ENGY.4340 Nuclear Reactor Theory

Fall 2016

HW #1 -- Review of Some Key Concepts from Fundamentals of NSE

Problem 1 Neutron Density, Flux, and Current (10 points)

Two beams of 2 eV neutrons intersect at an angle of 60^o. The density of neutrons in both monoenergetic beams is 5×10^7 neutrons/cm^3.

a. Compute the neutron intensity (i.e. flux) of each beam.

b. What is the neutron flux where the two beams intersect?

c. What is the neutron current where the two beams intersect?

State any assumptions and explain/justify your assumed geometry…

Problem 2 Neutron Density, Flux, and Current (10 points)

Consider two beams of monoenergetic neutrons traveling along the same guide tube. The guide tube contains a vacuum so there are no interactions within the tube. One beam has neutrons with an energy of 10 keV and density of 1×10^6 n/cm^3 and the other beam contains neutrons at 1 eV with a density of 2×10^7 n/cm^3.

Compute the energy-integrated neutron flux and energy-integrated net neutron current in the guide tube for the following two cases:

1. beams travel in the same direction
2. beams are in the opposite direction

State any assumptions and explain/justify your result…

Problem 3 Macroscopic Cross Sections for Uranium Carbide (10 points)

The fuel for an experimental thermal reactor contains uranium carbide (UC) with the uranium enriched to 4.8 w/o. The density of UC is 13.6 g/cm^3.

a. With this information and suitable microscopic cross sections (see Note below), calculate the macroscopic thermal (at E = 0.0253 eV) absorption and fission cross sections for the uranium carbide fuel pin.

b. Assuming 1/v behavior, also compute both macroscopic cross sections at E = 1 eV.

Note: There are several good sources for thermal microscopic cross section data on the web. Unfortunately, most of these only tabulate the cross sections for the individual isotopes, which then requires that the user perform a weighted average to obtain data for the naturally occurring elements. However, the tabulated data at the ***Japan Nuclear Data Center*** give information for both the naturally occurring elements and the individual isotopes. In addition, the ***Tables of Nuclear Data*** at this website (http://wwwndc.jaea.go.jp/NuC/) also contain lots of other useful information (the non-1/v factor, the resonance integral, etc.). Thus, I find this website quite useful for obtaining the data needed for performing preliminary reactor physics calculations -- and I suggest that you become familiar with this website and use it to obtain the needed microscopic data for problems like this throughout the semester.

Problem 4 Beam-Target Interactions (10 points)

a. Let ϕ_o be the mono-directional mono-energetic uniform neutron flux incident on a thick slab target of thickness L and surface area A. If the target material has $\sigma_a >> \sigma_s$ so that $\sigma_t \approx \sigma_a$, derive an expression for the rate at which neutrons interact (i.e. get absorbed) in the full target.

b. In a specific case, the incident beam intensity is 4×10^{10} neutrons/cm^2-sec, the target cross sectional area is 1.2 cm^2 and the target is 5.6 cm thick. The target atom density is 2.5×10^{22} atoms/cm^3 and the total cross section at the energy of the beam is 24.8 b. For this specific situation, how many neutron interactions per second occur in the target?

Hint: Integration over the spatial variable is usually necessary when computing total reaction rates when the collision density varies with position (i.e. which is the case with thick planar targets).

Problem 5 Activation Analysis (10 points)

A small 0.12 g gold sample (pure Au197) is placed in an experimental location in a research reactor where the average neutron flux is 5×10^{11} n/cm^2-s. The properly averaged capture cross section for Au197 is approximately 85 b. The sample is irradiated for 4 hr after which it is removed from the neutron field. During the irradiation, radioactive Au198 is produced at a constant rate via neutron capture in Au197 (that is, the flux is constant and the amount of Au197 does not change significantly during the irradiation interval). The half-life of Au198 is about 2.7 days and its neutron absorption cross section is small.

Based on the above description, estimate the activity (in curies) of the gold sample at the following times:

a. immediately upon removal from the reactor and

b. 72 hrs after removal from the reactor.

Explain your solution logic and support your results with a set of formal calculations/analyses.