No. of Printed Pages : 5 **MPHE–026**

M. SC. (PHYSICS)

(MSCPH)

Term-End Examination

June, 2025

MPHE-026 : ELEMENTS OF REACTOR PHYSICS

Time : 2 Hours *Maximum Marks : 50*

Note : *Answer any* ***five*** *questions. Symbols have their usual meanings. The marks for each question are indicated against it. You may use a calculator.*

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1. (a) Calculate the macroscopic fission cross-section of ^{235}U. Given microscopic cross-section fission cross-section ^{235}U is 400 barns. Natural uranium contains only 0.7% of ^{235}U. Use density of natural uranium as 18.95 g cm^{-3}. 5

(b) Differentiate between prompt and delayed neutrons released in fission and discuss the role of delayed neutrons for reactor safety. 5

2. (a) Suppose n is the number of neutrons in one generation, k is the multiplication factor, ^ is the generation time and $n(t)$ is the neutrons at time 't', then derive the relation :

$$n(t) = n(0)\exp\left(\frac{k-1}{\wedge}\right)t$$

Calculate the neutron population in terms of initial population $n(0)$ after 1s if the generation time is 0.1 millisecond and the increment in k per generation is 0.001. 5

(b) Define breeding ratio and doubling time. Which is the best fissile material for ^{238}U based fast breeder reactor ? Discuss your argument. 5

3. (a) Obtain an expression for kinetic energy of a neutron after collision in the lab system in terms of energy decrement per collision (α).What will happen if the neutron collides with hydrogen nucleus ? 4+1

(b) Show that for large mass numbers,

$$\xi = \frac{2}{A + \frac{2}{3}}$$

where ξ is the average logarithmic energy decrement per collision. 5

4. (a) Obtain Fermi age equation for slowing down an infinite weakly absorbing medium. 5

(b) If 10 MeV neutron collides with ^{238}U nucleus, compare the average energy loss of neutrons if they undergo : 5

(i) inelastic scattering; and

(ii) elastic scattering.

5. (a) Write down the assumptions made in arriving at neutron transport equation for a heterogeneous multiplying assembly. 5

(b) What do you understand by constant cross-section approximation ? Derive the steady state transport equation for a plane geometry. The neutron scattering and external sources are taken to be isotropic. 5

6. (a) Starting with the general form of transport equation, obtain the diffusion equation :

$$\mathrm{J}(\vec{r}, \vec{\mathrm{E}}, t) = -\mathrm{D}(\vec{r}, \mathrm{E}', t)\nabla\phi(\vec{r}, \vec{\mathrm{E}}, t)$$

what does $\mathrm{D}(\vec{r}, \mathrm{E}', t)$ represent in this equation ? 5

(b) The scattering cross-section of carbon at 1 eV is 4.8 b. Estimate the diffusion coefficient of carbon. Assume that Σ_a is almost negligible. 5

7. (a) Show that the root mean square crow-flight distance for neutrons from an infinite plane source in an infinite medium is $\sqrt{2}\,L$. 5

(b) Write two-group diffusion equations for a critical homogeneous reactor in steady state. Obtain two-group criticality condition for a bare homogeneous reactor. 5

8. Neutrons are assumed to slow down to thermal energies in one group approximation given by diffusion theory. The energy distribution of these neutrons is Maxwellian. Obtain expression for one-group thermal flux in this approximation. 10

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