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Collimator Design for Neutron Radiography Systems Using a Reactor Flux

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Abstract: This paper is aimed to investigate the design of an optimized neutron radiography system that utilizes the flux of a reactor. Moderation, collimation aspects are studied. A Monte Carlo code, MCNP4C, was utilized to achieve a maximum and more homogenous neutron flux in the collimator outlet next to the image plane, taking into account geometric characteristics. It was possible to obtain a normalized thermal neutron flux, at the image plane, equals 10^{-9}n cm^{-2} s^{-1} by a collimator design set.

Key words: Neutron radiography · Collimator · MCNP4C · Reactor flux

INTRODUCTION

Neutron Radiography (NR) is an important tool in non-destructive examining, which has been adopted in industrial, medical, metallurgical, nuclear and explosive inspections [1]. From the 1990s to the present day, the imaging systems used to do neutron radiography and the methods used to analyze the images have continued to advance [2]. The improvement of existing detectors with, for example, thinner scintillation screens and the development of new detectors like the microchannel plate, have supplied continual increases in resolution. The amount of scattering or absorption of neutrons by atomic nuclei varies in an obviously random style through the periodic table. Hydrogen especially has a very large scattering cross-section. Neutrons can therefore make good contrast for light atoms in the presence of heavy atoms.

To construct NR systems can use of nuclear reactors with high neutron fluxes [3] However, their costs and their impossibility of transporting is caused they have some limiting factors to application. Moreover, it is necessary to design and build transportable systems of NR for more applications. Then to have systems which can easy to act, handle and repair, these systems must be well shielded and the neutron radiography rules and regulations followed. Hence, the size, weight, shielding and operational stipulations of the system must be optimized. For this purpose, the size, weight, shielding and operational stipulations of the system must be optimized. The objective of this work is to optimize NR parameters, such as moderator thickness, collimation and shielding in order that a NR system can be constructed with maximum and as uniform as possible thermal neutron flux at the image plane.

For the designing of the suggested system, MCNP version 4B and the ENDF/B-VI cross section set were used. MCNP is a general purpose, Monte Carlo, radiation transport code for three-dimensional, continuous energy, time-dependent neutron, photon and electron transport [4].

Moderation Design: The thermalization of the neutrons is the first step in the optimization system of a neutron beam, i.e. they are slowed down up to they obtain an equivalent temperature equal to the environment. The material which can use as a moderator should have a considerable scattering cross section, a reduced cross section for absorption (Silva *et al.*, 2001). There are not exist materials that have all these properties. However, combine some material such as iron, high density polyethylene, boron carbide, titanium dioxide and Air can use for this purpose. The properties of moderators consist of chemical composition, density (g/cm^3), high average logarithmic energy loss (ξ) and the moderating ratio (R_m) which are defined by [5].

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Table 1: Properties of the moderator materials studied

Material	Chemical composition	Density (g/cm^3)	ξ	R_m
Iron	Fe	7.87	0.035	0.150
Bismuth	Bi	9.747	0.0095	1.644
Lead	Pb	11.35	0.0096	0.580
High density polyethylene	C_2H_4	0.98	0.914	64.00
Boron Carbide	B_4C	2.52	0.397	0.003
Titanium dioxide	TiO_2	4.23	0.143	0.137

$$\xi = \frac{\sum_{i=1}^{N} \xi_i \sigma_s^i}{\sum_{i=1}^{N} \sigma_s^1} \quad (1)$$

$$\sum\nolimits_{a,s}^{X_n Y_m} = \frac{\rho^X n^Y m N_A}{A^X n^Y m} \sum\nolimits_i (nf_X^i \sigma_{a,s}^{X,i} + mf_Y^i \sigma_{a,s}^{Y,i}) \quad (2)$$

$$R_m = \xi \frac{\sum_s}{\sum_a} \quad (3)$$

where σ_s^i is scattering cross section of ith of element and $\sum_{a,s}^{X_n Y_m}$ is absorption and scattering cross section for $X_n Y_m$ chemical composition. The results of moderator's properties are shown in Table 1. From this comparison it is evident that C_2H_4 is more suitable for moderator and other materials can use for diaphragm and other part of NR collimator system.

Thermalization Factor: When reactors is used as a source for thermal neutron radiography, the moderating body that hosts the source is applied in order to supply the largest possible flux of thermal neutrons in a region from which the beam may be extracted by the collimator. Then, an important parameter is the thermalization factor (TF), which is determined by (3):

$$TF \propto \frac{Fast\ neutron\ yield}{Peek\ thermal\ flux} \quad (4)$$

The best moderator to use with a reactor neutron source is commonly the one in which a small value for TF can be accepted [6]. The first step of analysis, calculations were carried out to evaluate the efficiency of the thermal moderation of some selected materials. The chosen geometric configuration was a cylindrical system with 208 cm length and 12.73 cm inlet diameter of primary collimator. A reactor source is placed in distance of 86.29 cm of inlet aperture of the collimator.

Collimator Design: In order to deal with the indispensability to enhance the area of radiography inspection of the target, together with the isotropy and point wise geometry of the source, a divergent-type collimator was planned in this study. The collimator optimization was carried out by considering the geometric unsharpness and the probability of obtaining the maximum intensity of thermal neutrons at the image plane. The geometric configuration of the recommended system of thermal neutrons is shown in Fig. 1. The system is made up of a reactor source, a divergent collimator with three sections (primary, intermediate and secondary collimators) which arrives an image plane.

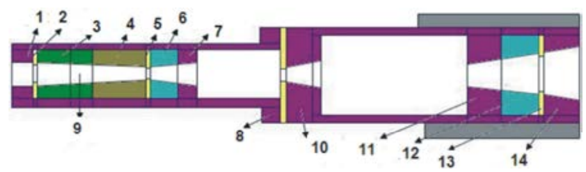

Fig. 1: The geometry of neutron radiography system by MCNP4C

RESULTS

The flux of thermal neutrons obtained in the moderator, per neutron source rate [7] which called the moderation efficiency, the flux of fast neutron and the ratio of thermal to fast flux can be seen in Fig. 2 for different material. TF values for the examined material are proportional to the ratio of fast to thermal flux. As shown in Fig. 2, the best moderator to use with a reactor neutron source is polyethylene (38 cm) and Fe (10 cm) and polyethylene (28 cm) which they have small value for TF. This result is consistent to obtained data of moderating ratio. The neutron flux for iron with length of 10 cm mid poly ethylene with length of 28 cm is better than others. Placing iron mid poly ethylene not only works as a moderating piece but also the most role of iron is to protect poly ethylene against gamma rays.

The moderating piece does not affect the intensity of thermal neutrons in the image plane because thermal neutrons pass through the hole of the primary collimator. The results show that the intensity of fast neutrons in the image plane significantly has been reduced after placing the moderating piece (Fig. 3).

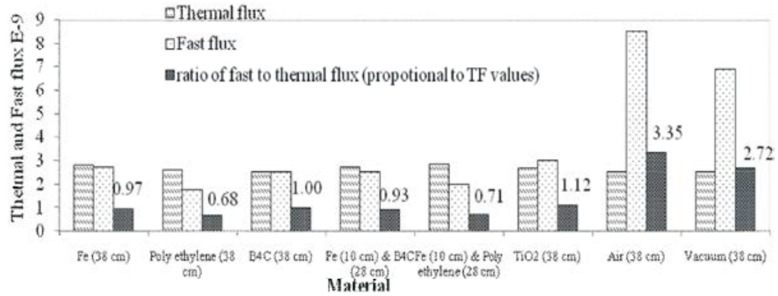

Fig. 2: Variation of thermal neutrons flux, fast flux and the ratio of thermal to fast flux for different material

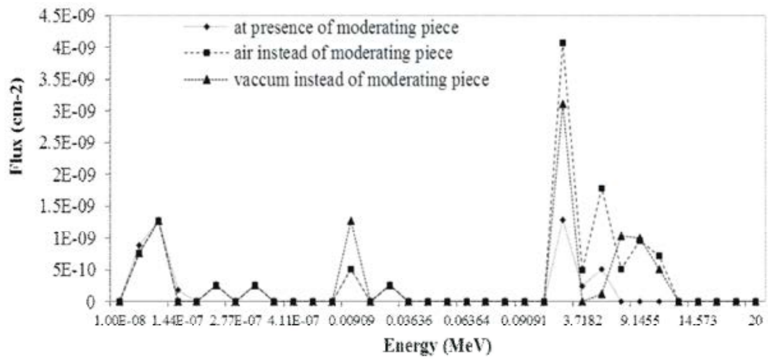

Fig. 3: Variation of neutrons flux as a function of energy at presence of moderating piece or without that

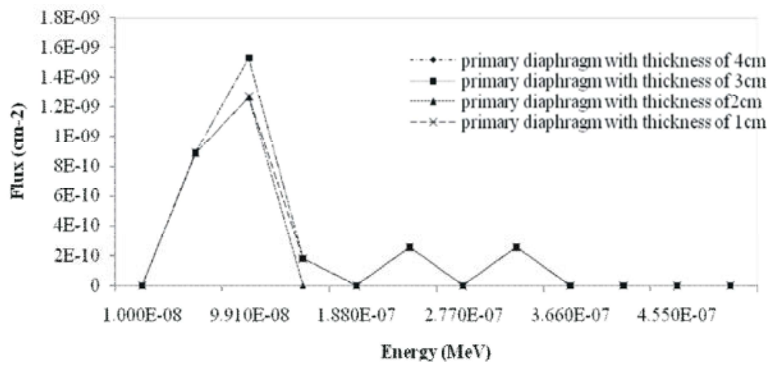

Fig. 4: Variation of thermal flux as a function of energy at different primary diaphragm diameters

In association with these materials, thermal neutrons flux as a function of energy is investigated with different primary diaphragm thickness. Variation of the thermal neutrons flux with primary diaphragm thickness is approximately same (Fig. 4). For this reason and in addition the high cost and available of boron carbide sheet in the marketing, the 1 cm thickness is used.

In the optimization procedure of the collimator, the collimator length L, the inlet aperture D andthe diameter of the collimator inlet next to the image plane, D_O, were varied in order to achieve the values for which the maximum thermal flux is obtained in the image plane. The distance between object and detector are 10 mm. The best obtained results and the geometric unsharpness, U_g and the collimation ratio, L/D are shown in in Table 2.

For reducing the exposure time, it is necessary to have a high neutron flux at the object. The L/D rate has been reduced by increasing D or decreasing L, which results the neutron flux been larger. In other hand, increasing of neutron flux will decrease the geometric unsharpness. So, the lower L/D is required for higher neutron flux and the higher L/D is required for higher resolution. In this article, the L/D rate is equal to 16.33 that if the object to detector distance be 10mm, the geometric unsharpness will be 0.61. This yields almost $10^4 n/cm^2\, s^{-1}$ neuton flux, that the exposure time with this flux at high quality digital neutron radiography is too short about 10 minute or less. Also, for obtaining the different geometric unsharpness, we consider different position of object and detector.

Table 2: Thermal flux at image plane (φ_{th}), geometric unsharpness (U_g) for different L/D ratios (L_f=10 mm)

D (cm)	L (cm)	$\phi_{Th}(cm^{-2})$	U_g(mm)	L/D
12.7	706.5	8.3E-11	0.18	55.5
10	300	9.5E-10	0.33	30
12	300	5.8E-10	0.40	25
12.5	250	3.2E-10	0.50	20
12.7	208	2.9E-9	0.61	16.33

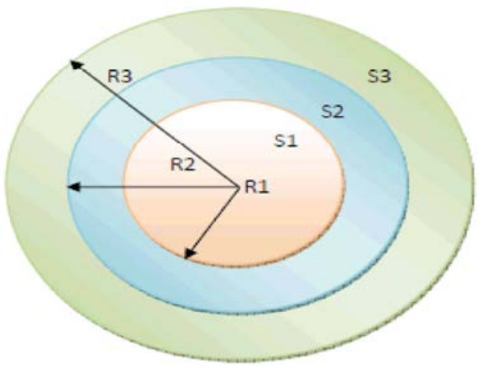

Fig. 5: Schematic design of the image plane feather which divided to three part with same area

In order to investigate of accuracy of flux calculations in simulation eight different nps is considered and the total flux in the image plane for all nps is obtained. The data show same results for them with coefficient variation value (CV) equal 13.57%.

In this study, homogenous of flux in image plane is studied by dividing image plane to three part with same area as shown in Fig. 5. Fsn tally is used for this purpose and by comparison calculated neutron fluxes in each area, the number of neutron per unit area were same as a result of neutron distribution in image plane is homogenous.

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