A Theoretical Evaluation of the Effectiveness of the Biological Shields for the 0.5MCi Irradiation Facility

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ABSTRACT

The primary shielding and maze designs for the 0.5 MCi Co-60 Gamma Irradiation Facility (GIF) of NNRI at Kwabenya have been theoretically investigated for their effectiveness.

The 1-D transport code ANISNPC was used to investigate the effectiveness of the primary shields, which include the water pool and the concrete walls, by determining the transmission through them. The intensity of photons scattered along the maze plus that transmitted through the intervening walls were used to evaluate the shielding effectiveness of the goods and personnel mazes.

The results show that, the facility could accommodate a 1 MCi Co-60 source if this is consistent with other safety requirements of the facility.

Keywords primary shield, maze design, transmission, dose rate

INTRODUCTION

Ghana is about to commission the first multi-purpose pilot scale irradiation facility at Kwabenya with the assistance of the International Atomic Energy Agency (IAEA). The supplier of the facility is the Radiation Technology Department of the Institute of Isotopes of the Hungarian Academy of Sciences, Budapest, Hungary. The design was jointly undertaken by a Ghanaian architect and the Supplier with an expected loading capacity of 18.5 pBq (500 KCi) Co-60 cage type source. The shields are expected to attenuate the radiation to an average maximum of 2.2 μGy/h at the occupied areas as recommended by the International Commission on Radiological Protection (ICRP). Where the occupancy factor is less than unity, a maximum dose rate of 17.4 μGy/h may be allowed in small areas adjacent to the shield, provided the average dose to individuals of 100 μSv/wk or 50 mSv/yr is not exceeded [1].

In the design and construction of irradiation facilities, the biological shields constitute one of the very important considerations to ensure safety of personnel and equipment in the working areas. In particular, the shield must

protect personnel working in the irradiation chamber when source is in the storage position and

take account of the operators and other workers who may stay close to the maze entrances or adjacent walls.

Dry (concrete or other shielding materials) or we (water pool) storage may be used to achieve the first objective. Concrete is normally used for the walls and maze designs to bring about the attenuation of photons by multiple scattering. Lead-lined or steel doors may be used at the maze entrances to further attenuate photons whilst serving also as a barrier against inadvertent entry into the irradiation chamber when source is unshielded. In each case theoretical calculations are used initially to obtain the optimum thickness or layer of shield required. In the case of the maze the number of breaks or bends and their respective lengths determine the doserate at their entrances. This may also be determined through calculation.

In this report an assessment of the concrete and water pool as shields have been investigated using the 1-D transport code ANISNPC to obtain the transmission and hence the *doserate* at occupied areas including the roof of the chamber where the *hoist and batteries* are to be located. The doserate at the maze entrances are the combined contributions of direct transmission through the concrete walls and that due to multiple scatter of photons along the maze. The ANISNPC code was again used for the direct transmission components at the maze entrances while the scattered components were obtained by analytical calculation.

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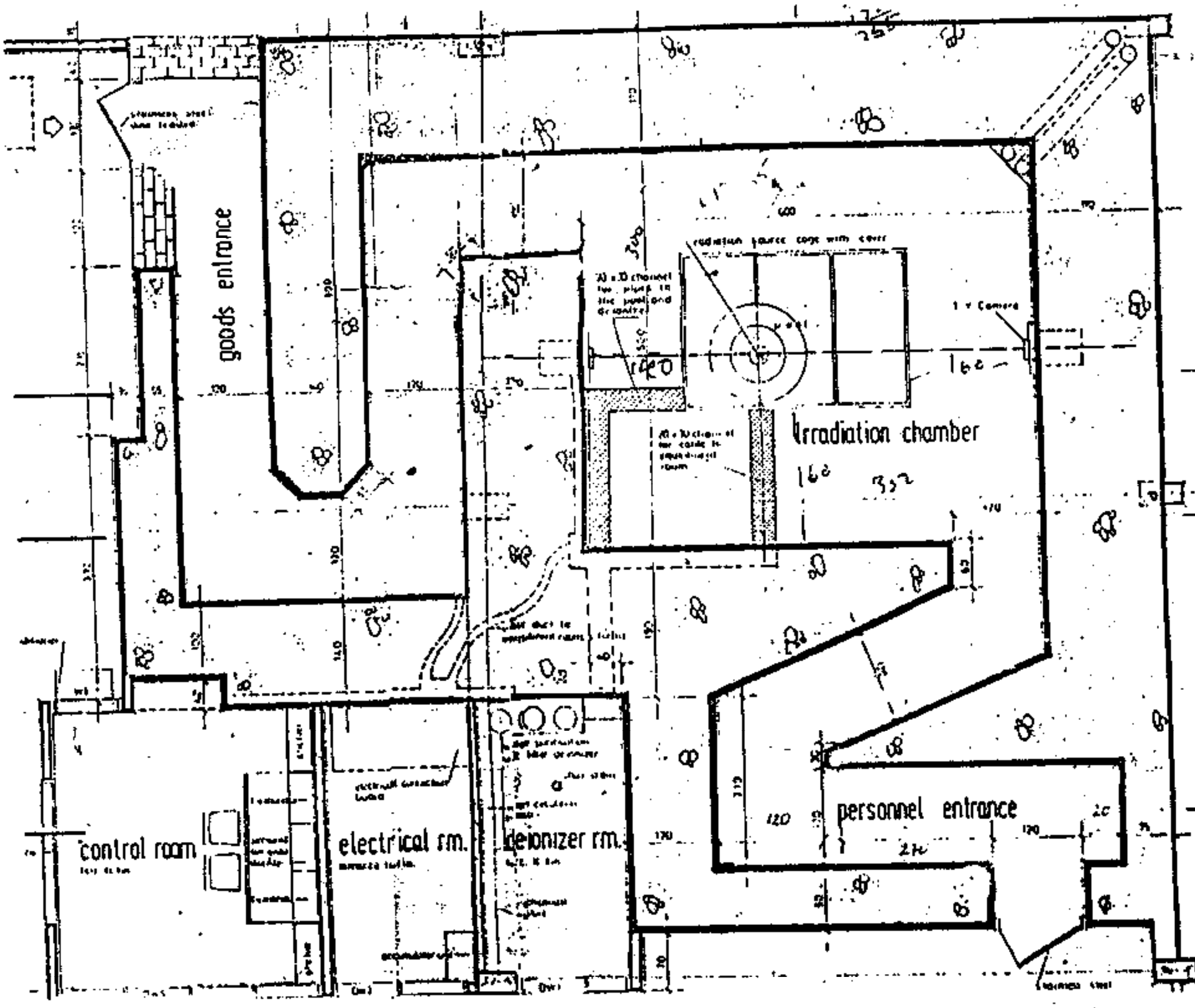

Fig. 1: Plan of Gamma Irradiation Facility

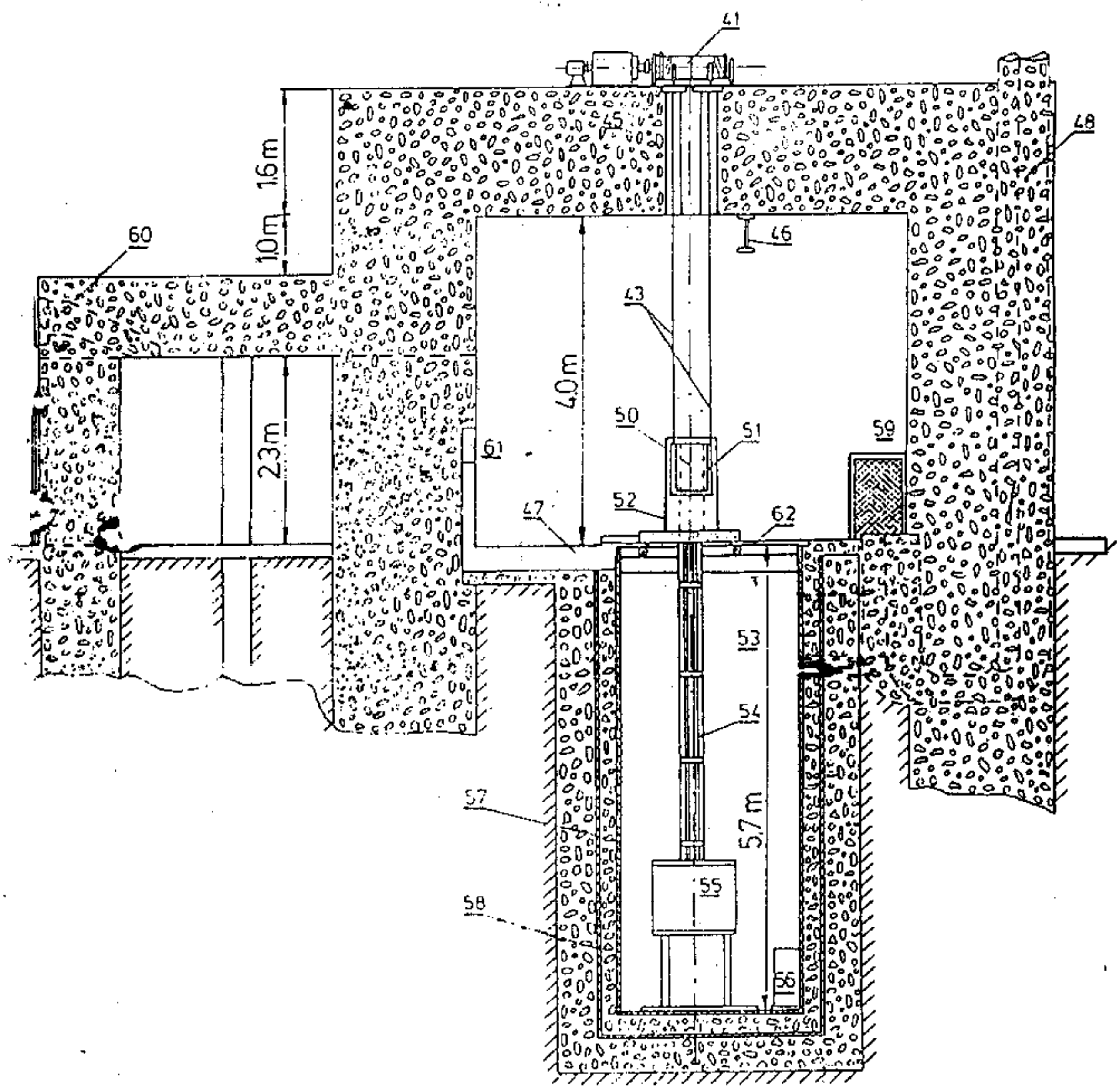

Fig. 2 Vertical Section of Facility showing the Chamber and the water Pool

DESCRIPTION OF FACILITY

The design layout of the irradiation chamber is shown in Figure 1 where an average wall thickness of 1.7m concrete. The wall between de-ionizer room and the irradiation chamber is 1.9m thick. The goods and personnel mazes are 1.2m wide. The height of the goods maze is 2.0m whilst the personnel maze is 1.5m high.

In Figure 2, the cross-section of the chamber and the water pool are shown. The water pool is 5.7m deep, 2m wide and 3m long, however the rest position of the sources would allow only 4m of the water for the attenuation of the photons.

There will be 20 source pencils arranged in the form of a cage of diameter *280mm*. Each pencil will be 208mm long with an outer diameter of 40mm and wall thickness of 1.5mm. When at the irradiation position, the horizontal plane of symmetry of the sources will be 80cm from the floor to allow varied product boxes to be treated.

CALCULATIONAL METHODS

Numerical approach based on ANISNPC was used to study the primary shields including *doserate* at the maze entrances due to direct transmission of photons through intervening walls. However the doserate at a maze entrance is also based on scatter of the photons along the maze and this is evaluated by analytical calculations. The theory of the two approaches are given below. (a) Theory of ANSIN for Primary Shields (Concrete and Water pool)

The 1-D Transport Code ANISNPC solves the linearised transport equation sometimes referred to as Boltzmann's Equation. The equation depicts the fact that particles (neutral) move in straight lines at constant speed between collisions with nuclei. The dependent variables ø is the angular particle flux and is a function of position r, direction $\hat{\Omega}$, energy E and time t.

ie $\quad \phi = \phi(r, \hat{\Omega}, E, t) \qquad (1)$
$\quad\quad = v N(r, \hat{\Omega}, E, t)$

and $N(r, \hat{\Omega}, E, t)\, dr\, d\hat{\Omega}\, dE$ is at time t the number of particles in the volume element dr about r in a solid angle $d\hat{\Omega}$ about $\hat{\Omega}$ with energies in the range dE about E and where v is the particle speed.

The scalar particle flux is $\phi(r, E, t) = \int \phi(r, \hat{\Omega}, E, t)\, d\hat{\Omega}$. The Boltzmann's equation can be obtained simply by balancing the various mechanism by which particles can be gained or lost from an element in a phase-space, $(r, \hat{\Omega}, E)$):

$$\begin{bmatrix}\text{Time rate}\\ \text{of particle}\\ \text{density}\\ \text{change}\end{bmatrix} = \underset{(1)}{\begin{bmatrix}\text{change due}\\ \text{to}\\ \text{physical}\\ \text{leakage}\end{bmatrix}} + \underset{(11)}{\begin{bmatrix}\text{change due}\\ \text{to}\\ \text{collisions}\end{bmatrix}} + \underset{(111)}{\begin{bmatrix}\text{sources}\end{bmatrix}}$$

which can be expressed mathematically as,

$$\frac{1}{v}\frac{\partial}{\partial t}\phi(r, \hat{\Omega}, E, t) = \underset{(1)}{-\hat{\Omega}\cdot\nabla\phi(r, \hat{\Omega}, E, t)} + \underset{(111)}{s(r, \hat{\Omega}, E, t)}$$

$$+\Sigma_t(r, E, t)\phi(r, \hat{\Omega}, E, t) + \int dE' \int d\hat{\Omega}'\, \Sigma(r; E' \to E; \hat{\Omega}' \to \hat{\Omega}; t)\, \phi(r, \hat{\Omega}', E', t) \qquad (2)$$

where $\phi(r, \hat{\Omega}, E, t)$ is the angular flux or angular particle track length

$\Sigma_t(r, E, t)$ is the total macroscopic cross-section

$\Sigma(r; E' \to E, \hat{\Omega}, \Omega'; t)$ is the differential transfer cross-section describing the probability that a particle with an initial energy E' and direction $\hat{\Omega}'$ undergoes a collision at r in a time t, resulting in a change of direction and energy; and

$s(r, \hat{\Omega}, E, t)$ is the source term.

Equation 2 can be applied to neutrons or photons, depending on the correct physical interpretation of the interaction between the particles and the host medium (cross-section). For neutrons, besides a scattering reaction, a collision of neutrons with a host medium may generate other neutrons by fission (multiplying medium) and so the transfer kernel may include another term [2].

Using a multigroup treatment of the energy dependence with interest in the group angular flux

$$\phi_g(r, \hat{\Omega}) = \int_g dE\, \phi(r, \hat{\Omega}, E) \qquad (3)$$

the integration being taken over the entire range covered by group g. The multigroup equation comes from a simple integration of the whole equation over the same energy range to give

$$\frac{\partial}{v_g}\phi_g + \Sigma_g\phi_g + \hat{\Omega}\nabla\phi_g = \sum_{g'} \int d\hat{\Omega}'\, \Sigma_{gg'}(r, \hat{\Omega}' \to \hat{\Omega})\, \phi_{g'}(r, \hat{\Omega}') + s_g(r, \hat{\Omega}). \qquad (4)$$

where v_g is the average velocity for the group g

Σ_g is the group total cross section, sg is the group fixed source

$\Sigma_{gg'}(\hat{\Omega}' \to \hat{\Omega})$ is the group to group differential cross section and these quantities are defined by the requirements that the group equation is just the integral of the energy dependent equation is

$$\frac{\partial}{v_g}\phi_g(r, \hat{\Omega}) = \int_g dE\, \frac{\partial}{v}\phi(r, \hat{\Omega}, E)$$

$$\Sigma_g\phi_g(r, \hat{\Omega}) = \int_g dE\, \Sigma(r, E)\, \phi(r, \hat{\Omega}, E)$$

$$s_g(r, \hat{\Omega}) = \int_g dE\, s(r, \hat{\Omega}, E) \qquad (5)$$

Now the group velocity and the group cross-sections depend formally upon the angular flux ϕ (r, $\hat{\Omega}$, E) and the multigroup method depends firmly upon the concept of that, prior to the calculation, group velocities and cross sections have to be calculated for each material.

The differential cross section for its directional variation depends only on the angle between the incoming and outgoing directions in the scattering process and can be approximated by a truncated Legendre polynomial series expansion.

$$\Sigma_{gg'}(\hat{\Omega}' \rightarrow \hat{\Omega}) = {}_{4\pi} \sum_{l=0}^{L} (2l+1) \Sigma^{l}_{gg'} P_{l}(\hat{\Omega} . \hat{\Omega}') \quad (6)$$

It is the component Σ^{l}_{gg}, which form part of the multigroup cross section set as material data for the transport calculation. For many calculations only one term in the series is used or is needed for adequate accuracy and this is the isotropic scattering approximation in multigroup theory.

Now restricting problems to time independent ones, the transport equation [4] becomes

$$\Sigma_g \phi_g (r, \hat{\Omega}) + \hat{\Omega}. \nabla\phi_g = s_g(r, \hat{\Omega}) + {}_{4\pi} \Sigma(2l+1) \Sigma \Sigma^{l}_{gg'} \int d\Omega\, P_l(\Omega . \Omega')\phi_{g'}(r, \Omega') \quad (7)$$

in which the leakage term ($\hat{\Omega}\nabla\phi_g$) for spherical geometry consideration as used in this study is given by

$$\hat{\Omega}. \bar{\nabla}\phi_g = -\frac{\mu}{r^2}\frac{\partial}{\partial r}\left[r^2\phi_g(r, \mu)\right] + \frac{1}{r}\frac{\partial}{\partial\mu}\left[(1-\mu^2)\phi_g(r, \mu)\right] \quad (8)$$

In the discrete ordinate method applied in ANISN, the angular variables is broken into a number of discrete directions (angular quadratures, with associated weights, to approximate integrals); the spatial variables is treated by a finite difference scheme, and the energy variable, by multigroup method ie

$$r, \hat{\Omega}, E) \xrightarrow[\text{Ordinate}]{\text{Discrete}} (r_j; \hat{\Omega}_m; E_g) \quad (9)$$

The transport equation is therefore discretised into a discrete phase-space cell, resulting in a system of algebraic equations at the discrete mesh points r_j; $\hat{\Omega}_m$ for each energy group ie

$$A\phi = B\phi + S \quad (10)$$

where A is the matrix representing the streaming collision operator (Ω, + Σ); B is a matrix representing the group in scattering term; S is a matrix representing the source term (external plus group to group scattering sources, including fission where applicable) and ϕ is the angular flux vector at the discrete group mesh points. The numerical solution of the algebraic equation can be obtained by an interaction scheme, inner interaction

$$\phi^{(n)} = A^{-1} B \phi^{(n-1)} + A^{-1}S \quad (11)$$

using some acceleration method (coarse mesh rebalancing; linear diffusion synthetic etc).

The selection of the discrete ordinates quadrature set ; $\hat{\Omega}_m$ directions and ω_m weights to discretise the transport equation, by approximating integral angular terms;

$$\int_{\hat{\Omega}} \phi(r) d\hat{\Omega} = \sum_{m=1} \phi(r, \hat{\Omega}_m) \omega_m. \quad (12)$$

is basic to obtain an accurate numerical solution.

For one dimensional geometries, the following criteria must be satisfied by the quadrature set ; μ_m; ω_m

(i) invariance projection, ie μ_m symmetry about μ=o or

$\mu_i = -\mu_{m-i}$, $w_i = \omega_{m+1-i}$ (symmetry upon reflection)

(ii) positively of the solution ie $\omega_m > 0$, $\sum_{m=1}^{M} \omega_m = 1$

From (i) and (ii), the condition to be satisfied is

$$\sum_{m=1}^{M} \mu_m \omega_m = 0 \quad (13)$$

ANISN code employs the quadrature scheme which satisfies these conditions and also satisfies the diffusion theory" condition that

$$\sum_{m=1}^{M} \omega_m \mu_m = 3 \quad (14)$$

The full discretised multigroup-discrete ordinate 1-D equation with isotopic scattering (L-O) for geometries of slab, cylinder and sphere can be written in the general form as

$$\mu_m\left[A_{i+1/2}\phi^g_{m,i+1/2} - A_{i-1/2}\phi^g_{m,i-1/2}\right] + \left[a_{m+1/2}\,\phi^g_{m+1/2,i} - a_{m-1/2}\,\phi^g_{m-1/2,i}\right]\omega_m + \Sigma^g_{t,i}\,\phi^g_{m,i}V_i = s^g_{m,i}V_i$$

$$i=1,2,\ldots\ldots\ldots I$$
$$m=1,2,\ldots\ldots\ldots M \qquad (15)$$

where the area A and the volume V for slab, cylinder, and spherical geometries are given in Table 1.

The curvature coefficients, a, is obtained from the recurrence relation

$$a_{m+1/2} = a_{m-1/2} - \mu_m\omega_m\left\{A_{i+1} - A_i\right\} \qquad (16)$$

The centre fluxes are related to the edge fluxes such that

$$\phi^g_{m,i} = \alpha\phi^g_{m,i+1/2} + (1-\alpha)\phi^g_{m,i-1/2} \qquad (17)$$

$$\phi^g_{m,i} = \alpha\phi^g_{m+1/2,i} + (1-\alpha)\phi^g_{m-1/2,i}$$

$$i=1,2,\ldots\ldots I$$
$$m=1,2,\ldots\ldots M \qquad (18)$$

The diamond scheme ($\alpha=1/2$) is used in ANISN.

ANISN code is a FORTRAN program, with the following basic operations.

- problem set up operation
- iterative calculations
- concluding calculations

Geometry	$A_{i\pm1/2}$	V_i
1-D Slab	1	Δx_i
1-D Sphere	$4\pi r^2_{i\pm1/2}$	$4\pi r^2_i\,\Delta r_i$
1-D Cylinder	$2\pi r_{i\pm1/2}$	$2\pi r_i\,\Delta r_i$

Table 1 : Area A and Volume V for Different Geometries.

The input for ANISN is divided into the following data sets:

i Problem Title
ii Problem Parameters (integer and floating point)
iii Cross section data
iv Flux or fission guess data
vi Remainder of data

The photon source in this report is assumed to be located at the centre of a spherical (ie point source) material and the necessary correction applied in the end. Using a fixed source in the fish mesh, s_6 angular quadrature, spherical geometry and P_3 scattering order the calculations were carried out.

Microscopic cross section for various elements were taken from FLUNGP. LIB master library with energy groups listed in Table 2 [2] for the water. In the case of concrete IRAN3. LIB with 25 energy groups (Table 3) was used [4]. In either case a utility program called LMOD was used to generate the library files for the different materials. The number densities of water and concrete used in the code are given in Tables 4 and 5 respectively [3].

The output from an ANISN-PC problem can be obtained on the screen, on printed paper or in a file. The information displayed among others include:

i. title problem with a brief edit of problem definition with a short explanation
ii a list of read arrays (input information) with the number of entries
iii a list of the zone numbers by interval; radii, areas and volumes, fission density, and density factor if any
iv a list of the zone; fission spectrum; velocities; right albedos (RT); left albedo (LFT); diffusion calculation marker (DIFF MARKER); material number (MAT'L/ZONE); order of scattering by zone (L of P(L) and the radius modifiers (RADIUS MOD)
v a list of cross section mixing table including mixture component, and number density and a list of angular quadrature constants including direction cosine, weight, the reflected direction indices, and the product of cosines and weights (WTxCOS). If ISCT > 1, The Legendre coefficients used in the anisotropic scattering source is displayed.
vi Summary tables for each zone and by group, including sum for all groups in the last line, and also a summary of all system are displayed for the quantities: Fixed source(s); Fission sources; in scatter; self scatter; out scatter; absorption; net leakage; balance; right boundary flux; right boundary current J^+; left boundary J^- ; right leakage; left leakage; fission rate; total flux; density. All reaction rates have units of reactions per second; the total flux is the sum over the appropriate intervals of the product of the scalar flux and the interval volume and density is the total flux divided by the group velocity.
vii The transmission which is the important parameter of interest now is given by the right leakage current J^+ multiplied by the surface area.

The exposure rate from 1 Becquerel source of Co-60 is 8.42×10^{-17} Gy/s at 1 metre and varies as the inverse square of the distance. The correction factor for transforming a point source to the cage type is 0.7 and hence

Table 2: Flungp. lib 21 energy group structure

Group	Energy Range(eV)	
1	1.4 E 07	1.2 E 07
2	1.2 E 07	1.0 E 07
3	1.0 E 07	8.0 E 06
4	8.0 E 07	7.5 E 06
5	7.5 E 06	7.0 E 06
6	7.0 E 06	6.5 E 06
7	6.5 E 06	6.0 E 06
8	6.0 E 06	5.5 E 06
9	5.5 E 08	5.0 E 06
10	5.0 E 06	4.5 E 06
11	4.5 E 06	4.0 E 06
12	4.0 E 06	3.5 E 06
13	3.5 E 06	3.0 E 06
14	3.0 E 06	2.5 E 06
15	2.0 E 06	2.0 E 06
16	2.0 E 06	1.5 E 06
17	1.5 E 06	1.0 E 06
18	1.0 E 06	4.0 E 05
19	4.0 E 05	2.0 E 05
20	2.0 E 05	1.0 E 05
21	1.0 E 05	1.0 E 04

Table 3: Energy group structure of Iran 3. lib

NEUTRON		GAMMA	
Group	Upper Energy (eV)	Group	Upper Energy (eV)
1	1. 7333 E +7	8	1.4 E +7
2	5.2205 E +6	9	9.0 E +6
3	1.0026 E +6	10	6.0 E +6
4	4.9787 E +5	11	4.0 E +6
5	9.8037 E +4	12	3.0 E +6
6	9.1188 E +3	13	2.6 E +6
7	5.3156 E -1	14	2.0 E +6
		15	1.5 E +6
		16	1.0 E +6
		17	7.0 E +5
		18	4.5 E +5
		19	3.0 E +5
		20	1.5 E +5
		21	1.0 E +5
		22	7.5 E +4
		23	4.5 E +4
		24	3.0 E +4
		25	2.0 E +4

Table 4: Number Densities of Elements of Water

Element	Number Density x 10^{24} (atoms/cc)
Hydrogen	6.68E-02
Oxygen	3.34E-02

Table 5: Number Densities of Elements of Ordinary Concrete

Element	Number Density x 10^{24} (atoms/cc)
Hydrogen	7.77E-03
Oxygen	4.39E-02
Sodium	1.05E-03
Magnesium	1.49E-04
Aluminium	2.45E-03
Silicon	1.58E-02
Potasium	6.93E-04
Calcium	2.92E-03
Iron	3.13E-04

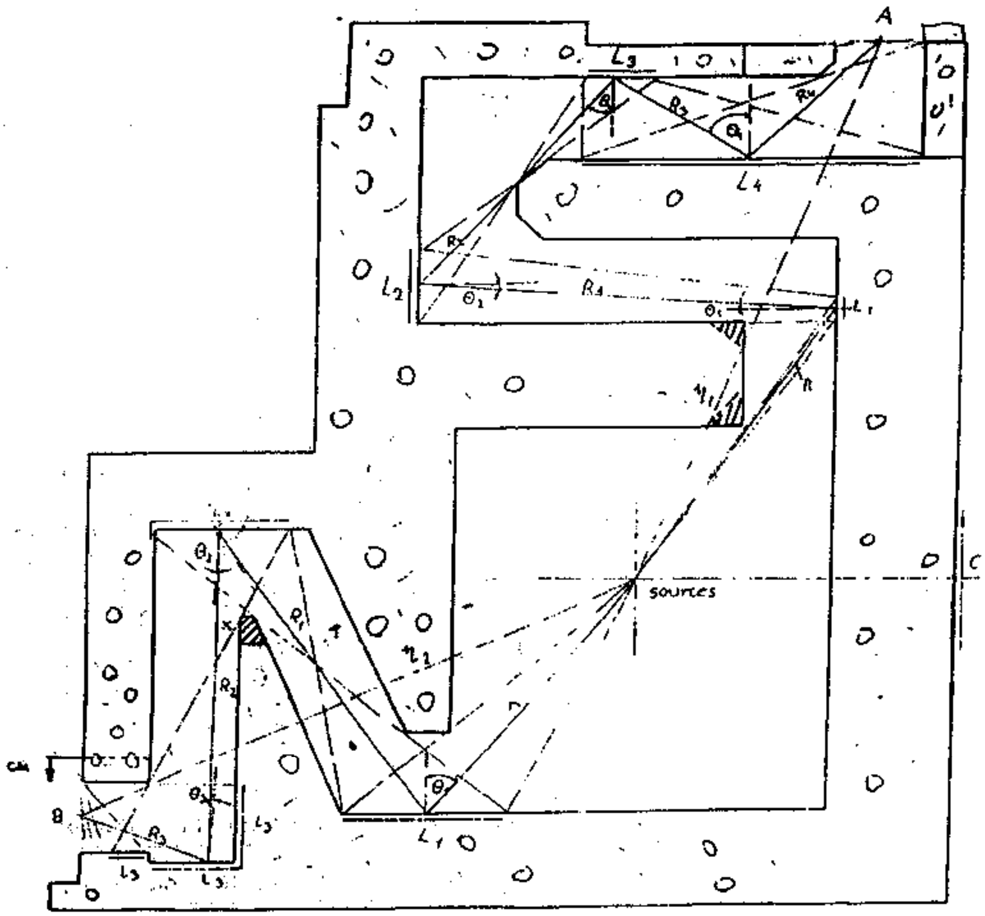

Fig. 3: Diagram showing Scattered and Transmitted Photons to the Maze Entrances

Table 6: Dose Rate at Selected Points due to Trans,mission of Photons through Primary Shields

Material	Thickness m	Distance m	Transmission	Dose Rate /µGy/h 0.5MCi	1.0MC1
Water	4.0	4.0	5.962E-11	0.02	0.03
Concrete	1.7	4.0	5.707E-09	1.39	2.77
Concete	1.9	4.6	1.407E-09	0.27	0.53
Concrete (roof)	1.6	5.6	1 721E-08	2.15	4.31

Table 7: Dose Rate at Maze Entrance due to Transmitted and Scattered Photons

Mazes	Total Length of Maze/m	Total Wall Thickness/m	Dose >Rate (µGy/h) Scattered	Transmitte	Total
Personnel	8.25	2.1	48.8	5.28E-3	48.8
Goods	9.1	2.4	0.25	1.51E-4	0.26

the dose rate outside a shielding wall is given by

$$D = \frac{3600 \times 8.42 \times 10^{-17} \times 0.7 \times T \times Activity(Bq)}{d^2} \; Gy/h \qquad (19)$$

where d is the distance from the source to the outside surface of the shield

T is the transmission (obtained from ANISN)

Accurate calculations of the exposure rate and energy spectrum at points along concrete mazes are difficult and as such detailed calculations of exposure rate attenuation in concrete mazes have mostly been confined to two-legged concrete ducts. The basis of radiation attenuation in the maze is the fact that, a large portion of the gamma rays falling on the surface of a wall is absorbed, with only a small part scattered back into the maze. The gamma rays scattered from a wall surface is calculated from the equation;

$$D_s = D_o \frac{s_i \ \alpha_i \cos \theta_i}{2\pi \ R_i^2} \qquad (20)$$

where

D_o is the dose rate at the first wall surface

s_i is the area of the scattering surface (m^2)

s=L x H; H = height of surface;

L = length of surface

θ_i is the angle between the vertical axis of the scattering surface and the incident gamma ray (o)

α_i is the differential dose albedo, ie. the ratio of the doses incident on the surface to that scattered from it.

R_i is the distance from the source to the first wall surface or the distance between two scattering surfaces or distance from the last scattering surface to the plane of the maze door (m^2).

Cos θ_i is the "surface density" of the gamma rays incident at θ angle

The differential dose albedo strongly depends on the energy of the gamma rays, the incident angles and the material of the wall. The scattering surface of interest and the distances between them are selected in a subjective way, mainly based on experience.

In case of Co-60 sources, the energy of the gamma rays on the first scattering surface is E_1 = 1.25 MeV, on the second E_2 = 0.5 MeV, on the third and further on $E_{3,4...}$ = 0.25 MeV [1]

In the case of mazes, the *doserate* at an entrance is the combined contribution of direct transmission through the intervening concrete walls and that through scatter along the maze (Figure 3). The ANISN code as usual was used to obtained the contribution by direct transmission whilst the analytical method was used to calculate the scatter contribution along the maze.

Finally calculations of the *doserate* at selected points were obtained for the facility if loaded with (i) 0.5 MCi and (ii) 1.0 MCi Co-60 sources for the final assessment.

RESULTS AND DISCUSSION

The results of doserate shown in table (6) and (7) reveal that the biological shields would provide adequate protection for personnel and equipment.

The doserate above the water pool in the irradiation chamber is 0.02 μGy/h if the capacity of the facility is increased to 1 MCi. At the back of the chamber the dose rate is expected to 2.77μGy/h with the increased capacity but this is acceptable because the occupancy factor of this location is far less than 1. The same is also true the 4.31μGy/h expected on top of the irradiation chamber.

PRIMARY SHIELDS

The dose rate at the maze entrances are observed to be essentially due to scattered photons along the length of the mazes. As shown in table 5, the dose rate at the personnel entrance due to scattering is expected to be 48.8μGy/h and 5.28 nGy/h through transmission.

The results indicate that the provision of a lead-lined door at the goods entrance would serve essentially as a physical barriers since the dose rate is already relatively low (ie 0.26 μGy/h) and would not change even f the facility is loaded with 1 MCi. This provision would allow the use of the conveyer system since the removal of the door would not significantly increase the risk of personnel who would be loading the conveyer. The expected dose rate at the entrance of the personnel maze is rather high (ie 48.8 μGy/h) so that the 9mm lead-lined door provided is intended to serve both as a physical barrier and to improve on the biological shielding of the photons.

CONCLUSIONS

The primary shields and maze designs have been proved to be able to provide adequate protection for personnel and equipment's of the GIF even when the facility is loaded with 1 MCi Co-60 sources instead of the intended 0.5 MCi capacity.

The electrical and mechanical structural designs are however to be evaluated to ascertain the efficacy of increasing the loading capacity of the facility to 1 MCi before embarking on it.

It is recommended that actual dose rate data be acquired during commissioning to augment the deduction so far derived from this theoretical assessment.

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