Exercise b1:

BACKGROUND: One of the most useful features of mcnp and serpent is the ability to perform burnup calculations. These are calculations in which the materials present in the simulation are allowed to evolve as a function of time, just like in a real reactor. A common way to perform these calculations is to create an input deck representing a reactor, and input a power at which to burn. The Monte Carlo codes will run transport calculations to compute a normalized neutron spatial and energy distribution, and the user-supplied power will be used to compute the resulting reaction rates and absolute neutron flux levels.

B1 is the same godiva sphere as used previously in g1, with a burnup calculation added. It is burned at a power of 5e6 W and data is collected at 10 days and then again at 110 days after the start of the simulation.

GOAL: Learn how to add burnup calculations to mcnp and serpent simulations.

INSTRUCTIONS:

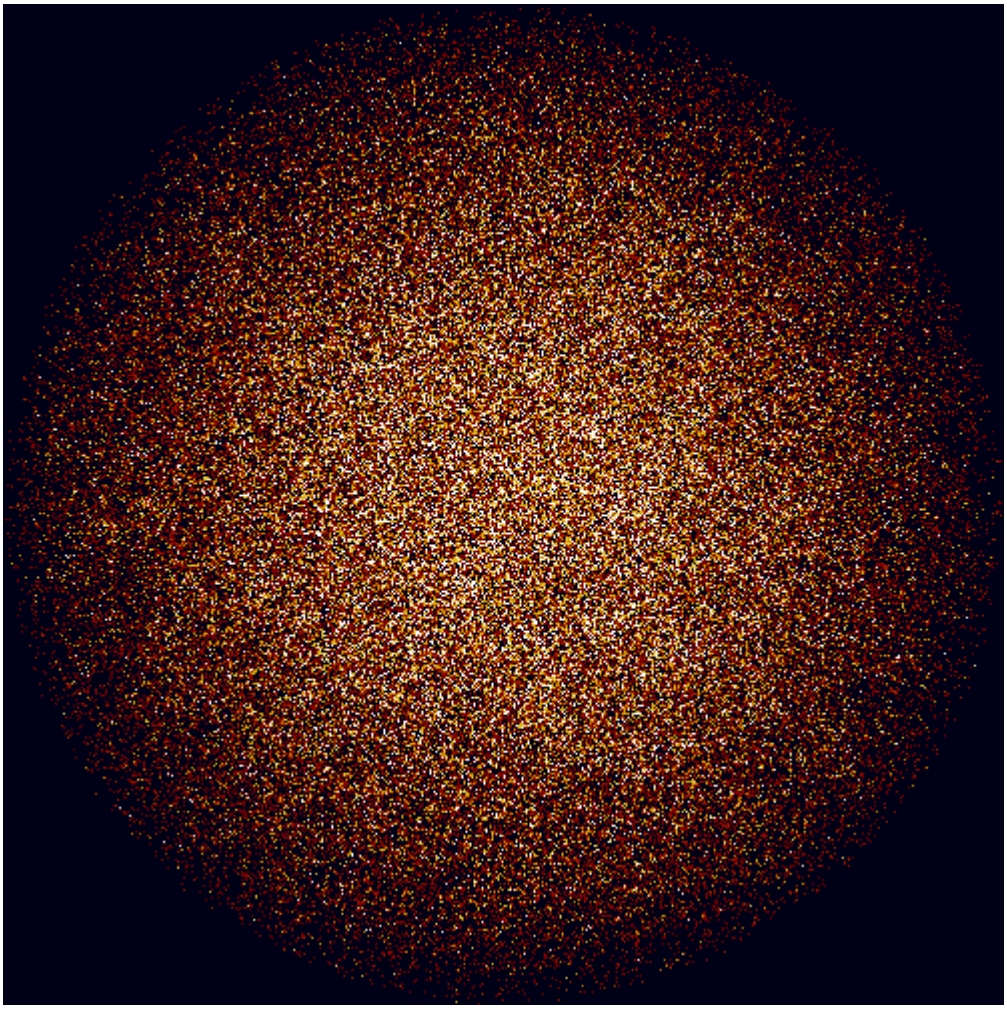
In mcnp:
Copy g1 to b1
Add information on the sphere's volume in the cell card for inside the sphere.
Add a "BURN" command in the data card of the input deck (last section)
Specify "time" to equal 10 and 100. These are time steps, this will measure time in 10 days and then again 100 days later.
Specify the power fraction "Pfrac" to equal 1 for each time step
Specify the "power" to equal 5 MW
Specify the material you want burned "mat"
Specify the burnup option using "bopt = 1.0 4"
Omit 2 isotopes from the calculations, 90234 and 91232. You may need to omit more depending on what libraries are installed on your computer.

In serpent:
Open the template provided
Add information on the sphere's volume in the material card using the command "vol"
Add a burn up of type 1 to the material
You may notice there are some new physics options for burnup in this deck. Feel free to look up their meanings
In the area provided for burnup set the power equal to 5e6 W
Below this add depletion steps using the command "dep"
Add steps at 10 days and 110 days. There are multiple ways of doing this. I like the "daytot" option but you can try one of the others.

EXERCISES:
a) Run both input decks.

b) Have a close look at the output files for each program. Which program is tracking more isotopes?

c) Notice that serpent conveniently allocates all of its outputs to a .m file, so feel free to open this in matlab. Take a look at what data is output by each program and compare any results.



Putting it all Together, Part 1.

So far you have learned to use many of the features available in Serpent and MCNP. A common use for these codes is reactor design, and the next series of exercises will instruct you how to simulate a reactor. The initial model will be simple, but will become increasingly complex as each exercise adds more realism. We will use Serpent for these exercises.

GOAL: To simulate a light water reactor (LWR) using simple geometry. To learn how to use the "pin" card in Serpent in order to define fuel pins.

BACKGROUND: Serpent provides a number of features to simplify the creation of input decks for reactor analysis. One of these features is the pin card which can be used to quickly define common pin type fuel geometry, without having to define separate surfaces and cells for fuel, cladding and air gap regions. The syntax for the pin card is shown in chapter 3.4 of the Serpent manual.

```
pin <id>
<mat 1> <r1>
<mat 2> <r2>
...
<mat n>
```

where	<id>	is the pin identifier (universe number)
	<mat 1> <mat 2> ...	are the materials
	<r1> <r2> ...	are the outer radii of the material regions

LWR fuel pins are typically uranium dioxide pellets stacked inside a cylinder of zircaloy cladding, and there is a small (but important) air gap between the cladding and the fuel. One way to use the pin card to define a LWR fuel pin could be:

```
pin 1
fuel    0.396
gap     0.416
clad    0.466
coolant
```

Here we specify a pin with the ID number 1, with the material fuel included up to a radius of 0.396cm, then the material gap occupying the next 0.02cm, then the material clad occupying the next 0.05cm giving an overall pin radius of 0.466cm. Everything beyond that is filled with the material coolant. It is up to the user to define the fuel material. The gap, cladding, and coolant materials are provided. The lattice pitch (center-to-center pin separation) should be 1.3 cm.

EXERCISES:

1) Modify the provided template to create an input deck which consists of a lattice of fuel pins in a water moderator, bounded by a cylinder 4m in height and diameter.

2) Run a criticality calculation on your reactor. Note down the k_{eff} value.

3) Increase the lattice pitch by 10% and re-run your simulation. Is your k_{eff} bigger or smaller, and why?

4) Reduce the water density by 10% and re-run your simulation. Is your k_{eff} bigger or smaller, and why?