STUDY OF A PORTABLE NEUTRON RADIOGRAPHY SYSTEM USING ^{241}Am-Be NEUTRON SOURCE FOR INDUSTRIAL APPLICATIONS

Robson Ramos
CNEN/IEN/DIRA/SECIC
Caixa postal 68 550
21945-970 Ilha do Fundão
Rio de Janeiro, RJ, Brasil
robson@ien.gov.br

Ademir X. da Silva
[PEN/COPPE – DNC/EE]CT/UFRJ
Caixa Postal 68 509
21945-970 Ilha do Fundão
Rio de Janeiro, RJ, Brasil
ademir@lmn.con.ufrj.br

Verginia R. Crispim
[PEN/COPPE – DNC/EE]CT/UFRJ
Caixa Postal 68 509
21945-970 Ilha do Fundão
Rio de Janeiro, RJ, Brasil
verginia@lmn.con.ufrj.br

Abstract. *This paper shows a study of a portable neutron radiography system using ^{241}Am-Be source aiming industrial applications. Moderation, collimator and shielding are considered. A Monte Carlo Code, MCNP4B, has been used to obtain a maximum and more homogeneous thermal neutron flux in the collimator outlet next image plane. Among the various moderator materials analyzed, the high-density polyethylene showed to be the most efficient, with a thermalization factor of 271 cm^2. In association with the presented herein, it was possible to obtain an L/D ratio 6.4 for the thermal neutron flux, at the collimator outlet next to the image plane, for neutron flux of up 120 n.cm^{-2}s^{-1} considering an ^{241}Am-Be neutron source with 185 GBq of activity.*

Keywords. *Neutron Irradiators, Moderation, Collimation, Neutron Radiography, Nondestructive testing.*

1. Introduction

Non-destructive testing (NDT) is a method of materials characterization very important to the materials engineer. Problems and defects of all kinds arise in the development and use of mechanical devices, electrical equipment, hydraulic systems, transportation mechanisms and the like. However, an extremely wide range of NDT methods are available to help the engineer to examine these different problems and various defects in an assortment of materials and under varying circumstances. The Neutron Radiography (NR) has been recently recognized in the military, industrial communities, as a necessary and essential non-destructive testing method that has been applied in especial cases of inspection, where it is difficult to take radiographs by X-rays or gamma rays (Crispim, 1993). Typical uses for neutron radiography include finding cracks or early corrosion in material structures such as metal castings, determining process flow problems, real-time dynamic imaging of two phase material flow processes, commercial applications for the semiconductor industry and a growing research portfolio (Bastürk et al., 2002). The study of portable NR systems with small neutron flux is paramount importance, mainly as a result of the current evolution status of NR techniques that can handle image processing in real time. Nuclear reactors supply high fluxes and can be used to build up NR systems, nevertheless, the high technology involved, the costs and the impossibility of transporting them are limiting factors to the applications (in loco) and development of new fields of NR (Sinha et al., 1996). Alternatively, small accelerators and radioisotopes are used. Sources using (α,n) nuclear reaction with beryllium (Be) are common and ^{241}Am-Be source, with 2.2x10^6 neutrons per second per curie yield, is an example. In order to enlarge the range of application of the NR technique, it is necessary to design and construct portable systems for in loco inspections with easy operation, handling, repair, besides properly shielded and following radiography rules and regulations. Therefore, size, weight, shielding and operation conditions of the system are supposed to be optimized. The major goal of this work consists in optimizing NR parameters, such as moderator thickness, collimation and shielding so that a portable NR system can be constructed with maximum and as uniform as possible thermal neutron flux at the image plane. The ^{241}Am-Be was the neutron source chosen among the available commercial radioisotopes because of its a long half-life (458 years) and low emission of gamma radiation. For the modeling of the proposed system, the Monte Carlo code transport MCNP-4B was used (Briesmeister, 1997).

1.1. Moderator

The first step in optimization process of a neutron beam consists of the thermalization of the particles, i.e. they are slowed down until they reach an equivalent temperature equal to the environment. Thus, the moderator material must have high average logarithmic energy loss, ξ, (Duderstadt and Hamilton, 1976) which is defined by

$$\xi = 1 - \frac{(A-1)^2}{2A} \cdot \ln\left(\frac{A+1}{A-1}\right) \tag{1}$$

Where A is the mass number of the moderator material, This material should have a high scattering cross section, (Σ_s), and a low absorption cross section, (Σ_a), because, otherwise, too many neutrons are lost due to absorption. No existing material possesses all these properties, however, it is possible to put together these parameters and define a moderating ratio, R_m, by means of the expression:

$$R_m = \xi \frac{\Sigma_s}{\Sigma_a} \tag{2}$$

The moderating ratio is a relative measure of the capacity of a moderator in spreading neutrons without absorbing a great number of them. It is supposed to be as large as possible so that a good moderating material can be met. Properties of some moderators are shown in Tab. (1).

Table 1. Properties of the moderator materials studied.

Material	Chemical composition	Density (g/cm^3)	ξ	$\xi\Sigma_s$ (cm^{-1})	R_m
Light Water	H_2O	1.0	0.920	1.35	71
Paraffin	$C_{25}H_{52}$	0.89	0.917	1.69	64
High Density Polyethylene	C_2H_4	0.98	0.914	1.80	64

These moderators were selected for the optimization study of the moderating/shielding system of neutrons arising from a ^{241}Am-Be source with 185 GBq of activity (5 Ci). Figure (1) shows the geometric configuration of the neutron source used in this work. Fabricated with a compacted mixture of americium oxide with beryllium metal powder (Amersham, 76/7) and doubly encapsulated in vacuum melted stainless steel (grade AISI.316) sealed by argon arc welding.

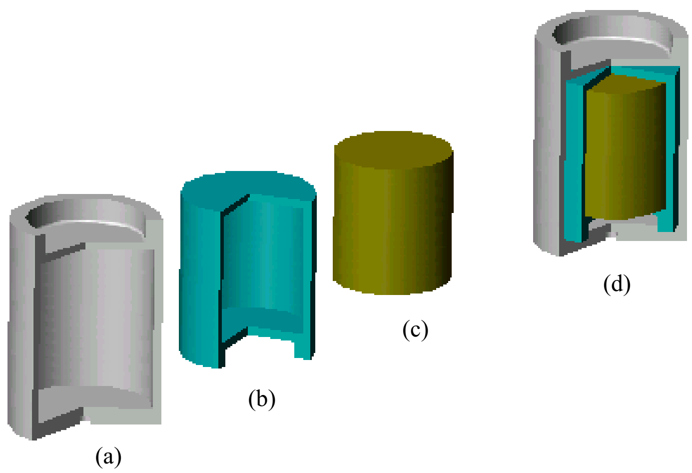

Figure 1. Geometric configuration of ^{241}Am-Be source. Here (a) outer encapsulated of 3 cm diameter, 6cm of height and 0.12 cm of wall-thickness; (b) inner encapsulated of 2.76 cm diameter, and 0.12 cm of wall-thickness; (c) ^{241}Am-Be cylindrical source and (d) general configuration.

Thermalization Factor

When the use of radioisotopes as sources for thermal neutron radiography is desired, the role of the moderating body that hosts the source is to provide the largest possible flux of thermal neutrons (energies less than 1 eV) in a region from which the beam may be extracted by the collimator. Hence, a very important parameter is the Thermalization Factor, TF, which is defined by (Barton, 1972):

$$\mathrm{TF(cm^2)} = \frac{\text{Fast neutron yield (n/s)}}{\text{Peak thermal flux } (n/cm^2.s)} \tag{3}$$

The best moderator to use with an isotope neutron source is usually the one in which a small value for TF. For the first analysis, calculations were carried out to evaluate the efficiency of the thermal moderation of some selected materials. The chosen geometric configuration was a cylinder system with radius of 20 cm. The ^{241}Am-Be source was placed in the center of the simulated cylinder system.

1.2. Collimator

In order to deal with the necessity to enhance the area of radiography inspection of the object, along with the isotropy and cylinder geometry of the source, a divergent-type collimator was considered in this work. The collimator optimization was feasibility consideration to reaching the maximum intensity of thermal neutrons at the image plane. Figure (2) shows the geometric configuration of the proposed system of thermal neutrons. The system is composed of a ^{241}Am-Be source imbedded in a 15 cm-thick moderating cube of high density polyethylene and a conic divergent collimator with a 1 mm-thick layer of cadmium attached to the internal walls, what ensures that only the thermal neutrons from the primary bundle can actually arrive at the image plane. The collimator base was positioned tangentially at 3.25 cm from the source, which is the region of maximum of thermal neutron produced in high density polyethylene, see Tab. (2). Three conic divergent collimator of different sizes were analyzed so as to estimate the thermal neutron flux at the image plane.

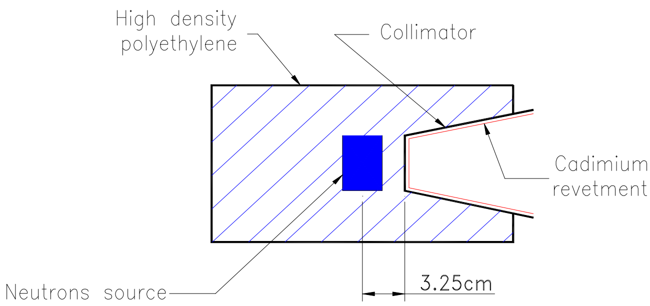

Figure 2. Geometric configuration of the moderating system.

1.3. Shielding

Among the many materials that can be used to shield the proposed NR system, the polyethylene-boron (Gujrathi and D'auria, 1972) was selected because, besides being effective shielding materials and requiring a smaller thickness to cover the irradiating system than usual materials found in the literature, it can also be utilized in external structures, are suitable for low budget projects and are satisfactory heat-resistant. The geometric configuration originally chosen for the study on the shielding efficiency of the material pointed out is depicted in Fig. (3).

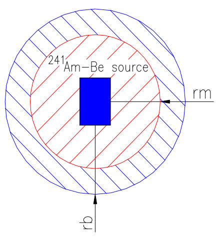

Figure 3. Simulated geometric configuration: rm, radius of the moderating cylinder of high density polyethylene (15 cm) and, rb, radius of the shielding cylinder.

The model of the irradiating system is comprised of a high density polyethylene moderating cylinder of fixed radius 15 cm (rm), with a fixed source at de center, and a variable external radius of the shield (rb), aiming at obtaining rate estimates of the dose equivalents at the external surface of the shield.

2. Monte Carlo MCNP Simulation

The Los Alamos National Laboratory Monte Carlo N-Particle radiation transport code, MCNP in 4B version (Briesmeister, 1997) was employed to perform the calculations in this work. MCNP is a general purpose, three-dimensional general geometry, time-dependent, Monte Carlo N-Particle code that is widely used around the word for many radiation protection and shielding applications. Photon interactions in the MCNP4B code are performed by MCNP photon library MCPLIB, which is based on evaluated data from Evaluated Nuclear Data Files (ENDF). MCNP has a generalized input file capability with allows the user to specify an infinite variety of source and detector conditions without having to make modifications to the Monte Carlo source code itself. The size, shape and spectrum of the radiation source, the composition and configuration of the medium through which photons are transported, and the detector geometry and type (energy, flux detection, etc) are all user-defined parts of this input file.

3. Results

The moderation efficiency, defined as the flux thermal neutrons obtained in the moderator per neutron source (IMEL, 1996), to analyzed materials can be seem in Fig. (4). The flux of thermal neutrons for paraffin ($C_{25}H_{52}$), high density polyethylene (C_2H_4) and light water (H_20) is about the same, since they possess approximately the same number of hydrogen atoms per volume unit. In association with these materials, the flux of thermal neutrons decreases rapidly as the radius of the moderating cylinder increases.

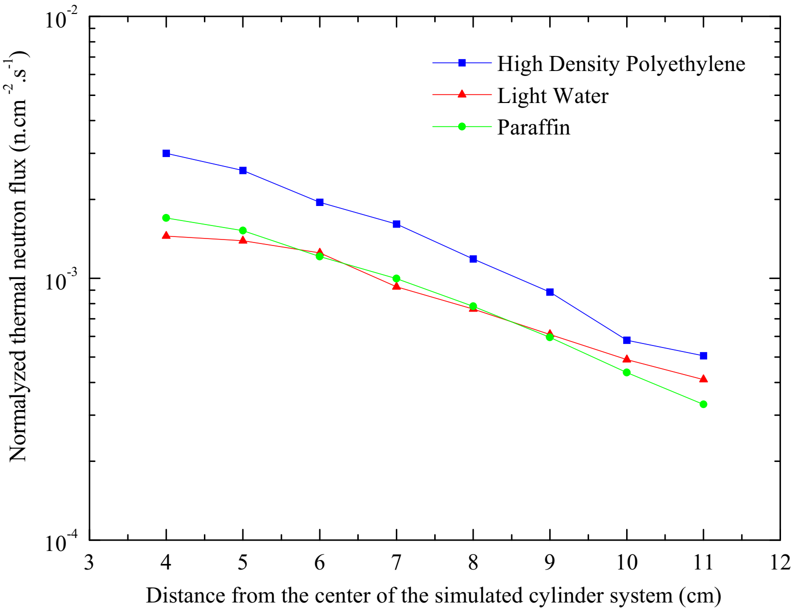

Figure 4. Normalized thermal neutron flux value, considering to ^{241}Am-Be source immersed in moderator materials: high density polyethylene, light water and paraffin.

Table (2) shows the thermalization factor (TF) values obtained from the normalized thermal neutron flux (1/thermal flux), the position at which the peak of thermal neutrons flux takes place in the moderating cylinder with its respective value, the epithermal and fast flux for a ^{241}Am-Be source. The estimates are normalized by source particle.

Table 2. Moderating efficiency of the analyzed materials.

Material	Composition	TF (cm^2)	Position (cm)	Thermal Flux[a]	Epithermal Flux[a]	Fast Flux[a]
Light Water	H_2O	552	4.0	1.81E-3	9.22E-4	5.42E-3
Paraffin	$C_{25}H_{52}$	552	3.75	1.81E-3	1.25E-3	5.75E-3
High Density Polyethylene	C_2H_4	271	3.25	3.69E-3	2.39E-3	8.06E-3

[a] *(n.cm^{-2}.s^{-1} per source neutron rate)*

In the collimator optimization process, the collimator length, L, the inlet aperture, D, and collimator diameter inlet next to the image plane, D_0, are showed in Fig. (5), were varied in order to attain the values for which the maximum thermal flux is obtained in the image plane

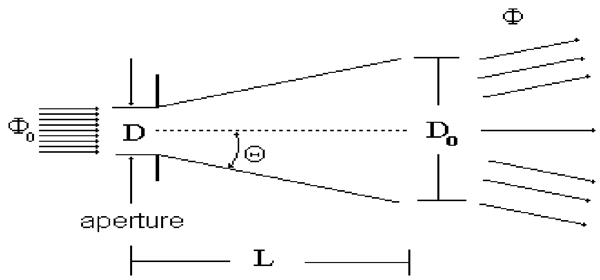

Figure 5. Geometric configuration of the divergent collimator.

The results are showed in Tab. (3), as well as the divergence angle of the beam, (θ) and the collimator ratio, L/D, considering the ^{241}Am-Be source with 185 GBq ($1.1x10^7$n/s) of activity. These values are in good agreement with available literature data (Kobayashi and Mochiki, 1996) to neutron radiography system based on mobile radioisotope neutron sources.

Table 3. Thermal flux at image plane (θ_{th}) and divergence angle of the beam (θ) for different L/D ratios.

D (cm)	L (cm)	D_0 (cm)	Φ_{th} ($n.cm^{-2}.s^{-1}$)	θ (degree)	L/D
5	16	10.5	1.3E2	10	3.2
5	30	15.4	3.8E1	10	6
5	40	19	2.1E1	10	8
2.5	16	8	1.2E2	10	6.4
2.5	30	13	1.1E1	10	12
2.5	40	16.5	6.3E0	10	16

The estimated dose equivalent rates for different values of thickness of polyethylene-boron (CH-B) shielding are shown in Fig. (6). If the rate of source emission is considered $1.1x10^7$ n/s (185 GBq of ^{241}Am-Be), the thickness of the polyethylene-boron shield must be least at 17.5 cm, so that the dose equivalent rate with respective neutrons does not exceed the limit of 25 μSv/h for human operators, established by CNEN (Comissão Nacional de Energia Nuclear, 1987) and recommended by ICRP (International Commission on Radiological Protection, 1977), at the shield surface.

The estimated dose equivalent rates arising from gamma rays generated by the interaction between the neutrons and the moderator, as well as the irradiating shield are also presented in Fig. (6). In this case, a 5cm-thickness shielding of polyethylene-boron is enough so that the dose equivalent due secondary gamma rays can be kept under the recommended limit.

4. Conclusions

The aim of modeling calculations performed using the Monte Carlo MCNP-4B code was the optimization of a portable neutron radiography system using ^{241}Am-Be source. This system should used for the industrial application to help the engineer to examine different problems and various defects in an assortment of materials and under varying circumstances.

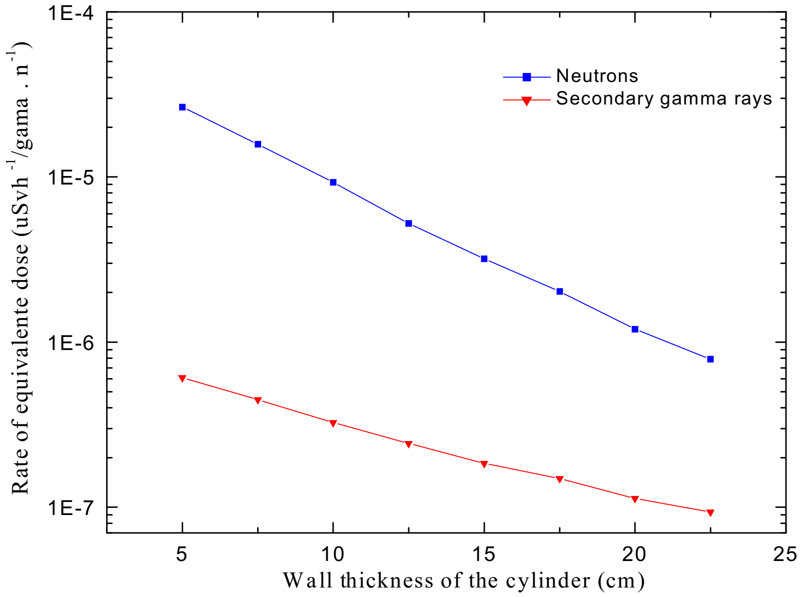

Figure 6. Dose equivalent rates due to neutrons and secondary gamma rays versus wall thickness of the polyethylene boron shielding cylinder. The estimates are normalized by source particle.

Observing Fig. (4) and the results shown in Tab. (2), to conclude that the high density polyethylene material is the most effective among them, it was chosen moderator of this work because it has the lowest thermalization factor (271 cm^2) and the maximum thermal neutrons flux at 3.25 cm away from the ^{241}Am-Be source.

According to the results presented in Tab. (3), for an inspection where sample details are intended to be observed with no restrictions with respect to neutron beam exposition time, a collimator with L/D ratio of 16 is suitable. On the order hand, for a less detailed inspection a collimator with L/D ratio of 6.4 is recommended.

Figure (6) shows that polyethylene-boron has good properties to shield the neutron radiography system proposed in this work. If a neutron source with emission rate of $1.1x10^7$ n/s is considered, the estimated shielding thickness must be at least 17.5 cm, so that the dose equivalent rate with respective neutrons and gamma rays generated by the interaction between the neutrons and moderator at the surface, does not pass the limit recommended by CNEN of 25 μSv/h for human operators. The results suggest that the optimization of the shielding is necessary and relevant, not to economic matters, but it is very important to the radiation protection.

5. References

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