ENGINEERING PHYSICS 4D03/6D03

DAY CLASS Dr. Wm. Garland
DURATION: 50 minutes
McMASTER UNIVERSITY FINAL EXAMINATION December 6, 2003

Special Instructions:
1. Closed Book. All calculators and up to 6 single sided 8 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] A stream of free neutrons are travelling through an absorbing media of $\Sigma_a = 0.022 cm^{-1}$. The neutrons can be absorbed or they can beta decay with a half-life of 11.7 minutes. They are travelling at a thermal velocity.
   a. Assuming for the moment that it could only decay, what is the governing neutron balance equation?
   b. Assuming now that they could only be absorbed, what is the governing balance equation?
   c. Assuming that both processes can occur, what is the combined governing equation?
   d. By examining the relative size of the terms for the two processes, determine the relative probability that a thermal neutron will undergo beta decay compared to being absorbed.

2. [5 marks] Construct a general and comprehensive depletion / buildup rate equation for a nuclide, accounting for self decay, parental decay, neutron capture and transmutation. Explain each effect briefly.

3. [10 marks] What must the reactivity insertion be for a reactor undergoing a power excursion with a measured period, T, of 1 second (ω= 1/T)? To simplify the calculation, assume the presence of only one delayed precursor group with half life of 20 seconds. Assume a neutron lifetime, ℓ, of 5 x 10^{-5} seconds and the delayed fraction, β, is 0.007.

4. [15 marks] Consider an infinite sub-critical reactor with a <u>uniformly</u> distributed source S neutrons/cm^3-s. Assume one-speed neutrons.
   a. In terms of the cross sections and other parameters of the standard neutron balance equation, determine how many neutrons are eventually absorbed per source neutron produced. Since every neutron produced is eventually absorbed, this ratio is a ratio of the total production rate to the original source emission rate. Call this ratio M.
   b. In terms of k, show from the basic definition of k that the multiplication ratio, M = 1/(1-k). You might use a particular numerical value of k to illustrate and generalize from there.
   c. Relate k to the cross sections to show that both evaluations of M are equal.
   d. How can this ratio M be used in starting up a reactor (called the approach to criticality)?
   e. Why is it so important to approach criticality very carefully?

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5. [10 marks] 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?

6. [10 marks] In regards to the numerical solution of the steady state flux equations:
   a. How does the Gauss-Seidel method differ from the Jacobi-Richardson method? Illustrate using a simple example.
   b. What is SOR (Successive over-relaxation). Illustrate.
   c. How is the $\nabla \bullet D \nabla \phi$ approximated for the case of non-uniform properties and non-uniform grid spacing? Illustrate using the 1 dimensional case.
   d. How is criticality achieved numerically?

7. [10marks] Consider a rectangular tank that is 100 cm square at the base and contains a homogeneous mixture of fuel and moderator. Known parameters are $\Sigma_a = 0.500$ cm^{-1}, D = 10.0 cm. The tank was slowly filled with the mixture until criticality was achieved at a height of precisely 100 cm. Then, a small amount of absorber material was added uniformly to the mixture, causing the reactor to go subcritical. More fuel / moderator mixture was added to bring the reactor back to criticality at a height of 110.0 cm. Assume that the absorber does not displace any mixture material. Use one-group diffusion theory to find:
   a. the inferred value of $\nu\Sigma_f$,
   b. the effective macroscopic cross section of the absorber.

8. [10 marks] For the one-group transient neutron diffusion model of a one dimensional homogeneous bare slab reactor:
   a. State the neutron balance equation and the appropriate initial and boundary conditions.
   b. Derive the stability criteria for the explicit numerical scheme.
   c. Show how this condition is relaxed when an implicit scheme is used.

9. [10 marks] For an infinite slab reactor of extrapolated thickness, "a", derive the criticality condition for the two-group approximation (fast and thermal neutrons, no upscatter, no fast fissions, no neutrons born in the thermal region). Assume the slab is surrounded by a vacuum on both sides and that the slab is a homogeneous mixture of fuel and moderator.

10. [10 marks total] A typical fuel channel in a CANDU reactor puts out a nominal power of 6.5 MW. The channel is 6 m. long and consists of 12 bundles, each weighting roughly 20 kg. The fuel heat capacity is 221 J/(kg K). Estimate how long it would take to start melting the fuel once cooling was completely lost to the fuel. Assume that the melting point of UO_2 is 1000 °C above the nominal operating temperature of the fuel.

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