Experience with the Monte Carlo Method

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ABSTRACT

Monte Carlo simulation of radiation transport provides a powerful research and design tool that resembles in many aspects laboratory experiments. Moreover, Monte Carlo simulations can provide an insight not attainable in the laboratory. However, the Monte Carlo method has its limitations, which if not taken into account can result in misleading conclusions. This paper presents the experience of its author, over almost three decades, in the use of the Monte Carlo method for a variety of applications.

Key Words: Monte Carlo Simulation, Particle Transport

INTRODUCTION

The Monte Carlo method was introduced by Nicholas Metropolis and Stanley Ulam; both worked with John von Neumann on the Manhattan Project during World War II[(1)]. The concept was devised to liberate scientific calculations from the traditional pure combinatorial calculation approach that progresses through deterministic logical steps. The stochastic approach of the Monte Carlo method, called after the city of Monte Carlo in the principality of Monaco (known for its gambling houses), solves problems without defining a particular system of coordinates. Other numerical solution methods start with a differential, integral or an integro-differential equation describing the simulated physical process. The relevant equation is then either re-written in a finite-difference form, averaged over finite elements, or expanded in terms of a polynomial series into different modes, each mode in turn handled via a finite difference or finite-element formulation. Any of those formulations require the designation of a specific coordinate system: cartesian, cylindrical or spherical. This naturally restricts the use of these methods to problem geometries that can be confined to a parallelepiped, cylindrical or spherical shapes. The Monte Carlo method circumvents this by relying on random sampling from statistical distributions derived from the original mathematical/physical problem. The mean of a sampled quantity provides an acceptable estimate of the quantity of interest, regardless of the initial distribution from which the quantity is sampled. This is because the central limit theorem, also known as the law of large numbers, stipulates that any sequence of *independent* and *identically distributed* random variables with a common mean and variance approaches a normal distribution as the number of samples approaches infinity. Therefore, with every mean one can calculate a standard deviation that defines a level of confidence in the quantity. This is a unique feature of the Monte Carlo method, which unlike deterministic methods, associates with each given answer a measure of the confidence one can put in the obtained result.

The main advantage of the Monte Carlo method is its ability to handle complex geometries.

This enables one to simulate realistic geometries of experiments and systems. The method can then be applied as an "experimental" tool prior to performing actual laboratory experiments. This not only saves time, effort, acquisition of equipment, and personnel exposure to radiation, but it also assures to a great extent eventual success of experiments and designs. However, results from Monte Carlo simulations can be misleading, if careful attention is not given to setting up the problem and to interpreting results. This paper gives some of such encounters from experience. We begin, however, by showing how the Boltzmann radiation transport equation is formulated to suit the probabilistic formulations required by the Monte Carlo method. The discussion below is from a book by the author[2].

PARTICLE TRANSPORT (BOLTZMANN) EQUATION

The Boltzmann transport equation, which governs the transport of neutral radiation particles (neutrons and photons) is an integro-differential equation. For neutrons, it takes the form:

$$\begin{aligned}\frac{1}{v}\frac{\partial\phi(\vec{r},E,\vec{\Omega},t)}{\partial t} &= -\vec{\Omega}\cdot\nabla\phi(\vec{r},E,\vec{\Omega},t)-\Sigma_{total}(\vec{r},E,t)\phi(\vec{r},E,\vec{\Omega},t)\\ &+\int\left(\Sigma_{scatter}(\vec{r},E'\to E;\vec{\Omega}'\to\vec{\Omega},t)+\nu\Sigma_{fission}(\vec{r},E'\to E;\vec{\Omega}'\to\vec{\Omega},t)\right)\\ &\times\phi(\vec{r},E',\vec{\Omega}',t)\,dE'\,d\vec{\Omega}'+Q(\vec{r},E,\vec{\Omega},t)\end{aligned} \tag{1}$$

where $\phi(\vec{r},\vec{\Omega},E,t)$ is the neutron flux at the space location $\vec{r}$ with energy E and direction $\vec{\Omega}$ at time, t, per unit energy per steradian; Σ is the macroscopic cross section for the designated interaction, ν is the number of neutrons produced per fission and Q refers to an independent (external source) at the designated coordinates. For this equation to be amenable for use in Monte Carlo simulations, it needs to provide probability density functions. This is done by converting Eq. (1) into an *integral emergent particle density equation*, written in terms of $\chi(\vec{r},E,\vec{\Omega},t)$, the density of particles leaving a source or emerging from a collision at phase space coordinates $\vec{r}$ in direction $\vec{\Omega}$ at time t with energy E. Let us introduce the integral operators:

$$T(\vec{r}'\to\vec{r},E,\vec{\Omega})\equiv\int_0^\infty dR\,\Sigma_t(\vec{r},E)\exp[-\beta(\vec{r},R,\vec{\Omega})] \tag{2}$$

$$C(\vec{r},E'\to E,\vec{\Omega}'\to\vec{\Omega})\equiv\int_{E'=E}^\infty\int d\vec{\Omega}'\,\frac{\Sigma_s(\vec{r},E'\to E,\vec{\Omega}'\to\vec{\Omega})}{\Sigma_t(\vec{r},E')} \tag{3}$$

with $\beta(\vec{r},R,\vec{\Omega})=\int_0^R\Sigma_t(\vec{r}-R'\vec{\Omega})dR'$. The transport operator, $T(\vec{r}'\to\vec{r},E,\vec{\Omega})$, is an integral operator (kernel) that transports a particle from position $\vec{r}'$ to position $\vec{r}$, along direction $\vec{\Omega}$ while maintaining the energy E, i.e. with no interaction. The collision integral operator, $C(\vec{r},E'\to E,\vec{\Omega}'\to\vec{\Omega})$, causes a particle at $\vec{r}$ to collide, changing its energy and direction as it scatters. Bookkeeping of particles in the phase space results in the particle transport equation:

$$\chi(\vec{r},E,\vec{\Omega},t)=S(\vec{r},E,\vec{\Omega},t)+C(\vec{r},E'\to E,\vec{\Omega}'\to\vec{\Omega})T(\vec{r}'\to\vec{r},E',\vec{\Omega})\chi(\vec{r}',E',\vec{\Omega}',t) \tag{4}$$

where $S(\vec{r},E,\vec{\Omega},t)$ is the density of particles generated by an external source.

A solution of Eq. (4), like any other integral equation, can be approximately expressed by the sum:

$$\chi(\vec{r},E,\vec{\Omega},t)=\sum_{n=0}^{\infty}\chi_n(\vec{r},E,\vec{\Omega},t) \tag{5}$$

with

$$\chi_0(\vec{r},E,\vec{\Omega},t) = S(\vec{r},E,\vec{\Omega},t) \tag{6}$$

Then using, Eq. (4), one has:

$$\chi_n(\vec{r},E,\vec{\Omega},t) = C(\vec{r},E' \to E,\vec{\Omega}' \to \vec{\Omega})T(\vec{r}' \to \vec{r},E',\vec{\Omega})\chi_{n-1}(\vec{r}',E',\vec{\Omega}',t) \tag{7}$$

Recall that T and C are integral operators, defined respectively by Eqs. (2) and (3). Eq. (7) represents a cumulative density function (cdf) that can be employed in a Monte Carlo experiment.

A particle transport stochastic trial is called a *random walk*, or a *history*. Each random walk starts by sampling a source particle from $S(\vec{r},E,\vec{\Omega},t)$, to determine the particle's coordinates: $(\vec{r}_0,E_0,\vec{\Omega}_0,t_0)$. Obviously, $S(\vec{r},E,\vec{\Omega},t)$ has to be given in the form of a probability density function (pdf) that describes the source's geometry, energy spectrum, and change in intensity with time (if relevant). A flight distance R is then sampled using the transport kernel, $T(\vec{r}' \to \vec{r},E,\vec{\Omega})$, where R is the distance the source particle travels with energy E in the direction $\vec{\Omega}$ before encountering a collision at $\vec{r}_1 = \vec{r}_0 + R\vec{\Omega}_0$, where $\vec{r}_1$ is now the site of the first collision. The particle will arrive at this site at time $t_1 = t_0 + R/v_0$, where v_0 is the particle's speed corresponding to energy E_0. Time in Monte Carlo experiments is often referred to as the particle's "age".

The probability of scattering at the new site, as the collision kernel of Eq. (3) indicates, is $\frac{\Sigma_s(\vec{r}_1,E_0)}{\Sigma_t(\vec{r}_1,E_0)}$, where Σ is the macroscopic cross section of the material at position $\vec{r}_1$ and the subscripts s and t designate, respectively, the scattering and total cross sections. In order to avoid "wasting" a particle in random walks, particle transport codes perform the so-called "non-analog" sampling, a process in which particle-absorption is forbidden. This is a form of importance sampling, or biasing, aimed at avoiding early termination of the random walk, and allows the random walk to continue until the tracked particle escapes the system's geometry or is terminated by other means (e.g. a low-energy cutoff). The bias introduced by this process is compensated for by assigning to each sampled source particle some weight, (typically unity, or the source intensity), and multiplying the weight at the collision point by the non-absorption probability, $1 - \frac{\Sigma_s(\vec{r}_1,E_0)}{\Sigma_t(\vec{r}_1,E_0)}$. After some collisions, when the particle weight becomes low, Russian roulette sampling can be performed, or the random walk can be terminated altogether when the weight becomes too insignificant.

At the first collision at $\vec{r}_1$, the weight is adjusted, and post-collision particle energy, E_1 and direction $\vec{\Omega}_1$ are sampled from the distribution:

$$\int_{4\pi} d\vec{\Omega} \frac{\Sigma_s(\vec{r}_1,E_0 \to E_1,\vec{\Omega}_0 \to \vec{\Omega}_1)}{\Sigma_s(\vec{r}_1,E_0)}$$

Then, a new collision site is sampled from the transport kernel, and so on. The sampling process is repeated until the particle is terminated by escaping the system (to an artificially created external void), or reaching a pre-assigned cut-off limit for age, energy, or weight.

Estimators

The random walk process follows the transport of a particle, and can be repeated N times by sampling new source particles. It does not however estimate an answer to a physical quantity of interest, unless a "scoring" of this quantity is performed during the random walk, and a "tally" is kept of these scores. The quantity of interest is usually the particle fluence, or some related

value. The scoring process is determined by a variety of estimators which evaluate the fluence, or fluence-like quantities, at a point or a region. Statistical estimates, including the average and the variance of the average are calculated at the end of the random walk process.

The *body crossing estimator* evaluates the flux crossing a surface, by accumulating the weight of particles crossing the surfaces divided by the absolute value of the cosine of the angle between the normal to the surface and direction of the incident particle. Provisions must be made to avoid angles with small cosine values. The *track length estimator* evaluates the fluence by summing the track length of particles crossing a given zone, divided by the volume of the zone. This is usually suitable for evaluating the fluence in void or air regions, and regions containing a low density material. The *collision density estimator* adds up the weight of particles colliding within a zone, divided by the total cross section of the material and the volume of the zone. This estimator provides adequate estimates for the fluence in regions of high density materials, where a large number of collisions are anticipated.

In all the above estimators, the particle must visit the region or surface of interest. In situations where the probability of the particles reaching the region of interest is low, an expected-value estimator can be used. The *next event estimator* is such estimator. This estimator is particularly useful for point detectors, where there is only one possible position for the "next collision". This estimator determines at every collision the probability of the next event taking place at the detector site, and scores $\frac{W p(\mu) \exp(-\lambda)}{2\pi R^2}$, where W is the particle's weight, $p(\mu)$ is the value of the probability density function at μ, the cosine of the angle between the particle trajectory and the direction to the detector, λ is the total number of mean-free-paths encountered over the trajectory from the collision point to the detector and R is the distance between the collision point and the detector. This is done by tracing a "pseudo-particle" from the collision site to the detector (scoring point), without altering the original random walk path. The same process is also performed for source particles to provide the uncollided component. However, collisions very close to the detector site can lead to unrealistically high scores, as $R \longrightarrow 0$. This estimator should not therefore be used in regions with high collision density, and scores close to the detector site should be excluded when this estimator is utilized.

PROBLEM SETUP

As indicated above, a random walk is controlled by the transport and collision kernels. The distance of flight is sampled from a distribution described by the transport kernel, while the status of the particle after collision is governed by a distribution defined by the collision kernel. Both kernels are defined by the material's cross sections presented in the form of probability tables. Throughout the transport process, a particle encounters different geometric configurations and materials. Some biasing techniques may also be applied to discard particles that are unlikely to significantly contribute to the quantity of interest, and to promote particles of importance. Eventually, of course, estimates of the quantity of interest must be obtained, otherwise the entire exercise is fruitless. One has however to start with a source. Below we summarize the essential steps that must be specified before executing a Monte Carlo experiment.

Source

The position, geometry, directional distribution and energy distribution of the particle's source must be specified. In transient analysis, a function describing the change of the source's intensity with time must also be given. Fission sources and secondary particle sources, (e.g. a gamma-

ray following a neutron interaction, an electron following a photon absorption, or bremsstrahlung photons), are determined by the cross sections of the material and need not be specified as input parameters. The fission distribution with energy needs, however, to be specified, in order to determine the energy of the emerging neutrons. A source particle is also assigned a weight, typically a unity. When fission occurs, or when a secondary particle is produced, the attributes of the new particle (weight, position, energy, direction and age) are stored in a bank for later processing, once the tracking of the initial source particle is completed.

Geometry

The Monte Carlo method can handle complex geometries. The geometry must however be specified in such a way that enables tracking of the particle throughout the system. This tracking process determines not only the particle's spatial position but also the type of material encountered hence cross sections, at the encountered site. The geometry can be specified via analytical geometry procedures, which define the surfaces of different geometric objects. Surfaces are then combined (with logical operators) to define volume cells. This is the method used in the MCNP code[4]. Alternatively, geometry may be specified via a set of elementary bodies, combined together using logical operators to form a zone of a particular material. This is the so-called combinatorial geometry method utilized in the MORSE code[5].

Material Cross Sections

The cross sections for the different materials encountered must be supplied as a function of energy. The Legendre expansion coefficients for each material are also needed, in order to account for anisotropic scattering. The cross sections are processed prior to the execution of the random walks, to provide the probability tables. The tabulated values are then used to determine the distance to be traveled by a particle until it encounters the collision (interaction), the outcome of the collision, and the outgoing energy and angle of a scattering event; in addition to the number of neutrons per fission for fissile materials.

Importance Sampling

Some initial trials should be performed without importance sampling. If the variance of a quantity of interest is high, then these initial trials can provide insight into the problem and assist in assigning the spatial regions and energy regions within which importance sampling can be effective in reducing the variance. Importance sampling can be performed at the source, during the random walk and at scoring. The expected-value estimates of the next-event estimator can be viewed as a form of importance sampling. Source biasing should allow the production of more source particles, with suitably reduced weights, in the more important ranges of each variable: position, energy and direction. For example, source particles directed to a region of interest should be sampled more often than those directed away from it, provided of course that the weight of the source particle is accordingly adjusted. Variance reduction can also be achieved by the cut-off parameters, discussed below, where insignificant particles are not allowed to continue to score endlessly to the quantity of interest. However, the main methods of importance sampling in particle transport codes are splitting, Russian roulette and exponential transformation.

Splitting should be applied in regions or energies that are expected to significantly contribute to the quantity of interest but is unlikely to be reached; the opposite is true for Russian roulette. It is important however to control the amount of splitting to avoid the unnecessary creation of too many particles. Russian roulette takes a particle of weight W and turns it into a particle of weight $W' > W$ with probability $\frac{W}{W'}$ and kills it with probability $1 - \frac{W}{W'}$. In general, Russian roulette increases

the history variance but decreases the time per history, while splitting achieves the opposite effect.

The exponential transformation is a process that stretches or shrinks a particle's path-length between collisions. This is done by artificially reducing the macroscopic cross section in the preferred direction and increasing it in the opposite direction. A fictitious cross section, Σ^*, is related to the actual cross section, Σ, by: $\Sigma^* = \Sigma_t(1 - p\mu)$, where μ is the cosine of the angle between the preferred direction and the particle's direction and p is a biasing parameter, $|p| < 1$; a constant or equal to $\frac{\Sigma_a}{\Sigma_t}$. For $p = \frac{\Sigma_a}{\Sigma_t}$ and $\mu = 1$, $\Sigma^* = \Sigma_s$ and the particle path is sampled from the distance to the next scattering, rather than from the mean-free-path ($\frac{1}{\Sigma_t}$) for all interactions. The weight is, consequently, adjusted by a factor of $\exp(-\Sigma_a d)$, where d is the distance of travel. Therefore, the exponential transformation works best in highly absorbing media and very poorly in highly scattering media. Exponential transformation is useful in deep penetration problems.

Tallys

Tallying is the process of scoring the parameters of interest to provide the required answer for the problem at hand. One or more of the estimators discussed in this section can be used to calculate various quantities of interest. For example, the particle current (directional flux) over a surface can be evaluated using the body-crossing estimator. This estimator, by including all directions, can be used to estimate the particle fluence (or flux when the source weight (strength) is given in terms of particles per unit time). The particle flux can also be evaluated within a volume using the track-length or the collision density estimators; obviously the latter estimator is not applicable in void and is not reliable in low-density regions where very few collisions take place. The next-event estimator can be used for estimating the flux at point in a voided zone where no collision can occur near the detector site. When using this estimator in a low-density region, an exclusion zone should be assigned around the point detector to avoid singular estimates. Scoring the particle flux multiplied by the material's total cross section in the region where the flux is evaluated provides an estimate of the interaction rate. For each of these tallies, the user can designate the particle energy, direction or range(s) within which the final answer is desired. The scored quantity for the flux is basically the particle weight. This weight multiplied by the particle energy provides an estimate of energy deposition. One can also supply a detector-response function by which the particle flux is multiplied to simulate the response of a physical detector.

Termination

Since Monte Carlo particle transport codes usually employ non-analog sampling, the user must specify some criteria to terminate a random walk. Termination of a random walk can be effected by one of the following criteria: an upper bound for particle age, an energy threshold, a cut-off weight, and by defining a full absorption (also called external void) region outside the domain of interest. In addition, the user must also specify a criterion for terminating the entire Monte Carlo experiment. This is typically done by specifying the number of random walks (histories) to be performed. However, it is often desirable to also assign a maximum computer execution time, just in case a random walk is trapped endlessly within a particular zone in the problem geometry. When performing a criticality calculation, the number of cycles (generations) should also be specified.

Computer Codes

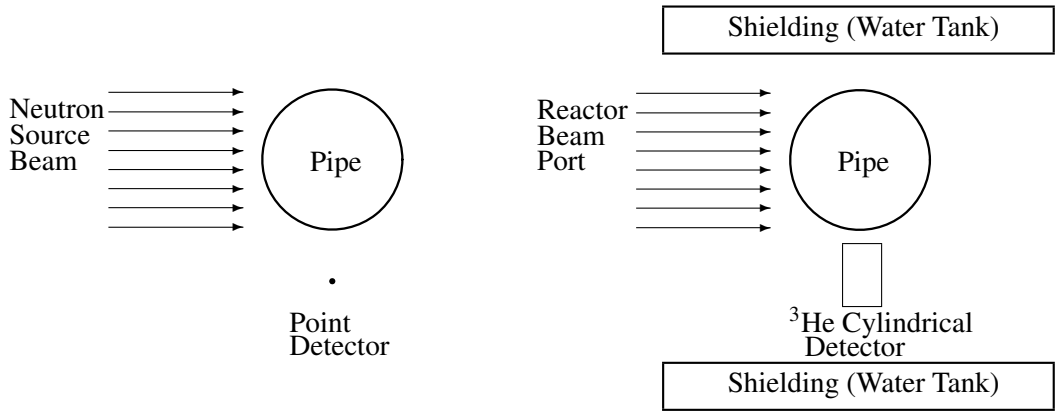
A number of Monte Carlo computer codes are readily available and can be acquired through the Radiation Safety Information Computational Center, Oak Ridge, TN (`http://www-rsicc.ornl.gov`), or the OECD Nuclear Energy Agency, France (`http://www.nea.fr`). However, the

most widely used code for particle transport analysis is perhaps the MCNP code[4], for neutrons, photons and electrons, and its extension the MCNPX code[6] which applies to other particle types as well. The COG code[7] can "simulate complex radiation sources, model 3D system geometries with 'real world' complexity, specify detailed elemental distributions, and predict the response of almost any type of detector[8]. The MCBEND[9] code is commercially available, and is designed for "the every day (or occasional) user". The EGS4 code[10] and the TIGER series of codes[11] are also used in simulating the transport of photons and electrons. The Geant4 toolkit[12] includes also the simulation of high-energy particles. Special-purpose Monte Carlo codes, as those described by Prettyman et al.[13] and Hussein[14], have been developed for specific tasks. Those interested in writing their own Monte Carlo code for photon transport will find the analytical expressions for the cross-sections given in Baro et al.[15] quite useful.

FROM EXPERIENCE

The author's first experience was in simulating a neutron scattering technique for the measurement of the void (steam) content of boiling water in a pipe[16]. The simulations was conducted for the purpose of explaining the unanticipated results that were encountered in experiments conducted in the same laboratory by someone else. The technique relies on measuring the amount of neutrons thermalized by the water phase which should to be directly indicative of the liquid, and consequently the void, content. The simulation setup differed from the experimental one in that the former used a point detector, weighted with the (n,p) cross section of ^{3}He, rather than an actual cylindrical ^{3}He detector. The source beam was also considered to be a Watt fission source, while the actual beam was extracted from the center of the core of a research reactor via a beam port, and as such was neither a perfect parallel beam nor had an idealistic fission spectrum. However, a similar set of experiments conducted in another research reactor produced the expected results, and the above stated differences were considered not to be too important. After careful viewing of the local experiment, it was realized that there was two water tanks on each side of the setup, as shown in Figure 1, that were absent in both the original simulation and the experiments conducted elsewhere. These water tanks were too close to the setup that they reflected slowed-down neutrons towards the setup, which competed with the neutrons thermalized within the test section (pipe) and reduced the detector's sensitivity to the water content in the pipe. Upon incorporating these tanks in the simulation, the results agreed with those of the experiment. Lesson learned: pay attention to the surroundings of an experimental setup.

Many years later, upon designing a self-shielded device (scatterometer) for void-fraction measurement using the scattering of neutrons emerging from an isotopic source, ^{252}Cf, though accounting carefully for the surrounding shielding, the measured contrast of the device (ratio between count rate for a full-of-water pipe and an empty one) was much lower than the simulated one. This difference persisted even upon lining up the interior walls of the shielding with cadmium sheets to eliminate any thermal (below the 0.5 eV cadmium cutoff energy) neutrons. Full simulation of the detector, as a metallic cylinder containing a ^{3}He gas at the same pressure as that of the actual detector, did not alleviate this difference. After some investigation, it was realized that the difference can be attributed to the fact that the code used (MCNP[4]), being a neutral particle code, does not simulate the recoil of the ^{3}He nuclei upon their elastic scattering with neutrons. This interaction results in a detector signal corresponding to fast neutrons that far exceeds that produced by the $1/v$ cross section of the (n,p) interaction which is used as a detector response function in the simulations; where v refers to neutron velocity. Lesson learned: every code has its own limitations and some of the physical processes are not simulated. Realizing



(a) Simulated Geometry (b) Experimental Geometry

Figure 1: Simulated and actual geometries for neutron-scattering void-fraction measurement.

this limitation, a successful design of the scatterometer was arrived at using a combination of simulation and experimental knowledge[17]. Efforts to further enhance the performance of the scatterometer and expand its capabilities using Monte Carlo simulations are discussed in detail elsewhere[18].

Simulating the detector response can also be problematic with photon detectors. A code such as MCNP[4] has the ability to simulate electron transport and hence fully simulate the detector response, and as such one can avoid the use of a detector response function. However, full simulation of the detector response can be quite time consuming and inefficient, distracting from the ability to focus the random walk process on the studied system. It is therefore advisable to perform separate simulations to generate a detector response matrix, that can be used to adjust unadulterated photon energy spectrum data that can be readily obtained from photon-transport simulation. However, in generating the detector response matrix attention should be paid to the direction of photon incidence on the detector as it affects the response function as our simulations of an HPGe detector indicated[19].

Our experience also indicates that different results can be obtained using different cross-section libraries, random-walk cut-off parameters, and different levels of simulation details. Even simulating the same problem with the same setup can produce slightly different results, as we found when comparing the results of radiation energy deposition in a CANDU reactor channel using the MCNP and EGS4 codes[20].

The numerical values obtained by Monte Carlo simulations are typically normalized to one source particle. Multiplying the obtained value by the source strength (particles/second) can provide values comparable to what is measured experimentally. It should be however kept in mind that many factors prevent one from obtaining values that match exactly the experimental results, due to simplifications in the simulation setup, limitations of simulation models, the presence of experimental background radiation and electronic noise, etc. Calibrating simulations to a well-defined experiment can resolve those problems. But Monte Carlo simulations, like all other types of numerical simulations, are best in examining trends, determining the parameters of most influence on the results, optimizing an experiment and in proof-of-concept studies.

We have used Monte Carlo simulations for designing a number of instruments; including the measurement of gadolinium concentration using neutrons scattering[21], and the inspection of concrete structures[22] and composite materials[23] using Compton scattering. We have also employed the method for investigating techniques to detect landmines using isotopic sources[24]. Monte Carlo simulations are quite useful in design as they can provide insight into the physical nature of the problem at hand, that is not easily measurable or observable experimentally. For example, in the gadolinium measurement system we were able to obtain a footprint of the thermalized neutrons, and make sure that all points within the test section had received more or less the same number of thermalized neutrons, so that a representative response of the contents of the various regions of the test section is obtained.

We have also found Monte Carlo simulations to be quite useful for providing independent data for testing numerical algorithms for decoding measurements[25] and for image reconstruction[26;27]. Monte Carlo simulations are also useful in designing radiation shielding and in estimating radiation dose due to contamination[28].

CONCLUSION

As indicated in this paper, the Monte Carlo method has occupied the about thirty-year research career of this author, who employed it in a number of ways and for different objectives. It is a powerful method of simulation, but should be used with great care. The Monte Carlo method is best suited for developing new concepts, performing parametric studies, optimizing system design, or providing independent numerical data in the absence of readily available experimental measurements. This simulation method can also provide data not measurable experimentally, as for example tracing the motion of radiation flow. When attempting to obtain absolute values that can be directly compared with experimental data, attention should be given to the setup details and the capabilities of the models employed in the simulation code. The power of this simulation method lies in its ability to handle complex geometries. Its main limitation is that it only provides solutions at specified locations, unlike deterministic methods which provide values at all points in the space considered.

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