ENGINEERING PHYSICS 4D3/6D3

DAY CLASS — Dr. Wm. Garland
DURATION: 3 hours — Page 1 of 3
McMASTER UNIVERSITY FINAL EXAMINATION — December 1998

Special Instructions:

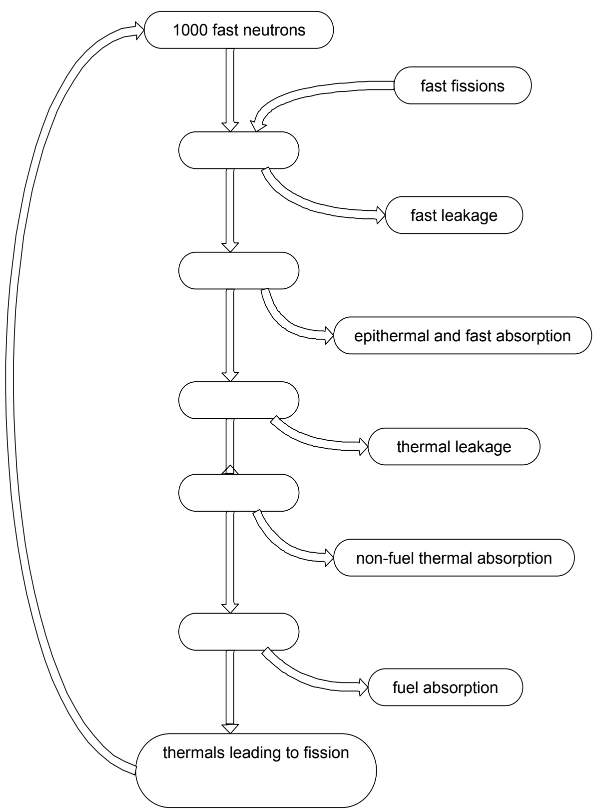
1. Closed Book. All calculators and up to 8 single sided 8 ½" by 11" crib sheets are permitted.
2. Do all 7 questions.
3. The value of each question is as indicated.
4. Point form is sufficient for discussion type questions.

TOTAL Value: 100 marks

THIS EXAMINATION PAPER INCLUDES 3 PAGES AND 7 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) Distinguish between neutron flux and neutron current.
   (b) Distinguish between critical and prompt critical.
   (c) Distinguish between $\Sigma_s(E_1$ 6 $E_2)$ and $\Sigma_s(E_2$ 6 $E_1)$.
   (d) Distinguish between η and f of the four factor formulae.
   (e) Distinguish between reactivity, ρ, and multiplication factor, k.

2. [10 marks]
   Using the figure as a guide, deduce how many fast neutrons escape from a critical reactor from the following facts. When this reactor is critical, per unit time:
   1. Every thermal fission produces on average 2.4331 fast neutrons
   2. 25 fast neutrons are produced by fast fissions
   3. The ratio of the fission and absorption macroscopic cross sections for the fuel is 0.4835
   4. 50 thermal neutrons are absorbed elsewhere than in the fuel
   5. 20 thermal neutrons escape from the reactor
   6. 100 non-thermal neutrons are absorbed in the reactor.



3. [15 marks]
For a planar source of neutrons, S neutrons / cm^2 sec, in an infinite absorbing medium, we know that the flux distribution is given by: $\varphi = \frac{SL}{2D}\exp(-\frac{|x|}{L})$, where L is the diffusion length, D is the diffusion coefficient, and x is the distance from the planar source.
   a) Integrate over space to find the total absorption rate of neutrons in the right hand half of the absorbing medium.
   b) Compare (a) to the current at the source plane.

4. [15 marks]
Consider a rectangular tank that is 100 cm square at the base and contains a homogeneous mixture of fuel and moderator. Known parameters are Σ_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.

5. [15 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.

6. [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, 2.2x10^6 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)	C_p (J / kg °K)
Aluminum	24.76	903.5
Uranium	7.539	201.6

7. [25 marks]

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

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

$$\frac{\partial C_i}{\partial t} = -\lambda_i C_i + \beta_i\sum_{g'=1}^{G}\nu_{g'}\Sigma_{f_{g'}}\varphi_{g'}$$

Note that φ_g and C_i are functions of $\underline{r}$ and t but the notation has been dropped for clarity. The poison equations are:

$$\frac{\partial I(\underline{r},t)}{\partial t} = \gamma_I\sum_{g'=1}^{G}\Sigma_{f_{g'}}\varphi_{g'}(\underline{r},t) - \lambda_I\, I(\underline{r},t)$$

$$\frac{\partial X(\underline{r},t)}{\partial t} = \gamma_X\sum_{g'=1}^{G}\Sigma_{f_{g'}}\varphi_{g'}(\underline{r},t) + \lambda_I\, I(\underline{r},t) - \lambda_X\, X(\underline{r},t) - \sum_{g'=1}^{G}\sigma^X_{a_{g'}}\varphi_{g'}(\underline{r},t)\; X(r,t)$$

and the fuel depletion equation is:

$$\frac{\partial N_f}{\partial t} = -N_f(\underline{r},t)\sum_{g'=1}^{G}\sigma^f_{a_{g'}}\varphi_{g'}(\underline{r},t)$$

a) Assuming two neutron groups (fast and thermal), no upscatter, no fast fissions, and no neutrons born in the thermal region, what are the transient flux and precursor equations?

b) Ignoring fuel depletion for the moment, what are the steady state xenon and iodine concentrations at a given flux (which is steady in time but may vary in space)? Which terms in the flux and precursor equations of (a) are dependent on these poison concentrations?

c) For the two-group approximation, what is the steady state precursor concentration, C_i, given the flux (which is steady in time but may vary in space)?

d) What do the two-group steady state flux equations look like if the steady state value of C_i is substituted in?

e) To numerically solve the transient fluxes, precursors, poisons, etc, a controller is introduced to keep the flux at some prescribed set point (which may be steady or may vary in time). This controller alters the absorption terms in the flux equations. Yet in the steady state algorithms, we used a fudge factor, k, in the fission terms. Explain the rational behind the two different schemes.

THE END