國 立 清 華 大 學 命 題 紙

九十一學年度 工程與系統科學 系（所） 丙 組碩士班研究生招生考試

科目 核工原理 科號 3802 共 二 頁第 一 頁 *請在試卷【答案卷】內作答

1. Describe the difference between following terms:
   a) elastic scattering (n,n) vs. inelastic scattering (n,n')
   b) fissile vs fertile
   c) thermal shield vs. biological shield.

   (12%)

2. Evaluate the activity of 5g of ^{239}Pu. What will the activity be 10,000 years from now?
   (Note: half-life of ^{239}Pu is 24,390 year)
   (10%)

3. Calculate the mass attenuation coefficient of a fuel pellet (UO_2 having a density of 10 g/cm^3) for 1 MeV gamma rays. What is their mean free path?
   (Note: The mass attenuation coefficient of U and O are 0.076 and 0.0636 cm^2/g, respectively.)
   (10%)

4. (a)What is fission product yield? Plot the fission product yield curve for thermal fission in ^{235}U.
   (b)What is Bragg curve? Plot the curve.
   (c)Let η be the number of neutrons released in fission per neutron absorbed by fuel.
   Calculate the value of η for natural uranium at 0.0253 eV.

   Data: ^{238}U has $\sigma_f = 0$, $\sigma_a = 2.71\,barn$, $abundance = 99.27\%$

   ^{235}U has $\sigma_f = 582.2\ barn$, $\sigma_a = 680.8\ barn$, $\nu = 2.418$.

   (ν is the average number of neutrons released per fission.)

   (18%)

5. Two thermal reactors, one is homogeneous and the other is heterogeneous. If the amount of fuel and moderator in the reactor core are the same for both systems, how does the multiplication factors k_∞ compared with each other? And why?
   (10%)

6. In the criticality calculation of thermal reactors it is often necessary to use two-group diffusion method.

(a)Write down the two-group diffusion equations using appropriate two-group constants such as:

diffusion coefficient: D_1, D_2;

transfer cross section from group 1 to group 2: $\Sigma_s^{1\text{-}2}$;

absorption cross section: Σ_{a1}, Σ_{a2};

removal cross section: Σ_{R1} ($=\Sigma_{a1} + \Sigma_s^{1\text{-}2}$), Σ_{R2} ($=\Sigma_{a2}$);

fission cross section: Σ_{f1}, Σ_{f2};

no. of neutron released per fission: ν_1, ν_2;

(b) For bare reactors, all group fluxes have the same spatial dependence. Derive the critical equation.

(20%)

7. (a) Describe the difference between "prompt (fission) neutrons" and "delayed neutrons", especially in terms of where they are originated from.

(b) Consider an infinite homogeneous thermal reactor which may or may not be critical. Let

n(t) be the thermal neutron density,

$\phi(t)$ ($= n v$) be the thermal neutron flux,

Σ_a be the thermal absorption cross section,

Σ_f be the thermal fission cross section,

ν be the no. of neutrons released per fission

Assuming all the fission neutrons are prompt, write down the time-dependent diffusion equation for the thermal neutron density n(t) and solve the equation for n(t).

(c) In reality, there are delayed neutrons. Let

β be the delayed neutron fraction,

C(t) be the delayed neutron precursor concentration,

write down the equations for the time rate of change of n(t) and C(t).

Explain clearly the meaning of each term.

(20%)