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MASTER'S DEGREE PROGRAMME IN PHYSICS [M. SC. (PHYSICS)] (MSCPH)

Term-End Examination

June, 2026

MPHE-026 : ELEMENTS OF REACTOR PHYSICS

Time : 2 Hours *Maximum Marks : 50*

Note : *(i) Answer any **five** questions.*

(ii) Symbols have their usual meanings.

(iii) The marks for each question are indicated against it.

(iv) You may use a calculator.

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1. (a) A monoenergetic beam of neutrons having an intensity of 2×10^{10} n cm^{-2} s^{-1} impinges on a target 1 cm^2 in area and 0.1 cm thick. There are 0.048×10^{24} atoms cm^{-3} in the target and σ_t at the energy of the beam is 5.0 b. Calculate the macroscopic total cross section. 5

   (b) Using Weiszäcker's formula, calculate binding energy per nucleon for ^{235}U. Use $a_v = 15.8$, $a_s = 17.8$, $a_c = 0.7$, $a_A = 23.7$. 5

2. (a) (i) Can neutrons slow down to thermal energies by scattering in the uranium fuel ? Justify your answer. 3

(ii) What is the difference between slowing down density and neutron flux ? 2

(b) India plans to have three-stage nuclear energy programme. Discuss in detail. 5

3. (a) Obtain an expression of scattering angle of a neutron in lab system with particular to hydrogen. 4+1

(b) If E and E' are respectively the energies before and after collision, obtain an expression of average logarithmic energy decrement, ξ in terms of A. Also, plot ξ *versus* A for different nuclides using this formula. 5

4. (a) Show that the scattering probability function $P(E_i \to E_f)$ of a neutron due to elastic scattering can be expressed in terms of lethargy 'u' as :

$P(u_i \to u_f) =$

$$\begin{cases} \dfrac{e^{u_i - u_f}}{1-\alpha} & , u_f - l_n\left(\dfrac{1}{\alpha}\right) < u_i < u_f \\ 0 & \text{otherwise} \end{cases}$$

where E_i is the initial energy and E_f is its energy after elastic scattering. 5

(b) Calculate the width of the energy distribution of inelastically scattered neutrons for heavy nuclides in the LAB system. 5

5. (a) Write the general form of transport equation in terms of flux for a

multiplying system. Express this equation in terms of transport (L) and fission (F) operators. What will be the form of this equation for a non-multiplying homogeneous system ? 5

(b) Using integral form of neutron transport equation, obtain Peierl's equation for steady state. 5

6. (a) Discuss initial and boundary conditions used to solve diffusion equation. Does the flux actually become zero at the extrapolated boundary ? 5

(b) Suppose that components of angular neutron current along the x-axis, y-axis and z-axis in a cartesian coordinate system are $\vec{J}_x = 0{\cdot}25 \times 10^{14}$ neutrons cm^{-2} s^{-1}, $\vec{J}_y = 0{\cdot}433 \times 10^{14}$ neutrons cm^{-2} s^{-1} and $\vec{J}_z = 0.866 \times 10^{14}$ neutrons cm^{-2} s^{-1}. Calculate the neutron density $n(\vec{r}, \vec{\Omega}, \mathrm{E}, t)$ at $r = 0$ if the average velocity of neutrons is 2200 ms^{-1}. 5

7. Write the assumptions related to neutron current density at any point in a medium. Hence, derive Fick's law :

$$\mathrm{J} = -\,\mathrm{D}\ \mathrm{grad}\ \phi$$

where D is the diffusion coefficient. 10

8. (a) Calculate the flux at a distance of 150 cm from a plane source isotropically emitting 10^4 neutrons cm^{-3} s^{-1} in a heavy water slab. 3

(b) Write the two-group critical equation, assuming that all neutrons in a group slow down into the next group of lower energy and fission is induced by thermal neutrons only. Using this equation, obtain Fermi-age critical equation for a bare reactor in terms of non-leakage probability of fast neutrons. 7

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