1) Principles of Fission
   a. On average, how much energy is released by a fission reaction in total?
   b. How is this energy distributed among its products?
   c. What are the two ranges of A values for prompt, thermal neutron induced fission products that you expect?
   d. Select two realistic fission products and write a reaction you would expect.
   e. What is the difference between fissile, fissionable, and fertile nuclei?
   f. What is the significance of Pu-239 in reactors? Where does it come from?

2) A material has a total microscopic neutron cross section of $2.5 \times 10^{-24}\ cm^2$, and contains $5.20 \times 10^{23}\ \mathrm{nuclei/cm^3}$

a) What is the macroscopic cross section?

b) What is the mean free path of neutrons in this material?

c) If neutrons impinge perpendicularly on a slab of the material that is 2.0 cm thick, what fraction of them will penetrate the slab without making a collision?

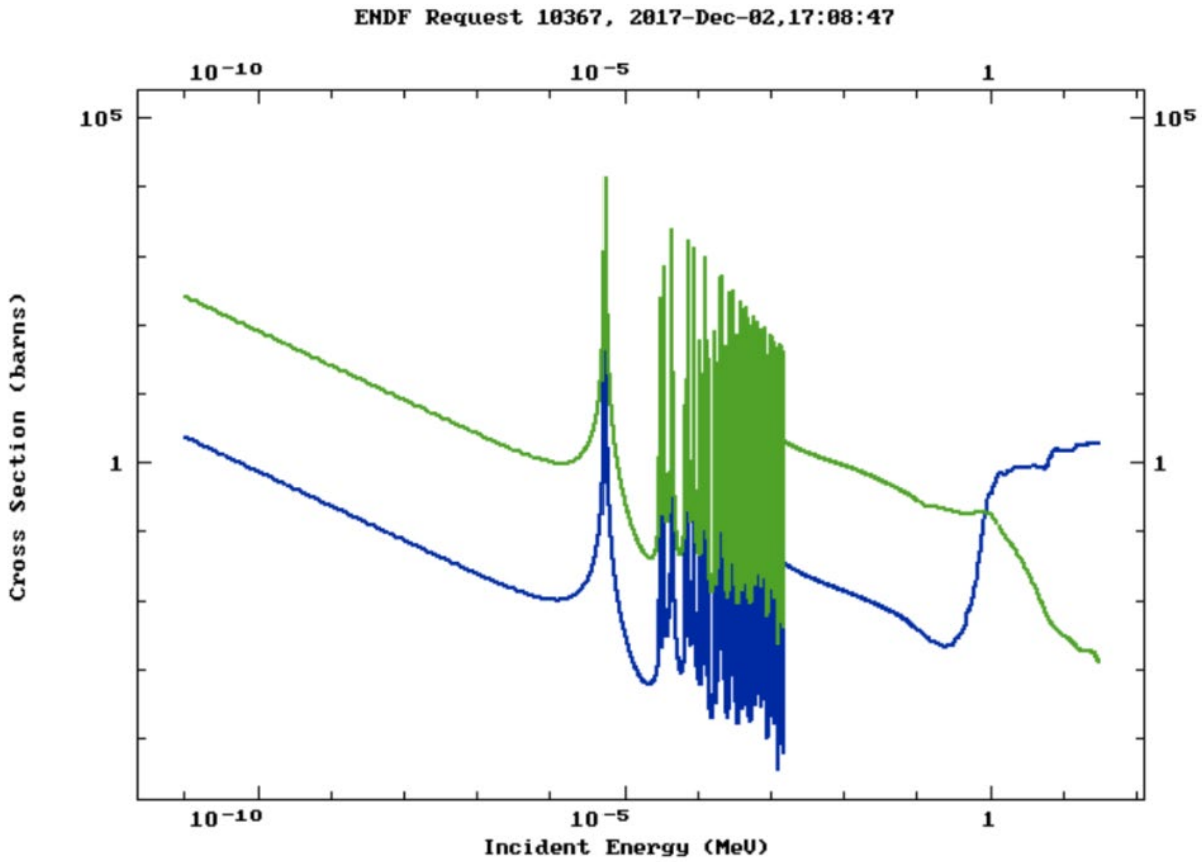
d) If the neutron beam has an intensity of $7 \times 10^7\ \frac{n}{cm^2 s}$, and a cross sectional beam area of $3\ cm^2$, how many collisions per second will the neutrons undergo?

3) A light water reactor is operating at a steady state power of 1000 MWth. Assume all of the neutrons are thermal.

a. Estimate the fission rate of the reactor.
b. How many neutrons are being produced from fission per second?
c. How many neutrons are either being absorbed or leaking out of the core per second?

4) Free neutrons ($mass = 1.67 \times 10^{-27} kg$) undergo $\beta^-$ decay with a half life of 10.4 minutes. Determine the probability that a 5 eV neutron will decay before being absorbed in an infinite, purely-absorbing material with $\Sigma_a = 0.022\ cm^{-1}$.

5) The $(n, f)$ and $(n, \gamma)$ reactions are shown below for U-236. Which cross section is which?



6) Define the following:

a. Macroscopic cross section (definition and units)
b. Fission reactor rate in a homogenous reactor with volume V (definition, formula, units)
c. Neutron mean free path
d. Fuel enrichment
e. Natural abundance

7) Scattering

a. What is the difference between elastic and inelastic scattering?
b. What does it mean if scattering is isotropic?

8) Say you have a reactor that is a cube with an edge length of 0.8m composed of a homogenous fissile material. The thermal neutron flux is uniform and $1 \times 10^{15}\ n/cm^2 s$. The macroscopic thermal fission cross section is 0.1 $cm^{-1}$. What is the total reactor power in MW?

9) Fermi's Six Factor Formula

a. Define each term in the six factor formula.
b. Draw a diagram that shows the possible events that can happen to a neutron in a reactor.
c. What are two assumptions used in the derivation of the six factor formula?
d. Write down the four factor formula and state when it is applicable.
e. What assumptions do you make when you express the multiplication factor as:

$$\frac{\nu\Sigma_f^{fuel}\Sigma_f^{fuel}}{\Sigma_a^{fuel}+\Sigma_a^{other}}$$