University of New Brunswick

Department of Chemical Engineering

CHE 3804 NUCLEAR ENGINEERING

QUESTION BANK

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Prepared by

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Reference Text

INTRODUCTION TO NUCLEAR ENGINEERING

by

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CHE 3804 NUCLEAR ENGINEERING

QUESTION BANK

CONTENTS

REFERENCE EQUATIONS

Reaction Rate:

$$R = \phi ßN$$

Decay Activity:

$$\propto = \lambda N$$

Build up Activity:

$$\propto = R(1 - e^{-\lambda t})$$

Decay Equations:

$$dN / dt = - \lambda N$$

$$N_t = N_o e^{-\lambda t}$$

Build up - Decay Equations:

$$dN_x / dt = \phi \sigma_y N_y - \lambda_x N_x$$

$$N_x = N_{eqx} [1 - e^{-\lambda x t}]$$

$$N_{eqx} = \phi\ \sigma_y N_y / \lambda_x$$

Burn up Equations:

$$dN / dt = - \phi\ \sigma N$$

$$N_t = N_o e^{-\sigma \phi t}$$

Build up - Burn Up Equations:

$$dN_x / dt = \phi\ \sigma_y N_y - \phi\ \sigma_x N_x$$

$$N_t = N_{eqx} [1 - e^{-\sigma_x \phi t}]$$

$$N_{eqx} = \sigma_y N_y / \sigma_x$$

REFERENCE EQUATIONS

NEUTRON DIFFUSION

$$J_x = - D\,(d\phi / dx)$$

$$D \approx \lambda_{tr} / 3$$

$$\Sigma_{tr} = \Sigma_s(1 - \mu)$$

$$\mu = 2 / 3A$$

$$L^2 = D / \Sigma_a$$

$$L^2 = r^2 / 6$$

$$\lambda_{tr} = 3D$$

$$\lambda_{tr} = 1 / \Sigma_{tr}$$

REACTOR THEORY

$$\nabla^2\phi + B^2\phi = 0$$

$$B^2 = (k_\infty - 1) / L^2$$

$$B^2 = (k_\infty - 1) / M_T^2$$

$$M^2_T = L^2_T + \tau_T$$

$$\tau_T = D_1 / \Sigma_1$$

REFERENCE EQUATIONS

REACTOR KINETICS

Reactor Kinetics Equation:

$$P_t = P_O \left[\frac{\beta}{\beta - \Delta k} e^{\frac{\lambda \Delta k}{\beta - \Delta k} t} - \frac{\Delta k}{\beta - \Delta k} e^{-\frac{\beta - \Delta k}{\ell} t} \right]$$

Prompt Drop Approximation:

$$P_t = P_O \frac{\beta}{\beta - \Delta k} e^{\frac{\lambda \Delta k}{\beta - \Delta k} t}$$

Prompt Fission Neutron Equation:

$$P_t = P_o e^{\frac{\Delta k}{\ell} t}$$

Reactor Period Equations:

$$T = \frac{\ell}{\Delta k} \quad \text{(prompt neutrons only)}$$

$$T = \frac{\beta - \Delta k}{\lambda \Delta k} \quad \text{(with delayed neutrons)}$$

TABLE 6.2
Bucklings and fluxes for critical bare reactors

Geometry A	Dimensions ϕ_{mas}/ϕ_{av}	Buckling	Flux		
Infinite slab	Thickness a	$(\pi/a)^2$	$A \cos(\pi x/a)$	$1.57\ P/aE_R\Sigma_f$	1.57
Rectangular parallelepiped	a x b x c	$(\pi/a)^2 + (\pi/b)^2 + (\pi/c)^2$	$A \cos(\pi x/a) \cos(\pi y/b) \cos(\pi z/c)$	$3.87P/VE_R\Sigma_f$	3.88
Infinite cylinder	Radius R	$(2.405/R)^2$	$AJ_o\,(2.405r/R)$	$0.738P/R^2E_R\Sigma_f$	2.32
Finite cylinder	Radius R Height H	$(2.405/R)^2 + (\pi/H)^2$	$AJ_o\,(2.405r/R) \cos(\pi z/H)$	$3.63P/VE_R\Sigma_f$	3.64
Sphere	Radius R	$(\pi/R)^2$	$A\ 1/r \sin(\pi r/R)$	$P/4R^2E_R\Sigma_f$	3.29

Cross Section of Sphere	$A = (\pi/4)\ D^2$
Volume of Sphere	$V = (\pi/6)\ D^3$
Surface of Sphere	$S = \pi D^2$

CONSTANTS

Use the following constant values where applicable

Constant Values:
Density of Water: $\rho = 1000\ kg/m^3$
Gravitational Acceleration: $g = 9.81\ m/s^2$
Avogadros Number $N_A = 0.6022 \times 10^{24}$
Boltzmann Constant $k = 8.6170 \times 10^{-5}$ eV/°K
Boltzmann Constant $k = 1.3806 \times 10^{-23}$J/°K

Energy release in fission = 200 MeV
Conversion Factors
1 barn = $10^{-24}\ cm^2$
1 curie = 3.7×10^{10} disintegrations/s
1 amu = 931 MeV
1 MeV = 1.6022×10^{-13} J

Reactor Kinetics:
Delayed neutron fraction: $\beta = 0.0065$
Average neutron lifetime: $\ell = 0.001$ s
Delayed neutron time constant: $\lambda = 0.1\ s^{-1}$

If not given asume that the atomic (or molecular) mass (in amu) is equal to the atomic mass number A.

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QUESTION BANK SECTION 1

CHAPTER 2

FUNDAMENTAL CONCEPTS

2.0 ATOMIC STRUCTURE AND DIMENSIONS

PROBLEM 2.01 (Text Book Problem 2.1)

How many neutrons and protons are there in the nuclei of the following atoms: (a) ^{7}Li, (b) ^{24}Mg, (c) ^{135}Xe, (d) ^{209}Bi, (e) ^{222}Rn?

PROBLEM 2.02 (Text Book Problem 2.3)

How many atoms are there in 10 g of ^{12}C?

PROBLEM 2.03 (Text Book Problem 2.9)

Compute the mass of a proton in amu.

PROBLEM 2.04 (Text Book Problem 2.10)

Calculate the mass of a neutral atom of ^{235}U (a) in amu; (b) in grams.

PROBLEM 2.05 (Text Book Problem 2.12)

Using Eq. (2.3), estimate the radius of the nucleus of ^{238}U. Roughly what volume fraction of the ^{238}U atom is taken up by the nucleus?

PROBLEM 2.06 (Text Book Problem 2.13)

Using Eq. (2.3), estimate the density of nuclear matter in g/cm^3. Take the mass of each nucleon to be approximately 1.6×10^{-24} g.

PROBLEM 2.07 (Text Book Problem 2.14)

The planet earth has a mass of approximately 6×10^{27} g. If the density of the earth were equal to that of nuclei, how big would the earth be?

PROBLEM 2.08

Determine the fraction of volume taken up by the nucleus in the following atoms

(i) H
(ii) He
(iii) Na
(iv) Al
(v) Pb

Compare the values with the physical densities of these elements.

2.1 MASS AND ENERGY

PROBLEM 2.11 (Text Book Problem 2.15)

The complete combustion of 1 kg of bituminous coal releases about 3×10^7 J in heat energy. The conversion of 1 g of mass into energy is equivalent to the burning of how much coal?

PROBLEM 2.12 (Text Book Problem 2.16)

The fission of the nucleus of ^{235}U releases approximately 200 MeV. How much energy (in kilowatt-hours and megawatt-days) is released when 1 g of ^{235}U undergoes fission?

PROBLEM 2.13 (Text Book Problem 2.17)

Compute the neutron-proton mass difference in MeV.

2.2 ATOMIC AND MOLECULAR MASSES

PROBLEM 2.21 (Text Book Problem 2.56)

Calculate the atom density of graphite having density of 1.60 g/cm^3.

PROBLEM 2.22 (Text Book Problem 2.58)

What is the atom density of ^{235}U in uranium enriched to 2.5% in this isotope if the physical density of the uranium is 19.0 g/cm^3?

PROBLEM 2.23 (Text Book Problem 2.60)

It has been proposed that uranium carbide (UC) be used for the initial fuel in certain types of breeder reactors, with the uranium enriched to 25 w/o. The density of UC is 13.6 g/cm^3. (a) What is the atomic weight of the uranium? (b) What is the atom density of the ^{235}U?

PROBLEM 2.24 (Text Book Problem 2.61)

Compute the atom densities of ^{235}U and ^{238}U in UO_2 of physical density 10.8 g/cm^3 if the uranium is enriched to 3.5 w/o in ^{235}U.

PROBLEM 2.25 (Text Book Problem 2.6)

Natural uranium is composed of three isotopes, ^{234}U, ^{235}U, ^{238}U. Their abundances and atomic weights are given in the table below. Compute the atomic weight of natural uranium.

Isotope	Abundance, %	Atomic weight
^{234}U	0.0057	234.0409
^{235}U	0.72	235.0439
^{238}U	99.27	238.0508

2.3 NUCLEAR STRUCTURE

PROBLEM 2.31

All stable isotopes lie in a band where the ratio of neutrons to protons ranges from 1.0 to about 1.5. Should an isotope have (i) too many and (ii) too few neutrons for stability explain what reactions are likely to occur. Write typical equations for these reactions.

PROBLEM 2.32 **(First Test 1992) (Final Exam 1993) (Final Exam 1994) (First Test 1995)**

On axes of neutrons versus protons sketch the distribution of naturally occurring isotopes. Illustrate on this diagram and define in words the following processes:

(a) alpha particle emission
(b) beta particle emission
(c) positron (positive beta) particle emission.
(d) fission of nucleus
(e) decay of fission products

For each case state where in the diagram the process is likely to occur and what the result of the process will be.

Description (10 marks)

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QUESTION BANK SECTION 2

CHAPTER 2

ATOMIC AND NUCLEAR PHYSICS

2.4 RADIO-ACTIVITY

PROBLEM 2.41 (Text Book Problem 2.28)

Using the chart of the nuclides, complete the following reactions. If a daughter nucleus is also radioactive, indicate the complete decay chain.

a) ^{18}N ————————.   b) ^{83}Y ————————.
c) ^{135}Sb ————————.   d) ^{219}Rn ————————.

PROBLEM 2.42 (First Test 1995)

Tritium (^3H) decays by negative beta decay with a half-life of 12.26 years. The atomic weight of ^3H is 3.016. (a) Determine to what nucleus ^3H decays. (b) Calculate the mass in grams of 1 mCi of tritium.

Calculation (5 marks)

PROBLEM 2.43 (First Test 1994) (First Test 1996)

Uranium in nature consists of 99.3% U-238 with a half-life of 4.47×10^9 years and 0.7% U-235 with a half life of 7.04×10^8 years. Calculate the activity of one gram of natural uranium.

Calculation (5 marks)

PROBLEM 2.44 (First Test 1994)

Plutonium-239 undergoes α-decay with a half-life of 24,000 years. Compute the activity in curie of one gram of plutonium dioxide, $^{239}PuO_2$.

Calculation (5 marks)

PROBLEM 2.45

Write equations for the radioactive decay of the following and indicate if the final product is stable or not. If unstable continue to write equations until a stable element is reached:

(i)	n-1	(vi)	C-14
(ii)	H-3	(vii)	Co-60
(iii)	Li-5	(viii)	Xe-135
(iv)	He-7	(ix)	U-235
(v)	Li-8	(x)	U-238

PROBLEM 2.46 (Final Exam 1994)

Radio-active Nitrogen-16 is formed by the activation of Oxygen-16 in a water cooled reactor. Water or steam leaving the reactor is thus radioactive but N-16 has a short half life and this activity disappears quickly. If the initial activity of 1 kg of water is 1 μ Ci determine the number of N-16 atoms in the sample. Calculate the time for the activity of the sample to decay to one millionth of its initial value that is to 1 pCi. The half-life of N-16 is 7.13 seconds.

Calculation (5 marks)

2.5 RADIO-ACTIVE DECAY

PROBLEM 2.51 (First Test 1994)

Approximately what mass of ^{90}Sr ($T_{1/2}$ = 28.8 yr) has the same activity as 1 g of ^{60}Co ($T_{1/2}$ = 5.26 yr)?

Calculation (5 marks)

PROBLEM 2.54 (First Test 1991)

As of early 1981, approximately 1500 000 litres (1500 m^3) of radioactive water were located in the basement of the containment building of the Three Mile Island Unit 2 nuclear plant. One of the principal sources of this radioactivity was ^{137}Cs measured at 156 μCi/cm^3. Calculate how many atoms per cm^3 of this radionuclide were in the water at that time? The half life of ^{137}Cs is 30 years.

Calculation (5 marks)

PROBLEM 2.55 (First Test 1995)

Since the half-life of U-235 (7.13×10^8 years) is less than that of U-238 (4.51×10^9 years), the isotopic abundance of U-235 has been steadily decreasing since the earth was formed about 4.5 billion years ago. Determine how long ago the isotopic abundance of U-235 was equal to 3.0 weight percent, that is, the enrichment of the uranium used in many nuclear power plants. The present day isotopic abundance of U-235 is 0.72%.

2.6 RADIO-ACTIVE CHAINS

PROBLEM 2.61

When one radioactive isotope A decays into another B and that in turn decays into a third C a differential equation indicating the quantity of B present may be established as follows:

$$dN_B = \lambda_A N_A \, dt - \lambda_B N_B \, dt$$

The solution of this differential equation is different in different text books:

$$N_B = N_{BO}\, e^{-\lambda Bt} + N_{AO}\, \lambda_A/(\lambda_B - \lambda_A)(e^{-\lambda At} - e^{-\lambda Bt})$$

$$N_B = N_{AO}\, \lambda_A/\lambda_B - \lambda_A)\,(e^{-\lambda At} - e^{-\lambda Bt})$$

Explain why different solutions are apparently obtained from the same equation. If there is an error in one or other text book explain what it is. Plot the relative values of N_B as given by the above equations and show clearly the influence of each term.

PROBLEM 2.62

Isotope A decays to form Isotope B in a radioactive decay chain. Plot the relative quantities of A and B over a period of 100 hours for the following three cases:

(a) Half life of A = 25 hrs, half life of B = 4 hrs
(b) Half life of A = 10 hrs, half life of B = 10 hrs
(c) Half life of A = 4 hrs, half life of B = 25 hrs

PROBLEM 2.63

Xenon-135 is a strong neutron absorber and is hence detrimental to the operation of a nuclear reactor. Xenon-135 is formed from the decay of Iodine-135 which is a fission product and Xenon-135 in turn decays to Cesium-135. If initially there is a sample of 1 kg of Iodine-135 plot the quantities of Iodine-135 and Xenon-135 in the sample every 10 hours up to 100 hours. Determine at what point in time the Xenon-135 concentration reaches a maximum.

PROBLEM 2.64 (Final Exam 1994)

Uranium-238 can absorb neutrons to become Uranium-239. This decays fairly rapidly by β-particle emission to become first Neptunium-239 and then Plutonium-239. Thus effectively Plutonium-239 builds up directly from neutron absorption in Uranium-238. If 1 kg of Uranium-238 is subject to a neutron flux of 10^{14} neutrons/cm^{2}s determine the rate of buildup of Plutonium-239 and the quantity of Plutonium-239 present after 1 year.

Microscopic absorption cross-section of U-238 σ_a = 2.7 b
Microscopic absorption cross-section of Pu-239 σ_a = 1013 b

Calculation (5 marks)

QUESTION 2.65 (First Test 1991)

Each isotope can be represented by its chemical symbol, its atomic number, and its atomic mass number as follows:

$_ZX^A$, where Z = atomic number,
X = chemical symbol,
A = atomic mass number.

$_{92}U^{238}$ is the first isotope of a long radioactive decay chain which ends with a stable isotope of lead. This chain begins with $_{92}U^{238}$ emitting an å, the following four isotopes emitting β^-, β^-, α and α respectively. Give the chemical symbol, atomic number and atomic mass number of the first five radioactive daughters of $_{92}U^{238}$.

GIVEN:

Z	79	80	81	82	83	84	85	86	87	88	89	90	91	92
X	Au	Hg	T"	Pb	Bi	Po	At	Rn	Fr	Ra	Ac	Th	Pa	U

(5 marks)

2.7 RADIO-ACTIVE BUILD-UP

PROBLEM 2.71

When a radioactive isotope builds up due to neutron activation a differential equation indicating the quantity of that isotope present may be established as follows:

$$dN = R\,dt - \lambda N\,dt$$

The solution of this differential equation is different in different text books:

$$N = N_o e^{-\lambda t} + (R/\lambda)(1 - e^{-\lambda t})$$

$$N = (R/\lambda)(1 - e^{-\lambda t})$$

Explain why different solutions are apparently obtained from the same equation. If there is an error in one or other text book explain what it is. Plot the relative values of N as given by the above equations and show clearly the influence of each term.

PROBLEM 2.72 (First Test 1992) (Final Exam 1995)

If 1 g of Cobalt-59 is irradiated in a neutron flux of 10^{14} neutrons/cm^{2}s for a long period illustrate in a sketch the build-up of Cobalt-60 due to neutron absorption. If it is removed from the reactor during this period show sketch graphically how it would subsequently decay. Calculate the activity of the sample if it were removed from the reactor after a period of 1 year. Determine how long it would take for the activity of the sample to drop to one third of its activity after removal from the reactor.

Use the following data:

Half life of Co-60 = 5.27 years
Microscopic cross section of Co-59 = 37.2 barns
Density of Cobalt = 8.8 g/cm^3

(10 marks)

PROBLEM 2.73 (Final Exam 1992) (Final Exam 1993)

Cobalt-60 is widely used as a strong emitter of γ-rays for irradiation purposes. If 10 g of Cobalt-59 is placed in a reactor having a neutron flux of 10^{14} neutrons/cm^{2}s, calculate the amount of Cobalt-60 built up over a period of 1 year. Determine also the activity of the sample on removal from the reactor after this period of irradiation.

Use the following data:

Half life of Co-60 = 5.27 years
Microscopic cross section of Co-59 = 37.2 barns
Density of Cobalt = 8.8 g/cm^3

Calculation (5 marks)

PROBLEM 2.74 (First Test 1993) (Final Exam 1993) (Final Exam 1994)

The fission product I-131 has a half-life of 8.05 days and is produced in fission with a yield of 2.9 percent, that is, 0.029 atoms of I-131 are produced per fission. Calculate the equilibrium activity of this radionuclide in a reactor operating at 3300 MW.

Calculation (5 marks)

2.8 BINDING ENERGY

PROBLEM 2.81 (First Test 1991) (Final Exam 1992) (First Test 1993)

(a) Explain what is meant by:

- mass defect
- binding energy

(b) Sketch the relationship between binding energy and mass number.

(c) Explain why energy is released from:

- the fusion of two light nuclei;
- the fission of a heavy nucleus.

(d) As the atomic number increases, the ratio of neutrons to protons in the nucleus increases. Briefly explain why this trend occurs and why the ratio is not linear.

Description (10 marks)

PROBLEM 2.83

Calculate the amount of energy released when a neutron decays. In what form is this energy.

PROBLEM 2.84 (First Test 1991) (Final Exam 1992) (First Test 1994)

Consider the following fission reaction:

$$n + {}_{92}U^{235} \rightarrow {}_{54}Xe^{139} + {}_{38}Sr^{94} + 3n + E$$

Calculate the energy E (in MeV) released in this reaction.

Refer to the data below for atomic masses.

DATA:

Atom (or Particle)	Mass (amu)
^{3}H	3.016
^{2}H	2.014
^{4}He	4.002
^{235}U	235.044
^{139}Xe	138.918
^{94}Sr	93.915
n	1.009

(5 marks)

PROBLEM 2.85 (Final Exam 1994) (First Test 1996) (Final Exam 1996)

Plutonium-239 can undergo fission after absorbing a neutron. If the fission products are Cesium-133 and Palladium-104 determine the amount of energy released in the fission process. Use the following data:

Atom or particle	mass (amu)
Cs-133	132.905
Pd-104	103.90403
Pu-239	239.05216
n	1.00867

Calculation (5 marks)

PROBLEM 2.86

Compare the binding energy per nucleon of Uranium-235 plus a neutron and the binding energy per nucleon of Uranium-236. Compare the results with Table 3.3 in the Text Book.

PROBLEM 2.87 (First Test 1992) (Final Exam 1992) (First Test 1993) (First Test 1996)

(a) Calculate the average binding per nucleon of Uranium-235.

(b) Estimate the energy added when a thermal neutron enters a Uranium-235 nucleus.

(c) State the difference between fissile and fissionable nuclides.

Use the following data:

Mass of proton = 1.007277 amu
Mass of neutron = 1.008665 amu
Mass fo U-235 nucleus = 234.993453 amu
Mass of U-236 nucleus = 235.995092 amu

Calculation (5 marks)

PROBLEM 2.88 (Final Exam 1991) (Final Exam 1993) (Final Exam 1995)

(a) Calculate the binding energy per nucleon in MeV of helium.

(b) Consider a fusion reaction in which two deuterium atoms react to form tritium and hydrogen. Calculate the energy released in MeV in such a reaction given the following data:

atom or particle	mass (amu)
^{4}He	4.00260
^{3}H	3.01605
^{2}H	2.01410
^{1}H	1.00782
n	1.008665
p	1.007277

Calculation (5 marks)

2.9 NUCLEAR ENERGY

PROBLEM 2.91 (First Test 1992)

Calculate the power production when a nuclear plant completely consumes 3 kg of U-235 per day. Assume a thermal efficiency of 30% for the plant.

Calculation (5 marks)

PROBLEM 2.92 (First Test 1994)

A nuclear plant with an output of 1000 MW (electrical) operates at full load for one year. During this time the enriched uranium fuel burns down from an enrichment of 3.3% to an enrichment of 2.8%. Determine the quantity of fuel initially loaded into the reactor. If this fuel is in the form of uranium dioxide determine the mass of UO_2 in the core. Assume that the plant has an overall thermal efficiency of 32%.

Calculation (5 marks)

PROBLEM 2.93

If 1 kg of Pu-239 in a reactor is consumed by fission in 1 day determine the thermal power output of that reactor.

PROBLEM 2.94 (First Test 1992) (Final Exam 1994)

Polonium-210 decays to the ground state of ^{206}Pb by the emission of a 5.305-MeV α-particle with a half-life of 138 days. Calculate the mass of ^{210}Po is required to produce 1 MW of thermal energy from its radioactive decay.

Calculation (5 marks)

PROBLEM 2.95 (Final Exam 1991) (Final Exam 1993) (Final Exam 1995)

The radioisotope generator SNAP-9 was fueled with 475 g of ^{238}PuC (plutonium-238 carbide), which has a density of 12.5 g/cm^3. The ^{238}Pu has a half-life of 89 years, and emits 5.6 MeV per disintegration, all of which may be assumed to be absorbed in the generator. The thermal to electrical efficiency of the system is 5.4 percent. Calculate (a) the thermal output and (b) the electrical power of the generator.

Calculation (5 marks)

PROBLEM 2.96 (First Test 1993)

The yields of nuclear weapons are measured in kilotons, where 1 kiloton = 2.6×10^{25} MeV. Calculate how much ^{235}U is fissioned when a 100 kiloton bomb is exploded? The density of Uranium is 19.1 g/cm^3.

Calculation (5 marks)

PROBLEM 2.97 (Final Exam 1991) (Final Exam 1992) (Final Exam 1993) (Final Exam 1996)

Calculate (a) the power output (in MW) and (b) the rate of fuel consumption (in kg U-235/day) of a typical PWR reactor given the following data:

Number of Fuel Assemblies in Reactor	157	
Number of Fuel Rods per Assembly	264	(17 x 17 array)
Fuel Rod Outside Diameter	9.5	mm
Fuel Rod Cladding Thickness	0.57	mm
Fuel Pellet Diameter	8.19	mm
Fuel Rod Lattice Pitch	12.6	mm
Fuel Rod Effective Length	3.658	m
Uranium Dioxide Density	10.4	g/cm^3
Average U-235 Enrichment	2.8	%
Effective U-235 Fission Cross Section	380	barns
Average Neutron Flux	4.5×10^{13} neutron/cm^{2}s	

Calculation (5 marks)

CHE 3804 NUCLEAR ENGINEERING

QUESTION BANK SECTION 3

CHAPTER 3

INTERACTION OF RADIATION WITH MATTER

3.0 NUCLEAR INTERACTIONS

PROBLEM 3.01 (First Test 1995) (First Test 1996) (Final Exam 1996)

Describe briefly five different neutron interactions with nuclei. In each case state clearly the steps in the process and the products of the interaction.

(10 marks)

PROBLEM 3.02 (First Test 1992)

In a CANDU reactor the following reactions occur in the moderator and coolant:

(a) Neutron activation of the Deuterium in the heavy water creates Tritium. The Tritium subsequently decays by negative beta emission.

(b) High energy neutrons absorbed by the Oxygen-16 in the water create Nitrogen-16. Nitrogen-16 subsequently decays by negative beta and high energy gamma ray emission.

Write down the equations for these four reactions showing the balance of A and Z numbers.

PROBLEM 3.03 (First Test 1996)

In a CANDU reactor the following reactions occur in the fuel:

(a) Neutrons are absorbed by Uranium-238 in the fuel to create Uranium-239. This subsequently decays to Neptunium-239 and then to Plutonium-239.

(b) Iodine-135 is created directly by fission. This decays to Xenon-135 and then to Cesium-135.

Write down the equations for these absorption and decay reactions showing the balance of A and Z numbers.

Equations (5 marks)

PROBLEM 3.04 (First Test 1993) (Final Exam 1993) (First Test 1994) (First Test 1995) (Final Exam 1996)

Attached are two pages of the Chart of Nuclides. With reference to this chart write full equations (indicating mass number and charge or atomic number of all constituents) for the following reactions:

(a) Interacting Reactions

- (i) Neutron producing reaction by the impact of α-particles on Boron-11.
- (ii) Neutron producing reaction by the impact of α-particles on Beryllium-9.
- (iii) Neutron producing reaction by the effect of high energy γ-rays on Deuterium.
- (iv) Nuclear transmutation by the reaction of fast neutrons with Oxygen to form Nitrogen.
- (v) Fusion of Tritium and Deuterium to produce Helium.

(b) Decay Reactions

- (i) Decay of a neutron
- (ii) Decay of Nitrogen-16
- (iii) Decay of Carbon-14
- (iv) Decay of Lithium-5
- (v) Decay of Helium-7

Equations (10 marks)

3.1 NEUTRON CROSS SECTIONS

PROBLEM 3.11 (Final Exam 1994)

A beam of 2-MeV neutrons is incident on a slab of heavy water (D_2O). The total cross sections of deuterium and oxygen at this energy are 2.6 b and 1.6 b, respectively. Calculate the following:

(a) The macroscopic total cross section of D_2O at 2 MeV.

(b) The thickness of the slab in order to reduce the intensity of the uncollided beam by a factor of 10.

(c) If an incident neutron has a collision in the slab, what is the relative probability that it collides with deuterium.

Calculation (10 marks)

PROBLEM 3.12 (First Test 1996)

Stainless steel type 304 having a density of 7.86 g/cm^3 has been used in some reactors. The nominal composition by weight of this material is as follows: carbon, 0.08 percent; chromium, 19 percent; nickel, 10 percent; iron, the remainder. Calculate the macroscopic absorption cross section of SS-304 at 0.0253 eV.

The microscopic absorption cross section at thermal conditions and atomic weights are as follows:

Carbon	$\sigma(n, \gamma)$ =	0.0034 barn	MW = 12.0
Chromium	$\sigma(n, \gamma)$ =	3.1 barn	MW = 52.0
Nickel	$\sigma(n, \gamma)$ =	4.43 barn	MW = 58.7
Iron	$\sigma(n, \gamma)$ =	2.55 barn	MW = 55.8

Calculation (5 marks)

PROBLEM 3.13 (First Test 1991) (First Test 1993) (Final Exam 1993) (First Test 1994)

Calculate at 0.0253 eV the total macroscopic absorption cross section of uranium dioxide (UO_2) in which the uranium has been enriched to 3% by weight in ^{235}U. The density of UO_2 is approximately 10.5 g/cm^3. The microscopic cross sections at 0.0253 eV are as follows

U-238	$\sigma(n,\gamma) = 2.7$ barn
U-235	$\sigma(n, \gamma) = 99$ barn
U-235	$\sigma(n, f) = 582$ barn

Calculation (5 marks)

3.2 NEUTRON ATTENUATION

PROBLEM 3.21 (First Test 1996)

Calculate the mean free path of 1-eV neutrons in graphite. The total cross section of carbon at this energy is 4.8 barn.

Calculation (5 marks)

PROBLEM 3.22 (First Test 1995)

Calculate at 0.025 eV the moderating ratio ($\zeta\, \Sigma_a / \Sigma_a$) of heavy water ($D_2O$). The density of D_2O is approximately 1.10 g/cm^3. The microscopic cross sections at 0.0253 eV are as follows:

Heavy water σ_a = 0.00133 barn
σ_s = 13.6 barn
ζ = 0.51

Determine also the average number of collisions required for D_2O to reduce neutron energy from 2 MeV to 0.025 eV.

Calculation (5 marks)

PROBLEM 3.25 (First Test 1991) (First Test 1996)

Suppose that a fission neutron, emitted with an energy of 2 MeV, slows down to an energy of 0.025 eV as the result of successive collisions in a moderator. If, on the average, the neutron gains in lethargy the amount ξ in each collision, how many collisions are required if the moderator is graphite consisting of C-12. The logarithmic mean energy decrement ξ is given by the following equation:

$$\xi = 1 + \frac{(A - 1)^2}{2A} \mathrm{Ln} \left(\frac{A - 1}{A + 1}\right)$$

(5 marks)

PROBLEM 3.27 (First Test 1993)

Suppose that a fission neutron, emitted with an energy of 2 MeV, slows down to an energy of 0.025 eV as the result of successive collisions in a moderator. Determine how many collisions are required if the moderator is Beryllium consisting of Be-9. The logarithmic mean energy decrement ξ is given by the following equation:

$$\zeta = 1 + \frac{(A - 1)^2}{2A} \mathrm{Ln} \left(\frac{A - 1}{A + 1}\right)$$

Calculation (5 marks)

3.3 NEUTRON REACTIONS

PROBLEM 3.31 (Text Book Problem 3.1)

A monoenergetic beam of neutrons having an intensity of 4 x 10^{10} neutrons/cm^2-sec impinges on a target 1 cm^2 in area and 0.1 cm thick. There are 0.048 x 10^{24} atoms per cm^3 in the target and the total cross section at the energy of the beam is 4.5 b.

(a) What is the macroscopic total cross section?
(b) How many neutron interactions per second occur in the target?
(c) What is the collision density?

PROBLEM 3.32 (Text Book Problem 3.2)

The β^--emitter ^{28}Al (half-life 2.30 min) can be produced by the radiative capture of neutrons by ^{27}Al. The 0.0253-eV cross section for this reaction is 0.23 b. Suppose that a small, 0.01-g aluminum target is placed in a beam of 0.0253-eV neutrons having an intensity of 3 x 10^8 neutrons/cm^2s, which strikes the entire target. Calculate (a) the neutron density in the beam; (b) the rate at which ^{28}Al is produced; (c) the maximum activity (in curies) which can be produced in this experiment.

PROBLEM 3.33 (First Test 1992) (First Test 1994) (First Test 1995)

The 2200 meters-per-second flux in an ordinary water reactor is 1.5 x 10^{13} neutrons/cm^2s. Calculate the rate at which thermal neutrons are absorbed by the water.

Use the following date:

Microscopic cross section of Hydrogen = 0.332 barns
Microscopic cross section of Oxygen = 0.00027 barns

Calculation (5 marks)

PROBLEM 3.34 (Final Exam 1995) (First Test 1996)

The 2200 meters-per-second flux in a heavy water reactor is 1.5×10^{13} neutrons/cm^{2}s. Calculate the rate at which thermal neutrons are absorbed by the water.

Use the following date:

Microscopic cross section of Deuterium = 0.00053 barns
Microscopic cross section of Oxygen = 0.00027 barns

Calculation (5 marks)

3.4 FISSION ENERGY

PROBLEM 3.42 (First Test 1991)

Consolidated Edison's Indian Point No. 2 reactor is designed to operate at a power of 2758 MW. Assuming that all fissions occur in ^{235}U, calculate in kg per day the rate at which ^{235}U is fissioned.

Calculation (5 marks)

3.5 FISSION PRODUCT DECAY

PROBLEM 3.51 (Final Exam 1996)

The fission product I-131 has a half-life of 8.05 days and is produced in fission with a yield of 2.9 percent, that is, 0.029 atoms of I-131 are produced per fission. Calculate the equilibrium activity of this radionuclide in a reactor operating at 3300 MW.

Calculation (5 marks)

PROBLEM 3.53 (Text Book Problem 3.43)

The spontaneous fission rate of ^{238}U is 1 fission per gram per 100 sec. Show that this is equivalent to a half-life for fission of 5.5×10^{15} years.

PROBLEM 3.54 (Second Test 1992)

The thermal neutron cross section for the production of 1n-116 via the (n, γ) reaction with In-115 is 157 barns. A thin indium foil of mass 0.15 g is placed in a neutron beam tube having a thermal flux of 2×10^{13} neutrons/cm^{2}s. Determine the following:

(a) Rate of absorption of neutrons in foil.
(b) Activity of In-116 after 1 hour of irradiation

Assume that the initial sample contains only In-115.
The half life of In-116 is 54 minutes and the density of indium is 7.31 g/cm^3.

Calclulation (5 marks)

PROBLEM 3.55 (First Test 1991)

Given that the half-life for spontaneous fission of ^{238}U is 5.5×10^5 years, calculate the spontaneous fission rate in 1 kg of pure ^{238}U.

Calculation (5 marks)

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QUESTION BANK SECTION 4

CHAPTER 4

NUCLEAR REACTORS AND NUCLEAR POWER

4.0 REACTOR TYPES

PROBLEM 4.01

Describe the main differences and similarities between a Canadian Deuterium Uranium Reactor (CANDU) and a Pressurised Water Reactor (PWR).

PROBLEM 4.02 (Second Test 1996)

Describe the main differences and similarities between a Boiling Water Reactor (BWR) and a Pressurised Water Reactor (PWR). Emphasis should be on the reactor vessel internals and the reactor core.

Description (5 marks)

PROBLEM 4.03 (Final Exam 1994)

With respect to fuel element configuration and control rod arrangement compare the Boiling Water Reactor (BWR) and the Pressurised Water Reactor (PWR)

Description (5 marks)

PROBLEM 4.04 (First Test 1992) (Second Test 1995)

State the main differences between a PWR (Pressurised Water Reactor) and a CANDU (Canadian Deuterium Uranium) power plant. Where appropriate explain why these differences exist to accommodate the different technologies.

Description (5 marks)

PROBLEM 4.05

Explain why there is a variation in core size for a given output of the following reactors:

BWR	Boiling Water Reactor
PWR	Pressurised Water Reactor
CANDU	Canadian Deuterium Uranium Reactor
AGR	Advanced Gas Cooled Reactor
HTGR	High Temperature Gas Cooled Reactor
FBR	Fast Breeder Reactor

Arrange these in order of core power density and state what puts one reactor ahead of another.

4.1 NEUTRON CYCLE

PROBLEM 4.11 (Final Exam 1996)

Sketch the typical configuration of a thermal nuclear reactor showing typical neutron paths. Show also how the neutron cycle is controlled and how heat is removed. Describe the functions of the main components of a reactor.

PROBLEM 4.12 (Second Test 1991) (Second Test 1993) (Second Test 1994) (Second Test 1995)

(a) Give the four factor formula for neutron multiplication. Define each of the factors appearing in the formula in words or mathematically.

(b) Sketch a neutron cycle and show where and how each of the four factors affects the neutron population.

Description (10 marks)

PROBLEM 4.13 (Final Exam 1991) (Final Exam 1992) (Final Exam 1994)

(a) Write the 6 factor formula for the neutron multiplication factor and define each of the terms in the formula.

(b) Sketch the neutron cycle for one generation and show where neutrons are created and lost. Label the sketch with the factors defined in (a) above.

(c) State which factors can be varied by the operator and which cannot. Explain how the variable factors can be manipulated.

Description (10 marks)

4.2 REACTOR SAFETY

PROBLEM 4.21 (Second Test 1995)

Explain the purpose and principles of reactor shielding and containment.

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QUESTION BANK SECTION 5

CHAPTER 5

NEUTRON DIFFUSION AND MODERATION

5.0 NEUTRON DENSITY

PROBLEM 5.01 (Text Book Problem 5.1)

Two beams of 1-eV neutrons intersect at an angle of 90°. The density of neutrons in both beams is 2×10^8 neutrons/cm^3.

(a) Calculate the intensity of each beam.

(b) What is the neutron flux where the two beams intersect?

PROBLEM 5.02 (Text Book Problem 5.2)

Two monoenergetic neutron beams of intensities $I_1 = 2 \times 10^{10}$ neutrons/cm^{2}s and $I_2 = 1 \times 10^{10}$ neutrons/cm^{2}s intersect at an angle of 30°. Calculate the neutron flux in the region where they intersect.

PROBLEM 5.03 (Second Test 1996)

The thermal flux at the center of a graphite research reactor is 5×10^{12} neutrons/cm^{2}s. The temperature of the system at this point is 120°C. Compare the neutron density at this point with the atom density of the graphite.

Boltzmann constant: $k = 8.6170 \times 10^{-5}$ eV/°K
$k = 1.3806 \times 10^{-23}$ J/°K

PROBLEM 5.04 (Final Exam 1991)

The thermal (0.0253 eV) cross section for the production of a 54-minute isomer of ^{116}In via the (n, γ) reaction with ^{115}In is 157 b. A thin indium foil weighing 0.15 g is placed in the beam tube having a thermal neutron flux of 2×10^{13} neutrons/cm^{2}s. Calculate (a) The rate at which thermal neutrons are absorbed by the foil and (b) the 54 minute activity of ^{116}In after 1 hour in the beam tube. *Note :* The isotopic abundance of ^{115}In is 95.7%

(5 marks)

5.1 DIFFUSION PARAMETERS

PROBLEM 5.11 (Text Book Problem 5.31)

Calculate the thermal neutron diffusion length at room temperature in water solutions of boric acid (H_3BO_3) at the following concentrations:

(a) 10 g/litre,
(b) 1 g/litre, and
(c) 0.1 g/litre.

Hint: Because of the small concentration of the boric acid, the diffusion coefficient for the mixture is essentially the same as that of pure water.

PROBLEM 5.12 (Second Test 1991) (Second Test 1994)

Calculate the following for thermal neutrons in water:

(a) Diffusion coefficient.
(b) Diffusion length.
(c) Average direct ("crow flight") distance travelled.

For water: $\Sigma_a = 0.0222\ cm^{-1}$
$\Sigma_s = 3.44\ cm^{-1}$

(5 marks)

PROBLEM 5.13 (Second Test 1996)

Calculate the following for thermal neutrons in pure heavy water:

(a) Diffusion coefficient.
(b) Diffusion length.
(c) Average direct ("crow flight") distance travelled.

For pure heavy water: $\Sigma_a = 4.42 \times 10^{-5}\ cm^{-1}$
$\Sigma_a = 0.452\ cm^{-1}$

Calculation (5 marks)

PROBLEM 5.14 (Second Test 1992) (Second Test 1995)

Calculate the following for thermal neutrons in graphite:

(a) Diffusion coefficient
(b) Diffusion length
(c) Average direct ("crow flight") distance travelled

For graphite: $\Sigma_a = 0.0002728\ cm^{-1}$
$\Sigma_s = 0.3811\ cm^{-1}$

Calculation (5 marks)

5.2 NEUTRON FLUX VARIATION

PROBLEM 5.21 (Second Test 1991) (Final Exam 1994)

The neutron flux in a bare spherical reactor of radius 50 cm is given by

$$\phi = 5 \times 10^{13} \frac{\sin 0.0628r}{r} \text{ neutrons/cm}^2\text{s}$$

where r is measured from the centre of the reactor. The diffusion coefficient for the system is 0.80 cm. Determine the following:

(a) The maximum value of the flux in the reactor.

(b) The leakage rate of neutrons from the reactor.

Calculation (5 marks)

PROBLEM 5.22 (Second Test 1992) (Final Exam 1993)

The neutron flux in a bare spherical reactor is given approximately by

$$\phi = (A/r)(\sin \pi r/R) \quad \text{neutrons/cm}^2\text{s}$$

where r is measured from the centre of the reactor. For a reactor 1 m in diameter measurements indicate a maximum neutron flux at the centre of the reactor of 3×10^{12} neutrons/cm^{2}s and a leakage rate at the surface of the reactor of 5×10^{10} neutrons/cm^{2}s. Determine the diffusion coefficient of the material in the reactor.

Calculation (5 marks)

PROBLEM 5.23 (Second Test 1992) (Second Test 1994)

For a point source in an infinite medium the neutron flux at a distance r from the source is given by

$$\phi = \frac{Se^{-r/L}}{4\pi\ Dr}\ \text{neutrons/cm}^2\text{s}$$

where S is the source strength and D the diffusion coefficient. If a source of strength 10^7 neutrons/s is located in ordinary water determine the neutron flux 15 cm from the source.

For water: $\Sigma_a = 0.0222\ cm^{-1}$
$\Sigma_s = 3.44\ cm^{-1}$

Calculation (5 marks)

PROBLEM 5.24 (Second Test 1996)

For a point source in an infinite medium the neutron flux at a distance r from the source is given by

$$\phi = \frac{Se^{-r/L}}{4\pi\ Dr}\ \text{neutrons/cm}^2\text{s}$$

where S is the source strength and D the diffusion coefficient. If a source of strength 10^7 neutrons/s is located in pure heavy water determine the neutron flux 15 cm from the source.

For pure heavy water: $\Sigma_a = 4.42 \times 10^{-5}\ cm^{-1}$
$\Sigma_s = 0.452\ cm^{-1}$

Calculation (5 marks)

PROBLEM 5.25 (Second Test 1995)

For a point source in an infinite medium the neutron flux at a distance r from the source is given by:

$$\phi = \frac{Se^{-r/L}}{4\pi Dr} \text{ neutrons/cm}^2\text{s}$$

where S is the source strength and D the diffusion coefficient. If a source of strength 10^7 neutrons/s is located in graphite determine the neutron flux 15 cm from the source.

For graphite: $\Sigma_a = 0.0002728 \text{ cm}^{-1}$
$\Sigma_s = 0.3811 \text{ cm}^{-1}$

Calculation (5 marks)

PROBLEM 5.26 (Second Test 1993) (Second Test 1995)

The thermal flux in a bare spherical reactor 1 m in diameter is given approximately by

$$\phi(r) = 2.39 \times 10^{14} \frac{\sin 0.0628r}{r} \text{ neutrons/cm}^2\text{s}$$

The reactor is moderated and cooled by water of normal density which takes up to one third of the reactor volume. Consider the production of Deuterium ^2H in the water due to neutron absorption.

(a) Develop an equation to obtain the reaction rate at any point in the reactor.
(b) Determine the reaction rate at the centre.
(c) Determine the average reaction rate over the whole reactor.
(d) Calculate the number of Deuterium atoms produced in the reactor per unit time.

Microscopic absorption cross section for Hydrogen is $\sigma_a = 0.33$ barn.

Calculation (5 marks)

5.3 NEUTRON FLUX PROFILE

PROBLEM 5.31 (Second Test 1994) (Second Test 1996) (Final Exam 1996)

Sketch the neutron flux profile at the boundary between a diffusing medium and a non-diffusing medium (vacuum). Explain with reference to neutron diffusion why the actual flux profile changes as it does near the surface. Explain also how the neutron flux predicted by diffusion theory can be made to match the actual neutron flux within the diffusing medium. Show the relationship between the boundary geometrical dimensions and the diffusion coefficient D.

Description (5 marks)

PROBLEM 5.32 (Second Test 1991) (Second Test 1994) (Final Exam 1995)

(a) Sketch the neutron flux profile at the boundary of a diffusing medium and a non-diffusing medium (vacuum). Explain how the neutron flux predicted by diffusion theory can be made to match the actual neutron flux within the diffusing medium. Show the relationship between the boundary geometrical dimensions and the diffusion coefficient D.

(b) Sketch the thermal neutron flux profile from the centre of the core to the boundary of a bare reactor and of a reflected reactor. Explain why the neutron flux is different for the two cases. Indicate clearly the origin and direction of diffusion of the thermal neutrons.

Description (10 marks)

PROBLEM 5.33 (Second Test 1994)

A bare spherical reactor of radius 50 cm operates at a power level of 100 megawatts. If $\Sigma_f = 0.0047\ cm^{-1}$, calculate the maximum and average values of the flux in the reactor.

Calculation (5 marks)

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QUESTION BANK SECTION 6

CHAPTER 6

NUCLEAR REACTOR THEORY

Note: Make all computations at room temperature, unless othewise stated, and take recoverable energy per fission to be 200 MeV.

6.0 REACTOR PARAMETERS

PROBLEM 6.01 (Text Book Problem 6.1)

Calculate the fuel utilization and infinite multiplication factor for a fast reactor consisting of a mixture of liquid sodium and plutonium, in which the plutonium is present to 3.0 w/o. The density of the mixture is approximately 1.0 g/cm^3.

PROBLEM 6.02 (Text Book Problem 6.2)

The core of a certain fast reactor consists of an array of uranium fuel elements immersed in liquid sodium. The uranium is enriched to 25.6 w/o in ^{235}U and comprises 37 percent of the core volume. calculate for this core

(a) the average atom densities of sodium, ^{235}U, and ^{238}U;
(b) the fuel utilization;
(c) the value of η;
(d) the infinite multiplication factor.

PROBLEM 6.03 (Second Test 1996)

A homogeneous solution of ^{235}U and H_2O contains 10 grams of ^{235}U per liter of solution. Compute

(a) the atom density of ^{235}U and the molecular density of H_2O;
(b) the thermal utilization;
(c) the thermal diffusion area and length;
(d) the infinite multiplication factor.

Calculation (5 marks)

PROBLEM 6.04 (Text Book Problem 6.20)

A bare spherical reactor 50 cm in radius is composed of a homogeneous mixture of ^{235}U and beryllium. The reactor operates at a power level of 50 thermal kilowatts. Using modified one-group theory, compute:

(a) the critical mass of ^{235}U;
(b) the thermal flux throughout the reactor;
(c) the leakage of neutrons from the reactor;
(d) the rate of consumption of ^{235}U.

PROBLEM 6.05 (Text Book Problem 6.21)

It is proposed to store H_2O solutions of fully enriched uranyl sulfate ($^{235}UO_2SO_4$) with a concentration of 30 g of this chemical per litre. Is this a safe procedure when using a tank of unspecified size?

PROBLEM 6.06 (Second Test 1996)

(a) Explain the difference between homogeneous and heterogeneous reactors.
(b) Sketch the variation in the microscopic absorption across section of typical uranium fuel.
(c) With reference to the sketch in (b) above explain the advantages of a heterogeneous reactor over a homogeneous one.

Description (5 marks)

PROBLEM 6.09 (Second Test 1992) (Final Exam 1992)

A heterogeneous uranium-water reactor lattice consists of an array of natural uranium metal rods 1.50 cm in diameter with a pitch of 2.80 cm. Calculate the following:

(a) Radius of equivalent cell
(b) Uranium to water ratio
(c) Thermal utilization factor assuming a uniform flux profile
(d) Reproduction factor
(e) Resonance escape probability
(f) Fast fission factor
(g) Neutron multiplication factor

The factor f may be calculated assuming the same neutron flux in the fuel as in the moderator.

The factor p for this lattice is given by:

$$p = \exp\left[-\frac{N_F \times 10^{-24} \times V_F \times 12.92}{1.46 \times V_M}\right]$$

where N_x is the number of fuel atoms per unit volume, and V_F and V_M are the volumes of fuel and moderator respectively.

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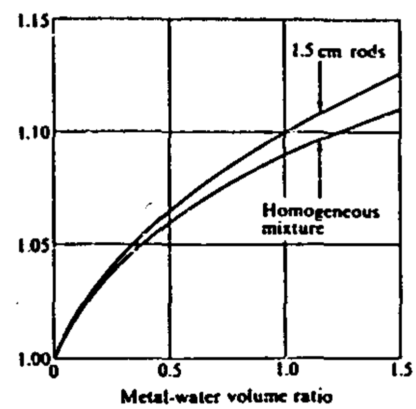

Fig. 6.10 The fast fission factor as a function of metal–water volume ratio.

The factor ϵ for this lattice is given by the adjoining diagram.

For natural uranium: $\Sigma_a = 0.3668\ cm^{-1}$
$\Sigma_f = 0.2025\ cm^{-1}$
$v = 2.418$
$\rho = 19.1\ g/cm^3$

For ordinary water: $\Sigma_a = 0.0222\ cm^{-1}$

Calculation (10 marks)

6.1 POWER LEVEL

PROBLEM 6. 11 (Second Test 1994)

The core of a certain reflected reactor consists of a cylinder 3 m high and 3 m in diameter. The same fluid is used as moderator and coolant and the fuel rods are 9.5 mm in diameter on a pitch of 12.6 mm in a square array. The measured maximum to average neutron flux is 1.5. When the reactor is operated at a power level of 825 MW calculate the following:

(a) Maximum power density in kW/litre.
(b) Maximum heat flux at fuel element surface in kW/m^2.

Calculation (5 marks)

PROBLEM 6.12 (Text Book Problem 6.3)

A bare cylindrical reactor of height 100 cm and diameter 100 cm is operating at a steady-state power of 20 MW. If the origin is taken at the center of the reactor, determine the power density at the point $r = 7$ cm and $z = -22.7$ cm?

PROBLEM 6.13 (Text Book Problem 6.4)

In a spherical reactor of radius 45 cm the fission rate density is measured as 2.5×10^{11} fissions/cm^{3}s at a point 35 cm from the center of the reactor.

(a) At what steady-state power is the reactor operating?

(b) What is the fission rate density at the center of the reactor?

PROBLEM 6.14 (Second Test 1996)

The core of a certain reflected reactor consists of a cylinder 3.048 m high and 3.048 m in diameter. The measured maximum-to-average flux is 1.5. When the reactor is operated at a power level of 825 MW, what is the maximum power density in the reactor in kW/litre?

Calculation (5 marks)

PROBLEM 6.15 (Second Test 1992) (Second Test 1994)

The core of a certain reflected reactor consists of a cylinder 3 m high and 3 m in diameter. The same fluid is used as moderator and coolant and the fuel rods are 9.5 mm in diameter on a pitch of 12.6 mm in a square array. The measured maximum to average neutron flux is 1.5. When the reactor is operated at a power level of 825 MW calculate the following:

(a) Maximum power density in kW/litre.
(b) Maximum heat flux in kW/m^2.

Calculation (5 marks)

PROBLEM 6.16 (Second Test 1993) (Second Test 1995) (Final Exam 1995)

A bare rectangular reactor of height 120 cm and each side 100 cm is operated at a steady-state power of 20 MW. If the origin is taken at the center of the reactor, calculate the power density at the point x = 7 cm, y = 30 cm and z = - 22 cm?

Calculation (5 marks)

6.2 REACTOR DESIGN

PROBLEM 6.23 (Second Test 1991) (Second Test 1994) (Second Test 1995)

A large research reactor consists of a cubical array of natural uranium rods in a graphite moderator. The length a of each side of the reactor is 7.62 m and it operates at a power P of 20 MW. The average macroscopic cross section for fission Σ_f is 2.5 x 10^{-3} cm^{-1}. Calculate the following:

(a) Buckling
(b) Maximum value of thermal neutron flux.
(c) Average value of thermal neutron flux.

Calculation (5 marks)

PROBLEM 6.24 (Final Exam 1992)

A large research reactor consists of a cubical array of natural uranium rods in a graphite moderator. The length a of each side of the reactor is 7.62 m and it operates at a power P of 20 MW. The average macroscopic cross section for fission Σ_f is 2.5 x 10^{-3} cm^{-1}. Calculate the following:

(a) Buckling
(b) Maximum value of thermal neutron flux.
(c) Average value of thermal neutron flux.

Calculation (5 marks)

PROBLEM 6.26 (Second Test 1996)

The original version of the Brookhaven Research Reactor consisted of a cube of graphite which contained a regular array of natural uranium rods, each of which was located in an air channel through the graphite. When the reactor was operated at a thermal power level of 22 MW, the average fuel temperature was approximately 300°C and the maximum thermal flux was 5×10^{12} neutrons/cm^{2}s. The average values of L^2_T and τ_T were 325 cm^2 and 396 cm^2, respectively, and k_∞ was equal to 1.0735.

(a) Calculate the critical dimensions of the reactor.
(b) Calculate the total amount of natural uranium the reactor.

Calculation (5 marks)

PROBLEM 6.27 (Second Test 1991) (Second Test 1993)

The original version of the Brookhaven Research Reactor consisted of a cube of graphite which contained a regular array of natural uranium rods each of which was located in an air channel through the graphite. When the reactor was operated at a thermal power level of 22 MW the maximum thermal flux was 5×10^{12} neutron/cm^{2}s. The average values of L^2_T and τ_T were 325 cm^2 and 396 cm^2 respectively and k_∞ was equal to 1.0735. Determine the following:

(a) Critical dimensions and volume of the reactor in cubical form.
(b) Critical dimensions and volume of an equivalent reactor in spherical form.

Calculation (5 marks)

6.3 NEUTRON FLUX FLATTENING

PROBLEM 6.31 (Second Test 1992) (Second Test 1996)

State the difference between single group, two group and multigroup calculations as applied to neutron moderation and diffusion in a reactor. Explain why it is necessary to use multigroup calculations and what the advantage is of a two group calculation.

Description (5 marks)

PROBLEM 6.32 (Second Test 1995) (Final Exam 1996)

(a) Explain why neutron flux flattening is desirable.
(b) Give three methods for achieving flux flattening.

Description (5 marks)

PROBLEM 6.33 (Second Test 1991) (Final Exam 1992) (Final Exam 1993)

(a) Explain why neutron flux flattening is desirable.

(b) Give three methods for achieving flux flattening.
For each method explain how the flux profile is changed and sketch the flattened profile along with the original profile to show the change.

Description (10 marks)

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QUESTION BANK SECTION 7

CHAPTER 7

THE TIME DEPENDENT REACTOR

7.0 REACTOR KINETICS

PROBLEM 7.03 (Final Exam 1994)

A ^{235}U fueled reactor originally operating at a power of 1 milliwatt is placed on a positive 10 minute period. Calculate the time that the reactor will take to reach a power level of 1 megawatt.

Calculation (5 marks)

PROBLEM 7.04 (Text Book Problem 7.16)

It is desired to double the power level of a ^{235}U fueled reactor in 20 minutes.

(a) On what period should the reactor be placed?
(b) How much reactivity should be introduced?

PROBLEM 7.06 (Final Exam 1991) (Final Exam 1996)

During test-out procedures a thermal reactor is operated for a time at a power of 1 megawatt. The power is then to be increased to 100 megawatts in 8 hours. Calculate the following :

(a) The stable period on which the reactor should be placed.
(b) The positive reactivity insertion required (in mk).

Calculation (5 marks)

PROBLEM 7.07 (Final Exam 1992) (Final Exam 1995)

During power raising on a power reactor 5 mk of positive reactivity is inserted by manipulation of the control system. Calculate how long it will take for the power level to increase from 20% to 100%.

Calculation (5 marks)

PROBLEM 7.08 (Final Exam 1993)

(a) During power raising on a power reactor 0.1 mk of positive reactivity is inserted by manipulation of the control system. Calculate how long it will take for the power level to increase from 20% to 100%. Determine also the reactor period during this increase in power.

(b) A reactor is scrammed by the instantaneous insertion of 50 mk of negative reactivity after having reached a power level of 1 megawatt. Calculate how long it takes the power level to drop to 1 milliwatt.

Calculation (10 marks)

PROBLEM 7.09 (Final Exam 1991) (Final Exam 1993) (Final Exam 1994) (Final Exam 1996)

During approach to criticality in a CANDU reactor the strength of the photoneutron source is 0.001 % of the neutron power level at full load.

(a) If the value of the neutron multiplication factor k is 0.975 calculate the measured neutron power level.

(b) Determine how much positive reactivity must be added to double the neutron power level.

Calculation (5 marks)

7.1 REACTOR CONTROL

PROBLEM 7.11 (Final Exam 1991) (Final Exam 1995) (Final Exam 1996)

A reactor is scrammed by the instantaneous insertion of 50 mk of negative reactivity after having reached a power level of 1 megawatt. Calculate how long it takes the power level to drop to 1 milliwatt.

Calculation (5 marks)

PROBLEM 7.12 (Text Book Problem 7.13)

When a certain research reactor operating at a power of 2.7 megawatts is scrammed, it is observed that the power drops to a level of 1 watt in 15 minutes. How much reactivity was inserted when the reactor was scrammed?

Calculation (5 marks)

PROBLEM 7.13 (Final Exam 1991)

Explain what is meant by "chemical shim" and how and when it is used. State what chemical is commonly used and what its characteristics are.

Calculation (5 marks)

PROBLEM 7.14 (Final Exam 1992) (Final Exam 1996)

Explain the basic principles concerning the approach to criticality in a nuclear reactor. State what measurements are obtained from the instrumentation and how these are interpreted by the operators to predict the point of criticality.

Description (5 marks)

7.2 REACTOR POISONS

PROBLEM 7.21 (First Test 1993) (Final Exam 1993)

The fission product I-131 has a half-life of 8.05 days and is produced in fission with a yield of 2.9 percent, that is, 0.029 atoms of I-131 are produced per fission. Calculate the equilibrium activity of this radionuclide in a reactor operating at 3300 MW.

Calculation (5 marks)

PROBLEM 7.22 (Final Exam 1993) (Final Exam 1995)

Explain how Xenon-135 is created and destroyed in an operating reactor. If appropriate state the equations governing the rate of change of Xenon-135 in the reactor. The effect of changes in neutron flux on the rate of change of Xenon-135 concentration as well as the effect on the neutron flux of the Xenon-135 must be clarified.

Description (5 marks)

PROBLEM 7.25 (Final Exam 1991)

(a) Give the equations for the variation with time of the concentration of Iodine-135 and Xenon-135 in a reactor.

(b) A reactor which has been operating at 100% power for an extended period of time has its power reduced to 50% at a certain point in time. Show graphically a plot of the Iodine-135 and Xenon-135 concentrations relative to their equilibrium values before and after the step change in power. The plot must include a time scale and the concentrations must be consistent with the time.

Iodine-135 $\gamma = 0.063 = 6.3\%$
$\lambda = 2.87 \times 10^{-5}\ s^{-1} = 0.1035\ hr^{-1}$
$t_{1/2} = 6.7$ hr

Xenon-135 $\gamma = 0.003 = 0.3\%$
$\lambda = 2.09 \times 10^{-5}\ s^{-1} = 0.0753\ hr^{-1}$
$t_{1/2} = 9.2$ hr
$\sigma = 3.5 \times 10^{6}$ b

(10 marks)

PROBLEM 7.26 (Final Exam 1992)

Explain how Xenon-135 is created and destroyed in an operating reactor. If appropriate state the equations governing the rate of change of Xenon-135 in the reactor. The effect of changes in neutron flux on the rate of change of Xenon-135 concentration as well as the effect on the neutron flux of the Xenon-135 must be clarified.

Description (5 marks)

7.3 FUEL BURNUP

PROBLEM 7.31 (Final Exam 1995)

A nuclear plant with an output of 1000 MW (electrical) operates at full load for one year. During this time the enriched uranium fuel burns down from an enrichment of 3.3% to an enrichment of 2.8%. Determine the quantity of fuel initially loaded into the reactor. If this fuel is in the form of uranium dioxide determine the mass of UO_2 in the core. Assume that the plant has an overall thermal efficiency of 32%.

Calculation (5 marks)

PROBLEM 7.32 (Final Exam 1991) (Final Exam 1992) (Final Exam 1995) (Final Exam 1996)

A CANDU reactor which has been operating for a period without refuelling has, at the time in question, an average fuel burnup of 0.4 n/kb. ($\phi t = 0.4 \times 10^{-3}$ neutrons/barn) Using the data given below determine the following at this level of burnup:

(a) Concentration of U-235 (g/kg U)
(b) Concentration of Pu-239 (g/kg U)
(c) Reactor Thermal Power Output (MW)

Mass of uranium dioxide in core	m = 97 tonnes
Density of uranium dioxide in core	$\rho = 10.4$ g/cm^3
Natural abundance of U-235 in new fuel	$f = 0.72\%$
Microscopic absorption cross section of U-238	$\sigma_a = 2.7$ b
Microscopic absorption cross section of U-235	$\sigma_a = 678$ b
Microscopic fission cross section of U-235	$\sigma_f = 580$ b
Microscopic absorption cross section of Pu-239	$\sigma_a = 1013$ b
Microscopic fission cross section of Pu-239	$\sigma_f = 742$ b
Average prevailing neutron flux in core	$\phi = 7 \times 10^{13}$ neutrons/cm^{2}s

Calculation (10 marks)

PROBLEM 7.33 (Final Exam 1991)

Describe the fundamental principles and differences in the fuel management of a CANDU and of a PWR. The following should be addressed: refuelling frequency, reactivity variation, burnup characteristics, enrichment levels and flux flattening due to fuel.

Description (10 marks)

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QUESTION BANK SECTION 8

SUPPLEMENTARY NOTES

NUCLEAR WASTE MANAGEMENT

8.0 WASTE CHARACTERISTICS

PROBLEM 8.01

Describe the characteristics of used nuclear fuel on discharge from a nuclear reactor and subsequently when in storage.

Description (5 marks)

PROBLEM 8.02 (Final Exam 1994)

With respect to radiation explain when in the fuel cycle from fuel element manufacture to disposal a fuel element is dangerous to man.

Description (5 marks)

PROBLEM 8.03 (Final Exam 1994)

Explain the difference between transuranic waste and fission products and state in what way these are a health hazard to man.

Description (5 marks)

8.1 WASTE DISPOSAL

PROBLEM 8.11 (First Test 1994)

State the hazards of used nuclear fuel and show how these hazards change with time. Describe how used CANDU fuel bundles are stored and explain what criteria govern the mode of storage at various times after removal from the reactor. Clarify what types of barriers are required at various stages to provide protection against radiation and contamination.

Description (10 marks)

PROBLEM 8.12 (Final Exam 1995)

State and explain the general philosophies behind the following methods for the storage of spent fuel elements:

- Spent fuel storage pools at plant
- Dry storage canisters above ground
- Permanent disposal vaults below ground

Description (5 marks)

PROBLEM 8.13 (Final Exam 1994)

Describe the method of above ground dry storage and explain what precautions have to be taken to prevent environmental contamination.

Description (5 marks)

PROBLEM 8.14

Describe the method of below ground permanent disposal and explain what lessons have been learned from nature with regard to the containment of radioactive products.

PROBLEM 8.15 (First Test 1991)

Explain the general philosophies regarding the permanent underground storage of spent reactor fuel elements.

Description (5 marks)

PROBLEM 8.16 (Final Exam 1992)

Explain the general philosophies regarding the temporary and semi-permanent above ground storage of spent reactor fuel elements.

Description (5 marks)

PROBLEM 8.17 (Final Exam 1993)

Explain the general philosophies regarding the permanent below ground storage of spent reactor fuel elements.

Description (5 marks)

PROBLEM 8.18 (Second Test 1993)

Write an essay of about 300 words explaining the technicalities of a particular aspect of nuclear waste storage and disposal. The essay should be in good English and in simple language as if it were to be published in a public relations brochure.

Essay (10 marks)

8.2 ENVIRONMENTAL CONSIDERATIONS

PROBLEM 8.21 (Final Exam 1996)

Discuss the environmental impact of power production. Compare the effects of conventional power generation with those of nuclear power generation. The discussion should be of at least 300 words and in good English.

PROBLEM 8.22 (Final Exam 1996)

Write an essay of at least 300 words explaining the risks of power production from various forms of energy. The essay should be in good English and in simple language as if it were to be published in a public relations brochure.

Essay (10 marks)

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QUESTION BANK SECTION 9

CHAPTER 9

RADIATION PROTECTION

9.0 BIOLOGICAL EFFECTS

PROBLEM 9.01 (Final Exam 1995)

Explain the following radiological terms

- Exposure Rate
- Absorbed Dose
- Relative Biological Effectiveness
- Quality Factor
- Dose Equivalent

Description (5 marks)

PROBLEM 9.02 (Second Test 1994)

(a) Explain the following radiological terms

- Exposure Rate
- Absorbed Dose
- Relative Biological Effectiveness
- Quality Factor
- Dose Equivalent

(5)

(b) Describe the biological effects of radiation. Give the probable early effects of different levels of acute whole body radiation doses.

(5)

Description (10 marks)

PROBLEM 9.03 (Final Exam 1991) (Final Exam 1995)

Describe the most important biological effects of large acute doses of radiation with respect to humans. Make a clear distinction between early effects and late effects and, where appropriate, give values of radiation dose received.

Description (10 marks)

PROBLEM 9.04 (Final Exam 1992)

Describe the biological effects of radiation and relate these to similar chemical effects. Distinguish between stochastic and non-stochastic effects.

Description (5 marks)

PROBLEM 9.05 (Final Exam 1993)

Write an essay of not less than 240 words explaining some aspect of the important biological effects of radiation with respect to humans. The essay should be in good English and in simple language as if it were to be published in a public relations brochure.

Essay (10 marks)

PROBLEM 9.06 (Final Exam 1994)

Write an essay of not less than 300 words related to man and radiation. Consider natural and manmade radiation and radiation received by humans in everyday life. The essay should be in good English and in simple language as if it were to be published in a public relations brochure.

Essay (10 marks)

PROBLEM 9.07 (Final Exam 1995)

Write an essay of not less than 300 words on the benefits of nuclear radiation or nuclear power to mankind. The essay should be in good English and in simple language as if it were to be published in a public relations brochure.

Essay (10 marks)

PROBLEM 9.08 (Second Test 1992)

Describe the most important biological effects of large acute doses of radiation with respect to humans. Make a clear distinction between early effects and late effects. Where appropriate give values of radiation dose received and its observed effect.

Description (10 marks)

CHE NUCLEAR ENGINEERING

QUESTION BANK

ANSWERS TO PROBLEMS

SECTION 1

Problem 2.01	See Chart of Nuclides
Problem 2.02	$N = 0.502 \times 10^{24}$ atoms
Problem 2.03	
Problem 2.04	
Problem 2.05	
Problem 2.06	$\rho = 1.96 \times 10^{14}$ g/cm^3
Problem 2.07	
Problem 2.08	
Problem 2.11	m = 3280 short tons
Problem 2.12	
Problem 2.13	$\Delta m = 1.293$ MeV
Problem 2.21	$N = 0.0802 \times 10^{24}$ atmos/cm^3
Problem 2.22	
Problem 2.23	
Problem 2.24	
Problem 2.25	
Problem 2.31	Explanation
Problem 2.32	Sketch and Explanation

SECTION 2

Problem 2.41	
Problem 2.42	(a) He-3
	(b) m = 0.103 μg
Problem 2.43	$\alpha_{uranium} = 12.9 \times 10^3$ disintegrations/s, $\alpha_{uranium} = 0.349$ μ curie
Problem 2.44	$\alpha = 55$ m curie
Problem 2.45	
Problem 2.46	$N_0 = 381 \times 10^3$ atoms, t = 142 s
Problem 2.51	$m_S = 8.21$ g
Problem 2.54	$N = 7.88 \times 10^{15}$ nuclei/cm^3
Problem 2.55	t = 1.77 billion years
Problem 2.61	
Problem 2.62	
Problem 2.63	
Problem 2.64	m = 2.56 g
Problem 2.65	See Chart of Nuclides

Problem 2.71	
Problem 2.72	α = 126 Curies
	t = 8.36 years
Problem 2.73	α = 1260 Curie
Problem 2.74	α = 80.7×10^6 Curie
Problem 2.81	Description and Sketch
Problem 2.83	
Problem 2.84	E = 180 MeV
Problem 2.85	E = 210 MeV
Problem 2.86	
Problem 2.87	(a) BE = 7.587 MeV/nucleon
	(b) BE = 6.541 MeV
Problem 2.88	(a) BE = 7.07 MeV/nucleon
	(b) E = 4.03 MeV
Problem 2.91	P = 855 MW (electrical)
Problem 2.92	m = 276 tonnes
Problem 2.93	
Problem 2.94	m = 7.06 kg
Problem 2.95	(a) P = 253.4 W
	(b) = 13.7 W
Problem 2.96	m = 5.07 kg
Problem 2.97	(a) P = 2842 MW
	(b) Fuel consumption = 3 kg/day

SECTION 3

Problem 3.01	Description
Problem 3.02	Equations
Problem 3.03	Equations
Problem 3.04	Equations
Problem 3.11	(a) Σ = 0.205 cm^{-1}
	(b) x = 11.2 cm
	(c) P = 0.765
Problem 3.12	$\Sigma_{ss\text{-}304}$ = 0.24 cm^{-1}
Problem 3.13	Σ_a = 0.545 cm^{-1}
Problem 3.21	λ = 2.6 cm
Problem 3.22	N = 36 collisions
Problem 3.25	N = 116 collisions
Problem 3.27	N = 88 collisions
Problem 3.31	(a) $\Sigma_t = 0.216\ cm^{-1}$
	(b) $R = 8.64 \times 10^8\ s^{-1}$
	(c) $F = 8.64 \times 10^9\ cm^{-3}\ s^{-1}$
Problem 3.32	
Problem 3.33	R = 0.333×10^{12} neutrons / cm^3 s

Problem 3.34	$R = 0.600 \times 10^{9}$ neutrons/cm^3 s
Problem 3.42	Consumption = 2.9 kg/day
Problem 3.51	
Problem 3.53	
Problem 3.54	(a) $R = 0.2466 \times 10^{13}$ reactions/s
	(b) $\alpha = 0.0001324 \times 10^{16}$ disintegrations/s
	(c) $\alpha = 35.8$ curie
Problem 3.55	$\alpha = 10$ fissions/s

SECTION 4

Problem 4.01	
Problem 4.02	
Problem 4.03	
Problem 4.04	
Problem 4.05	
Problem 4.11	
Problem 4.12	Explanation
Problem 4.13	Explanation
Problem 4.21	

SECTION 5

Problem 5.01	(a) $I = 2.77 \times 10^{14}$ neutrons / cm^2 s
	(b) $\phi = 5.53 \times 10^{14}$ neutrons / cm^{2}s
Problem 5.02	
Problem 5.03	$n = 19.6 \times 10^{6}$ neutrons / cm^3
	$N = 8.03 \times 10^{22}$ atoms / cm^3
Problem 5.04	(a) $R = 2.36 \times 10^{12}$ neutrons/s
	(b) $\alpha = 34.3$ Curies
Problem 5.11	(a) $L_T = 1.37$ cm
	(b) $L_T = 2.47$ cm
	(c) $L_T = 2.80$ cm
Problem 5.12	(a) $D \approx 0.10$ cm
	(b) $L = 2.13$ cm
	(c) $r = 5.22$ cm
Problem 5.13	
Problem 5.14	(a) $D = 0.926$ cm
	(b) $L = 58.26$ cm
	(c) $r = 143$ cm
Problem 5.21	(a) $\phi_{max} = 3.14 \times 10^{12}$ neutrons/cm^{2}s
	(b) Leakage = 1.58×10^{15} neutrons/s
Problem 5.22	$D = 0.833$ cm
Problem 5.23	$\phi = 458$ neutrons/cm^{2}s

Problem 5.24	
Problem 5.25	ϕ = 44 290 neutrons/cm^{2}s
Problem 5.26	(b) R = 0.00111 x 10^{14} reactions/cm^{3}s
	(c) R = 0.03326 x 10^{12} reactions/cm^{3}s
	(d) R = 17.6 x 10^{15} reactions/s
Problem 5.31	Sketch and Explanation
Problem 5.32	Sketch and Explanation
Problem 5.33	ϕ_{max} = 4.17 x 10^{15} neutrons / cm^2 s
	ϕ_{ave} = 1.27 x 10^{15} neutrons / cm^2 s

SECTION 6

Problem 6.01	f = 0.887
	k_∞ = 2.315
Problem 6.02	
Problem 6.03	(a) N = 2.56 x 10^{-5} x 10^{24} atoms/cm^3
	(b) f = 0.435
	(c) L_T^2 = 4.58 cm^2
	L_T = 2.14 cm
	(d) k_∞ =0.898
Problem 6.04	
Problem 6.05	k_∞ = 1.237 Not safe for unspecified size.
Problem 6.06	
Problem 6.09	(a) a = 158 cm
	(b) Uranium/Water = 0.291
	(c) f = 0.828
	(d) η = 1.335
	(e) p = 0.883
	(f) ϵ = 1.05
	(g) k_∞ = 1.025
Problem 6.11	(a) P_{max} = 58 kW / litre
	(b) $(q / A)_{max}$ = 310 kW / m^2
Problem 6.12	P = 67.9 W/cm^3
Problem 6.13	
Problem 6.14	
Problem 6.15	P_{ave} = 37.1 kw/litre, P_{max} = 55.6 kw/litre
Problem 6.16	P_p = 31 MW/m^3
Problem 6.23	(a) B^2 = 0.000051 cm^{-2}
	(b) ϕ_{max} = 2.19 x 10^{12} neutrons / cm^2 s
	(c) ϕ_{ave} = 0.56 x 10^{12} neutrons / cm^2 s
Problem 6.24	(b) ϕ = 2.19 x 10^{12} neutrons/cm^{2}s
	(c) ϕ_{ave} = 0.564 x 10^{12} neutrons/cm^{2}s
Problem 6.26	(a) Length of side = 539 cm
	(b) m = 8.54 x 10^4 kg

Problem 6.27	(a) a = 539 cm, V = 156 m^3
	(b) D = 622 cm, V = 126 m^3
Problem 6.27	(a) V = 156 m^3
	(b) V = 126 m^3
Problem 6.31	Explanation
Problem 6.32	Explanation
Problem 6.33	Explanation and Sketches

SECTION 7

Problem 7.03	t = 4 hrs 36 mins
Problem 7.04	
Problem 7.06	(a) t = 6254 s
	(b) t = 6256 s
Problem 7.07	t = 40 minutes
Problem 7.08	(a) t = 17 minutes
	(b) t = 3.1 minutes
Problem 7.09	(a) S_∞ = 0.04%
	(b) Δk = +12.5 mk
Problem 7.11	t = 3.5 minutes
Problem 7.12	ρ = - 28.7%
Problem 7.13	
Problem 7.14	
Problem 7.21	α = 80.7 x 10^6 Curie
Problem 7.22	Explanation and Equations
Problem 7.25	Explanation and Equations
Problem 7.26	Explanation and Equations
Problem 7.31	m = 94 tonnes UO_2
Problem 7.32	(a) $C_{U\text{-}235}$ = 5.49 g/kg U or 4.78 g/kgUO_2
	(b) $C_{Pu\text{-}239}$ = 0.88 g/kgU or 0.78 g/kg UO_2
	(c) P = 1858 MW
Problem 7.32	(a) $C_{t\,U\text{-}235}$ = 5.49 g/kg U
	= 4.84 g/kg UO_2
	(b) $C_{t\,Pu\text{-}239}$ = 0.88 g/kg U
	= 0.78 g/kg UO_2
	(c) P = 1858 MW
Problem 7.33	

SECTION 8

Problem 8.01
Problem 8.02
Problem 8.03
Problem 8.11

Problem 8.14
Problem 8.15
Problem 8.16
Problem 8.17
Problem 8.18
Problem 8.21
Problem 8.22

SECTION 9

Problem 9.01
Problem 9.02
Problem 9.03
Problem 9.04
Problem 9.05
Problem 9.06
Problem 9.07
Problem 9.08