February 1998

Ph.D. Qualifying Examination
Reactor Theory and Experimentation

1. (15 min) Calculate an expression for the extrapolation distance in plane geometry if the flux is extrapolated linearly beyond a vacuum boundary. Use one-group diffusion theory. [Hint: set the incident partial current equal to zero.]

2. (20 min) For the NSC TRIGA reactor you are required to calibrate a certain control rod by the rod drop technique.

   a) Describe the experimental procedures.

   b) What would be the approximate mean neutron generation time a few minutes after the rod was dropped from its 100% out position if the total rod worth was found to be $4.83 and the total effective delayed neutron fraction is 0.0065? [Hint: For large negative reactivity insertion, $T \approx l/\Delta k$.]

3. (15 min) A nuclear engineer has completed a neutronic analysis of a fuel assembly. The fuel assembly is a 17×17 array of pin cells, with each pin cell approximated as being homogeneous with known fine-group cross sections. Each pin cell's material properties are potentially different from those of any other pin cell.

   The neutronic analysis used a calculation with 70 energy groups in the fine-group set, so now you have available to you a set of 70-group pin-cell-averaged scalar fluxes and a set of 70-group pin-cell cross sections for each pin cell.

   a) If the top 20 groups are "fast" and the bottom fifty are "thermal," what is an expression for the assembly-averaged thermal fission cross section?

   b) Again with the 20/50 fast/thermal split, what is an expression for the assembly-averaged fast-to-thermal scattering cross section in terms of what you have available?

4. (10 min) Write the integral form of the transport equation for mono-energetic neutrons in plane-parallel ("slab") geometry. Assume isotropic scattering, azimuthal symmetry, no extraneous source, no fission, and given incident angular fluxes on the boundaries.

5. (25 min.) Consider a CRITICAL, thermal, homogeneous parallelpiped reactor that contains a point (extraneous) source of thermal neutrons at its center. The point source's strength is $1x10^7$ neutrons/sec. All neutrons in the reactor can be treated as "thermal". The macroscopic fission and absorption cross sections in the reactor are .0500000 cm-1 and .118753019 cm-1, respectively. There are 2.43 neutrons produced per fission.

In this reactor, starting from an initial flux of zero everywhere in the reactor, and neglecting delayed neutrons, the time dependent thermal neutron flux can be shown to be given as follows:

$$\phi\,(x,y,z,t) = \sum_{m=1,3,5..}^{\infty}\sum_{n=1,3,5..}^{\infty}\sum_{o=1,3,5..}^{\infty} \frac{S_{mno}}{\Sigma_a}\,\frac{1}{1+L^2B^2_{mno}}\left(\frac{e^{\left(\frac{k_{mno}-1}{t_{mno}}\right)}-1}{k_{mno}-1}\right)\cos\frac{m\pi x}{a}\cos\frac{n\pi y}{b}\cos\frac{o\pi z}{c}$$

where the origin of the coordinate system is at the reactor's center,

a, b and c are the lengths of the sides of the reactor parallel to the x, y and z axes, respectively, and a=b=c= 80 cm,

L^2 is the diffusion area, and $L^2 = 5\ cm^2$

υ is the thermal neutron speed and $v = 220000$ cm/sec,

B^2mno is the buckling of the parallelpiped reactor (extrapolation distances may be neglected), and

$$k_{mno} = \frac{k_\infty}{1+L^2B^2{}_{mno}}$$

where

$$t_{mno} = \frac{\dfrac{1}{\upsilon\Sigma_a}}{1+L^2B^2{}_{mno}}$$

(a) (40%) Find the value of the constant denoted as S_{111}. Assuming the extraneous source is represented as

$$S(x,y,z) = 1x10^7\delta(x)\delta(y)\delta(z)$$

where δ denotes the Dirac delta function

(b) (40%) Determine the asymtotic (at large time) rate of change of the thermal neutron flux in this CRITICAL parallelpiped reactor.

(c) (20%) If there are 200 MeV released per fission, calculate the asymtotic rate of change of the reactor power.

6. (15 min) A source of monoenergetic fast neutrons $(\tau(E_o)=0)$ is distributed throughout a semi-infinite slab of moderator of thickness, a, such that $q(x,\tau)=S\cos\frac{\pi x}{a}$. Ignore extrapolation distance. S is a constant.

Using age theory, $\frac{\partial^2 q}{\partial x^2}(x,\tau)=\frac{\partial q(x,\tau)}{\partial \tau}$ and the boundary conditions $q(x=\pm a/2,\tau)=0$, it can be shown that the solution for $q(x,\tau)$ is of the form $q(x,\tau)=\sum_{n=1,3,5,7...} A_n e^{-B_n^2\tau}\cos Bnx$ where $B_n^2=\left(\frac{n\pi}{a}\right)^2$ and A_n's are constants.

a) (50%) Find the expression for the slowing down density of the neutrons throughout the slab at the age, τ_{th}, i.e., solve for the A_n's in the expression for $q(x,\tau_{th})$.

b) (50%) Derive the relationship for the probability that a source neutron does <u>not</u> leak from the slab while slowing down to the age τ_{th}.

7. (10 min) The six-group point kinetics equations are given below as:

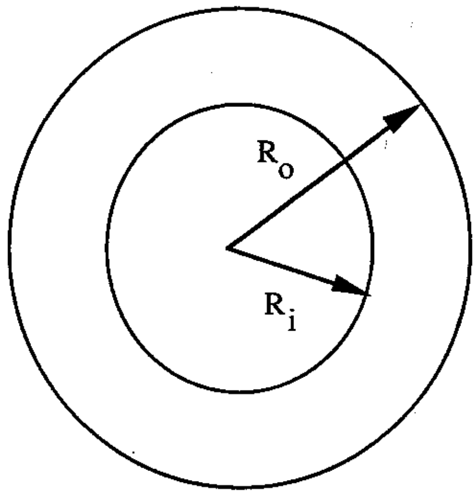
$$\frac{dN(t)}{dt}=\frac{\rho-\beta_{eff}}{\lambda}N(t)+\sum_{i=1}^{6}\lambda_i C_i(t)$$

$$\frac{dC_i(t)}{dt}=\frac{\beta_{i,eff}}{\lambda}N(t)-\lambda_i C_i(t);\ i=1....6.$$

(a) (50%) State the limitations of the point-kinetics equations.

(b) (50%) Assume that a U^{235}-fueled thermal reactor is critical and operating at steady-state at 100% of full power. Suddenly, a control rod having a worth of \$20 (β_{eff} = 0.007) is inserted into the core in a step manner. Derive an appropriate expression for the neutron power profile as a function of time following the sudden rod insertion. You can normalize your expression using initial operating power of 1,000 MW_{th}.

8. (10 min.) A bare spherical shell of fissile material has an outer radius R_o and an inner radius R_i. With a vacuum in the central cavity this configuration is just critical.



A low Z material containing some neutron absorber is inserted in the center of the shell. According to one-speed theory, is the new configuration subcritical, critical, or supercritical? **Explain** you answer.