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Designing a Transportable Neutron Radiography System

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ABSTRACT

Aspects of a neutron radiography setup, including the energy and geometry of the neutron source, the material choices and geometry of the moderation system, and the geometry of the imaging section and of the shielding system, via Monte Carlo methods are described. We evaluated the practicality of a transportable neutron radiography system. Calculations of neutron fluxes and energy distributions due to an americium-beryllium neutron source were performed using MCNP (Monte Carlo N-Particle) software package. The result shows that the safe source activity level is in the order of micro-Ci. This demonstrates the challenges in constructing a practical, efficient, and safe radiography system.

Key words: neutron, radiography, simulation, detector, radiation, Monte Carlo

INTRODUCTION

Radiography literally means making pictures by radiation. As the neutral component of the atomic nucleus, the neutron is a highly penetrating particle. We thus can use the neutron as a probe of material compositions of large sizes. Neutron radiography is a unique imaging technique which allows us to visually examine inner structures of materials. The process is non-destructive to the material. One of the important precautions is the possibility that the object might become activated and radioactive.

Neutron radiography is similar to x-ray radiography in the sense that, in both methods we let a radiation beam pass through the sample and record images on films. Neutrons, however, can image the light elements and pass through heavy elements better than x-rays. Neutrons also react differently to isotopes, while x-rays cannot distinguish between them. Moreover, neutrons usually are able to penetrate thicker objects, assuming that the available flux is sufficient. Therefore, neutron radiography has been widely applied as a quality assurance technique. Among tested samples are nuclear fuel materials, chemical explosives, electronic components, and plants.

Another difference between x-ray and neutron radiography is in the image creation step. Most photographic films consist of an emulsion of a silver halide (silver and halogen compound), which are sensitive to light and other electromagnetic radiation; therefore the films react directly to x-rays. Neutrons, however, need to be converted into ionizing radiation such as x-rays, gamma rays, or electrons. In neutron radiography, a converter material used in conjunction with the film is needed. Usual converters are boron (emits alpha), dysprosium (gamma), indium (gamma), and gadolinium (electron) (Berger, 1975).

In this study we focused on designing the passage of the neutrons as well as the shielding.

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We used MCNP (Monte Carlo N-Particle) version 4c computer code to design a neutron radiography apparatus. MCNP allows us to experiment with a variety of different materials and their combinations. A fission reactor usually provides a good source of neutrons, due to its high flux. However, the system that we design will utilize a radioisotope, which could potentially yield a portable and more economical setup.

MATERIALS AND METHODS

In designing a radiography system, we aim to achieve the following objectives: standard safety, high flux efficiency, good image quality, and sufficient energy moderation. After the design was completed, we calculated the radiation dose rate at relevant areas to insure safety. Neutron flux was measured in the target area (sample, film, neutron converter). We assessed what source strength was required to make the apparatus practical. The image quality is one of the restricting factors in planning our collimator dimensions. The energy of the neutrons is also important. We required that the neutrons arrive at the imaging area at an appropriate energy range.

Components

The main parts of the neutron radiography system are the neutron source, a moderator, a collimator, a gamma filter, shielding, and a target area. In this study, we made an assumption that an americium-beryllium source is used. However, our model will be easily adaptable for use with any isotropic source.

In neutron radiography, the neutrons react with the converter material. Usually some kind of an absorption reaction happens and an emission of a gamma or beta ray ensues. It is this secondary radiation which creates an image on film. If we examine the cross sections of materials, we can see that neutron absorption is much easier when neutrons are traveling slowly. By slowing down the neutrons before they reach the target area, we reduce the exposure time. Moderation is done by letting neutrons collide with nuclei. The most effective moderating nucleus is hydrogen, due to its mass.

The collimator minimizes stray neutrons and ensures that the neutron beam arrives at the target area in relatively mono-directional fashion. We need to have a highly neutron-absorbing material as the collimator walls. Boron carbine (B_4C) is chosen as our collimator wall material.

The sharpness of the images obtained is usually dependent on the ratio between the distance from the neutron source (L) and the aperture diameter (D) (Von Der Hardt and Rottger, 1981) If L/D is too small, it would result in blurry images; while a ratio too large would waste away the neutron flux. To avoid diffusiveness, the ratio should not fall below 10 (Von Der Hardt and Rottger, 1981).

We require a shielding material to reduce the amount of gamma going towards the target area. For the gamma filter, we use lead (Pb) for its effectiveness in attenuating the gamma and its durability. Bismuth (Bi) may also be used as the gamma filter; it is usually more expensive than lead, however.

Lastly we require a good shielding material. Similar to the moderator material, we desire materials which have high contents of hydrogen. We can also use something which absorbs neutrons and emits non penetrating radiation, like the alphas. With these considerations, borated paraffin is a good choice. Paraffin has a large hydrogen composition to slow down the neutrons, whereas boron-10 nuclei capture neutrons very well.

Design parameters

All the components were put together via a program's input file. In the final configuration, there is a cylindrical casket, lying on its side (axis parallel to ground). There is also a conical collimator along the axis. A cone offers an advantage over a cylinder in that it does not cut out neutron flux as much. The collimator ratio (L/D) is set to 10. The particle source is near the pole of the collimator cone. The exposure area is the base of the cone. There is an additional cover behind the target.

The ^{241}Am-^{9}Be source emits neutrons according to the reactions shown below, with the neutron energies in the range of 0-10 MeV (Figure 1). ^{241}Am has a half-life for alpha decay of 432 years. ^{9}Be then goes through (alpha, n) reaction (Krane, 1988). Approximately six neutrons are released per 10^5 Am decays (Krane, 1988).

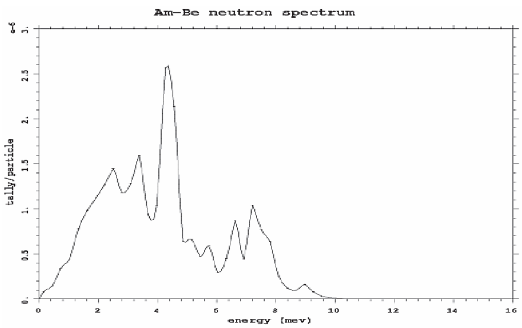

Figure 1 Energy spectrum of the neutrons from ^{241}Am-^{9}Be source.

$$^{241}\text{Am} \longrightarrow {}^{237}\text{Np} + {}^{4}\text{He} \quad (Q = 5.6\ \text{MeV})$$
$$^{9}\text{Be} + {}^{4}\text{He} \longrightarrow {}^{12}\text{C} + n \quad (Q = 5.7\ \text{MeV})$$

The source is a thin cylinder. Its radius is 1 cm and its length is 2 cm. A slab of paraffin (density 0.93 g/cm^3) is used as the moderator. A thickness of 5 cm is chosen. The lead (density 11.35 g/cm^3) used as the gamma filter is also 5 cm thick. B_4C collimator wall is 1 cm thick. The whole thing sits inside a borated paraffin shielding. The target area is designed to be a circle with a 20-cm diameter. An additional borated paraffin cover is placed behind the target area. Additionally, there is a thin lead disk near the source and a thin lead plate behind the target to further attenuate photons. A thin aluminum wall is used on the outermost part of the apparatus. Final geometry is shown in Figure 2. Overall, the target plate is 20 cm away from the source element. The number of particles started for this study is 200,000.

RESULTS AND DISCUSSION

Neutron fluxes

A key quantity that determines exposure time is the thermal neutron flux (thermal neutrons per unit area per second) at the target. Here, "thermal" means low energy that is about 0.02–

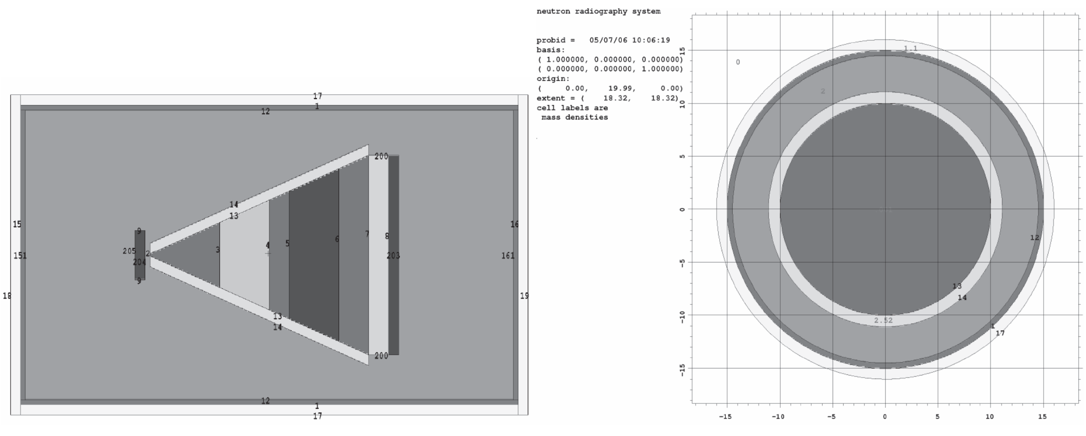

Figure 2 The design geometry: side view (left) and top view (right). On the side view figure, the source is the cylinder to the left. The target/film area is to the right. The whole setup appears as a large cylinder.

0.5 eV. The neutron flux is shown in Figure 3. We can see that it looks much different from the initial spectrum (Figure 1). After energy moderation, fast neutrons are slowed down and relative thermal neutron contribution increases.

Total neutron flux at target surface is 2.1×10^{-4} neutrons/cm^2 (± 1.6%). Thermal neutron, which we define the energy to be about 0.04 eV, flux is 5.0×10^{-6} neutrons/cm^2 (± 7.9%). The answer from MCNP is averaged per one neutron started. For a 1 Ci source, thermal flux = $5.0\times10^{-6}\times3.7\times10^{10} = 1.9\times10^5$ neutrons/cm$^2\cdot$ s. The reactor source which is normally used in doing neutron radiography gives thermal neutron flux of the order 10^5. Therefore, the flux should be sufficient if we use a 1 Ci source. Taken the one-tenth limit, the thermal neutron flux should not fall below 10^4 n cm^{-2} s^{-1} to be practical, otherwise the exposure time would be too long. Neutron flux is directly proportional to the source activity. Photon flux is shown in Figure 4. Total photon flux at target surface is 1.0×10^{-4} photons/cm^2 (± 2.8%).

Safety

A high thermal neutron flux at the target is one thing to achieve. User safety is a primary concern in performing any radiation operation. In this part, we estimate the radiation levels around the radiography system. The International Commission on Radiological Protection (ICRP) guideline says that the whole-body dose limit for a radiological worker is 5 rem^2 (or 50 mSv) per year (Eidelman *et al.*, 2004), as it is 0.1 rem/year for general public. Some organs may have higher limits, but we use the whole-body limit to be conservative. The safe limit is 2.5 mrem/hr for radiological workers and 0.05 mrem/hr for the public.

We tallied fluxes in locations surrounding the apparatus. Thin dosimetric cells, equivalent to human skin, were constructed to tally the energy deposited. MCNP flux-to-dose conversion factors change fluxes (neutrons/cm^2) to equivalent doses (rem/hr). Figure 6 shows the dose rates due to neutrons and photons.

Around the cylinder, the neutron dose rate is 2.1×10^{-9} rem/h (1.0 % error). At the cover near source, it is 2.0×10^{-9} rem/h (2.8 % error). Near target, it is 1.6×10^{-10} rem/h (8.8 % error). Photon dose rate around the cylinder is 1.1×10^{-10} rem/h (1.0 % error). At the cover near source, it is 1.1×10^{-10} rem/h (1.3 % error). Near target, it is 8.3×10^{-12} rem/h (4.8 % error). To get easy-to-use answers, we multiply the equivalent dose rates by activity (in dps). For a 1-Ci source, A = 3.7×10^{10} dps, equivalent dose rates are shown In Table 1.

Table 1 Equivalent dose rates around the shielding.

Location	Equivalent dose rate (rem/hr)	
	Neutron	Photon
Side	77.7	4.1
Near source	74.0	4.1
Near target	5.9	0.3

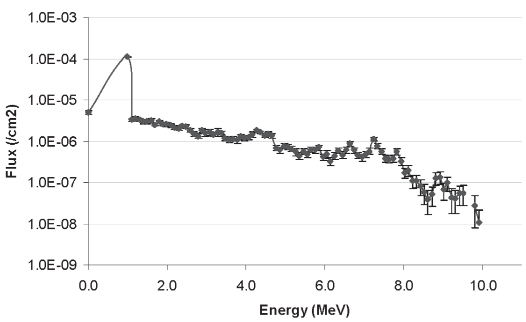

Figure 3 Neutron flux as a function of energy.

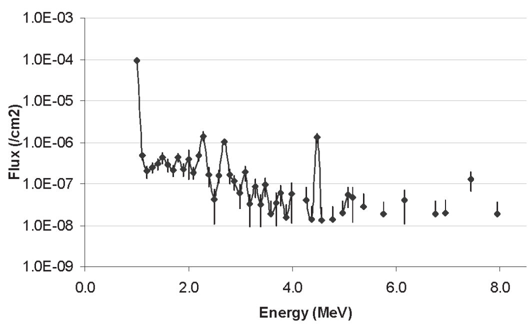

Figure 4 Photon flux as a function of energy.

The maximum dose rate is at the side. If we are to lower this figure to 2.5 mrem/hr (safe limit), the maximum activity for the source would have to be A_{max} = 1Ci × $\left(\frac{2.5\times10^{-3}}{77.7+4.1}\right)$ = 31mCi, which would yield the thermal neutron flux of $5\times10^{-6}\times31\times10^{-6}\times3.7\times10^{-10}$ = 5.7 neutrons/cm^2/s. This result shows that the compromise between safety and performance considerations yields a very low flux.

CONCLUSIONS

Neutron radiography is a useful nuclear analysis technique. Due to their mass and lack of charge, neutrons have properties that make them a unique probing tool. However, a truly portable system is a challenge because we require a bulk of material to adequately shield highly penetrating neutrons. The low thermal neutron flux found in this simulation (less than 10 neutrons/cm^2/s) is far from ideal for a robust portable radiography system. Thus, as it is utilized, the technique is largely reactor based. If traditional materials are used, the large dimensions of the instrument could prevent the portable neutron radiography system from being practical.

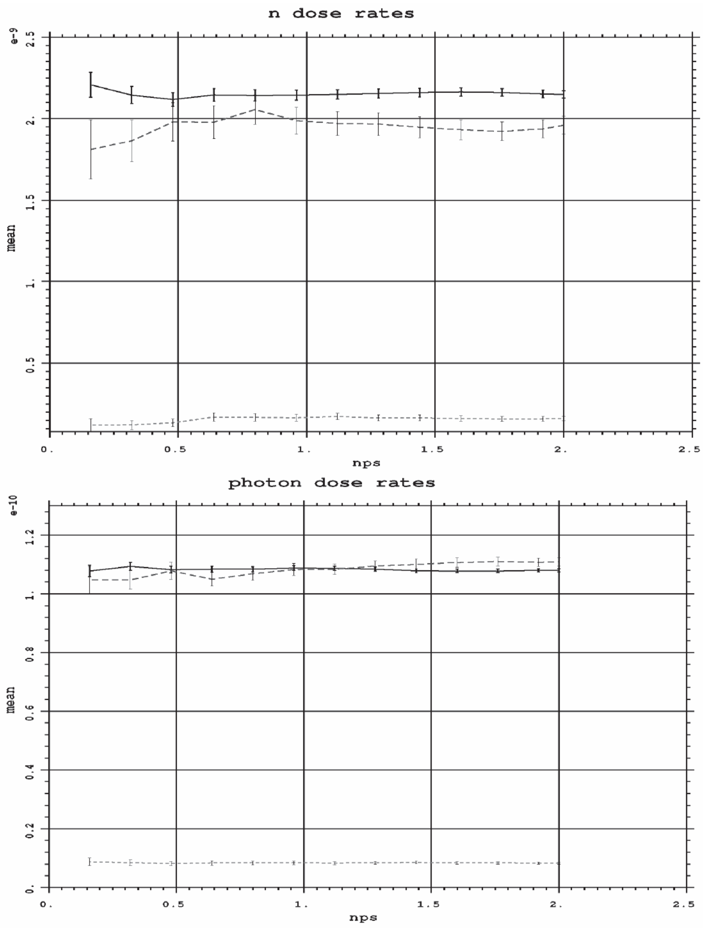

Figure 6 Equivalent dose rates (rad/hr) due to neutrons and photons in three surrounding regions. Colors: black (solid) = side, blue (dashed) = near source, red (short dashed) = near target.

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