

1. ER362 Reactor Physics

2. Credit Hours (3)/Contact Hours (4)

3. Course Director – Stephen R. McHale

4. Introduction to Nuclear Engineering, 4th Ed., by J.L. Lamarsh, A.J. Baratta; Pearson, 2018 – ISBN 978-0-13-457005-1

5. Specific course information

- a. Examines the principles underlying power generation in both a steady state and transient fission reactor. Subject areas include neutron diffusion and slowing down theory, criticality, bare, reflected and heterogeneous reactors, reactor kinetics, and reactor control. Experiments and classroom exercises include the determination of important reactor parameters, the use of computational methods, and the use of plant simulation software.
- b. Prerequisites: ER301
- c. Required for Nuclear Engineering Program, elective course for all other majors

6. Educational objectives for the course

- a. Solve problems involving nuclear reactions and neutron interactions with matter
- b. Solve problems involving the fission process, to include calculation of parameters for the infinite multiplication factor, the effective multiplication factor, and fuel consumption rates
- c. Solve problems involving diffusion theory applied to several geometries and neutron fluxes
- d. Solve problems involving calculation of criticality, critical mass of fuel, and critical size of homogeneous nuclear reactors
- e. Solve problems involving reactor dynamics, including reactivity insertion
- f. Solve problems involving reactor control, temperature coefficients, and effects of poison buildup.

7. Specific program outcomes addressed by this course

	(1)	(2)	(3)	(4)	(5)	(6)	(7)
Reinforced		X	X		X	X	X
Mastered	X						

8. Brief list of topics to be covered

- a. Lecture topics include
 - i. Mono and polyenergetic neutron interactions, transport, and slowing down
 - ii. Fission energetics and products
 - iii. Neutron diffusion, continuity, and sources
 - iv. Diffusion theory, leakage, and transport in finite media

- v. One-group reactor equation and buckling
- vi. One-group criticality, power distribution, density, flux, and four-factor formula
- vii. Modified one-group equation and calculation of critical mass given either materials or dimensions
- viii. Two-group theory, non-leakage, and six-factor formula
- ix. Reactor kinetics with and without delayed neutrons
- x. Prompt criticality and reactivity insertion
- xi. Reactor control, control rod worth, temperature effects
- xii. Xenon and samarium poisoning
- xiii. Fuel burnup and core life

b. Lab exercises include

- i. Introduction to MCNP
- ii. Neutron scattering reaction rate
- iii. Point source in an infinite diffusing medium
- iv. Diffusion in a finite sphere
- v. MCNP diffusion in layered media
- vi. MCNP comparison with multi-group diffusion
- vii. Criticality calculations with MCNP
- viii. MCNP cross section, geometry, and mesh tally plotting
- ix. Lattice geometry with MCNP
- x. MCNP comparison with one-group criticality
- xi. Heterogeneous reactor modeling with MCNP
- xii. Reflector savings
- xiii. Control rod worth analysis with MCNP
- xiv. Burnup calculations with MCNP