

# Nuclear Engineering 150 – Discussion Section

## Team Exercise Solutions #4

### Problem 1

A reactor contains 5% (by atom) enriched uranium dioxide, which is 15% of the entire core by volume.

- Calculate the macroscopic cross section for this core if we were to treat it as a homogeneous volume.
- If the reactor were a cube with a side length of 4 m and a beam of  $10^{15}$  thermal neutrons were incident on one face of the cube, how many neutrons would we expect to make it through to the other side?

Nucleus	Thermal $\sigma_t$ (b)	Mass (g/mol)
$^1\text{H}$	20.8	1.008
$^{16}\text{O}$	3.5	15.995
$^{235}\text{U}$	607.5	235.044
$^{238}\text{U}$	11.8	238.050
Compound	$\rho$ (g/cm <sup>3</sup> )	
$\text{H}_2\text{O}$	1.0	
$\text{UO}_2$	10.4	

### Problem 1 Solution

- A macroscopic cross section for a homogeneous mixture of materials can be found by summing the cross sections of those components, weighted by their volume fractions:

$$\Sigma = \sum_i \frac{V_i}{V_t} \Sigma_i \quad (1)$$

where  $\Sigma_i$  is the macroscopic cross section of the  $i^{\text{th}}$  material,  $V_i$  is the volume of the  $i^{\text{th}}$  material, and  $V_t$  is the total volume. For this problem, we are told that the fraction  $\frac{V_{\text{UO}_2}}{V_t} = 0.15$  and  $\frac{V_{\text{H}_2\text{O}}}{V_t} = 0.85$ . Using these in equation (1) the total macroscopic cross section of our reactor can be expressed just in terms of the macroscopic cross sections of  $\text{UO}_2$  and  $\text{H}_2\text{O}$ .

$$\Sigma = 0.15\Sigma_{\text{UO}_2} + 0.85\Sigma_{\text{H}_2\text{O}} \quad (2)$$

Calculating these two macroscopic cross sections requires knowledge of each of the individual microscopic cross sections of the compounds, since the macroscopic cross section is defined as

$$\Sigma_i = n_i \sigma_i, \quad (3)$$

where  $n_i$  and  $\sigma_i$  are respectively the number density and total microscopic cross sections of the compounds. While neither of these quantities are given explicitly, we can calculate each from the information provided.

We can find a compound's number density by dividing the weight density by the mass of one molecule of the compound.

$$n_i = \frac{\rho_i}{m_i}$$

The densities for  $\text{UO}_2$  and  $\text{H}_2\text{O}$  are provided, and we can find the mass of a molecule by dividing the molar mass of the material by Avogadro's number,  $N_A = 6.022 \times 10^{23}$  molecules/mol.

$$m_i = \frac{M_i}{N_A},$$

and so

$$n_i = \frac{\rho_i N_A}{M_i}.$$

For our two compounds,  $\text{UO}_2$  and  $\text{H}_2\text{O}$  these number densities are

$$n_{\text{UO}_2} = \frac{\rho_{\text{UO}_2} N_A}{M_{\text{UO}_2}} \quad \text{and} \quad n_{\text{H}_2\text{O}} = \frac{\rho_{\text{H}_2\text{O}} N_A}{M_{\text{H}_2\text{O}}}.$$

Before moving on to finding the microscopic cross sections, we can note that the molar mass of the compounds can be broken into the sum of the molar masses of their constituents according to their fraction in the compound.

$$n_{\text{UO}_2} = \frac{\rho_{\text{UO}_2} N_A}{f_{\text{U5}} M_{\text{U235}} + f_{\text{U8}} M_{\text{U238}} + 2M_{\text{O}}} \quad \text{and} \quad n_{\text{H}_2\text{O}} = \frac{\rho_{\text{H}_2\text{O}} N_A}{2M_{\text{H}} + M_{\text{O}}}.$$

Next, we look at microscopic cross sections. The microscopic cross section of a compound is given by the sum of the microscopic cross sections of its elemental components. The complete microscopic cross section for water can be found using this fact.

$$\sigma_{\text{H}_2\text{O}} = 2\sigma_{\text{H}} + \sigma_{\text{O}}$$

The microscopic cross section for  $\text{UO}_2$  can be found similarly, however the calculation is slightly more complicated due to its enrichment in  $^{235}\text{U}$ . First, we find the average microscopic cross section for uranium by weighting the cross sections of  $^{235}\text{U}$  and  $^{238}\text{U}$  by their abundance.

$$\sigma_{\text{U}} = f_{\text{U5}} \sigma_{\text{U5}} + f_{\text{U8}} \sigma_{\text{U8}}$$

where  $f_i$  is the atomic fraction of material  $i$  in the compound (for uranium in this case, this fraction is just the enrichment in atom %). Then, we can complete the process exactly as we did for  $\text{H}_2\text{O}$ ,

$$\begin{aligned} \sigma_{\text{UO}_2} &= \sigma_{\text{U}} + 2\sigma_{\text{O}} \\ &= f_{\text{U5}} \sigma_{\text{U5}} + f_{\text{U8}} \sigma_{\text{U8}} + 2\sigma_{\text{O}}. \end{aligned}$$

Combining these number densities and macroscopic cross sections, we can expand and rewrite the macroscopic cross sections described by equation (3) as Now we can substitute the given values.

$$\begin{aligned} \Sigma_{\text{UO}_2} &= \frac{\left(10.4 \text{ g/cm}^3\right) (0.05(607.5 \text{ b}) + 0.95(11.8 \text{ b}) + 2(3.5 \text{ b})) (6.022 \times 10^{23} \text{ mol}^{-1})}{0.05(235.044 \text{ g/mol}) + 0.95(238.050 \text{ g/mol}) + 2(15.995 \text{ g/mol})} \\ \Sigma_{\text{UO}_2} &= 112.7 \text{ m}^{-1} \end{aligned}$$

$$\begin{aligned} \Sigma_{\text{H}_2\text{O}} &= \frac{\left(1.0 \text{ g/cm}^3\right) (2(20.8 \text{ b}) + 3.5 \text{ b}) (6.022 \times 10^{23} \text{ mol}^{-1})}{2(1.008 \text{ g/mol}) + 15.995 \text{ g/mol}} \\ \Sigma_{\text{H}_2\text{O}} &= 150.8 \text{ m}^{-1} \end{aligned}$$

and plug these into equation (2) for the total cross section of the core.

$$\Sigma = 0.15(112.7 \text{ m}^{-1}) + 0.85(150.8 \text{ m}^{-1})$$

$$\boxed{\Sigma = 145.085 \text{ m}^{-1}}$$

b) Neutron attenuation follows a decaying exponential according to the equation

$$N = N(0) e^{-\Sigma x}$$

where  $N$  is the number of neutrons (you may be more familiar with this equation in terms of intensity; we have just eliminated the area and rate components from both sides).

We let  $N(0)$  be our initial number of incident particles,  $10^{15}$ ,  $\Sigma$  be the macroscopic cross section calculated in part (a), and  $x$  be the distance traveled by the neutrons, 4 m.

$$N = 10^{15} e^{(-145.085 \text{ m}^{-1})(4 \text{ m})}$$

This works out to be about  $9.153 \times 10^{-238}$ , or zero particles.

$$N = 0$$

Note that this number only indicates the quantity of particles which make it through the core uncollided. Since many of these interactions are scattering collisions, and not absorptive events, it is quite possible that some neutrons will eventually make it through the reactor. A more complicated derivation would be required in that case.

## Problem 2

A neutron beam with an intensity of  $2 \times 10^{12}$  neutrons/(cm<sup>2</sup>·s) is incident on an unknown shielding material and has a beam spot of 5 cm<sup>2</sup>. The shielding material has a thickness of 10 cm.

- On average,  $3.0 \times 10^9$  neutrons/s make it through the shield uncollided. What is the macroscopic cross section of the shield material?
- What is the mean free path of a neutron in the shielding material?
- If a single beam pulse is 10  $\mu$ s, how many collisions are expected to take place in the shielding material?

## Problem 2 Solution

- The transmitted uncollided intensity of a neutron beam through a material with macroscopic cross section  $\Sigma$  is given by

$$I = I(0) e^{-\Sigma x}.$$

Solving for  $\Sigma$ , we find

$$\Sigma = \frac{1}{x} \ln \left( \frac{I(0)}{I} \right)$$

Using the values provided (and considering the intensity as per the target, rather than per cm<sup>2</sup>), we have

$$\Sigma = \frac{1}{10 \text{ cm}} \ln \left( \frac{2 \times 10^{12} \text{ neutrons}/(\text{cm}^2 \cdot \text{s}) (5 \text{ cm}^2)}{3.0 \times 10^9 \text{ neutrons/s}} \right)$$

$$\Sigma = \frac{1}{10 \text{ cm}} \ln (3.33 \times 10^3)$$

$$\boxed{\Sigma = 0.811 \text{ cm}^{-1}}$$

- The mean free path of a particle is defined as  $\lambda \equiv \frac{1}{\Sigma}$ .

$$\boxed{\lambda = 1.233 \text{ cm}}$$

- We are told that the beam pulse is 10  $\mu$ s, so when considered in conjunction with the known intensity of the neutron beam, we can find the total number of particles produced by the beam per area. We define the neutron fluence,  $\Phi$ , as the number of neutrons per cm<sup>2</sup>, calculated as

$$\Phi = It = (2 \times 10^{12} \text{ neutrons}/(\text{cm}^2 \cdot \text{s}))(10^{-5} \text{ s}) = 2 \times 10^7 \text{ neutrons}/\text{cm}^2$$

The number of interactions,  $R$ , is

$$R = \Phi \Sigma V$$

where  $\Phi$  is the incident neutron fluence in  $\left[ \frac{\text{neutrons}}{\text{cm}^2} \right]$ ,  $\Sigma$  is the macroscopic cross section of the shielding material in  $\left[ \frac{1}{\text{cm}} \right]$ , and  $V$  is the volume of the shield in  $\left[ \text{cm}^3 \right]$ .

We have already found  $\Phi$ , we calculated  $\Sigma$  in part (a), and we can determine  $V$  by multiplying the area of the beam spot by the thickness of the target.

$$V = 5 \text{ cm}^2 \times 10 \text{ cm} = 50 \text{ cm}^3$$

The total number of collisions is then

$$R = (2 \times 10^7 \text{ cm}^{-2})(0.811 \text{ cm}^{-1})(50 \text{ cm}^3)$$

$$\boxed{R = 8.11 \times 10^8 \text{ collisions}}$$