A novel design of neutron survey instrument

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Almost all neutron survey instruments that are currently in use in the UK are based on the Andersson-Braun or Leake moderator type designs that date back to the 1960s. There has been relatively little innovation in the designs, despite the deficiencies of the dose equivalent response characteristics of these instruments. MCNP-4C2 has been used to investigate the possibility of producing a relatively light, moderator type instrument, with improved energy and angle dependence of response characteristics. The results are promising, with the fast and thermal neutron responses being matched and the over-response to intermediate energy neutrons being reduced when compared to the Andersson-Braun and Leake designs. Results for the calculated mono-energetic response characteristics are presented. These results are used to calculate the systematic errors in the reading in workplaces.

Brief Introduction to the Monte Carlo method and MCNP

For real situations it is generally not possible to calculate the radiation field at a location, or in this case the instrument response, analytically. The Monte Carlo method uses the simulation of individual particles based on the known physics and random numbers. The result is accurate provided: sufficient particles reach the region of interest, all pathways of significance are sampled, the problem can be specified precisely i.e. all geometries, materials, densities, cross-sections are specified and the physics of the problem should be well known. The Monte Carlo method is an ideal tool for optimization, where laboratory experimentation would be too expensive. Throughout this professional training year at HPA-RPD, a computer-based program called the Monte Carlo N-Particle Transport Code (MCNP) was used. MCNP is a general purpose, continuous energy, generalised geometry, time-dependant, coupled neutron/photon/electron Monte Carlo transport code. It can be used in several transport modes. For this project the neutron only mode was used. The neutron energy range is from 10^{-11} MeV to 20 MeV for all isotopes and up to 150 MeV for some isotopes. The user creates an input file that is subsequently read by MCNP. The file contains information about the problem such as geometric specification, materials and cross section evaluations.

Neutron Area Survey Instruments

Neutron area survey instruments are used to detect neutrons with a wide range of energies and directions. They are designed to have ambient dose equivalent response characteristics, $R_{H^*(10)}$, that are as independent of neutron energy and angle of incidence as possible, but given the difficulty of the problem it is unsurprising that all current designs are deficient in terms of both energy and angle dependence of response to some extent. Neutron survey instruments are used in workplaces where neutrons are present, for example nuclear reactors, fuel manufacture and fuel reprocessing. Neutrons can be dangerous to human health wherever significant doses are received and therefore workplaces where neutrons are present must be monitored. The first practical neutron survey instrument was produced in 1963 by Andersson-Braun [1,2,3]. It used a boron trifluoride (BF_3) proportional tube as a central thermal (0.0253 eV) neutron detector surrounded by cylindrical polyethylene moderating mass. Three years later John Leake published a paper on a design that uses a spherical helium-3 detector surrounded by spherical polythene moderating mass[1,4,5]. Both designs have a thermal neutron absorbing layer within the moderating mass and both designs over-respond in the intermediate energy region. MCNP-4C [6] was used to model the novel design.

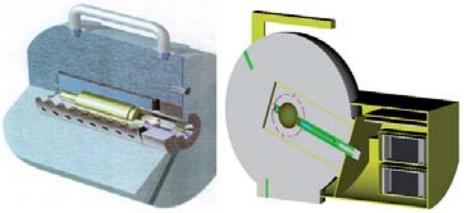

Figure 1 Modified Andersson-Braun design (left) and Leake design (right)

Neutron Physics

For descriptive purposes, the energy of neutrons is divided into three energy-ranges [7]:
Thermal (slow) neutrons: < 0.5 eV
Intermediate neutrons: 0.5 eV – 100 keV
Fast neutrons: >100 keV

Thermal neutrons are in thermal equilibrium with the surrounding matter. They have a Maxwellian velocity distribution with a mean velocity of about 2200 m s^{-1}, corresponding to an energy of 0.0253 eV at room temperature [7].

Neutrons do not produce direct ionization, as they have no charge and do not interact strongly with electrons. Neutrons can when they interact with nuclei, however, produce secondary charged particles that can cause ionization: principally this ionization is caused by elastic and inelastic scattering events but nuclear reactions such as (n,p), (n,α), (n,γ) or (n,fission) also contribute. Consequently, capture reactions, which have 1/*v* cross-sections (where *v* is neutron velocity), can contribute significantly to the dose deposited in tissue, especially in highly moderated neutron fields, primarily through the ^{1}H(n,γ) and ^{14}N(n,p) reactions.

Neutron detectors are commonly based on detecting the secondary particles produced by nuclear capture reactions. In this case we concentrate on (n,p) capture reactions. The type of detector used is a spherical ^{3}He detector.

The important reaction between a ^{3}He atom and a neutron is as follows: -

$$^3_2He + ^1_0n \rightarrow ^3_1t + ^1_1p \qquad (1)$$

The *Q*-value for this reaction is 0.764 MeV. For reactions induced by thermal neutrons, the *Q*-value of 0.764 MeV leads to oppositely directed reaction products with energies: E_p=0.573 MeV and E_t=0.191 MeV [8]. The thermal neutron cross-section for this reaction is 5330 barns and its value falls off with a 1/*v* relationship where *v* is neutron velocity. For fast neutrons the cross-section falls off continuously with increasing neutron energy. In any detector based on ^{3}He reactions several competing reactions must be considered. The most significant is simple elastic scattering of neutrons from the helium nuclei. The cross-section for elastic scattering is always larger than that for the (n,p) reaction in the fast neutron energy range (100 keV – 10 MeV), and this predominance becomes more pronounced as the neutron energy becomes larger. In addition a competing (n,d) reaction is possible at neutron energies exceeding 4.3 MeV, but the cross-section is low for energies below 10 MeV [8]. However, even for high energies the response of the detector is dominated by thermalized neutrons, because the fast neutron component of the field is attenuated and the cross section for ^{3}He(n, p) for thermal neutrons is four orders of magnitude higher than the elastic scattering cross section for fast neutrons.

The cross sections of many neutron-induced reactions potentially useful in detectors drop off rapidly with increasing neutron energy. The importance of scattering at higher energies becomes greater as the neutron can transfer an appreciable amount of energy in one collision and is therefore moderated (slowed down to a lower energy). The most efficient moderator is hydrogen. It is not practical, however, to simply surround a detector with hydrogen. A very good moderating material widely used is polyethylene (CH_2).

Dose Quantities

For a circular plane parallel source the fluence, Φ, is given by the ratio of the number of source particles, N, divided by the source area, A [9]:

$$\Phi = \frac{N}{A} \qquad (2)$$

The fluence response, R_Φ, is then given by the number of $^3He(n,p)^3H$ capture events, C, divided by the fluence:

$$R_\Phi = \frac{C}{\Phi} \qquad (3)$$

The exact description of the operational quantity for area monitoring is achieved by the definition of a geometric phantom replacing the human body and the definition of a specific point in that phantom, since dose equivalent is a dose quantity. The phantom has a mass density of 1 g cm^{-3} and a mass composition of 76.2% oxygen, 11.1% carbon, 10.1% hydrogen and 2.6% nitrogen. For area monitoring the International Commission on Radiation Measurements and Units (ICRU) recommends the use of the quantity ambient dose equivalent, $H^*(10)$, which is intended ‘to provide a conservative estimate of the effective dose’. $H^*(10)$ is defined for an expanded and aligned field in a very simple phantom, the ICRU sphere which has a diameter of 30 cm, as the dose equivalent at 10 mm depth [10]. The ambient dose equivalent can be found by using the following equation [11]:

$$H^*(10) = \int_L Q(L) D(L)\,\mathrm{d}L \qquad (4)$$

where $Q(L)$ is the quality factor for particles with linear energy transfer, L, and $D(L)$ is the absorbed dose. The method used to define $H^*(10)$ is best described using diagrams. Consider the Figure 2; this is the ICRU sphere with a point at 10 mm depth. The arrows indicate radiation fields and their size indicates energy; higher energies are indicated by big arrows and vice versa. In the sphere on the left the radiation fields are distributed and they deposit dose at different areas in the sphere. $H^*(10)$ is defined for expanded radiation fields, aligned to produce a plane-parallel beam, as shown in the right sphere. This is done using Monte Carlo calculations. The absorbed dose can then be used to calculate $H^*(10)$. On the left is the ICRU sphere with distributed radiation fields and on the right is the ICRU sphere with expanded and aligned fields. The point ‘x’ is at 10 mm depth, the depth generally used for penetrating radiation:

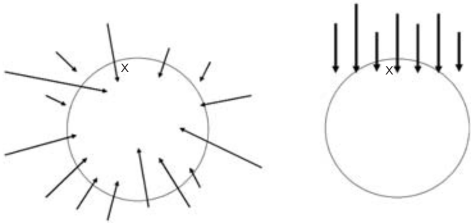

Figure 2 On the left is the ICRU sphere with a distributed radiation field and on the right is the ICRU sphere with an expanded and aligned field. The point ‘x’ is at 10 mm depth.

An ideal detector would have a constant ambient dose equivalent response, $R_{H^*(10)}$, which is basically a measure of the sensitivity of the detector and is given by:

$$R_{H^*(10)} = \frac{C}{H^*(10)} \qquad (5)$$

The ambient dose equivalent conversion coefficient, $h^*(10)$, can also be used to obtain $H^*(10)$ and is defined by [11]:

$$h^*(10) = \frac{H^*(10)}{\Phi} \qquad (6)$$

The dose equivalent response, $R_{H^*(10)}$, can then be written as:

$$R_{H^*(10)} = \frac{R_\Phi}{h^*(10)} \qquad (7)$$

For a detector to have a constant $H^*(10)$ response, the fluence response would have to rise rapidly at around 10 keV. The use of capture reactions that liberate energetic charged particles makes this difficult to engineer. The capture reactions are used primarily for their high cross-sections that allow the instrument to have high sensitivity, but their 1/v behaviour, means that the fast neutrons must be highly moderated to produce dose equivalent response. However, thermal neutrons must still penetrate the moderator and reach the central detector, or their response will be very low. Using a perforated absorber produces a balance between the two effects. For example, in this design a boron loaded polyethylene (BLP) layer is used to attenuate neutrons because the capture reaction $^{10}B(n,\alpha)^7Li$ has a very high cross-section for thermal neutrons.

Design Specifications

Over the 40 years since the first practical survey instruments were produced there has not been major innovation. Most designs are deficient to some extent: they are either heavy and a have a good fast neutron response or they are relatively light and have a good thermal response but a poor fast response. All moderator-type designs have an over-response to intermediate energy neutrons. The novel design is similar in concept to an earlier NRPB innovation [12] that uses a central ^3He detector with 6 photodiodes with ^6LiF converters located outside the boron loaded absorbing layer. That design uses the “outer detectors” to detect the thermal and intermediate energy range and an imperforated boron loaded absorbing layer to limit the response of the inner detector to fast neutrons. Whilst the energy dependence of response of this earlier design was superior to all prior designs, it was also complex and potentially expensive. The use of seven detectors instead of one and the requirement for high and low voltage power supplies limits its commercial potential. However, a project lead by the University of Lancaster is currently investigating a version of this earlier instrument for commercial viability.

The total radius of the design is 11 cm and the moderating mass weighs roughly 5 kg. This is lighter than other designs such as the Leake 0949 (weight: 6.2kg), Studsvik (10.5 kg), LB6411 (9.0 kg) and SWENDI (13.4 kg) [1].

Results and Discussions

The best data for $H^*(10)$ response are shown in Figure 3. The solid lines are the data for the novel design. The black line is for modelling using a plane parallel source on the z-axis and the red line is for an isotropic source. The dashed lines are the Leake 0949 that was shown in Figure 1. The Leake 0949 weighs roughly 6.2 kg whereas our design currently weighs roughly 5 kg. The aim is to have an $H^*(10)$ response of at least 1 nSv^{-1} at each energy especially in the thermal region. The data for our design look very promising compared to other designs; it has a good response and is light. As the energy of the neutrons rises above 10 eV so does the response, peaking at 2 MeV and then dropping again at the fast neutron range. The reason for this is because more of the fast neutrons escape from the instrument without being detected. As the moderator mass is roughly 5 kg a significant fraction of the fast neutrons are not moderated. The ambient dose equivalent response for the isotropic source in the thermal energy region is roughly 2.5 nSv^{-1} which is above the target of at least 1 nSv^{-1} as thermal neutrons are highly scattered and will hence never be unidirectional. The plane parallel ambient dose equivalent response is also above 1 nSv^{-1} in the fast neutron energy range; this is significant, as fast neutrons are more likely to be unidirectional. Our instrument is also much more sensitive than the Leake 0949.

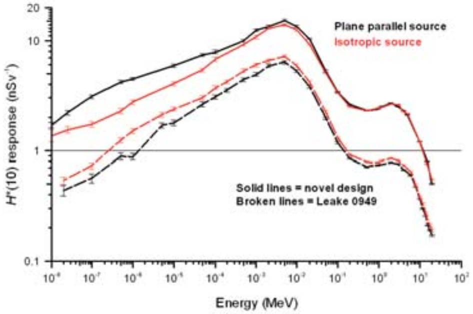

Figure 3 Calculated $H^*(10)$ response data for novel design and Leake 0949. The top two (solid) lines are data for the novel design, whereas the bottom two (broken) lines are data for the Leake design.

Comparison with other designs

We can also compare the novel design with other instruments using the $H^*(10)$ response of the calibration source for each instrument. The calibration source is ^{252}Cf for the novel design. Table 1 compares the novel design and Leake 0949 with the WENDI, SWENDI, Studsvik and LB6411 [1]. The ^{252}Cf source is a spontaneous fission source.

Comparison of sensitivity of various designs

Design	Calibration $R_{H^*(10)}$ (nSv^{-1})	Weight (kg)
Leake 0949	0.9	6.2
WENDI	2.6	13.4
SWENDI	4.0	13.4
Studsvik	1.3	10.5
LB6411	2.8	9.0
Novel Design	2.5	5.0

Conclusions

The novel design of neutron survey instrument was optimized using the Monte Carlo method. The design is lighter and smaller than most designs on the market, weighing 5 kg and measuring 11 cm in diameter and would therefore be relatively cheap to make and use. The design is also relatively sensitive, with a calibration response of 2.5 nSv^{-1}, compared to other designs such as the WENDI (weighing 13.4 kg) which has a calibration response of 2.6 nSv^{-1}. The WENDI is much heavier than the novel design yet its calibration response is almost the same. The Leake 0949 weighs 6.2 kg but its calibration response is 0.9 nSv^{-1}. The response in workplaces was also calculated, the design overestimates slightly in the thermal neutron energy range and underestimates slightly for fast neutrons in an isotropic field.

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