Chapter 10

TALLYING IN MCNP

Tallying is the process of scoring the parameters of interest, i.e. providing the required answers. For each answer the fractional standard deviation (fsd), relative error, is provided. Each tally is defined by an Fna number, where "n" is a unique (i.e. not repeated in the same job) number and "a" is the particle type (N,P, or E). The scoring of each quantity of interest is discussed below. Note that adding multiples of 10 do not alter the tally type, i.e. F1, F11, F21 are all type F1 tallies, specified for different reasons.

10.1 Current (F1)

The neutron, photon, or electron current (particle energy) integrated over a surface:

$$\mathrm{F1} = \int_A \int_\mu \int_t \int_E J(\vec{r}, E, t, \mu) dE dt d\mu dA$$

$${}^*\mathrm{F1} = \int_A \int_\mu \int_t \int_E E * J(\vec{r}, E, t, \mu) dE dt d\mu dA$$

Note that $J(\vec{r}, E, t, \mu) = |\mu|\Phi(\vec{r}, E, t, \mu)A$. The current, F1, calculated in MCNP is the number of particles crossing a surface in a given direction, therefore, it simply scores the particle's weight. *F1 is analogous to the radiative flux crossing an area in radiative-heat transport theory. The range of integration can be controlled by the cards FS for area, E for energy, T for time and C for angle cosines (relative to the normal to the surface, unless another reference vector is defined by the FRV option in an FT card). This tally can be used for all particle types, but for electrons the ELC option in the FT card can be used to segregate tallies by charge (positrons from electrons).

10.2 Flux (F2, F4 and F5)

Three estimators for flux are available: surface crossing (F2), track-length (F4) and next-event (point and ring detectors). These are estimates of the quantity $\int_t \int_E \Phi(\vec{r}, E, t) dEdt$. Note that the integral is over energy and time, resulting in units of particles per cm^2. The range of integration can be controlled by the E and T cards. The flux time units are controlled by the units of the source; that is the tally represents fluence tally if the source has units of particles and represents flux if the source has units of particles per unit time. The *F2, *F4 and *F5 estimate $\int_t \int_E E\Phi(\vec{r}, E, t) dEdt$. F5 is only available for photons and electrons.

10.2.1 Surface-Crossing Estimator (F2)

Using the relationship between flux and current, $J(\vec{r}, E, t, \mu) = |\mu|\Phi(\vec{r}, E, t, \mu)A$, the flux is estimated by scoring the $W/(|\mu| * A)$, where W is the particle weight crossing the surface within the designated time and energy range. MCNP sets $|\mu| = 0.05$ when $\mu < 0.1$ to avoid singularity. Note that when $W = 1$ and the $\mu = 1$, this estimator provides an estimate of $1/A$, i.e. it can be used to stochastically estimate the area.

10.2.2 Track Length Estimator (F4)

This estimator uses the fundamental definition of flux as the number of particle-track lengths per unit volume. Therefore, WT_L/V is scored within a cell all particles tracks in the designated time and energy range.

10.2.3 Next-Event Estimator (Point and Ring Detectors)

Unlike F2 and F4, F5 does not require a particle to reach the detection location. F5 scores at very collision the probability that the next event being at the detector site, and scores $Wp(\mu)\exp(-\lambda)/2\pi R^2$, where $p(\mu)$ 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 integrated over the trajectory from the collision point to the detector and R is the distance between the collision point and the detector. $p(\mu)$, and consequently F5, are available only for neutrons and photons. This is done by tracing a pseudo-particle, without altering the original random walk path, from the collision site to the detector. The same process is also performed for source particles to provide the uncollided component. The estimator should not be used near reflecting, white and periodic boundaries, since pseudo-particles travel only in straight lines (between two points). MCNP provides estimates of the quantities of interest for source particles alone (called direct contribution), due to uncollided particles, as well as due to the source and interactions combined (total contribution).

Russian roulette can be played on pseudo-particles using the variance reduction PD card, or the DD detector diagnosis card. The PD card is used to create pseudo-particles with a probability, p_i, or eliminating the creation of such particles in a cell if , $p_i = 0$; otherwise Russian roulette is played with probability p_i and the weight of the surviving particle is adjusted by the factor $1/p_i$. The DD card is used to play Russian roulette with unimportant pseudo-particles, i.e. particles with a small $Wp(\mu)/2\pi R^2$ value. MCNP by default applies Russian roulette to pseudo-particles with $k = 0.1$, where k is the parameter that determines the weight below which Russian roulette is applied. This the only variance reduction technique, aside from non-analog Monte Carlo, MCNP applies by default, because it is a very powerful and effective technique.

A sphere of exclusion of radius R_0 can be specified with F5 to exclude collisions that occur close to the detector site and cause singularities. The code corrects the flux estimator accordingly, see manual. The proper value of R_0, in cm or mean-free-paths, requires some experimentation. However, detectors with different values of R_0, called coincident detectors, can be used with little cost incurred, to experiment with the value of R_0. Coincident detectors can be also used to add more than one detector with, for example, a different response function, at the same location.

The next-event estimator is particularly attractive when the flux is to be estimated in a region where the particles are unlikely to reach the region on their own. It is however a deterministic statistical estimate that is expensive, as it requires scoring at every collision, and can result in ambiguous statistics, particularly when collisions occur near the detector. It is therefore most suited for evaluating the flux in air far away from the media where collisions occur.

A ring detector is simply a point detector in which the point detector location is not fixed, but sampled from some location on a ring. A ring detector is defined by F5X, F5Y or F5Z, with X, Y and Z correspond, respectively, to rings located rotationally symmetric about the x, y and z axes.

Although singularities due to the $1/R^2$ term in the next-event estimators can be controlled using an exclusion zone, singularities can be also introduced by $p(\mu)$, which is not a probability but is the derivative of a probability) and can exceed unity. It can approach singularity in highly forward peaked scattering, such as the coherent scattering of photons. It is recommended that coherent scattering be tuned off, when this estimator is used.

10.3 Energy Deposition (F6 and F7)

Enrage deposition tallies estimate:

$$F_{6,7} = \frac{\rho_a}{V\rho_g}\int_V\int_t\int_E H(E)\Phi(\vec{r},E,t)dEdtdV$$

where ρ_a and ρ_g are, respectively, the atomic and mass (gram) densities and $H(E)$ is the heating response (MeV/g or jerks/g for $^*F_{6,7}$, 1 MeV= 1.60219E-22 jerks). $H(E)$ has different meaning depending on the particle type, consult the manual. These tallies are merely track-length estimators of the flux with an energy-dependent multiplier, $H(E)$. Therefore, the F4 tallies with the proper energy-dependent multiplier, FM card, can made equivalent to the F6 or F7 tallies. Note that the FM card can be used with the surface-crossing tally (F2) and the next-event estimators (F5) to calculate heating as well. The *F8 tally can be used for photons and electrons to calculate energy deposition on a surface.

The F6 tally includes all reactions and scores the quantity $WT_l H(E)\, \rho_a/(\rho_m V)$. F7 scores fission energy deposition, $WT_l H(E)\rho_a/(\rho_m V)$, and is therefore available only for neutrons. In a neutron problem, F7 gamma heating is deposited locally (all fission photons are immediately captured), while in F6 gamma heating is deposited elsewhere, when the photons are tracked. Then the true heating is obtained by combining the neutron and photon tallies in a coupled neutron/photon calculation, with the F6:N,P tally.

10.4 Pulse Height (F8)

This tally is not recommended for use with neutrons, and does not work with most various reduction schemes, since it is inherently an analog process. In a cell that models a physical detector. F8 is a surface-crossing estimator in a cell. The source cell is first credit with the energy times the weight of the source particle. When a particle crosses a surface, the energy times the weight is subtracted from the account of the cell it is leaving and is added to the account of the cell it is entering. At the end of the history, the account in each tally cell is divided by the source weight. The resulting energy determines which energy bin the score is put in. The value of the score is the source weight for an F8 tally and the source weight times the energy in the *F8 tally. The value of the score is zero if no track entered the cell during the history. When *F8 energy deposition tally is used and no energy bins are specified, variance reduction is allowed, since the total energy deposition is to be obtained.

10.5 Tally Controls

10.5.1 Binning

The E, T, C and FS cards control, respectively, the energy, time, angle cosines and the surface or cell segments for scoring.

Binning by the number of collisions can be done with the INC option in the FT card and an FU card. This is useful for example to obtain first-collision effects.

Binning by a cell, to determine what potion of a detector tally comes from what cell, can be done with the ICD option on the FT card and an FU card.

The SCX and SCD options on the FT card can be used to bin a tally score according to what source distribution caused it . SCX limits scoring to a single specified distribution, while SCD allows more than one distribution, and the results are binned to each individual distribution.

Binning by particle type (for multigroup calculations) and particle charge (electron or positron) is allowed also, see the FT card PTT and ELC options.

10.5.2 Flagging

CF and SF cards can be used to print contributions form particular cells. For example, an F4 1 tally followed by an CF4 2 3 4 tally permits printing the total flux in cell 1, as well as that part of the flux contributed only by particles having passed through cells 2, 3 and 4.

10.5.3 Response Functions

The EM, TM and CM cards multiply, respectively, each energy, time and angle cosine by a different constant. This is useful for introducing response functions or changing units.

The DE and DF cards allow modeling of an energy-dependent dose function that is a continuos function of energy from a table whose points do not coincide with the tally energy bin structure (card E). Useful for example for flux-to-dose conversion factors.

The FM card multiplies the F1, F2, F4 and F4 tallies by any continuos-energy quantity available in the data libraries. These include for neutrons the absorption cross section, average heating number (MeV/collision), gamma-ray production cross section, total fission cross section, fission ν, and fission Q (MeV/fission). For photons, available functions include incoherent and coherent scattering cross sections, photoelectric, pair-production and total cross sections and the photon heating number.

Gaussian energy broadening, useful in simulation of radiation detectors is available through the FT tally (GEB) option.

Time convolution, useful for example to simulate a square source pulse can be introduced through type TMC option in the FT card.

A user supplied tally can be provided by the TALLYX subroutine, see manual. The tally output format can also altered by the FC, FQ TF, DD PRINT cards.

10.6 Work Problems

In a particular problem, the following parameters are of interest:

1. The average neutron fluence in cell 10.
2. The fission heating per unit volume of material number 10001 at the origin point (0,0,0).
3. The total tritium production per cm^3 from natural lithium in cell 1.
4. The photon flux at the origin point, contributed only from cells 1 and 3.

Write the tally cards required for the problem.