Transport Calculations of Radiation in Different Materials

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Abstract:-

Transport calculations were made for radiation doses and fluxes in different materials for different ranges. The Monte Carlo multigroup experimental transport code MORSE-CG [1] was used for these calculations. Some modifications were introduced to this code. These modifications include the addition of subroutine called SOURCE, and the addition of a function called DIREC. The subroutine SOURCE allows the radiation source of particles to have certain geometry instead of the built-in point source. While the function DIREC makes the radiation to transport in a certain direction to reduce the time of calculations.

The obtained results of calculations were compared with experimental referenced data which show a good agreement.

Introduction:-

MORSE code solves the multigroup Boltzmann transport equation by Monte Carlo method for photons and neutrons radiations. It uses the energy group cross-sections (which represent all the probabilities of all kinds of radiation interactions with matter) with energy transfer coefficients (represented by legender expansion of an arbitrary order) .The scattering moments of photons are computed from the Klein –Nishina equation. The logic of random walk process used in Monte Carlo method (transporting of radiation particles from one collision to another) is identical for both neutrons and photons.

The solution of the Boltzmann equation by Monte Carlo method generally involves a solution of the density of particles with phase space coordinates. Quantities of interest are then obtained by summing the contributions of all collisions at certain positions. The random walk governs the process of choosing a source particle and following it through its history of events. Particle splitting and Russian roulette are employed to reduce uncertainties associated with the finite number of particles examined. The history of a particle is terminated when the particle leaks from the geometry, reaches energy cut-off or is killed by Russian roulette process.

A very flexible geometry package is incorporated in the code. It describes general three dimensional configurations. The analysis of the results gives arbitrary locations of the detectors used. There are no numerical limitations rather than the available computer storage. At the detector locations, the desired quantities are calculated. There is a diagnostic of analysis at any point in the transport calculations of particles to gain more information's into the physics of all particles interactions.

Calculation Details:-

The multigroup cross section used in the calculations for some materials are from RSIC [2]. Otherwise they are from ENDF\B library using GGTC or AMPAX II codes [3]. Transport calculations for neutrons and gamma radiations through air have usually important considerations for any radiations studies. Transmission of radiation through air is identical to that through dense media. The difference is in the relaxation lengths of radiation in air is of the order of tenth that in dense media. In our calculations, air was considered to be made of Oxygen and Nitrogen with total density of 1.29 gm/letter. The geometry considered in the calculations was a concentric spherical shell of air surrounding the source.

Five problems have been studied for using a point source of radiations these were

1. Gamma ray dose in an infinite air media.
2. Neutron dose in an infinite air media.
3. Gamma ray spectra in water.
4. Gamma ray spectra in lead.
5. Neutron fission source in water.

The code was tested after modifying it by studying the effect of modifications on the results of calculations. The problem choosed for this checking was an isotropic sphere of a certain radius of iron emitting gamma rays from the whole body of the sphere. Three cases were examined for three different energies.

Modifications of the code:-

1:- Subroutine SOURCE:-

The only source option built in the code is appoint source. This source is described by a function. The particles start with direction cosines u, v, and w. If u=v=w=0.0 the particles starts isotropic. For a source of a spherical geometry, the start location for each particle is selected at random from a point inside the sphere. The probability density function of this particle can be written as:

$$\int_{-\infty}^{\infty} p(r) d^3 r = 1$$

Where the integration represents a normalization factor, and represents the unit volume. The cumulative distribution function (P (n)) can be used to obtain the particle location at r. It is written as:

$$P(r) = \int_{-\infty}^{r} p(r^-) d^3 r^-$$

The source strength (S) at a point r from the surface of the sphere will be:

$$S = \int_0^r S_0 dV = \frac{4\prod S_0}{3} r^3$$

Where dV is the differential volume at r (dV=4Π r^2 dr), S is the total number of source particles emitted by the sphere.

The cumulative probability of particles born in the shell r_s will be:

$$P(r_s) = \frac{4\prod S_0}{3S}$$

The translation of source coordinates from the center of the sphere (x, y, and z) to a shell at a radius r_s will be:

$$x_s = x_\circ + u.r_s$$

$$y_s = y_\circ + v.r_s$$

$$z_s = z_\circ + w.r_s$$

$$r_s = r.R^{1/3}$$

Where R is the radius of the source sphere.

2- Function DIREC:-

This function deals with the cosines of the motion of the particles in the random walk process. It affects the cosines between the flight of particles and the most important direction. It is used to vary the amount of path length biasing depending on whether the particle is traveling in an important direction or not. For a source of a spherical geometry of radius R the direction can be written as:

$$R = \sqrt{x^2 + y^2 + z^2}$$

$$DIREC = (x.u+y.v+z.w)/R$$

Results and Discussion:-

The first problem considered in our calculations was a point gamma radiation source in an infinite air medium. The results of these calculations are shown in figure (1). This figure also shows a comparison between our results and three different referenced data. Two of them are based on moment method, while the third set based on Monte Carlo method [4]. The comparison was for gamma doses of energies 2 and 6 MeV. There is an excellent agreement between our results and the referenced data. The energy dose conversion factors used in our calculations were from ANSI/ANS [5].

The second problem considered in our calculations was determining the neutron doses reaching different distances in air. Figure 2 shows these calculations for neutron energy of 14 MeV. It also shows a comparison of our results with two different references. The first one by Starker, Keith, and Shelton [4]. They used discrete ordinate method in there calculations. The second by Webster [4], he used Monte Carlo method. A good agreement was found between these calculations and our results. The small differences appeared were due to the different energy cross section used.

Gamma ray spectra from a point source (Cs-137) of 600-700 MeV energy in water was the third problem studied. The obtained results were compared with a referenced data [6] as shown in figure 3. Two and three mean free paths of water have been considered. The agreement between our results and the referenced data was excellent despite of using two different sources of energy group cross sections.

The fourth problem was studying the spectra of 8 MeV gamma rays in different depths of lead. This is shown in figure 4. Figure 5 shows the last problem that had been considered in our calculations. It shows the attenuation of radiation from a neutron fission source in water. The attenuation started from near the source and gradually increases with depths. This study shows that a large fraction of the total flux (the upper line) was due to the scattered neutrons (the line in the middle), as expected. The lower line in the figure was for the uncollided flux.

To study the effect of modifying the code, a comparison was made between using a point source and using a source of a spherical shape. This comparison is represented in figures 6, 7, 8 for three different gamma energies. These figures show that using a spherical source leads to a constant flux along the diameter of the sphere, then the flux decreases gradually outside the sphere. This study shows that the added subroutines allowed us to calculate the needed parameters with a relatively higher accuracy and better statistics.

REFERENCES

1. M.B. Emmett., (The Morse Monte Carlo Radiation Transport Code System), ORNL-4082, 1975.
2. D.K. Trubey., S.K. Penny, K.D. Lathrop., ORNL-9., 1965.
3. Atlas of Photoneutron cross sections, UCRL-78482., Lawrance, Livermore Lab., 1976.
4. N.M.Scheeffers, (Reactor shielding for nuclear engineers), AEC, U.C. 80, 1973.
5. Neutron and Gamma ray flux to dose rate conversion factors, ANSI/ANS., 1977.
6. Palta, R., Thesis, the University of Missouri., USA 1981.