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Extension of CASCADE.04 to estimate neutron fluence and dose rates and its validation

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Abstract. Capability to compute neutron dose rate is introduced for the first time in the new version of the CASCADE.04 code. Two different methods, 'track length estimator' and 'collision estimator' are adapted for the estimation of neutron fluence rate needed to calculate the ambient dose rate. For the validation of the methods, neutron dose rates are experimentally measured at different locations of a 5Ci Am–Be source, shielded in Howitzer-type system and these results are compared with those estimated using (i) modified CASCADE.04.d and (ii) MCNP4A codes and it is found that the agreement is good. The paper presents details of modification and results of the comparative study.

Keywords. Neutron dose rate; neutron fluence; Monte–Carlo, CASCADE.04 code.

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1. Introduction

The neutron sources like Am+Be, Ra+Be, Po+Be etc., reactors, cosmic rays and nuclear reactions are frequently used to measure neutron interaction cross-sections by various methods such as activation and transmission and used also for the characterization of materials, detection of trace elements and in many research and developmental activities. In such activities, for the estimation of heterogeneous neutron fluxes, one has to use a large number of detectors as one detector may not respond to a wide range of energy and that needs very long time of measurement and analysis. It is also known that radiation dose rate is frequently used in health, shielding and safety experiments, space flights and in many diagnostic facilities and it may be converted into gross neutron fluence by using standard fluence to ambient dose conversion coefficients but it is very difficult to get the fine energy spectrum of neutrons using dose rate measurements. Thus, there is a need to devise methods of revealing neutron energy spectra from the economical methods of measurements such as dose measurement. It may also be added that estimation of the energy distributions is more important for radiation protection purposes compared to direct measurement of dose equivalent quantities. This is because the dose equivalent is a quantity estimated from the neutron energy spectrum through fluence to dose conversion coefficients, which can be, and have been, changed by ICRP. On the other hand, neutron energy spectra remain invariant and can be used to compute the dose equivalent in the case of any future change in fluence to dose conversion coefficients, while the reverse is not possible.

Earlier the CASCADE code [1–3] predicted the number of neutrons escaping out of an assembly and there was no way to estimate the neutron fluence (defined as number per cm^2) inside the volume of an assembly where the neutron flow may be from many directions. This problem can be solved by incorporating the 'track length estimator' and 'collision estimator' methods [4] to calculate the neutron fluence. By including the fluence to ambient dose conversion coefficients one can estimate the dose rates from thermal to 5 GeV neutron energy. For its experimental validation, a Howitzer-type experimental set-up containing 5Ci Am–Be source is used for the measurement of neutron dose rate at different radial and axial positions. For the sake of comparison of theoretical results of the modified CASCADE code another popular code MCNP4A [5] has been used for the estimation of neutron fluence and the ambient dose.

The paper is organized as follows: a brief description of the CASCADE.04 is given in §2 along with the description of 'track length estimator' and 'collision estimator' methods. Description of the experimental set-up is given in §3 along with intercomparison of calculated neutron fluences from two codes as well as comparison of dose rate with the experimental results. Conclusions are given in §4.

2. Simulation of neutron transport in CASCADE.04

The CASCADE.04 is a well-established Monte Carlo transport code for nuclear interactions of intermediate and high energy nucleons, pions and nuclei. It incorporates intranuclear cascade, pre-equilibrium, evaporation and fission models to simulate the details of an interaction in a target in the energy range starting from the projectile energy to the stopping energy or till a particle escapes out of an assembly. In the case of thick targets, electronic energy loss, spallation processes and other nuclear reactions of primary as well as secondary particles have been taken into account. In the code, cross-sections of the hadron–nucleus collision are calculated based on the compilations of the experimental data [6,7]. The code uses 26-group neutron transport cross-sections [8] below 10.5 MeV. This library contains the multigroup data from 14.5 MeV down to thermal energy, for the microscopic elastic, inelastic (n, n'), $(n, 2n)$, fission and capture cross-sections, and the average cosine (μ) for the angle of elastic scattering also. In the case of resonances, two or three equal energy sub-groups are made. Group-wise transition probabilities for the inelastic cross-sections are given and used for the selection of the next energy group. Fission neutron spectrum and mean neutron multiplicities for fissile nuclei are also given.

CASCADE.04 code is based on the well-described axially symmetric geometry. Calculation of the surface boundary and surface crossing is described in detail in ref. [9] and other references therein. Interactions of neutrons are simulated using the Monte–Carlo method in the following steps: (i) identification of the initial zone number and interaction site of the neutron, (ii) selection of the collision nuclide, and (iii) type of interaction, i.e. elastic, inelastic, capture and fission.

Angular distribution of elastically scattered neutron is considered to be isotropic in the centre of mass system, and energy is calculated using two-body collision kinematics. The free gas thermal treatment is considered for the neutron interaction (energy <4 eV) with light nuclei which can be switched off by defining higher energy cut-off. In the case of capture, we have considered that the neutron has been lost after the capture event has taken place and a new neutron has to be selected for the transport. In the modified CASCADE.04.d code survival probability has been considered for large systems and specially, for the light moderating media. In the case of inelastic interaction, the energy and angles have been simulated using scattering matrices and fission neutron multiplicity has been taken from the tabulated values given in the library [8].

In this way, the CASCADE.04 code is able to estimate the current, i.e. number of particles e.g. neutron, charge particles, and mesons. The weights of the particles are stored to estimate this current tally. The units of the tally are derived from the units of the source. If the source has units of particles per unit time the current tally will also be particles per unit time. Fluence tally is introduced first time in the code which is defined in units of particles/cm^2, but it represents the flux (particles/cm^2/s) if the source units are particles per unit time. The fluence tally can be made per unit energy by dividing the numbers in each energy bin with the width of the energy bin. The 'collision estimator' and the 'track length estimator' methods are introduced in the code to estimate the fluence. Details about these estimators can be found in ref. [4].

2.1 *Method of collision estimator*

The collision estimator method [4] is based on the probability of collision. We have considered spheres around the point of interest where fluence is to be calculated. Fluence (n/cm^2) for this sphere is given by

$$\begin{aligned}\varphi(\text{neutron/cm}^2) &= \frac{\text{number of collisions}}{\Sigma_t V} \\ &= \frac{1}{V}\int \mathrm{d}E \int \mathrm{d}t \int \mathrm{d}V \int \mathrm{d}\Omega \Psi(\vec{r}, \hat{\Omega}, E, t), \qquad (1)\end{aligned}$$

where V is the volume of the ring or sphere at the point, $\vec{r}$, $\hat{\Omega}$, E, t are particle position vector, direction vector, energy, and time, respectively. In this case we have counted the number of collisions weighted over mean free path, $1/\Sigma_t$ corresponding to different energy neutrons and divided by the volume of that detector (ring or sphere). Here, Σ_t is the macroscopic cross-section.

2.2 *Method of track length estimator*

The track length estimator adds the lengths of the neutron tracks in the ring or the sphere and one can define the fluence as

$$\varphi(\text{neutron/cm}^2) = \frac{\text{length of the tracks}}{V} = \frac{1}{V}\int \mathrm{d}E \int \mathrm{d}t \int \mathrm{d}V \int \mathrm{d}\Omega \Psi(\vec{r}, \hat{\Omega}, E, t). \quad (2)$$

The surface fluence tally is also introduced in similar way as described in ref. [4] and given in the following equation as

$$\varphi(\text{neutron/cm}^2) = \frac{\text{number of particles}}{|\mu| A} = \frac{1}{A}\int \mathrm{d}E \int \mathrm{d}t \int \mathrm{d}V \int \mathrm{d}\Omega \Psi(\vec{r}, \hat{\Omega}, E, t), \quad (3)$$

where A is the area of the hypothetical detector used for counting particles and μ is the cosine angle between normal to the surface and particle trajectory. If more particles travel in the volume more will be the interactions and consequently higher will be the fluence. This method can be applied in the case of vacuum also where no collision takes place but finite number of tracks can be collected in the volume. Thus, the track length estimator may be considered superior to the collision estimator in the case of vacuum. The track length estimator gives better estimation than the collision estimator for small number of events also because the number of un-collided tracks may then become relatively substantial. The track length estimator is similar to the collision estimator for small thickness of the detector where length of the tracks = number of collisions/Σ_t (here, mean free path $\lambda = 1/\Sigma_t$).

3. Experimental set-up and comparison of results

The neutron dose rate measurements are carried out using a 5Ci Am–Be cylindrical (diameter = 22.4 mm, height = 31 mm) source [10] available at the University of Rajasthan which was supplied by National Physical Laboratory, Teddington, England. The spectrum of the neutron source depends on the design and amount of AmO_2+Be. The neutron spectrum is given in figure 1 which shows several peaks at 3.3, 4.5 MeV etc. including a significant number of low energy neutrons below 1 MeV. The source is situated at the centre of a cylindrical tank of size $D \times L = 90 \times 92$ cm^2 filled with paraffin (density = 0.7 g/cm^3) all around the source. There are eight vertical holes at radial distances, r = 16.5, 17.5, 19, 20.5, 24, 26, 27.5 and 29 cm from the centre of the source, and ninth hole located at the centre is used for the experiments in close proximity of the source (see figure 2). The diameter of eight holes is 2 cm each and the diameter of the central hole is 5.3 cm. Neutron transmission from the central hole is blocked using an Al-cylinder filled with paraffin when there is no experiment. The neutron doses inside the

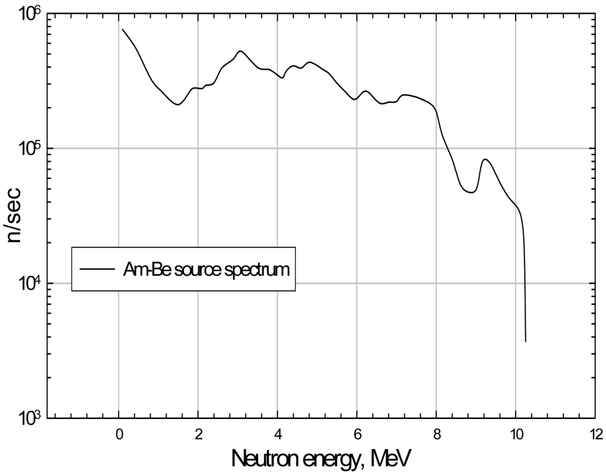

Figure 1. Intensity of neutrons from the 5Ci Am–Be bare source.

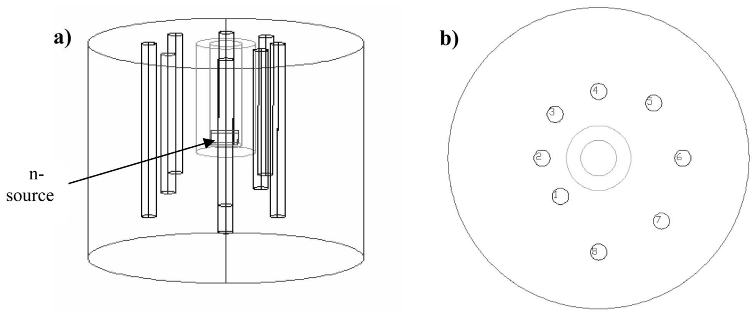

Figure 2. Schematic of (**a**) container of the Am+Be source with one central hole and eight probe holes and (**b**) top view of the container/tank.

above-mentioned eight holes are measured using 5-G-1 BF_3 proportional detector (Rem meter).

We have done calculations of the fluence using the 'collision estimators' method in the CASCADE.04.d and MCNP4A codes for isotropic neutron source of 1 MeV situated at the centre of a hydrogen medium (cylinder of $D \times L = 90 \times 92$ cm^2 with assumed density 0.7 g/cm^3) where probability of inelastic events is zero. The neutrons are found thermalized after about 18 collisions in both the codes (CASCADE.04.d and MCNP). Later on, we performed similar calculations for 5Ci Am–Be isotropic neutron source situated at the centre of the paraffin cylinder as described above and obtained neutron fluences at different radial distances from both the codes. The results of fluence per unit incident neutron are compared in figure 3. The calculated values from CASCADE.04.d code are close to that of the values obtained from MCNP code at neutron energy above 4 MeV but have difference up to a factor of 3 below this energy range. The reason of the discrepancy is conjectured to be the treatment of angular distribution of the scattered neutrons in CASCADE.04.d

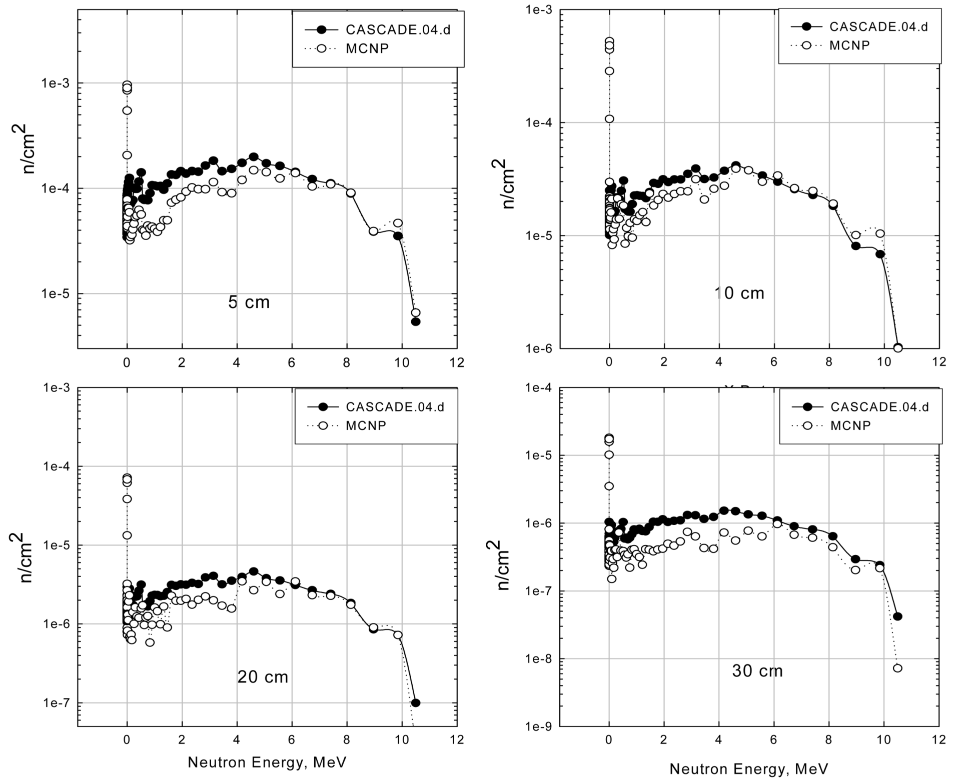

Figure 3. Comparison of neutron fluence (n/cm^2) at different radial distances from the centre of the paraffin cylinder predicted by CASCADE.04.d and MCNP for the 5Ci Am–Be neutron source for one incident neutron.

where the code uses the first-order Legendré polynomial only. This will be confirmed and modified in our future works where we plan to include the ENDF point data library with higher Legendré moments for the angular distribution.

Similar calculations have been done using the 'track length estimator' method also and the results are the same. We have used the weight cut-off variance reduction technique and Russian roulette is applied. Energy cut-off (0.0253 eV) has also been applied for faster calculations and to reduce the variance.

Geometrical representation of the paraffin cylindrical shield was realized in MCNP4A using the CZ and PZ geometry card options available in the code. The shield medium was split into several concentric cylindrical shells in order to improve neutron transport towards the outer edges of the shield, thereby improving the collision sampling efficiency and hence the precision of the tally scores. The main MCNP tally options utilized for this problem are the F2 (surface current) and the F5 (point detector estimate). Neutron cross-section library containing data from the ENDF/B-VI (evaluated nuclear data file) is included with the MCNP and used in the present work. All neutron reactions given in the cross-section library (such as ENDF/B-VI) are accounted. Thermal neutrons are described by both the free

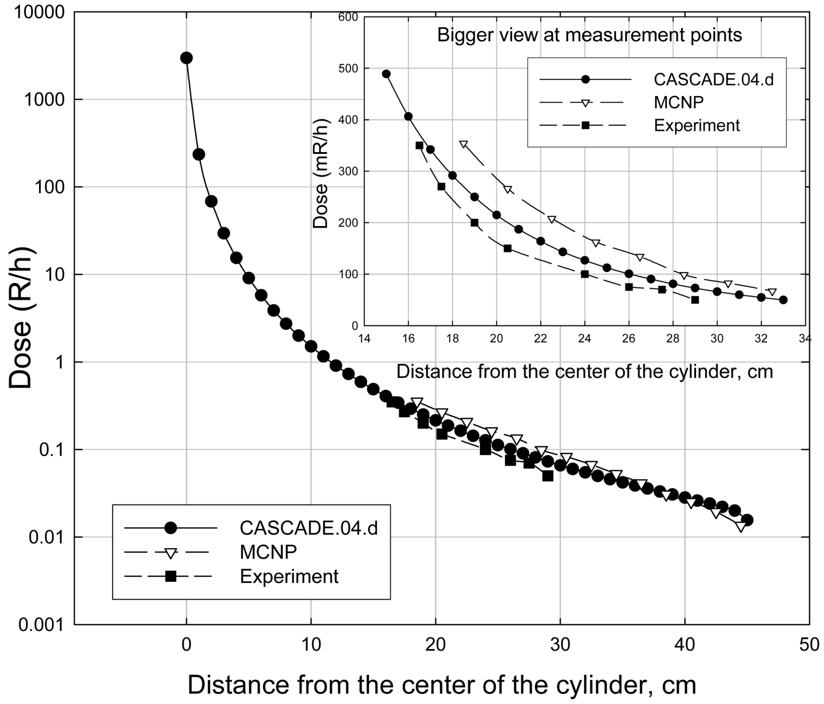

Figure 4. Comparison of the measured dose rate (R/h) at different radial distances of the 5Ci Am–Be Howitzer system, with calculated results using CASCADE.04.d and MCNP codes. In the inset bigger view at distances >14 cm are shown in zoomed mode.

gas and $S(\alpha, \beta)$ models. The tally errors varied between 3 and 6% in most of the energy bins including the source bin, that contribute to the total fluence and hence to the dose. In other energy bins, the bin errors ranged between 4 and 20%. The errors mentioned above are statistical errors only.

We have included the NCRP-38 [11], ICRP-21 [12], and ICRP-74 [13] for the fluence rate to dose rate conversion data in units of (Rem/h)/(n/cm^2/s) for the calculations in CASCADE.04.d. The fluence estimated using the two methods given in §2 is converted using the log–log interpolation as recommended by the NCRP-38, ICRP-21, and ICRP-74 reports. The dose rate calculation for higher energy neutron conversion factors up to 10 TeV are taken from ref. [14] and that is useful for the shielding calculation of the accelerator driven sub-critical systems.

The CASCADE.04.d and MCNP predictions are compared with the measured dose rate (Rem/h) and given in figure 4 as a function of the distance from source. It may be seen from figure 4 that the neutron dose rate decreases exponentially with the radial distance and the calculated results from the CASCADE.04.d code are in better agreement within a factor of 2 with the experimental data than MCNP. At larger distances agreement is better than at smaller distances. It may be inferred that the code can be used successfully for estimation of both the neutron fluence rate and the ambient dose rate.

4. Conclusions

Options to calculate neutron fluence rate and dose rate have been incorporated in the CASCADE.04.d code. The calculations of the neutron fluence from the modified code show that it is in reasonable agreement with the values estimated from the MCNP code. Results of the dose rate from the CASCADE.04.d code show better agreement with the experimental data. It may be pointed out that the CASCADE code generates the fluence as a function of neutron energy and the neutron dose will be different for different fluence distributions. Thus, using the iterative approach, measurement of dose may determine approximately distribution of the fluence as function of energy. As pointed out earlier that in the CASCADE code, fluence to dose conversion is given up to TeV range, the code may be used for similar calculations up to ∼TeV energy. However, we have not experimentally verified the code beyond 10 MeV neutron energy so far by the dose measurements but the code has been validated for the neutron fluence rate, i.e neutron flux up to several GeV energy [3,9,15].

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