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Effect of Water Loss on The Neutron Transmission of The Reactor Shielding Concrete

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Review Article

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ABSTRACT

The conditions, surrounding the shielding concrete (moisture and temperature), affect the water content within it and therefore the neutron transmission of the shielding concrete. In this study, the impact of changing the amount of water, in four types of candidate shielding concretes was investigated. The neutron transmission of concrete has been identified experimentally and computationally using the MCNP code. The results show a significant increase in the neutron transmission of each the four concrete mixes due to the water losses, especially between 20°C-100°C. Acceptable compatibility was noticed between the measured and calculated results of the neutron transmission of the studied concrete.

INTRODUCTION

Usually, thick layers of shielding concrete are used in nuclear facilities. Concrete has two functions: to support the facility and prevent the high level radiation emitted from the core from reaching the surrounding environment. The shielding concrete contains water molecules; which contribute to the neutron transmission. The properties of concrete are affected by the surrounding conditions of operation; such as the temperature and humidity which subsequently affect the neutron transmission.

As the temperature increases, the properties of the concrete are modified and the water disappears by evaporation. The interaction of a neutron with a nucleus depends on the type of nucleus, and on the neutron energy. Also, usually, the absorption of thermal neutrons in materials is more likely than the absorption of fast neutrons. Also, the likelihood of a reaction depends on the form of the reaction.

The probability of a certain interaction occurring between a neutron and a nucleus is known as the microscopic cross-section (σ) of the nucleus for a given interaction. This cross section varies with the energy of the neutron. The microscopic cross section is also defined as the active region that the nucleus gives to the neutron for a given reaction. So the higher the effective area, the higher the likelihood of a reaction. Since the microscopic cross section is an area, it is expressed in units of area or square centimeters.

The square centimeter is very large compared to the effective area of the core, and where the square centimeter scales is "as large as a barn" when it applied to nuclear processes. The microscopic cross sections are expressed in terms of barns (10-24 cm^2). If a neutron interacts with a specific volume of matter, then it depends not only on the microscopic cross section of the individual nucleus but also on the number of nuclei within that volume.

MATERIALS AND METHODS

The concrete samples were prepared in the form of 10 cm x 10 cm x 10 cm cubes. These samples were placed in a water bath of 22°C for 28 days after that they left to dry at room temperature. The Samples were heated to a different temperature (from 20°C to 500°C).

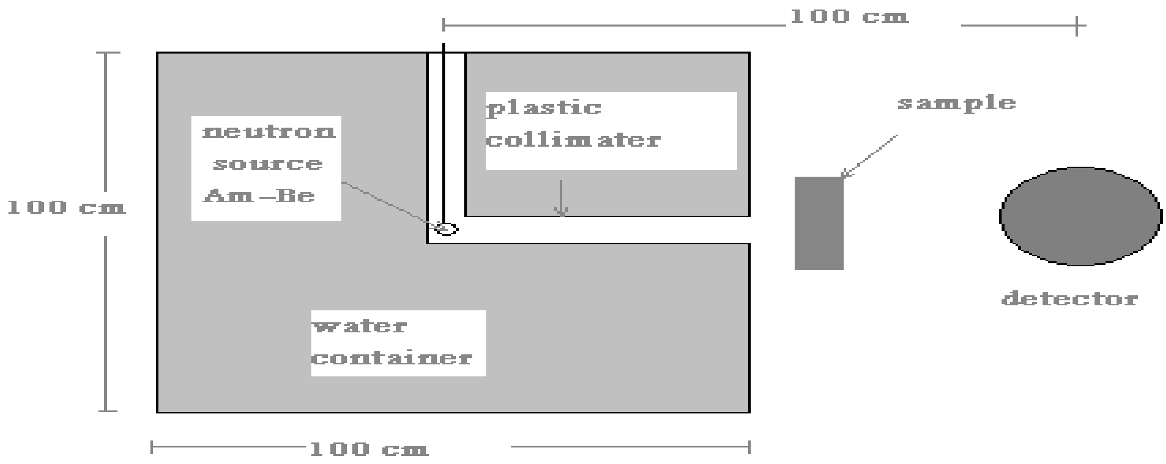
The samples were taken out after being cooled and placed in plastic bags and tied tightly to minimize the exchange of moisture with the surrounding medium. The sample weight was measured and determined the water content of each case. Groups of RS BH, RH, and BS aggregates were obtained by crushing and sieving natural hematite and serpentine rocks [1,2]. The sieve classification of the resulting aggregate. The symbols and mixing ratios for the four mixtures under consideration.

The Am-Be neutron source was used inside a sealed empty plastic tube, located in the center of 1 x 1 x 1 m^3 the container of water, the beam of neutrons Collimated. Berthold Spherical Neutron Detector LB-6411 was used for this experiment. The source center is 100 cm away from the detector center. The ambient dose equivalent was measured after the detector is calibrated in a fast-neutron field.

The detectors of neutrons usually have a high uncertainty. that will change the precision of the results as it happens in many similar experiments applied for the calculation of shielding, for example, the famous ORNL Lid-Tank shielding facility results, these results are useful for hand-made approximate calculation in many practical cases of neutron shielding.

The coefficients of calculated attenuation will consider just the fast neutrons into account, and the coefficients of resulted attenuation are the effective fast-neutron removal cross-sections. The concept of effective removal cross-section is used widely for calculations of neutron attenuation in the shielding of concrete, and most neutrons that pass through the thick shields are fast neutrons [3,4]. The neutron transmission for concrete mixes was measured using broad beam experimental setup shown in Figure 1. The relation between water content and neutron transmission was investigated.

Figure 1. Experimental drawing of the setup of the experiment using the source of Am-Be.



MCNP calculation methods

Due to the cost and difficulty of applying some of radiation measurements, the MCNP simulation can be used to calculate the desired results quickly and accurately at low cost. The MCNP code is one of the means used to simulate radiation measurements.

Currently, the random sampling method is the most popular method. However, the Monte Carlo methods do not provide a solution to the problem, but rather only approximate the average behavior of the particles. In radiation–transport (R–T) problems, this approximation is accomplished by simulating the paths of "particles" (photons) from creation to end (absorption or escape). In code MCNP-4c, the physical quantities of each particle are calculated, such as particle flux over a surface, volume, or energy deposition in a specific part of the geometry (cell). Advances in computer technology greatly influence the use of MC methods, because the statistical sampling process in R-T problems involves time-consuming mathematical operations [5].

The mixing ratios mainly affect the composition of concrete and the chemical composition of the materials used. Loosing of the water by heating the concrete cubes causes loss of weight which means density decreasing.

The MCNP simulation was done for the Am-Be source, composition and dimensions of samples, geometry and conditions of the experiments.

The simulation of the detection process in order to calculate the linear neutron transmission for the modeled concrete shields, were done by using the Monte-Carlo method. The neutron transmission for the above mentioned shields were calculated using the MCNP simulation for each shield. In this study by using of MCNP code the Am-Be point source is located in front of concrete shield with thickness of 10 cm was simulated. Firstly, the amount of flux was determined by point detector tallies. After that the amount of flux was determined for each of concrete mixes, using the neutron transmission [6-10].

RESULTS AND DISCUSSION

The experimental and calculated neutron transmission of the D,F,H and S mixes as a function of water loss rate. Analyzing these results shows the following: Experimentally, the neutron transmission rate increases versus the water loss rate increase in all mixes between 20°C-100°C due to the evaporation of unbounded water. The water loss rate in the mixtures (H-MIX and D-MIX) is larger than (F-MIX and S-MIX mixtures). The (H-MIX and D-MIX) have a small changing of the water loss rate therefore the changing of neutron transmission is slight. The errors of measuring neutron transmission and water loss rate are less the 5%, 6% respectively. The errors of the MCNP-4c calculated results do not exceed 1% because the nps was about 106. The discrepancy difference between the experimental and the MCNP-4c results were less than 5%. Both experimental and simulation results in all cases gave good agreement.

CONCLUSION

Losing of the shielding concrete water decreases the neutron attenuation characteristics for all concrete mixes especially, Consideration must be given in the design of shielding concrete for nuclear reactors as well as for neutron sources, for mixtures H-MIX and D-MIX. The neutron transmission rate of local concrete mixes was simulated using the MCNP-4c code. The simulation results and experimental ones were compared. MCNP-4c simulation results and experimental ones gave acceptable compatibility.

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