Shielding Calculations for the Design of Neutron Radiography Facility Around Parr

Mirza Muhammad Ashraf
Abdur Rahim Khan

NON-DESTRUCTIVE TESTING GROUP
RADIATION & ISOTOPE APPLICATIONS DIVISION
Pakistan Institute of Nuclear Science & Technology
P. O. Nilore Islamabad.
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MIRZA MUHAMMAD ASHRAF
ABDUR RAHIM KHAN

NON-DESTRUCTIVE TESTING GROUP
RADIATION & ISOTOPE APPLICATIONS DIVISION
PAKISTAN INSTITUTE OF NUCLEAR SCIENCE AND TECHNOLOGY
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ABSTRACT

Shielding calculations for Neutron Radiography facility, proposed to be established around PARR have been carried out using two group diffusion theory and other shielding formulae.

A 65 cm thick graphite block is provided as a source block. A divergent beam collimator with a collimating ratio of 127 and source size 16.2 mm $\phi$ has been designed to provide more imaging area. 25 cm thick shield of bismuth is provided to reduce high energy gamma radiation and to slow down fast neutrons. The neutron to gamma ratio is $2.95 \times 10^7$ cm$^{-2}$ mRem$^{-1}$. Beam shutter is 39.6 cm thick and comprises of B$_4$C, paraffine wax, aluminium, lead and steel as shielding materials for neutron and gamma rays. The front and side wall thicknesses of beam trap are 39.6 cm and 20 cm respectively. The front and side wall thicknesses of enclosure are 50 cm and 30 cm respectively.

The expected dose rate in the beam and lateral direction of the radiation enclosure is 0.6 and 0.02 mRem/hr respectively.
1. INTRODUCTION

There are a number of non-destructive techniques used for inspection of materials. Neutron radiography is one of them. Neutron radiography is analogous to X-ray and gamma ray radiographic techniques. It can be used to examine non-active as well as active materials and components which is an advantage over the other non-destructive testing techniques.

To design a neutron radiography facility around a research reactor using one of the beam tubes, estimates of thermal flux reaching the outer end of the beam tubes are required for radiographic purposes. Shielding calculations for the beam shutter, beam trap and radiation enclosure are necessary for safety consideration. There are quite a number of methods used to calculate neutron penetration. Some of these methods are mentioned below.

i) Straighthead Approximation

ii) Diffusion Theory

a) Modified One Group Theory

b) Two Group Theory

iii) Moments Method (Transport Theory)

iv) Monte Carlo Method

v) Direct Integration of Boltzmann equation

In neutron radiography only thermal neutrons play effective role in making radiographic images. For radiographic purposes, neutrons coming from the reactor core can be divided into two groups i.e thermal and fast groups. To calculate thermal flux resulting directly from the core and from thermalization of fast neutrons in graphite block and bismuth metal, two group diffusion
theory has been used in this report [1,4]. Calculation of gamma intensity at various depths of shielding materials are based on attenuation methods [1-4]. In addition, the functions of collimator, source block, beam filters, beam shutter and beam trap have been discussed.

2. METHODS OF SHIELDING CALCULATIONS

A number of shielding formulae and approximations are used to calculate neutron and gamma radiation intensities at various depths of shields. Diffusion theory is used to calculate the intensities of thermal and fast neutrons. Simple attenuation methods are applied to compute gamma radiation intensities [1-3]. These are discussed in the following sections.

2.1 NEUTRON SHIELDING

2.1.1 DIFFUSION THEORY

The physical basis of this theory implies that the current in a uniform medium is governed by \(-D\nabla^2 \phi\) and the balance of neutrons must be conserved [1].

Production = Absorption + Leakage

Keeping in view the production of neutrons and their absorption along with leakage, diffusion theory results in the following mathematical formulation.

\[-D\nabla^2 \phi_{\text{th}} + \Sigma_a \phi_{\text{th}} - S = 0\]  \hspace{1cm} (1)

where in cartesian coordinates, \((x, y, z)\)

\[\nabla^2 = \frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2} + \frac{\partial^2}{\partial z^2}\]

\[S = \text{source of neutrons, neutrons cm}^{-3} \text{ sec}^{-1}\]
\[ \Sigma_a = \text{macroscopic absorption cross-section, cm}^{-1} \]

\[ D = \text{diffusion co-efficient, cm} \]

\[ \nabla = \text{Laplacian operator} \]

\[ \phi_{th} = \text{flux of thermal neutrons i.e. neutrons cm}^{-2} \cdot \text{sec}^{-1} \]

Equation (1) is valid if the following conditions are true.

a) the thermal flux must be finite and non-negative in the region where this equation implies.

b) the flux \( \phi \) tends to zero as the distance increases to infinity.

c) the flux \( \phi \) and the current \(-D\nabla\phi\) must be continuous at the interface of two media.

d) \( \Sigma_a/\Sigma_s \) and \( d\phi_{th}/dx \) must be small where \( \Sigma_s \) is the macroscopic scattering cross section.

2.1.1.1 MODIFIED ONE GROUP TREATMENT

In one group treatment, to simplify the neutron penetration calculations, the source term in equation (1) can reasonably be approximated by

\[ S = \Sigma \phi_f(x) \]  

(A)

where \( \Sigma \) is the effective macroscopic slowing down cross-section and \( \phi_f(x) \) represents the fast neutron flux distribution at any point \( x \).

The fast flux \( \phi_f(x) \) is defined by
\[
\phi_f(x) = \phi_f(0)e^{-\Sigma x} \quad \text{(B)}
\]

where \(\phi_f(0)\) is the fast flux at \(x=0\) i.e. the fast flux at the inner face of the shield. Substituting equations (A) & (B) into equation (1) and rewriting the obtained equation in one dimensional cartesian coordinates only.

\[
D_{th}(d^2\phi_{th}/dx^2) - \Sigma_a \phi_{th} + \Sigma \phi_f(0)e^{-\Sigma x} = 0 
\]

(2)

Equation (2) can be rearranged as

\[
d^2\phi_{th}/dx^2 = (\Sigma_a / D_{th})\phi_{th} + (\Sigma / D_{th})\phi_f(0)e^{-\Sigma x} = 0
\]

(3)

Solving the equation, the general solution can be written as

\[
\phi_{th}(x) = Ae^{kx} + Be^{-kx} + Ce^{-\Sigma x}
\]

(4)

where \(k^2 = \Sigma_a / D_{th}\) and \(\phi \to 0\) as \(x \to \infty\)

therefore \(A = 0\)

Using the conditions mentioned above, the solution of the equation is

\[
\phi_{th}(x) = \phi_{th}(0)e^{-kx} + \Sigma\phi_f(0)/(D_{th}(k^2-\Sigma^2)) \{e^{-\Sigma x} - e^{kx} \}
\]

(5)

Equation (5) gives the thermal flux distribution through a shield of thickness \(x\)

2.1.1.2 TWO GROUP APPROXIMATION

To find the nature of flux distribution, the two-group equations can be written as follows [2].
\[-D_f \nabla^2 \phi_f + \Sigma_{af} \phi_f = S\] \hspace{1cm} (6)

and

\[-D\nabla^2 \phi_{th} + \Sigma_{ath} \phi_{th} = \Sigma_{af} \phi_f\] \hspace{1cm} (7)

The source term is taken to be equal to zero i.e. \(S = 0\) as it represents fast neutrons in the beam tube. So, the one dimensional equations can be rewritten as follows.

\[d^2 \phi_f / dx^2 - K_f^2 \phi_f = 0\] \hspace{1cm} (8)

\[d^2 \phi_{th} / dx^2 - K_{th}^2 \phi_{th} = \Sigma_{af} \phi_f / D_{th}\] \hspace{1cm} (9)

The solution of the above mentioned equations can be written as:

\[\phi_f(x) = \phi_f(0)e^{-k_f x}\] \hspace{1cm} (10)

\[\phi_{th}(x) = \phi_{th}(0)e^{-k_{th} x} + \Sigma_{af}\phi_f(0) / [D_{th}(K_f^2 - K_{th}^2)].(e^{-K_{th} x} - e^{-k_f x}]\] \hspace{1cm} (11)

where

\[K_{th} = \sqrt{\Sigma_{ath} / D_{th}}\] and \[K_f = \sqrt{\Sigma_{af} / D_f}\]

2.1.1.3 AGE DIFFUSION APPROXIMATION

To estimate the thermal flux distribution, provided fast neutron flux is known, age-diffusion approximation can be used. The results obtained by using the following formula are approximate and are helpful to assess the shield thickness required for different facilities.
The thermal flux distribution is given by [1]

\[ \phi_{th}(x) \approx (\Sigma / \Sigma_c) \phi_f(0) e^{-\Sigma (x-\tau/\lambda)} \]  \hspace{1cm} (12)

where

- \( \phi_{th} \) = thermal flux distance at \( x \), neutrons/cm\(^2\)/sec
- \( \lambda = 1/\Sigma \), \( \lambda \) is the relaxation length, cm\(^{-1}\)
- \( \Sigma_c \) = capture cross-section of shield material, cm\(^{-1}\)
- \( \phi_f(0) \) = fast neutron flux at \( x = 0 \), neutron.cm\(^{-2}\).sec\(^{-1}\)
- \( \tau = \) Fermi age, cm\(^2\)

2.2 GAMMA SHIELDING

In a beam port of a reactor, the emerging beam of radiation consists of gamma rays and neutrons of different energies. These radiation are as a result of direct radiation from the reactor core and produced through radiative capture of neutrons. The gamma radiation present in the radiation beam will spoil the image on the radiograph. To have better quality images, it is required that gamma-ray contribution in the beam should be very small. To find out the gamma radiation intensity at any depth of the shield, the following relation can be used [1, 4, 9].

\[ \phi_o(x) = \phi_{in} e^{-\mu x} B(\mu x) \]  \hspace{1cm} (13)

where

- \( \phi_{in} \) = gamma flux entering the shield, photons.cm\(^{-2}\).sec\(^{-1}\)
\( \phi_o(x) \) = gamma flux at the x depth of the shield, photons cm\(^{-2}\) sec\(^{-1}\)

\( \mu \) = linear attenuation coefficient, cm\(^{-1}\)

\( x \) = thickness of the shield, cm

\( B \) = build up factor.

The build up factors for gamma rays have been computed using

\[ B = 1 + \mu r/4 + (\mu r)^2/10 \]  \hspace{1cm} (14)

and Taylor's method for different shields \([5,17]\).

### 2.2.1 GAMMA RADIATION FROM RADIATIVE CAPTURE OF NEUTRONS

The intensity of gamma rays resulting from radiative capture at any depth of a shield can be calculated by using the following mathematical relation \([2]\).

\[ \phi_{\gamma}(t) = 1/2 \int_0^T \phi_{th}(x) \sum_i \gamma_i E_i(\mu(x-t)) dt + 1/2 \int_0^T \phi_{th}(x) \sum_i \gamma_i E_i(\mu(x-t)) dt \]  \hspace{1cm} (15)

where

\[ \phi_{\gamma}(t) = \text{gamma intensity at thickness } t \text{ of the shield, Mev cm}^{-2} \text{ sec}^{-1} \]

\[ \phi_{th}(x) = \text{thermal neutron flux at point } x \text{ in the shield, neutron cm}^{-2} \text{ sec}^{-1}. \]

\( T \) = maximum thickness of shield, cm

\( E_i \) = exponential integral

\( \gamma_i \) = energy of gamma ray emitted by the constituent of shield through radiative capture, Mev.
\[ \Sigma_{a,i} = \text{thermal neutron capture cross-section of } ith \text{ constituent of shield material, cm}^{-1} \]

The total gamma radiation in the beam port will then be the sum of gamma rays coming directly from the core and those resulted from radiative capture in the source block and bismuth filter placed in the beam tube to extract a beam of thermal neutrons for radiographic purposes [3].

3. SOURCE BLOCK

The major factor governing the beam intensity for radiographic applications is the thermal neutron flux available at the base of the collimator. This flux depends upon the thermal neutron flux from the reactor core and the moderator material placed inside the beam tube to thermalize the fast neutrons in the radiation beam. The moderators which can be used to thermalize the fast neutrons are graphite, \( \text{H}_2\text{O}, \text{D}_2\text{O}, \text{ZrH}_2 \) etc. [6,18-19]. Graphite is preferred as a moderator (source block) because of its large diffusion length. However, for accelerators and neutron sources polyethylene can be a good choice. A minimum separation between source block and the base of the collimator is required to get a uniform flux. For graphite moderator, the separation between the source block and the base of the collimator is 14 cm [6,7].

4. BISMUTH FILTER

The primary purpose of this filter is to reduce the high energy gamma radiation in the radiation beam coming from the core and through radiative capture i.e., \((n,\gamma)\) [6,14-16,20]. The effect of bismuth
filter on thermal neutrons is negligible because of its smaller absorption cross section for thermal neutrons. Therefore this filter is used to filter out the beam of radiation which otherwise may assist to reduce image quality of the radiograph.

5. COLLIMATOR

Reactor sources, for neutron radiography, are present all over the world and have high thermal neutron intensities, $10^6 n/cm^2/sec$ or higher. So, collimators are needed to provide a collimated beam suitable for radiography [5,7,9]. One thing to be pointed out is that beam tubes which have high ratio of thermal to fast neutron intensities and low gamma intensities are although desirable but not necessary. The desirable features for the purpose of radiographic inspection are as follows:

i) high thermal neutron intensity.

ii) low gamma radiation intensity

iii) relatively low fast neutron intensity

iv) large area coverage by the neutron beam.

v) low angular divergence.

To achieve the above objectives, extra moderating material is used. For low angular divergence, collimators of many types are used. In ANL, collimators are made by filling stainless steel shells with high density concrete. This technique results in better resolution. Even it is possible to achieve the same results by fashioning simple apertures from some neutron absorbing materials i.e. cadmium, $B_4C$ or Indium etc.

To limit the divergence of the beam multi-slit technique
is applied and the collimator is lined with neutron absorbing material i.e. cadmium & B_4 C etc. The detailed layout diagram of the collimator is shown in Figure No. 1.

5.1. COLIMATING MATERIALS

Collimating materials for neutrons of lower energy are phosphor, aluminium, bronze & steel etc [4,7].

To improve the definition of the image formed, L/d (length to diameter ratio) should be increased. This is analogous to the focal spot size and source to film distance in x-ray or gamma-ray radiography but increase in the focal to film distance will reduce the intensity of radiation reaching the film and need more exposure time to produce good quality images. It is advisable to use very large L/d ratio. Different neutron radiographic facilities are utilizing different L/d ratios depending on type of applications but L/d = 250 is sufficient for all kind of neutron radiography applications (i.e. the image sharpness is not affected) [6,12]. One point which does not go in favour of straight beam collimator is that the flux at the emerging end of the collimator does not vary inversely as L^2. So for large imaging area, divergent type collimators is a better choice. The estimated flux at the emerging end of the collimator is governed by the following equation [6,7,18].

\[ \tilde{\phi} = \frac{A}{16} \left( \frac{d}{L} \right)^2 \left( \phi_1 + 1/\Sigma \frac{\partial \phi}{\partial x} \right) \]  \quad (16)

where

\[ \phi_1 = \text{neutron intensity at the base of the collimator}, \]
\[ \phi = \text{neutron intensity at exit, neutron cm}^{-2} \text{sec}^{-1} \]
\[ A = \text{collimator area at exit, cm}^2 \]
\[ d = \text{diameter of inlet aperture, cm} \]
\[ L = \text{length of collimator, cm} \]
\[ \Sigma = \text{macroscopic total cross section of the moderator, cm}^{-1} \]
\[ \partial \phi / \partial x = \text{flux gradient at the inner face of moderator} \]

Since the flux gradient is usually very small, the above equation reduces to

\[ \phi = \frac{1}{16} (\frac{d}{L})^2 A \phi_i \]  \hspace{1cm} (17)

Thus the respective neutron and gamma ray fluxes reaching at the emergent end of the collimator with a collimating ratio \( L/d \) are computed by equations (18) and (18-A) \([6,7]\).

\[ \phi_n = \frac{1}{16} (\frac{d}{L})^2 \phi_i \hspace{1cm} \text{for neutrons} \]  \hspace{1cm} (18)
and

\[ \phi_g = \frac{1}{4} (\frac{d}{L})^2 \phi_i \hspace{1cm} \text{for gamma rays} \]  \hspace{1cm} (18-A)

5.2. DESIGN OF COLLIMATOR

The collimating ratio \( L/d \) plays an important role in the resolution of images obtained on the radiographic film. To determine \( L/d \) ratio, collimator ratio, the following relation can be used \([6-7,9-10]\).

\[ g = \frac{d L_i}{L_s} \]  \hspace{1cm} (19)

where

\[ d = \text{source size, aperture diameter} \]
If \( L_0 = L \), the length of the collimator, then

\[
\frac{L}{d} = \frac{L_f}{U_g} \tag{20}
\]

Since \( L_1 \) is usually taken as the specimen thickness, and \( U_g \) from 0.1 mm to 0.5 mm depending on the type of radiographic technique, so one can easily estimate \( L/d \) ratio. For better neutron intensity at the imaging end a collimator with \( L/d = 127 \) and 16.2 mm source size will serve the purpose. A choice for \( L/d = 250 \) will reduce the source size to 1 cm which in turn will reduce the beam intensity.

5.3. APERTURE SIZE

Since the resolution of the image obtained through radiography is directly proportional to the source size i.e.

\[
U_g \propto d
\]

where \( d \) is the source size, so the aperture of the collimator should be well defined \([7]\). To achieve better resolution, the material for the inlet face should be opaque to neutrons (especially for those neutrons to which converter screens are sensitive) e.g. gadolinium, indium and dysprosium, etc. More over boron carbide and boral etc are also used for defining inlet aperture.

The thickness of the aperture can be calculated by equation
\[ A = 1 - e^{-\Sigma x} \]  \hspace{1cm} (21)

Where

- \( A \) = attenuation factor
- \( \Sigma \) = macroscopic cross section of the aperture material, \( \text{cm}^{-1} \)
- \( x \) = thickness of the aperture, \( \text{cm} \)

For a better choice of aperture thickness, \( A \approx 0.95 \) can be used. The thickness of gadolinium and indium required are 1.6 mm and 4.1 mm respectively [7,12].

6. BEAM SHUTTER

To use beam tube frequently for neutron exposures to different objects in case of neutron radiography, beam shutters are required [9,10].

There are a number of materials which are used in the fabrication of beam shutters. The neutrons are slowed down first and then absorbed. The materials used for beam shutters are paraffine wax with borax, lead, boral sheets, cadmium and \( \text{B}_4\text{C} \) etc. The design of beam shutters should be such that there is no streaming of gamma rays and neutrons. Layout sketch of the beam shutter is shown in Figure 2.

7. BEAM TRAP

In most experiments of nuclear physics, the reactors are used as neutron source. The experimental set up used for collimating the beam only intercepts a few percent of radiations and the dose rate from such a beam of neutrons will be very high, \( 10^5 \) times the maximum
permissible level. It is very essential to have a beam catcher to be placed as close to the experimental apparatus as possible [3, 9, 11, 20]. Furthermore, it must be small because of the space problems around reactors in the presence of other experimental set-ups used for other purposes. To facilitate other experimentalists to carry out their research work, a beam catcher has been provided to shield the radiation.

8. ENCLOSURE

Since there is scattering of γ-rays and neutrons from beam trap and the object under test, an additional shielding is required. Moreover, the scattering of neutron and γ-rays will be in all directions, so an enclosure of high density concrete is necessary to facilitate the radiation workers to carry out radiography and other experiments in the vicinity of neutron radiographic facility [11, 12, 21].

9. RESULTS

9.1. NEUTRON FLUXES

The thermal and fast fluxes of neutrons at the starting point of beam tube no. 6 are as follows [13, 24].

\[ \phi_{\text{fast}} = 2.1 \times 10^{13} \text{ neutrons/cm}^2/\text{sec} \]

\[ \phi_{\text{th}} = 4.2 \times 10^{13} \text{ neutrons/cm}^2/\text{sec} \]

The neutron fluxes mentioned above are incident fluxes on the source.
block, the moderator, placed inside the tube No. 6 to thermalise fast neutrons.

9.1.1. SOURCE BLOCK

Graphite block is used as source of thermal neutrons, the thermal and fast fluxes calculated at 65cm depth of graphite shield using equation (10) and (11) are

\[ \phi_{\text{fast}} = 5.427 \times 10^{11} \text{ neutrons/cm}^2/\text{sec} \]
\[ \phi_{\text{th}} = 2.40 \times 10^{13} \text{ neutrons/cm}^2/\text{sec} \]

The trend of neutron flux in graphite shield is shown in Figure 3. An increase in the thermal neutron flux inside the source block is obviously because of thermalization of fast neutrons. The neutron parameters for various shielding materials are listed in Tables I to III. The neutron and gamma radiation fluxes also go through a decrease because of air attenuation between graphite block and base of the collimator.

9.1.2. BISMUTH FILTER

Bismuth filter is used to reduce high energy gamma rays and also to slow down fast neutrons i.e. thereby increasing thermal neutron flux, the values of the fast and thermal fluxes as calculated by equations (10) and (11) at 25 cm depth of bismuth shield, are as follows:

\[ \phi_{\text{fast}} = 4.567 \times 10^{10} \text{ neutrons/cm}^2/\text{sec} \]
\[ \phi_{th} = 1.23 \times 10^{13} \text{ neutrons/cm}^2/\text{sec} \]

The flux trend through bismuth filter is shown in Figure 4. The neutron fluxes reaching at the emergent end of beam collimator with \( L/d = 127 \), using equation (17) and (18) are as follows.

\[ \phi_{fast} = 1.7 \times 10^7 \text{ neutrons/cm}^2/\text{sec} \]

and

\[ \phi_{th} = 4.76 \times 10^7 \text{ neutrons/cm}^2/\text{sec} \]

9.2. GAMMA RADIATION

The gamma radiation accompanied by the neutron beam are due to fission, radiative capture of moderator and fission fragments, etc. Each of these emit gamma radiation of different energies, therefore it is necessary to consider convenient energy groups for estimating shielding and filtering thicknesses.

The source term of neutrons is given by

\[ \sum_{a,i} \phi_{th} \gamma_i \]

where \( \phi_{th} \) is the thermal flux, \( \Sigma_{a,i} \) is the absorption cross-section of \( i \)th material in the core and \( \gamma_i \) is the gamma ray energy associated with \( i \)th material. Gamma flux at any point \( x \) in the shield from the core of the reactor is given by

\[ \phi_\gamma = \left( S_v / 2 \mu c \right) \epsilon_i (\mu x) B(\mu x) \]

(22)

where \( \mu \) is the linear absorption coefficient of core, \( S_v \) is the volumetric source and \( B \) is the build up factor. The gamma radiation fluxes have been calculated using equation (22). The attenuation and
buildup are neglected because the beam tube starts very close to the core. The gamma fluxes are divided into four convenient energy groups listed below.

<table>
<thead>
<tr>
<th>Group No</th>
<th>Energy Range, Mev</th>
<th>Energy flux, Mev/cm²/sec</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0-1</td>
<td>$6.15 \times 10^{13}$</td>
</tr>
<tr>
<td>2</td>
<td>1-2</td>
<td>$5.05 \times 10^{13}$</td>
</tr>
<tr>
<td>3</td>
<td>2-4</td>
<td>$1.976 \times 10^{13}$</td>
</tr>
<tr>
<td>4</td>
<td>4-6</td>
<td>$4.55 \times 10^{12}$</td>
</tr>
</tbody>
</table>

As the beam port is very close to the core, the attenuation of water can be neglected. The attenuation coefficients of different materials used for gamma shielding are listed in Table IV.

9.2.1. ATTENUATION THROUGH GRAPHITE BLOCK (SOURCE BLOCK)

The gamma intensity at 65 cm depth of graphite shield for each energy group is calculated by equation (22). The fluxes are:

<table>
<thead>
<tr>
<th>Group No</th>
<th>Energy Range, Mev</th>
<th>Energy flux, Mev/cm²/sec</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0-1</td>
<td>$1.320 \times 10^{12}$</td>
</tr>
<tr>
<td>2</td>
<td>1-2</td>
<td>$3.203 \times 10^{12}$</td>
</tr>
<tr>
<td>3</td>
<td>2-4</td>
<td>$2.52 \times 10^{12}$</td>
</tr>
<tr>
<td>4</td>
<td>4-6</td>
<td>$7.831 \times 10^{11}$</td>
</tr>
</tbody>
</table>

During the interaction of thermal neutrons with graphite shield (source block), 4.95 Mev and 3.68 Mev gamma rays are emitted due to radiative capture. The gamma contribution resulting at 65 cm depth of graphite block calculated by equation No.(15) is $5.67 \times 10^{11}$.
and $1.44 \times 10^{11}$ Mev/cm$^2$/sec respectively. The gamma intensity at the end of source block i.e. the gamma intensity incident on bismuth filter can be rearranged for each group.

<table>
<thead>
<tr>
<th>Group No</th>
<th>Energy Range, Mev</th>
<th>Energy flux, Mev/cm$^2$/sec</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0-1</td>
<td>$1.32 \times 10^{12}$</td>
</tr>
<tr>
<td>2</td>
<td>1-2</td>
<td>$3.203 \times 10^{12}$</td>
</tr>
<tr>
<td>3</td>
<td>2-4</td>
<td>$2.660 \times 10^{12}$</td>
</tr>
<tr>
<td>4</td>
<td>4-6</td>
<td>$1.350 \times 10^{12}$</td>
</tr>
</tbody>
</table>

Bismuth is used to filter out the high energy contents of gamma rays, and also to slow down fast neutrons in the beam port. The gamma fluxes for each energy group at 25 cm depth of bismuth shield is calculated by equation (13) and are listed as follows.

<table>
<thead>
<tr>
<th>Group No</th>
<th>Energy Range, Mev</th>
<th>Energy flux, Mev/cm$^2$/sec</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0-1</td>
<td>$1.69 \times 10^5$</td>
</tr>
<tr>
<td>2</td>
<td>1-2</td>
<td>$1.66 \times 10^8$</td>
</tr>
<tr>
<td>3</td>
<td>2-4</td>
<td>$3.37 \times 10^8$</td>
</tr>
<tr>
<td>4</td>
<td>4-6</td>
<td>$1.57 \times 10^8$</td>
</tr>
</tbody>
</table>

The gamma dose rate through graphite, bismuth filter and the beam collimator is plotted in Figure 5. For the worst possible case the attenuation of gamma rays between the source block and the base of the collimator has been neglected because it involves complicated calculations. Thus the neutron flux leaving the graphite
is assumed as the flux reaching the base of the collimator. The intensity of gamma rays resulting from radiative capture of neutrons from bismuth block calculated by equation (15) is $5.89 \times 10^{11}$ Mev/cm²/sec as 4.17 Mev gamma ray is associated per neutron capture.

Since this energy flux is in the range of group 4, so the energy flux due to gamma rays of various energies after adding contribution from radiative capture can be rewritten as.

<table>
<thead>
<tr>
<th>Group No</th>
<th>Energy Range, Mev</th>
<th>Energy flux, Mev/cm²/sec</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0-1</td>
<td>$1.69 \times 10^{5}$</td>
</tr>
<tr>
<td>2</td>
<td>1-2</td>
<td>$1.66 \times 10^{8}$</td>
</tr>
<tr>
<td>3</td>
<td>2-4</td>
<td>$3.37 \times 10^{5}$</td>
</tr>
<tr>
<td>4</td>
<td>4-6</td>
<td>$5.91 \times 10^{10}$</td>
</tr>
</tbody>
</table>

The gamma ray fluxes at the end of the beam collimator with $L/d = 127$ are calculated by equation (18-A).

Thus the group fluxes and calculated dose rates at the outer end of the beam tube number 6 can be written as follows.

<table>
<thead>
<tr>
<th>Group No</th>
<th>Energy flux, Mev/cm²/sec</th>
<th>Dose Rate, mRem/hr</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>2.62</td>
<td>$4.60 \times 10^{-3}$</td>
</tr>
<tr>
<td>2</td>
<td>$2.57 \times 10^{3}$</td>
<td>7.59</td>
</tr>
<tr>
<td>3</td>
<td>$5.224 \times 10^{3}$</td>
<td>24.89</td>
</tr>
<tr>
<td>4</td>
<td>$9.145 \times 10^{5}$</td>
<td>$5.805 \times 10^{3}$</td>
</tr>
</tbody>
</table>

The cumulative dose rate due to gamma rays is $5.813 \times 10^{3}$ mRem/hr and the n/r ratio is $2.93 \times 10^{7}$ cm⁻².mR⁻¹, which is more than $5.0 \times 10^{4}$ cm⁻².mRem⁻¹, a threshold to produce
good radiographic images with minimal gamma interference [6].

9.3. DESIGN LEVEL

Considering the dose rate due to all types of radiation encountered in extraction of a beam from reactors, a surface dose rate of 55 mRem/hr is taken as the design level for the shutter as well as for the beam trap calculations. This limit of design level is taken keeping in view the exposure of neutrons to the worker during the setting up of radiographic conditions and mobility of the shutter system. As the NDT group deals with other types of radiation such as X-rays and gamma rays, the exposure from this type of radiation should be a part of the total permissible dose per year.

Number of exposures per year = 200
Average time for setting exposure = 6 minute.
Total exposure to a worker per year = \(6 \times 200 = 1200\) minutes.
Dose level at the surface of the shutter = 55 mRem/hr
Total dose received per year = \((1200/60) \times 55\) mRem
= 1100 mRem
= 11.0 Rem

The dose permissible for a radiation worker per year is 5 Rem, so 1.1 Rem dose per year due to neutrons and gamma radiation to the worker is well below the permissible level.

9.4. BEAM SHUTTERS

Beam shutters are required to use the neutron beam
ports frequently for experimental purposes. The neutron and gamma fluxes reaching the beam shutter from core and radiative capture from the shielding materials in the beam tube calculated by the equations (18) and (18-A) are.

\[ \phi_{\text{fast}} = 1.77 \times 10^5 \text{ neutrons/cm}^2/\text{sec} \]
\[ \phi_{\text{th}} = 4.76 \times 10^7 \text{ neutrons/cm}^2/\text{sec} \]
\[ \phi_{\gamma} = 9.145 \times 10^6 \text{ Mev/cm}^2/\text{sec} \]

With gamma radiation dose rate \(= 5.813 \times 10^3\) mRem/hr, the materials used in the design of beam shutter are 2 cm aluminium, 20 cm paraffine wax, 0.6 cm boron carbide, 15 cm of lead and 2 cm steel respectively. The neutron fluxes and gamma radiation dose rate leaving the beam shutter are:

\[ \phi_{\text{th}} = 1.640 \times 10^3 \text{ neutron/cm}^2/\text{sec} \]
\[ D_{\gamma} = 49.2 \text{ mRem/hr} \]
\[ D_n = 3.4 \text{ mRem/hr} \]

Total dose rate \(= 52.6\) mRem/hr

As the shielding thickness estimated by two group theory is approximate, if required additional shielding can be provided after measurement of the dose rates experimentally. The gamma radiation attenuation coefficients for different shielding materials are listed in Table-IV.
9.5. BEAM TRAP

FRONT WALL:

The beam trap shielding thickness in the direction of radiation beam is 39.6 cm and it contains 4 cm of aluminium, 15 cm of lead, 20 cm of borated wax and 0.6 cm of boron carbide. The fluxes leaving the beam trap surface are [20, 21].

\[
\phi_{\text{fast}} = 1450 \quad \text{neutron/cm}^2/\text{sec}
\]

\[
\phi_{\gamma} = 7.27 \times 10^2 \quad \text{Mev/cm}^2/\text{sec}
\]

The thermal neutrons are completely attenuated. Hence the surface dose at the trap surface in the beam direction.

\[
\text{Dose Rate} = \phi_{\text{fast}} U_f + \phi_{\gamma} U_{\gamma}
\]

where \( U_f \) and \( U_{\gamma} \) are the dose conversion factors [3].

Dose Rate = 1450 \times 0.03 + 7.27 \times 10^2 \times 6.35 \times 10^{-3} = 48.12 \text{ mRem/hr}

SIDE WALL:

When the neutron beam strikes on aluminium .1 cm in thickness, the radiation (neutron and gamma radiation) are scattered in \( 4\pi \) direction [21]. The neutron and gamma fluxes after traversing 4 cm aluminium, 8 cm paraffine wax and 8 cm lead in the lateral direction are:
\[ \phi_{\text{fast at } 90^\circ} = 19 \quad \text{neutron/cm}^2/\text{sec} \]

\[ \phi_i \text{ at } 90^\circ = 9.7 \times 10^2 \quad \text{MeV/cm}^2/\text{sec} \]

and the thermal neutrons are completely attenuated.

Hence, the Surface Dose Rate

\[ = 19 \times 0.03 + 9.7 \times 10^{-2} \times 9.2 \times 10^{-4} \]

\[ = 0.57 \text{ mRem/hr} \]

The conversion factor \(9.2 \times 10^{-4}\) corresponds to lower energy of the gamma radiation scattered at right angle to the radiation beam. The design of the beam trap is shown in Figure 6.

9.6. Radiation Enclosure

To avoid radiation exposure from neutron radiography facility to workers carrying out other research experiments in the vicinity of the facility and from air scattering of radiation from beam trap structure and specimens etc, radiation enclosures are built. Considering radiography of 10 cm thick steel plate, the radiation dose rate due to scattered radiation from the plate after traversing 30 cm baryte concrete wall is

\[
\text{Dose rate} = \phi_f U_f + \phi_{\text{th}} U_{\text{th}} + \phi_i U_i
\]

\[ = 0.03 \times 13.14 + 0.0036 \times 0.09 + 9.2 \times 10^{-4} \times 232 \]

\[ = 0.6 \text{ mRem/hr} \]

where \(U\)'s are the respective dose conversion factors [3].
The dose rate from neutrons and gamma radiation after passing through 50 cm thick baryte concrete wall will be less than 0.02 mRem/hr. Prediction for scattering of radiation from objects being radiographed is not too easy because it depends upon the type and thickness of the material, its orientation to the radiation beam and scattering cross section of the material etc. More, the scattered radiation will suffer a loss in intensity through inverse square law as the enclosure walls are to be built at 75 cm away from the trap. Space provided for setting up the geometric conditions for radiography, away from the surface of the beam trap on each side. However, in the direction of the beam, a 50 cm thick wall of baryte concrete is to be built at 1.5 meter away from the reactor wall. The opening of the trap at a distance of 60 cm from the reactor wall is 35 cm in diameter. The beam shutter can slide along the wall and beam trap position can be adjusted depending upon the geometrical conditions of the radiographic workpiece. The movement of the shutters will be controlled manually. The radiographic enclosure along with radiographic equipment is shown in Figure 7.

9.7. CONCLUSION

Neutron penetration calculation for the estimation of shielding required for beam shutter, beam trap and radiation enclosure are based on two group diffusion approximation. Gamma radiation penetration calculations have been carried out using simple attenuation methods. Graphite is provided as moderator and thermal neutron booster to provide more neutron intensity at the inner end of the divergent beam collimator. Collimator will be made of aluminium.
with an inside lining of boron carbide inside. The inlet aperture of the collimator is 1.62 cm$^2$ and the collimating ratio is 1:127 which would be suitable for number of applications of neutron radiography. The neutron to gamma ratio is $2.95 \times 10^7$ cm$^{-2}$.mRem$^{-1}$ and is suitable for direct neutron radiographic method. Both the beam shutter and trap are 39.6 cm thick each and provide enough attenuation to reduce the radiation exposure to permissible level. The design level is 1.1 Rem per year for 200 exposures each for six minute duration. This can be further reduced by minimizing exposure setting time. The practical experience gained through the fabrication and installation of the neutron radiography facility would provide the basis for designing a better collimating system and would help establish underwater radiography facility for the inspection of highly radioactive materials and components etc.
REFERENCES


### Table I: Two Group Diffusion Parameters of Shielding Materials [3,22]

<table>
<thead>
<tr>
<th>Material</th>
<th>Fast Diffusion Length (cm)</th>
<th>Thermal Diffusion Length (cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>H₂O</td>
<td>5.75</td>
<td>2.90</td>
</tr>
<tr>
<td>D₂O</td>
<td>11.00</td>
<td>1.71</td>
</tr>
<tr>
<td>Be</td>
<td>9.90</td>
<td>24.00</td>
</tr>
<tr>
<td>C (Graphite)</td>
<td>17.30</td>
<td>50.00</td>
</tr>
</tbody>
</table>

### Table II: Two Group Properties of Shielding Materials [2,14,23]

<table>
<thead>
<tr>
<th>Material</th>
<th>Density (gm/cm³)</th>
<th>( \Sigma_{rem} (cm⁻¹) )</th>
<th>( D_f (cm) )</th>
<th>( L_{th} (cm) )</th>
<th>( D_{th} (cm) )</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bismuth</td>
<td>9.85</td>
<td>0.117</td>
<td>2.17</td>
<td>32</td>
<td>1.126</td>
</tr>
<tr>
<td>Boron Carbide (B₄C)</td>
<td>2.52</td>
<td>0.129</td>
<td>0.88</td>
<td>0.084</td>
<td>0.59</td>
</tr>
<tr>
<td>Lead</td>
<td>11.35</td>
<td>0.132</td>
<td>1.69</td>
<td>12.82</td>
<td>0.918</td>
</tr>
<tr>
<td>Paraffine Wax (C_{25}H_{52})</td>
<td>1.14</td>
<td>0.099</td>
<td>0.83</td>
<td>3.95</td>
<td>0.105</td>
</tr>
<tr>
<td>+ Boric Acid</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Iron</td>
<td>7.85</td>
<td>0.1576</td>
<td>0.37</td>
<td>1.27</td>
<td>0.345</td>
</tr>
<tr>
<td>Graphite</td>
<td>1.6</td>
<td>0.0785</td>
<td>1.016</td>
<td>50.00</td>
<td>0.917</td>
</tr>
<tr>
<td>Aluminium</td>
<td>2.7</td>
<td>0.0788</td>
<td>2.27</td>
<td>20.00</td>
<td>4.06</td>
</tr>
</tbody>
</table>
Table III: Neutron Cross Sections for some Shielding Materials [14]

<table>
<thead>
<tr>
<th>Material</th>
<th>Fast Group</th>
<th>Thermal Group</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$\Sigma_c$</td>
<td>$\Sigma_{tr}$</td>
</tr>
<tr>
<td>Bismuth</td>
<td>$5.64 \times 10^{-2}$</td>
<td>0.167</td>
</tr>
<tr>
<td>Iron</td>
<td>$2.51 \times 10^{-4}$</td>
<td>0.168</td>
</tr>
<tr>
<td>$B_4C$</td>
<td>-</td>
<td>0.527</td>
</tr>
<tr>
<td>Graphite</td>
<td>-</td>
<td>0.340</td>
</tr>
</tbody>
</table>

Table IV: Mass Attenuation Co-efficient of Different Shielding Materials, cm /gm [3,9,12,14]

<table>
<thead>
<tr>
<th>Energy, Mev</th>
<th>$H_3BO_3$ $\rho=1.52$ gm/cm$^3$</th>
<th>Paraffine Wax $\rho=0.9$ gm/cm$^3$</th>
<th>Borated Wax $\rho=1.19$ gm/cm$^3$</th>
<th>$B_2O_3$ $\rho=1.85$ gm/cm$^3$</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>0.0562</td>
<td>0.0727</td>
<td>0.0702</td>
<td>0.0468</td>
</tr>
<tr>
<td>2.</td>
<td>0.0393</td>
<td>0.0506</td>
<td>0.049</td>
<td>0.0314</td>
</tr>
<tr>
<td>3.</td>
<td>0.0316</td>
<td>0.0405</td>
<td>0.0392</td>
<td>0.0253</td>
</tr>
<tr>
<td>4.</td>
<td>0.0272</td>
<td>0.03446</td>
<td>0.0334</td>
<td>0.0218</td>
</tr>
<tr>
<td>5.</td>
<td>0.0242</td>
<td>0.0304</td>
<td>0.02948</td>
<td>0.0195</td>
</tr>
<tr>
<td>6.</td>
<td>0.0222</td>
<td>0.0275</td>
<td>0.0267</td>
<td>0.01793</td>
</tr>
<tr>
<td>8.</td>
<td>0.0195</td>
<td>0.0236</td>
<td>0.023</td>
<td>0.0158</td>
</tr>
<tr>
<td>10.</td>
<td>0.0178</td>
<td>0.0209</td>
<td>0.0204</td>
<td>0.0145</td>
</tr>
</tbody>
</table>
DIVERGENT BEAM COLLIMATOR

WITH \( L/d = 127 \)
AND \( N/ = 2.93 \times 10^7 \text{ cm}^{-2} \cdot \text{ mR}^{-1} \)

**FIGURE 1 NEUTRON BEAM COLLIMATOR**
FIG. 3 NEUTRON FLUX ATTENUATION THROUGH GRAPHITE BLOCK
FIG. 4. NEUTRON FLUX ATTENUATION THROUGH BISMUTH BLOCK
FIG. 5 GAMMA DOSE RATE ATTENUATION THROUGH GRAPHITE AND BISMUTH BLOCKS
FIGURE 6 BEAM TRAP
SECTION 1

FIGURE: 7 - SKETCH OF NEUTRON R
SECTION 2

SKETCH OF NEUTRON RADIOGRAPHY FACILITY

<table>
<thead>
<tr>
<th>ITEM</th>
<th>DESCRIPTION</th>
<th>MATERIAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>ENCLOSURE WALLS</td>
<td>CONCRETE</td>
</tr>
<tr>
<td>2</td>
<td>TRAP SUPPORT STRUCTURE</td>
<td>CONCRETE</td>
</tr>
<tr>
<td>3</td>
<td>NEUTRON BEAM TRAP</td>
<td>CONCRETE</td>
</tr>
<tr>
<td>4</td>
<td>BEAM SHUTTER</td>
<td>CONCRETE</td>
</tr>
<tr>
<td>5</td>
<td>BEAM TUBE NO 6</td>
<td>ALUMINUM</td>
</tr>
<tr>
<td>6</td>
<td>SOURCE BLOCK</td>
<td>CONCRETE</td>
</tr>
<tr>
<td>7</td>
<td>BISMUTH FILTER</td>
<td>BISMUTH</td>
</tr>
<tr>
<td>8</td>
<td>BEAM COLLIMATOR</td>
<td>ALUMINUM</td>
</tr>
<tr>
<td>9</td>
<td>NEUTRON STREAMING STOP</td>
<td>CONCRETE</td>
</tr>
<tr>
<td>10</td>
<td>TROLLEY</td>
<td>ALUMINUM</td>
</tr>
<tr>
<td>11</td>
<td>SHUTTER WHEELS</td>
<td>STEEL</td>
</tr>
<tr>
<td>12</td>
<td>TROLLEY WHEELS</td>
<td>STEEL</td>
</tr>
<tr>
<td>13</td>
<td>RAYON WHEEL</td>
<td>STEEL</td>
</tr>
<tr>
<td>14</td>
<td>GUIDE CHANNEL</td>
<td>STEEL</td>
</tr>
</tbody>
</table>

NON DESTRUCTIVE TESTING GROUP
RADIATION AND ISOTOPE APPLICATION DIVISION PISTECH
NLCORE ISLAMABAD

NEUTRON RADIOGRAPHY FACILITY AROUND
PAKISTAN RESEARCH REACTOR

RIAD - 2 (NDT)