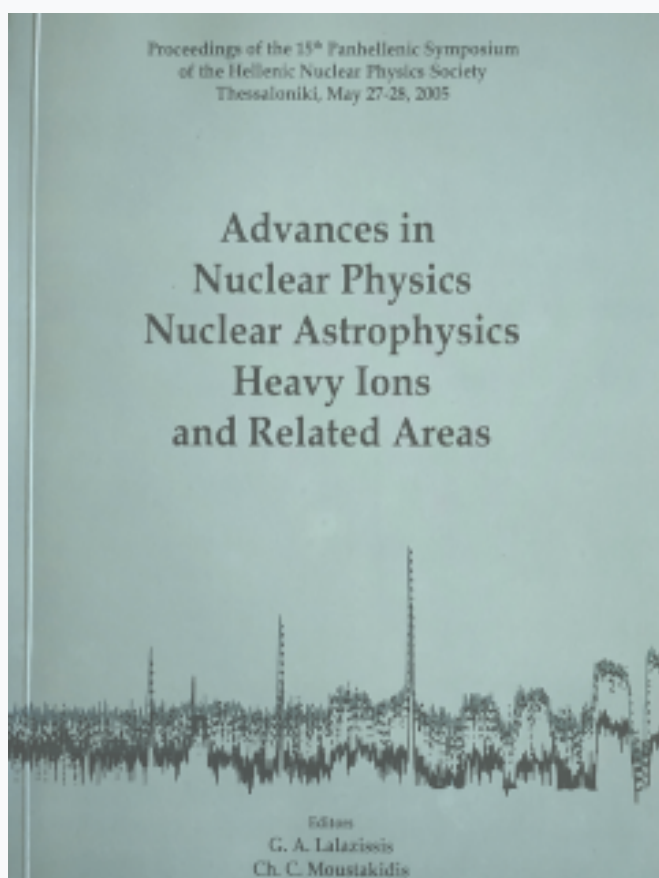


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Shielding around spallation neutron sources

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Abstract

Spallation neutron sources provide more intense and harder neutron spectrum than nuclear reactors for which a substantial amount of shielding measurements have been performed. Although the main part of the cost for a spallation station is the cost of the shielding, measurements regarding shielding for the high energy neutron region are still very scarce. In this work calculation of the neutron interaction length in polyethylene moderator for different neutron energies is presented. Measurements which were carried out in Nuclotron accelerator at the Laboratory of High Energies (Joint Institute for Nuclear Research, Dubna) and comparison with calculation are also presented. The measurements were performed with Solid State Nuclear Track Detectors (SSNTDs).

1 Introduction

Several experiments have been performed in order to study the neutron shielding around nuclear reactors. Spallation neutron sources can generate harder neutron densities and harder spectrum than nuclear reactors [1]. The neutron spectrum of spallation source is similar to the neutron spectrum around nuclear reactors up to 10 MeV. The difference between the two spectra is observed at higher neutron energies, from 10 MeV up to few GeV. In this neutron energy region, shielding experiments are still very scarce. Since this energy region is higher than the energy region around nuclear reactors and lower than the energy region around high energy accelerators. In addition the cost for the radiation shielding contributes a considerable part of the total

cost, since massive shields for high energy neutrons having strong penetrability are required. So the material to be chosen should be readily available and not expensive. Calculations or experiments to get the dose attenuation length are quite important in order to found the most appropriate material to surround a spallation source. In radiation shielding research, non-charged particles, photons and neutrons, are the main radiation to be considered. In this study calculations to design the optimal shielding were performed taken into account mainly the neutron contribution. Regarding the selection of an appropriate shielding material to surround a spallation source the interaction length of high energy neutrons in shielding material must be calculated. The ENDF/B-VI cross section set was used in order to calculate the interaction length according to the relationship between the cross section and interaction length [2]. Shielding experiments are important to investigate the accuracy of calculations. Hence measurements using Solid State Nuclear Track Detectors (SSNTDs) were performed in Nuclotron accelerator at the Laboratory of High Energies at Joint Institute for Nuclear Research (JINR) in Dubna Russia. To compare the calculations with the experimental results the transmission factor of neutrons was estimated. The transmission factor is the ratio of the ambient dose equivalent values (neutron or gamma) with shield and the neutron ambient dose equivalent without shield [3]

2 Experimental

The spallation neutron source consists of cylindrical Pb target 50 cm in length and 8 cm in diameter. Around the Pb target U rods were placed hexagonally. Four sections of natural Uranium blankets constitute the spallation source. Each section consists of 30 U-rods with 10.4 cm length and 3.6 cm diameter. The whole system is surrounded, for radiation protection purpose, by polyethylene moderator [4].

Neutron distributions on the surface of the U-blanket and on the top of the shielding were performed. On the surface of the each section of the U-blanket, four sets of SSNTD were positioned in the vertical direction relative to the target axis. At the corresponding position on the top of the shielding, three sets of SSNTDs were placed vertical to the beam axis to measure the neutrons escaping the set-up. Each set of the detectors consist of three parts. One part is bare CR39 which is used to detect intermediate-fast neutrons from 0.3 up to 3 MeV from proton recoils on the detector itself. The other two parts are CR39 covered with $\text{Li}_2\text{B}_4\text{O}_7$ neutron converter, while one of these parts is also covered on both sides by 1mm Cd. By this detector arrangement information about thermal-epithermal neutrons can be obtained [5].

3 Results and Discussion.

3.1 Calculations

Most shielding designs for high-energy accelerators have been performed by using a point kernel method, Moyer model [6,7]. Moyer model is based on exponential attenuation of neutron dose equivalent for neutrons reaching the equilibrium state after thick shield. This method uses a single built-up factor and an attenuation length. However, for lower energies, the attenuation length depends on the neutron energy and the simple Moyer model is no longer applicable. In the practical neutron shield designs of spallation neutron sources there is a need for quick calculation method such as point kernel code. Neutron attenuation calculations are sometimes made by Monte Carlo method based on intra-nuclear cascade model. However this method is very time consuming and complicated. The determination of the shield necessary around a spallation source usually requires the estimation of the neutron fluence to be expected outside a given shield. The relationship used for calculation behind the shielding is related to that used for dose calculations behind shielding of high energy proton accelerators. In the general case this relationship is the following:

$$\Phi(x, \theta) = \frac{\Phi_o(\theta)}{r^2} \exp\left[-\frac{x}{g(\theta)\lambda}\right] [1]$$

$\Phi_o(\theta)$ is taken as $\Phi_o(90^\circ)$ describes the number of neutrons crossing at 90° the moderator surface. The r corresponds to the distance between the source and the point of interest, x is the depth inside the shielding, $g(\theta)$ is defined as $\sin\theta$ and λ is the interaction length[8]. The interaction length is depending on the energy of the neutrons arriving the shielding. For each neutron energy the interaction length has been calculated using the relationship between the interaction length and the inelastic cross section [2].

The ENDF/B-VI cross section data used in order to calculate the interaction length of neutrons inside the polyethylene. In table 1 the interaction length of neutrons in a polyethylene moderator is presented. Applying the neutron interaction length, as presented in table1, in the equation 1 the attenuation of neutron in a polyethylene moderator, for various neutron energies can be calculated (figure 1). The neutron attenuation becomes stronger as the neutron energy decreases. Except from the thermal neutron region in which a built-up effect is observed. This is due to the fact that neutron interaction length in polyethylene becomes smaller as the neutron energy decreases. Inside the polyethylene moderator neutrons interacts by elastic reaction with the hydrogen. So the neutron average energy decreases quickly and more thermal neutrons are produced behind the polyethylene shielding.

Table 1
Neutron interaction length in a polyethylene moderator for different energies

Neutron Energy	σ_{in} (barn)	λ (1)
1eV-10KeV	40	0.625
100 KeV	28	0.893
500 KeV	14	1.79
1 MeV	6	4.17
5 MeV	4	6.25
10 MeV	2.5	10
20 Mev	1.5	16.7
30 MeV	1.3	19.2
100 MeV	0.42	59.5
1 GeV	0.28	89.3

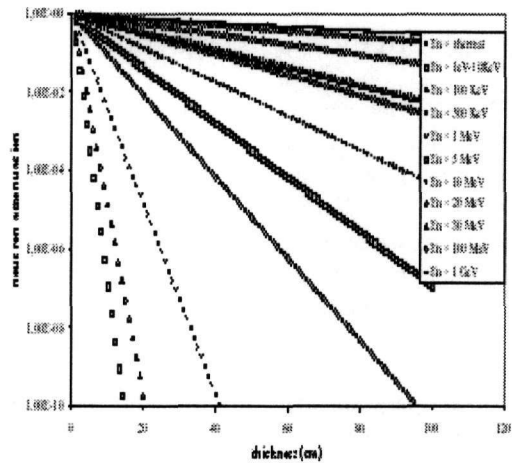


Fig. 1. Neutron attenuation in polyethylene moderator

The neutron spectrum produced by irradiation of Pb plus Uranium Blanket set up with 1.5 GeV proton beam was theoretically determined using the high energy transport code DCM/DEM. The DCM/DEM code is a Dubna version of cascade-evaporation approach, based to Bertini model. According to this calculation mostly intermediate and fast neutrons were produced [9].The neutron spectrum that penetrated through the shield assembly of polyethylene

has been estimated using the relationship 1. The overall view of this spectrum reveals the following facts. The neutron spectrum behind the polyethylene shielding still has a broad peak at intermediate-fast neutron energy region. But this component has decreased behind the shielding. This is because the neutrons in polyethylene are mainly slowed down by elastic scattering on hydrogen. The high elastic reaction cross section of neutrons in polyethylene causes shifting of fast neutrons below the resonant region. Therefore more thermal neutrons are produced after the 26 cm of the polyethylene shielding. The total number of escaping neutrons is at least one order of magnitude less than on the U-blanket surface and half of those neutrons are in the thermal energy range.

3.2 Experimental results

In order to compare calculations with experimental results neutron flux measurements on the U-blanket surface and on the top of the shielding were performed using SSNTDs. Among the four set of the detectors placed on the U-blanket no significant variations were observed. The same behaviour is also observed between the three set of the detectors placed on the top of the shielding [4]. Thermal-epithermal neutrons are one order of magnitude less than fast neutrons. The total number of escaping neutrons from the shielding is at least one order of magnitude less than the total number of neutrons produced by the spallation neutron source, while the half of these neutrons are in the thermal energy range. These results are in good agreement with calculations.

To compare calculations with experimental results the transmission factor of neutrons was calculated. In general the transmission factor is the ratio of the ambient dose equivalent values (neutron or gamma) with shield to the neutron ambient dose equivalent without shield [3]. More specific the transmission factor of neutrons is the ratio of neutron fluence outside the shielding material to the neutron fluence produced by the spallation neutron source, in specific energy region. In table 2 the transmission factor of neutron obtained by calculation and experimental results for both thermal-epithermal and intermediate-fast neutron components is presented.

Table 2

Transmission factor of neutrons after 26 cm polyethylene moderator

Thermal-Epithermal neutrons		Intermediate Fast neutrons (0.3-3) MeV	
Calculated	Experimental	Calculated	Experimental
$8,9 \cdot 10^{-2}$	$(8,6 \pm 0,34) \cdot 10^{-2}$	$3 \cdot 10^{-2}$	$(2,9 \pm 0,8) \cdot 10^{-2}$

4 Conclusion

The transmission factor of neutrons for both thermal- epithermal and intermediate-fast neutron components behind the 26 cm of the polyethylene has been estimated. According to these calculations the transmission factor for thermal-epithermal and intermediate-fast neutron components is 0.089 and 0.03 respectively. The goal is to design shielding, using relative available and not expensive materials, with as less as possible transmission factor (about 0.004 for ambient dose equivalent). Therefore calculations in order to found the most appropriate shielding to surround a spallation source are quite important. The good agreement between calculations and experimental results obtained by SSNTDs shows that by calculating neutron interaction length in various shielding materials it is possible to design the proper shielding for a spallation source.

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