AN INVESTIGATION ON THE ATTENUATION OF NEUTRON AND GAMMA RAYS IN REACTOR CONCRETE BIOLOGICAL SHIELD

by

ABDUS SATTAR MOLLAH

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has been accepted as satisfactory in partial fulfilment for the degree of Master of Philosophy in physics and certify that the student demonstrated a satisfactory knowledges of the field covered by this thesis in an oral examination held on 30th December, 1987.

Board of Examiners

(1) Prof. Gias Uddin Ahmad
Dept of Physics, BUET.

Gias uddin Ahmad 30/12/87
Supervisor and Chairman

(2) Dr. S.R. Husain.
Chief Scientific Officer
Bangladesh Atomic Energy Commission

30.12.87
Co-Supervisor and Member

(3) Prof. Tafazzal Hossain
Head, Physics Dept., BUET.

T Hossain 30.12.87
Member

(4) Prof. M.A. Asgar
Physics Dpet., BUET.

30/12/87
Member

(5) Dr. M. Ahsan
Director, Commercial Division,
Bangladesh Atomic Energy Commission

30.12.87
Member (External)

CERTIFICATE

This is to certify that the author is solely responsible for the work reported in this thesis and that this work has not been submitted to any university for the award of a degree.

Abdus Sattar Mollah

(ABDUS SATTAR MOLLAH)
Candidate

Giasuddin Ahmad

(DR. G. U. AHMAD)
Guide

S.R. Husain

(DR. S. R. HUSAIN)
(CO-SUPERVISOR)

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ABSTRACT

The biological shield of a nuclear reactor is required to attenuate both neutrons and gamma rays from the reactor core. Generally, materials having high densities are used as biological shield materials in a nuclear reactor. Concrete has been widely used as shielding materials due to its satisfactory material strength and good shielding properties. The density of conventional ordinary concrete is not adequate for a biological shield in a nuclear reactor. Types of concrete possessing high density should be developed in order to attenuate both neutron and gamma radiation emitted from the nuclear reactor. The heavy aggregates such as barite, magnetite, ilmenite, limonite etc. have been widely used for preparing a high density concrete for biological shield in a nuclear reactor. While fabricating the high density concretes, due consideration should be given to the local availability of the aggregates. Since the magnetite and ilmenite aggregates are available as by products in the beach sand processing pilot plant at Cox's Bazar of the Bangladesh Atomic Energy Commission, this work was undertaken to design and fabricate high density concrete shields indigenously with the objective of avoiding costly imports and also to develop local know how.

In this study, high density concrete with and without boron additives using ilmenite and magnetite sand have been studied. For comparison, ordinary concrete with and without boron additives have also been studied. In addition, mixed

concrete used in the biological shield of 3 MW TRIGA Mark-II research reactor of the Bangladesh Atomic Energy Commission has been studied with and without boron additives for comparison with the other three types of concretes with and wihtout boron additives. The density, compressive strength, modulus of elasticity and splitting strength were measured for these concretes. The densities of ilmenite and magnetite concretes are found considerably higher than that of ordinary concrete.

The nuclear parameters such as attenuation coefficient, removal cross section, half value thickness, and relaxation length, which are of interest for reactor shielding have been evaluated for these concretes with a spontaneous fission source (^{252}Cf). The investigation concerning the attenuation properties reveals that the shielding properties of boron loaded heavy concretes are better than those of ordinary concretes and slightly better than those of heavy concretes without boron additives.

CONTENTS

CHAPTER 1

INTRODUCTION

1.1 Scope and objectives

A nuclear reactor is a prolific source of potentially dangerous radiation; it is unavoidably so, since most of the radiation released originates with the fission process itself. In addition to the energetic neutrons and gamma rays that are emitted simultaneously with the fission event, the fission fragments formed are highly radioactive nuclides that emit alpha, beta and gamma radiations. That nuclear radiation can be injurious to man is beyond dispute. To enable personnel to work in the vicinity of a reactor in operation, it is necessary to attenuate this radiation in a shield surrounding the core, so as to reduce the radiation dose to a tolerable level in the region beyond the shield. The biological hazards of radiation arise from the fact that the radiation interacts with human tissues, losing some of its energy in the process. This loss of energy is sufficient to ionize atoms of cells, upsetting the delicate chemical equilibrium of the cell and even causing its death[1].

The radiation must, therefore, be attenuated sufficiently so that the residual radioactivity cannot render permanent damage to the persons exposed to it. The attenuation of this biologically harmful ionizing radiation to an acceptable dose level, is usually achieved by surrounding the reactor core with a sufficiently large mass of suitable materials like Pb, Fe, etc. or high density concrete. The main reactor shield used for this purpose is frequently referred to as the biological shield[2,3].

There are two general classes of radiations to be considered in the design of biological shields of a reactor: (1) primary radiation, and (2) secondary radiation. Primary radiation is defined as the radiation that results from nuclear fission which includes - (i) charged particle radiation, (ii) gamma radiation from the decay of the radioactive fission products, and (iii) neutron radiation. Secondary radiation includes all other sources of radiation. An important component of secondary radiation is the gamma radiation generated in the shield materials themselves, due to the interaction of the primary neutrons with the materials of the shield.

The group of charged particles consists of heavy, positively charged particles, like alpha particles, protons, deuteron, and the fission fragments, and the much lighter beta particles. The presence of alpha particles in nuclear reactors arises from the radioactive decay of the fuel which consists of naturally radioactive materials. Protons and deuterons are present in negligible numbers only, generally as the result of neutron reactions with the nuclei of the moderator, coolant, and the surrounding air as, for example, from the (n,p) reaction with ^{14}N. The beta particles originate from the decay of the radioactive fission fragments which undergo a series of consecutive beta transformations, and from the radioisotopes that are formed in the reactor as a result of neutron capture. Electrons are also liberated through the interaction of electromagnetic radiation i.e., gamma radiation with atoms and nuclei. The charged particles interact primarily with the atoms of matter as a whole, or with the atomic electrons, and only very infrequently with the nuclei. Nuclear collisions are rare and the probability of their occurrence is very small for the charged particles. Since proton, alpha and beta particles are charged, they interact with the electric field surrounding the atoms of the shielding material and lose their energy considerably. Generally, they do not constitute a separate shielding problem. The most important mode of interaction between gamma radiation and matter which leads to a reduction of the gamma ray energy is that between gamma radiation and the electrons of the absorbing material. Although interactions between gamma radiation and nuclei occur sometimes, the frequency of this occurrence or the cross sections for these processes are relatively small compared to those for the electron interaction. Neutrons, on the other hand, cannot interact electrically with atoms, since they are electrically neutral and, therefore lose energy only through interactions with atomic nuclei.

Ionizing radiation produces ion pairs in its passage through a substance, both directly and/or indirectly. Directly ionizing particles are charged particles, such as electron, proton and alpha particle, having sufficient kinetic energy to produce ionization by collision. Indirectly ionizing particles are uncharged particles, such as neutrons and gamma rays, which can liberate directly ionizing particles or initiate a nuclear transformation. Thus neutrons and gamma rays can only cause ionizations as a result of secondary processes. Although many types of nuclear particles are released directly and indirectly in and around the reactor core, the essential shielding problem is the attenuation of the penetrative fast neutrons and gamma rays that are released in the reactor core and reactor shield. In a nuclear reactor, even though the radioactivity is primarily well contained within the core assembly, penetrating radiations like neutrons and gamma rays constitute the major external exposure hazard to occupational workers, the population at large and the related environment. The potential hazard exists not only due to the primary radiations originating from the reactor but also from secondary radiations such as capture gammas. All nuclear reactors, except those operating near zero power level, are sources of intense neutron and gamma radiation* and therefore, represent a serious health hazard to the operating personnel and research workers.

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*The total amount of activity[4] in curies contained within a nuclear reactor can be estimated by multiplying the thermal power, P(in watts) by the factor 10. Thus, A(curies)=10xP.

Electronic equipment, which may be part of the reactor instrumentation system, can be severely damaged or rendered unreliable by being exposed to a strong radiation field. Therefore, instruments and other engineering equipments, may also require to be shielded from the harmful effects of nuclear radiation. In order to protect the occupationally exposed personnel, the population at large, equipment and the related environment from the deleterious effects of ionizing radiations, it is mandatory to shield the reactor assembly adequately so as to reduce the radiation level within the permissible limit as recommended by the International Commission on Radiological Protection (ICRP)[5]. Thus, the application of biological shielding around the reactor assembly resolves basicaly into the problem of protection of radiation workers against gamma rays and neutrons.

The distinction that is sometimes made in the role of the biological shield, is between the neutron shield and the gamma ray shield. This distinction arises because of the fundamental difference in the basic physical processes by which the two species of radiation interact with matter. A heavy material such as lead or iron, is a relatively good absorber of gamma rays, but is much less effective against neutrons with energies in the range of 1 eV to 1 MeV. By way of contrast, a hydrogen-containing material such as water is particularly effective in shielding against neutrons in this energy range but is not particularly effective as an attenuator of gamma rays. Concrete, which is an homogeneous mixture of light and heavy nuclei that possesses the essential characteristics for both neutron and gamma ray attenuation. It has satisfactory mechanical and structural properties, and it also involves relatively low manufacturing as well as maintenance cost. Also the easy process of construction makes concrete specially suitable for radiation shielding. Hence, to day it is being widely and extensively used as biological shield material. Since the attenuation to be provided by the biological shield for nuclear radiations is very large, a need to improve the quality and effectivenes of concrete as a radiation shield has always been felt. For protection mainly against gamma radiations, it may be desirable to use heavy concretes with densities of more than 2.3 gm/cm^3, prepared from heavy aggregates of natural and/or artificial origin. For protection against the simultaneous action of gamma and neutron radiation, it may be advantageous to use heavy concretes containing a large amount of bond or semi-bond water bearing aggregates and/or boron bearing aggregates. In choosing the type of aggregates due consideration should be given to the local availability of aggregates and their characteristic propeties. Depending on the local aggregates available, many combinations in concretes are possible. In order to use locally available materials, a research and development work was undertaken to design and fabricate the reactor biological concrete shield with the objective of avoiding costly imports and also to develop local know how.

1.2 Interaction of neutrons and gamma rays with matter

The biological shield of a nuclear reactor is required to attenuate both gamma rays and neutrons emerging from the core. The energies of these radiations cover a wide range. The interaction processes of neutrons and gamma rays with matter are useful in making preliminary estimates of the attenuation of neutrons and gamma rays by the shield. In principle, the problem of reactor shielding has three aspects:

(a) slowing down of fast neutrons;

(b) capture of the slowed down (or initially slow) neutrons; and

(c) attenuation ofgamma radiation originated in reactor including primary radiation from the core and secondary radiation formed as a result of various interaction between neutrons and nuclei of the surrounding materials of the reactor core.

Processes by which fast neutrons and gamma radiations are attenuated in a reactor shield are well understood. Generally, they are attenuated in the shield by their interaction with matter. In view of the many excellent treatments in existence[6-8], it is deemed unnecessary to discuss the interaction of neutrons and gamma rays with matter in detail here. A brief outline will be given here to provide the various processes by which the neutrons and gamma rays are attenuated in material. They are as follows:

(i) Interaction of neutrons with matter; and

(ii) Interaction of gamma rays with matter.

1.2.1 Interaction of neutrons with matter

Neutrons may be either captured or scattered by nuclei, with different probabilities for each constituent of the shield. Scattering process may be either elastic or inelastic and they may be isotropic or non-isotropic. Elastic scattering, in which a neutron is deflected without loss of energy except for the kinetic energy which it imparts to the recoiling nucleus. Inelastic scattering, in which there is an interaction between the nucleus and the neutron resulting in the emission of a neutron with an energy different from that of the incident one. Capture, in which a neutron is absorbed into a nucleus which then undergoes some reaction (other than the re-emission of a neutron), usually the emission of a photon. One of the principal factors in neutron shielding is the reduction of their energy. When the neutron energy has been brought down to a value approaching thermal energies the capture probability increases rapidly and the neutrons will be completely absorbed with the emission either of another particle or of a gamma ray. Elements of low mass number is often used as the shield for slowing down neutrons of high energy; the cross section for scattering is, however, small -especially for elements of low mass number. Although hydrogen alone (as water) could be used as a neutron shield, a shield consisting entirely of hydrogen in the form of heavy water would be unsatisfactory. It will be advantageous to utilize inelastic scattering by introducing an element (or elements) of moderate or high mass number. They do not decrease the neutron energy to an appreciable extent in elastic collisions, but they are able to reduce the energies of very fast neutrons as a result of inelastic scattering collisions. However, a combination of a moderately heavy or heavy element with hydrogen will effectively slow down neutrons of very high energies. A single inelastic scattering collision would decrease the neutron energy to about 1 MeV or less, but subsequent collisions would be elastic in nature. Many such collisions would then be necessary to reduce the neutron energy to a value at which the capture cross section is significant. In the mean time, the neutrons would have penetrated a considerable distance into the shield. The next aspect of neutron shielding to consider is the capture of neutrons after they have been slowed down. Actually this is a relatively simple matter, even though the shield does not contain an element with a large or fairly large cross section for the capture of slow neutrons. In fact, it is generally accepted that in a shield containing sufficient hydrogen, almost every neutron which has suffered inelastic scattering may be regarded as removed, since the probability of further slowing down and subsequent capture is large. Concrete is useful for permanent shields around neutron generator and reactors, and heavy elements are sometimes included in the concrete because of their inelastic properties. In all cases where large attenuations are necessary, the secondary gamma radiation from inelastic scattering and capture processes will become important and additional thickness may be required to reduce the gamma ray dose rate. Boron is sometimes included in small proportions in shielding materials because it has a high capture cross-section and the gamma radiation emitted is of low energy so that it is easily attenuated.

1.2.2 Interaction of gamma rays with matter

There are three ways in which gamma rays interact with an absorbing material; the photoelectric effect, the Compton effect,and pair production. These interactions wil now be discussed briefly.

(i) The photoelectric effect:

In the photoelectric effect, a gamma-ray photon, with energy greater than the binding energy of an orbital electron in an atom interacts with the latter in such a way that the whole of the gamma-ray energy is transferred to an electron which is consequently ejected from the atom. The probability of this occurrence, and hence the value of the absorption coefficient, is greatest when the photon has just enough energy to remove the electron from its orbit; below this it cannot occur and with increasing photon energy the absorption coefficient falls fairly rapidly. The extent of the photoelectric interaction depends on both the energy E of the gamma radiation and the atomic number Z of the absorbing material. As a rough approximation, the probability of photoelectric interaction is given by

$$\sim \text{Constant} \times Z^n/E^3 \quad \ldots \quad (1)$$

where n varies from about 3, for gamma rays of low energy, to 5, for high energy gamma rays. It is apparent, therefore, that the photoelectric effect increases with increasing atomic number of the absorber and with decreasing energy of the gamma rays. In actual practice, it is found that photoelectric absorption of gamma radiation is important only for energies less than about 1 MeV, and then only for absorbers of high atomic number. It is evident, from the foregoing discussion, that the photoelectric effect leads to the virtually complete removal of the gamma-ray photon; it may be replaced, to some extent, by X-ray photons of low energy, but few of these escape as such from the absorbing material.

(ii) The Compton scattering:

In Compton scattering the photon collides with a loosely bound electron and loses some of its energy in imparting momentum to the electron. The degraded photon and the electron then move off at an angle to each other like collisions between billiard balls. This part of the energy of the photon is absorbed and the rest is scattered. Since the Compton effect involves interaction between a photon and an electron, its magnitude is dependent on the number of orbital electrons in the atom of the absorber; this is same as the atomic number. The Compton interaction is thus directly proportional to the atomic number of the absorber, so that, like the photoelectric effect, it is more significant for materials of high atomic number. As a very rough approximation, the probability of Compton interaction is given by

$$\sim \text{Constant} \times Z/E \qquad \ldots \qquad (2)$$

However, there is a significant difference between the photoelectric and Compton effects to which attention must be drawn. The photoelectric effect is a true absorption process; that is to say, the photon is absorbed, as stated above. In experiments made with a narrow beam and a collimated detector the scattered radiation is lost from the beam and so appears to be absorbed and consequently the effect of this scattering is included in the linear absorption coefficient. Compton absorption is of importance upto energies of several MeV.

(iii) The pair production:

Pair production occurs when a sufficiently energetic photon passes close to the nucleus. The whole of the energy of the photon is converted into two electrons, one positive and other negative. Since the mass energy equivalent of an electron is 0.51 MeV, this effect can only occur with photons having energy of 1.02 MeV and above. The excess energy of the photon appears as momentum of the electrons. The positron has a very short life as it quickly combines with a negative electron and both are annihilated with the emission of two 0.51 MeV photons. The extent of pair

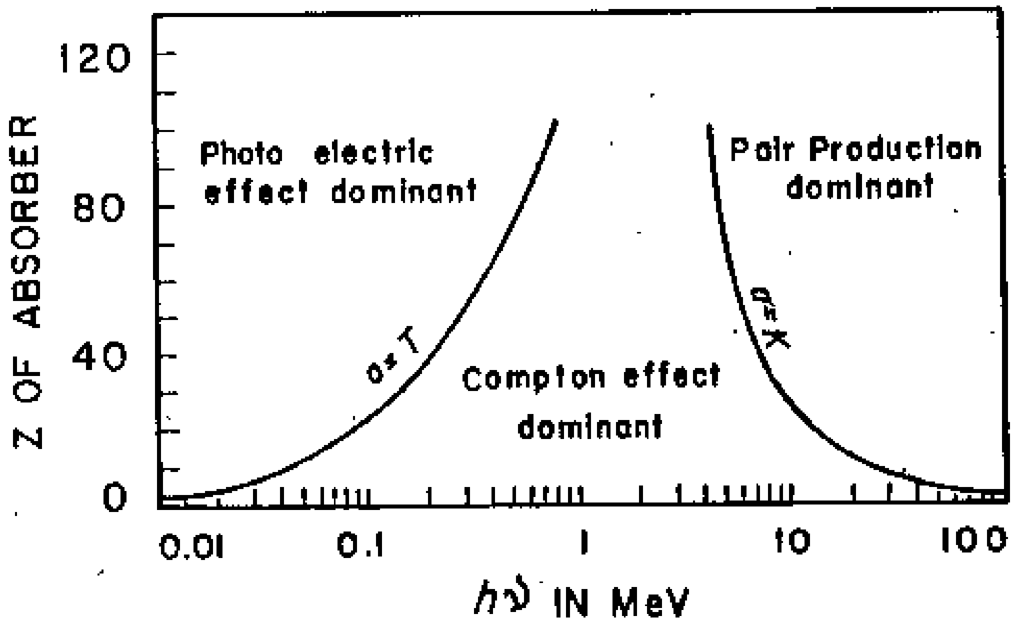


Fig. 1 The relative importance of the three major types of gamma ray interactions.

production by gamma radiation of energy E (in MeV) is related to the atomic number Z of the absorber. As a very rough approximation, the probability of pair production is given by

$$\sim \text{Constant} \times Z^2 \,(E-1.02) \qquad \ldots \qquad (3)$$

It therefore, increases with the atomic number of the absorbing material and with increasing photon energy in excess of 1.02 MeV. Since both the photoelectric and Compton effects decrease with increasing gamma-ray energy, whereas pair production increases, it is evident that the latter process will become of major importance at high energies. For absorbers of high atomic number it becomes the dominant type of interaction of gamma rays with energies in excess of about 5 MeV. It is clear from this section that the total photon cross-section to be used in attenuation calculations is given by the sum of the cross section for photoelectric effect, Compton scattering effect, and pair production. The relative importance of the three processes described[7] above for different absorber materials and gamma ray energies is illustrated in Fig.1.

1.3 Definition of some nuclear parameters

To evaluate the effectiveness of any new shielding material, the following nuclear parameters should be measured:

(i) Attenuation coefficient (μ);

(ii) Half-value thickness (HVT);

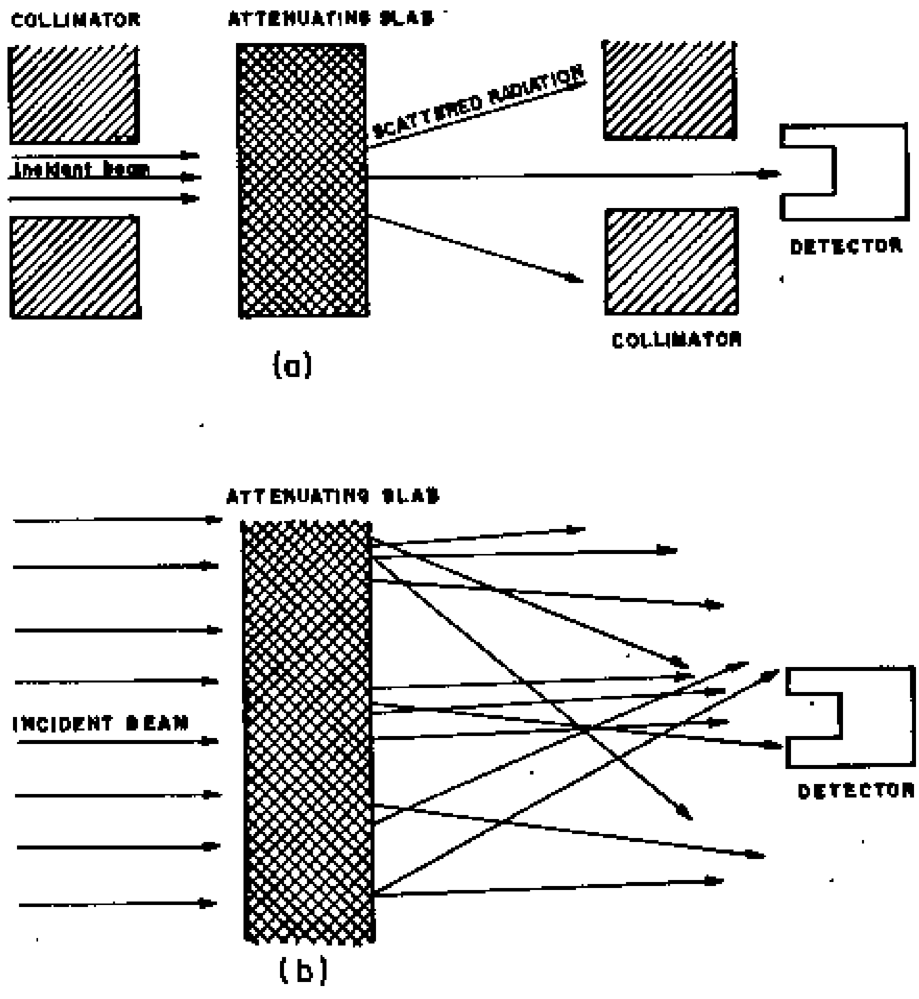


Fig. 2. Experimental arrangements for measurments of attenuation coefficient:
(a) narrow beam or good geometry and (b) broad beam or poor geometry.

(iii) Buildup factors (B);

(iv) Removal cross section (Σ R); and

(v) Relaxation length (λ).

A brief description of each term mentioned above and their relationship is discussed below:

(i) Attenuation coefficient (μ)

The most important quantity characterizing the penetration and diffusion of gamma radiation in extended media is the attenuation coefficient, μ . This quantity depends on the photon energy E and on the atomic number Z of the medium and may be defined as the probability per unit path length that a photon will interact with the medium. The rate at which gamma rays are attenuated is determined pirmarily by the atomic number and density of the shielding material, and to a lesser extent by its geometrical configuration. Consider a slab of material of a given/or certain thickness located between a narrowly collimated beam of gamma ray photons and a narrowly collimated detector, as indicated in Fig.2a. The distinctive feature of a narrow beam experiment (i.e., one having 'good geometry') is that only the photons which traverse the specimen absorber without experiencing an interaction of any kind, reaches the detector; and scattered radiation, for example, is prevented from reaching the detector. Assuming that the beam intensity incident on the slab has the value I(o), one obtains for the intensity transmitted, I(t), through the slab the value

$$I(t) = I(o)\ e^{-\mu t} \quad \ldots \quad (4)$$

where μ is the narrow beam attenuation coefficient. In this case the attenuation is found to be exponential.

(ii) Half-value thickness (HVT)

The absorption effect of radiation shielding materials may be characterized, by their half-value thickness (HVT), which is the thickness required to reduce the intensity to one-half. Similarly, the tenth value thickness (TVT) is the thickness required to reduce the intensity to one-tenth. HVT, is generally obtained from the graph by plotting the transmitted intensity vs. thickness of the absorbing materials.

(iii) Buildup factors

Two of the three main processes, the photoelectric effect and pair production effectively represent pure absorption of the gamma radiation. Furthermore, calculations of the attenuation of gamma rays would be relatively simple if the interaction processes were purely absorptive, i.e., if each collision resulted in the disappearance of a photon. Unfortunately, gamma ray interaction processes are not always completely absorptive. It is the existence of the third process, Compton scattering, that complicates the issue. The scattered photons which are not counted in narrow beam-geometry experiments still have to be reckoned with in shielding problems. However, in most shielding arrangements of practical importance, the point of interest (e.g., the human body) will consist of, in addition to the uncollidal component, a component consisting of:

(i) photons that have undergone scattering interactions in the intervening medium, and

(ii) low intensity, subsidiary radiation released locally in the intervening medium due to the interactions between the source photons and the medium.

Therefore, in the shielding arrangements most frequently encountered in practice, it is the total radiation received that is of interest; such arrangements are often described as possessing 'broad beam geometry', to distinguish them from the well collimated (a good geometry) arrangement as illustrated in Fig. 2a. A typical arrangement of source and shield having broad beam geometry is shown schematically in Fig.2b. When a broad beam of gamma photons passes through a medium, various absorption and scattering processes cause a decrease in the number of original photons present in the beam as indicated in Fig.2b. Due to the interaction between photon radiation and the absorbing materials secondary radiation arises which must not be neglected, since it may cause considerable increase of the residual radiation leaving the absorber. In such situation the simple attenuation equation (4) no longer holds for the radiation intensity behind a shield and a correction must be made for the effect of scattered photons. It is convenient to describe the difference between the results of measurements in narrow and broad-beam cases by means of buildup factors.

In studying gamma ray attenuation through a medium (Fig.2b), the concept of buildup factor is generally used. This is defined as follows[9]:

$$B = \frac{\text{Quantity of interest at a point due to the total number of particles}}{\text{Quantity of interest at a point due only to noninteracted particles}}$$

The 'quantity of interest' might be the number of particles, their energy, or the dose imparted by them; thereby giving rise to number, energy, and dose buildup factors, respectively. Therefore, the corrected attenuation equation (4) becomes

$$I(t) = I(o)\, B\, e^{-\mu t} \quad \ldots \quad (5)$$

where the buildup factor, B, takes into account the increase in radiation intensity due to scattering within the absorber, I(o) is the initial intensity of the radiation source, I(t) is the attenuated intensity after passing throughthe shielding materials of thickness t(cm) and μ is the attenuation coefficient (cm^{-1}). If all three types of interaction of gamma-ray photons with matter are included, then

$$\mu = \mu_{pe} + \mu_c + \mu_{pp} \quad \ldots \quad (6)$$

where μ_{pe}, μ_c and μ_{pp} represent the attenuation coefficients due to photoelectric, the Compton effect, and pair production, respectively. Usually, buildup factors are given at different relaxation lengths or attenuation units, μt. Like the attenuation coefficient, the value of the B varies with absorber material and radiation energy. It also varies with geometry of the source and depth of penetration in the absorbing medium. The total attenuation coefficient is calculated for a given energy by

$$\mu = 0.693/HVT \quad \ldots \quad (7)$$

where HVT is the half value thickness. In practical gamma ray shielding calculation, the buildup factor is being widely used to determine the total effect of the radiation at the point of interest. In principle the buildup factor concept is also applicale to neutron attenuation, but in practice it is found to be much less successful when applied to neutrons, and consequently it is much less important a concept in the context of neutron shielding calculation.

(iv) Removal cross section (ΣR)

The removal cross section of a material as used in shielding calculations, is the cross section which describes the probability that a fast neutron will reach a given detection point in a material without being absorbed or scattered out of the direct path, i.e., if t is the thickness of the material through which fast neutrons are passing, then the ratio of the transmitted intensity, I(t) to the incident intensity, I(o), is given by

$$I(t) = I(o)\, e^{-\Sigma_R t} \quad \ldots \quad (8)$$

where Σ_R is the macroscopic removal cross section for the material.

(v) Relaxation length (λ)

The relaxation length of any material is that thickness of material in which the incident radiation diminishes to a factor e^{-1} (i.e., $1/e = 0.368$) of its incident intensity, where e is the base of natural logarithms. This factor is also known as mean-free-path(mfp) which represents the average distance travelled by a photon between successive interactions. If $I(t)$ is the radiation (neutron or gamma ray) intensity or related quantity at a depth t within a shield, then the relaxation length λ, for the given radiation and medium, is given by

$$\frac{1}{\lambda} = -\frac{1}{I(t)} \frac{dI(t)}{dt} \quad \ldots \quad (9)$$

at the point t. As defined in this manner, the relaxation length may vary with the depth t within the shield. However, it is often found that, owing to a combination of circumstances, the radiation intensity decreases in an approximately exponential manner over a range of distances within the shield; λ remains constant over this range and the flux at t can be represented by

$$I(t) = I(o)\, e^{-t/\lambda} \quad \ldots \quad (10)$$

where $I(o)$ is the flux at the origin of the t coordinate. Comparison with the attenuation equation for gamma rays (4) and equation for neutrons (8) shows the λ is formally equivalent to $1/\mu$ or $1/\Sigma_R$, respectively. Since the relaxation length of a particular radiation in a given material is equivalent to the reciprocal of the attenuation coefficient (for gamma rays) or the removal cross section (for neutrons), the relaxation lengths (or their reciprocals) provide a good means for comparing the shielding effectiveness of different materials.

Many research workers have tried in various ways to determine the structural properties and nuclear parameters for different shielding materials. To put things in perspective, a critical review of the earlier work is envisaged.

1.4 Review of earlier works

An excellent review of the methods for the preparations of different types of concretes to be used in shielding various nuclear installations and the data pertaining to constituents and characteristic properties of the concretes has been made by Hungerford et al[10].

Schmidt[11] has made an indepth review of fast neutron (<15 MeV) transmission through concretes. This review includes the details of different types of concrete and their compositions used for the studies of transmitted spectra, dose rates and evaluation of relaxation lengths- both experimental and computational.

In a review article, Egorov[12] has described the attenuation of fast neutron through layers of different shields. The shielding materials for which relaxation lengths have been measured for neutrons and gamma rays are: polyethylene, graphite, iron, lead and serpentine concrete. Experiments have shown that the relaxation lengths of neutrons and gamma radiation in serpentine concrete are about 12 cm and 15 cm respectively.

Relaxation lengths for fast neutron (>0.5 MeV), and gamma rays, total dose rates, and buildup factors relevant to polymer-based concretes were measured by Belayakov[13].

The attenuation of fast neutrons in ilmenite concretes has been studied by Adams and Lokan[14,15]. The densities of concretes studied varied from 2.4 to 3.8 g/cm^3. No comparison has been made between ordinary concrete and ilmenite concrete.

The barium serpentine cement concrete used by Voskrenesky et al[16] has a density of 4.6 to 4.75 g/cm^3. They have studied several engineering properties and relaxation lengths in the shielding materials.

Veselkin et al[17] have studied the reactor neutron transmission through 15 different concrete compositions. These are: ordinary concretes (densities: 2.32, 2.35, 2.42 g/cm^3), hydrate concretes (densities: 2.36, 2.41, 2.60, 2.90, 3.15, 3.20 g/cm^3), heavy concretes (densities: 3.1, 3.3, 5.0, 6.0 g/cm^3), serpentine concrete (density: 2.36 g/cm^3) and iron-serpentine concrete (density: 3.6 g/cm^3). Dose buildup factors were measured as an index of shielding properties.

Dubrovski et al[18] have studied the shielding properties of stone concretes by measuring the relaxation lengths, and they have established that the stone concretes (density: 2.38 g/cm^3) are better than the ordinary concretes (density: 2.2 g/cm^3) from radiation shielding point of view.

Transmitted spectra for neutrons through concrete, water, paraffin, iron and boron-loaded polyethylene have been computed by Makra et al[19]. They, however, did not compare the relative efficiencies for different types of concretes. Transmission of fission and reactor neutron through several materials (water, iron, lead, concrete + iron + boron) has also been measured by Makra et al[20]. Again, no attempt seems to have been made by them for comparing the relative efficiencies of these different types of concrete.

Szmendera et al[21] have computed the total attenuation (neutron +gamma) for different types of concrete aggregates: basalt - magnetite, barite - magnetite, serpentine - barite and serpentine - magnetite, with the densities ranging from 2.44 to 4.42 g/cm^3.

Dikarvev[22] has studied the shielding properties of concrete. He has measured the relaxation lengths of neutron and gamma rays in ordinary and limonite concretes. His investigation shows that for the given neutron and gamma ray spectra, limonite concrete has better shielding properties than ordinary concrete.

Vasilev et al[23] have studied the attenuation of reactor radiations in serpentine sand. They have measured the relaxation lengths of fast neutrons, thermal neutrons and gamma rays in serpentine sand and compared their data with the relaxation lengths of such radiations in serpentine concrete and iron-briquelles concrete. They have found that the shielding properties of monolithic serpentine blocks are considerably better than those of iron-briquelles and some what better than those of serpentine concrete.

Attenuation of neutron and photons transmitted through ordinary concrete were studied by Uwamino et al[24]. Source neutrons and photons were obtained from Californium-252 fission source. Relative shielding properties of amonia meta tungsten for neutron and gamma rays from ^{252}Cf fission source have been studied by Senffle and Philbin[25].

Dose buildup factors for a plane non-directional source of Co-60 were obtained experimentally for plane parallel barriers composed, respectively, of water, ordinary concrete and heavy concrete by Furuta et al[26]. They have measured the dose buildup factors in water (ρ=1.0 g/cm^3), graphite (ρ=1.77 g/cm^3), ordinary glass (ρ=2.49 g/cm^3), aluminium (ρ=2.75 g/cm^3), iron (ρ=7.83 g/cm^3) and lead (ρ=11.3 g/cm^3). Similar study has also been performed by Tamura and Tsurue[27] using a Cs-137 source.

The broad beam attenuation coefficients of 4-32 MV X-rays in ordinary concrete, heavy concrete, iron and lead have been studied by Maruyama et al[28].

Broerse and Werven[29] have measured the absorption characteristics of various materials for 3 MeV and 15 MeV neutron. Attenuation curves are reported for water, concrete, paraffin and borated laminated wood.

Shielding characteristics of concrete with iron (ρ=2.25 g/cm^3) has been studied with fission source by Palfalvi[30]. Polyethylene with 1% by weight boron has also been used as the shielding material.

Attenuation of 14 MeV neutrons in shields of concrete and paraffin wax has been studied by Bozyap and Day[31]. Removal cross sections and buildup factors were measured using a different attenuation model.

Experimental attenuation curves have been measured for uncollimated bremsstrahlung radiation, incident on ilmanite-loaded concretes of density 2.88, 3.85 and 4.33 g/cm^3 by Lokan et al[32].

Gamma ray buildup factors in ordinary concretes have been measured by Eisenhauer et al[33] and Husain et al[34] using point isotropic ^{60}Co source. Harris et al[35] have studied shielding characteristics of ordinary concrete having density of 2.25 g/cm^3.

Megahid et al[36] have measured the attenuation and distribution of gamma ray doses in a multilayered shield consisting of water, iron and ordinary concrete. The relaxation length of gamma rays in ordinary concrete was evaluated. They have also measured the attenuation of reactor thermal neutrons in a bulk shield of ordinary concrete. In addition the thermal neutron relaxation lengths in ordinary concrete were derived for disc collimated beam and infinite plane mono directional sources.

Attenuation characteristics of 6.2 MeV source photons penetrating concrete, iron and lead shields have been studied by Bishop and Marafie[37]. They have measured the exposure buildup factors.

Attenuation of 14 MeV neutrons from a point source in concrete shields (density: 2.3 g/cm^3) has been studied by Robsen and Kroon[38].

The relative effectiveness for varying hydrogen contents in concretes have been computed by Belayakov et al[39]. Concretes of density 2 to 2.4 g/cm^3 and concrete with organic substances (polymer - cement and polymer) have been considered.

The attenuation of fast neutrons and gamma rays through shields have been reported for magnetite concretes, ilmenite concretes, serpentine concretes and ordinary concretes by several authors[40-45]. They have measured the dose buildup factors, relaxation lengths, linear attenuation coefficient.

The use of suitable additives in concrete is encouraged to enhance the shielding properties. The only additive used in concrete is boron either in elemental form or in other forms such as

i) boron-bearing aggregate like colemanite;

ii) pyrex glass ground to sand;

iii) boric acid, borohydrides or complex borates dissolved in mix water (water used for casting concretes); and

iv) boron in cementing agent.

Since the presence of boron retards the setting process of concrete, generally very good care has to be taken in the preparation of concrete by confining the amount of boron to about one percent. To enhance the shielding capabilities and suppress the unwanted capture gammas, boron is used in aggregates containing a large amount of bond or semi-bond water.

The use of boron or its compounds in concretes, the preparation and properties of such concretes have been thoroughly reviewed by Hungerford et al[10]. The properties of boron and boron compounds have also been reviewed by Akerhielm[46]. Hungerford[47] has described the borated materials used in Enrico Fermi Reactor biological shield.

Broader et al[48], have studied the shielding properties of borated-heat resistant chromite concretes. They estimated the neutron and gamma dose rate relaxation lengths as a function of shield thickness. Broader et al[49] also studied the shielding properties of concretes containing boron. They used concretes of densities 1.8 and 2.2 g/cm^3. They have also investigated concretes of different densities (2.3 to 4.2 g/cm^3) containing different weight fractions of boron and measured the relaxation lengths and dose buildup factors.

The shielding properties of borated concretes have also been measured by Dubrovski et al[50]. They concluded that having more than 15 kg/m^3 of boron in concrete does not produce any additional effectiveness. Further, boron is less effective in denser concretes. They also found that boration is most effective in heat resistant non-hydrogeneous concretes.

Veselkin et al[51], have studied the effect of boron and iron as additives on concretes and the relevant shielding properties for reactor neutrons. They used 20 kg/m^3 of boron carbide in ordinary concrete (density 2.36 g/cm^3) and 72 kg/m^3 of boron carbide in serpentine concrete (3.36 g/cm^3).

Kelly[52] and Quetier Monique[53] have described the use of neutron absorbing materials in their review.

Mashkovich and Tsypin[54] have discussed the present trend towards experimental shielding research. Petrove[55] has discussed the possibility of applying an experimental method for optimisation of multilayered biological shields. They concluded that the method provides one with a careful tool for transforming the results of simulating experiments into practical design of physical facilities. Petrove[56] has also described the optimisation of two - layered - shield for gamma and neutron shielding based on computations.

1.5 The present work

The Bangladesh Atomic Energy Comission has recently commissioned its 1st 3MW(t) TRIGA Mark-II research reactor at the Atomic Energy Research Establishment (AERE), Savar and the proposed nuclear power reactor at Rooppur, Pabna is likely to be implemented in near future. The potential hazard associated with nuclear installations due to ionizing radiations has already been mentioned. In nuclear reactors, adequate shielding is applied to attenuate the intensity of emitted radiations considerably. The biological shield of a nuclear reactor is required to attenuate both gamma rays and neutrons emerging from the core. The energies of these radiations cover a wide range. Generally, materials having high densities are used as shielding materials. The present literature review reveals the different types of concretes considered for biological shielding of reactors. Heavy concrete with density greater than that of ordinary concrete (density: 2.3 g/cm^3) is suitable for reactor biological shield. High density may be obtained in concretes by the use of the following aggregates: barite, magnetite, ilmenite, hematite, limonite, steel punchings or shot, ferrophosphorus. The high density aggregates will improve both the neutrons and the gamma ray shielding effectiveness of concretes. Aggregates containing a large amount of bond or semi-bond water and/or boron contents improve neutron shielding effectiveness. Boron increases the neutron shielding effectiveness of concretes and also acts as a suppressor of secondary (capture) gamma rays. The presence of heavy constituents and hydrogen (mix water) in the concretes provides for an adequate solution for the problem of attenuating gamma rays and fast neutrons. However, in order to capture thermal neutrons without the ensuing production of hard (penetrating) gamma rays, use must be made of elements with a very large neutron cross section with the accompanying emission of only soft gamma rays which are readily absorbed within the shield. Boron, containing isotope ^{10}B is such an element. For use in concretes the boron compounds must be added to the other constituents to increase the probability of neutron capture. A proportion of 0.9 to 1% of boron in relation to the volume of the concrete is recommended by several authors[10]. According to some authors, increasing the boron content beyond 1.5% does not result in any increase in the effectiveness of shielding. On the contrary it creates setting problems for the concretes. Boron may be added to ordinary concrete and high density concretes in a variety of ways to enhance the thermal attenuating properties of the concrete and to suppress secondary gamma ray generation. Boron-10, because of its large thermal-neutron capture cross-section (4000 barn) competes quite successfully with other elements in capturing neutrons when added to concrete even in small quantities. The reaction is $^{10}B(n, alpha)^{7}Li$, which emits a 0.49 MeV capture gamma-ray. This low energy gamma-ray is easily absorbed in the concrete. The detailed literature survey reveals that in the past decade extensive research was carried out on heavy concretes using two or more combinations of magnetite, ilmenite, limonite, hematite, bryte, etc., and the resultant aggregates were successfully applied to shield the nuclear installations. The shielding performance depends on the properties of component substances; and by a suitable choice of constituents it is possible to optimize the shielding characteristics for various types of radiation. The present study was undertaken to design and fabricate concrete shields indigenously with the objective of avoiding costly imports and also to develop local know how. Such a study should be of particular interest to Bangladesh because of the availability of magnetite and ilmenite as by-products in the beach sand processing pilot plant at Cox's Bazar. It may be mentioned here that concrete shields using magnetite sand from Cox's Bazar have been studied, designed and fabricated by Ahmed et al[57,58] for possible use as the biological radiation shield. They have measured only point isotropic gamma ray (^{60}Co) buildup factors in two aggregates concrete shield (density: 2.72 g/cm^3). The concrete slabs used by them were made of cement, mixed sand and stone chips in the ratio of 1:2:3 (mixed sand composed of magnetite and Sylhet sand in the ratio of 3:2). In order to use concretes prepared from locally available materials as a reactor biological shield, the detailed research and development works on the evaluation of structural and nuclear shielding properties of concretes undertaken. This thesis describes the investigation carried out pertaining to

(1) fabrication of concrete shielding by mixing magnetite, ilmenite, stone and Sylhet sand in standard proportion with and without boron additives;

(2) evaluation of physical and structural properties of the concrete; and

(3) evaluation of the radiation shielding properties of the concretes.

In chapter 2, the fabrication of concretes of different types are described. Several physical and structural properties were also evaluated for these concretes.

The experimental method used in this study is described in chapter 3. The chapter 4 indicates the experimental results obtained in the present investigation. The summary and conclusion of the whole study as well as future research programme are given in chapter 5.

CHAPTER 2

FABRICATION AND EVALUATION OF STRUCTURAL PROPERTIES OF CONCRETES

2.1 Introduction

Concrete is an inexpensive, easily handled material. Nuclear properties of concretes may be enhanced by using special aggregates, adding heavy metals to the mix or boron additives to it. Any concrete with a water content of $\sim$ 5% makes a good neutron shield and the use of boron and heavy aggregates enhances the neutron-shielding properties of concrete. The standard composition (mixing ratio) of ordinary concrete with a conventional method of preparation is 1 part cement: 2 parts sand: 4 parts rock aggregate by volume. These ratios vary for special and high-density concretes according to need. Variation of size of the aggregate and the sand particles as well as the mix proportions will produce variations in appearance, density, and strength properties. Use of special aggregates, sand, cements, or additives to the mix water may enhance both the physical as well as nuclear properties of the concrete.

The biological shield concrete must provide the desired primary shielding against the penetrating radiations and also possess the adequate load bearing characteristics. Thus, for any new type of concrete to qualify itself as biological shield its structural properties as well as nuclear properties should fulfill the established criteria and its properties can be changed to suit special needs. There are several properties of interest which one must consider while utilizing concretes for radiation shielding purposes. The desired physical and mechanical properties include: (1) density; (2) compressive strength; (3) thermal conductivity; (4) hydrogen content; and (5) rate of dehydration with temperature and aging. The nuclear parameters of interest are: (1) gamma-ray attenuation coefficient which governs the characteristic relaxation (or e-folding) length and half value thickness (HVT) in the material; (2) removal cross section which governs the characteristic neutron relaxation length in the material; and (3) the hydrogen content which makes possible the slowdown of fast neutrons to lower energies. The thermal neutron attenuation properties should be increased to the maximum attainable and the production of capture hard gamma rays should be suppressed as much as possible. To achieve it the concrete may be loaded with boron-containing materials. The fabrication of concretes shield with and without additives and evaluation of their structural properties are described in this chapter. In this study, boron in the form of boric acid (H_3BO_3) is added (1% by vol.) in the concrete shield to enhance the neutron shielding effectiveness.

2.2 Fabrication of concretes

Concrete is artificial brick manufactured from a mixture of binding inert materials in combination with water. Concrete is considered as a chemically combined mass where inert material acts as a filler and the binding material acts as a binder. The inert materials used in concrete are termed as 'aggregates' and are used in concrete to give strength to it. Usually they are graded coarse and fine. The aggregates give volume to the concrete, around the surface of which the binding materials adheres in the form of a thin film. In nuclear application use of high density aggregates increases the density of the concrete, an important consideration for shielding. Certain aggregates may increase the fixed-water content in the concrete, another property desirable from a nuclear view-point. Impurities in the aggregates or other ingredients in concrete may seriously affect the setting, aging, physical and nuclear properties of concretes; undesirable minerals produce deleterious effects on the setting characteristics, which in turn affects the aging and strength properties. Debrish and dirt particles in unwashed aggregate may cause concrete to crack upon setting and crumble with age; gas pockets formed by too great a heat generation during the setting process may create voids which weaken the concrete and decrease its shielding value. It is important that no voids be formed in the pouring of concrete shields. Proper consistency will aid in eliminating this problem. Thus, the homogeneity and consistency should be maintained. The proportion of water in the mix is vital to its consistency. A good concrete shield should have a slump of around 10 to 15 cm. In theory, the voids in the coarse aggregates are filled up with fine aggregates and again the voids in the fine aggregates are filled up with the binding materials. Finally, the binding material, as the name implies, binds the individual units of aggregates into a solid mass in presence of water.

Concrete is usually prepared by thoroughly mixing portland cement, sand and stone chips. The casting of the mixture is done in a mould of desired shape and size and allowed to hydrate or cure in a humid atomosphere. After a 28 day cure, the concrete reaches about 80 to 90% of its ultimate strength. Concrete at this point contains $\sim$3.5% by weight of bond or hydrated water and $\sim$3.5% by weight of free water. In this study, concrete slabs with and without boron additives with different composition were prepared in the concrete laboratory, Department of Civil Engineering, Bangladesh University of Engineering and Technology (BUET). The American Society of Testing Materials (ASTM)[59] procedure was followed for making the concrete slabs. The concrete slabs were made with cement, sand and stone chips in the ratio of 1:2:3 by volume. In addition, identical type concrete slabs used in the reactor biological shield of 3MW TRIGA Mark-II research reactor was also made with cement, sand (20% ordinary Sylhet sand + 40% ilmenite sand + 40% magnetite sand) and stone chips in the ratio of 36:100:100 by volume. The detailed composition of cement, sand and stone chips are given in Table 1. From the prepared mixture, twenty slabs each having dimension of 12.5cm x 25cm x 12.5cm were fabricated for radiation shielding studies. In addition, nine solid cylindrical (15 cm dia. x 30 cm length) slabs were made and their structural properties were studied in the concrete laboratory at BUET. The basic properties of the aggregate used for the preparation of concretes and the chemical analysis of sand are shown in Tables 2 and 3 respectively.

2.3 Structural properties of concrete

There are several physical and structural properties of interest which one should consider while evaluating the use of concretes for radiation shielding[59-61].

2.3.1 Density:

The density of a concrete plays a vital part in its nuclear performance, because it is a measure of the atomic concentration of the constituent elements. High density is a desirable property for stopping both gamma rays and neutrons. The greater is the density of the material the more will be the number of nuclear interactions and hence it becomes more effective in absorbing radiation. The density may be increased by use of heavier aggregates. The density of a given mix formula may be increased a few percent by addition of heavy aggregates to the normal mix without affecting the properties of the resultant concrete.

Table 1

Summary of concrete compositions

Type of Concrete	Symbol	Mix ratio by volume			Boron additives % by volume
		Cement	Sand	Course aggregates (Stone chips)	
Ordinary concrete	01	1	2	3	-
Boron-loaded Ordinary concrete	01B	1	2	3	1
Ilmenite concrete	I1	1	2	3	-
Boron-loaded ilmenite concrete	I1B	1	2	3	1
Magnetite concrete	M1	1	2	3	-
Boron-loaded magnetite concrete	M1B	1	2	3	1
Ilmenite-Magnetite-ordinary mixed concrete	IM01	36	100*	100	-
Boron-loaded ilmenite-magnetite-ordinary mixed concrete	IM01B	36	100*	100	1

*Ilmenite sand = 40%; Magnetite sand = 40%;

Sylhet (ordinary) sand = 20%

Table 2

Basic properties of aggregate

Properties	Fine aggregate			Course aggregate	
	Sylhet	Ilmenite	Magnetite	White	Black
Finness modulus	2.84	0.74	0.68	-	-
Bulk unit weight (kg/m^3)	1554	2563	2563	1538	1602
Bulk specific gravity (S.S.D. basis)	2.54	4.78	4.55	2.54	2.86
Absorption (%) (oven dry basis)	1.52	0.10	0.46	0.49	0.34
Effective absorption (%)	1.00	0.04	0.38	0.47	0.33

Table 3

Chemical analysis of sand

Name of sand	Organic content (%)	Chloride content (%)
Sylhet sand	0.60	-
Ilmenite sand	0.09	0.06
Magnetite sand	0.21	0.07

The overall density of concretes varies from 1.3 g/cm^3 for certain aggregate to 4.8 g/cm^3 for certain steel-loaded iron ore-aggregate concretes[10].

2.3.2 Compressive strength:

Compressive strength is of paramount importance since most of the concrete shield constitute the entire structural assembly. Preservation of individual shield integrity over its expected lifetime is, therefore, essential. Proper use of aggregates can enhance the strength of concrete. For neutron shielding, where a high water content is desirable, strength may have to be sacrificed to some extent in order to obtain the desired water content. A recommended mixing ratio of the dry ingredients for a high-strength portland concrete is 1 cement: 2 sand: 4 aggregate by volume with a water-cement ratio of 0.53 by weight. This formula will produce a concrete with compressive strength lying between 210 kg/cm^2 and 420 kg/cm^2 under properly controlled curing conditions. The increasing cement content will tend to increase the strength of concrete, while increasing the water content will tend to decrease its strength. For heavy and other special concretes the desirable water-cement ratio may be less than 0.53. Furthermore, curing or proper aging of concrete is important both from nuclear as well as structural points of view. A 28 day aging period under moist conditions will develop the maximum strength characteristics[60-61].

2.3.3 Thermal conductivity:

Thermal conductivity is the rate at which heat passes through cm-cube of a material when there is a unit temperature difference between the two faces of the material. The thermal conductivity of ordinary concrete varies between 0.0008 to 0.004 cal/s-cm-^{o}C, depending on the mix. In general, heavy aggregates added to ordinary concrete has a conductivity[10] of around 0.004 cal/s-cm-^{o}C.

2.3.4 Hydrogen content:

Hydrogen occurs in concrete mainly in the form of fixed water and small amounts of organic impurities. The main shielding effectiveness of concrete for neutrons is due to the water content of concrete[10].

2.3.5 Dehydration:

Ordinary concrete will retain its water content for a long period of time. Dehydration is so slow at ordinary temperature that many concrete still retain half of their original water content after a period of 20 to 30 years. Heat and radiation rapidly accelerate the dehydration process. At temperature above 200^{o}F dehydration is noticeable, and above 600^{o}F it becomes very rapid[10].

The densities and compressive strength including tensile strength and modulus of elasticity were determined with cylindrical shaped slabs of 15 cm dia. X 30 cm length in the laboratory. For each type of measurement, a minimum of three samples were fabricated. The above properties were evalauated using conventional techniques prescribed by ASTM. All properties were evaluated for all types of concretes after 28 days curing. The results are summarised in Table 4. More detailed information about aggregate, cement, concretes and their relevant properties are found elsewhere[59-61].

Table 4

Structural and physical properties of 30 cm x 15 cm dia. cylinder concrete aged 28 days

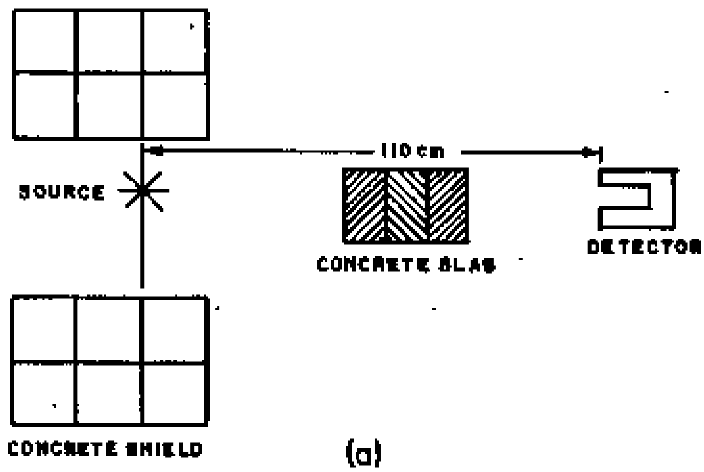
Type of concrete	Density (g/cm^3)	Compressive strength (kg/cm^2)	Modulus of elasticity (kg/cm^2)	Splitting tensile strength (kg/cm^2)
01	2.33	315.00	27.09×10^4	27.86
01B	2.24	256.90	22.87×10^4	21.56
I1	2.88	217.70	22.47×10^4	16.10
I1B	2.78	190.40	20.63×10^4	12.60
M1	2.82	211.54	19.39×10^4	25.48
M1B	2.72	198.21	17.52×10^4	21.08
IM01	2.75	238.01	33.85×10^4	33.35
IM01B	2.69	233.80	27.44×10^4	30.12

CHAPTER 3

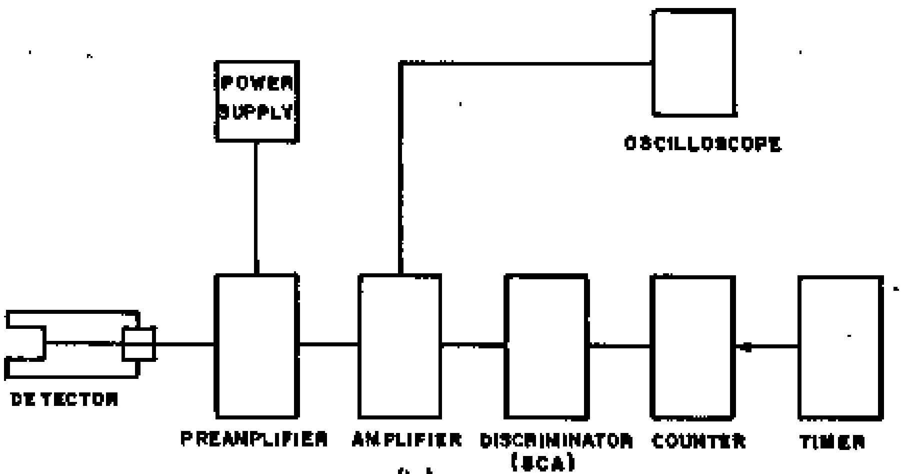
EXPERIMENTAL METHOD

3.1 Experimental arrangement

Even if the theoretical understanding of shield behaviour are sufficiently accurate and extensive to allow detailed calculation of reactor shield design, common prudence would require the experimental determination of shield behaviour prior to reactor shield design. Therefore, it is understandable that measurements of the shielding properties of matter in the bulk have provided the most important basis for the study of neutron and gamma ray attenuation. The method used for the detection of neutron and gamma radiation are almost exclusively based on the ionization produced by particles in their passage through matter. In the interaction of gamma radiation with matter, electrons are produced, which in their turn produce ionization effects that can be detected. Similarly, neutrons are detected through nuclear reaction which result in energetic charged particles (such as protons, alphas and so forth) that reveal their presence by ionization they produce. The experimental arrangement for the measurement of attenuation of radiations (both neutrons and gamma rays) consists of a radiation source, a detection system and slabs of the material being investigated. A detection system can be considered to be made up of two parts: a detector and a recording system. The interaction of the radiation with the system takes place in the detector. The recording system



(a)



(b)

Fig. 3. Experimental arrangement: (a) a block diagram of experimental set up and (b) elements of a typical signal chain for pulse counting.

receives the output from the detector and after amplification and subsequent discrimination records the number of events. A block diagram of the experimental arrangement for the attenuation measurements is shown in Fig. 3a. In our experiment, the broad beam geometry was employed in order to include the maximum amount of scattered radiation in the transmitted beam. A BF_3 gas proportional detector was used to monitor neutron components, while a NaI (Tl) scintillation detector was used to monitor gamma ray components in this study. The distance between the source and the detector was 110 cm. The incident neutron or gamma-ray intensity was first measured without any concrete slab between the source and the detector. Then the first slab of the shielding material was placed nearest to the detector and the transmitted intensity was measured by the detector. The next slabs were successively introduced in front of the slabs already placed. In this way the shielding thickness was increased, while the detector remained in the same position. The measurements were repeated for the addition of each 5 cm thick slab of concrete up to a maximum of 100 cm (totalling 20 identical slabs). The output from the detector was fed into the counting system. Figure 3b is a schematic diagram of the counting system. The detailed description of the radiation source and the detectors used in the shielding experiments are furnished in the sections 3.2, 3.3 and 3.4. Some of the electronics associated with the counting system are described below.

Preamplifier

The output of the detector appears as a pulse of voltage or charge which is so small that it is practically impossible to deal with the signal pulses without an amplification step. Therefore, the first element in a signal processing chain is often a preamplifier, provided as an interface between the detector and the pulse processing and analysing electronics that follow. From the stand-point of signal-to-noise ratio, it is always preferable to minimize the capacitative loading on the detector, and therefore long interconnecting cables between the detector and preamplifier should be avoided as much as possible. The preamplifier should, therefore, be located as close as possible to the detector. Another function of the preamplifier is to quickly terminate the capacitance, and therefore to maximize the signal-to-noise ratio. It also serves as an impedence matcher, presenting a high impedence to the detector to minimize loading, while providing a low impedence output to drive succeeding components. The preamplifier conventionally provides no pulse shaping and its output is a linear tail pulse. The rise time of the output pulse is kept as short as possible, consistent with the charge collection time in the detector itself.

The signal processing section of preamplifiers intended for either solid-state diode detectors or gas-filled ionization or proportional counters is quite similar. However, preamplifier intended for proportional counter applications must be designed to withstand several thousand volts, whereas solid-state detectors seldom require bias supplies of more than a few hundred volts. Preamplifiers intended for scintillation counters are usually quite different. Because the signal level from a photomultiplier (PM) tube is rather high, the gain and noise specifications of a scintillation preamplifier is different from those used with proportional counter. Although it is quite possible to use a scintillation counter without a preamplifier in many applications, it is convenient to include a preamplifier simply to avoid the charges which can occur in the time constant of the equivalent anode circuit of the PM tube if it is connected directly to a linear amplifier or other measuring circuits.

The preamplifier is also normally used to supply bias voltage to the detector. Arrangements can always be made to supply voltage to the detector independent of the preamplifier, but it is usually convenient to do so through the preamplifier. A signal cable between the preamplifier and detector provides both the voltage to the detector and the signal pulse to the input of the preamplifier.

Most preamplifiers in use today are charge sensitive, and provide an output pulse with an amplitude proportional to the integrated charge output from the detector. Canberra Model 2001 preamplifier was used in our experiment. Its salient features are as follows:

- Low noise-better than 600 eV

- Ideal for high rate γ-ray counting applications
- Charge-sensitive
- First rise time-better than 40 nsec
- Pole/zero cancelled unipolar output pulse
- Gain stability-better than $\pm$ 0.005%

High-voltage supplies (E.H.T. supply)

All radiation detectors require the application of an external high voltage for their proper operation. This voltage is conventionally called "detector bias", and high-voltage supplies used for this purpose are often called detector bias supplies. Some characteristics of detector bias supplies which can be important in specific applications are:

- The maximum (and minimum) voltage level and its polarity;
- The maximum current available from the supply;
- The degree of regulation against long-term drifts due to changes in temperature or power line voltage; and
- The degree of filtering provided to eliminate ripple at power line frequency or other low-frequency noise.

The sophistication required of the bias supply varies greatly with the detector type. Typical scintillation high-voltage supplier must be capable of providing up to 3000 volts with a current of a few miliamperes. The output must also be well regulated to prevent gain shifts in the PM tube due to drifts in the high-voltage level. Bias supplies for proportional counters must also supply relatively high voltages, but the current demands are considerably less. However, the degree of regulation and filtering is again important because any high voltage fluctuations appear superimposed on the signal.

Canberra Model 3102 High voltage power supply was used in this study. It's noise and ripple characteristics make it suitable for operation with a wide variety of detectors. By design, the Model 3102 can supply all types of detector requiring 2 kV or less. It's salient features are as follows:

- o Regulated 0 to $\pm$ 2000 VDC, 2 mA ouput
- o Noise and ripple <2 mV peak to peak
- o Overload and short circuit protected

Linear amplifer

The linear amplifier accepts input pulses, often of either polarity, and shapes them into linear pulses with standard polarity and span-width. They are designed to be capable of amplifying pulses with rise time of the order of 10^{-8} second. The amplification factor or gain required varies over a wide range, but is typically between 100 and 50000. The gain is normally adustable over a wide range through a combination of coarse and fine controls. A good amplifier should have low input noise.

Canberra Model 2012 amplifier was selected for the present study. It has all the characteristics necessary to make it useful with scintillation detector and proportional counters. Its salient features include high gain, low noise, selectable time constants, and count rate optimization. The gain range, temperature stability and non-linearity specifications make this Model 2021 to be suitable for many applications requiring long counting times.

Differential discriminator (single channel analyser)

In order to count pulses properly, the shaped linear pulse must be converted into logic pulses. A differential discriminator or single channel analyzer (SCA) produces a logic pulse only if the amplitude of the input linear pulse lies between two levels. The action of the unit is therefore to select a band of amplitudes or 'window' in which the input amplitude must fall in order to produce an output pulse. In counting systems, the SCA can serve to select only a limited range of amplitudes from all of those generated by the detector. This is governed by the lower level discriminator (LLD) and upper level discriminator (ULD). The discrimination level is normally adustable by a front-panel control. In many counting situations, the LLD and ULD are chosen to exclude very small noise pulses at the lower end and very large pulses beyond the range of interest at the upper end so that the maximum sensitvity for counting detector pulses of all sizes is realized. A Canberra Model 2030 SCA was selected for this study.

Scalars or counters

As the final step in a counting system, the logic pulses must be accumulated and their number recorded over a fixed period of time by a timer. The function of a timer is simply to start and stop the accumulation period for an electronic counter or other recording device. The device used for this purpose may be a simple digital register which is incremented by one count each time a logic pulse is presented to its input. In nuclear pulse counting applications such devices are often called scalars.

Canberra Model 2071A Dual counter/timer was chosen for the present study. The Canberra Model 2071A Dual counter/timer provides, two eight decade counter, a crystal time base, and presetting logic. Normal operation is as a preset timer and event counter. The Model 2071A accommodates input counting rates of 100 million counts per second (100 MHZ) for negative inputs and 25 million counts per second (25 MHZ) for positive inputs.

3.2 Radiation source

Since reactor shielding involves the attenuation of fast neutrons and gamma radiations, the experiment was carried out using a ^{252}Cf fission source which emits both fission neutrons and fission gamma rays. The ^{252}Cf source is being widely used in studies of shielding properties[25]. The strength of the source used in this study was 2 μg on November 20, 1985. This source was chosen because of its

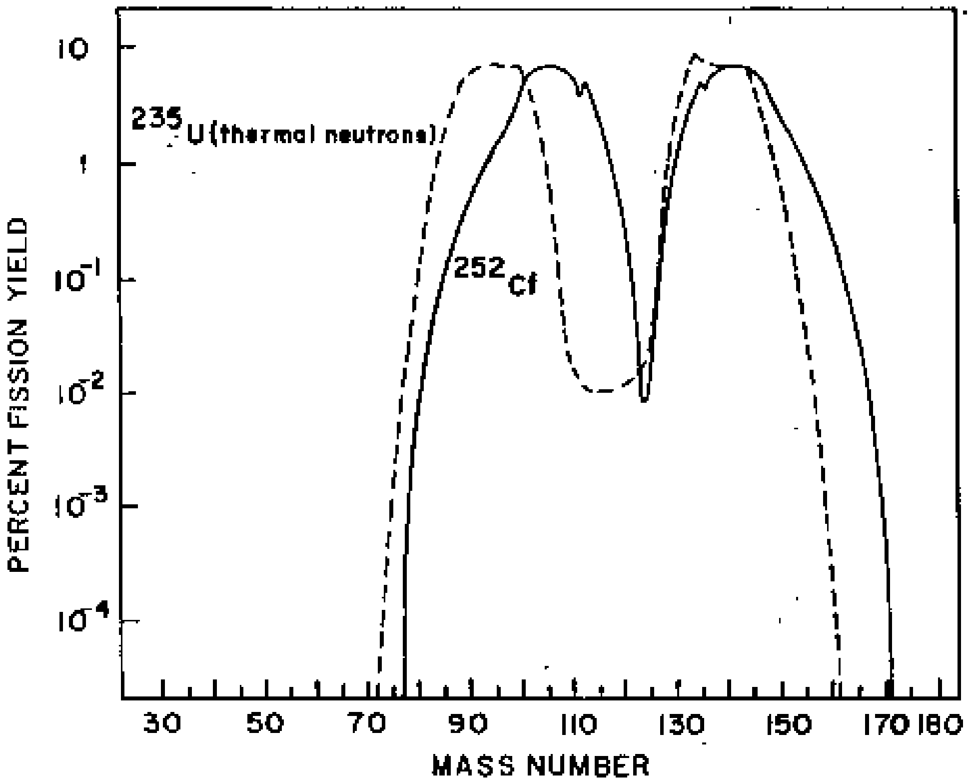
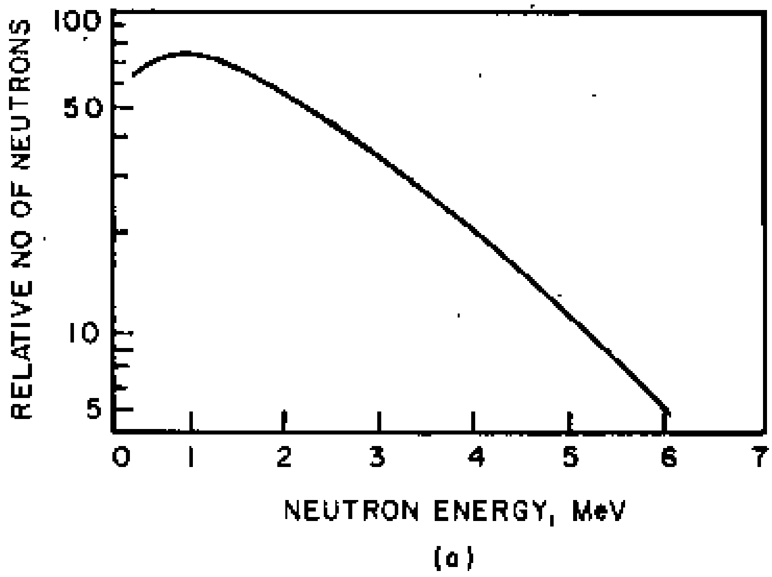


Fig. 4. The mass distribution of fission fragments and the corresponding distribution from fission of ^{235}U induced by thermal neutrons in a nuclear reactor.

fission yield being identical to the fission yield of ^{235}U which undergoes nuclear fission with thermal neutrons in a reactor core[3]. A typical fission yield distribution for both ^{252}Cf and ^{235}U is shown in figure 4. The average neutron energy of the fission spectrum due to ^{252}Cf has a mean value of 2.3 MeV, on the other hand, the average neutron energy of the fission spectrum due to ^{235}U has a mean value of 2 MeV.

Californium-252, an alpha emitter, undergoes spontaneous nuclear fission at an average rate of 10 fission for every 313 alpha transformation. The half-lives of ^{252}Cf for alpha emission and spontaneous fission are 2.72 years and 85.5 years respectively, its effective half-life, therefore, being 2.65 years. The neutron emission rate has been found to be 2.34×10^6 neutrons per second per microgram of ^{252}Cf. The gamma ray production rate is 1.3×10^7 photons per second per microgram of ^{252}Cf. A typical neutron and gamma ray fission spectrum of ^{252}Cf is shown in fig. 5. Its physical parameters[62] are shown in Table 5.



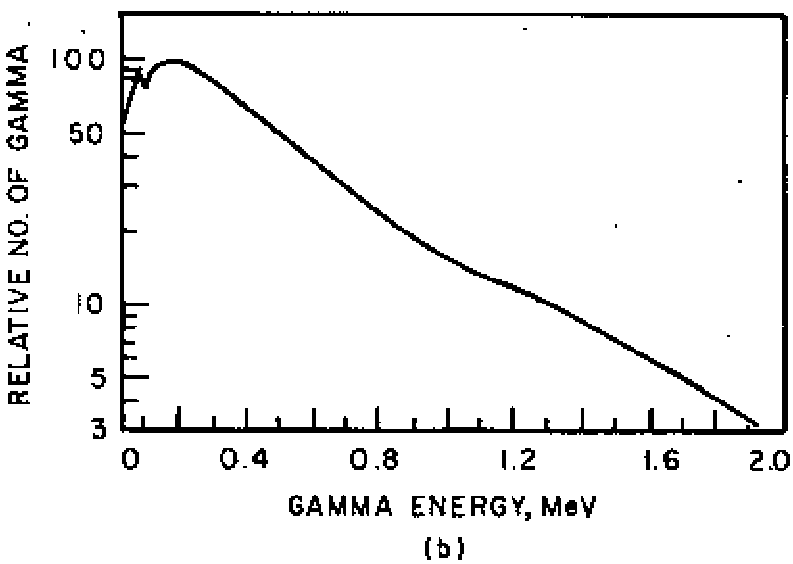


Fig. 5(a) Neutron energy spectrum of ^{252}Cf and

(b) Gamma energy spectrum of ^{252}Cf.

Table 5

Physical properties of ^{252}Cf

Atomic number	98
Atomic weight	252
Half-life for alpha emissions	2.73 y
Half-life for spontaneous fission	85.5 y
Half-life (effective)	2.65 y
Fraction of decays by alpha emission	97%
Fraction of decays by spontaneous fission	3%
Average number of neutrons per spontaneous fission	3.8
Decay heat (5% from fission, 49% from decay)	39 $\mu W/\mu g$
Specific activity	530 $\mu Ci/\mu g$
Neutron emission rate	$2.3 \times 10^{6} . s^{-1} . \mu g^{-1}$
Neutron spectrum	see Fig.5a
Photon emission rate	$1.3 \times 10^{7} . s^{-1} . \mu g^{-1}$
Gamma radiation spectrum	see Fig.5b.

3.3 Neutron detector

Several techniques are available for measuring fast neutrons[7]. The commonly used ones are nuclear emulsion, threshold detectors, ^{3}He or ^{6}Li sandwich detectors, proton-recoil detectors, semi-conductors with proton radiators and liquid scintillation detectors. In addition, there exist several fast neutron detectors based upon the slowing down of fast neutrons and then measuring these slow neutrons. A widely used detector for slow neutrons is the BF_3 proportional detector. A BF_3 gas proportional detector was used to monitor fast neutron components in this experiment. The specifications of this detector are shown in Table 6. In this device, boron trifluoride serves both as the target for con- version of slow neutrons into secondary particles as well as a counter gas. The detection efficiency, $\varepsilon(E)$ for the neutron incident along the axis of a BF_3 detector is given approximately by[7]

$$\varepsilon(E) = 1 - \exp [- \Sigma a(E) \, L] \quad . . . \qquad (11)$$

where $\Sigma a(E)$ is the macroscopic absorption cross section of ^{10}B at energy E, and L is the active length of the detector. Using equation (11), the calculated efficiency for our 30 cm long BF_3 counter is 91.5% at thermal neutron energies (0.025 eV) but drops to 3.8% at 100 eV. Hence, a BF_3 detector exposed to neutrons with mixed energies will respond principally to the slow neutron component and will have an extremely low detection efficiency for fast neutrons.

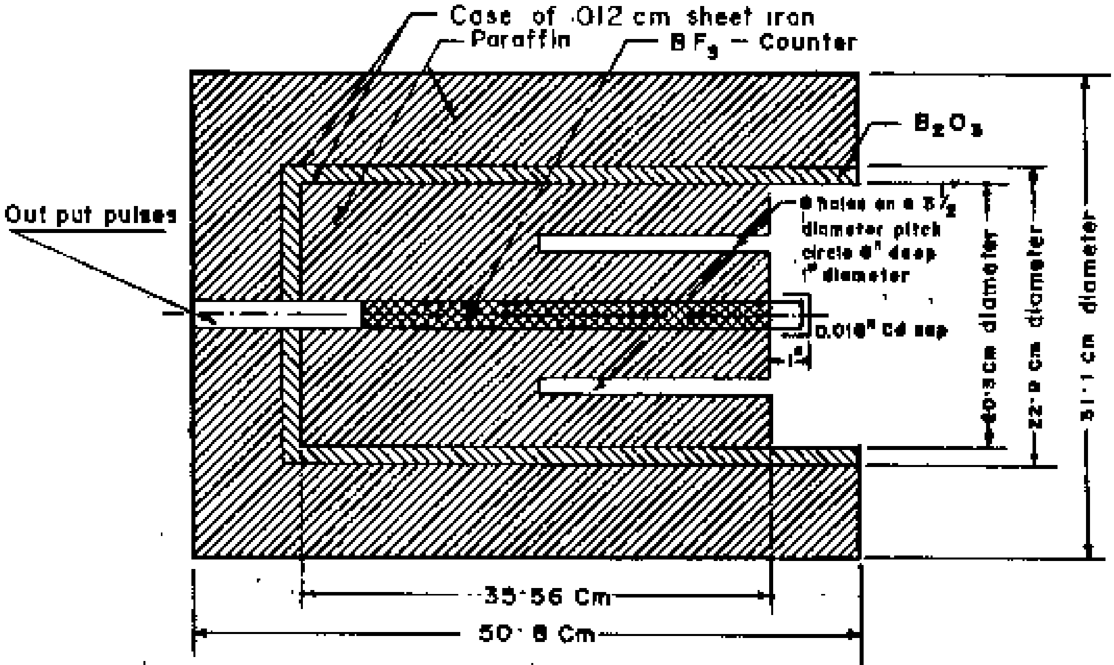
Table 6

BF_3 detector specifications

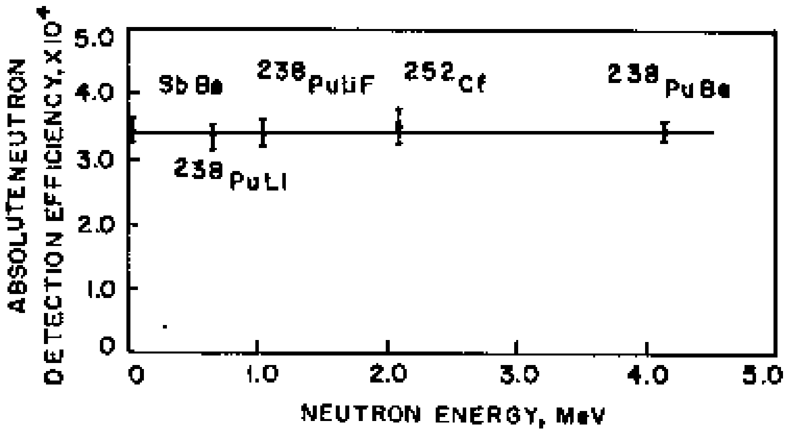
Manufacturer	:	Canberra Inc.
Model	:	4528
Diameter	:	2.54 cm
Active length	:	30 cm
Outer shell	:	0.089 cm thick
Anode	:	0.005 cm diameter tungsten
End window	:	0.203 cm thick alumina (Al_2O_3)
Fill gas	:	BF_3 (90% ^{10}B) at 120 cm Hg.

However, a device for measuring mixed energy neutron must, as a rule, have an acceptable detection efficiency over the entire energy range. The inherently low detection efficiency for fast neutróns of a slow neutron detector can be somewhat improved by surrounding the detector with a few centimeter of hydrogenous moderating materials. Moreover, a detector whose counting efficiency does not depend on the neutron energy can be a very useful device in many areas of neutron physics. For an ideal detector of this type, a graph of the detection efficiency versus neutron energy should be a horizontal line, which led to the name of 'flat response' detector. The most popular flat response neutron detector has been the so called "Long Counter". It is based on the principle of placing a slow neutron detector at the center of a moderating medium.

The combination of BF_3 detector and cylindrical paraffin moderator was first suggested as a flat response neutron detector (or long counter) by Hanson and McKibben[63]. In principle, the incident fast neutron can lose a fraction of its initial kinetic energy in the moderator before reaching the detector as a low energy neutron for which the detector efficiency is generally higher. By making the moderator thickness greater, the number of collisions in the moderator will tend to increase, leading to a lower value of the most probable energy when the neutron reaches the detector. One would therefore expect the detector efficiency to increase with moderator thickness if that was the only



(a)



(b)

Fig. 6 (a) Cross-section of the long counter and
(b) A typical efficiency curve of the long counter.

factor under consideration. A second factor, however, tends to decrease the efficiency with increasing moderator thickness; a neutron may be absorbed within the moderator before it has a chance of reaching the detector. The absorption probability will increase rapidly with increasing moderator thickness because absorption cross sections generally are larger at lower neutron energies. As a result of all these factors, the efficiency of a moderated slow neutron detector when used with fast neutron source will show a maximum at a specific moderator thickness. Thies and Bottcher[64] demonstrated how one can construct a flat response fast neutron detection system using a larger moderator with a BF_3 detector embeded in it in a suitable geometry. In this context, a design by McKTaggart[65], comprising of a BF_3 detector and suitable cylindrical paraffin moderator has achieved fairely widespread acceptance as the standard long counter (Fig. 6a). This device has proved to be highly successful as a neutron monitor in experiments using fast neutron. Moreover, it is simple, reliable and has a long term stability. Its sensitivity is reasonably uniform with neutron energy. A typical efficiency curve versus neutron energy of a long counter[7] shown in Fig. 6b.

Fig. 6a shows the basis of a long counter which was constructed following the design of Hanson-McKibben incorporating the modifications suggested by other workers[64-67]. The BF_3 detector was enclosed by a large cylindrical paraffin moderator 50.8 cm (20") in length and

38.1 cm (15") in diameter. The moderator had eight 2.54 cm (1") diameter holes, 8.89 cm (3.5") deep, in its front face, on a circumference 22.86 cm (9") in diameter. The holes provided in the front surface prevent a fall off in the efficiency at neutron energies below 1 MeV by allowing the lower energy neutrons to penetrate further into the moderator. The inner paraffin cylinder is 20.32 cm (8") in diameter. The density of the paraffin wax is 0.84 g/cm^3. The BF_3 detector used in this study was Canberra Model 4528 (Table 6), which had an active length as that of the paraffin cylinder, and was filled at 120 cm Hg pressure with enriched BF_3. A cadmium cap covered the front end of the BF_3 detector. The outside of the paraffin may be covered with a thin sheet of cadmium to absorb thermal neutrons while allowing fast neutrons to pass through into the paraffin. A common method employed for the detection of slow neutrons is based on the reaction

$$ {}^{10}_{5}B + {}^{1}_{0}n \longrightarrow \begin{cases} {}^{7}Li + {}^{4}He + 2.79\ \text{MeV}\ (6\%) \\ {}^{7}Li^{*} + {}^{4}He + 2.31\ \text{MeV}\ (94\%) \end{cases} $$

which has a thermal cross section of 4000 barn for ^{10}B isotope reaction. When thermal neutrons (0.025 eV) are used to induce the reaction, about 94% of all reaction lead to the excited state and only 6% directly to the ground state. In the (n, alpha) reaction with ^{10}B, an energy of 2.31 MeV is shared by the alpha particles and the recoiling ^{7}Li nucleus, of which the alpha particle carries away 1.47 MeV and the ^{7}Li nucleus retains 0.84 MeV as kinetic energy and the rest if any, as excitation energy.

The slow neutron measurements were made with a BF_3 detector which feeds into a standard amplifier and scaling circuit. The pulses from the detector were amplified and counted with conventional circuits (Fig.3b). The BF_3 detector exhibited a flat plateau from 1750 to about 2000 V and was operated at 1850 V. The long counter was calibrated against a standard neutron source. A 3 mCi Ra-Be was used as the standard neutron source. Fig.7 shows the pulse height distribution of the long counter as recorded by single channel analyser. The $^7Li^* + \alpha$-peak at 2.31 MeV and the $^7Li + \alpha$-peak at 2.79 MeV are clearly resolved. The 0.84 MeV ^7Li peak and 1.46 MeV alpha peak are unresolved and appear as small steps in the long tail of the $^7Li + \alpha$-peak. The cumulative effect of this type of process is known as the "wall effect" in gas counters. The wall effect continuum extends from E_{Li} (0.84 MeV) upto the full energy peak at ($E_{Li} + E_{alpha}$ = 2.31 MeV). Only those reactions leading to the ^7Li excited state were considered because the wall effect continuum associated with the much less probable ground state is normally so small as to be submerged by the remainder of the spectrum. In addition, the wall effect continuum is also expected from gamma rays which are often found along with the neutron flux to be measured. Gamma rays interact primarily in the wall of the counter and create secondary electrons which may produce ionization in the gas. Because the stopping power for electrons in gases is quite low, a typical electron will deposit only a small fraction of its initial energy within the gas before reaching the

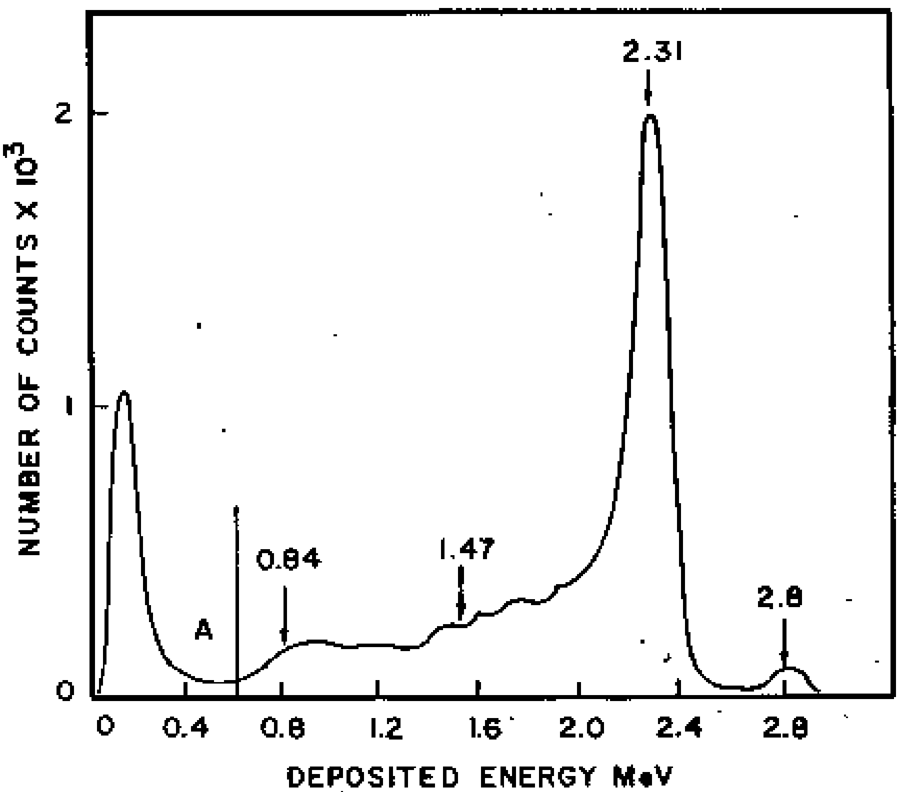


Fig. 7. Energy spectrum from the long counter due to Ra-Be source.

opposite wall of the counter. Thus we expect that most gamma ray interactions will result in low-amplitude pulses which will lie in the tail to the left of the point A in Fig.7. The pulse height spectrum from long counter tells us nothing about the energy spectrum of the incident radiation but is a function only of the size and geometry of the detector itself. Instead, we are interested in a stable operating point or a counting plateau for which small drifts in operating parameters do not significantly affect the neutron sensitivity of the counter. That objective would be met by setting a fixed discriminator level at the point labelled A in Fig.7. Under these conditions all the neutrons will be counted, whereas low-amplitude events will be rejected thus eliminating the gamma rays without sacrificing neutron detection efficiency.

3.4 Gamma ray detector

A gamma ray photon is uncharged and creates no direct ionization or excitation of the material through which it passes. The detection of gamma rays is therefore dependent on causing the gamma ray photon to undergo an interaction that transfers all or part of the photon energy to an electron in the absorbing material. Because the primary gamma ray photons are "invisible" to the detector, it is only the fast electrons created in gamma ray interactions that provide any clue to the nature of the incident gamma rays. These electrons have a maximum energy equal to the energy of the incident gamma ray photon and will slow down and loose their energy while ionizing the absorbed material. A detector to serve as gamma ray spectrometer, it must carry out two distinct functions. First, it must act as a conversion medium in which incident gamma rays have a reasonable probability of interaction and thus to yield one or more fast electrons, and second, it must function as a conventional detector for these secondary electrons. The scintillators are commonly used in gamma ray detection. Scintillators are group of solids, liquids and gases which emit light (or photon) from their atoms or molecules, excited (directly or indirectly) by the passage of radiations through them. This process of absorption of energy by a substance and its subsequent re-emission as visible or near visible (3000 to 7000 Å) is known as luminescence which is of two kinds - phosphorescence and fluorescence. The process where the light emission occurs during the excitation or whthin 10^{-8} sec is known as fluorescence while that occuring after the excitation has ceased (emission occurs within 10^{-6} sec to hrs.) is known as phosphorescence. Scintillators with shorter duration (10^{-12} sec) are useful for detection of nuclear radiations. The desirable qualities for scintillators are as follows:

(i) High fluorescent light output;

(ii) Transparency to its emitted light;

(iii) Rapid light emission; and

(iv) Less phosphorescence.

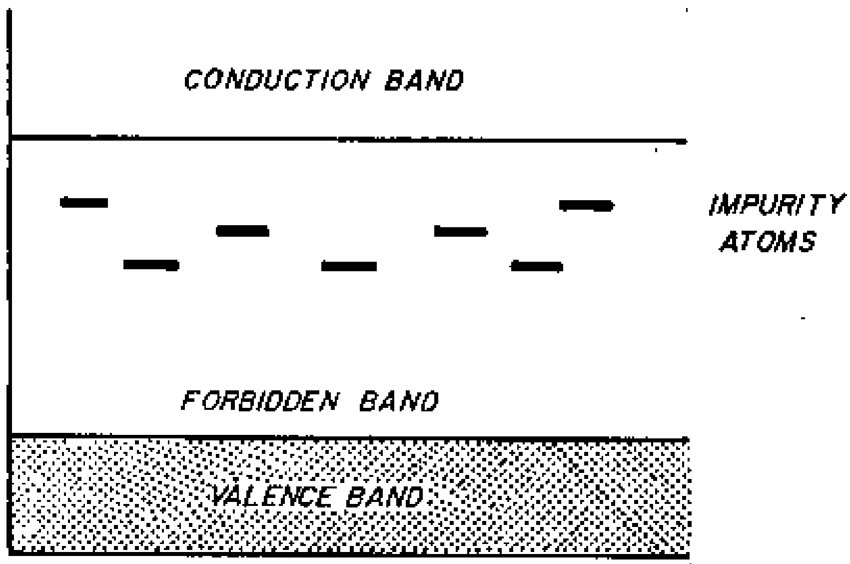


Fig. 8. Energy level diagram of inorganic scintillator crystal.

The properties of different scintillators have been discussed in detail elsewhere[1-4]. When ionizing radiation, for example a gamma ray passes through a scintillator part or all of its energy is absorbed in the scintillator. In organic scintillators, this absorbed energy can raise the molecule form their electronic ground states to their electronic excited states. A molecule in the electronic excited state may loose its energy to lattice vibrations or may return to its electronic ground state with the emission of visible light. The luminescence in an inorganic scintillator is an electronic process which may be represented in the band theory by a valence band, a forbiden band and a conduction band of energies. Normally, the valence band is completely filled with electrons and the conduction band which lies above the valence band is empty (Fig.8). Between these two bands, there lies the forbidden band in which electrons cannot normally exist. The presence of impurity atoms can create energy levels in the forbidden band at isolated points throughout the crystal.

The passage of an ionizing radiation through an scintillator crystal can excite the electrons and raise them from the valence band to the conduction band. An electron in the conduction band can wander through the crystal until it comes to the vicinity of an impurity atom or a lattice vacancy where it can drop to the energy level associated with such an imperfection. It can return to the valence band, which is its ground state, either directly from the conduction band or from its new energy level in the forbidden band. While returning to its ground state it gives up its excess energy in the form of fluorescent radiation.

The first two considerations, while setting up a gamma ray spectrometer are the choice of a suitable phosphor (or scintillator) and the size of the phosphor. Although a large number of phosphors namely NaI (TI), CsI(TI), Li(Eu), CsFl etc. are available for the purpose of gamma ray detection, NaI(Tl) crystal, ever since its introduction as a gamma ray detector by Hofstadter[68], has remained the most important detector for gamma ray scintillation spectrometry. The thallium activator which is present as an impurity in the NaI crystal structure to the extent of about 0.2% enhances the conversion of the energy absorbed in the crystal to light. Its high density (3.67 g/cm^3) together with its high effective atomic number (Z=53) makes it a suitable material for the efficient absorption of electromagnetic radiation. The presence of iodine gives a high cross-section for photoelectric and pair production interactions. The emission spectrum with a maximum of 4100 Å is suitable for good matching to many types of commercial photomultiplier. The speed of response makes it suitable for most applications, its decay time is of the order of 0.25 microsecond. These properties combined with the high transparency of the crystal to its own fluorescence mean that there is inherently good resolution making it a convenient detector for counting of gamma ray photons. The Model 802 scintillation detector, which is monoline crystal assembly including a high resolution NaI(Tl) crystal, a photo-multiplier (PM) tube, an internal magnetic/light shield, and a chrome plated aluminium housing was selected in our experiment to detectly gamma ray photons. Model 802 assembly plugs directly into the Model 2007P, which combines the bias network with a built-in preamplifier. Its salient features are shown in Table 7.

Table 7

Specifications of NaI(Tl) detector

Dimension	:	7.6 cm x 7.6 cm.
Window	:	0.05 mm aluminum (density:147.9 mg/cm^2)
Reflector-oxide	:	16 mm thick (density 88 mg/cm^2)
Magnetic/light shield	:	Conecting lined steel
Bias	:	1000 V
Resolution	:	7.5% at 662 keV peak of ^{137}Cs.

Gamma rays interact with the crystal by three main processes. These are the photoelectric effect, the Compton effect, and the pair production process. These processes have already been discussed in chapter 1(section 1.2.2). The amount of energy absorbed depends on the type of interaction which a photon undergoes with an electron in the scintillator. In a photoelectric interaction the entire photon energy is absorbed in the scintillator, whereas in

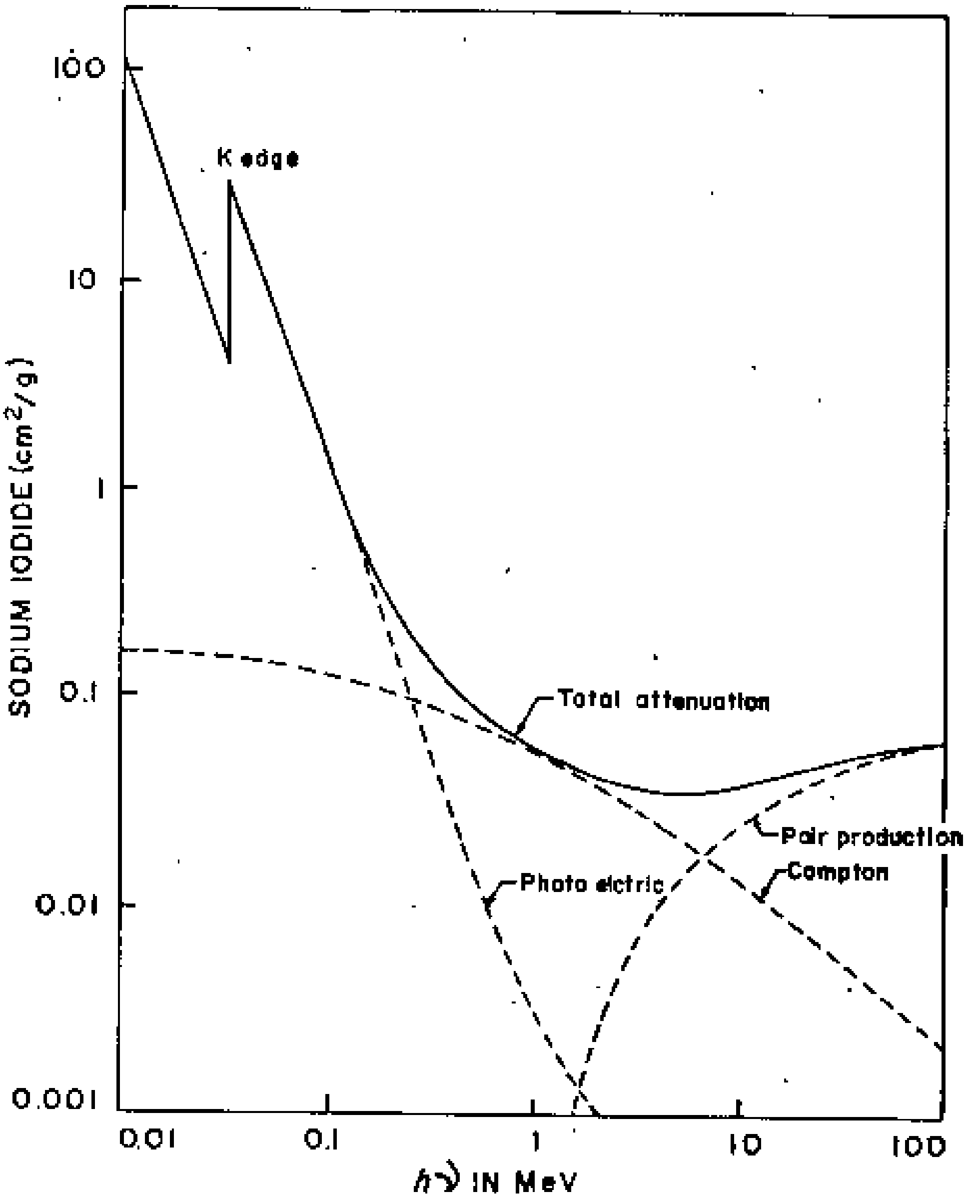


Fig. 9. Energy dependence of the various gamma ray interaction processes in sodium iodide crystal.

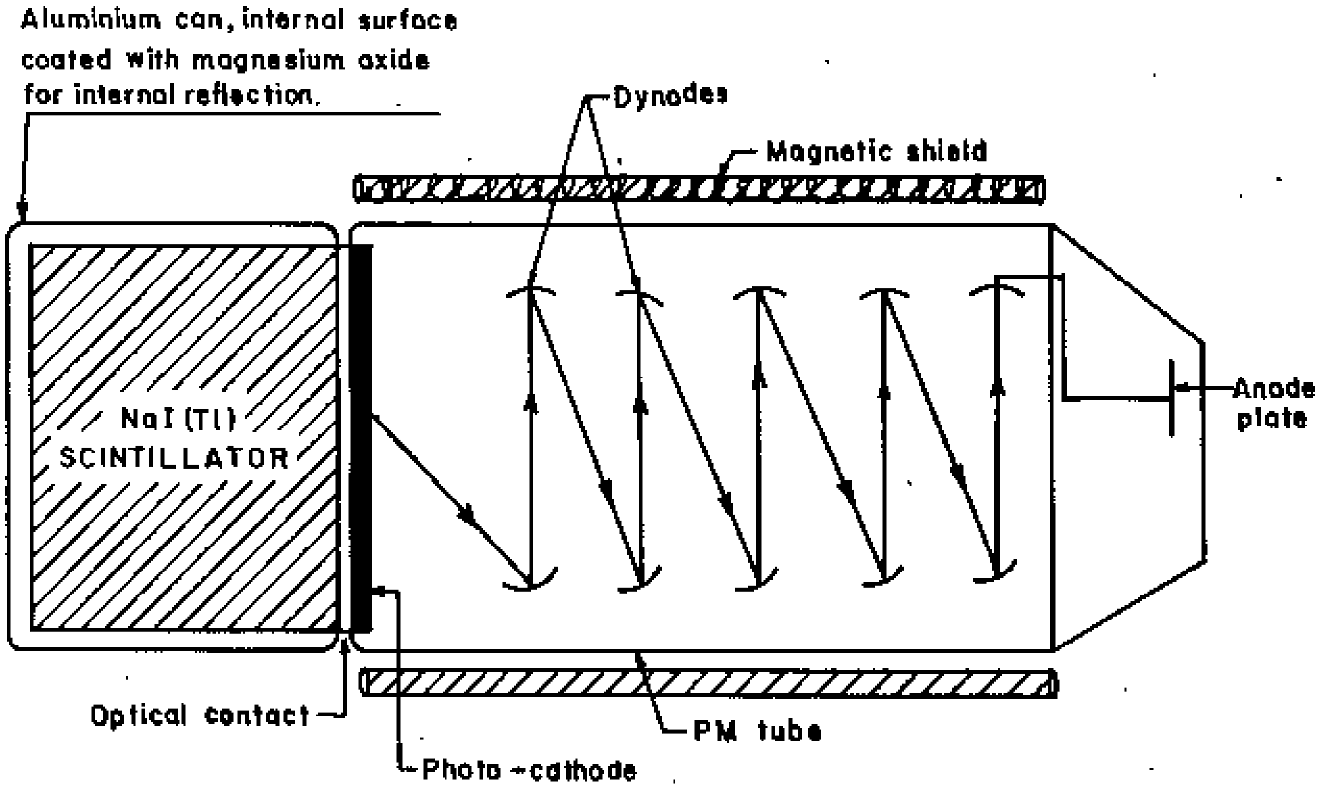


Fig. 10. A schematic diagram of NaI-PM tube system.

the Compton or pair production processes only part of the photon energy is absorbed. Fig.9 shows a typical curve which explains the interaction processes of gamma rays with NaI(Tl) crystal. The primary ionizing particles resulting from the gamma ray interactions namely the photoelectrons, Compton electrons, and positron - electron pairs - dissipate their kinetic energy by exciting and ionizing the atoms in the crystal. The excited atoms return to the ground state by the emission of light quanta. The intensity of the light flash produced in the scintillator is proportional to the amount of energy absorbed. The light quanta emitted by the scintillator are converted back into electrons by a photo-cathode. These electrons are subsequently multiplied into electric signals by photo-multipliers. A typical arrangement of the scintillator PM tube assembly is shown in figure 10. The scintillation light after reflection at the inside walls (which are light-tight) enclosing the scintillators falls on the photo-cathode of the PM tube which is optically coupled to the scintillator. The photo electrons are emitted from the photo cathode and are directed toward the first dynode of the photomultiplier (Fig.10), whose potential is about 100 V positive with respect to the photo cathode. Each electron that strikes the dynode causes several other electrons to be ejected from the dynode, thereby 'multiplying' the orignal photocurrent. This process is repeated about 10 times before all the electrons, thus produced, are collected by the plate of the PM tube. The resulting current pulses, whose magnitude is proportional to the energy of the primary ionizing radiation, produces voltage pulses at the input of the preamplifier. The voltage pulse can then be amplified and analysed for counting through conventional circuits (Fig.3b).

CHAPTER 4

RESULTS AND DISCUSSION

The results of the attenuation measurements of neutron and gamma ray for ordinary concrete, ilmenite concrete and magnetite concrete with and without boron additives are presented in the Figures 11-18. Figure 11 shows the variation of neutron attenuation factor, I/I(O) as a function of thickness for ordinary concrete with and without boron additives. It is seen that the addition of boron contribute to significant attenuation in neutron transmission. For an initial thickness of 5 cm, the attenuation observed is 33% for ordinary concrete (Type 01) without boron additives compared to 40% for boron additives ordinary concrete (Type 01B). For 30 cm, the attenuation increase to 87% and 88% respectively. The attenuation curves for gamma ray in the same types of concretes are shown in Fig. 12. The curve shows that the decrease in gamma ray intensity is less in the ordinary concrete without boron additives and this is due to the low absorption of newly produced gamma rays arising from thermal neutron capture in the concrete. For an initial thickness of 5 cm, the attenuation of gamma ray in ordinary concrete (without boron) is 29% compared to 34% when boron is added to the same concrete. The addition of boron to this concrete results in a greater attenuation of the emergent gamma ray. This is a consequence of the strong absorption of thermal neutrons by boron which results in the production of soft gamma rays, easily absorbed in the concrete would otherwise lead to capture gammas from concrete.

In Figures 13 and 14 the attenuation factor for neutron are given as a function of the thickness for ilmenite and magnetite concretes with and without boron additives. In case of ilmenite concrete (Type I1), the attenuation of neutron intensity observed is 38% compared to 33% for ordinary concrete (Type 01) at the thickness of 5 cm. For thickness of 30 cm, the attenuation of neutron intensity is 88% for type I1 concrete compared to 87% for type 01 concrete. The attenuation of neutron intensity observed is 37% for magentite concrete (Type M1) compared to 33% for ordinary concrete for the thickness of 5 cm and their attenuation effects increase to 89% and 87% respectively for the thickness of 30 cm. For an initial thickness of 5 cm, the attenuations of neutron intensity observed are 39% for boron loaded ilmenite concrete (Type I1B) and 38% for boron loaded magnetite concrete (Type M1B) compared to 40% for boron loaded ordinary concrete (Type 01B). The effect of boron additives to ilmenite and magnetic concretes has a similar trend for boron additivies to ordinary concrete. However, the attenuation effect of boron additives to ilmenite and magnetite concretes is better than that of ilmenite and magnetite concretes without boron additives. Addition of boron to ilmenite and magnetite concrete do not contriute to a significant attenuation in neutron transmission compared to boron loaded ordinary concrete.

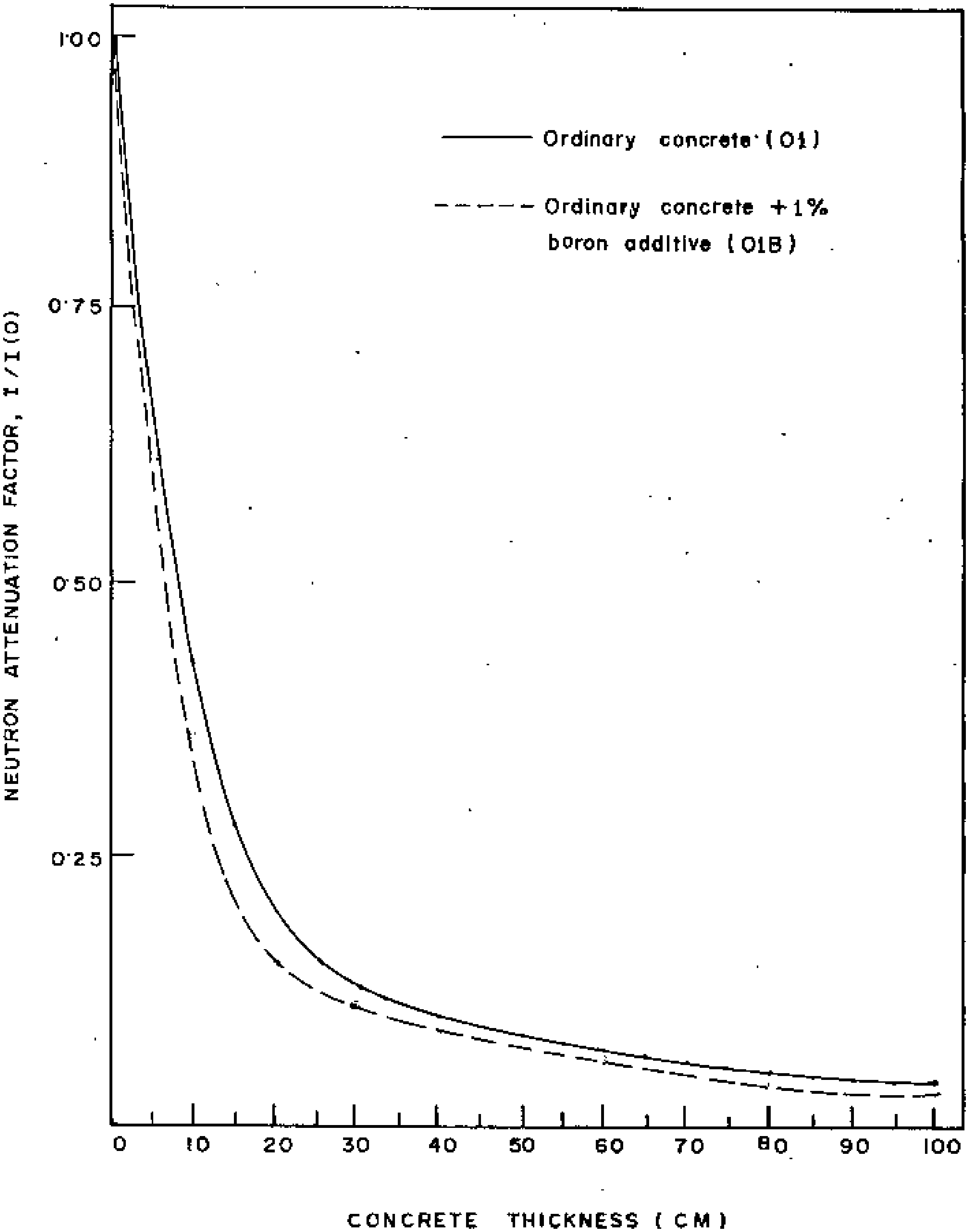


Fig. 11. Neutron attenuation curves for ordinary concretes O1 and O1B.

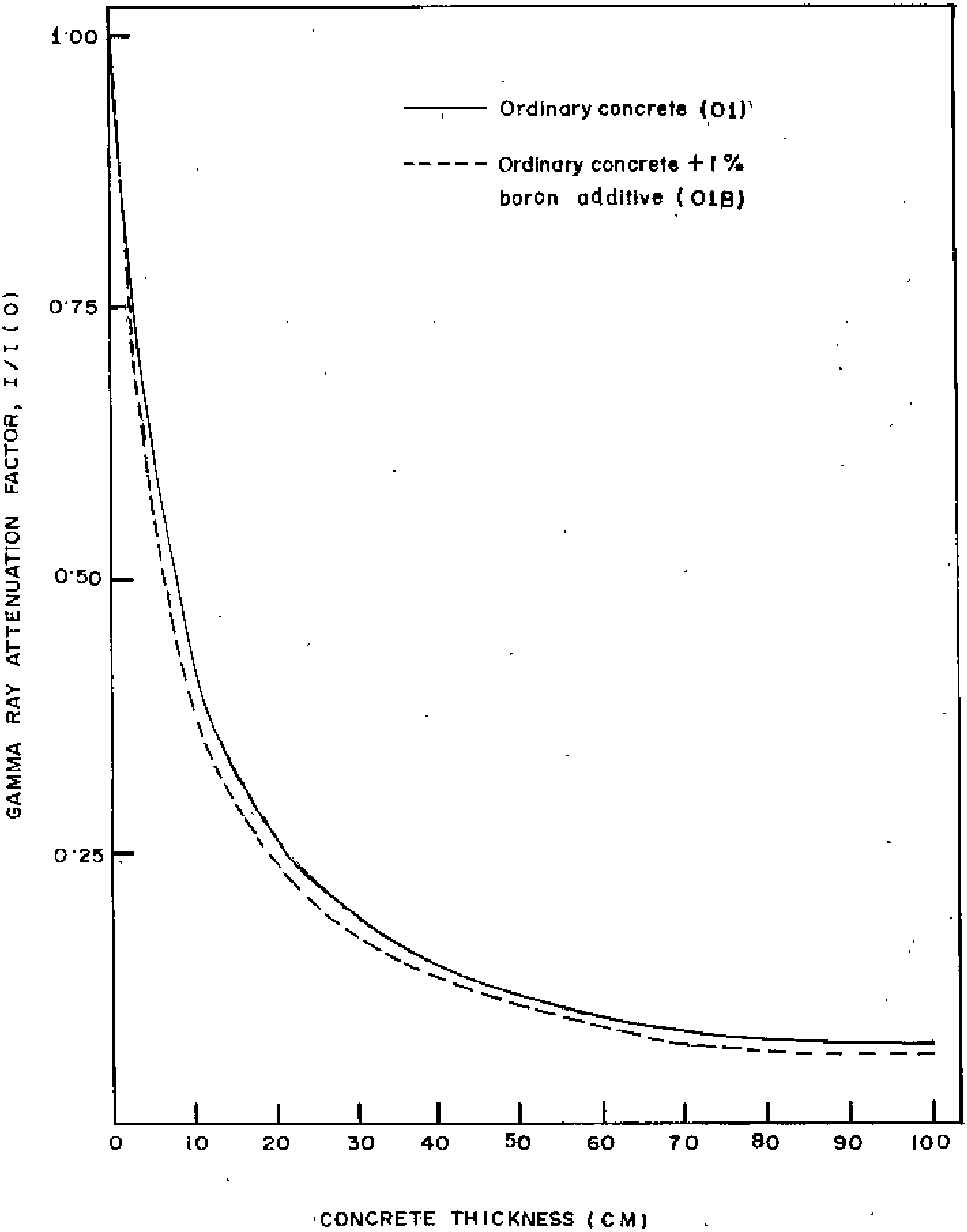


Fig. 12. Gamma ray attenuation curves for concretes O1 and O1B.

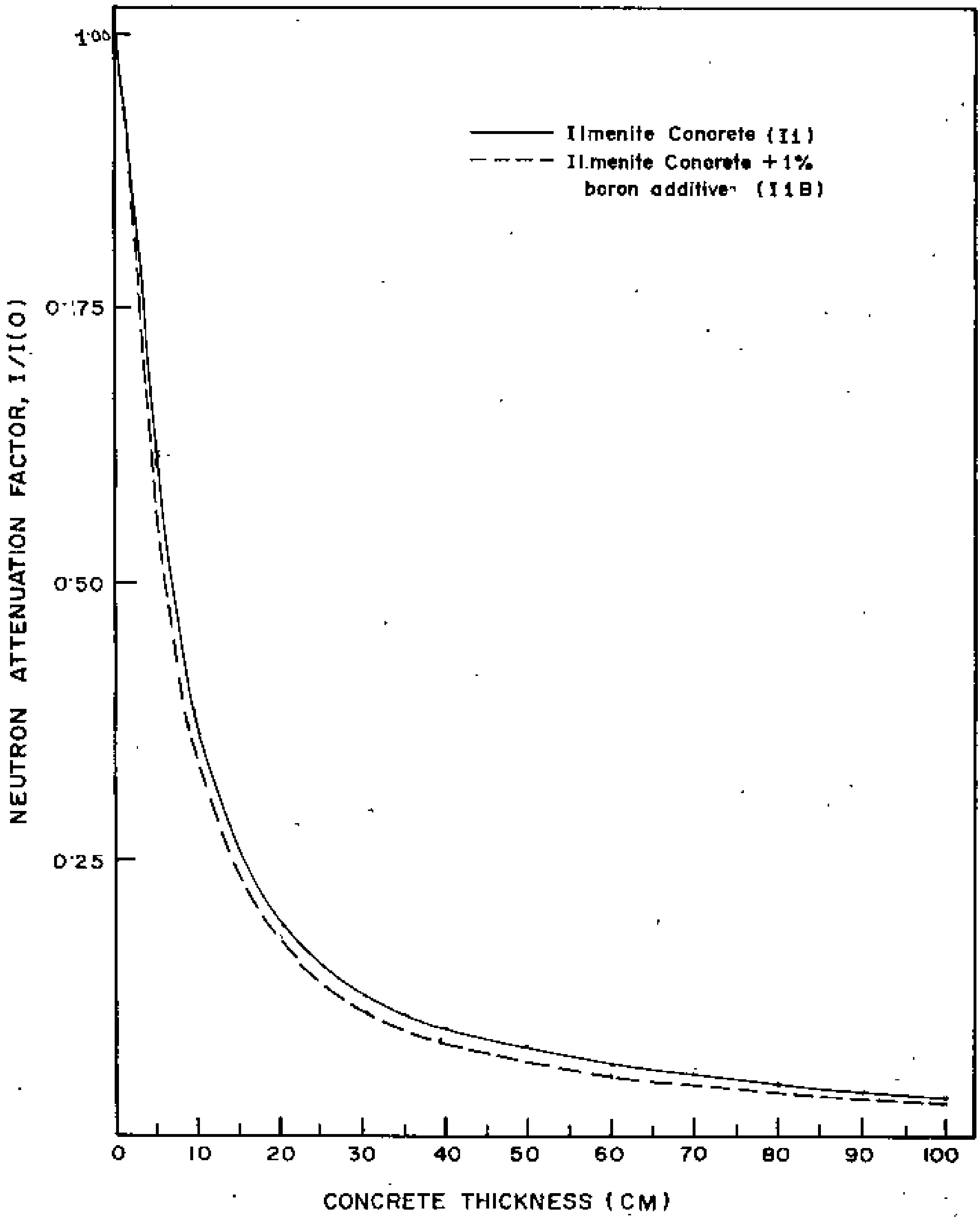


Fig. 13. Neutron attenuation curves for concretes Ii and IiB.

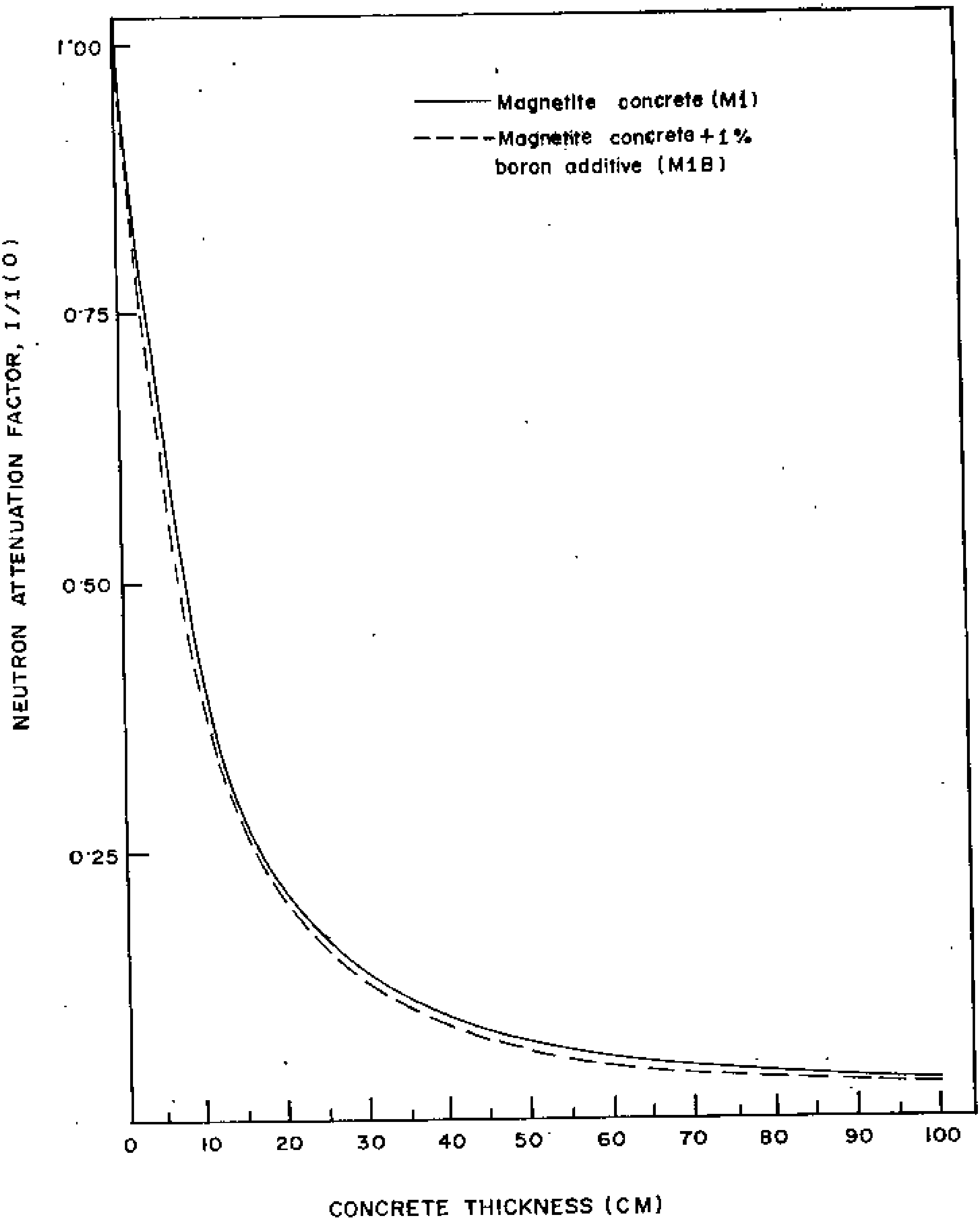


Fig. 14. Neutron attenuation curves for concretes M1 and M1B.

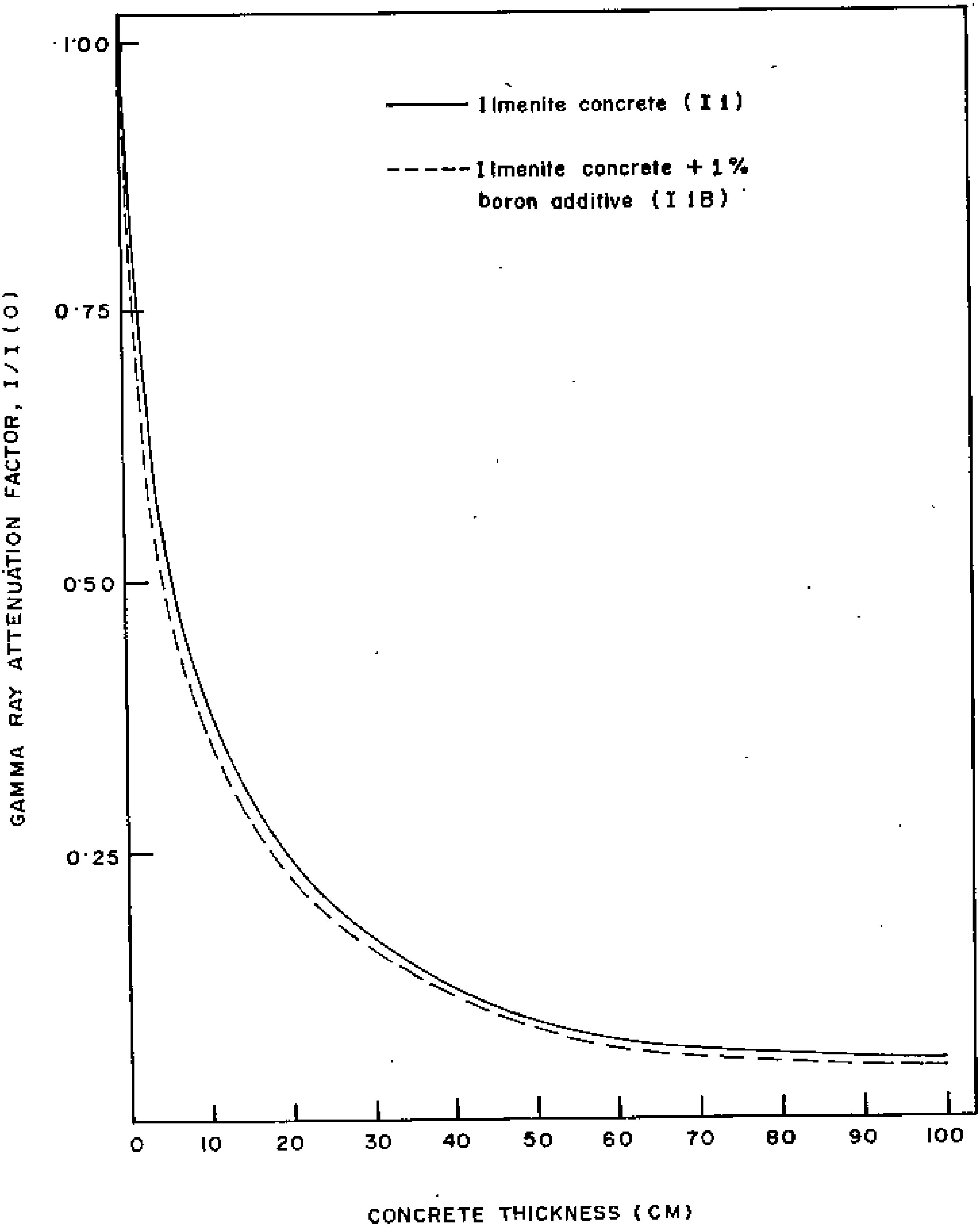


Fig. 15. Gamma ray attenuation curves for concretes I 1 and I 1B.

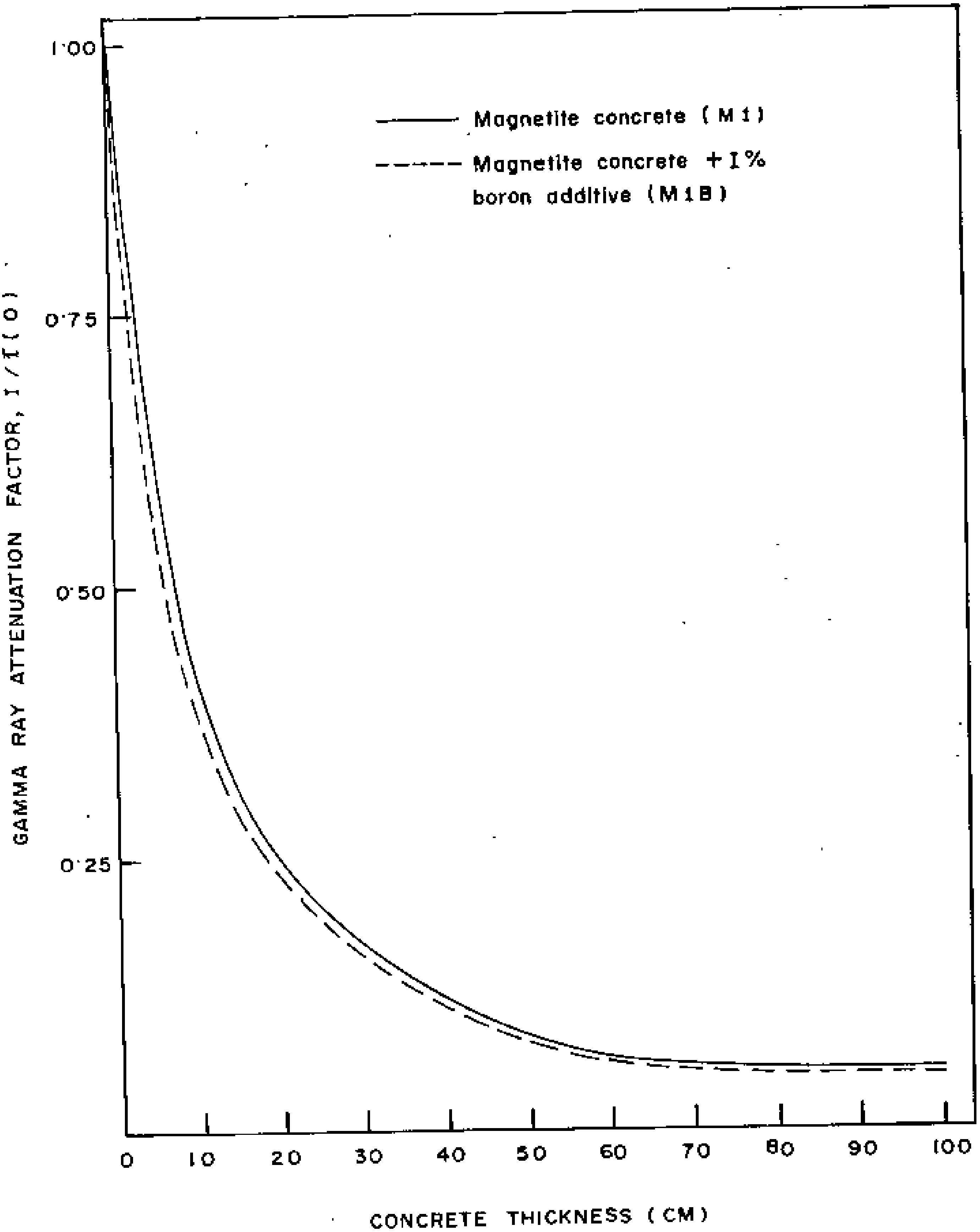


Fig. 16. Gamma ray attenuation curves for concretes M1 and M1B.

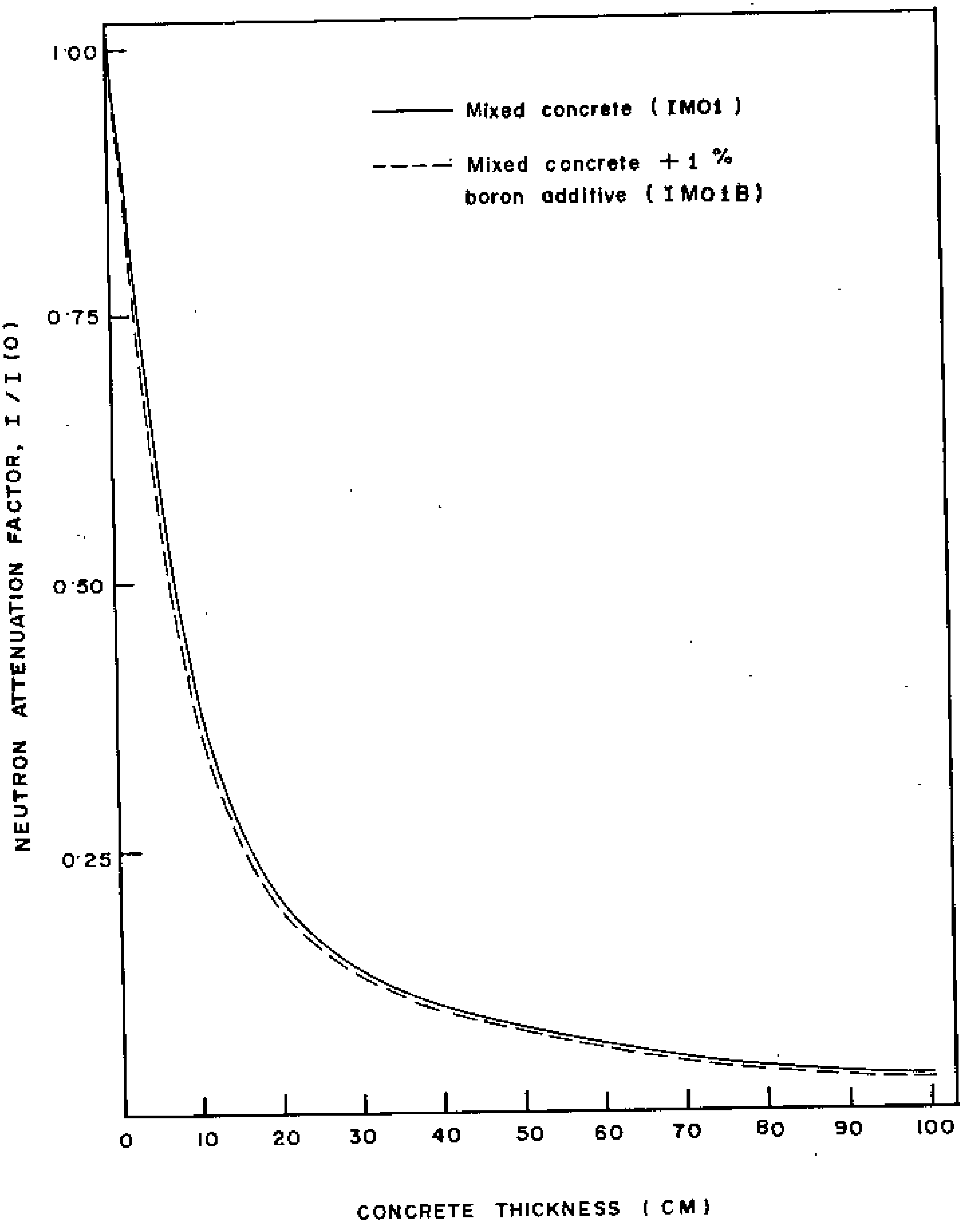


Fig. 17. Neutron attenuation curves for concretes IMO1 and IMO1B.

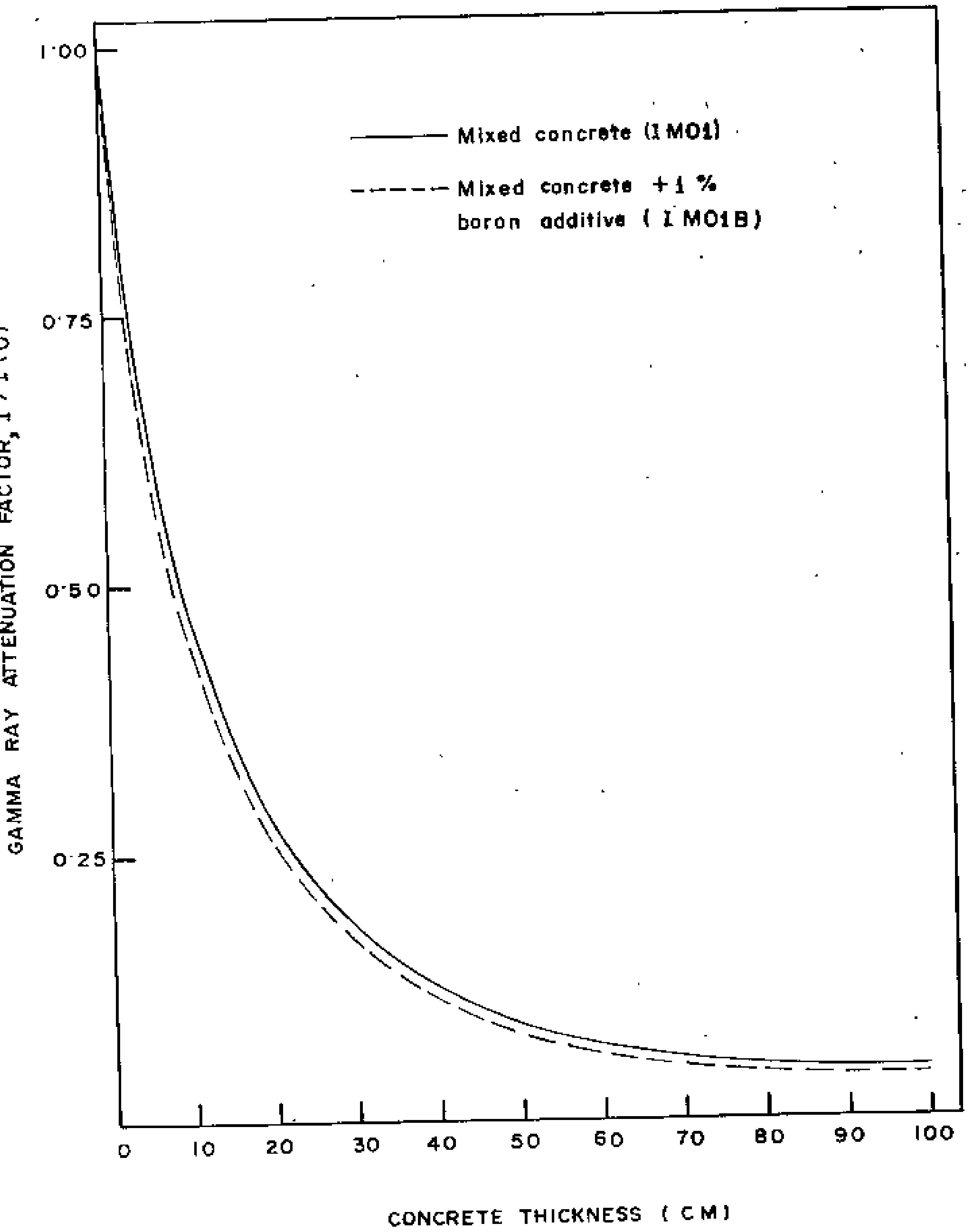


Fig. 18. Gamma ray attenuation curves for concretes IMO1 and IMO1B.

This is due to strong neutron inelastic scattering from iron which is present in the form of iron oxide in ilmenite ($FeO.TiO_2$) and magnetite ($FeO.Fe_2O_3$) concrete. This is in good agreement with the results obtained by several authors[10] who also reported that the addition of boron compound to high density concretes such as ilmenite, magnetite, barite, limonite etc. is not much effective for neutron attenuation.

Figures 15 and 16 show gamma ray attenuation factor as a function of the thickness for ilmenite and magnetite concretes with and without boron additives. The attenuation of gamma ray intensity is 38% for ilmenite concrete(Type I1) and 37% for magnetite concrete (Type M1) compared to 29% for ordinary concrete (Type 01) at the thickness of 5 cm. The attenuation in gamma ray transmission in I1 and M1 types of concretes is quite significant compared to type 01. This is due to the presence of iron oxide which strongly attenuates gamma ray transmission. The attenuation of gamma ray intensity is 39% for ilmenite concrete and 38% for magnetite concrete with boron additives compared to 34% for ordinary concrete with boron additives for the thickness of 5 cm. The attenuation of gamma ray intensity in boron loaded ilmenite and magnetite concretes is almost same with that produced by ilmenite and magnetite concretes without boron additives, but has a slightly better effect when compared to the attenuation produced in a boron loaded ordinary concrete. However, in all cases, the attenuation of gamma ray intensity in boron loaded concretes is slightly better than that in concretes without boron additives although concretes without boron additives have slightly higher densities (Table 4). The gamma ray intensity comprises of the primary gamma ray from the source (^{252}Cf) and the secondary gamma ray created by the interaction of neutrons with nuclei in the concrete. In case of boron loaded concretes, the high absorption of thermal and slow neutrons by ^{10}B through (n, alpha) reaction, results in the generation of soft secondary gamma radiation.

The mixed concrete that has been used for the biological shield in the 3 MW TRIGA Mark-II research reactor of the Bangladesh Atomic Energy Commission was also studied in this work. The composition and constituents of this concrete are givne in Table 1 (Type IM01). While some engineering properties have been studied for this concrete, the shielding effectiveness for this concrete has not been studied. These properties were evaluated in the present study along with other concretes. The mixed concrete with boron additives were also fabricated (Type IM01B) for comparison. The effectiveness of this concrete is compared with the other three types of boron loaded concretes, all having 1% boron. In Figures 17 and 18 the attenuation factor of neutron and gamma ray is shown as a function of the thickness in cm for these concretes (Type IM01 and IM01B). For an initial thickness of 5 cm, the attenuation of neutron intensity observed is 38% for concrete type IM01 compared to

41% for concrete type IMO1B. For larger thickness of 30 cm, their attenuation effects rise to 87% and 88% respectively. The attenuation of gamma ray intensity is 39% for concrete type IMO1 compared to 40% for concrete type IMO1B. The attenuation of gamma ray intensity of these concretes is similar to that of ilmenite concrete and magnetite concrete with and wihout boron additives but is better than that of ordinary concretes with and without boron additives.

In Figures 19 to 26 the counts for neutron and gamma ray are given as a function of the thickness in cm for different types of concretes. The attenaution of neutron and gamma ray intensity is generally characterised by half value thickness (HVT), removal cross section (ΣR), attenuation coefficient (μ) and relaxation length(λ). The attenuation characteristics based on the measured values from the attenuation curves for neutron and gamma ray are given in Tables 8 and 9. The values presented in these tables are deduced from the slops of the curves (Figures 19-26).

Since the relaxation length of a particular radiation in a given material is equivalent to the reciprocal of the attenuation coefficient (for gamma rays) or the removal cross section (for neutrons), the relaxation lengths provide a good means for comparing the shielding effectiveness of different materials. In all cases, the relaxation length for neutron in boron additives concrete is much smaller than in concrete wihout boron additives. This is explained by the presence of boron in concrete which results in a increase of

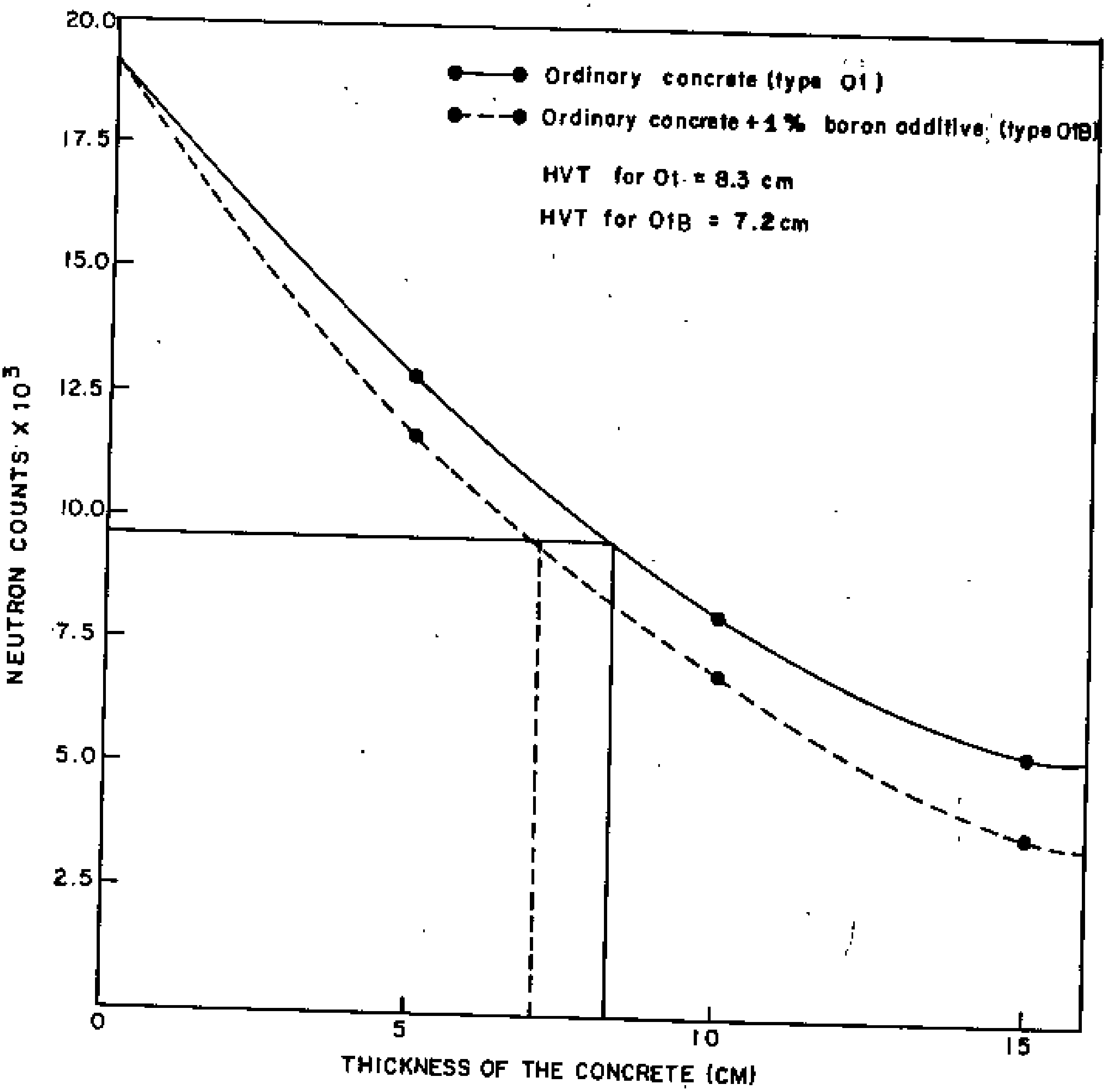


Fig. 19. Neutron counts vs. thickness of the concretes O1 and O1B.

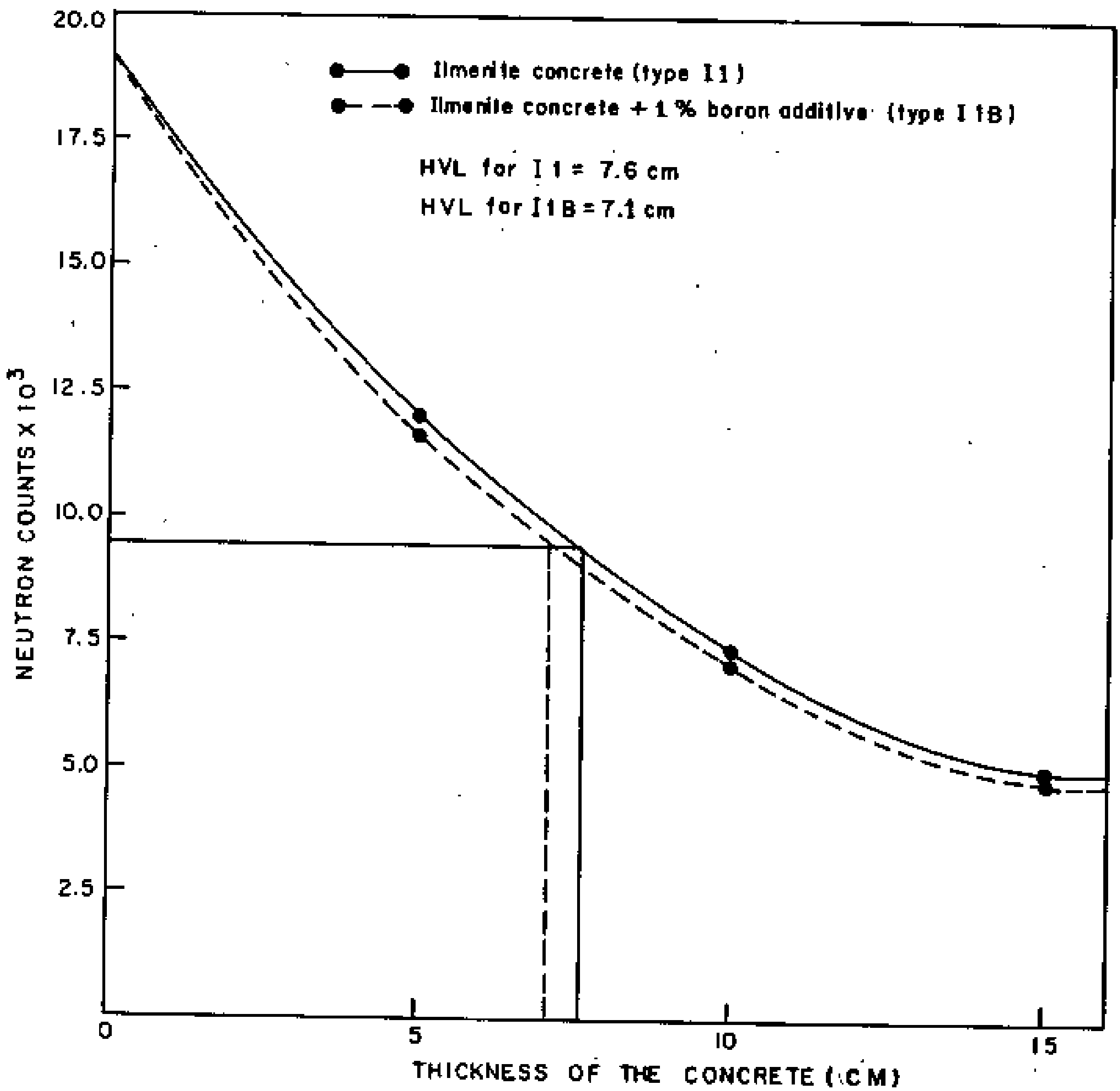


Fig. 20. Neutron counts vs. thickness of the concretes I1 and I1B.

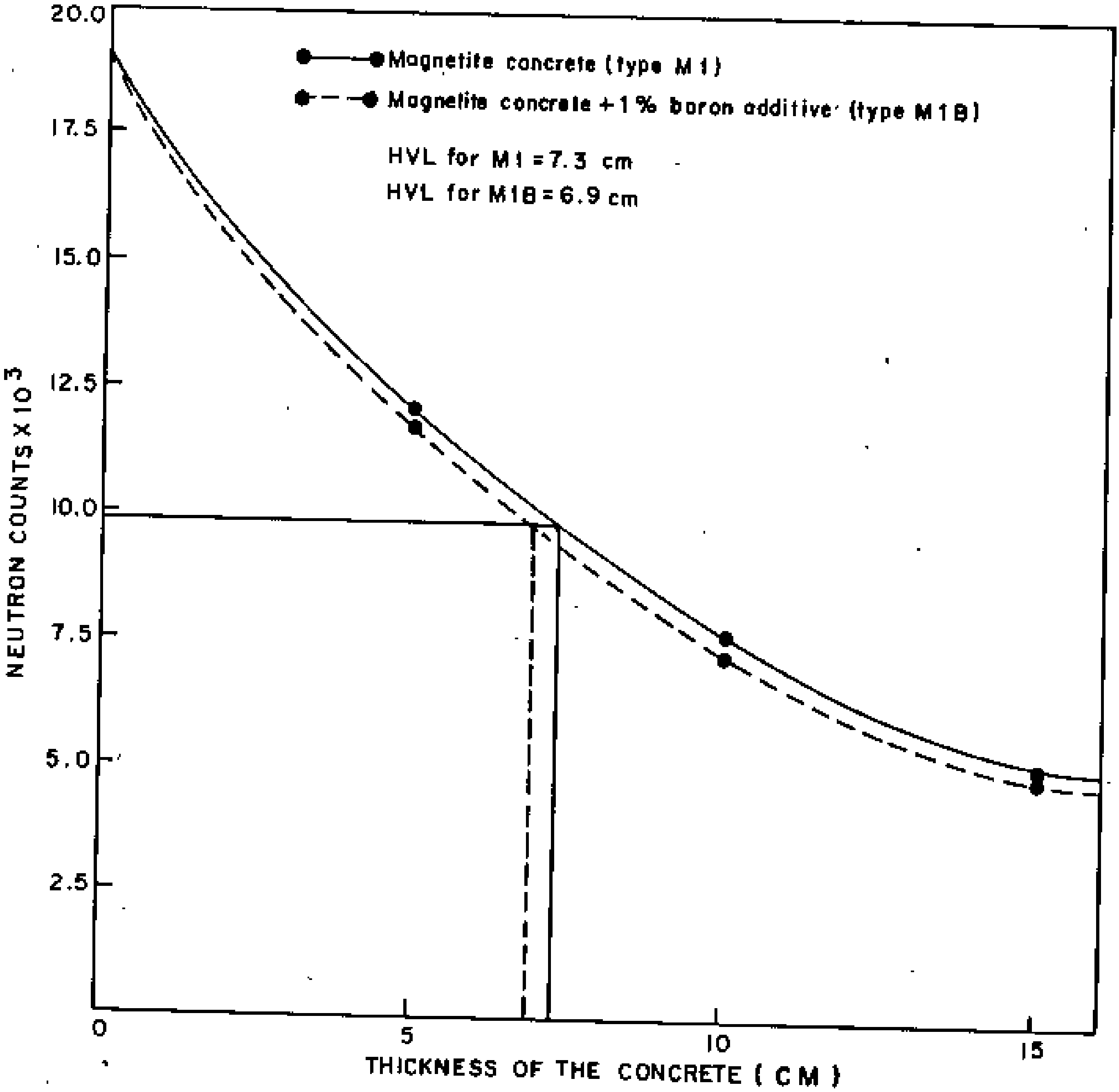


Fig. 21. Neutron counts vs. thickness of the concretes M1 and M1B.

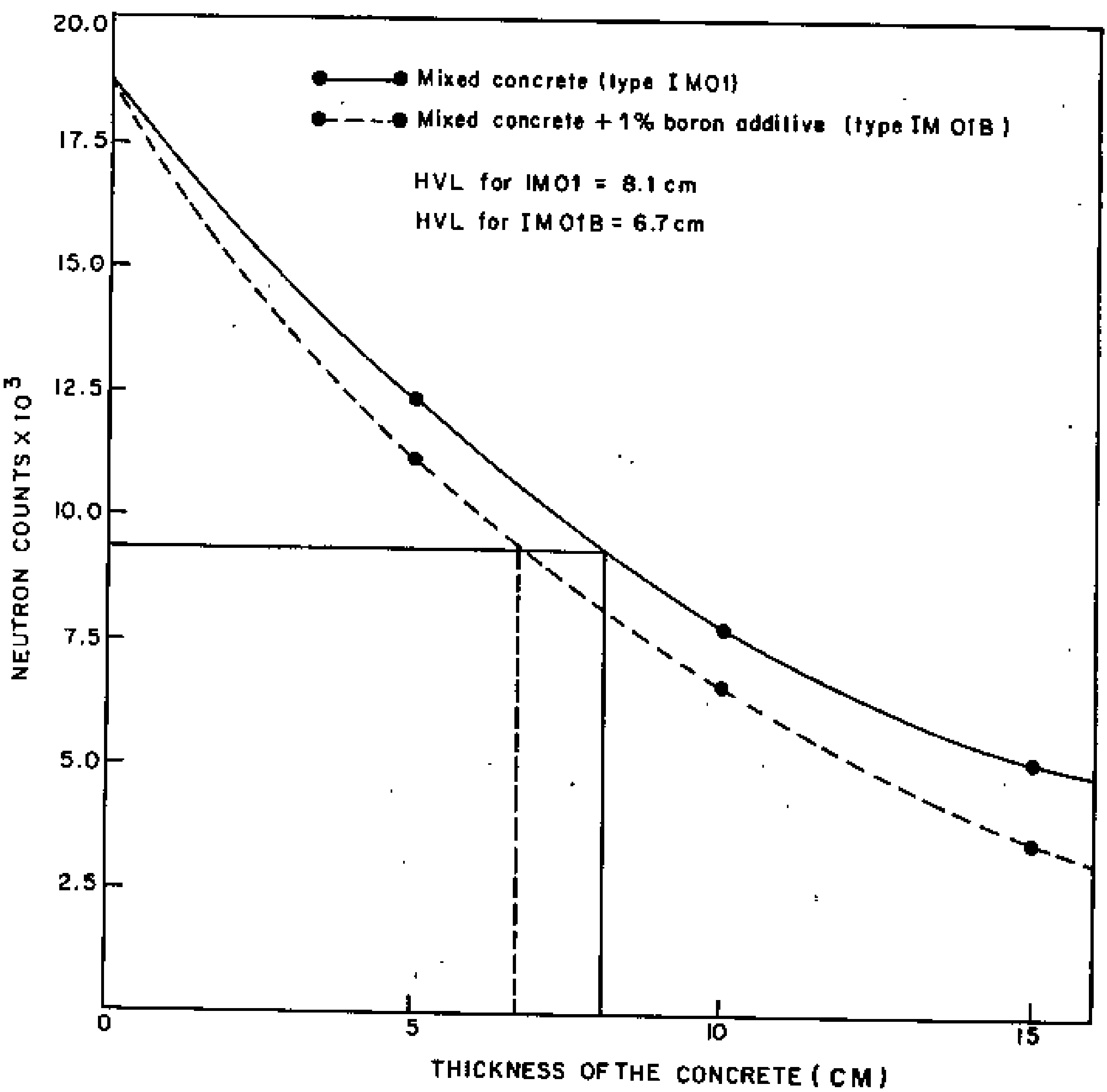


Fig. 22. Neutron counts vs. thickness of the concretes IMO1 and IMO1B.

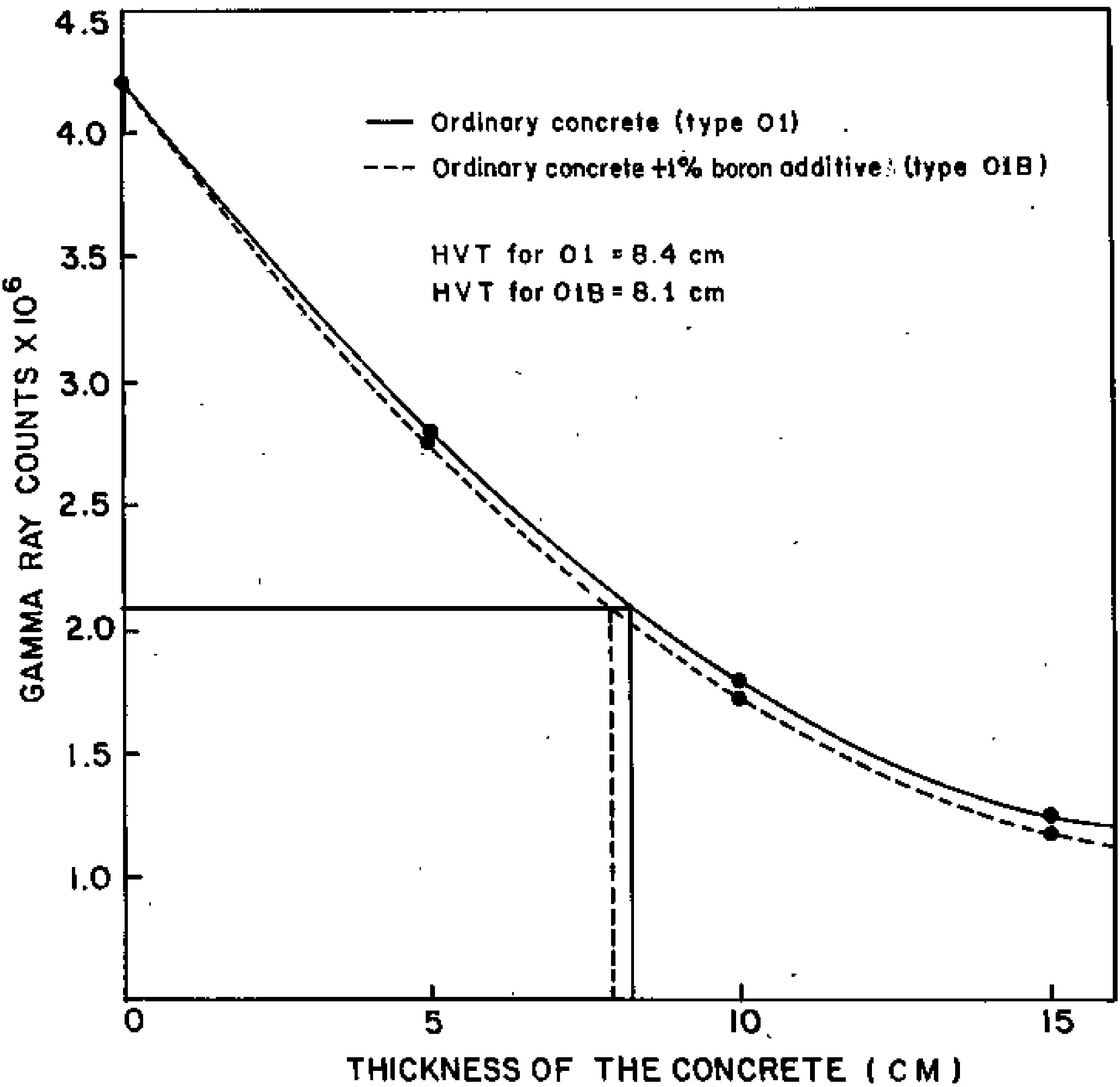


Fig. 23. Gamma ray counts vs. thickness of the concretes O1 and O1B.

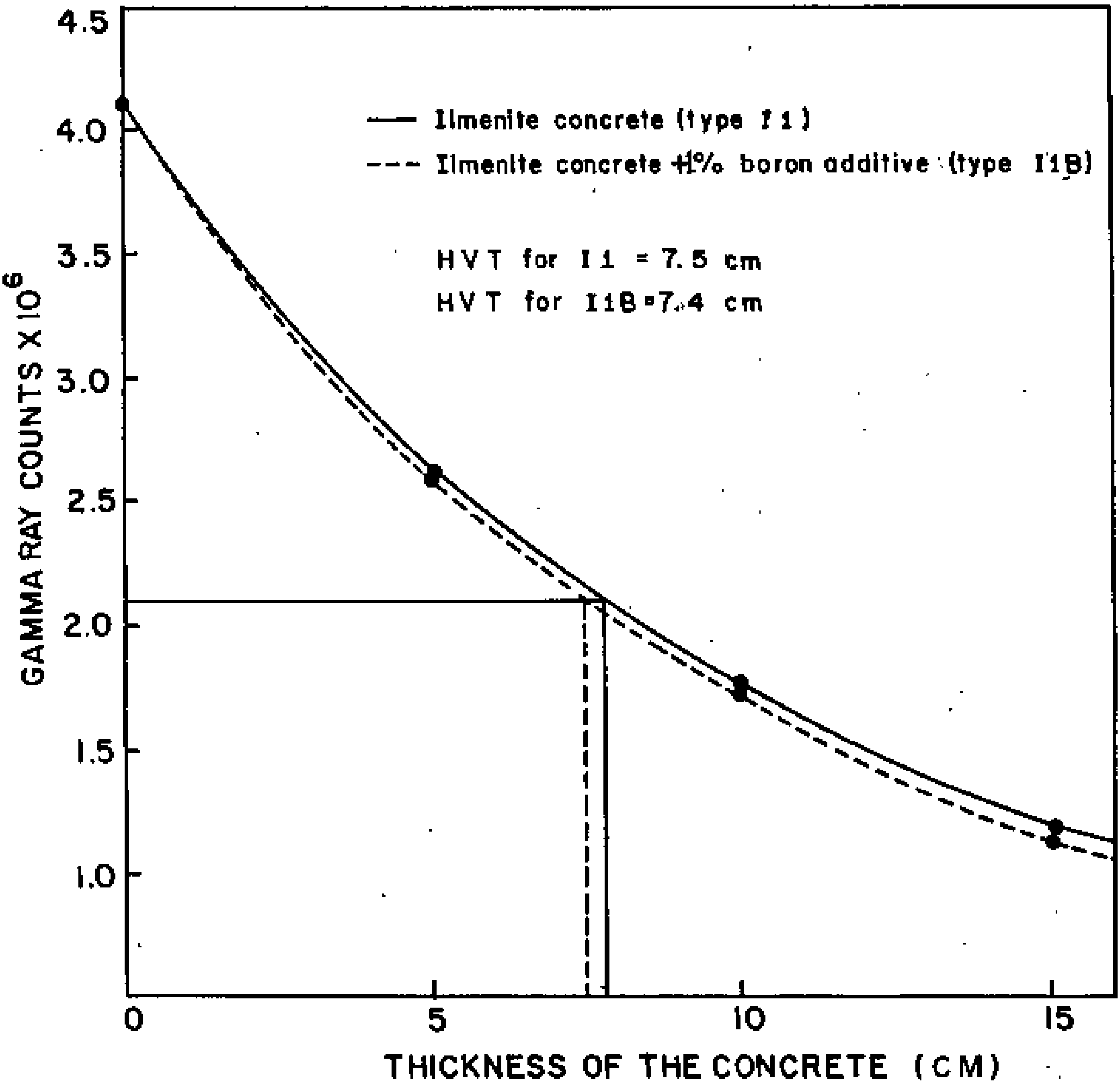


Fig. 24. Gamma ray counts vs. thickness of the concretes I1 and I1B.

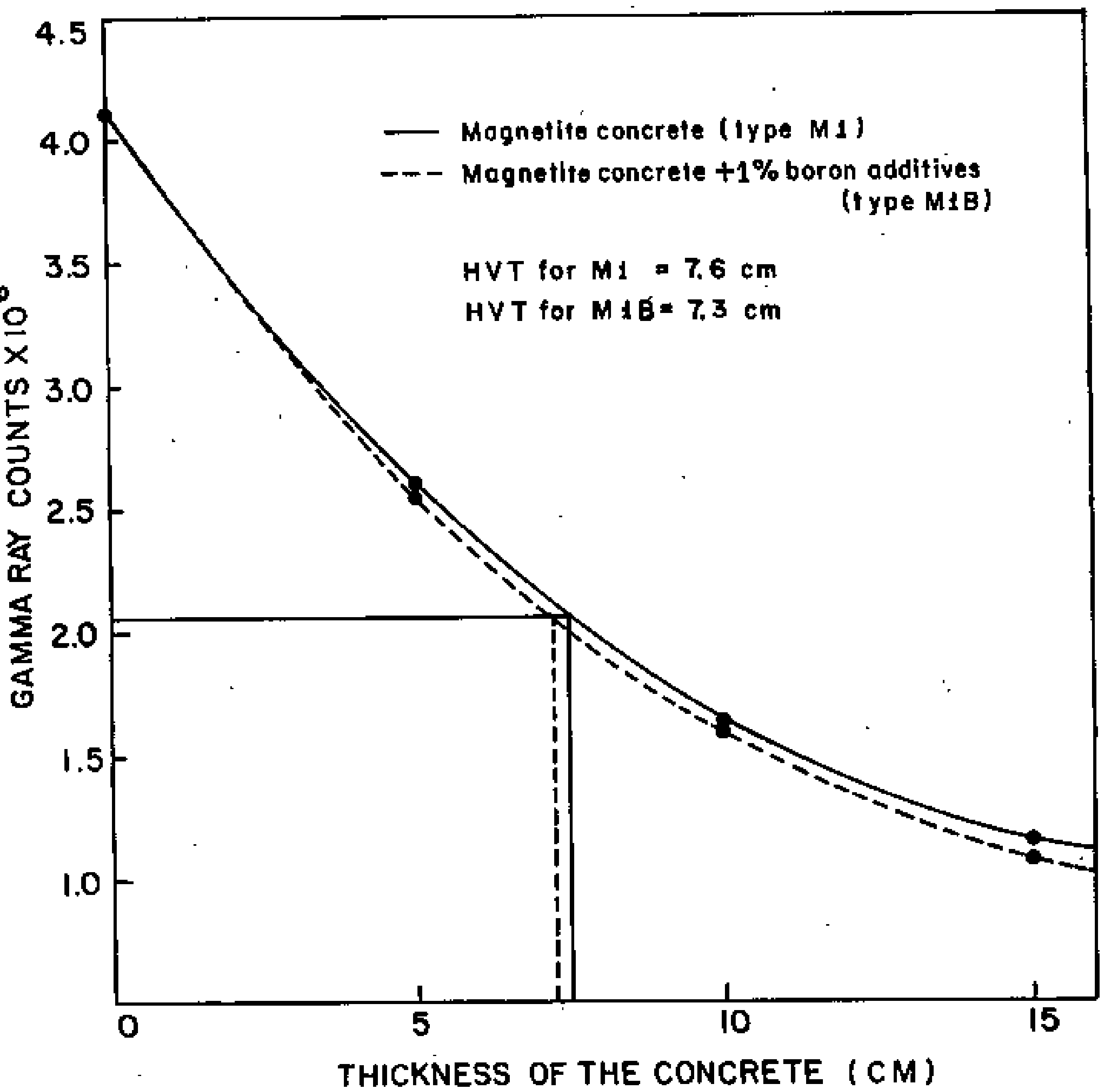


Fig. 25. Gamma ray counts vs. thickness of the concretes M1 and M1B.

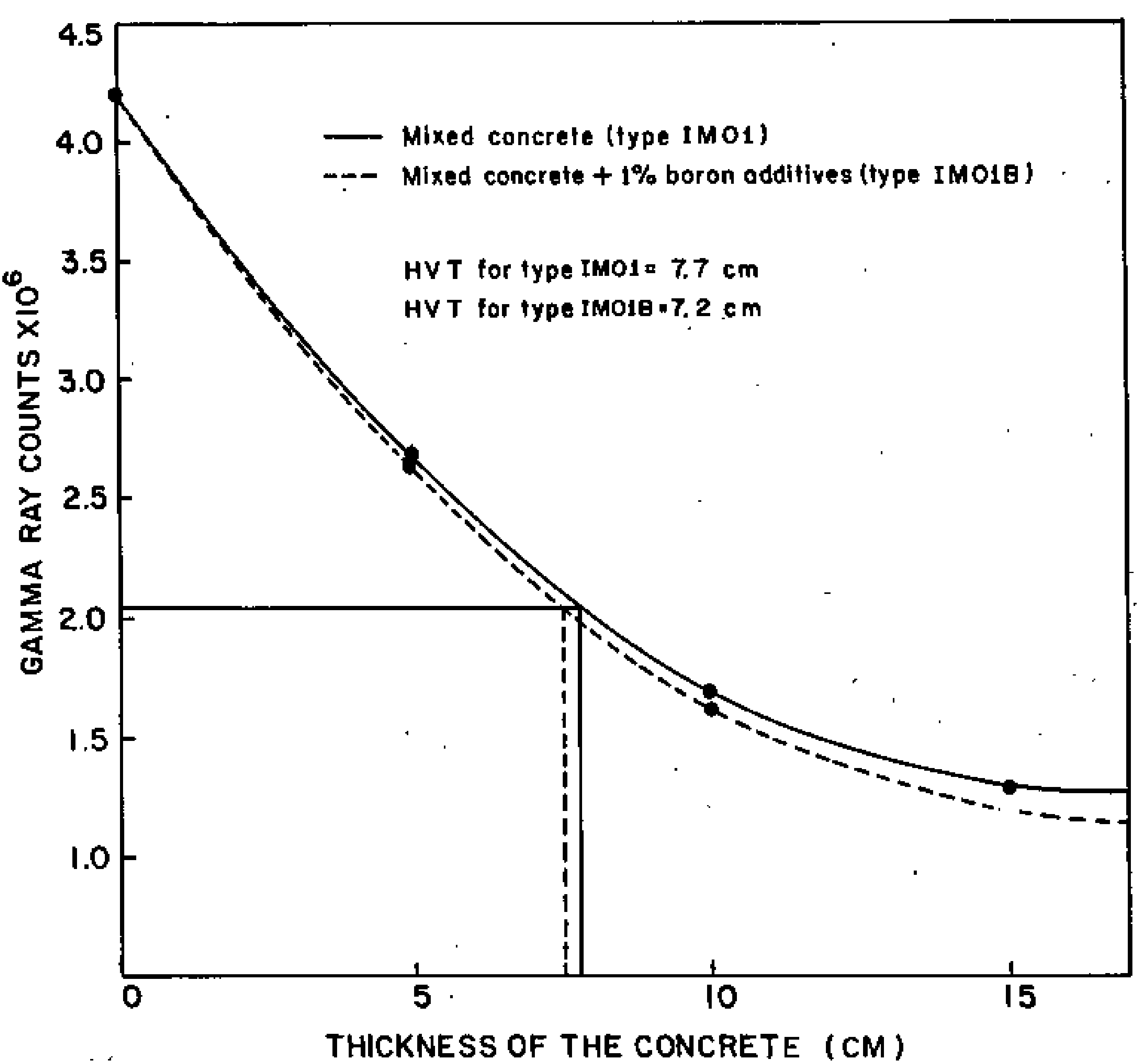


Fig. 26. Gamma ray counts vs. thickness of the concretes IMO1 and IMO1B.

the accumulation of moderated neutron. The relaxation length of gamma ray in boron aditives concrete is also slightly smaller than in concrete without boron additives because accumulation of moderated neutron does not produce secondary gamma radiation. In addition, the absorption of neutron by ^{10}B decreases the number of slow and thermal neutrons in the concrete. This in turn decreases the number of gamma rays emitted from capture of thermal neutrons.

A comparison of the present measurement of neutron and gamma ray attenuation characteristics (Tables 8 & 9) with those obtained in other investigations cannot be made directly because different composition and constituents of the concretes have been used[69]. Furthermore, data on the attenuation of radiation from ^{252}Cf for concretes could not be found in the literature. However, for comparison, the available data on relaxation length for neutron in different concretes are shown in Table 10. The relaxation lengths for neutron obtained in this study are generally comparable with the published values (Table 10), being slightly better for a few cases. The shielding performance depends on the properties of component substances; and by a suitable choice of constituents it is possible to increase the density of concrete and optimize the shielding characteristics for various types of radiation. The locally available magnetite and ilmenite sand have great economic potential for use in the development of high density concrete for biological shielding in a nuclear reactor.

Table 8

Nuclear parameters for neutrons in concretes

Type of concrete	Half value thickness (cm)	Removal cross section (cm^{-1})	Relaxation lengths (cm)
01	8.3	0.0835	11.98
O1B	7.2	0.0963	10.39
I1	7.6	0.0912	10.98
I1B	7.1	0.0976	10.24
M1	7.3	0.0949	10.53
M1B	6.9	0.1004	9.96
IMO1	8.1	0.0856	11.96
IMO1B	6.7	0.1034	9.67

Table 9

Nuclear prameters of gamma ray in concretes

Type of concrete	Half value thickness (cm)	Attenuation coefficients (cm^{-1})	Relaxation lengths (cm)
O1	8.4	0.0825	12.12
O1B	8.1	0.0856	11.69
I1	7.5	0.0924	10.82
I1B	7.4	0.0936	10.68
M1	7.6	0.0912	10.97
M1B	7.3	0.0949	10.54
IMO1	7.7	0.0900	11.11
IMO1B	7.2	0.0963	10.38

Table 10

Typical values of the relaxation lengths for neutron in concretes (Ref. 69)

Type of concrete	Density (g/cm^3)	Relaxation length (cm)
Ordinary	2.30	12.2
Ordinary	2.50	11.1
Ilmenite	3.52	13.4
Magnetite	3.29	8.9
Magnetite	3.50	10.5
Limonite	2.70	9.0
Barytes	3.46	8.0

CHAPTER 5

SUMMARY AND CONCLUSIONS

A nuclear reactor is a copious source of intense radiations. To enable personnel to work in the vicinity of a reactor in operation, it is necessary to attenuate the radiation in a biological shield surrounding the reactor core, so as to reduce the radiation dose to a tolerable level in the region beyond the shield. This is usually achieved by surrounding the reactor core with a sufficiently large mass of high density materials. Concrete is inexpensive and has a satisfactory structural strength as well as shielding characteristics. In view of these advantages, concretes have been widely used as shielding materials. The density of ordinary concrete is often found to be inadequate for a particular shielding application. Heavy concrete with density greater than that of ordinary concrete (2.3 g/cm^3) is suitable for biological shield in a nuclear reactor and other installations having radioactive sources. High density enhances the neutron attenuating properties of concrete moderatly and the gamma attenuating properties strongly. In addition, shields made of high density concrete may save as much as half the thickness for the same effect when ordinary concrete is used.

The most commonly used types of high density concretes are those based on barite, limonite, ilmenite, magnetite, hematite etc. At the beach sand processing pilot plant at

Cox's Bazar, ilmenite and magnetite sand come out in huge quantity as tailings. Investigation showed that these materials have great potential in the development of high density concrete to be used as biological shield in a nuclear reactor[57-58]. Further literature survey reveals that in the past decade many works have been done with ilmenite and magnetite concretes which were successfully applied for radiation shielding[69]. The present study was undertaken to design and fabricate high density concrete by using locally available ilmenite and magnetite sand with the objective of avoiding costly imports and also to develop local know how. The ilmenite and magnetite sand are oxide of iron. One disadvantage of all iron containing concrete is that neutron capture in them produces copious quantities of high energy secondary gamma radiation, which may offset some of the advantages of its use. It is, therefore, necessary to modify iron containing concretes by incorporating elements with a large effective removal corss section for neutrons in order to attenuate . Boron-10, because of its large thermal neutron capture cross section (4000 barn) competes quite successfully with other elements in capturing neutrons when added to concrete even in small quantities. A proportion of 0.9 to 1% by volume of boron in the concrete is recommended by several authors[69]. The reaction involved is ^{10}B (n, alpha) ^{7}Li. ^{7}Li emits 0.49 MeV, capture gamma ray. This low energy gamma ray is easily absorbed in the concrete. Other elements such as iron, will also capture thermal neutrons, while they produce very high energy gamma ray which may escape the shield without being absorbed. For instance, capture of a neutron in iron may produce an 8.0 MeV gamma ray, which has a high probability of escape from the shields. The presence of ^{10}B in an iron bearing concrete will therefore, suppress the generation of the 8.0 MeV gamma ray from neutron capture by the iron present in the concrete shield. In this study, ^{10}B in the form of boric acid (H_3BO_3) is added (1% by volume) to the concrete to suppress the capture gamma rays and thereby to enhance the shielding effectiveness. Concretes, both ordinary and high density type (i.e., ilmenite and megnetite concrete) with and wihout boron additives having dimensions of 12.5 cm x 25 cm x 12.5 cm, for studying attenuating properties of neutrons and gamma rays. Solid cylindrical (15 cm dia. x 30 cm length) slabs were made for studying the physical and structural properties. The density, compressive strength, modulus of elasticity and splitting tensile strength have been measured for these concretes. The nuclear parameters such as half value thickness, attenuation coefficient, removal cross section and relaxation length have been evaluated for these concretes. These parameters were measured using a spontaneous fission source (^{252}Cf) which emits both fission neutrons (thermal to <10 MeV) and fission gamma rays (several keV to <3 MeV). A long counter comprising of BF_3 gas proportional counter and suitable cylindrical moderator was used to monitor neutron components, while a NaI(Tl) detector was used to monitor gamma ray components. The results of the attenuation measurements of neutron and gamma rays for these concretes are presented in the Figures 11-18 and Tables 8-9. Tables 8-9 give a better understanding of the shielding efficiency of the different types of concretes for use as biological shields in nuclear reactors. As a result of these measurements, the following conclusions were drawn:

(i) The densities of heavy concrete (i.e., ilmenite, magnetite and mixed concrete) with and without boron additives are considerably better than those of ordinary concrete with and without boron additives.

(ii) The shielding properties of boron loaded heavy concretes are better than those of ordinary concretes with and without boron additives and slightly better than those of heavy concretes without boron additives.

(iii) For gamma ray attenuation, the ilmenite concrete (Type I1) is slightly better than the magnetite concrete (Type M1) and mixed concrete (Type IMO1) but much better than ordinary concrete (Type O1).

(iv) For the attenuation of both neutron and gamma radiation, boron loaded mixed concrete (Type IMO1B) is more or less as good as boron loaded magnetite concrete (Type M1B), but is slightly better than boron loaded ilmenite concrete (Type I1B) and considerably better than ordinary concrete both with and without boron additives.

The present investigation shows that heavy concretes (i.e., ilmenite, magnetite and mixed concrete) can be used as biological shield in nuclear reactor, gamma irradiation plant, particle acce le rator, neutron generation, x-ray installation and other similar installations having radiation sources. It is, therefore, hoped that this investigation will contribute significantly to a better understanding of the shielding efficiency of the different types of concretes with and without boron additives.

In continuation of the present study, it is proposed to undertake the following studies in connection with the shielding properties of concretes:

(1) To perform integral experiments to study transport properties of neutron and gamma ray in concretes (both heavy and ordinary) using neutron and gamma radiation beam from the 3 MW TRIGA Mark-II research reactor at Atomic Energy Research Establishment (AERE), Savar of Bangladesh Atomic Energy Commission, Dhaka.

(2) To develop simplified emperical model and computer codes for neutron and gamma transport calculations.

(3) To furnish comprehensive supporting data bases for shielding design requirements.

(4) To study the sensitivity properties of individual element in the concrete to get an optimum mixture.

(5) To investigate variation of water content in concretes with temperature and their effects on structural and shielding properties.

(6) To study the radiation effect on the concrete.

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