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A transportable neutron radiography system

J. G. Fantidis · G. E. Nicolaou · N. F. Tsagas

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Abstract A transportable neutron radiography system, incorporating a 50 mg ^{252}Cf source, has been simulated using the MCNPX code. The materials considered were compatible with the European Union Directive on 'Restriction of Hazardous Substances' (RoHS) 2002/95/EC, hence excluding the use of cadmium and lead. The design was optimized with respect to neutron moderation, shielding and collimation. High density polyethylene was chosen as the material for moderator and also shielding, which was further enhanced with layers of bismuth and borated polyethylene. Variable values for the collimator ratio were calculated. With suitable aperture and collimator design it was possible to optimize the neutron radiography parameters. Beam filters also were treated in order to improve the results. The proposed system has been considered with a wide range of radiography parameters, which are comparable with neutron radiography facilities from low power reactors.

Keywords Neutron radiography · MCNP · Thermal neutrons · Shielding material · Gadolinium · RoHS Directive

Introduction

Neutron radiography (NR) is a technique widely used for the non-destructive testing (NDT) of objects in industry and medicine [1]. The technique is frequently used either on its own or as complementary to photon-based imaging, particularly when light elements and different isotopes of an element are present in the object. A variety of neutron beams are available for NR, with their choice been a compromise between beam intensity and transportability for in situ testing. Nuclear reactors and accelerator-driven neutron sources provide high intensity neutron beams at the expense of transportability. On the contrary, a range of commercially available isotopic neutron sources can be easily incorporated in transportable radiography units, at the expense of beam intensity. These neutron sources offer the possibility to enlarge the range of applications of NR.

In this work, a transportable unit for radiography, based on the use of thermal neutrons from a ^{252}Cf isotopic neutron source incorporated within it, has been simulated using the MCNPX Monte Carlo code [2]. The aim is to optimize the design of the system in terms of its moderator, collimator and shielding, rendering it suitable for quality non-destructive testing, while ensuring appropriate radiation protection and safety standards for the personnel in the vicinity of the unit. The materials considered in the design of the unit have been chosen according to article 4 of the RoHS Directive 2002/95/EC. Hence lead, mercury, cadmium, hexavalent chromium, polybrominated biphenyls (PBB) and polybrominated diphenyl ethers (PBDE) have been excluded [3].

Materials and methods

The top view of the neutron radiography unit simulated in this work is shown in Fig. 1. Effectively, it comprises: (1) a cylindrical irradiation unit, with 100 cm height and 50 cm radius, made of high density-polyethylene (HD-PE); (2) a neutron source placed half way along the cylinder, for the irradiation of the object been analyzed; the source is situated behind a conic divergent collimator (3), at a distance,

J. G. Fantidis · G. E. Nicolaou (✉) · N. F. Tsagas
Laboratory of Nuclear Technology, School of Engineering, Democritus University of Thrace, Xanthi, Greece
e-mail: nicolaou@ee.duth.gr

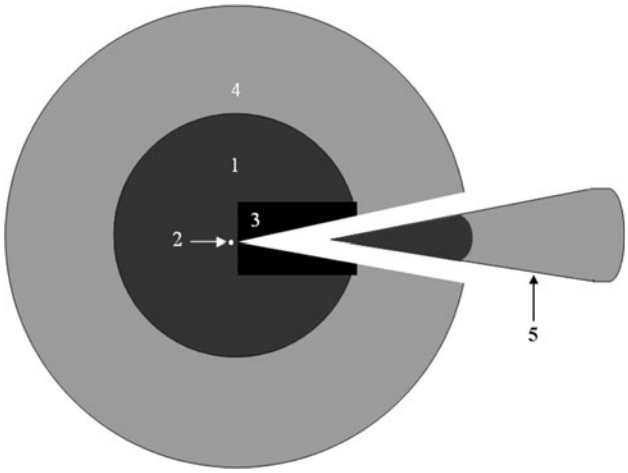

Fig. 1 Geometric configuration of the irradiating system (not in scale)

which permits good moderation of the neutrons, as this would be established in the optimization steps; the collimator is 50 cm in length, running on a radius halfway along the cylinder and providing a path for the neutrons from the source towards the object been imaged; (4) a combination of layers of different materials surround the unit providing a 50 cm shielding which is necessary for radiation protection purposes. When the unit is in use, the conic shielding sector (5) is removed to allow placing the object analyzed in front of the conic collimator.

A ^{252}Cf neutron source is considered, with an isotropic emission of 2.314 × 10^6 neutrons s^{-1} per μg of ^{252}Cf. The spectrum of the emitted neutrons extends up to 10 MeV with a mean energy at 2.3 MeV and modeled as a Watt fission spectrum using the coefficients provided by the MCNPX code. Further to the neutron emission, ^{252}Cf emits 1.3 × 10^7 photons s^{-1} per μg with a mean energy of 0.8 MeV [4]. The advantages of ^{252}Cf neutron source for radiography are: (a) low cost per neutron yield, (b) the average energy from neutrons is 2.3 MeV allows NR using, besides the thermal neutrons, other neutron energies, (c) small size, enabling the source to be placed in a transportable system, (d) peak thermal flux several times greater than the fluxes obtained with (γ, n) sources of the same total fast neutron yield, (e) minimal gamma ray background, (f) medium investment cost of the complete unit, and (g) it does not require license for use. The Department of Energy USA has a loan program to provide intense neutron sources at significant discounts to government agencies and to universities for educational, research, and medical applications. Some international loans have also been made on a case-by-case basis [5].

The design steps followed, aimed to optimise the moderator, collimator and shielding. The design considered the RoHS Directive, hence excluding the use of the materials Cd (cadmium) and Pb (lead) for the collimator and shielding. These would be replaced instead by Gd (gadolinium) and commercial Pb-free materials respectively. The use of Gd would result to the reduction of the gamma radiation background on an imaging device, due to the absence of the 0.558 MeV gamma rays emitted by Cd.

In the case of thermal NR, the first step in the design of the unit is the choice of the moderator material needed to thermalise the neutrons from the source. The moderator should provide the largest possible flux of thermal neutrons which would be collimated towards the object. The candidate materials considered (Table 1), should thermalise the neutrons in a small number of collisions, while not absorbing them to a great extent. Hence, it should have a considerable scattering cross section (Σ_s) and a reduced absorption cross section (Σ_a). A meaningful parameter, which characterizes the moderator, is the thermalisation factor (TF) defined by Barton [6]:

$$\mathrm{TF}\,(\mathrm{cm}^2) = \frac{\text{Fast neutron yield for a given source }(\mathrm{n\,s^{-1}})}{\text{Peak thermal flux in the moderator }(\mathrm{n\,cm^{-2}\,s^{-1}})} \tag{1}$$

The best moderator is usually the one with a small value for TF [7].

The collimator in a radiography system, would affect the quality of the image for a given radiation source type. The material used in the design of the collimator should direct a high intensity of neutrons towards the object. Furthermore, it should prevent stray and scattered neutrons from reaching the object through absorbing them, hence improving the unsharpness of its image. In this respect, the lining of the collimator, towards reducing the scattering of neutrons within it, is particularly important and should be made of a neutron absorbing material [8]. A characteristic parameter of a collimator that defines the degree of divergence of the neutron beam is the collimator ratio (L/D), where L is the length of the collimator and D is the diameter of the entrance aperture. The quality of the NR imaging, for a given design of the collimator, is governed, mainly by the ratio (L/D) and the associated equations

$$\phi_i = \frac{\phi_\alpha}{16\left(\frac{L_s}{D}\right)^2} \tag{2}$$

and

$$u_g = L_f \frac{D}{L_s} \tag{3}$$

where:

L_f is the image surface to object distance,
L_s is the source to object distance
D is the inlet aperture diameter,
ϕ_i is the neutron flux at the image plane,

Table 1 Moderator materials investigated and thermalisation characteristics

Material	Density (g cm^{-3})	Depth (cm)	TF (cm^2)	Thermal neutron flux[a]
Water	1.00	1.5	102	9.75E−3
Heavy water	1.10	1.75	341	2.93E−3
Paraffin	0.89	1.5	76	1.31E−2
Zirconium hydride	5.61	1.5	78	1.27E−2
High density polyethylene	0.98	1.5	63	1.60E−2
Polyethylene	0.91	1.5	71	1.41E−2

[a] n cm^{-2} s^{-1} per starting neutron

ϕ_α is the neutron flux at the aperture,
u_g is the geometric unsharpness.

Furthermore, the beam divergence is a significant measure of the usefulness of the beam near its periphery. If the neutron beam diverges very rapidly, then the outer portion of the images produced would suffer a significant distortion. The half-angle of the beam divergence (θ) is given by [8]

$$\theta = \tan^{-1}\left(\frac{I}{2L}\right) \tag{4}$$

where I and L are the maximum dimension of the image plane and the length of the collimator. The imaging quality of a system could be further characterized by the Thermal Neutron Content (TNC), which describes the number of thermal neutrons within the neutron beam

$$\mathrm{TNC} = \frac{\text{Thermal neutron flux}}{\text{Total neutron flux}} \tag{5}$$

and the relative intensities of the neutron (n) and the photon (γ) components of the beam, which typically should be

$$\frac{n}{\gamma} > 10^4\,\mathrm{n\,cm^{-2}\,mSv^{-1}} \tag{6}$$

The gamma radiation present in the imaging system would be due to possible photon emissions directly by the source and (n, γ) reactions within the unit and the object. Considering that the ^{252}Cf is associated with a photon spectrum, a filter is used to minimize the reduction in the contrast for the direct NR method due to these photons (Fig. 2).

Finally, the necessary shielding for radiation protection purposes was optimized using the MCNPX 2.5.0 code. Materials containing Pb were excluded from the design according to the EU Directive 2002/95/EC. Hence, a range of materials were considered (Table 2), which would provide effective shielding while still rendering the unit transportable.

Borated polyethylene (CH-B) is polyethylene with 5% boron and is widely used in neutron shielding applications because of its good nuclear and physical characteristics. Krafton-HB is a synthetic resin with boron, developed as an advanced shielding material for fast breeder reactors. Premadex is based on an organolithium salt used in the construction of lightweight shielding containers for the transport of highly active isotopic neutron sources such as ^{241}Am/Be and ^{252}Cf. EnviroShield is a homogeneous mixture of polymer polyethylene with 80% elemental bismuth homogeneously distributed within it. SWX-262 Gamma Putty is a gamma shielding free of lead, containing 90% bismuth uniformly distributed throughout. The 304 Stainless Steel is a good shielding material for gamma rays. Bismuth (Bi) is an excellent material for gamma-ray filtering. These materials were chosen because they are commonly used in 'generic' shielding problems.

The shielding material placed outside the HD-PE cylinder, comprise different materials with variable thicknesses, bearing in mind the transportability of the unit. The Dose Equivalent Rates (DER) was calculated on the external surface of the shield at the height of the source, where the dose has the maximum value.

Results and discussion

The efficiency of different materials to moderate neutrons to thermal energies was examined through the calculation of the thickness necessary to achieve the best thermal neutron flux and the lower parameter TF. The materials considered were light and heavy water, paraffin, zirconium hydride, high-density polyethylene (HD-PE) and polyethylene. The thermal neutron flux was calculated with the MCNPX Monte Carlo code, using the F5:N tally. The tally gives the neutron flux at a point detector in neutrons cm^{-2} per starting neutron. Cutoff (NPS) values up to 3×10^7 histories were considered yielding an accuracy of <2% in the calculations. An energy boundary of 0.01–0.3 eV was used to score the thermal neutron flux [9].

The trend of the variation of the thermal neutron flux, in terms of the moderator thickness, in the case of paraffin, HD-PE, polyethylene, zirconium hydride and light water is similar since they possess approximately the same number of hydrogen atoms per unit volume. In these materials, the flux decreases very rapidly with increasing moderating thickness. In the case of heavy water moderator, a smoother decay is observed due to the lower absorption of the thermal neutrons by the heavy water. The thickness at which the peak thermal neutron flux occurs and the TF parameters are shown in Table 1 for the different materials considered. The material chosen as moderator is the high density polyethylene (HD-PE) with a thickness of 1.5 cm,

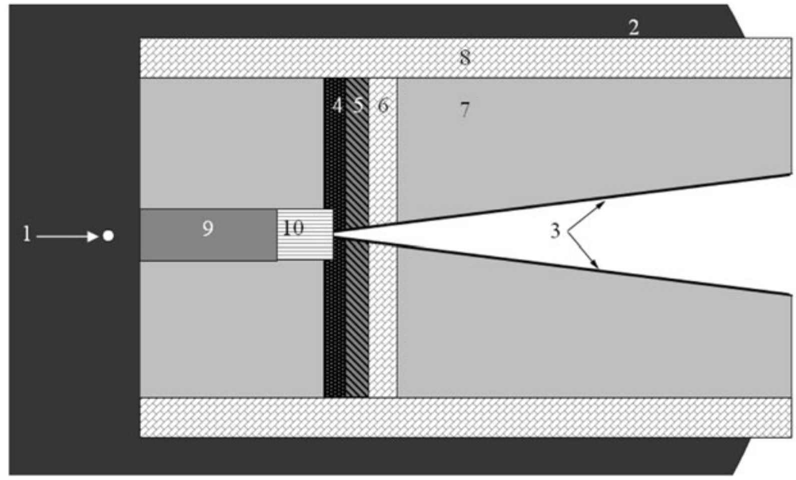

Fig. 2 The aperture geometry and the collimator with the two filters in front of them (not in scale)

Table 2 Shielding compositions in mass fraction

Element	Shielding materials						
	Krafton-HB	Borated-polyethylene	SWX-262	Premadex	SUS304	Enviroshield	Bismuth
H	0.1066	0.116	0.0144	0.114		0.028	
C	0.7529	0.612		0.474	0.0066	0.165	
O	0.1069	0.222	0.0866	0.399			
Bi			0.9			0.807	1.000
B	0.0078	0.05					
Li				0.013			
Si	0.0038						
N	0.022						
Cr					0.18		
Ni					0.08		
Fe					0.7394		
Density (g cm^{-3})	1.08	0.94	3.8	1	7.85	3	9.75

yielding the highest thermal neutron flux and at the same times the lowest TF.

The ^{252}Cf source, further to neutrons emits also gamma rays with a mean energy of 0.8 MeV [4]. Furthermore, the 50 cm thick HD-PE moderator has high hydrogen content, rendering it an effective neutron shield but at the same time a strong emitter of secondary 2.2 MeV gamma-rays from the capture of thermal neutrons by hydrogen. These capture gamma-rays can be minimized by adding boron or lithium in the HD-PE. This would reduce the dose, despite the presence of the capture 0.478 MeV gamma-rays resulting from capture of neutrons by boron. The total DER would contain three components, notably from the neutrons (DER1), the gamma-rays emitted by the source (DER2) and induced gamma-rays from the interaction of the neutrons and the moderator material (DER3) [10].

The shielding was designed for a 50 mg ^{252}Cf source, taking into consideration: the RoHS directive, the weight and dimensions which would render the unit transportable, the occupational dose limit of 25 μSv h^{-1} by ICRP-26 [11] and different combinations of the shielding materials shown in Table 2. The total dose (Sv h^{-1}) was calculated with the MCNPX Monte Carlo code, using the F4, and Fm4 tallies combined with the DE and DF cards. The tallies describe the neutron flux within a cell, while the two cards convert the absorbed dose to equivalent dose. The materials based on the polyethylene matrix (Krafton-Hb, CH-B and Premadex) were effective in shielding neutrons but ineffective in shielding the gamma-rays. Calculations with the MCNPX code have shown that thickness of at least 35, 33, and 37 cm are required for Krafton-HB, CH-B and Premadex respectively in order to keep DER1 within the recommended limit. Furthermore, the CH-B is a better solution for neutron shielding.

The total DER estimates, obtained for combinations of different layers of the materials shown in Table 2, are

Table 3 Estimates of the dose rate and weight for different shielding configurations

	Shielding materials thickness (cm)					Weight (kg)	Dose rate (μSv h^{-1})			
	CH-B	SUS-304	Enviroshield	SWX-262	Bi		DER1	DER2	DER3	Total
Layer1			55			19450	1.36	6.51	10.15	18.02
Layer1					15	13450	4.82	2.52	7.83	15.17
Layer2	35									
Layer1					10	13650	4.04	10.05	8.08	22.17
Layer2		5								
Layer3				5						
Layer4	30									
Layer1					10	13900	7.28	8.18	8.68	24.14
Layer2				15						
Layer3	25									
Layer1					10	13950	3.59	11.51	9.68	24.78
Layer2		5								
Layer3			5							
Layer4	30									
Layer1					10	14850	5.53	7.15	7.27	19.95
Layer2		10								
Layer3	25									
Layer1		15				16050	2.10	13.01	8.51	23.62
Layer2			20							
Layer3	15									

given in Table 3. Thus, the necessary shielding surrounding the unit comprises a layer of 15 cm thickness of Bi on the inside and a 35 cm thickness layer of CH-B on the outside.

The design of the collimator, aiming to reduce the gamma component while improving the *n* component at the object, comprises a combination of several different materials (Fig. 2). It would be desirable neutrons entering in the collimator with the direction of the exit window; otherwise, they should be captured by the walls of the collimator.

The proposed collimator has a length (L) of 50 cm, a diameter of the collimator aperture next to the image plane (D_0) of 16 cm and divergence angle (θ) of the beam of 9°. The distance (L_f) between the object and the imaging detector is considered at 0.5 cm [8]. The inlet aperture of the collimator is situated 1.5 cm in front of the source (1), which corresponds to the position that a peak thermal neutron flux is achieved in the HD-PE moderator (2). A 0.5 mm-thick layer of gadolinium covers the internal walls of the collimator (3). The aperture is a combination of three materials: a 0.5 cm-thick layer of gadolinium (4), ensuring that only the neutrons from the source arrive at the object; a 0.1 cm-thick indium filter (5) which captures unwanted epithermal neutrons preventing them from entering the neutron beam towards the object; and, a 2.4 cm-thick layer of bismuth (6) used to prevent gamma-rays generated within the other materials from arriving at the object. The borated polyethylene (7) was chosen among the available materials as a filling material in the collimator, since it provides a good shielding against stray neutrons. Bismuth with 1 cm-thickness was chosen as the collimator casing (8).

Single rectangular sapphire (Al_2O_3) and bismuth crystals of variable thicknesses were chosen as fast neutron and gamma-ray filters respectively [7, 12]: the former with dimensions of 1.5 cm height and 1.5 cm width (9); and, the latter with dimensions of 1.5 cm height and 1.5 cm width (10). High quality single-sapphire crystals offer at room temperature a better fast neutron filtering than silicon and quartz even when they are cooled by liquid nitrogen [13]. In addition, the transmission properties of sapphire are not altered by irradiation even after years within the neutron beam tube of a reactor [14]. Bismuth is commonly used for gamma-ray filtration, as it has a lower neutron-attenuation coefficient than lead, while nearly identical gamma-ray attenuation. Furthermore, bismuth single-crystal filter is often used in place of an equiaxed polycrystal resulting in an improved performance.

The calculated NR parameters, using the collimator described previously, are shown in Table 4 for different values of D. Neutron and gamma fluxes were calculated for the ^{252}Cf neutron source spectrum, with the aid of the MCNPX code using the F2 tally. The tally gives the

Table 4 The NR calculated parameters using the proposal aperture design and beam filters

D (cm)	*L/D*	U_g (cm)	Crystal filter (cm)		f_{th}	*t* (min)	n/γ (n cm^{-2} mSv^{-1})	TNC (%)
			Al_2O_3	Bi				
1	50	0.01	0	0	1.57E+4	10.6	9.71E+3	0.51
1	50	0.01	4	1	1.24E+4	13.5	1.40E+4	1.77
1	50	0.01	9	1	1.04E+4	16.0	1.77E+4	5.55
1	50	0.01	13	2	7.98E+3	20.9	1.76E+4	11.94
1	50	0.01	17	3	6.28E+3	26.5	1.41E+4	30.06
0.5	100	0.005	0	0	4.18E+3	39.9	9.92E+3	0.15
0.5	100	0.005	4	1	3.31E+3	50.3	1.54E+4	0.67
0.5	100	0.005	9	1	2.76E+3	60.3	1.53E+4	1.97
0.33	150	0.0033	0	0	1.89E+3	88.4	1.16E+4	0.08
0.33	150	0.0033	4	1	1.50E+3	111.1	1.25E+4	0.39

neutron flux averaged over a surface in neutrons cm^{-2} per starting neutron. Calculations were carried out with NPS $= 3 \times 10^7$ histories yielding an accuracy $< 2\%$ in the calculations. The thermal neutron flux at the object varies from 1.5×10^3 up to 1.6×10^4 n cm^{-2} s^{-1} for the 50 mg ^{252}Cf source, depending on the (*L/D*). The flux is comparable with a low power research reactor [15]. The TNC varies from 0.08 to 30.06% and the (n/γ) range from 9.71×10^3 up to 1.77×10^4 n cm^{-2} mSv^{-1}.

In each (*L/D*) case, the presence of the two crystal filters has reduced the gamma radiation parameter (γ) while increased the (n/γ) and TNC ones. Good-quality thermal neutron images require exposures of the order of 10^7 n cm^{-2} [16], with the exposure time being proportional to the thermal neutron flux. The exposure time has been calculated for the 50 mg ^{252}Cf source emitting 1.157×10^{11} n s^{-1}. In the case of $L/D = 50$ the exposure time varies between 10.6 and 26.5 min. Higher *L/D* values would require higher exposure time: in the cases of $L/D = 100$ and 150, exposure times in the range 40–60 and 88–111 min would be required respectively.

Conclusions

A transportable unit comprising a ^{252}Cf neutron source has been simulated, for radiography purposes, using the MCNPX Monte Carlo code. The materials considered, for the design of the system, were chosen according to the EU Directive 2002/95/EC, hence excluding lead and cadmium. The system was designed under the constraint that the *DER* should remain below the annual occupational dose limit. The design was optimized with respect to neutron moderation, shielding and collimation. High density polyethylene was chosen as the material for moderator and also shielding. The latter was further enhanced with layers of bismuth and borated polyethylene on the outside. This combination of materials rendered the irradiation unit lighter in weight and effective in reducing the dose to its surrounding. A conic divergent collimator was considered with gadolinium covering its internal walls. The use of crystal sapphire and bismuth within the collimator has led to improved parameters of the NR. According to the results obtained, the proposed system has a wide range of values for the parameters characterising the neutron radiography resulting in radiographs of variable quality. The (*L/D*) and (n/γ) values are comparable to the NR facilities based on low power research reactors.

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