

University of New Brunswick
Department of Chemical Engineering

UN 0702 POWER PLANT THERMODYNAMICS

LECTURE NOTES

MODULE 3

NUCLEAR HEAT REMOVAL

2007

PREPARED BY

PROFESSOR R. A. CHAPLIN

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UNIVERSITY OF NEW BRUNSWICK
DEPARTMENT OF CHEMICAL ENGINEERING

THERMODYNAMICS

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POWER PLANT THERMODYNAMICS

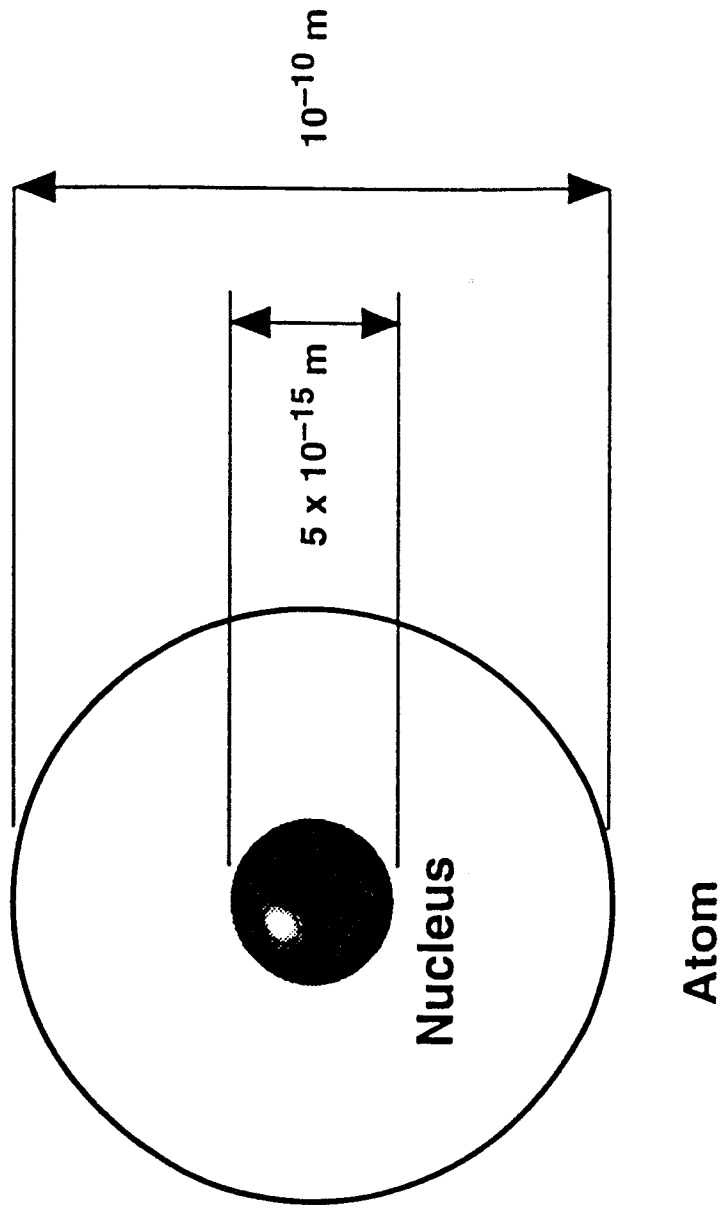
MODULE 3

NUCLEAR HEAT REMOVAL

- 1. NUCLEAR REACTOR PRINCIPLES**
- 2. REACTOR FLUX PROFILES**
- 3. HEAT CONDUCTION EQUATIONS**
- 4. FUEL HEAT CONDUCTION**
- 5. CLADDING CONDUCTION**
- 6. BOILING CHANNEL CONDITIONS**
- 7. STEAM GENERATOR LEVEL CONTROL**
- 8. STEAM GENERATOR PRESSURE CONTROL**
- 9. CONDENSER CHARACTERISTICS**

NUCLEAR REACTOR PRINCIPLES

Atom and Nucleus



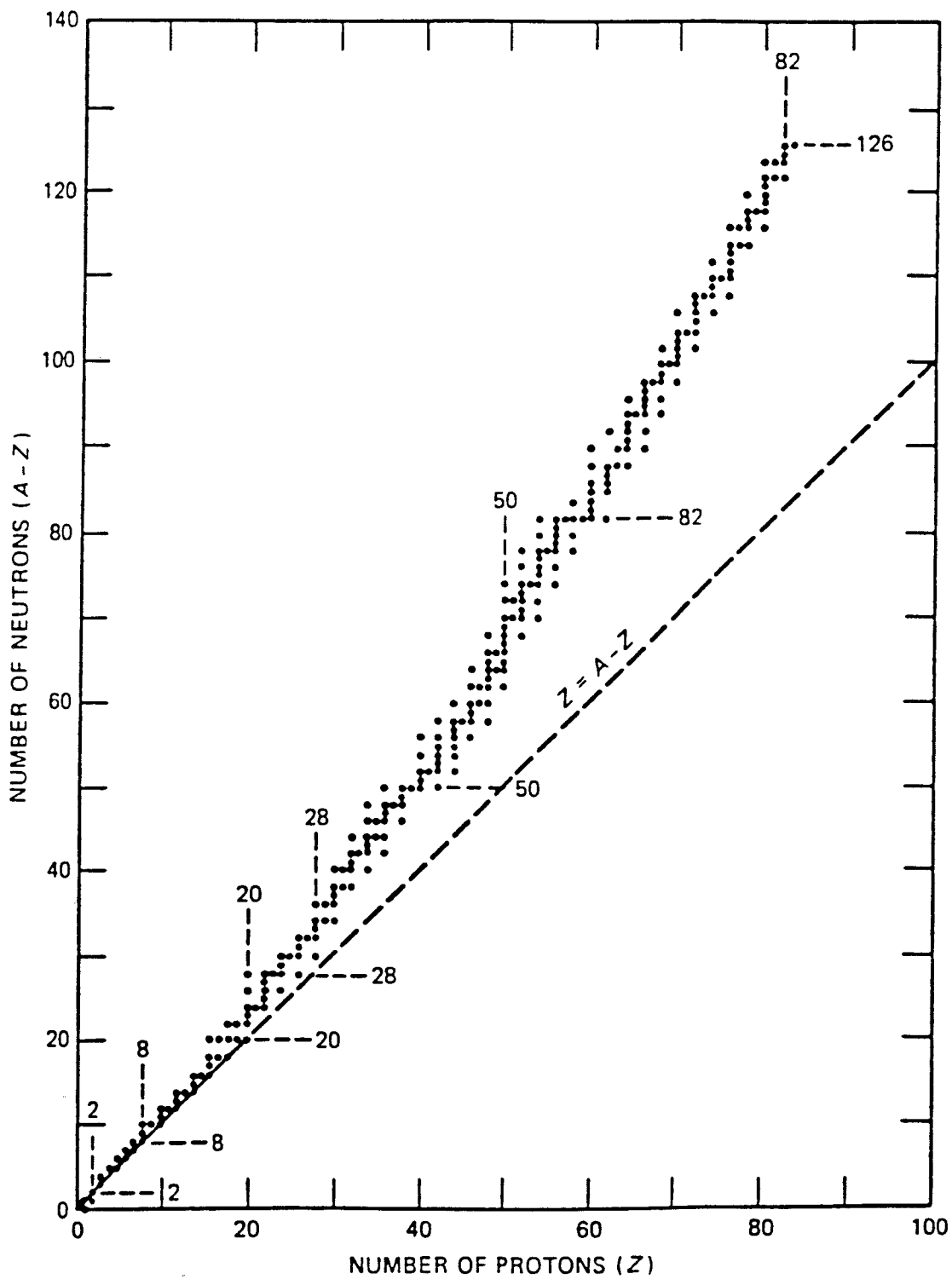
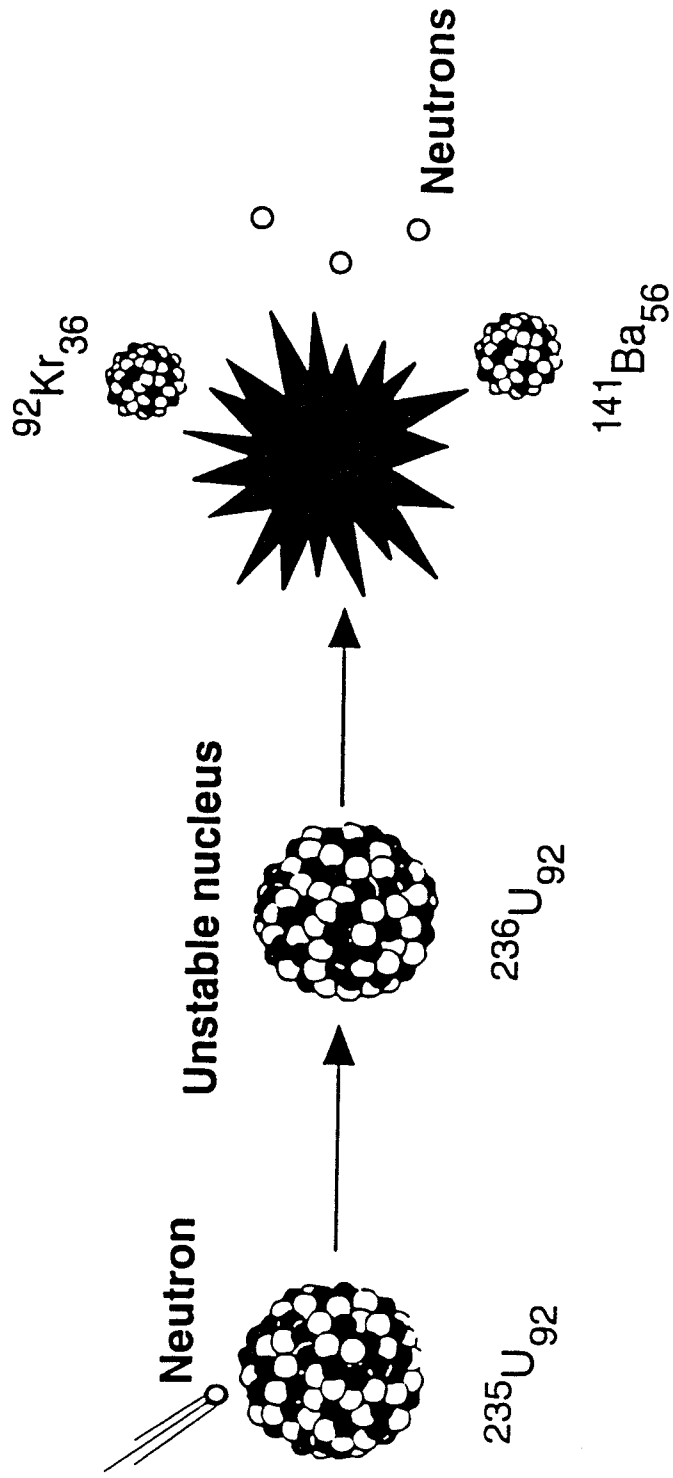


Fig. 1.2. Numbers of neutrons and protons in stable nuclei. (The short dashed lines indicate magic numbers of neutrons and protons.)

Fission Process



NUCLEAR REACTOR ENGINEERING

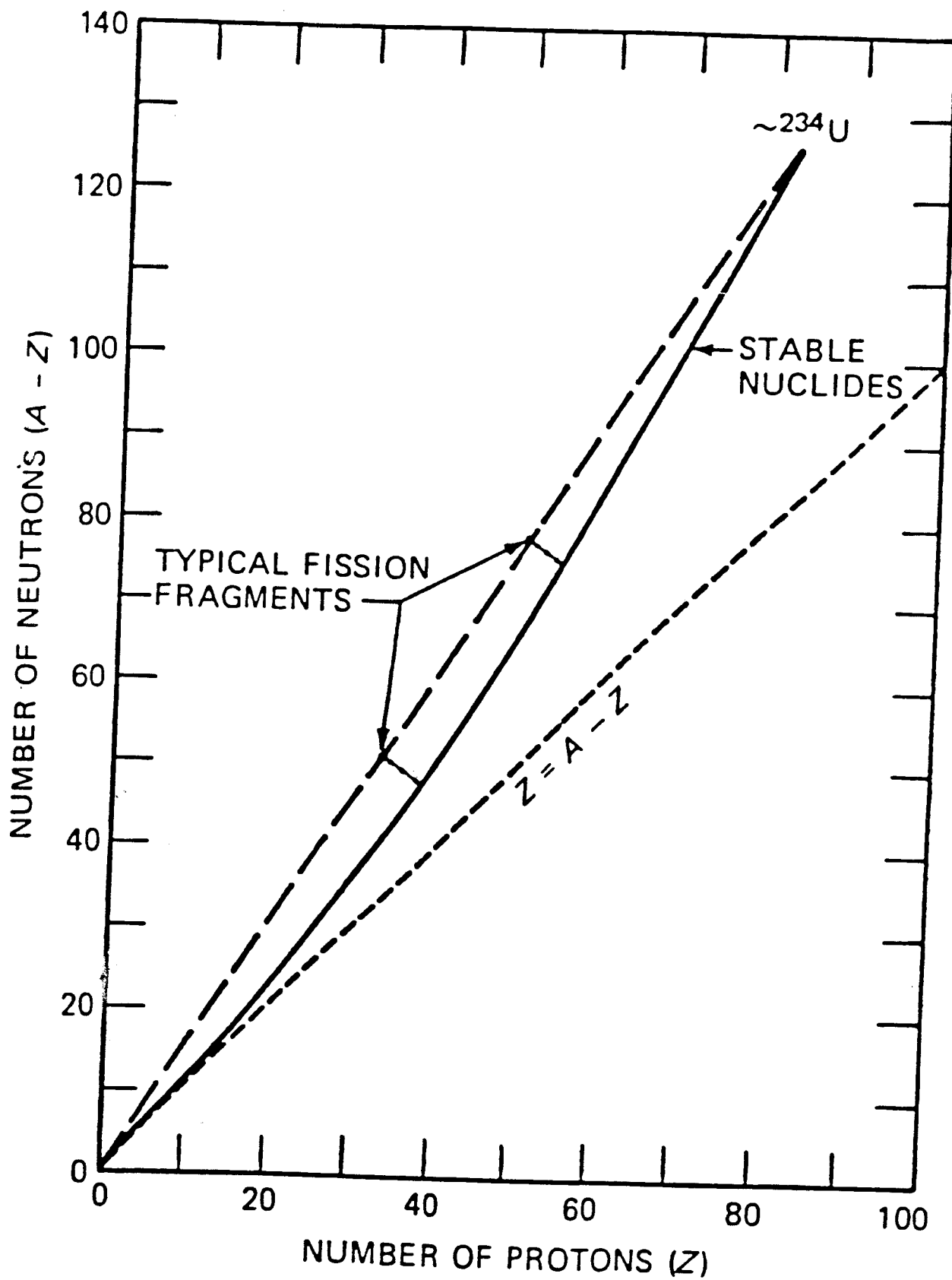
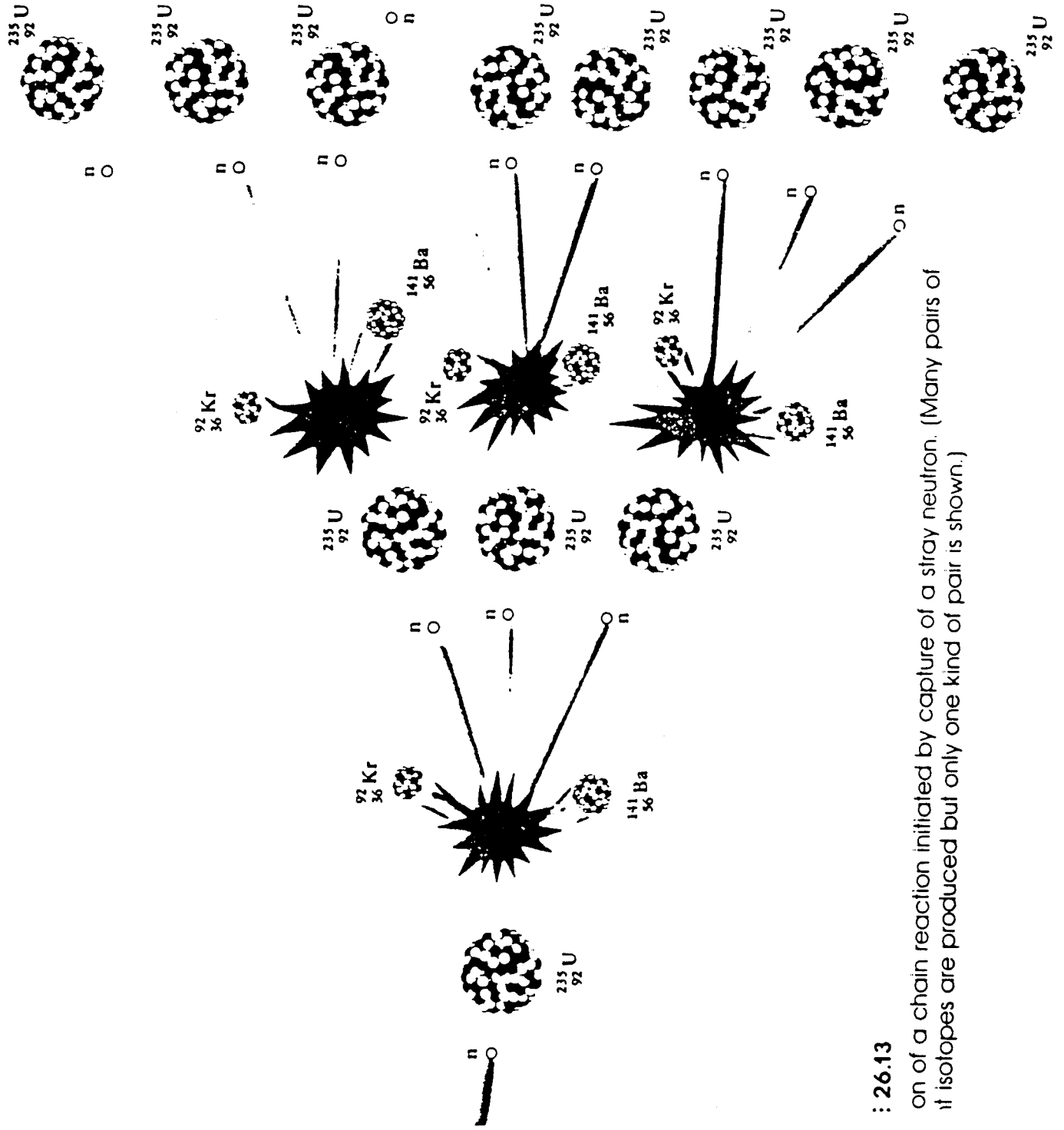


Fig. 2.29. Instability of fission fragments.



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on of a chain reaction initiated by capture of a stray neutron. (Many pairs of isotopes are produced but only one kind of pair is shown.)

Radiative Capture

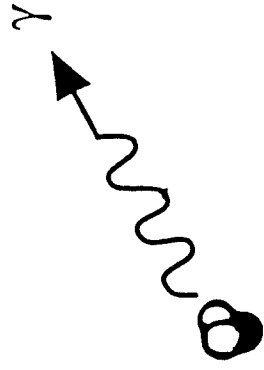


Fig 2.4

Elastic Collision

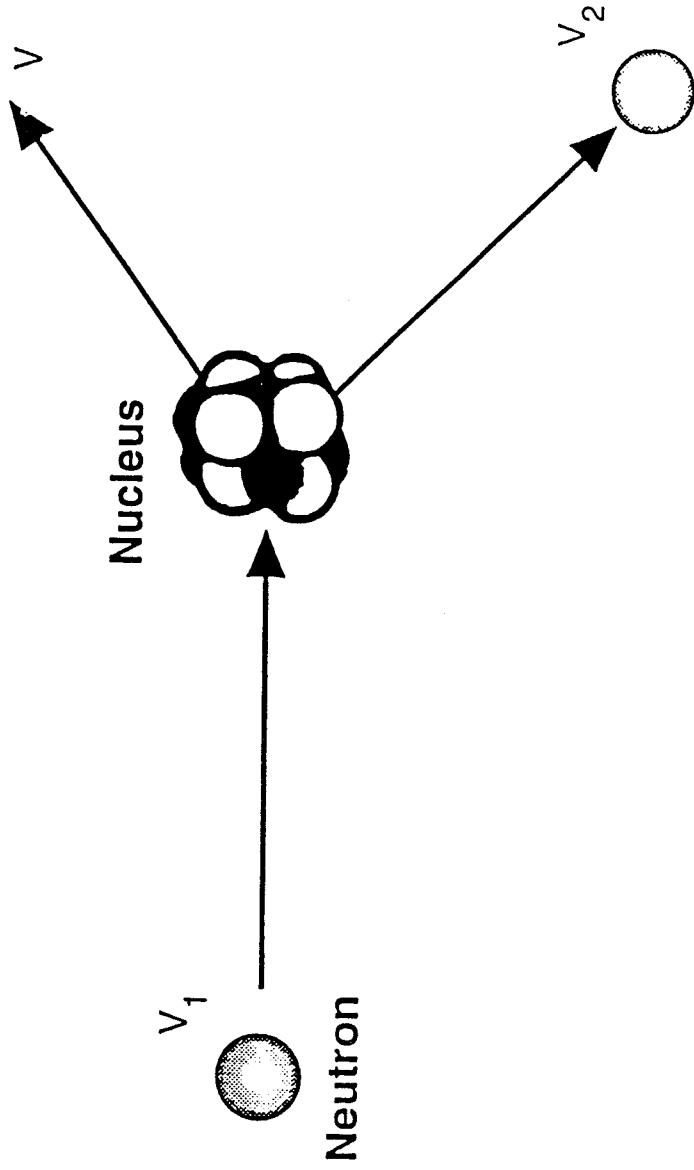
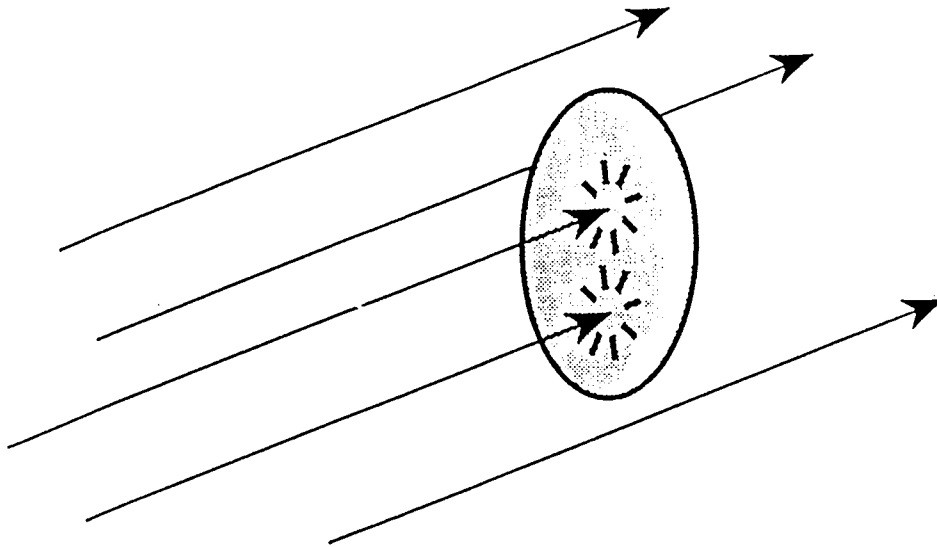
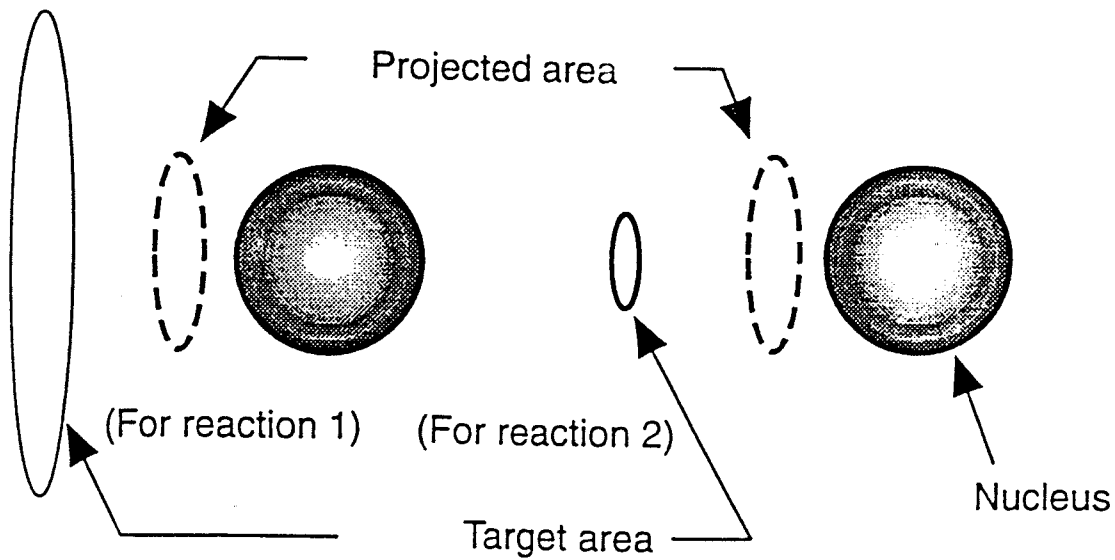


Fig. 2.1

Target Areas



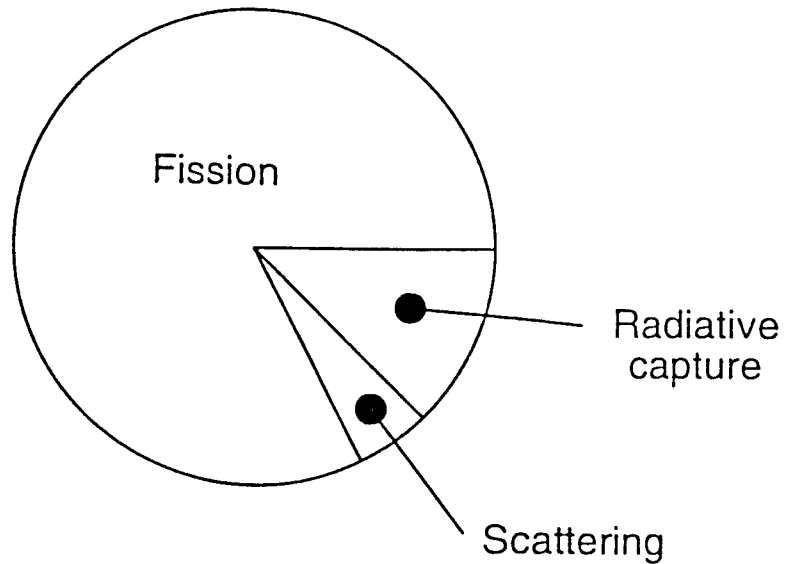
Target area may be smaller or larger than projected (physical) area of nucleus



Target areas are different for different nuclear reactions

Pie Diagram for Cross-Sections of U-235 and Nat-U

U-235



Nat-U

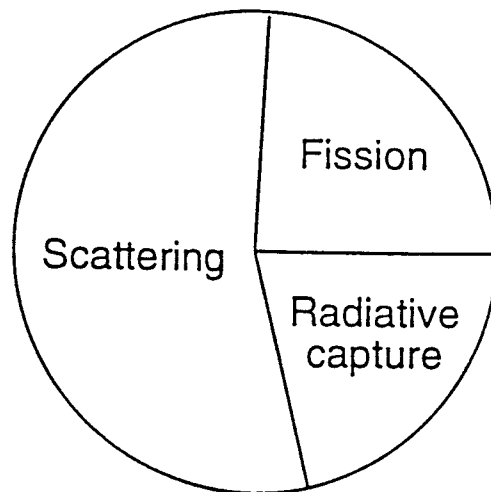
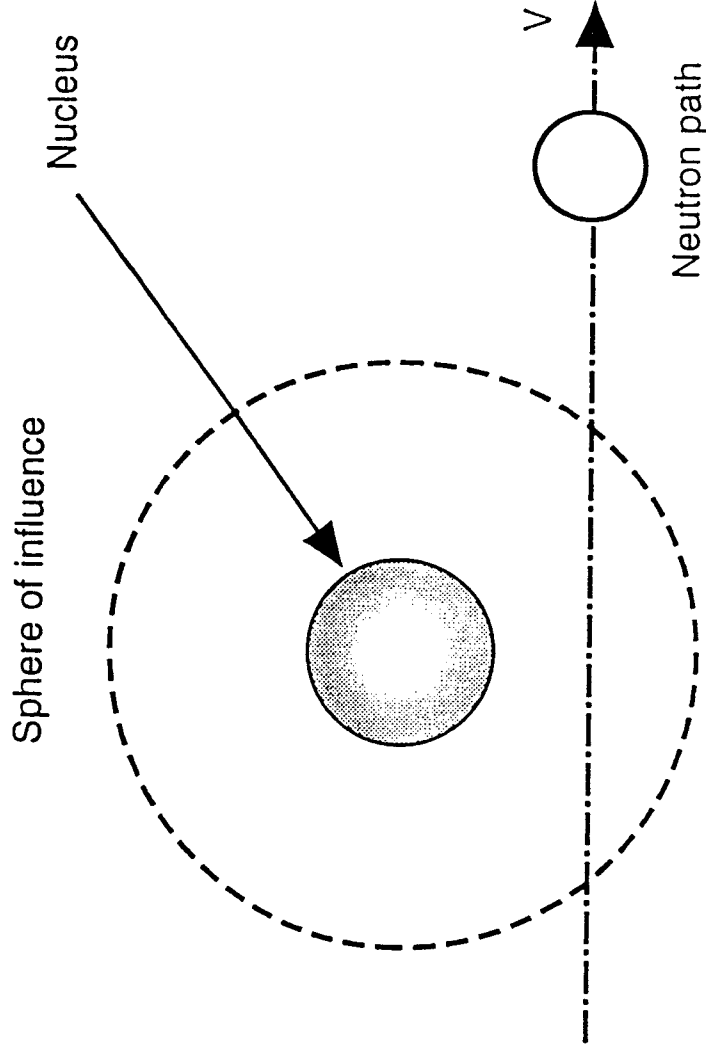


Fig. 3.2

Radiative Capture



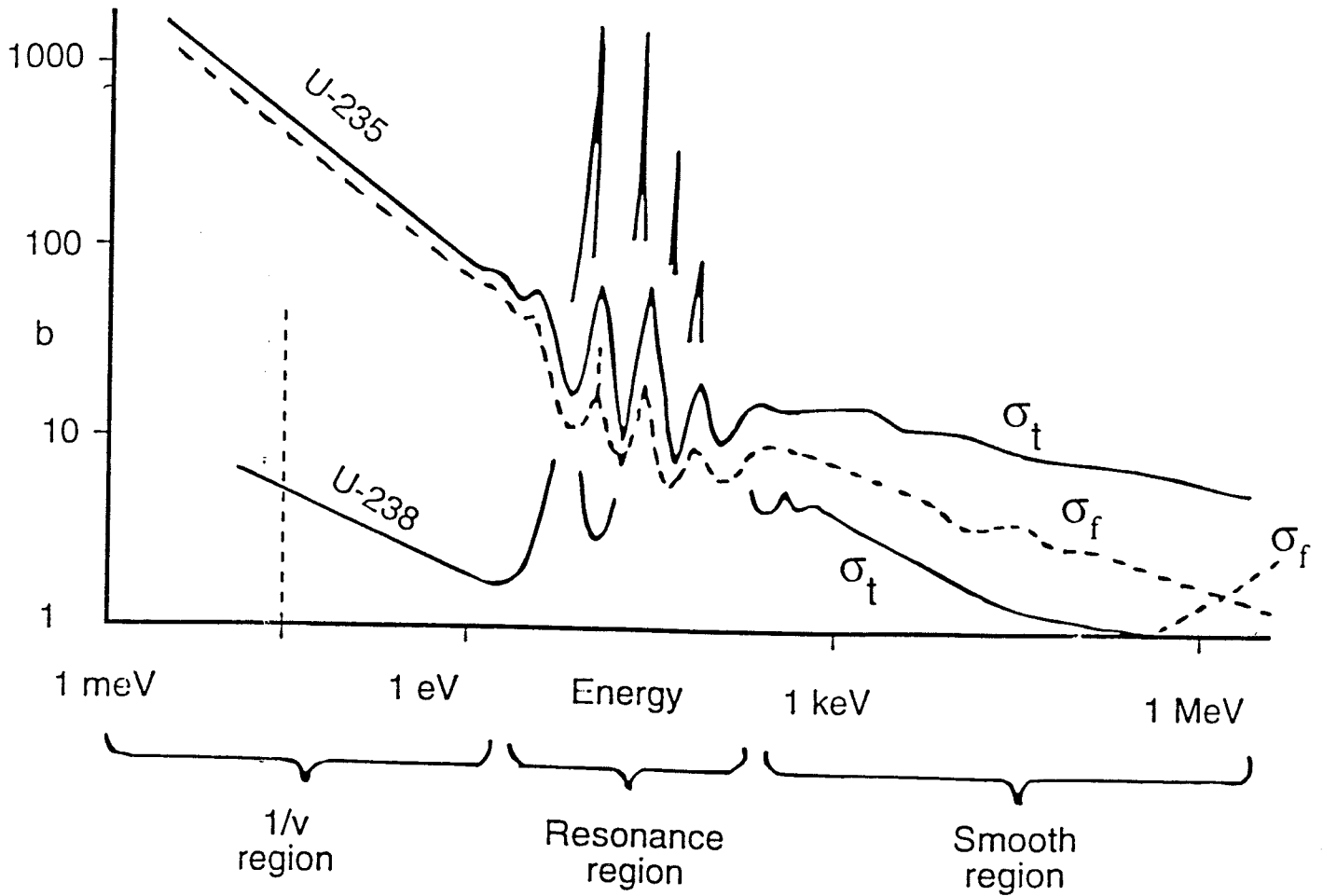
Time spent in sphere of influence

$$t \propto \frac{1}{v}$$

If probability of capture \propto time

$$\sigma_a \propto \frac{1}{v}$$

Fission Characteristics



Interactions of importance

$$\left. \begin{array}{l} \sigma_s = \text{Scattering} \\ \sigma_\gamma = \text{Radiative capture} \\ \sigma_f = \text{Fission} \end{array} \right\} \sigma_a = \text{Absorption}$$

$$\text{Capture/fission ratio: } \alpha = \sigma_\gamma / \sigma_f$$

$$\text{Probability of fission: } p = \sigma_f / \sigma_a$$

Elastic Collision

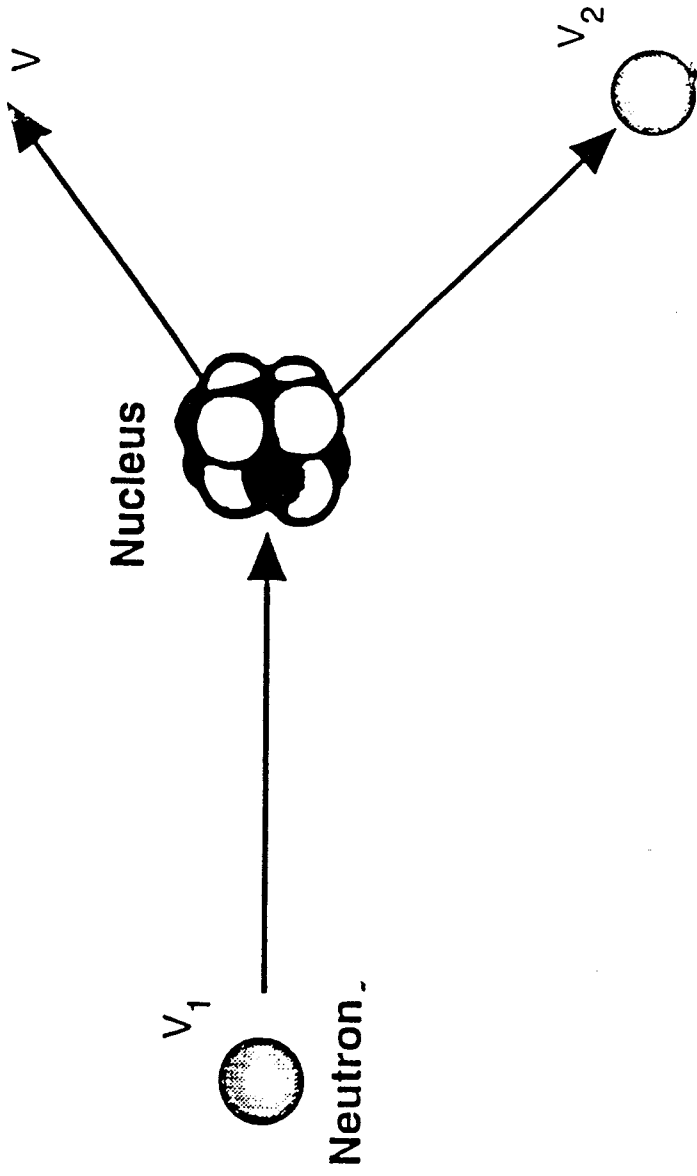


Fig. 2.1

Average Number of Elastic Collisions to Thermalize Fission Neutrons in Various Materials

H	18
D (Deuterium)	25
H ₂ O (Light water)	20
D ₂ O (Heavy water)	36
C-12 (Graphite)	115
U-238	2172

Mean Logarithmic Energy Decrements

Material	ξ	Collisions to thermalize
H ^{1*}	1.000	18
H ^{2*}	0.725	25
He ^{4*}	0.425	43
Be ⁹	0.206	83
C ¹²	0.158	115
H ₂ O	0.927	20
D ₂ O	0.510	36
BeO	0.174	105

Path of a Neutron from Birth to Absorption

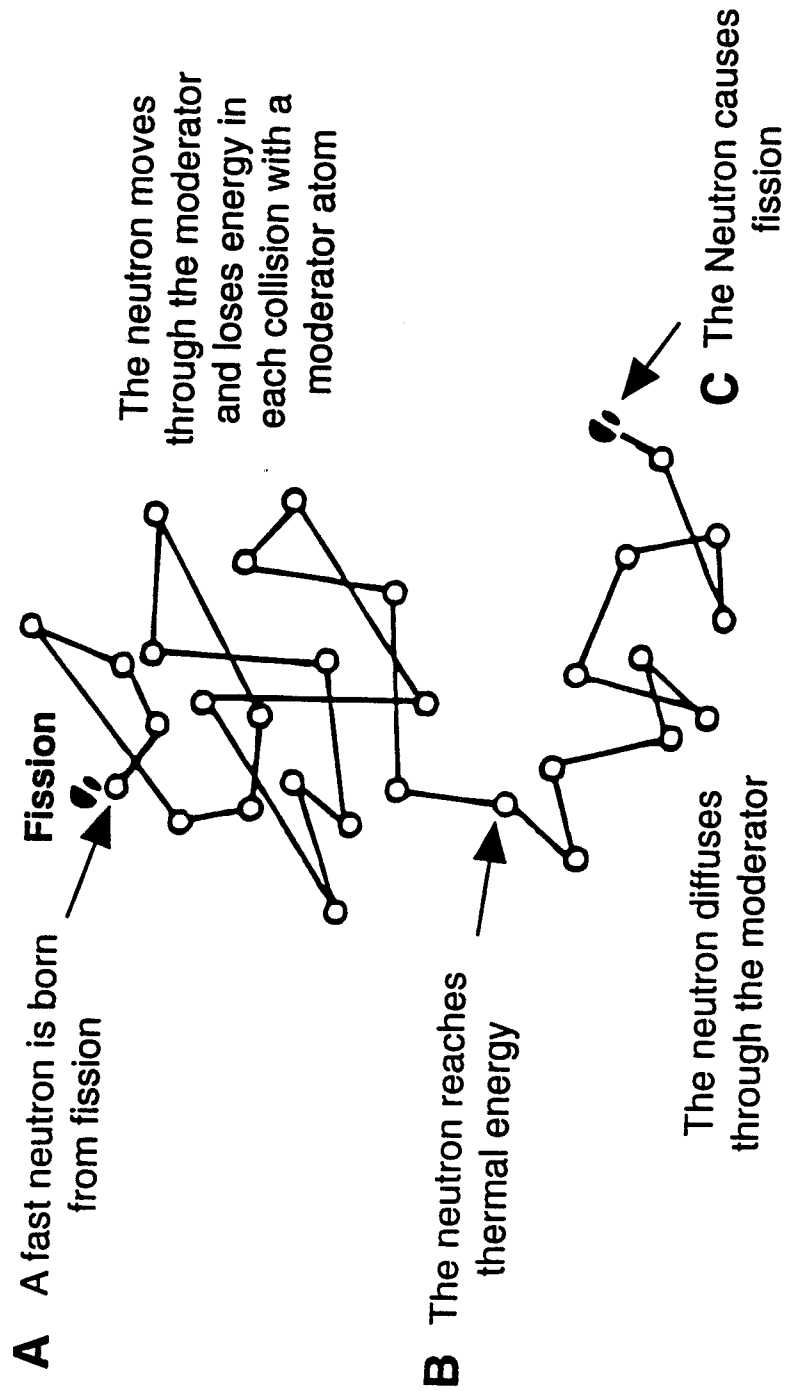
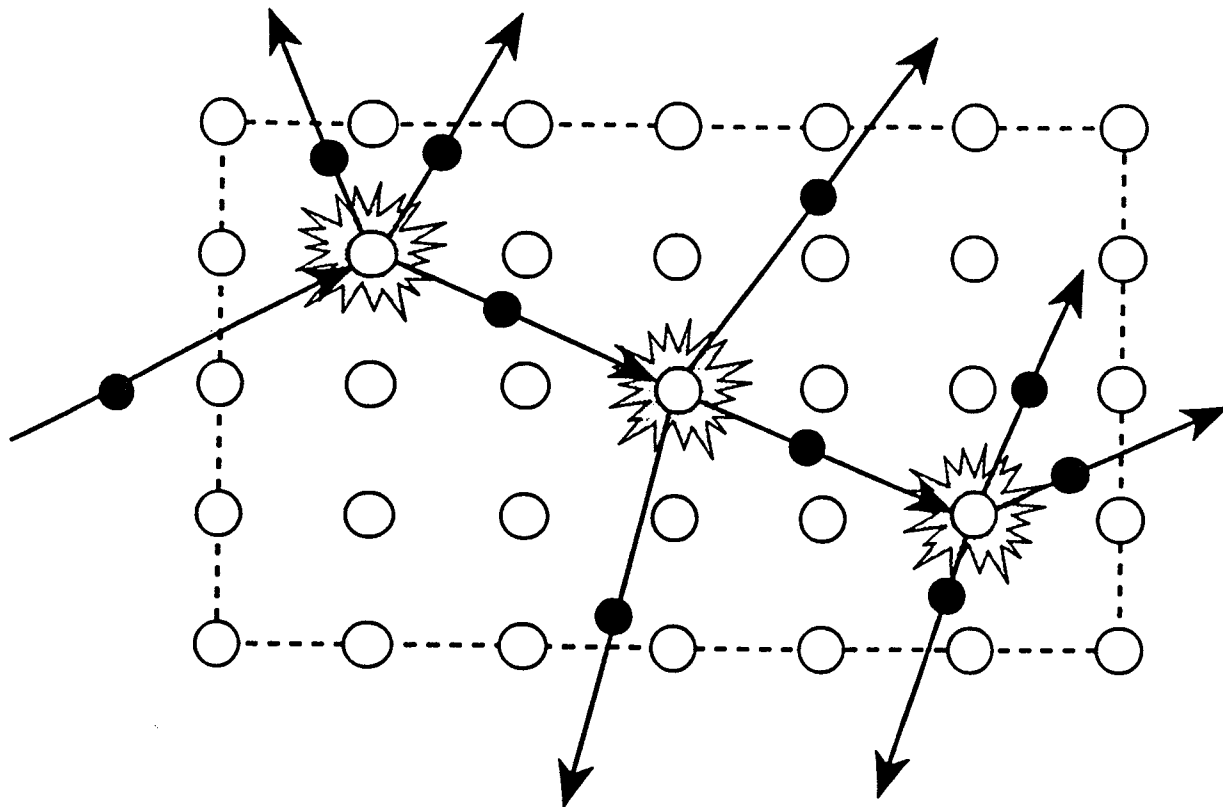
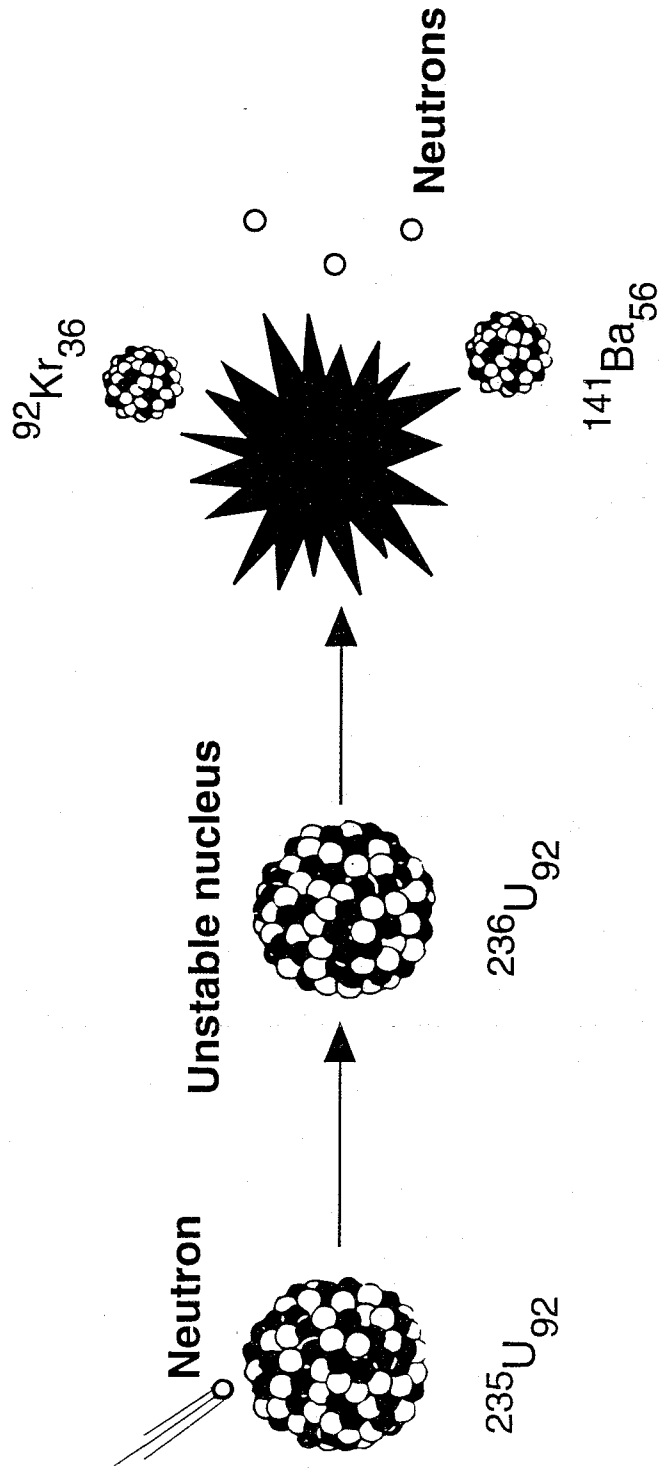


Fig. 4.3

Fission Chain Reaction



Fission Process



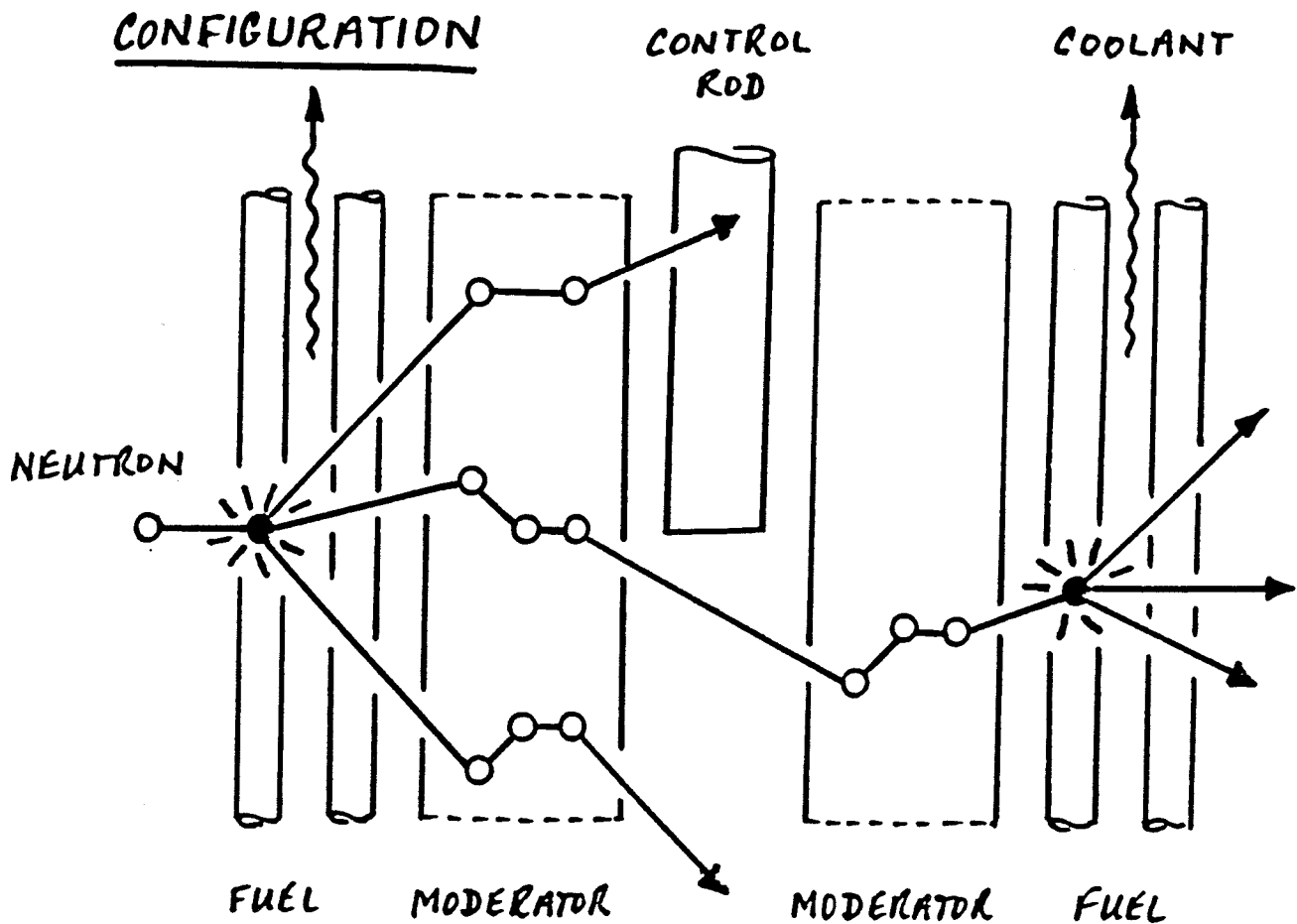
Approximate Distribution of Fission Energy Release in U-235

Kinetic energy of lighter fission fragment	100 MeV
Kinetic energy of heavier fission fragment	69 MeV
Kinetic energy of secondary neutrons	5 MeV
Energy of prompt γ rays	6 MeV
Beta particle energy gradually released from fission products	8 MeV
Gamma ray energy gradually released from fission products	6 MeV
Neutrinos (energy escapes from reactor)	11 MeV
Total	205 MeV

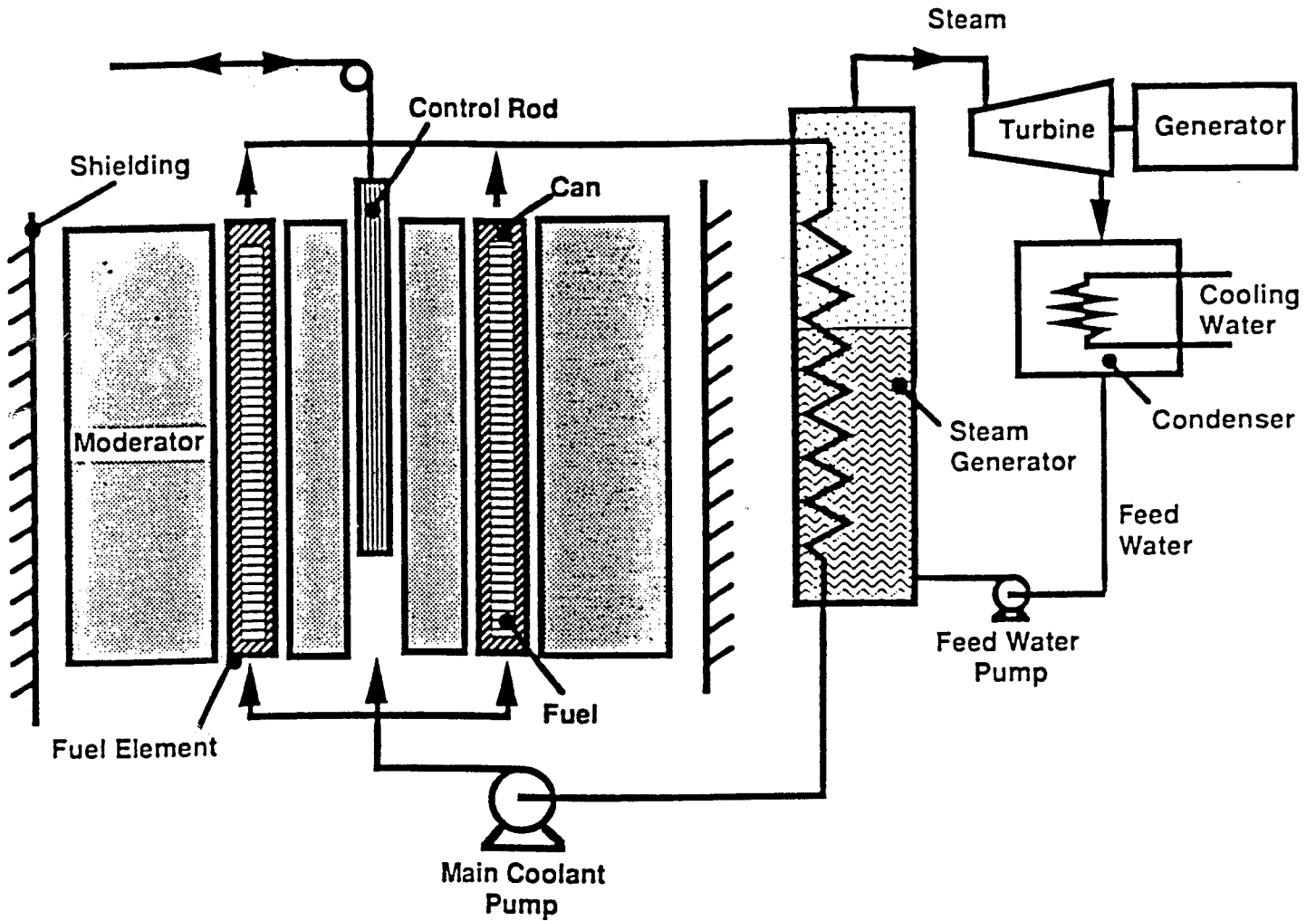
REACTOR PRINCIPLES

PRINCIPLES

- NEUTRON CHAIN REACTION
- CONTROL OF NEUTRONS
- MODERATION OF NEUTRONS
- REMOVAL OF ENERGY



BASIC REACTOR COMPONENTS



Avogadro's Number

$$N_A = 6.022 \times 10^{23}$$

Number of atoms or nuclei in a given sample

$$N = \frac{N_A}{A} \times \text{MASS (g)}$$

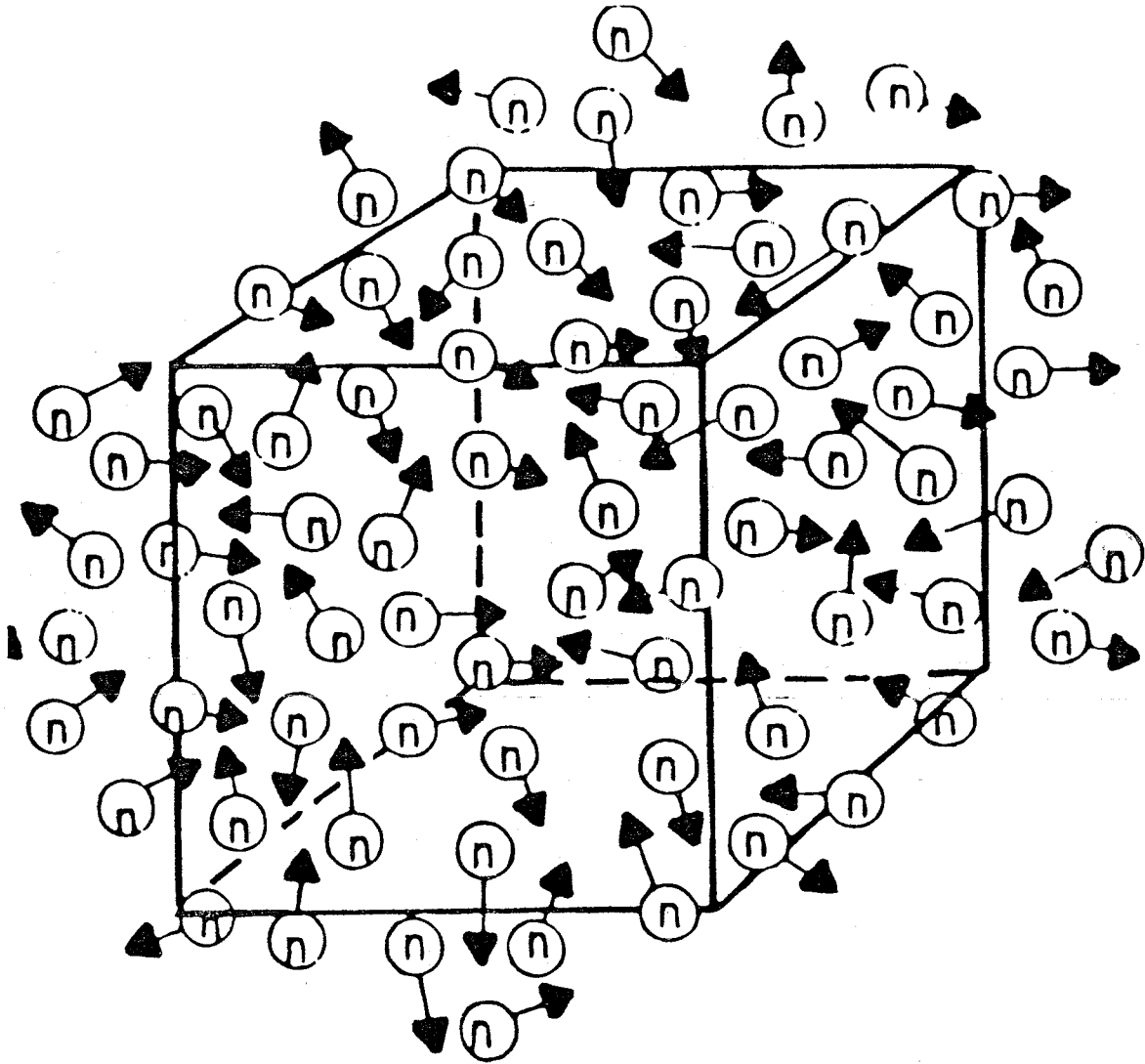
Example: Atoms in 1kg of U-235

$$\begin{aligned} N &= \frac{6.022 \times 10^{23}}{235} \times 1000 \\ &= 25.62 \times 10^{23} \text{ atoms} \end{aligned}$$

Example: 1kg of U-235 consumed in one day

$$\begin{aligned} N &= \frac{6.022 \times 10^{23}}{235} \times 1000 \text{ fission/day} \\ &= 25.62 \times 10^{23} \text{ atoms} / (24 \times 3600) \text{ Fissions/s} \\ &= 0.0002965 \times 10^{23} \times 200 \text{ MeV/s} \\ &= 0.05932 \times 10^{23} \times 1.6022 \times 10^{-13} \text{ J/s} \\ &= 0.09504 \times 10^{10} \text{ W} \\ &= 950 \times 10^6 \text{ W} \\ &= 950 \text{ MW} \end{aligned}$$

Neutron Reaction Rates



Definitions

Macroscopic cross-section

(Cross-section density in material)

$$\Sigma = N\sigma \quad \left(\frac{1}{\text{cm}} \right) \text{ or } \left(\text{cm}^{-1} \right)$$

$$N = \text{Nuclei per unit volume} \quad \left(\frac{\text{nuclei}}{\text{cm}^3} \right)$$

$$\sigma = \text{Microscopic cross-section} \quad \left(\text{cm}^2 \right)$$

Neutron flux

(Neutrons passing through given area per second)

$$\phi = nv \quad \left(\frac{\text{neutrons}}{\text{cm}^2 \text{ s}} \right)$$

$$n = \text{Neutrons per unit volume} \quad \left(\frac{\text{neutrons}}{\text{cm}^3} \right)$$

$$v = \text{Neutron velocity} \quad \left(\frac{\text{cm}}{\text{s}} \right)$$

Reaction rate

(Reaction rate of neutrons with material)

$$R = \phi\Sigma \quad \left(\frac{\text{reactions}}{\text{cm}^3 \text{ s}} \right)$$

$$\phi = \text{Neutron flux} \quad \left(\frac{\text{neutrons}}{\text{cm}^2 \text{ s}} \right)$$

$$\Sigma = \text{Macroscopic cross-section} \quad \left(\frac{1}{\text{cm}} \right)$$

Reactor Power

Neutron Flux

$$\phi = nv \text{ (neutrons/cm}^2\text{s)}$$

$$n = \text{Neutron density (neutrons/cm}^3\text{)}$$

$$v = \text{Neutron velocity (cm/s)}$$

Macroscopic cross-section

$$\Sigma_f = N\sigma_f \text{ (cm}^{-1}\text{)}$$

$$N = \text{Nuclei density (nuclei/cm}^3\text{)}$$

$$\sigma_f = \text{Microscopic cross-section (cm}^2\text{)}$$

Reactor power

$$P = \Sigma_f \phi V E_f \quad \frac{1}{\text{cm}} \times \frac{1}{\text{cm}^3\text{s}} \times \frac{\text{cm}^3}{1} \times \frac{\text{J}}{1}$$

$$V = \text{Volume of reactor (cm}^3\text{)}$$

$$E_f = \text{Energy release per fission (J)}$$