SHIELD DESIGN OF CONCRETE WALL BETWEEN DECAY TANK ROOM AND PRIMARY PUMP ROOM IN TRIGA FACILITY

M J H Khan, M Rahman, F U Ahmed[1], S I Bhuiyan[1], A Haque and A Zulquarnain

Atomic Energy Research Establishment, [1]Bangladesh Atomic Energy Commission, Bangladesh

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The objective of this study is to recommend the radiation protection design parameters from the shielding point of view for concrete wall between the decay tank room and the primary pump room in TRIGA Mark-II Research Reactor Facility. The shield design for this concrete wall has been performed with the help of Point-kernel Shielding Code Micro-Shield 5.05 and this design was also validated based on the measured dose rate values with Radiation Survey Meter (G-M Counter) considering the ICRP-60 (1990) recommendations for occupational dose rate limit (10 μSv/hr). The recommended shield design parameters are: (i) thickness of 114.3 cm Ilmenite-Magnetite Concrete (IMC) or 129.54 cm Ordinary Reinforced Concrete (ORC) for concrete wall A (ii) thickness of 66.04 cm Ilmenite-Magnetite Concrete (IMC) or 78.74 cm Ordinary Reinforced Concrete (ORC) for concrete wall B and (iii) door thickness of 3.175 cm Mild Steel (MS) on the entrance of decay tank room. In shielding efficiency analysis, the use of I-M concrete in the design of this concrete wall shows that it reduced the dose rate by a factor of at least 3.52 times approximately compared to ordinary reinforced concrete.

Keywords : Concrete, Shielding, Dose Rate and Micro-Shield

1. INTRODUCTION

The decay tank assembly consists of the decay tank and an inlet pipe of decay tank. This assembly contains water, which holding-up the radioisotopes ^{16}N & ^{19}O. Due to high activity of these isotopes the decay tank room was situated underground with enough shielding of surrounding walls & roof respectively. But there was no enough shielding wall (concrete) between the decay tank room and the primary pump room to protect the occupational personnel in the primary pump room during full power (3 MW) operation of TRIGA Mark-II Research Reactor. For enhanced radiation safety a concrete shield is proposed for the middle wall and this study will be helpful to construct this wall from the shielding point of view.

In this study, International Commission on Radiological Protection (ICRP)-60 (1990) is considered to estimate the radiation protection design parameters for concrete wall between the decay tank room and the primary pump room within the occupational dose limit [1]. The shield design parameters have been calculated using the Point-kernel Shielding Code Micro-Shield 5.05 [2] and these parameters are also validated based on the measured dose rate values with Radiation Survey Meter (G-M Counter), Berthold UMo LB 123 [3]. The shield design of concrete wall has been developed using ilmenite-magnetite concrete (IMC) of density 2.76 g/cm^3 and ordinary reinforced concrete (ORC) of density 2.35 g/cm^3. In addition, the purpose of this study is to carry out the radiation shield design using such materials developed locally, which would be both readily available & economically profitable for the country and also provides adequate shielding in the existing space around the concrete wall between the decay tank room and the primary pump room.

Corresponding author : M J H KHAN, jahirulkhan@yahoo.com,
Atomic Energy Research Establishment, GPO Box # 3787, Dhaka-1000, Bangladesh

2. CALCULATIONAL TECHNIQUES

Micro-Shield 5.05 [4] is a comprehensive gamma-ray shielding and dose assessment program. It is widely used for designing shields, estimating source strength from radiation measurements, minimizing exposure to people, and teaching shielding principles. This code was based on the point-kernel technique, which representing the transfer of energy by the un-collided flux along a line of sight path was combined with an approximate buildup factor to account for the contribution from the scattered (buildup) photons. With a distributed source, the point-kernel was integrated over the source volume for each source-energy considered. In this case, the gamma-ray dose rate, D(r), at any point due to isotropic emitting S photons of energy E per second per unit volume was calculated by the following formulae [5]

$$D(\vec{r}') = \int_v \frac{KS(\vec{r}')B(\mu|\vec{r}-\vec{r}'|, E)\exp(-\mu|\vec{r}-\vec{r}'|)dv}{4\pi|\vec{r}-\vec{r}'|^2}$$

Where, r is a point at which gamma dose rate is to be calculated, r' is a location of source in volume v, $|r-r'|$ is a distance between source point and point at which gamma intensity is to be calculated, B is a flux-to-dose conversion factor, K is a dose buildup factor, v is a volume of source region and μ is the total attenuation coefficient at energy E.

In this study, the shield design of the concrete wall was carried out by Micro-Shield 5.05 code and this code calculates photon flux, photon energy flux, exposure rate (air) and absorbed dose rate (air) with & without buildup factor respectively. Buildup factor was considered in the calculation of thickness of concrete wall (thick) between the decay tank room and the primary pump room because it was an important contributor to the total dose rate.

In addition, the Radiation Survey Meter (G-M Counter), Berthold UMo LB 123 [3] was used to measure the dose rates at different detector locations around the concrete wall (shown in Fig.1). The calculated dose rates were validated based on the measured values with this Radiation Survey Meter.

3. DESCRIPTION OF SOURCE AND SHIELD MATERIALS

3.1 Description of Source

The source originates from the reactor operation at full power (3 MW) while the cooling water passes through the reactor core. The radioactive products [6] ^{16}N and ^{19}O are produced through the fast neutron reaction $^{16}O(n,p)^{16}N$ & the thermal neutron reaction $^{18}O(n,\gamma)^{19}O$ respectively in this water. The dissolved water with ^{16}N (emitting 6 MeV gamma) and ^{19}O (emitting 1.36 MeV gamma) leaves the core and at last goes to the decay tank through an inlet pipe of the decay tank. In this case, the volumes of the decay tank and an inlet pipe of decay tank respectively act as a radiation source individually. The dimensions [7] of the decay tank and an inlet pipe of decay tank are taken from Reactor Operation and Maintenance Unit (ROMU). The total source activities for the decay tank and an inlet pipe of the decay tank considering the characteristics of the above radio-products [6] and the volumes [7] are 1.5×10^{12} photons/sec and 3.7×10^{12} photons/sec respectively.

3.2 Shield Materials

In this study, three shield materials are used to perform the radiation shield design of the concrete wall between the decay tank room & the primary pump room in TRIGA facility. The elemental compositions of these materials [8] are shown in Table 1. It may mention that Ilmenite-Magnetite Concrete (IMC) consists of 15

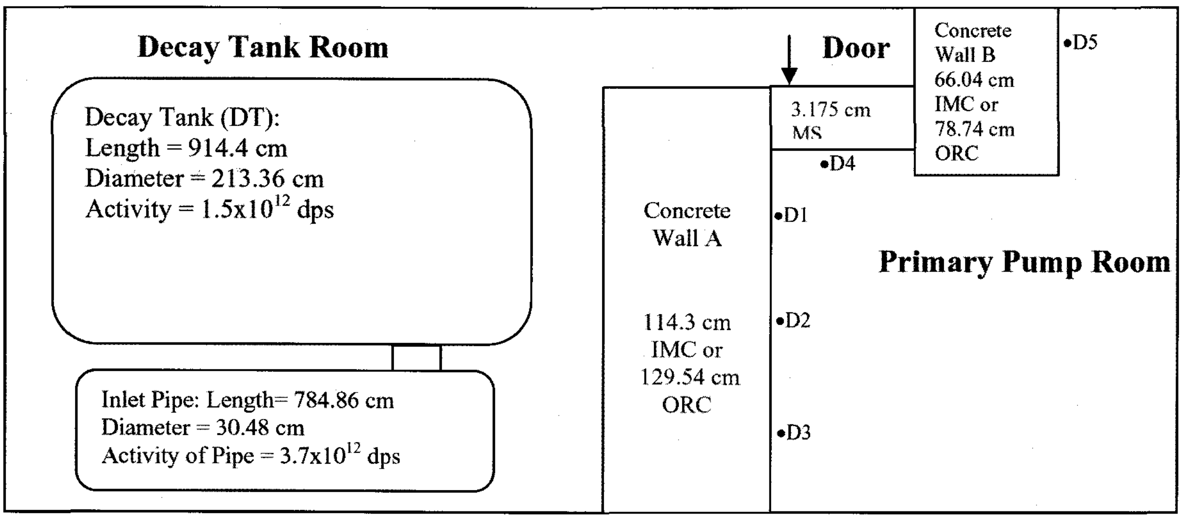

Fig.1. The Layout Plan of the Concrete Wall between the Decay Tank Room and the Primary Pump Room at TRIGA Facility.

Table 1. Elemental Compositions of Used Shield Materials [8]

Element	Ilmenite-Magnetite Concrete (IMC) Density, ρ= 2.76 g/cm^3	Ordinary Reinforced Concrete (ORC) Density, ρ= 2.35 g/cm^3	Iron Density, ρ= 7.86 g/cm^3
	Partial Density (g/cm^3)	Partial Density (g/cm^3)	Density (g/cm^3)
H	0.0157	0.013098	
C	0.0022		
O	1.0523	1.1645000	
Na		0.0399670	
Mg	0.1014	0.0060149	
Al	0.0497	0.1094900	
Si	0.1349	0.7366000	
P	0.0002		
S	0.0016	0.0028000	
K		0.0449940	
Ca	0.2469	0.1939300	
Ti	0.3563		
V	0.0021		
Cr	0.0010		
Mn	0.0084		
Fe	0.7863	0.0290250	7.86
Ni	0.0012		

Table 2. Comparison of Calculated Dose Rates with Measured Values at Different Detector Locations around the Middle Wall

Detector Location	Micro-Shield Code		Radiation Survey Meter [3]
	Calculated Dose Rates (μSv/hr)		Measured Dose Rates (μSv/hr)
	Direct Beam	With Buildup Factor	With Buildup Factor
D1	1.2456	6.7322	7.2325
D2	1.3456	7.4140	8.0053
D3	1.0045	5.5064	6.5654
D4	--------	--------	8.0157
D5	1.2305	7.2414	8.1065

elements and Ordinary Reinforced Concrete (ORC) consists of 10 elements. Due to elemental composition change IMC is more effective than ORC. IMC is a locally developed concrete, which has been effectively used in various radiation shield construction including the biological shield of the 3 MW TRIGA MARK-II Research Reactor [9].

4. LAYOUT PLAN OF THE CONCRETE WALL

The proposed layout plan of the concrete wall is presented in Fig.1 with self-explanatory engineering in details.

The physical positions of the source and the detector locations (D1, D2, D3, D4 and D5) have been shown to define the source-detector geometry and distances of shield materials. It can be seen that the detectors are positioned to face the most direct beam across the shield.

5. RESULTS AND DISCUSSIONS

The obtained results of the Micro-Shield Code and the Radiation Survey Meter (G-M Counter) from the shielding point of view are summarized in Table 2. The dose rates by both the techniques at different locations around the concrete wall are found below ICRP-60 (1990) recommended level for occupational dose limit (10 μSv/hr).

Table 3. Shows the Shielding Efficiency of IMC vs. ORC around the Middle Wall

Location	Micro-Shield Code		
	Shield Material	Total Dose Rate (mSv/hr)	*SE = ORC/IMC
D1	ORC	23.6974	3.52
	IMC	6.73220	

* SE=Shielding Efficiency

From Table 2 it is found that the calculated dose rate values for each detector position are lower than those values of radiation survey meter. Besides, at D4 position (outside of entry door of decay tank room) the dose rate is only measured with radiation survey meter due to limitation of Micro-Shield Code and this dose rate is also lower than ICRP recommended level for occupational dose limit. Since this code does not count back-scattering radiation.

In addition, the shielding efficiency of IMC in lieu of ORC was investigated with the help of the Point-kernel Shielding Code Micro-Shield 5.05. The calculated results on shielding effectiveness of IMC vs. ORC are shown in Table 3.

From Table 3 it may conclude that the use of I-M concrete in the design of the concrete wall reduces the dose rate by a factor of at least 3.52 times approximately compared to ordinary reinforced concrete. So, IMC is better than ORC for radiation shielding.

6. CONCLUSION

The shield design of the concrete wall has been carried out with the help of Point-kernel Shielding Code Micro-Shield 5.05 and this design was also validated based on the measured dose rate values with Radiation Survey Meter (G-M Counter) considering the ICRP-60 (1990) recommendations for occupational dose rate limit (10 μSv/hr). The recommended shield design parameters are: (i) thickness of 114.3 cm Ilmenite-Magnetite Concrete (density 2.76 g/cm^3) or 129.54 cm Ordinary Reinforced Concrete (density 2.35 g/cm^3) for concrete wall A (ii) thickness of 66.04 cm Ilmenite-Magnetite Concrete or 78.74 cm Ordinary Reinforced Concrete for concrete wall B and (iii) door thickness of 3.175 cm Mild-Steel (Iron density 7.87 g/cm^3) on the entrance of decay tank room. In shielding efficiency analysis, the use of I-M concrete shows that it reduced the dose rate by a factor of at least 3.52 times approximately compared to ordinary reinforced concrete in the design of concrete wall. Hence, IMC is better than ORC for radiation protection to provide adequate shielding in the existing space around the concrete wall and the civil construction of this concrete shield was also built based on this computational study.

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