NE506 HW5: Part II

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Abstract

What follows is a compilation of the responses to questions posed in Part III of homework #5 for NE506. All answers are already contained in the `readme.txt` files of the appropriate part's subdirectory, but they have been copied here for convenience.

Note that the input file in this directory does **not** contain the changes outlined in part 2.

Situation

Consider the simple aqueous reactor model in the file `hw5.2.i`. Assume a reactor power of 250 kW.

Part 1

Question

Determine the U-235 enrichment that will make this system critical. Give an answer to the nearest 0.1%.

Response

Via guess and check, the appropriate material was determined to be:

```
M1 1001 2 8016 1 92235 0.00649 92238 0.09351
```

This level of enrichment yields $k_{\text{eff}} = 0.99942$.

Part 2

Question

Show that this is subcritical with the central control rod fully inserted.

Code

Some changes were made to the geometry in order to model a fully inserted control rod. The lines

```
6   3 -4    -6   106
16  0       -6  -106
...
imp:n 1 10r 0
```

Were changed to:

```
6   3 -4    -6
...
imp:n 1 9r 0
```

Response

With these geometry changes, MCNP5 yields a new value: $k_{\text{eff}} = 0.96823$.

Part 3

Question

Create a Cartesian mesh tally with a 3 cm mesh over the entire system. Provide the following plots:

- across the core at axial heights of 30 cm, 55 cm, 75 cm
- on the y-z plane

Code

A mesh tally FMESH card was created for this problem.

```
c 3 cm mesh
FMESH4:n GEOM=xyz ORIGIN=-51 -51 -1
     IMESH=51  IINTS=34
     JMESH=51  JINTS=34
     KMESH=101 KINTS=34
```

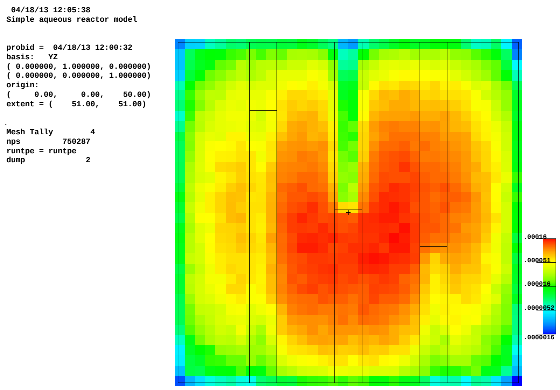
Results

The following plots (Fig 1) were obtained from the mesh tally:

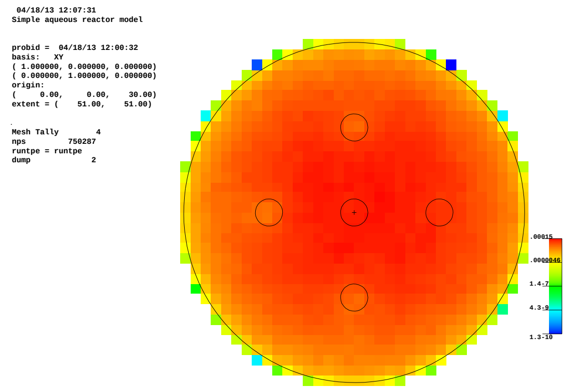
Part 4

Question

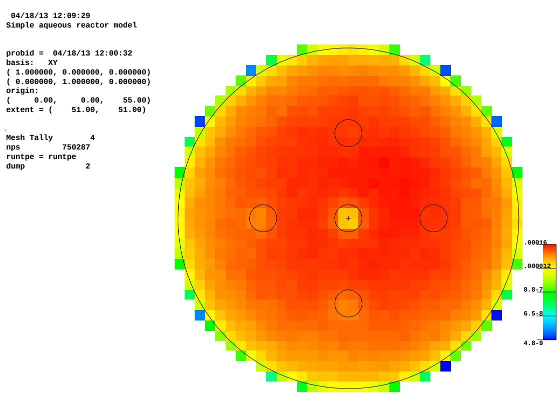
Approximate the dose to a worker by calculating the dose at a point 1 m off the ground and 5 m outside the reactor.



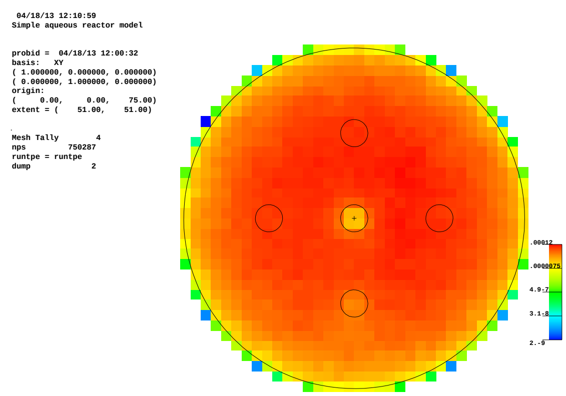
(a) The y-z plane.



(b) Across the core at an axial height of 30cm.



(c) Across the core at an axial height of 55cm.



(d) Across the core at an axial height of 75cm.

Figure 1: Plots of mesh tally results from various slices of the core.

Code

First, a fission "F7" tally was carried out to determine the source strength.

```
c Fission tally for source strength
F7:n (1 2 12 3 13 4 14 5 15 6 16)
SD7 1
FM7 1.6e-19
```

Then, using the results from the fission tally, the dose could be calculated.

```
c Detector tally
F5:n 0 550 100 0.1
c Neutron dose multipliers
DE5 2.5e-8 1e-7 5ILOG 1e-1 5e-1 1 2 5 10 20
DF5 3.85e-6 4.17e-6 4.55e-6 4.35e-6 4.17e-6 3.7e-6 3.57e-6
     2.08e-5 7.14e-5 1.18e-4 1.43e-4 1.47e-4 1.47e-4 1.54e-4
FM5 2.116473785E+16
```

Additionally, to ensure that the detector was not placed in a region of zero importance, some additional geometry was defined[1].

```
C Cell cards
1   1 -1    -1 2 3 4 5 6
2   2 -1    -2 -102
12  0       -2  102
3   2 -1    -3 -103
13  0       -3  103
4   2 -1    -4 -104
14  0       -4  104
5   2 -1    -5 -105
15  0       -5  105
6   3 -4    -6  106
16  0       -6 -106
101  0   1 -9
100  0   9

C Surf cards
9 RPP -50 50 -50 560 0 110
1 RCC   0    0 0 0 0 100 50
2 RCC  25    0 0 0 0 100  4
3 RCC   0   25 0 0 0 100  4
4 RCC -25    0 0 0 0 100  4
5 RCC   0  -25 0 0 0 100  4
6 RCC   0    0 0 0 0 100  4
102 pz 20
103 pz 40
104 pz 60
105 pz 80
106 pz 51
...
imp:n 1 11r 0
```

Response

First, MCNP5 was used to determine the fission energy tally for the geometry. The tally yielded $1.18121 \times 10^{-17} \frac{\mathrm{MW}}{\frac{n_{\mathrm{src}}}{\mathrm{s}}}$. To find the source strength, we take the reactor power, 250 kW, and divide by this value:

$$\frac{0.25\,\mathrm{MW}}{1.18121 \times 10^{-17} \frac{\mathrm{MW}}{\frac{n_{\mathrm{src}}}{\mathrm{s}}}} = 2.116473785 \times 10^{16} \frac{n_{\mathrm{src}}}{\mathrm{s}} \tag{1}$$

Now we can use this source strength as a multiplier to the detector tally in order to approximate the neutron dose at that point. We run MCNP5 again to find that the dose associated with uncollided neutron flux is $4.36349 \times 10^4 \ \frac{\mathrm{rem}}{\mathrm{hr}}$.

[1] Note that cell 101 and surface 9 have been added to the default geometry and cell 101 has been given a nonzero neutron importance.