Exercise 6 – Neutron mode:

Return to your file *ex3.ip* or construct a new file with the simple geometry described above; save this file as *ex6.ip*. Replace the definition for the material lead with that for water (you will have to change both the material card and the corresponding cell densities). Change the mode of the problem from photon to neutron (N) mode and the source energy to 2MeV. You will also need to change the cross-section libraries by adding the suffix .42C to all of the material definitions as described above. Modify the tally card to measure the *neutron* fluence crossing the surface of the water sphere.

Run this simulation for sphere radii of 0.2 cm, 2.0cm and 20 cm, and examine the results.

Are the results reliable in each case? If not make the necessary changes to the model.

Plot the surface fluence as a function of energy for the three spheres

Can you explain the observed differences between each simulation?

Exercise 6a (Optional) – A more realistic source:

Now replace the source with a more realistic one: a Watt fission spectrum. This is much closer to the kind of energy spectrum found for a spontaneous fission neutron source. The description for this is achieved using a source distribution card for the neutron energy:

We define the source using the SDEF card as before, but now we add additional parameters. The parameter ERG defines the energy of the source (there are many other physical properties you can change using other parameters). The card tells MCNP to use the distribution labelled 4 (D4) for the input values to the parameter given. We then define the distribution which will be used by ERG. This is done using the SP card followed by the distribution number (SP4). The arguments to this card then define the distribution used, -3 is defined as a Watt fission spectrum, -2 defines a Gaussian distribution (there are other distributions available with different codes). The final input card is given below:

```
SDEF ERG=D4
SP4 -3
```

There are many other arguments which can be used in this card, for example a string of numbers can be used to define points in the distribution. The Watt spectrum is commonly used and has therefore been defined elsewhere; using the argument -3 the distribution takes the form:

$$P(E) = C \exp^{(-E/a)} \sinh(bE)^{\frac{1}{2}}$$

By default in MCNP a = 0.965 MeV and b = 2.29 MeV^{-1}.

Run this simulation for sphere of radius 2.0cm and examine the results.

Are the results reliable in each case? If not make the necessary changes to the model.

Plot the surface fluence as a function of energy, is it much different to exercise 6 above?

How has the computational time and statistical error changed compared to above?

Exercise 7 – Criticality simulation:

In this exercise you will use the MCNP criticality functions with the simple geometry you have developed over the previous exercises to investigate how critical systems can be modelled. The file *ex7a.ip* provided describes the geometry and sets up the criticality simulation, use this if you are running out of time.

With the KCODE simulation a number of cycles are simulated using the final conditions of the previous cycle to define the problem. At the end of each cycle any new fission events determine the spatial distribution of sources for the next cycle etc.

Place a 5 cm radius sphere of natural uranium, 99.3% U-238, 0.7% U-235 with a density of 19.2 g/cm^3 inside the water cylinder.

Replace the SDEF source with the KCODE criticality definition. You must decide upon the number of starting neutron sources per cycle, *nsrck*, an estimate of the criticality constant k_{eff}, *rkk*, the number of cycles to run before k_{eff} is calculated, *ikz*, the number of cycles to run, *kct*, and their placement *(x, y, z)* at the beginning of the first cycle.
The KCODE definition has the following format:

KCODE nsrck, rkk, ikz, kct

The argument *nsrck* needs to be large enough so there are some new fission events generated upon each cycle, likewise if the guess for *rkk* is inaccurate then too many (leading to a memory overflow) or too few neutrons are generated for the simulation of subsequent cycles. Before the calculation begins it is worth waiting until a steady state is reached (as initial conditions will not match the real situation), so the value of *ikz* skips calculation for the first set of cycles. Finally the number of cycles, *kct,* should run for long enough that the value of k_{eff} reaches a constant value.

To define the starting positions for the initial cycle use the KSRC input definition. The KSRC card defines the starting positions of neutrons within the modelled geometry. Any number of points can be specified in the KSRC card including just 1 point. In subsequent generations the distribution will become more realistic (as the simulation uses the previous cycle as input and eventually should reach a steady state). (SDEF can also be used as alternative to KSRC).

You will need to think about the initial starting positions for neutrons within the system. If uniform sampling of a volume is chosen, early values of k_{eff} may be too low as too many neutrons will start where escape is more likely. This is opposed to the starting bias introduced by using one central starting neutron.

Investigate the estimate of k_{eff}, its error, and the figure of merit with cycle number, this is displayed at run-time and given in the output file.

What happens to k_{eff} when the surrounding material (water in the original problem) is replaced with graphite (density 1.7 g / cm^3) ?

From your results conclude which is the better moderator, water or graphite?

As an extension you may wish to consider the neutron economy of the moderating material using a similar method to that given in exercise 6 where the neutron population in the material is determined. How does this affect the materials worth as a neutron moderator?