ENGINEERING PHYSICS 4D03/6D03

DAY CLASS Dr. Wm. Garland
DURATION: 3 hours

McMASTER UNIVERSITY FINAL EXAMINATION December 16, 2005

Special Instructions:

1. Closed Book. All calculators and up to 6 single sided 8 1/2" by 11" crib sheets are permitted.
2. Do all questions.
3. The value of each question is as indicated. TOTAL Value: 100 marks

THIS EXAMINATION PAPER INCLUDES 2 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] Consider the equation for neutron beam attenuation, $I(x) = I_o e^{-\Sigma x}$.
   a. What is the simple differential equation it was derived from?
   b. Comment on its validity considering the following aspects:
      i. Multiple scatters
      ii. Space dependency of material properties
      iii. Cross section energy dependence
      iv. Neutron energy and moderation of neutrons
   c. What would be the differential equation that addresses the above concerns? Why?

2. [10 Marks] A free neutron beta decays with a half-life of 11.7 minutes. Determine the relative probability that a thermal neutron will undergo beta decay before being absorbed in an infinite medium. Calculate this probability using water, $\Sigma_a = 0.022\ cm^{-1}$, as the medium. HINT: Write a simple neutron balance equation for one-speed neutrons with an additional 'sink' of neutron decay. Ignore the diffusion term. You don't need to solve this equation but a comparison of the various terms is suggested. Assume thermal neutrons.

3. [10 marks] For a nuclide, say nuclear fuel:
   a. Construct a general and comprehensive depletion / buildup rate equation, accounting for self decay, parental decay, neutron capture and transmutation. Explain each effect briefly.
   b. If only neutron capture was important, solve the equation to determine the nuclide concentration as a function of time.

4. [10 marks] For a planar source of neutrons, S neutrons / cm^2 sec, in an infinite absorbing medium:
   a. Derive the neutron flux distribution (assume one speed).
   b. Why is the distribution in space different from the simple attenuation of a beam in an absorbing media?

5. [10 marks] Consider a homogeneous slab of fuel / moderator / coolant with known properties, extrapolated thickness 'a' and infinite in the other 2 dimensions. Assume the slab is surrounded by a vacuum on both sides. Assume steady state.
   a. Derive the neutron flux distribution (assume one speed).
   b. What is the criticality condition? What is the physical meaning of this?

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6. [10 marks] Consider a homogeneous slab of fuel / moderator / coolant with known properties, extrapolated thickness 'a' and infinite in the other 2 dimensions. Assume the slab is surrounded by a vacuum on both sides. Assume steady state.
   a. Derive the neutron flux distributions (assume two neutron speeds – fast and thermal). Make no simplifying assumptions about up-scatter or fissioning other than you can assume that the all prompt and delayed neutrons are born in the fast energy group.
   b. What is the criticality condition?
   c. Discuss which variables (and hence, terms) are likely to be small enough to ignore, given typical errors in experimentally determined parameters.

7. [10 marks] Consider a homogeneous slab of fuel / moderator / coolant with known properties, extrapolated thickness 'a' and infinite in the other 2 dimensions. Assume the slab is surrounded by a vacuum on both sides. Assume steady state.
   a. Derive the finite difference equations for the two-group approximation. Make no simplifying assumptions about up-scatter or fissioning other than you can assume that the all prompt and delayed neutrons are born in the fast energy group.
   b. Outline the numerical solution algorithm showing how the fluxes can be calculated and how criticality is achieved.
   c. Discuss why the numerically obtained criticality condition is different from the analytically obtained criticality conditions for the one and two group approximations. Suggest ways to improve the accuracy of the criticality estimate.

8. [10 marks] Consider a processing facility for the production of isotope B, consisting of a small container inside a reactor. The container is small enough and the absorption cross section is low enough that the neutron flux is uniform in the container. The isotope B is produced by the following reaction: A(n,γ)B. B decays to C with a half-life $T_{1/2}$. A and C do not decay. A, B and C all have finite neutron absorption cross sections and all can be added and removed from the container at will.
   a. State the governing balance equations for A, B and C.
   b. Propose a numerical solution algorithm, showing the finite difference equations and a simple flow diagram to illustrate how the algorithm works.

9. [10 marks] For a homogeneous, critical, one dimensional, bare slab reactor (modelled by the two-group neutron diffusion model):
   a. Derive the steady state xenon spatial distribution.
   b. Sketch this distribution in space for several cases (low flux, medium flux, high flux).

10. [10 marks] The McMaster Nuclear Reactor contains fuel plates that contain a homogeneous mixture of aluminum and uranium (called the fuel meat). For a postulated startup accident in which the control rods are inadvertently extracted causing the power to spike, 2.2×10^6 joules are deposited rapidly into the fuelled region. Given the data below and assuming that no heat is transferred out of the fuelled region, what is the average temperature rise in the fuel meat?

Material	**Mass (kg)**	**Cp (J / kg K)**
Aluminum	24.76	903.5
Uranium	7.539	201.6

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