ENGINEERING PHYSICS 4D3/6D3

DAY CLASS Dr. Wm. Garland

DURATION: 3 hours

McMASTER UNIVERSITY FINAL EXAMINATION December 6, 1993

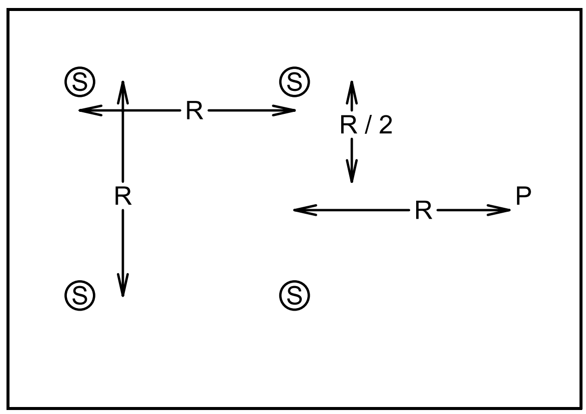
Special Instructions:

1. Open Book. All calculators and reference material permitted.
2. Do all questions.
3. The values of each question is as indicated.
   TOTAL Value: 100 marks

THIS EXAMINATION PAPER INCLUDES 3 PAGES AND 10 QUESTIONS. YOU ARE RESPONSIBLE FOR ENSURING THAT YOUR COPY OF THE PAPER IS COMPLETE. BRING ANY DISCREPANCY TO THE ATTENTION OF YOUR INVIGILATOR.

1. [10 Marks Total]

Consider four point sources of strength S neutrons/s in an absorbing medium, as shown in the figure. Calculate the flux at point P.



2. [10 Marks Total]

A U^{235}-fuelled reactor is operating at 2 MW (thermal). It is scrammed by the sudden insertion of the safety rods which have a total worth of -85 mk.

(a) Calculate the immediate change in power.

(b) Calculate the power level 10 minutes after shutdown. Assume one delayed group with a half-life of 55 seconds.

3. [10 Marks Total)

Consider a beam of neutrons incident upon a thin target with an intensity of 10^{12} neutrons/cm^2-s. If the total cross section for the nuclei in this target is 4 b, how long would one have to wait, on the average, for a given nucleus in the target to suffer a neutron interaction.

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4. [10 Marks Total]

Consider an infinite planar neutron source, emitting S neutrons/cm^2-s, surrounded by a homogeneous infinite mixture of absorbing material and fissile material. The mixture is sub-critical. In essence, this is an infinite subcritical pile with a planar source. Assume one group diffusion applies.

(a)Derive the steady state flux distribution as a function of space.

(b) What happens as the mixture approaches criticality?

5. [10 marks total]

What fraction of the initial fissile fuel of a highly enriched (experimental)) reactor remains following a fluence of 10^{20} neutrons / cm^2. Use $\sigma_a = 600$ b.

6. [10 Marks Total]

The general multigroup neutron diffusion equations with delayed precursors are given by:

$$\frac{1}{v_g}\frac{\partial \varphi_g}{\partial t} = \nabla \cdot D_g \nabla \varphi_g - \Sigma_{a_g}\varphi_g - \Sigma_{s_g}\varphi_g + \sum_{g'=1}^{G} \Sigma_{s_{g'g}}\varphi_{g'} + \chi_g \left[\sum_{g'=1}^{G} \nu_{g'}(1-\beta_{g'})\Sigma_{f_{g'}}\varphi_{g'} + \sum_{i=1}^{N} \lambda_i C_i \right] + S_g^{ext}$$

$$\frac{\partial C_i}{\partial t} = -\lambda_i C_i + \sum_{g=1}^{G} \beta_{i_g}\nu_g \Sigma_{f_g}\varphi_g$$

(a) Define each variable. Explain the significance of each term.

(b) In these equations, what does "directly coupled" imply?

(c) What does " no upscattering imply"?

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IMPORTANT NOTE: For the remaining questions, consider a simple reactor concept such as a pool-type reactor with an array of fuel elements cooled by water. Use the <u>general multigroup neutron diffusion equations</u> with delayed precursors (as given in question 6) plus any other pertinent equations to illustrate your answers.

7. [10 marks total]

Generate the steady state version of the flux equation of question 6. Discuss the short term (i.e. no change in poisons or fuel burnup) criticality issue from the point of view of this equation. What role do the delayed precursors play? Specifically, how can the reactor be made critical? Illustrate the physical processes involved. What parameters are involved? If this was a homogeneous bare core, what would the one and two group criticality conditions be?

8. [10 marks total] Ignore poison and burnup effects in this question.

How does a control rod work, i.e. what parameters in the governing equations of question 6 are affected? Where should the rod or rods be placed? Where should the flux be measured? How fast does the control rod movement need to be? Would more than one rod be necessary or desirable? Consider possible undesirable effects of control rods on the flux distributions. How could you achieve flux <u>shape</u> control as well as flux amplitude control? Consider the following algorithm to control the rod(s) position:

$$\text{desired rod position} = a_0 \left(\varphi_{measured} - \varphi_{set\ point} \right) + a_1 \int (\varphi_{measured} - \varphi_{set\ point}) \, dt,$$

where a_0 and a_1 are constants. Would it work? Would it be better to use thermal power instead of flux? Why or why not?

9. [10 marks total]

How do fission product poisons affect the neutron balance equations and criticality? How can you compensate for this effect? Show the additional equations necessary to account for the effect of poisons Estimate the $\Delta\rho$ due to poisons.

10. [10 marks total]

Discuss fuel depletion with respect to the neutron balance equations and criticality. How can you compensate for this effect? Show the additional equations necessary to account for burnup. Give a possible refuelling strategy designed to get to most power out of a given fuel inventory.

THE END