Problem 1.

Compute the macroscopic fission cross section of 1, 2, 3, 4, and 5 wt% enriched UO_2 with a density of $10.5 g/cm^3$. Use the $2200 m/s$ microscopic cross sections provided in Appendix A (pp. 606-610 in D&H). What is the probability that a neutron will be absorbed in ^{238}U (relative to all absorptions) in these mixtures?

Solution

Problem 2.

Using the Table of Nuclides at: http://atom.kaeri.re.kr/ton or that found at http://www.nndc.bnl.gov/sigm
find the following information:

Part (a)

What is the total fission cross section of U-233, U-235, Pu-239, and Pu-241 at 0.0253 eV?

Part (b)

What is the accumulated fission yield of ^{90}Sr from the thermal fission of ^{235}U?

Part (c)

Obtain a plot of the ^{241}Pu total absorption cross section at 300K over the range $10^{-9} to 20 MeV$. Print out the plot and label the major features of the cross section behavior. Turn in the labeled plot.

Solution

Problem 3.

How many collisions are required to slow down a neutron from 2MeV to 0.025 eV in

Part (a)

Hydrogen,

Part (b)

Deuterium,

Part (c)

Graphite, and

Part (d)

Lead?

Solution

From *Lamarsh*, Table 3.1:

Problem 4. Duderstadt & Hamilton, problem 3-8

Consider an infinitely large homogeneous mixture of ^{235}U and a moderating material. Determine the ratio of fuel-to-moderator density that will render this system critical for the following moderators: (a) graphite, (b) beryllium [*sic*], (c) water (H_2O), and (d) heavy water (D_2O). Use the thermal cross section data given in Appendix A.

Solution

Problem 5. Duderstadt & Hamilton, problem 3-1

What is the maximum value of the multiplication factor that can be achieved in any conceivable reactor design?

Solution

We know that the multiplication factor for a thermal reactor core is:

$$k_{eff} = \epsilon p \eta P_{fastnonleakage} P_{thermalnonleakage} f$$

. We can assume that an ideal core would be large enough to allow for no leakage, so $P_{fastnonleakage} = P_{thermalnonleakage} = 1$.