

22.54 Neutron Interactions and Applications

Date Due: 14 February 2002

Problem Set 1: Beginning to use MCNP

For this problem set you'll begin to learn to use the MCNP radiation transport code. Then, with each new problem set we will learn different features of the code. We start by learning how to construct an input file and how to find information in the output file.

Californium – 252

Half-life:	2.65 yr
Branching ratios:	alpha decay: 96.1%
	Spontaneous fission: 3.09 %
Neutron emission:	2.31×10^{12} n/s/g
Mean fission neutron energy:	2.14 MeV
Average prompt gamma energy:	0.8 MeV

For this problem you need to model a $120 \mu\text{g}$ ^{252}Cf source in a cylindrical shield. Make the shield from a combination of cylinders and plans. You will want to vary the thickness of the walls of the cylinder (see below), and you can model what's inside the cylinder as simply air. Assume that the ^{252}Cf is a point source which doesn't have to be explicitly described in the geometry specification portion of the input file. [You'll specify where the point is in terms of x, y, and z in the source definition of the input file.]

Although ^{252}Cf decay leads to alpha, neutron and gamma ray emission we will only model the neutron emission here. [However, we want to keep track of photons that might be induced in your shield so make sure you specify that you want to run a mode n p problem.] There are two ways we could model the emission of neutrons from the source. One way is to obtain a plot of neutron emission versus energy and bin-histogram it. You can then instruct MCNP to randomly start neutron from this user input distribution. Another way is to make use of a Watt fission spectrum that MCNP supplies via the following equation: $f(E) = C \exp(-E/a) \sinh(bE)^{1/2}$. Coefficients are given for several isotopes that fission. The coefficients for spontaneous fission of ^{252}Cf are: $a = 1.025$ (MeV) and $b = 2.926$ (MeV⁻¹). So, your sdef cards will look something like this:

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sdef  erg d1 pos x y z
sp1   -3 1.025 2.926
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Surround the cylindrical shield that contains the point source of Californium with air and place a tally region (a 'detector') a distance of approximately 1 meter from the source. There are several ways to do this. You could simply model a small air-filled cavity and measure flux through this cavity, for instance. Or, if you needed to account for interactions in the detector, you could try to model it explicitly (accurate dimensions, materials, etc.). If you just want to extract information about the shielding ability of different materials, this isn't necessary. A good way to increase

the efficiency of your calculations is to surround the source/shield with a spherical detector. While this has little to do with a realistic situation it does ensure that all particles that leave the shield (in any direction) contribute useful information to your tally. An f2:n tally will give you an estimate of neutron flux across a surface. Units will be particles/cm² per starting neutron. You will also want to measure photons induced in the shield (remember we are not starting any photons from the decay of californium). You can do this using the same tally surface. We'll use an f2: tally again but since tally numbers must be unique you should label the photon tally f12:p or f22:p, etc.

Now, simulate different shield materials. Try H₂O, D₂O, iron, air and boron. In other words, run the simulation five different times, each time changing the material card for the shield (don't forget to change the density of the material on the cell card for the shield). Vary the thickness of the shield and tabulate the approximate thickness needed to reduce the neutron flux tallied with an "air shield": by one order of magnitude for each material. Include the photon flux measured, the average mean free path of the neutrons in the shield, and the number of photons generated in the cell representing the shield. From the average mean free path you can also calculate the average microscopic cross section for each shield. [The average mean free path and the number of photons created can be found in the output file in tables given before the tally information.]

Your table will look something like this:

Shield Material	Thickness to Reduce Flux to 0.1	Avg. MFP	Avg σ	Photon Flux	#photons

So, you have just completed an 'experiment' looking at the shielding potential of a number of materials. Notice that you did not expose yourself to any radiation, you didn't have to purchase the californium (\$1 Million/g at 1992 prices), you didn't have to build any shields, and you didn't have to deal with the paperwork involved with purchasing, shipping and storing the californium.

Now, treat the data from this Monte Carlo experiment as you would the data from a laboratory exercise and discuss the data in your table (eg. attenuating properties of the materials, photo production etc.). Look through the output file and extract and comment on any other useful bits of information you find there. Make a note of what table the data appear in.