22.54 Neutron Interactions and Applications
(Spring 2002)

Problem Set No. 7 **Due: 16 May, 2002**

For this problem set you will once again be working with the MCNP code.

Imagine that you are conducting experiments with an accelerator-based neutron source and you are concerned about the dose equivalent you will receive during your work. You begin by determining dose equivalent from the source itself, and then design a shield to protect yourself.

Model the accelerator-based source as a monoenergetic point source at 14.0 MeV emitting monodirectionally in the direction of yourself. Do this using the **DIR** entry on the **sdef** card. A **DIR** of 1.0 will generate a pencil beam in the direction specified by the **VEC** entry.

Your **sdef** card will look something like this: **sdef vec 0 0 1 dir 1.0**
If your cylinder of tissue-equivalent material is not in the z-direction, change the **VEC** entry accordingly. If the source is not located at position 0,0,0 (default) you will have to give it specific co-ordinates. An energy of 14 MeV is the default value.

Model yourself as a right-circular cylinder (20 cm in diameter and 20 cm in height) made of tissue equivalent material (use the tissue composition listed in the caption of Figure 5.3, attached, or from the H.Cember table 6.12 for synthetic tissue composition, given with the class notes). Place yourself a meter or so away from the source with your cylindrical axis parallel with the beam direction.

Section the cylindrical tally region so that you can tally fluence as a function of depth in the phantom. Use an **f4** tally and convolute the fluence with energy-dependent kerma factors (these are given below) using the **de** and **df** tally cards.

Separately tally the following components:
- total neutron kerma
- fast neutron kerma ($E_n > 0.4$ eV)
- slow neutron kerma
- photon kerma

You can get all the neutron information from one tally by using the Tally Energy Card which will split a tally into separate energy bins. For example, **e4 0.4e-6**, will give you data in the bin up to 0.4 eV, a bin above 0.4 eV, and also information about the total tally over all energies.

You will need a separate tally for the photon kermas:

Photon kerma factors:
```
f14:p
de14   0.055 0.3 0.75 1.5 2.5 3.5 4.5 5.5 6.5
df14   7.42e-11 1.51e-10 3.82e-10 6.71e-10 9.63e-10
            1.21e-9 1.43e-9 1.64e-9 1.85e-9
```

1. Plot total neutron, fast neutron, thermal neutron, and photon kerma as a function of depth in the phantom. Comment on both the shape of the kerma versus depth curves, and on their magnitudes in terms of the various interactions going on in the tissue.

2. Discuss the difference between kerma and dose. Can you assume your tally is giving you dose information? Why?

3. Add an $S(\alpha,\beta)$ card to the materials specification. That is, if the tissue-equivalent phantom is composed of material X, then add the following card to your input deck: **mtX lwtr.01**

This will over-ride the free-gas model and instead ensure that MCNP begins, at energies lower than 4 eV, to sample from cross-sections that incorporate the molecular binding effects in water.

Re-run the simulation and plot fast neutron and thermal neutron doses with and without use of the $S(\alpha,\beta)$ cross-sections. Comment on the differences observed and the consequences of omitting the $S(\alpha,\beta)$ treatment.

3. Calculate the first-collision dose from the 14 MeV neutrons in your tissue-equivalent phantom. Discuss how the results compare with those obtained in part (2) above.

4. Using Figure 5.3, chose two fluence-to-kerma conversion factors, one for "thermal neutron" energies, and one for "fast neutron" energies. Re-run the simulation using only these two energy bins and compare the dose-versus-depth curves you generate with those from part 2 above. Comment on the differences.

5. Now, design an effective neutron shield to reduce your dose by three orders of magnitude, and insert it between you and the source. Try to make the shield as small as possible. You need to think about how to slow down the fast neutrons, and how to absorb the thermal neutrons. Try to avoid photon production in your shield or you will have to include some photon shielding. Re-run the simulations to confirm that you can substantially reduce the dose. If you have trouble getting good statistics in the deeper depths of the phantom (and of course you should if you have constructed an effective shield!), try increasing the importances in the deeper parts of the phantom and/or shield. Discuss your results and the quality of your shield.

6. What is your <u>dose</u> with and without the shield? What is your <u>dose equivalent</u> with and without the shield? State your assumptions.

<u>Neutron Fluence-Kerma Conversion factors</u>:

```
de414     0.2530000d-07 0.3600000d-07 0.6300000d-07 0.1100000d-06
     0.2000000d-06 0.3600000d-06 0.6300000d-06 0.1100000d-05
     0.2000000d-05 0.3600000d-05 0.6300000d-05 0.1100000d-04
     0.2000000d-04 0.3600000d-04 0.6300000d-04 0.1100000d-03
     0.2000000d-03 0.3600000d-03 0.6300000d-03 0.1100000d-02
     0.2000000d-02 0.3600000d-02 0.6300000d-02 0.1100000d-01
     0.2000000d-01 0.3600000d-01 0.6300000d-01 0.8200000d-01
     0.8600000d-01 0.9000000d-01 0.9400000d-01 0.9800000d-01
     0.1050000d+00 0.1150000d+00 0.1250000d+00 0.1350000d+00
     0.1450000d+00 0.1550000d+00 0.1650000d+00 0.1750000d+00
     0.1850000d+00 0.1950000d+00 0.2100000d+00 0.2300000d+00
     0.2500000d+00 0.2700000d+00 0.2900000d+00 0.3100000d+00
     0.3300000d+00 0.3500000d+00 0.3700000d+00 0.3900000d+00
     0.4200000d+00 0.4600000d+00 0.5000000d+00 0.5400000d+00
     0.5800000d+00 0.6200000d+00 0.6600000d+00 0.7000000d+00
```

```
     0.7400000d+00 0.7800000d+00 0.8200000d+00 0.8600000d+00
     0.9000000d+00 0.9400000d+00 0.9800000d+00 0.1050000d+01
     0.1150000d+01 0.1250000d+01 0.1350000d+01 0.1450000d+01
     0.1550000d+01 0.1650000d+01 0.1750000d+01 0.1850000d+01
     0.1950000d+01 0.2100000d+01 0.2300000d+01 0.2500000d+01
     0.2700000d+01 0.2900000d+01 0.3100000d+01 0.3300000d+01
     0.3500000d+01 0.3700000d+01 0.3900000d+01 0.4200000d+01
     0.4600000d+01 0.5000000d+01 0.5400000d+01 0.5800000d+01
     0.6200000d+01 0.6600000d+01 0.7000000d+01 0.7400000d+01
     0.7800000d+01 0.8200000d+01 0.8600000d+01 0.9000000d+01
     0.9400000d+01 0.9800000d+01 0.1050000d+02 0.1150000d+02
     0.1250000d+02 0.1350000d+02 0.1450000d+02 0.1550000d+02
     0.1650000d+02 0.1750000d+02 0.1850000d+02 0.1950000d+02
     0.2100000d+02 0.2300000d+02 0.2500000d+02 0.2700000d+02
     0.2900000d+02
 df414     0.1499720d-10 0.1270379d-10 0.9590374d-11 0.7260551d-11
     0.5389905d-11 0.4015983d-11 0.3044076d-11 0.2304105d-11
     0.1728048d-11 0.1306937d-11 0.1024115d-11 0.8409772d-12
     0.7513795d-12 0.7839956d-12 0.9706841d-12 0.1393574d-11
     0.2291770d-11 0.3682004d-11 0.6765059d-11 0.1168265d-10
     0.2100153d-10 0.3750360d-10 0.6442264d-10 0.1086916d-09
     0.1893317d-09 0.3130153d-09 0.4860338d-09 0.5871450d-09
     0.6072574d-09 0.6262392d-09 0.6452152d-09 0.6631995d-09
     0.6945842d-09 0.7373987d-09 0.7779963d-09 0.8166296d-09
     0.8541321d-09 0.8895289d-09 0.9228083d-09 0.9561770d-09
     0.9863175d-09 0.1016496d-08 0.1060750d-08 0.1117050d-08
     0.1166198d-08 0.1225152d-08 0.1274160d-08 0.1324001d-08
     0.1364879d-08 0.1418400d-08 0.1466305d-08 0.1534897d-08
     0.1668848d-08 0.1690515d-08 0.1654776d-08 0.1712547d-08
     0.1771585d-08 0.1833119d-08 0.1894731d-08 0.1953690d-08
     0.2002194d-08 0.2062326d-08 0.2114104d-08 0.2169431d-08
     0.2236361d-08 0.2340934d-08 0.2523829d-08 0.2559173d-08
     0.2537249d-08 0.2643879d-08 0.2731478d-08 0.2778630d-08
     0.2853773d-08 0.2964955d-08 0.3003800d-08 0.3128441d-08
     0.3146253d-08 0.3242025d-08 0.3289957d-08 0.3420520d-08
     0.3571735d-08 0.3711655d-08 0.3839066d-08 0.4181967d-08
     0.4266899d-08 0.4369313d-08 0.4300694d-08 0.4428566d-08
     0.4438795d-08 0.4685329d-08 0.4575192d-08 0.4786270d-08
     0.4902241d-08 0.5030422d-08 0.5238293d-08 0.5514162d-08
     0.5449712d-08 0.5406671d-08 0.5568274d-08 0.5650407d-08
     0.5717521d-08 0.5840606d-08 0.5979110d-08 0.6392402d-08
     0.6377405d-08 0.6612861d-08 0.6865127d-08 0.7032033d-08
     0.7127277d-08 0.7212924d-08 0.7317430d-08 0.7442321d-08
     0.7584216d-08 0.7564227d-08 0.7512547d-08 0.7539243d-08
     0.7409569d-08
c
```