OpenMC HW 1

Problem 1
Run the sample code provided. This will verify that OpenMC is working correctly. Report the "Combined k-effective" value (expect it to be very sub-critical).

Problem 2
Let's prepare to model a fuel pin assembly. Change the ceramic fuel to be HALEU (High-Assay Low-Enriched Uranium). Create a new material to be used in the fuel pin air gap; we will use pure argon. Make sure you specify the composition, temperature, and density of this new material, and make sure it's exported with the other materials. You should be specifying elements, not nuclides, where possible. Report your edited materials section.

Problem 3
Now we will design the fuel pin geometry. This will consist of a cylindrical fuel pellet with a diameter of 2 cm, a 2 mm thick air gap, and a 5 mm thick cylindrical cladding. The fuel pin should be surrounded by water. Report two "Universe.plot()" images of the new geometry, from the 'xy' and 'xz' bases, respectively.

Problem 4
Run and report k_{eff} (again, should be very sub-critical).

Problem 5
Now model an infinite fuel pin assembly of infinite height. Report the new k_{eff}. What is the geometric arrangement of the fuel pins? What is the pitch between each fuel pin?

Problem 6
Revert to a fuel enrichment of 3.0 ao%. Change the pitch between the fuel pins until you reach steady-state criticality (k_{eff} = 1.0). What is the new pitch?

Problem 7
Change the particle source such that neutrons are generated randomly throughout the fuel pellet. A box source can be used instead of a point source:
https://docs.openmc.org/en/stable/pythonapi/generated/openmc.stats.Box.html
How does this affect the k_{eff}? Is this new k_{eff} more accurate, or less?

Problem 8

Calculate and report the rate of fission reactions in the fuel pellet (with correct units).