UNIVERSITY OF PETROLEUM AND ENERGY STUDIES

End Semester Examination, December 2017

Program:	**M.TECH (NST)**	**Semester**	**: III**
Subject (Course):	**NEUTRON TRANSPORT THEORY**	**Max. Marks**	**: 100**
Course Code :	**NSAT 8004**	**Duration**	**: 3 Hrs**
No. of page/s:	**02**		

Section-1 (Each question carries 4 marks each)

1. Define continuous random variable and write expression for average and variance for the same?
2. Sample of N random variables X_1, X_2,…,X_N is given. Write expressions of sample average and sample variance.
3. Why the neutron transport equation is called Linear neutron transport equation?
4. What are the most important physical reactions in the neutron absorption cross section?
5. Define neutron scalar flux, neutron angular flux and neutron current vector?

Section-2 (Each question carries 8 marks each)

6. Briefly describe the construction of a continuous random variable. Give an example by choosing distance to collision " L " of a neutron as a random variable (that is, the distance between consecutive collisions). Total cross section σ is given.
7. List the assumptions (very briefly) made in the derivation of the neutron transport equation.
8. Describe Monte Carlo method for calculating the direction of motion ($\mu = \cos(\theta)$, φ) of neutron produced or scattered with isotropic distribution.
9. List few advantages and disadvantages of Monte Carlo method over the deterministic method in solving the neutron transport equation.
10. Using the K-eigenvalue problem, explain the method of power iteration to calculate the multiplication eigenvalue problem in a neutron multiplying system.

or

Write the time-independent fixed source neutron transport equation under continuous energy model. Explain the approximation to derive the multi-group cross sections and multi-group neutron transport equation.

Section-3 (Each question carries 20 marks each)

11. Write the slab geometry SN- discrete ordinates neutron transport equation. Specify the division of the slab geometry spatial domain into L intervals. Write the discrete neutron transport equation for the ith interval. Write the diamond difference relations in this interval. Explain the marching schemes, when both boundaries on the left and the right are vacuum boundaries.
12. Briefly describe the problem of penetrations of neutrons through a shielding block. Using Monte Carlo Method, compute (1) the probability p+ of a neutron penetrating the block, (2) the probability p- of a neutron reflected from the block and (3) the probability p0 of a neutron being absorbed by the block.

Or

What are alpha (α) time-eigenvalues and K multiplication-eigenvalues for specifying the criticality of a neutron multiplying system under supercritical, critical and subcritical state? For reactor criticality calculation, which eigenvalue is used? Give reasons.

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Section-1 (Each question carries 4 marks each)

1. Define discrete random variable and write expression for average and variance for the same?
2. Define neutron scalar flux, neutron angular flux and neutron current vector?
3. Sample of N random variables X_1, $X_2 \ldots X_N$ is given. Write the expression of sample average and sample variance.
4. What are the units of microscopic and macroscopic cross sections?
5. Plot fission spectrum as a function of neutron energy?

Section-2 (Each question carries 8 marks each)

6. Briefly describe the general scheme of the Monte Carlo Method using the example of computation of the area of a plane figure.
7. Briefly describe the construction of a continuous random variable. Give an example by choosing distance to collision " L " of a neutron as a random variable (that is, the distance between consecutive collisions). Total cross section σ is given.
8. Write the discrete ordinates form of slab geometry transport equation for a fixed source problem in a non-multiplying medium? Write the quadrature formula for approximating the scalar flux? Describe method for choosing weights and directions?
9. Describe Monte Carlo method for calculating the direction of motion ($\mu = \cos(\theta)$, φ) of neutron produced or scattered with isotropic distribution.
10. List few advantages and disadvantages of Monte Carlo method over the deterministic method in solving the neutron transport equation.

or

Write the time-independent fixed source neutron transport equation under continuous energy model. Explain the approximation to derive the multi-group cross sections and multi-group neutron transport equation.

Section-3 (Each question carries 20 marks each)

11. Briefly describe the problem of penetrations of neutrons through a shielding block. Using Monte Carlo Method, compute (1) the probability p^+ of a neutron penetrating the block, (2) the probability p^- of a neutron reflected from the block and (3) the probability p^0 of a neutron being absorbed by the block.
12. What are alpha (α) time-eigenvalues and K multiplication-eigenvalues for specifying the criticality of a neutron multiplying system under supercritical, critical and subcritical state? For reactor criticality calculation, which eigenvalue is used? Give reasons.

    Or

    Write the slab geometry S_N- discrete ordinates neutron transport equation. Specify the division of the slab geometry spatial domain into L intervals. Write the discrete neutron transport equation for the i^{th} interval. Write the diamond difference relations in this interval. Explain the marching schemes, when both boundaries on the left and the right are vacuum boundaries.