ENGY.3310 Fundamentals of Nuclear Science and Engineering

Spring 2016

HW #8: Neutron Cross Sections

Problem 1 Partial Cross Sections (5 points)

Several cross sections for U-235 at 1 MeV are as follows:

$\sigma_{el} = 4.0$ b $\quad$ $\sigma_{inel} = 1.4$ b $\quad$ $\sigma_f = 1.2$ b $\quad$ $\sigma_a = 1.3$ b

where we will assume that all the remaining cross sections for any neutron-producing and charged-particle reactions are negligible.

Under these conditions what are the total cross section and the capture-to-fission ratio in U-235 at 1 MeV?

Problem 2 Beam-Target Interactions (5 points)

A monoenergetic beam of neutrons, $I = 4\times10^{10}$ neutrons/cm^2-sec, impinges on a target 1 cm^2 in area and 0.1 cm thick. The target atom density is 4.8×10^{22} atoms/cm^3 and the total cross section at the energy of the beam is 4.5 b.

a. What is the macroscopic cross section?

b. How many neutron interactions per second occur in the target?

Problem 3 Beam-Target Interactions (5 points)

Let ϕ_o be the mono-directional mono-energetic neutron flux incident on a thick slab target of thickness L and surface area A. If the target material has $\sigma_a >> \sigma_s$, derive an expression for the rate at which neutrons are absorbed in the full 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 4 Beam-Target Interactions (10 points)

A beam of 2 MeV neutrons is incident on a slab of heavy water (D_2O). The total cross sections of deuterium and oxygen at this energy are about 2.6 b and 1.6 b, respectively. Pure heavy water has a density at room temperature of about 1.1 g/cm^3.

a. What is the macroscopic total cross section of D_2O at 2 MeV?

b. How thick must the slab be to reduce the intensity of the uncollided beam by a factor of 10?

c. If an incident neutron has a collision in the slab, what is the probability that it collides with deuterium?

Problem 5 SS-304 Macroscopic Thermal Absorption Cross Section (10 points)

Type 304 stainless steel, with a density of 7.86 g/cm^3, is often used as a structural material in a variety of nuclear systems. The nominal composition by weight of this material is as follows:

carbon 0.08% chromium, 19% nickel, 10% and iron, the remainder.

With these data and suitable microscopic cross sections, calculate the macroscopic absorption cross for SS-304 at 0.0253 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, you might want to become familiar with this dataset, and use it to obtain the needed microscopic data for the various components of SS-304...

Problem 6 Beam-Target Interactions and Mean Free Paths (10 points)

a. Determine the shield thickness needed (in number of mean free paths) to attenuate an incident neutron beam by a factor of 1000?

b. Boron is a common material used to shield against thermal neutrons. Estimate the thickness (cm) of natural boron at a density of 2.3 g/cm^3 needed to attenuate an incident thermal neutron beam to 0.1% of its original intensity.

c. Convert your result from Part b into mean free paths and comment on the result. Is this what you expected?

Problem 7 Point Source with and without a Shield (10 points)

Consider an isotropic point source and sample geometry where the sample is a 0.2 cm thick aluminum foil with a 5 cm^2 area that faces the source. The thermal neutron source strength is 10^8 neutrons/s. The aluminum sample density is 2.7 g/cm^3 and the total cross section for thermal neutrons is about 1.67 barns.

a. Estimate the uncollided flux at the sample location and the total Al interaction rate (reactions/sec) if the sample is placed 1 m from the source.

b. Re-compute the uncollided flux at the sample and the Al interaction rate if an 8 cm thick shield with $\Sigma_t = 0.5$ cm^{-1} is placed between the source and Al sample (assume that the total distance of 1 m between the sample and the source is unchanged from Part a).

Note: Neglect any neutron collisions in the air (that is, assume that the air represents a vacuum).

Problem 8 Cross Section Behavior versus Energy using the JANIS Code (15 points)

Go to the following website: http://www.oecd-nea.org/janis/download.html and download the JANIS program. This is a flexible java-based plotting program which allows you to plot various cross section profiles versus energy. It puts a significant database of nuclear data at your finger tips. For this exercise, you should become familiar with this program and use it to gain insight for the $\sigma(E)$ behavior for several types of materials and neutron reactions. In particular, you should plot and contrast/compare the following cross sections (each item should be a single plot, giving 7 separate plots for discussion and comparison):

1. Typical Fuel Material: U235 total, elastic, inelastic, fission, and capture cross sections
2. Typical Moderator Materials: H-1 elastic and capture **&** C_{nat} elastic and capture cross sections
3. Typical Structural Material: Zr_{nat} total, elastic, inelastic, and capture cross sections
4. Typical Poison Material: Cd113 total, elastic, inelastic, and capture cross section
5. Location of Resonance Region: Zr_{nat} total **&** U238 total cross sections
6. Fissile and Fissionable Materials: U235 fission **&** U238 fission cross sections
7. Behavior of Inelastic Cross Sections: Zr_{nat} inelastic **&** U238 inelastic cross sections

Carefully look at each plot and identify any key features or comparisons that exist (you should be able to find at least one or two interesting observations from each plot). The goal here is for you to gain some general insight into the behavior of $\sigma(E)$ for different materials and reaction types. We discussed some of these reactions in class, but now it is your turn to generate the plots and describe the $\sigma(E)$ behavior -- and your discussion should indicate what you have learned from this exercise!!!

Note: For consistency, select the NEA ENDFB-VI.8 (ver. 6.8) library for all your plots. The energy axis should be set to go from 0.001 eV to 20 MeV for each plot (except possibly for the inelastic cross section plot). Also set the y-axis to display a reasonable range (i.e. $\sigma < 0.1$ mb is negligible for most applications).

MT numbers: total = 1, elastic = 2, inelastic = 4, fission = 18, capture = 102, and n,α = 107