.0253 to ~ 1 eV (energies corresponding to room temperature and extending up to the sharp absorption cutoff by a cadmium absorber).
- iv. Epicadmium (neutrons that are transparent to a cadmium absorber (strong resonance capture of about 20,000 b at about 0.4 eV)) > 1 eV.
- v. Slow: 0.0253–100 eV.
- vi. Intermediate: $0.5\text{--}10^4$ eV.
- vii. Fast: 0.01–10 MeV.
- viii. High energy > 10 MeV.

2.7.2 Neutron cross section

For neutrons of a fixed energy, the probability per unit path length's a constant for any one of the interaction mechanisms. It is conventional to express this probability in terms of the cross section σ per nucleus for each type of interaction. The cross section has units of area and has traditionally been measured in units of the barn (10^{-28} m^2). For example, each nuclear species will have an elastic scattering cross section, a radioactive capture cross section, and so on, each of which will be a function of the neutron energy. When multiplied by the number of nuclei N per unit volume, the cross section σ is converted into the macroscopic cross section Σ , [21].

$$\Sigma = N \sigma \quad (2.14)$$

Which now has dimensions of inverse length Σ has the physical interpretation of the probability per unit path length for the specific process described by the (microscopic) cross section σ when all processes are combined by adding together the cross sections for each individual interaction, [21]. The resulting Σ_{tot} is the probability per unit path length that any type of interaction will occur.

$$\Sigma_{\text{tot}} = \Sigma_{\text{scatter}} + \Sigma_{\text{capture}} + \dots \quad (2.15)$$

This quantity has the same significance for neutrons as the linear absorption coefficient for gamma rays. If a narrow beam attenuation experiment is carried out for neutrons, the number of detected neutrons will fall off exponentially with absorber thickness. In this case the attenuation relation is written.

$$\frac{I}{I_0} = e^{-\Sigma_{tot} t} \quad (2.16)$$

The neutron mean free path λ is, by analogy with the gamma-ray case, given by $1/\Sigma_{tot}$. In solid materials, λ for slow neutrons may be of the order of a centimeter or less, where as for fast neutrons, it is normally tens of centimeters.

Under most circumstances, neutrons are not narrowly collimated so that typical shielding situations involve broad beam or (bad geometry) conditions. Just as in the case of gamma rays, the exponential attenuation of Eq (2.16) is no longer an adequate description because of the added importance of scattered neutrons reaching the detector .A more complex neutron transport computation is then required to predict the number of transmitted neutrons and their distribution in energy.

When discussing the rate of reactions induced by neutrons it is convenient to introduce the concept of neutron flux. If we first consider neutrons with a single energy or fixed velocity v , the product $v \Sigma$ gives the interaction frequency for the process for which Σ is the macroscopic cross section. The reaction rate density (reactions per unit time and volume) is then given by $n(r) v \Sigma$ where $n(r)$ is the neutron number density at the vector position r , and $n(r) v$ is defined as the neutron flux $\Phi(r)$ with dimensions of $\text{length}^{-2} \text{ time}^{-1}$. Thus, the reaction rate density is given by the product of the neutron flux and the macroscopic cross section for the reaction of interest, [21].

$$\text{Reaction rate density} = \Phi(r) \Sigma \quad (2.17)$$

This relation can be generalized to include an energy-dependent neutron flux $\Phi(r, E)$ and cross section $\Sigma(E)$:

$$\text{Reaction rate density} = \int_0^{\infty} \Phi(r, E) \Sigma(E) dE \quad (2.18)$$

2.7.3 Neutron moderation and absorption

It is practical, useful, and convenient to represent neutron intensity in terms of the number per unit area, either as fluence (n/cm^2) or as a fluence rate or flux ($\text{n/cm}^2 \text{ s}$). The interactions that slow neutrons down and cause their eventual removal from a beam are probabilistic: they either occur or they do not. Consequently, a flux of neutrons of intensity I will be diminished in a thickness x of absorber proportional to the intensity of the neutron source and the neutron removal coefficient Σ_{nr} , of the absorbing material:

$$-\frac{dI}{dx} = \Sigma_{nr} I \quad (2.19)$$

$$I(x) = I_0 e^{-\Sigma_{nr} x} \quad (2.20)$$

Where I_0 is the initial intensity and $I(x)$ refers to those neutrons that penetrate a distance x in an absorber without a collision; therefore, $e^{-\Sigma_{nr} x}$ represents the probability that a given neutron travels a distance x without an interaction. Conceptually, Σ_{nr} can be thought of as the probability per unit path length that a neutron will undergo an interaction as it moves through an absorber and be removed from the beam either by absorption or scattering.

In this context then it very much resembles the attenuation coefficient for photons in “good (or narrow-beam) geometry,” and can be similarly developed and used for neutron shielding and dosimetry.

The features of neutron beam including the concept of (narrow-beam) effects, [21]. The various interactions serve to remove a neutron from the beam such that it does not reach the receptor of interest e.g., a detector or a person. In this respect elastic and inelastic scattering interactions deflect neutrons out of the beam and Σ_{nr} accounts for all of the processes that do so.

However neutrons scattered from the narrow beam are likely to undergo other scattering interactions and be deflected back into the beam and reach the receptor. These more realistic or (poor geometry) conditions are accounted for with a (neutron buildup) factor.

When no hydrogenous materials are present the neutron removal coefficient Σ_{nr} is determined by the macroscopic cross-section $\Sigma = N\sigma_t$ where N is the number of target atoms/cm³ in an absorber and σ_t is the total cross-section in barns (10–24 cm²/atom) for each atom in a unit volume of absorber. Therefore Σ_{nr} has units of cm⁻¹ and is closely related physically to the attenuation coefficient for photons (and the beta absorption coefficient for electrons), and is in fact used in a similar way; it can be converted to a neutron mass coefficient (cm²/g) by dividing by the density of the absorber:

$$\text{Neutron mass coefficient (cm}^2\text{/g)} = \frac{\Sigma_{nr}}{\rho} \quad (2.21)$$

Another related concept is the mean free path which is the average distance a neutron of a given energy will travel before it undergoes an interaction. The mean

free path can also be thought of as the average thickness of a medium in which an interaction is likely to occur and is similar to the mean life of a radioactive atom.

$$\text{Mean free path} = \frac{1}{\Sigma_{nr}} \quad (2.22)$$

2.7.4 Neutron shielding

Neutron shields incorporate materials that enhance interactions that deplete neutrons from the primary beam. For fast neutrons, materials that slow neutrons down through elastic and inelastic scattering are used in combination with materials that enhance capture of neutrons when they reach thermal energies. Hydrogen is very effective in removing neutron especially energetic ones from a beam because collisions with hydrogen the dominant interaction diminish the energy by about one-half. The cross-section for the scattered neutrons is considerably greater because of the lower energy and the mean free path length is reduced considerably therefore a collision with hydrogen effectively removes a fast neutron from the beam.

2.7.4.1 Neutron Shielding Materials

Neutron shielding compounds are needed that slow neutrons to thermal energies so that they can be absorbed by a variety of other materials. As seen in table (2.2) and (2.3) Materials with relatively high fast neutron scattering cross sections include: lithium-6 lithium-7 boron-10 and cadmium-114, **Error! Reference source not found..** Which have cross sections near 1 barn for 1MeV Fast neutron capture cross sections tend to be lower than thermal cross sections thus thermalizing the neutrons will increase absorption rates and if this proves desirable given other engineering considerations, water, boron, and carbide compounds are excellent candidates for slowing down and absorbing neutrons. Water, an excellent

moderator of neutrons, requires pressure or heat to be effective as it may be prone to freezing in the Martian environment or vaporizing in the lunar environment. Solid materials such as concrete have the advantage over fluids of material stability. Boron, lithium and cadmium are attractive candidates given that these materials are solid and have high absorption cross-sections and/or attenuation ability for neutrons of a given energy, **Error! Reference source not found.** and Table Summarize the relative probabilities for interaction in various materials,[22].

Table (2.2) Fast Neutron (2 MeV) Capture Cross Sections

Material	Nuclide density (nuclei/cm³)	Microscopic capture cross section (cm²)	Macroscopic capture cross section (cm⁻¹)	Microscopic scatter cross section (cm²)	Macroscopic scatter cross section (cm⁻¹)
Water	3.346×10^{22}	2.5×10^{-29}	8.36×10^{-7}	5.0×10^{-24}	0.1673
Lithium-6	5.33×10^{22}	1.0×10^{-29}	5.33×10^{-7}	2.0×10^{-24}	0.1066
Boron-10	1.391×10^{23}	8.0×10^{-29}	1.11×10^{-5}	1.0×10^{-24}	0.01391

Table (2.3) Thermal Neutron (0.025 eV) Capture Cross Sections

Material	Nuclide density (nuclei/cm³)	Microscopic absorption cross section (cm²)	Macroscopic absorption cross section (cm⁻¹)	Fractional attenuation through 10 cm material
Water	3.346×10^{22}	3.32×10^{-25}	0.1111	0.6708
Lithium-6	5.33×10^{22}	3.85×10^{-26}	2.052×10^{-3}	0.0203
Boron-10	1.391×10^{23}	5.0×10^{-25}	6.955×10^{-2}	0.05012

2.7.5 Neutrons detection

Neutrons are generally detected through nuclear reactions that result in prompt energetic charged particles such as protons, alpha particles, and so on.

Virtually every type of neutron detector involves the combination of a target material designed to carry out this conversion together with one of the conventional radiation detectors. Because the cross section for neutron interactions in most materials is a strong function of neutron energy rather different techniques have been developed for neutron detection in different energy regions, [21].

2.7.5.1 Slow neutron detection

Slow neutrons are in the energy range below 0.5 eV. There are common requirements for slow neutron detection. As in photon detection, the interaction cross section must be as large as possible to achieve a high efficiency. Neutrons always produce secondary gamma-rays by interacting either with detector materials or with surrounding materials. Thus the Q-value of the interaction should be large to make the gamma-ray discrimination easy. The third requirement is that the kinetic energies of the interaction products should be fully absorbed [23]. The three useful interactions are ^{10}B (n, α), ^6Li (n, α) and ^3He (n, p). The Q-values are so large (compared to the neutron energy) that the interaction products cannot give energy information on the detected neutron. Thus, these detectors are useful for the pulse counting mode and a time-of-flight spectroscopy system must be employed if slow neutron spectroscopy is intended. The reaction products are emitted in opposite directions to conserve the linear momentum and the energy of each product can be calculated by conservation of energy and momentum, [23].

i. Boron-based detectors

The most famous type of boron-based neutron detector is the BF_3 proportional counter. In this detector BF_3 gas acts as both a proportional gas and a neutron detection material. Among various types of boron-containing gases, BF_3 has high concentration of boron and its gas multiplication performance is good as well. The

intrinsic efficiency of a 30 cm long detector (96% enriched ^{10}B) filled to 600 to 92 % at thermal neutron energies (25 MeV) Figure (2.3) Shows pulse height spectra for a BF_3 detector. Depending on the energy deposition of each product ion, the total deposited energy inside the gas volume changes, [23].

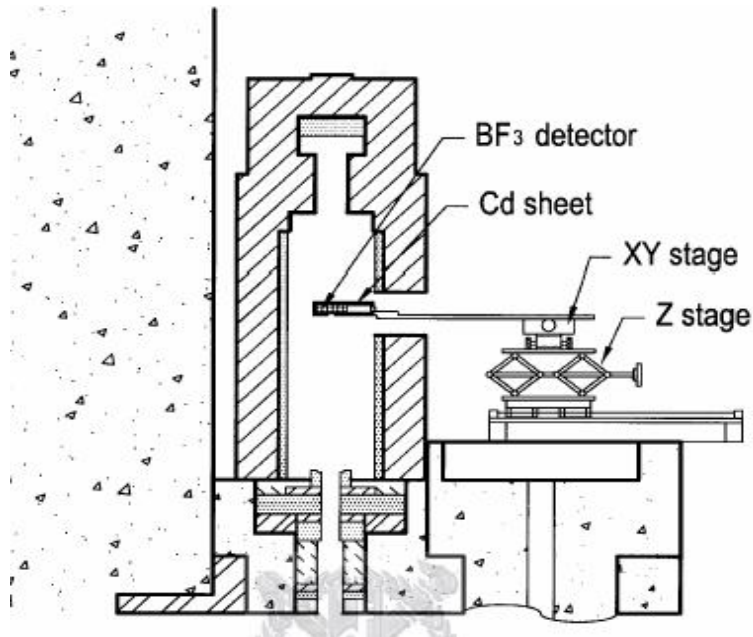


Fig (2.3) pulse height spectra for BF_3 detector

ii. Li-based detectors

Because a lithium-containing gas is not available, scintillation detectors are common as lithium-based slow neutron detectors. LiI is chemically similar to NaI, and therefore its scintillation performance is reasonably good. As an activator for LiI, Eu is doped. The thickness of a LiI (Eu) crystal is good enough to fully stop the neutron interaction products (alpha particle and triton). Thus, each neutron interaction event can make a signal pulse height equivalent to 4.78 MeV. When a gamma-ray with similar energy makes an interaction of full-energy deposition, the resultant pulse height is the same as the pulse height of the neutron event. Accordingly, the gamma-ray discrimination of LiI (Eu) neutron detectors is a major limitation of such type of gas neutron detectors, [23].

iii. ^3He -based detectors

Due to the natural property of helium, a gas proportional counter is the most popular type. Compared to the $^{10}\text{B}(\text{n},\alpha)$ interaction, the $^3\text{He}(\text{n},\text{p})$ reaction has a bigger cross section, which makes an ^3He proportional counter an attractive alternative to a BF_3 proportional counter. The lower Q-value of the $^3\text{He}(\text{n},\text{p})$ reaction makes gamma-ray discrimination more difficult than for a BF_3 counter,[23].

2.7.5.2 Fast neutron detection

Fast neutrons are often detected by first moderating (slowing) those to thermal energies where they can react more readily in certain detectors, such as gas-filled tubes containing ^3He or BF_3 . Fast neutrons are generally slowed by using a hydrogenous moderating material such as polyethylene where successive collisions between the neutron and hydrogen atoms efficiently transfer energy away from the neutron. However, during the slowing-down process the information on the original energy of the neutron its direction of travel and the time of emission is lost for many applications the detection of fast neutrons that retain this information is highly desirable,[24].

CHAPTER THREE

MONTE CARLO MODELING

CHAPTER THREE

MONTE CARLO MODLEING

3.1 Introduction

This chapter includes Monte Carlo modeling using Monte Carlo simulation program (MCNP5) to determine the effectiveness of shielding models (High density polyethylene with Borated Polyethylene, High , High density polyethylene with Boron carbide and High density polyethylene with Boric Acid)and examined against different moderation materials heavy water and Beryllium with optimal absorber as shielding materials for fast neutrons. This is done by utilizing ^{252}Cf as the fast neutron source and ^3He as thermal neutron detector.

3.2 Monte Carlo methods

Monte Carlo experiments are a broad class of computational algorithms that rely on repeated random sampling to obtain numerical results typically one runs simulations many times over in order to obtain the distribution of an unknown probabilistic entity. The name comes from the resemblance of the technique to the act of playing and recording results in a real gambling casino. They are often used in physical and mathematical problems and are most useful when it is difficult or impossible to obtain a closed-form expression or unfeasible to apply a deterministic algorithm. Monte Carlo methods are mainly used in three distinct problem classes: optimization, numerical integration and generation of draws from a probability distribution, [8].

3.2.1 Mont Carlo code

The MCNP5 Code developed and maintained by Los Alamos National Laboratory is the internationally recognized code for analyzing the transport of neutrons and gamma rays (hence NP for neutral particles) by the Monte Carlo method (hence MC). The code deals with transport of neutrons, gamma rays, and coupled transport, i.e., transport of secondary gamma rays resulting from neutron interactions. The MCNP5 code can also treat the transport of electrons, both primary source electrons and secondary electrons created in gamma-ray interactions, [25].

3.2.2 Structure of the MCNP5 Input File

An input file has the structure. Input lines have a maximum of 80 columns and command mnemonics begin in the first 5 columns. Free field format (one or more spaces separating items on a line) is used and alphabetic characters can be upper, lower, or mixed case. A continuation line starts with 5 blank columns or a blank followed by the end of the card to be continued, [25].

3.2.2.1 Annotating the Input File

It is good practice to add comments liberally to an input MCNP5 file so that it is easier for you and others to understand what problem is addressed and the tricks used. A comment line begins with C or c followed by a space. Such a line is ignored by MCNP. Alternatively, anything following a \$ sign on a line is ignored.

3.2.2.2 Units Used by MCNP5

The units used by MCNP are (1) length in cm, (2) energy in MeV, (3) time in shakes (10^{-8} s), (4) temperature in MeV (kT), (5) atom density in atoms $\text{b}^{-1} \text{cm}^{-1}$, (6) mass density in g cm^{-3} , and (7) cross sections in barns.

3.3 The proposed shielding structure

The proposed shielding system consist of ^{252}Cf nature source 10^7 n s^{-1} located at -10cm on the x-axis apart from a shield of dimension 25cm x20cm x 50cm with different filling materials . ^3He thermal neutron detector was located +100cm on the x-axis apart from the purposed shield.

3.3.1 Californium-252 (^{252}Cf) sources

Table (3.1) Characteristics of the ^{252}Cf neutron source

Item	Description
Atomic number	98
Half Life	2.65 Years
Spontaneous fission half-life	85.5 year
Neutron yield	2.34×10^6 n/ $\mu\text{s.g}$
Gamma-ray yield	1.3×10^7 γ / $\mu\text{s.g}$
Alpha-particle yield	1.9×10^7 α / $\mu\text{s.g}$
Average neutron energy	2.14 MeV
Average gamma-ray energy	1.0 MeV

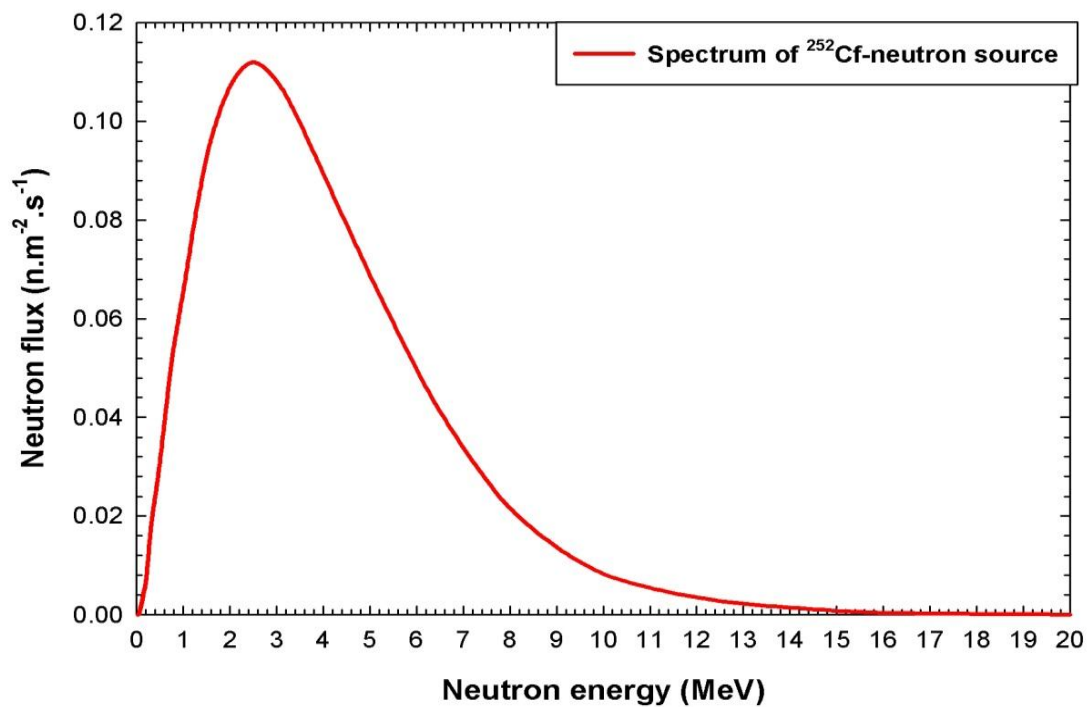


Fig (3.1) flux of fast neutron as function of neutron energy for ^{252}Cf neutron source

3.3.2 ^3He - Neutron detectors

Helium-3 is a most important isotope in instrumentation for neutron detection see figure (3.2) it has a high absorption cross section for thermal neutron beams and is used as a converter gas in neutron detectors. ^3He is an isotope of the noble gas helium. It is stable nonradioactive inert and nontoxic. The natural abundance of ^3He in helium is very small only 0.00014%. ^3He is a strong absorber of neutrons. The neutrons are captured by the reaction $^3\text{He} (n,p) ^3\text{H}$ building a proton and a triton with a reaction Q-value of 764 keV. The energy dependent cross-section of this reaction is one of the well known standards in neutron measurements. Both the proton and the triton are charged ions and will be registered by the proportional counter.

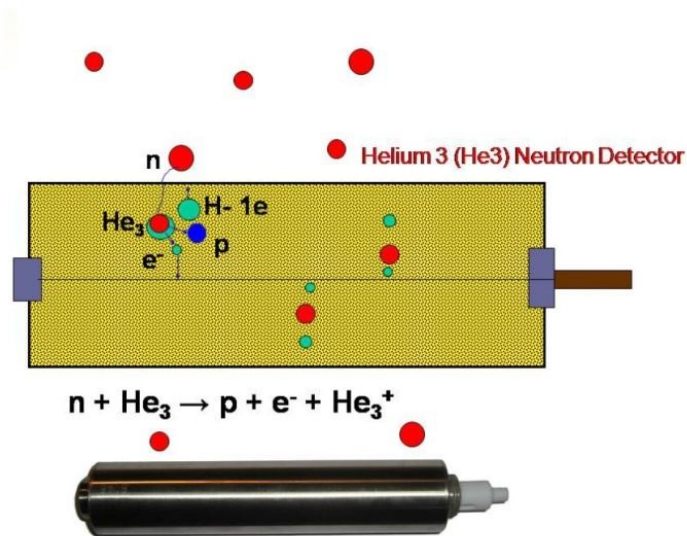


Fig (3.2) ^3He - neutron detector

3.3.3 Materials and shield model

The geometry

An input file was developed to simulate 20cm thick layer representing the moderator and of diminution 20cm x20cm x50cm. In ascend layer representing the absorber of diminution 5cmx20cm x50cm was model figure (3.3) shows the geometry used in the current simulation, while table (3.2) and table (3.3) represent the composition of the material used. the input file was present

∴ Appendix A.

Table (3.2) moderator as simulation in MCNP5 calculation, [26]

moderator	Density g\cm ³	Weight fractions				
		¹ H	² H	O	C	Be
heavy water D ₂ O	1.107		0.1118	0.8881		
Beryllium Be	1.82					1
High density polyethylene HDPE	0.95	0.143			0.857	

Table (3.3) absorber as simulation in MCNP5 calculation, [26]

Absorber	Density g\cm ³	Weight fractions					
		¹ H	² H	O	C	¹⁰ B	¹¹ B
Borated polyethylene BPE	0.95	0.116		0.222	0.612	0.05	
Boric acid H ₃ BO ₃	1.47	0.018		0.792		0.035	0.155
Boron carbide B ₄ C	2.52				0.2310	0.7690	

Geometry of the Proposed Model:

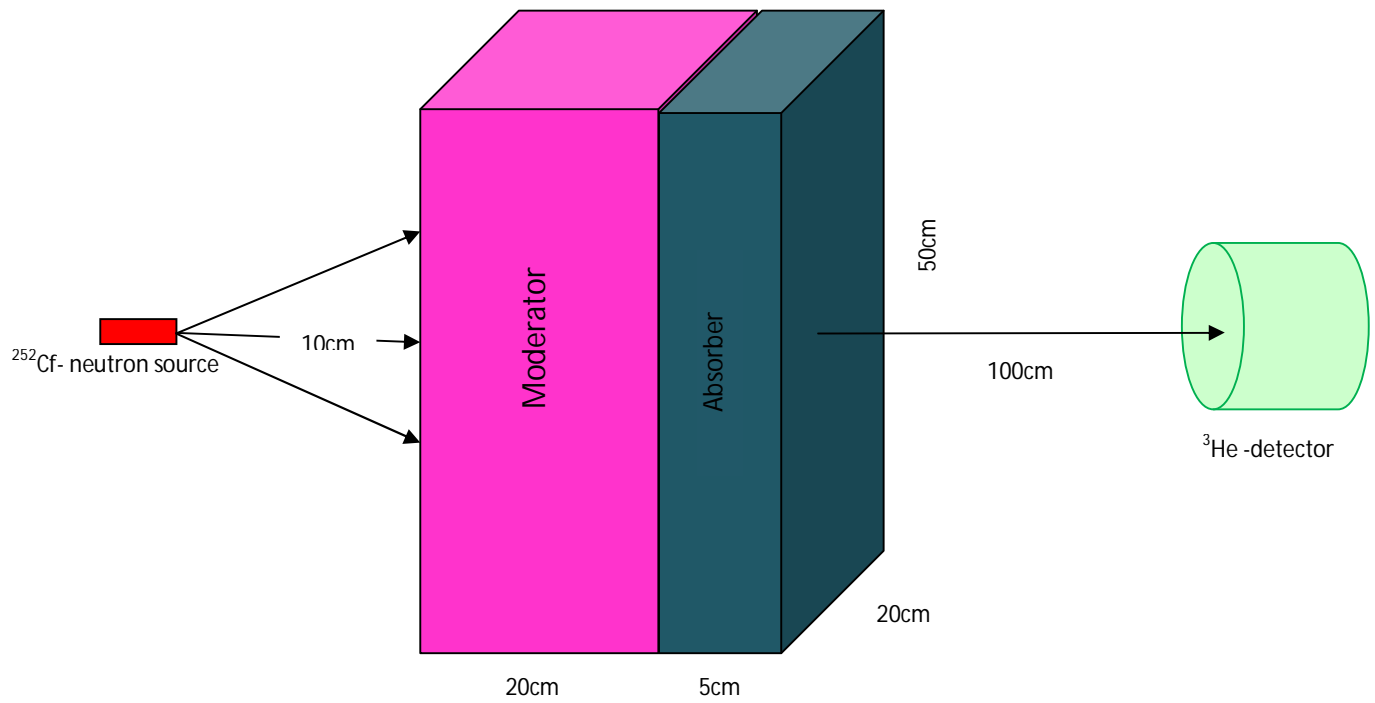


Fig (3.3) Dual layer shielding system for fast neutron

CHAPTER FOUR

RESULT AND DISCUSSION

CHAPTER FOUR

RESULT AND DISCUSSION

4.1 Calculate procedures

The first layer was simulated as container of dimension 20cm x 20cm x 50cm filled with high density polyethylene (HDPE), and being fixed as the moderator.

The second layer was simulated as container of dimension 5cm x 20cm x 50cm filled with three different borated materials; borated polyethylene (BPE), Boric acid (H_3BO_3) and Boron carbide (B_4C) Figure (4.1) and (4.2) shown the result in terms of transmitted neutron flux as function of energy.

As shown in figure (4.2) the optimal configuration is 20cm HDPE +5cm BC_4 the configuration has reduced the source energy 88% and intensity by 99%.

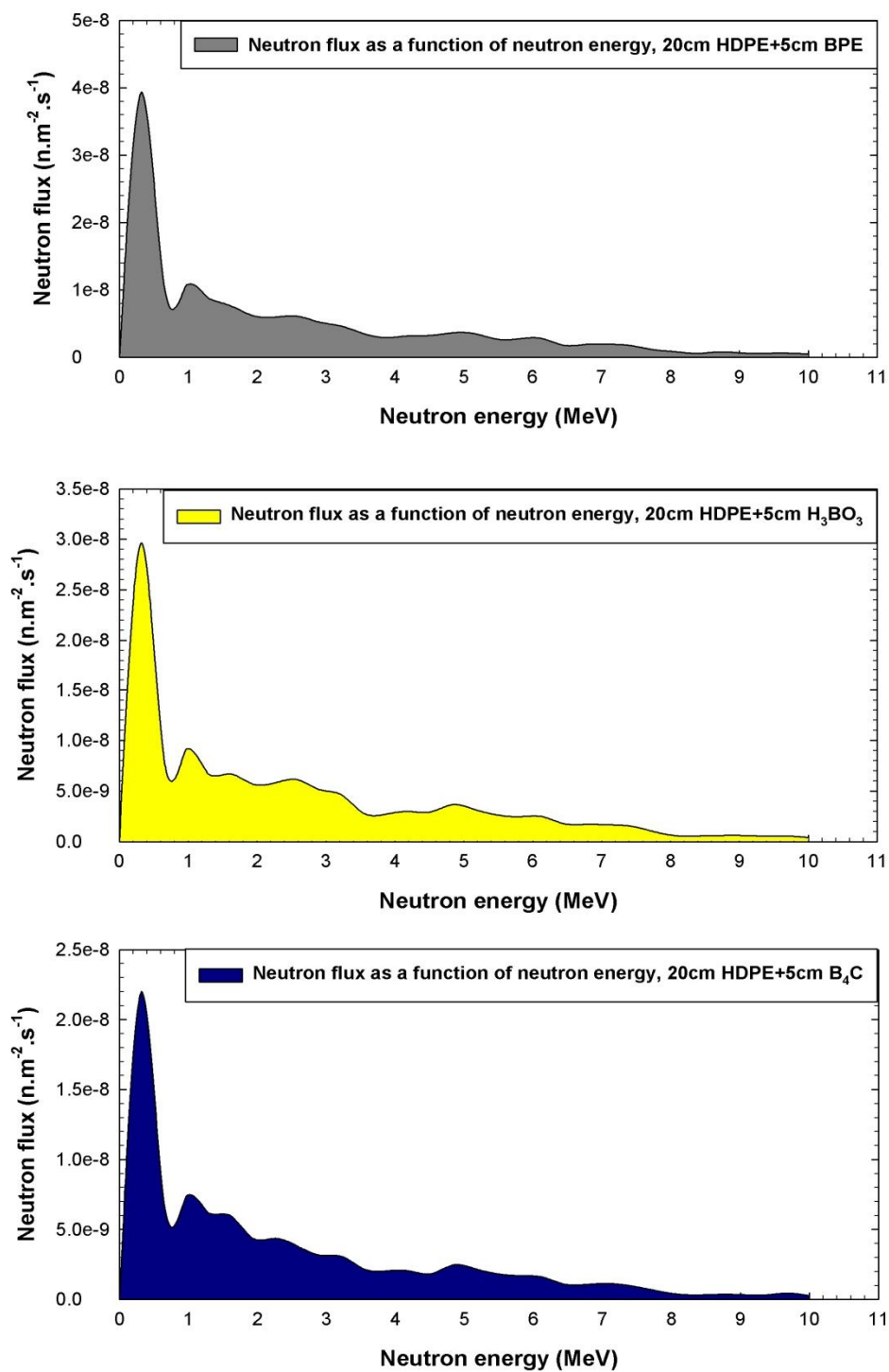


Fig (4.1) flux as function of neutron energy for different configuration

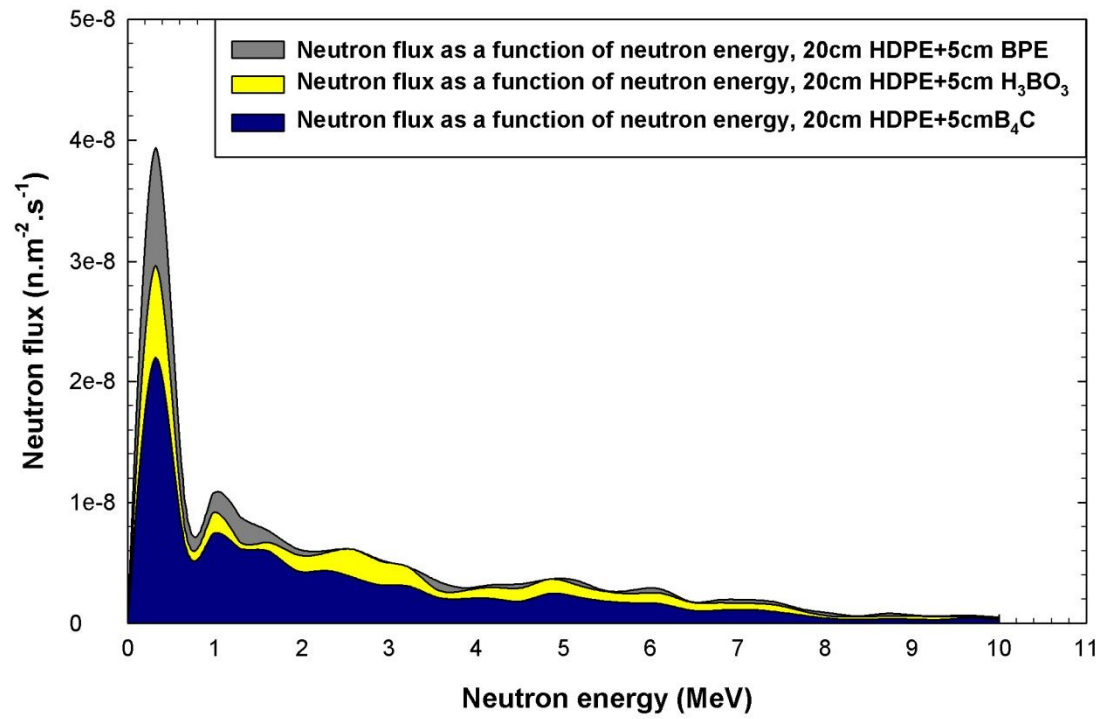


Fig (4.2) flux as function of neutron energy for optimal configuration

To confirm the stability of the optimal configuration it is perform has been examined against different moderation materials such as; heavy water (D_2O) and Beryllium (Be) .figure (4.3) show the result.

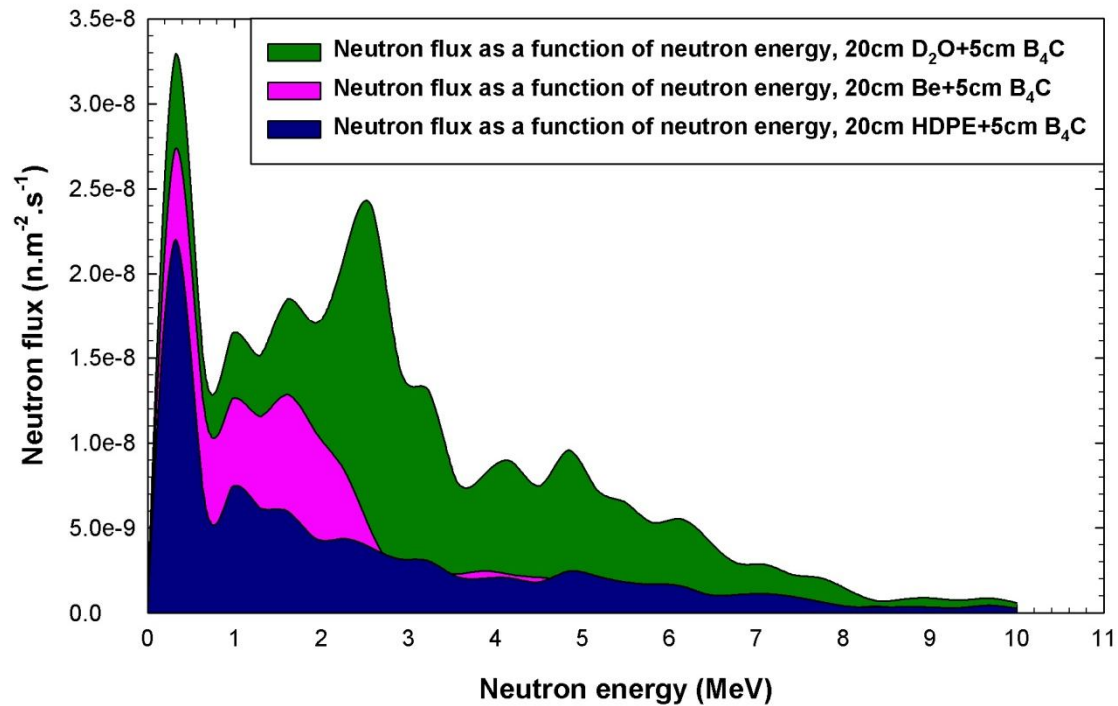


Figure (4.3) flux as function of neutron energy for optimal configuration

As shown the optimal configuration (20cm HDPE +5cm B_4C) x20cm x50cm still dominant ,(20cm HDPE+5cm B_4C) has reduced neutron intensity with respect to (20cm D_2O +5cm BC_4) by approximated 33% and with respect to (20cm Be+5cm B_4C) by approximated 18%.

4.2 Conclusions

Monte Carlo simulation code (MCNP5) was used to model a shielding system consisting of dual layers to shield fast neutrons emitted from ^{252}Cf -neutron source and to verify the performance of several moderating materials and absorbers as well.

The performance of high density polyethylene (HDPE) as moderating material was examined against different thermal neutron absorbers such as; borated polyethylene (BPE), Boric acid (H_3BO_3) and Boron carbide (B_4C). The optimal configuration was found to be (20cm HDPE +5cm B_4C) since it has reduced the source energy by 88% and the intensity by 99%.

To confirm the stability of the optimal configuration, its performance has been examined against two moderating materials such as; heavy water (D_2O) and Beryllium (Be). It is found that the configuration (20cm HDPE +5cm B_4C) is still dominant and has reduced neutron intensity with respect to configuration (20cm D_2O +5cm B_4C) by approximated 33% and with respect to configuration (20cm Be+5cm B_4C) by approximated 18%.

4.3 Recommendations

The current fast neutron shielding model was simulated using Monte Carlo computer code version 5 (MCNP5). Further experimental verification is required to examine the effectiveness of the proposed shielding system.

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Appendix A:

c	CELL	CARDS
1	1	-0.955 1 -2 -3 4 -5 6
2	4	-2.52 7 -8 -9 10 -11 12
3	5	-0.0012 (-1 :2 :3 :-4 :5 :-6)(-7 :8 :9 :-10 :11 :-12)-15
4	0	15

c	CERFACE	CARDS
1	pz	-10
2	pz	10
3	py	50
4	py	0
5	px	10
6	px	-10
7	pz	-10
8	pz	10
9	py	50
10	py	0
11	px	15
12	px	10
15	so	200

c DATA CARDS

c

c CELL IMPORTANCE

imp:n 1 1 1 0

c

c SOURCE SPECIFICATION

sdef pos= -20 25 0 erg=d1

SI1 3.16000E-02 3.98000E-02 5.01000E-02 6.31000E-02
7.94000E-02 1.00000E-01 1.26000E-01 1.58000E-01
2.00000E-01 2.51000E-01 3.16000E-01 3.98000E-01
5.01000E-01 6.31000E-01 7.94000E-01 1.00000E+00
1.26000E+00 1.58000E+00 2.00000E+00 2.51000E+00
3.16000E+00 3.98000E+00 5.01000E+00 6.31000E+00
7.94000E+00 1.00000E+01 1.26000E+01 1.58000E+01
2.00000E+01

sp1 0.00E+00 3.58E-05 1.97E-04 3.58E-04
1.02E-03 1.86E-03 2.91E-03 4.59E-03
6.42E-03 1.13E-02 1.81E-02 2.37E-02
3.08E-02 4.15E-02 5.38E-02 6.53E-02
8.07E-02 9.57E-02 1.07E-01 1.12E-01
1.06E-01 8.99E-02 6.87E-02 4.43E-02
2.22E-02 8.30E-03 2.68E-03 4.47E-04
7.09E-05

c

c TALLIES

f5:n 110 25 0 0.5

e5 0 30I 10

c

c MATERIAL SPECIFICATION

m1 1001. -0.1429 6012. -0.8571 \$HDPE

m2 1001. -0.116 6012. -0.612 \$BHDPE

5010. -0.010 5011. -0.040

8016. -0.222

m3 1001. -0.018 5010. -0.035 \$Boric acid

5011. -0.155 8016. -0.792

m4 5010. -0.769 6012. -0.231 \$Boron carbide

m5 07014. -0.7800 08016. -0.2200 \$Air

c

c PROBLEM CUT-OFF

nps 1000000

c

c PERIPHERAL CARDS

print