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Radiological assessment of high density shielding concrete for neutron radiography

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Abstract. A radiation shield is a physical barrier placed between a source of ionizing radiation and the object to be protected from the radiation. In this research, concrete was selected as the best shielding material. An investigation was carried out to evaluate a specific concrete mixture developed in [1] for shielding of radiation energies. The high density shielding concrete (HDSC) concrete had 28-day compressive cube strength of 30 MPa, w/c (water/cementitious ratio) of 0.51 and density of 4231 kg/m^3. It composed of CEM 52.5 N, silica fume, hematite aggregates, steel shots, colemanite and chemical admixtures.

This paper presents an evaluation of the shielding properties of the HDSC using foil activation method. It is shown that the HDSC mixture achieved the desired shielding capabilities within the first 250 mm thickness of the concrete block.

Keywords. Radiation shielding, curing, retardation, workability

Introduction

Concrete possesses good compromise thickness requirements for neutron and photon and can be cast into almost any complex shape [1]. It was chosen as a shielding material based on existing knowledge and literature. The objective of this research was to develop and evaluate a special concrete shield that would be used to contain radiations emerging from the core of a nuclear reactor and being transported by a beam port into the neutron radiography experimental chamber. In order to achieve this objective, the program was divided into two phases.

- Phase 1: This phase consisted of identifying raw materials, mix design and its optimization, testing of mechanical properties.
- Phase 2: Involved Monte Carlo Neutron Particle (MCNP) simulations using the identified aggregates and radiological assessment of the mixture developed in phase 1, to validate radiation shielding capability of the HDSC mix.

The results of Phase 1 of this work are presented in another paper [1] while this paper presents the findings of Phase 2 of this research.

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1. Background

A common industrial application of nuclear energy and related nuclear radiation issue is (but not exclusively) in power stations. Here, a radioactive material i.e uranium is used to produce electricity. However, in the process of their energy dissipation, radioactive materials are given out in form of energy referred to as radiation. It is a very powerful form of energy that is absorbed by all matter that it comes in contact with. Plants, animals, humans etc. are all affected by radiation. It attacks the cells which make up the body. Radiation energy remains active for a long time, as experienced in the nuclear explosion accident of 1986 at Chernobyl in the Soviet Union; Japan earthquake and Tsunami of March 2011. A report by the United Nations indicates that the natural exposure of a person to radioisotopes averages 2.0 millisievert (mSv) per year [4-5].

This study was geared towards shielding of radiations from nuclear reactors. In a nuclear reaction, the nucleus of heavy uranium-235 atoms absorbs a thermal neutron to initiate the fission process which splits into two nuclei called fission products. For each fission that occurs, between two or three neutrons are also emitted. These neutrons cause further fission of the enriched nuclei of uranium atom and hence release more energy, form more fission products and emit more neutrons, resulting in a chain reaction [3].

The HDSC concrete designed in this investigation was aimed at providing shielding against both, neutrons and photons emanating from a nuclear reactor. A literature survey on use of concrete for radiation shielding is given in another publication by the authors [1].

2. Materials and methods

Two techniques were used in the radiation shielding assessment namely, (1) the Monte Carlo Neutron Particle (MCNP-X) Simulation technique and (2) experimental radiological assessment of the designed HDSC.

2.1 Monte Carlo Neutron Particle simulations

The purpose of this radiological safety assessment was to model the source of radiation and determine the efficiency of the shielding material in attenuating radiation as the radioactive particles travels through the shield. A target of 1 μSv/h instead of 10 μSv/h (regulatory requirement) was to be achieved at the other side of the shielding material.

2.1.1. Inputs to the simulation model

To perform the simulations, a mix design with defined proportions was required. In this case, a preliminary mix design needed to be produced. From the chemical analysis of concrete ingredients used, economic considerations, and availability of aggregates, it was decided that hematite, colemanite and steel shots would be the main aggregates to be used in developing the HDSC. A mix design of 0.42 w/c (water/cementitious ratio) was used as an input into the MCNP simulation as shown in Table 1. The total (neutron and gamma) dose rate used was determined based on a model which included an equilibrium Low Enriched Uranium (LEU) core, internal geometry of the beam, layout and material composition of the experimental chamber.

Table 1. Mix input of the 0.42 w/c HDSC to MCNP simulations

Ingredients	OPC CEM 52.5 N	Water	Coarse hematite aggregates	Fine hematite aggregates	Fine Colemanite	Steel shots
Mass %	10.13	4.26	40.31	15.34	2.3	27.64

HDSC was modelled as per the mix composition given in Table 1, which gave the elemental composition summarized in Table 2. Although the density of about 4341 kg/m^3 was expected for this composition, the heavy weight concrete was modeled conservatively as 4000 kg/m^3.

Table 2. Elemental composition of the high density concrete used in the MCNP calculations

Element	w%
Al	0.4292
Ca	5.0126
Fe	66.8533
H	0.5296
Mg	0.1222
B-10	0.0673
B-11	0.2960
O	25.6002

2.1.2. Outcomes of the MCNP-X simulations

Calculations showed that the contact dose rates that were achieved, did meet the requirement of 1 μSv/h, with the only exception being the back wall (directly exposed to the open beam) where the contact dose rate at the hottest spot was more than 1 μSv/h but lower than 10 μSv/h. A thicker wall as well as a radiation beam stopper was included in the model in order to achieve the required dose rates.

2.2 Shielding experiment

The foil activation method was adopted for evaluating the attenuation properties of the shielding concrete, against radiations from the white spectrum neutron beam. Foils used in this experiment were made of gold. When a gold foil is activated by neutron beam, it emits characteristic gamma rays which can be counted and related to the neutron flux incident on the foil. By placing foils in front and between the shielding layers, the incident and transmitted neutron fluxes are obtained at each position of shielding thickness.

The simplified set-up of the experiment is given in Figure 1. The shielding material for evaluation consisted of 4230 kg/m^3 dense concrete blocks with surface area dimensions of 100 x100 mm. The thicknesses of the blocks used in the experiment are given in Table 3. The 20 mm thick slices where obtained by cutting the 100 x 100 x100 mm cubes using water jet cutting technology. This method was the only technique available that could cut through the hard (iron) aggregates of the concrete into slices, as shown in Figure 2.

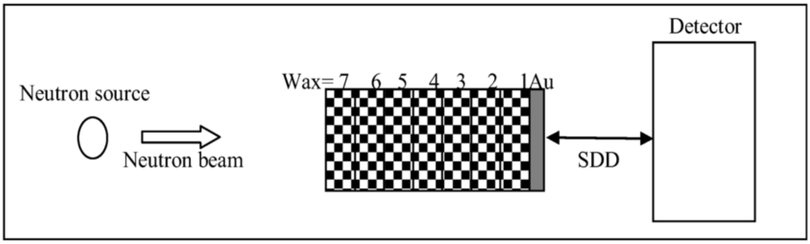

Figure 1. Top view of the experimental setup (SDD - silicon drift detectors)

Table 3. Thickness of shielding material blocks used for measurements of transmitted neutron beam

Thickness (mm)	Quantity
20	5
100	7

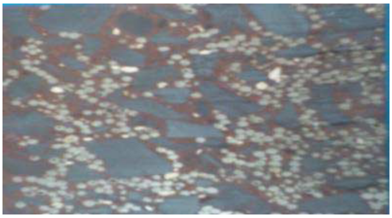

Figure 2. 20 mm thick concrete slice showing steel shot aggregates

The beam size was limited from 300 mm to 50 mm diameter circular area using wax and cadmium. The wax cylinder was prepared carefully to fit into the protrusion of the external shutter. This way the biological shield would protect the scattered neutrons from the wax. The sequence of beam limiting material consisted of 200 mm thick wax and a cadmium sheet of 1 mm thickness. The beam limiter ensured that there was negligible neutron background, arising from the scattering neutron beam multiples.

Foils were cut into 0.5 cm radius discs of 50 x10^{-4} cm thickness. After all layers of the shielding material were placed in the beam axis, ensuring that they are aligned to each other, foils were then placed between the shielding material as shown in Figures 3 and 4, positions F to B6. The 20 mm thick slices were employed in the set up except for the last position B6 where a 600 mm thick shielding concrete was used. Each foil was assigned a unique number and two foils were placed at each position on the left (L) and right (R) side. A thin foam was used to hold the foils in position and in line with the beam axis. Figure 4 is a photograph of the actual shielding experiment that was conducted.

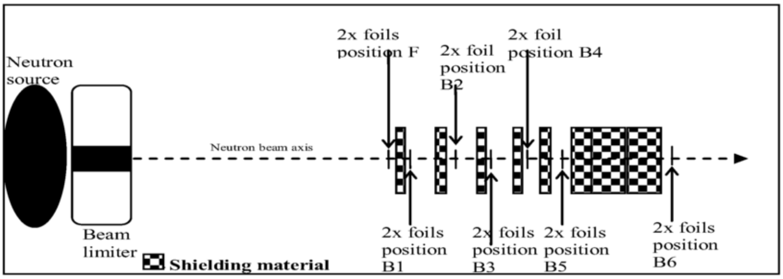

Figure 3. Side view of experimental set up

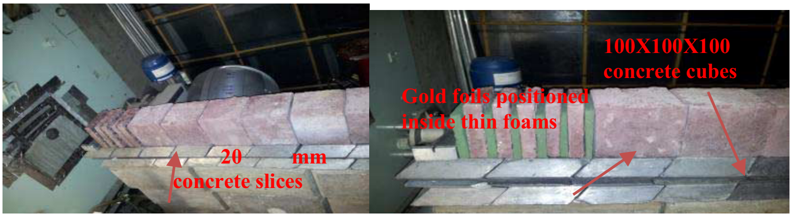

Figure 4. Experimental set up with foil

3. Experimental results

Measurements for flux calculation were conducted at thicknesses 0, 20, 40, 60, 80, 100 and 800 mm of the shielding material. The results for flux and dose rates are shown in Table 4. The linear attenuation coefficient was calculated to be 0.62 cm^{-1}, based on which the transmitted intensities after 200, 300, 400, 600 and 700 mm were calculated. The results in Figure 5 present a threshold line which is the dose rate threshold below which there is adequate estimated shielding for personnel safety from neutrons, as the dose rate is less than 10 μSv/h (20 mSv/y).

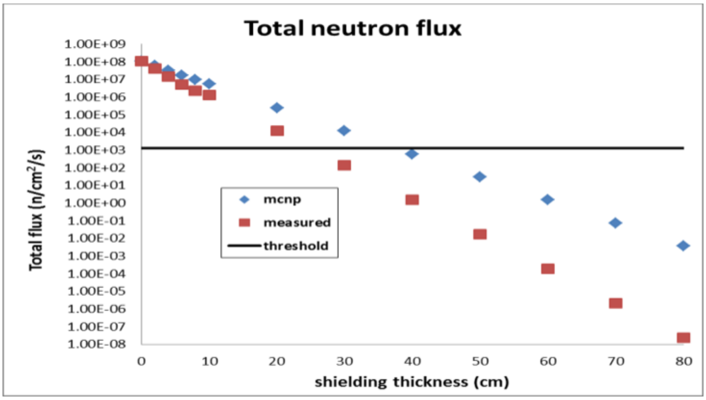

Figure 5. An extrapolated data of MCNP simulation and measured total flux at different thicknesses

Table 4. Experimental results showing dose rate attenuation

Foil position		**Flux ($n/cm^2/s$)**			**Dose rate (mSv/hr)**	
Face	Shielding thickness (mm)	Value	% difference	Average	Value	Average
F-L	0	1.10E+08	2.2	1.11E+08	4.02E+03	4.07E+03
F-R	0	1.12E+08			4.11E+03	
B1-L	20	3.98E+07	11.6	4.22E+07	1.77E+03	1.88E+03
B1-R	20	4.47E+07			1.99E+03	
B2-L	40	1.34E+07	12.5	1.43E+07	5.97E+02	6.37E+02
B2-R	40	1.52E+07			6.77E+02	
B3-L	60	4.72E+06	14.2	5.09E+06	2.10E+02	2.26E+02
B3-R	60	5.45E+06			2.42E+02	
B4-L	80	2.11E+06	19.6	2.34E+06	9.39E+01	1.04E+02
B4-R	80	2.57E+06			1.14E+02	
B5-L	100	1.22E+06	7.5	1.27E+06	5.23E+01	5.43E+01
B5-R	100	1.31E+06			5.63E+01	
Calculated[2]	200	8.64E+03	-	8.64E+03	3.70E-01	3.70E-01
Calculated[2]	300	9.60E+01	-	9.60E+01	4.11E-03	4.11E-03
Calculated[2]	350	1.01E+01	-	1.01E+01	4.33E-04	4.33E-04
Calculated[2]	400	1.07E+00	-	1.07E+00	4.57E-05	4.57E-05
Calculated[2]	500	1.18E-02	-	1.18E-02	5.08E-07	5.08E-07
Calculated[2]	600	1.32E-04	-	1.32E-04	5.64E-09	5.64E-09
Calculated[2]	700	1.46E-06	-	1.46E-06	6.26E-11	6.26E-11
L	800	2.32E-08	No signal	2.32E-08	8.52E-13	0.00E+00

4. Conclusions

The high density concrete mix that was developed [1] met all the requirements specified for shielding purposes. The aggregates used in the mix contained no long half-life decaying elements. The required minimum density of 4000 kg/m^3 as simulated in the MCNP was maintained

The performance of the concrete in shielding was more efficient than expected as the target levels of 1 μSv/h, which were more stringent than the regulatory limits of 10 μSv/h, were achieved at about 250 mm of the 600 mm thick wall of concrete.

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