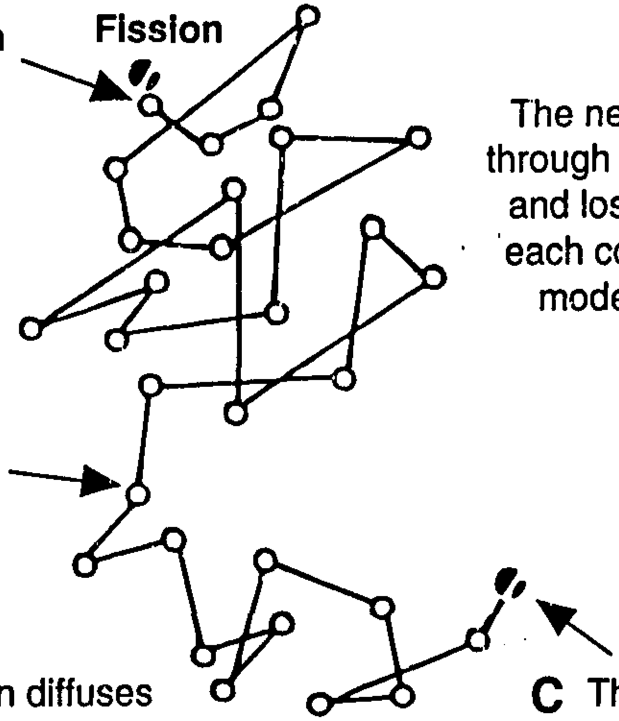
Path of a Neutron from Birth to Absorption



Fission

A A fast neutron is born from fission

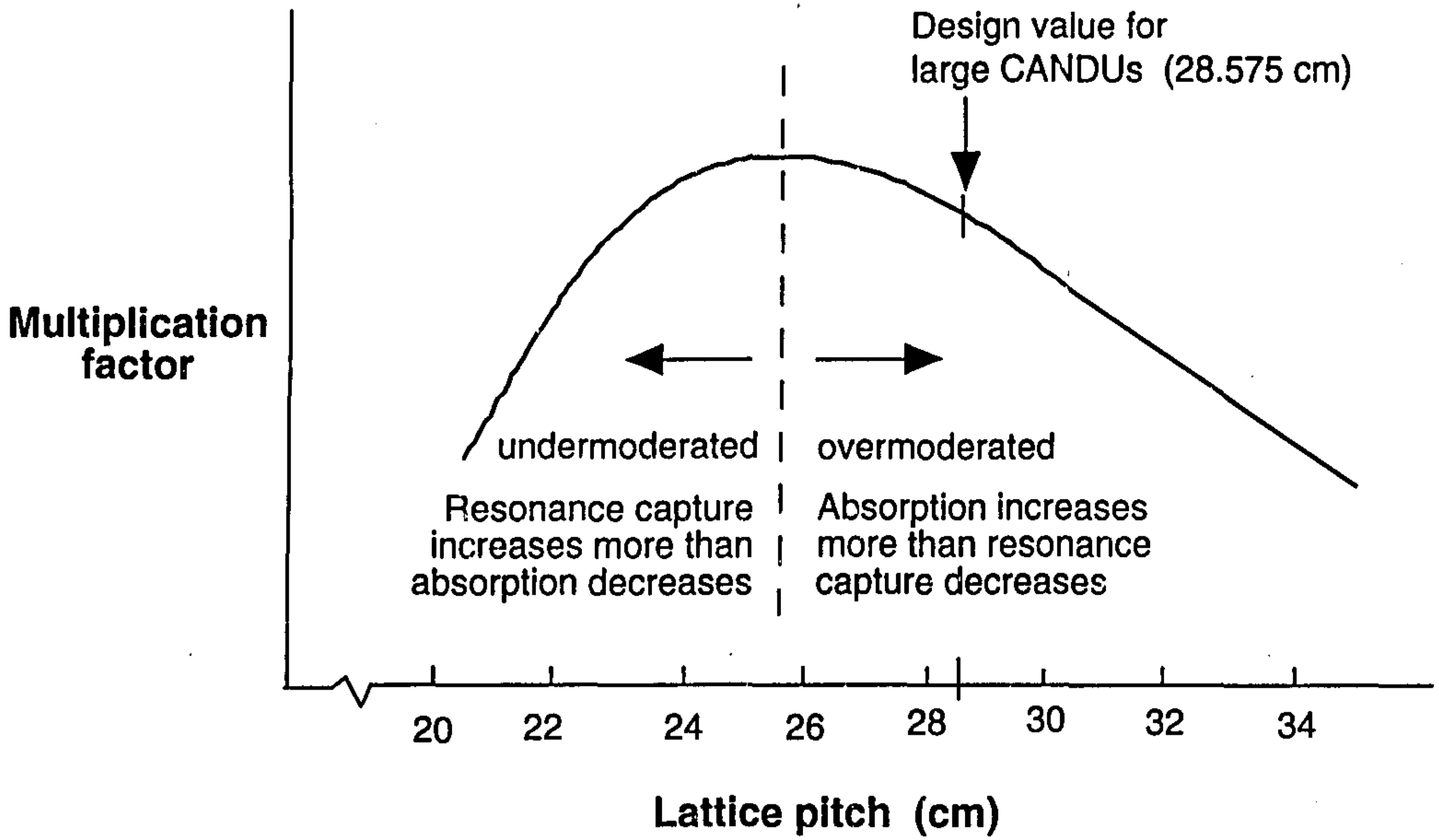
The neutron moves through the moderator and loses energy in each collision with a moderator atom

B The neutron reaches thermal energy

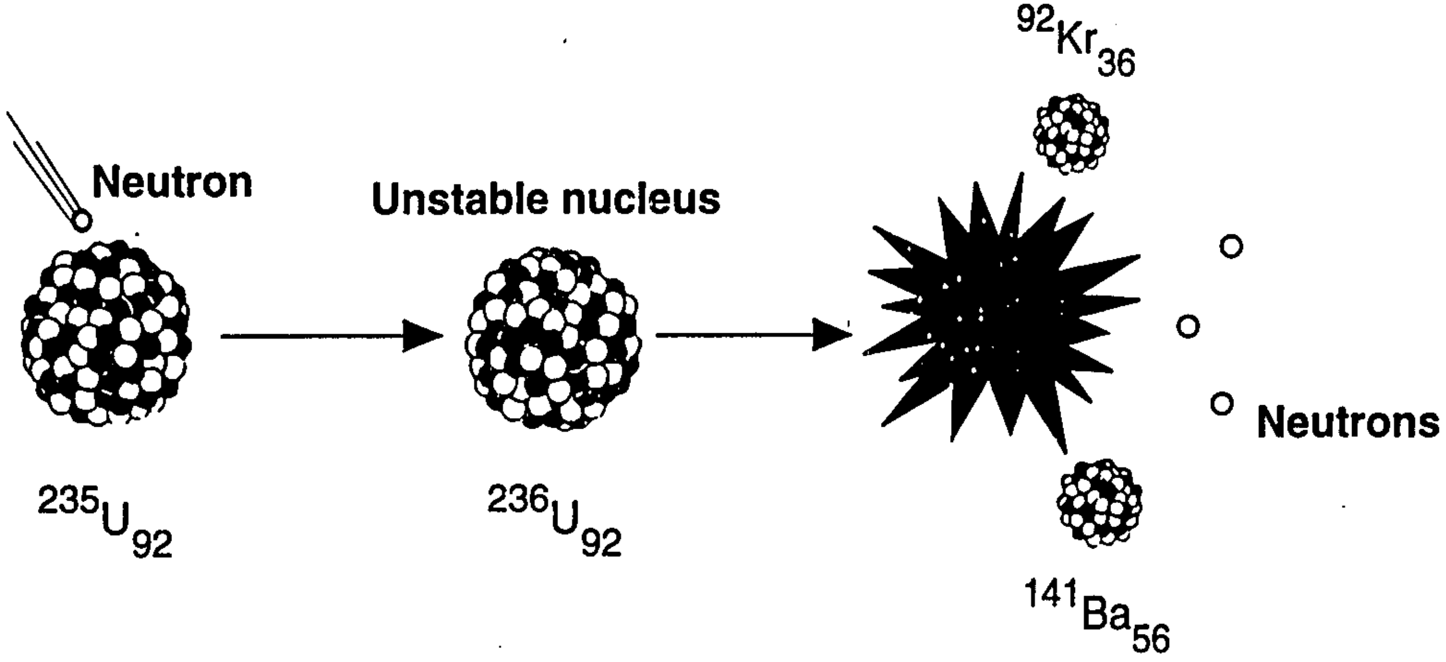
The neutron diffuses through the moderator

C The Neutron causes fission

Variation of k with Pitch of Lattice



Fission Process



$^{92}Kr_{36}$

Neutron

Unstable nucleus

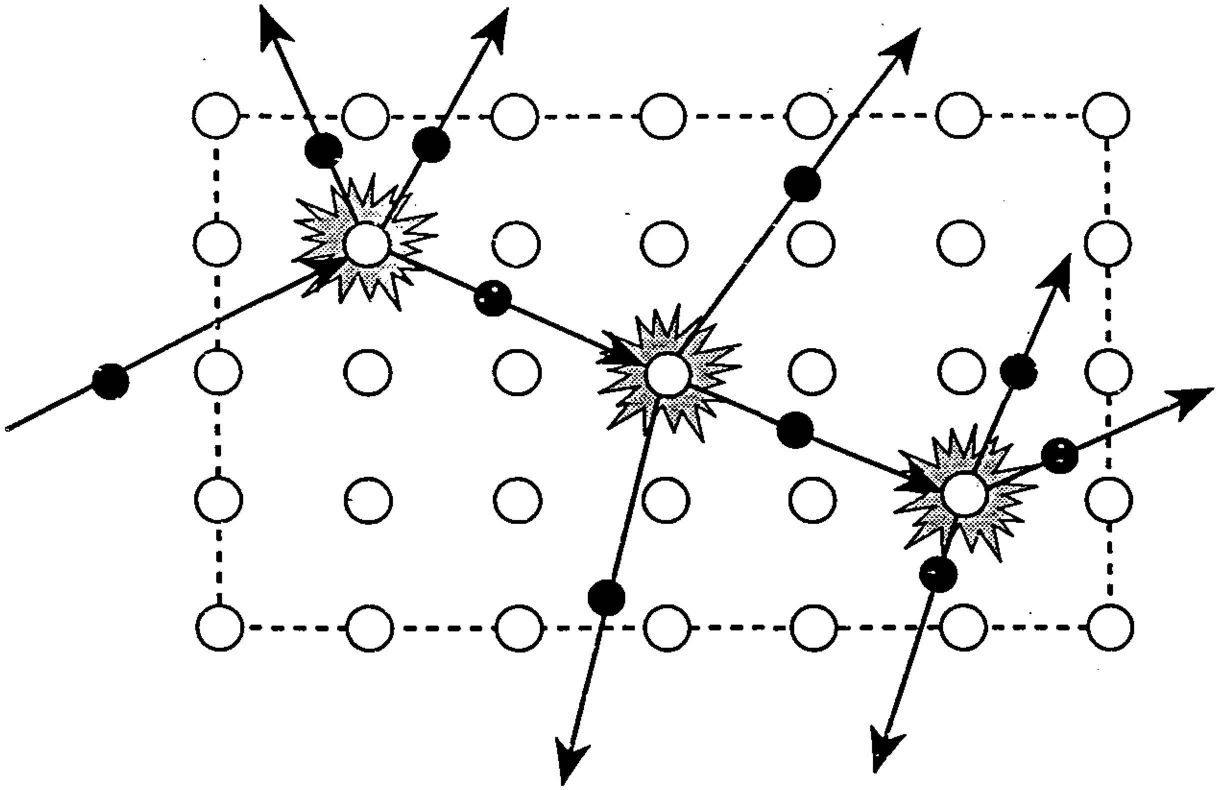
Neutrons

$^{235}U_{92}$

$^{236}U_{92}$

$^{141}Ba_{56}$

Fission Chain Reaction



Log Mean Energy Decrement

Logarithmic mean energy decrement

$$\xi = \overline{\mathrm{Ln}\,E_o - \mathrm{Ln}\,E}$$

$$= \overline{\mathrm{Ln}\,(E_o / E)}$$

$$= \overline{-\mathrm{Ln}\,(E / E_o)}$$

Value of ξ is given by

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

Approximate value of ξ are given by

$$\xi = \frac{2}{A + 2/3}$$

ξ is Greek letter Xi

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_2O	0.927	20
D_2O	0.510	36
BeO	0.174	105

Slowing Down Powers and Moderating Ratios

	ξ	$\Sigma_s(cm_{-1})$(a)	$\xi\Sigma_s$	Σ_a	$\xi\Sigma_{a/}\Sigma_a$
He(b)	0.425	2 x 10^{-6}	9 x 10^{-6}	? very small	? large
Be	0.206	0.74	0.15	1.17 x 10^{-3}	130
H(c)	0.158	0.38	0.06	0.38 x 10^{-3}	160
BeO	0.174	0.69	0.12	0.68 x 10^{-3}	180
H_2O	0.927	1.47	1.36	22 x 10^{-3}	60
D_2O	0.510	0.35	0.18	0.33 x 10^{-4}(d)	5500(d)
D_2O	0.510	0.35	0.18	0.88 x 10^{-4}(e)	2047(e)
D_2O	0.510	0.35	0.18	2.53 x 10^{-4}(f)	712(f)

Definitions

Logarithmic mean energy decrement ξ

$N\xi$ = $\ln \frac{E_i}{E_f}$

N = Number of Collisions

E_i = Initial energy (2 MeV)

E_f = Final Energy (0.025 eV)

Macroscopic scattering cross-section Σ_s

Σ_s = $N\sigma_s$

N = Nuclei per unit volume

σ = Microscopic cross-section

Slowing down power

= $\xi\Sigma_s$

Moderating ratio

= $\frac{\xi\Sigma_s}{\Sigma_a}$

Critical Assembly of 18 Bundles in D_2O

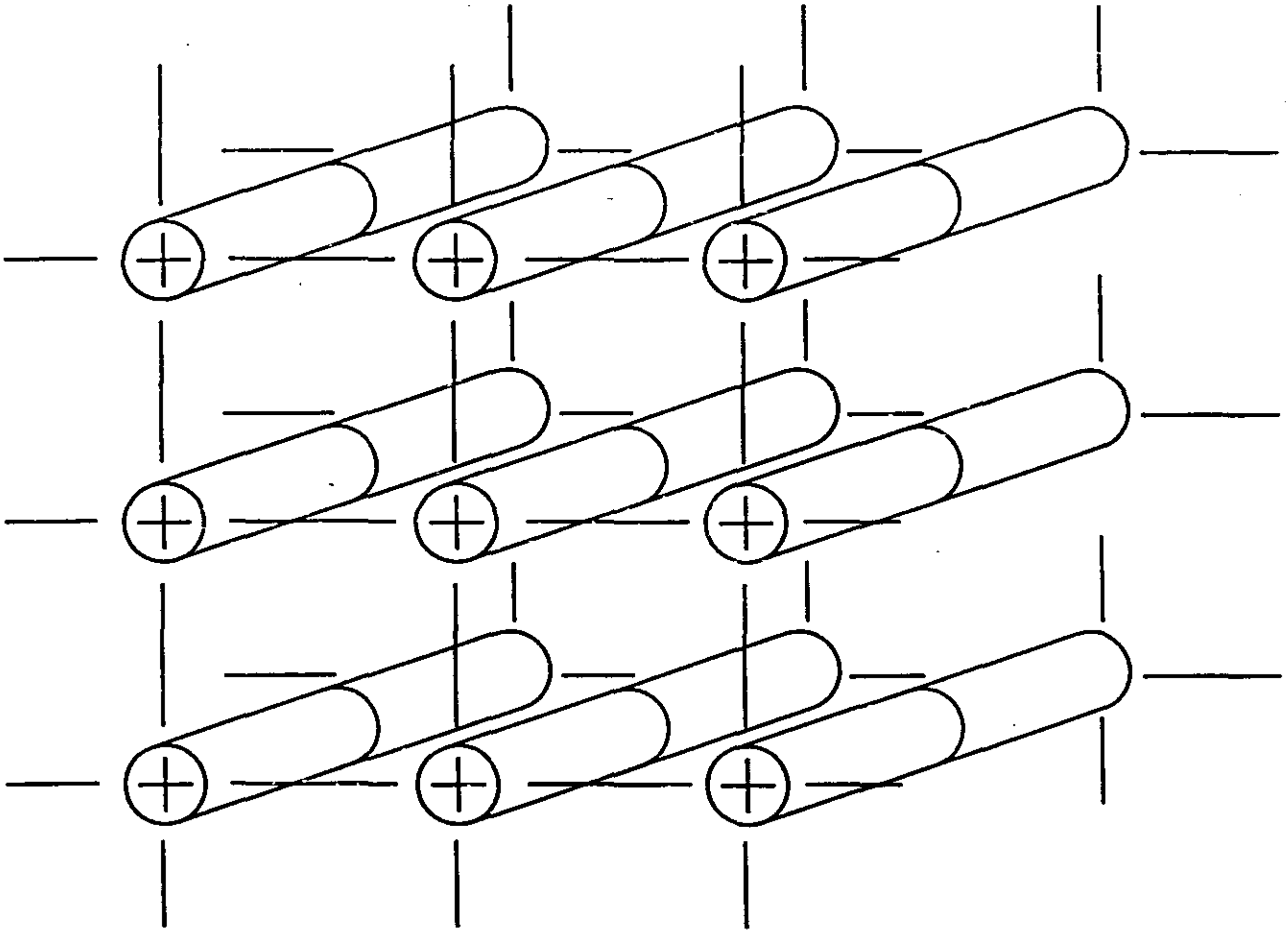
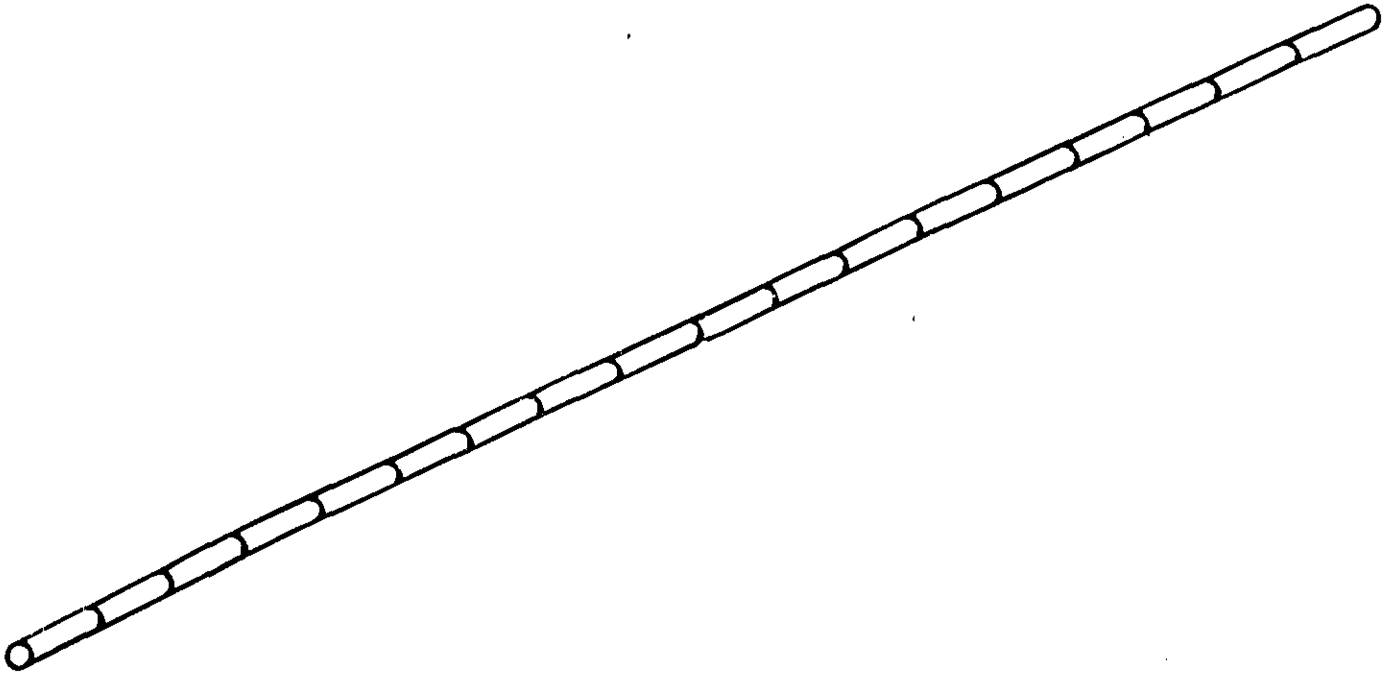
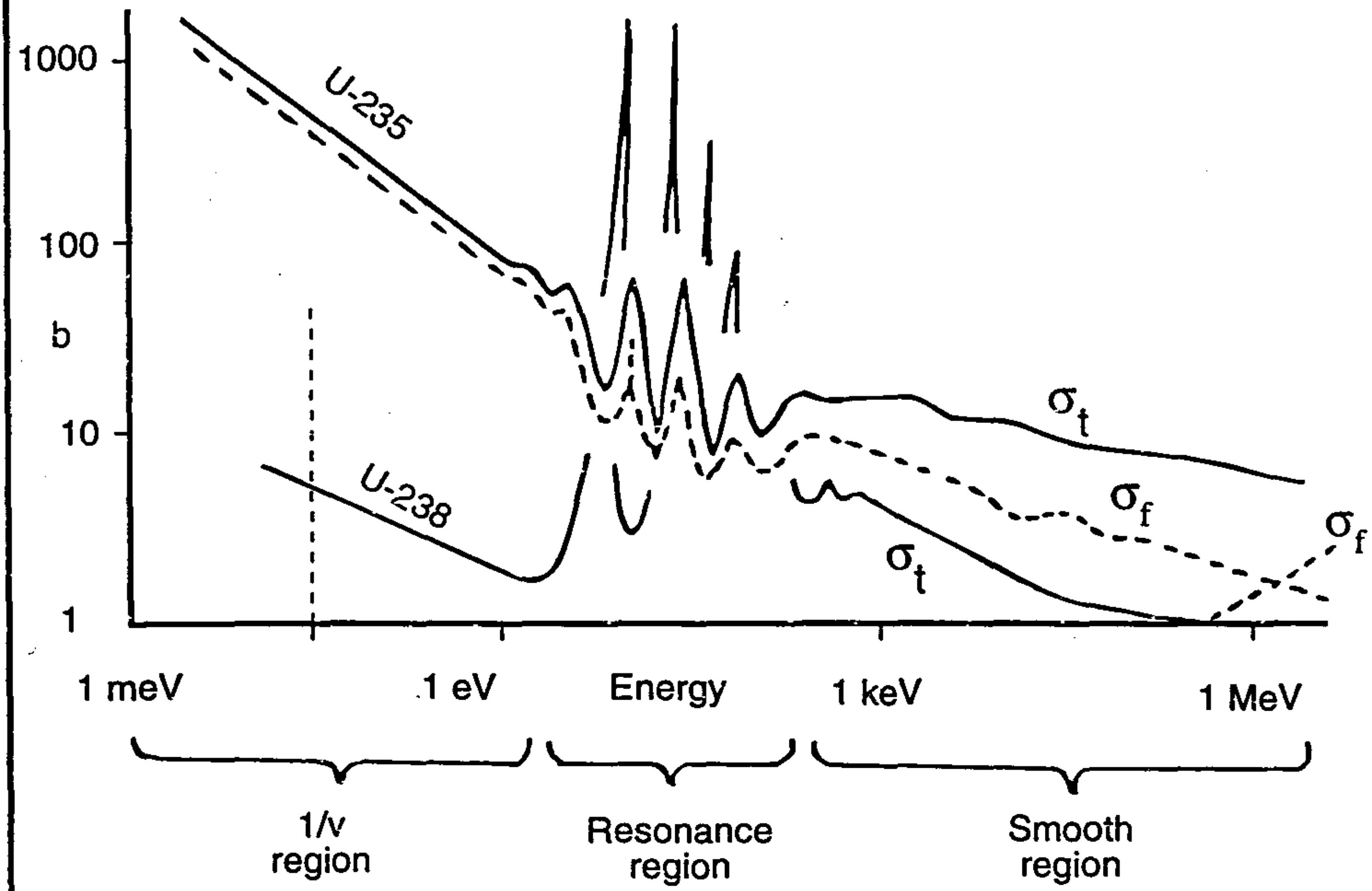


Fig.

Bundles Arranged in Single Line



Fission Characteristics



Interactions of importance

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

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

Probability of fission: $p = \sigma_f / \sigma_a$

Neutron Multiplication Factor

Neutron multiplication factor

$k_\infty = \varepsilon\, p\, \eta\, f$ (4 Factor)

$k = \varepsilon\, p\, \eta\, f\, \Lambda_f \Lambda_t$ (6 Factor)

ε = Fast fission factor

p = Resonance escape probability

η = Reproduction factor

$= \nu\, \Sigma_f^{FUEL} \Sigma_t^{FUEL}$

ν = Neutrons per fission

f = Thermal utilization factor

$= \Sigma_a^{FUEL} / \Sigma_a^{REACTOR}$

Λ_f = Fast neutron non-leakage probability

Λ_t = Slow neutron non-leakage probability

For reactor of finite size

$k = k_\infty \Lambda_f \Lambda_t$

k_∞ = k value for infinitely large reactor

Sphere

Volume: $\frac{\pi}{6} D^2$

Surface: πD^2

Surface/ volume ratio: $\pi D^2 / \frac{\pi}{6} D^2$

$= 6/D$

if D = 1 Ratio = 6

if D = 2 Ratio = 3

if D = 3 Ratio = 2

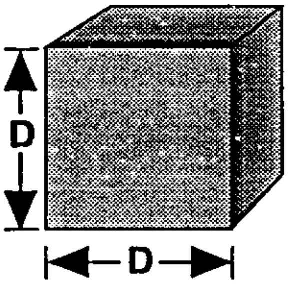
if D = 4 Ratio = 1.5

if D = 5 Ratio = 1.2

if D = 6 Ratio = 1

Surface-Volume Ratio

Cube of side D volume 100



$\text{Surface} = 6D^2$

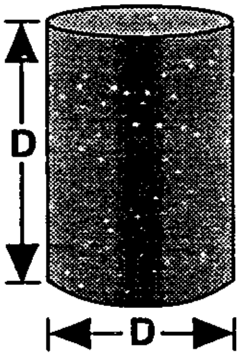
$\text{Volume} = D^3$

$D^3 = 100 \quad \therefore \quad D = 4.64$

$S = 6(4.64)^2 = 129$

$\text{S:V Ratio} = 129/100 = 1.29$

Cylinder of length D, diameter D, volume 100



$\text{Surface} = 2\frac{\pi}{4} D^2 + \pi D(D) = \frac{3}{2} D^2$

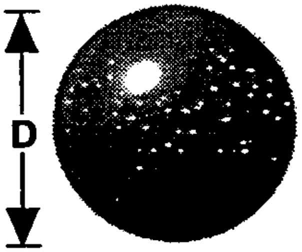
$\text{Volume} = \frac{\pi}{4} D^2 D = \frac{\pi}{4} D^3$

$D^3 = 100(\frac{\pi}{4}) \quad \therefore \quad D = 5.03$

$S = \frac{3}{2} \pi(5.03)^2 = 119$

$\text{S:V Ratio} = 119/100 = 1.19$

Sphere of diameter D, volume 100



$\text{Surface} = \pi D^2$

$\text{Volume} = \frac{\pi}{6} D^3$

$D^3 = 100(\frac{6}{\pi}) \quad \therefore \quad D = 5.76$

$S = \pi(5.76)^2 = 104$

$\text{S:V Ratio} = 104/100 = 1.04$

Variation of the Thermal Neutron Flux along the Axis of a Cylindrical Reactor

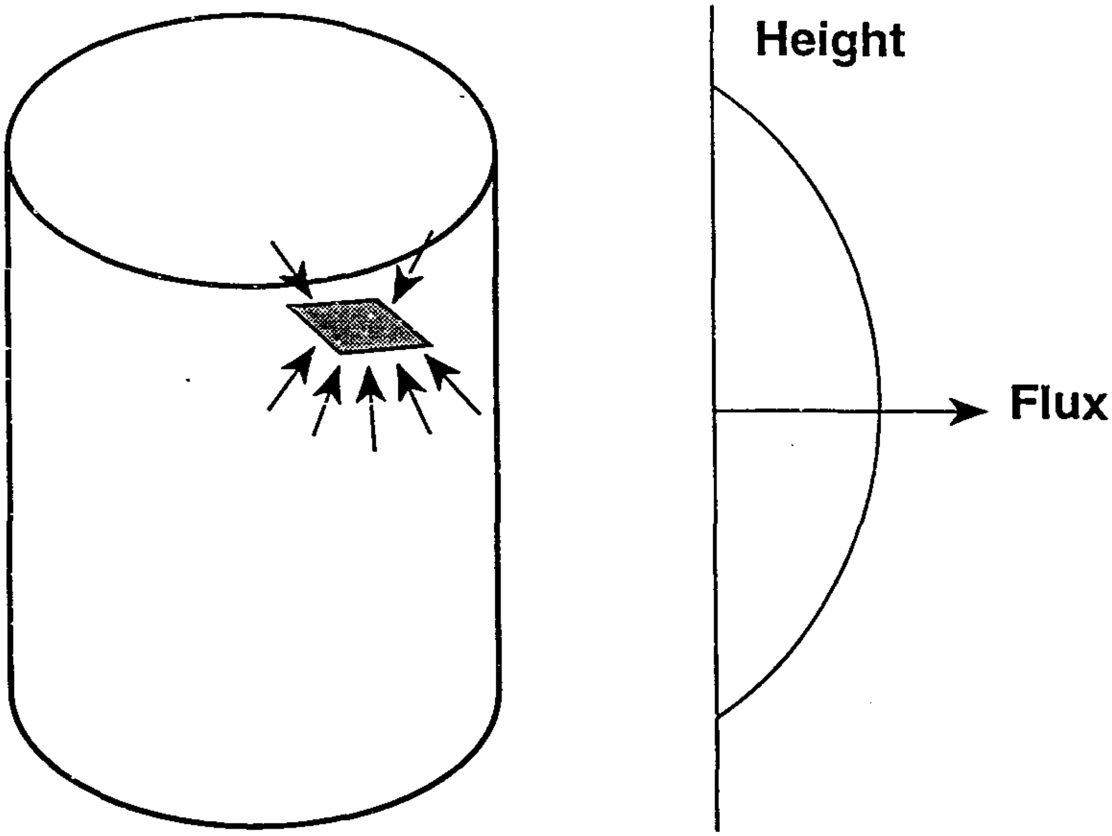
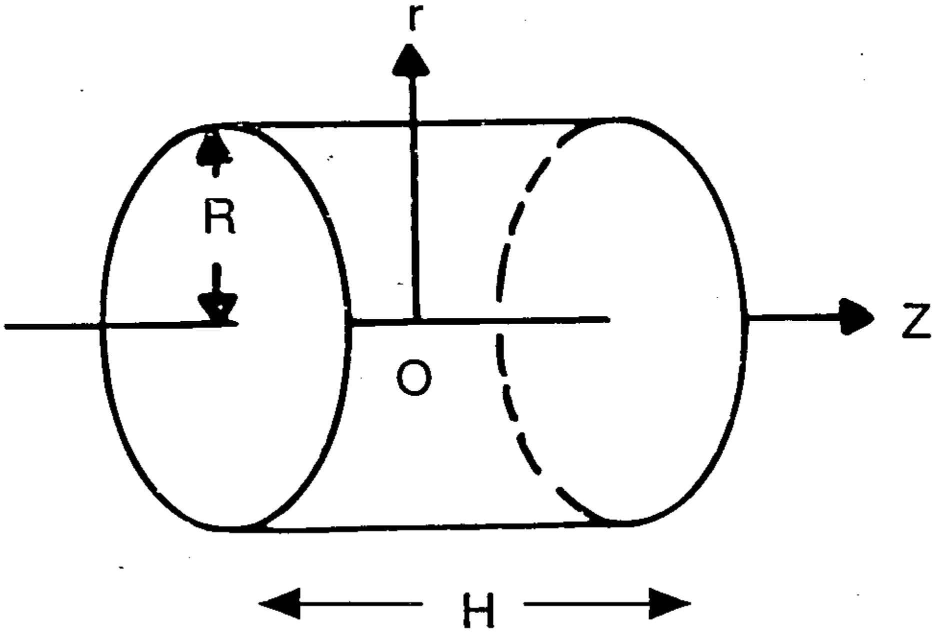


Fig. 6

Variation of Thermal Neutron Flux in Axial and Radial Directions in a Cylindrical Reactor



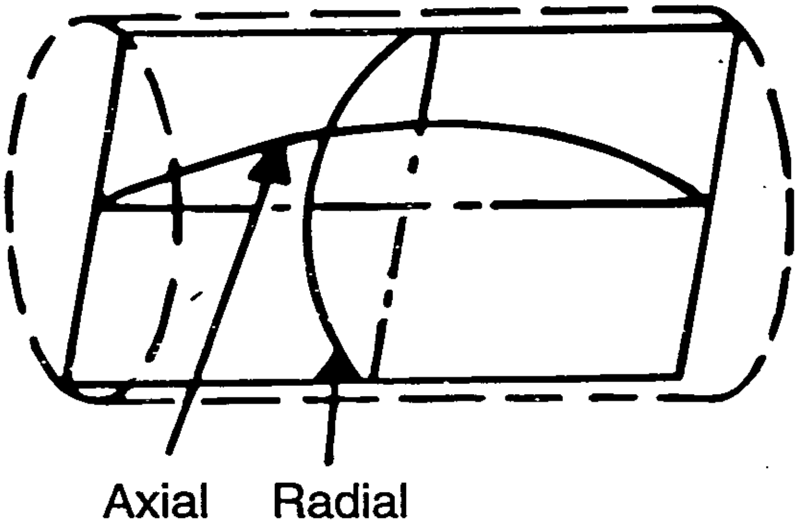
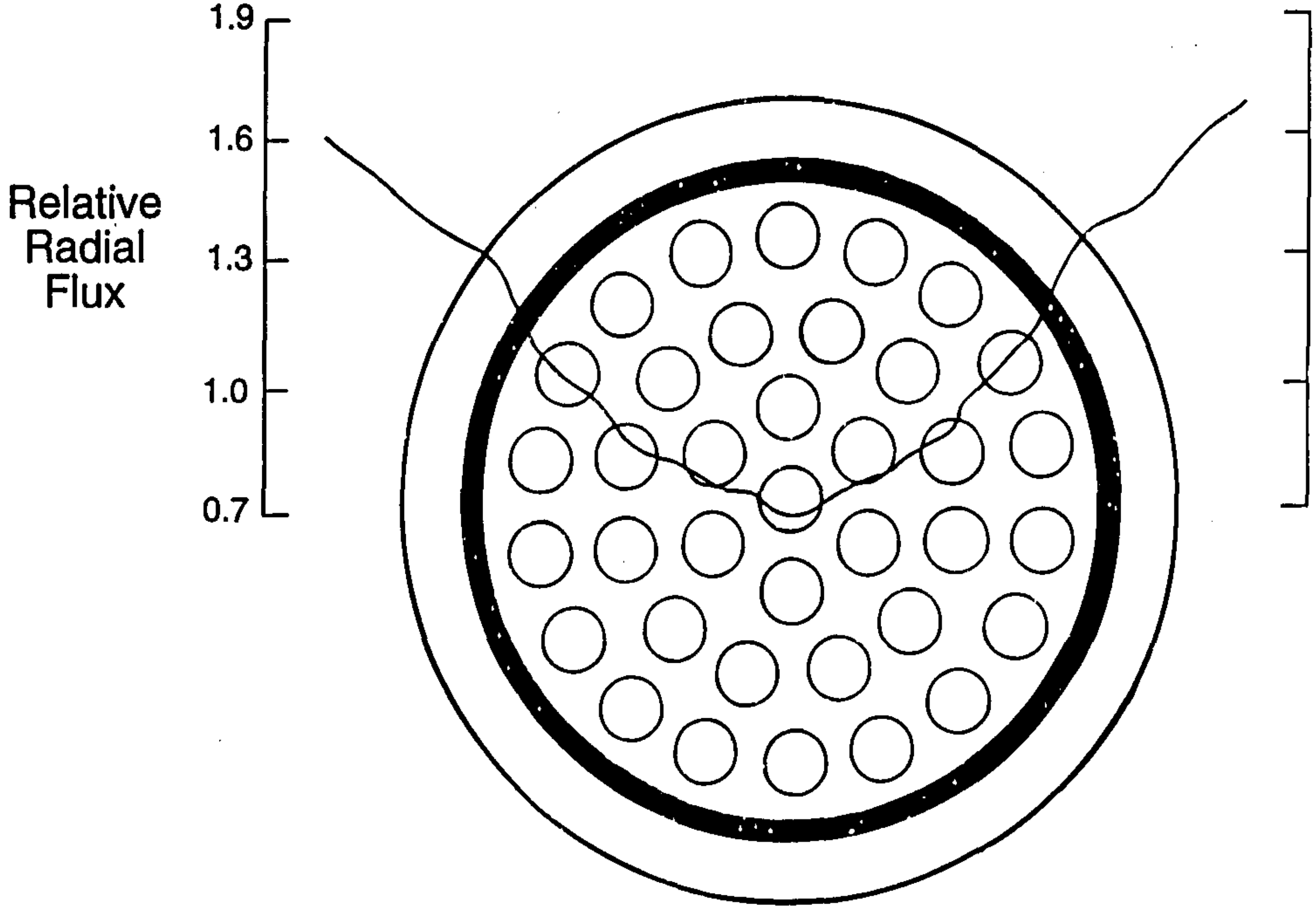
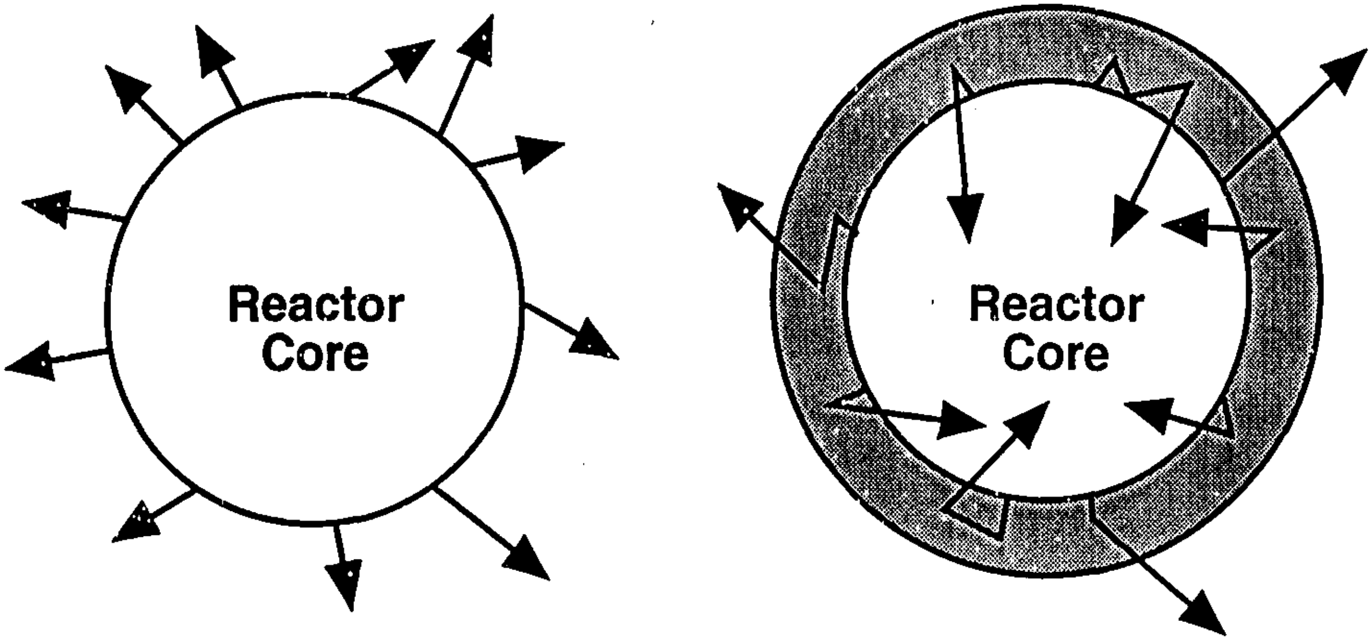


Fig. 6.

Depression of the Thermal Neutron Flux in the Interior of Fuel Bundle



Comparison of Neutron Leakage for Bare and Reflected Cores



(a) (b)

Effect of Reflector on Shape of Radial Flux

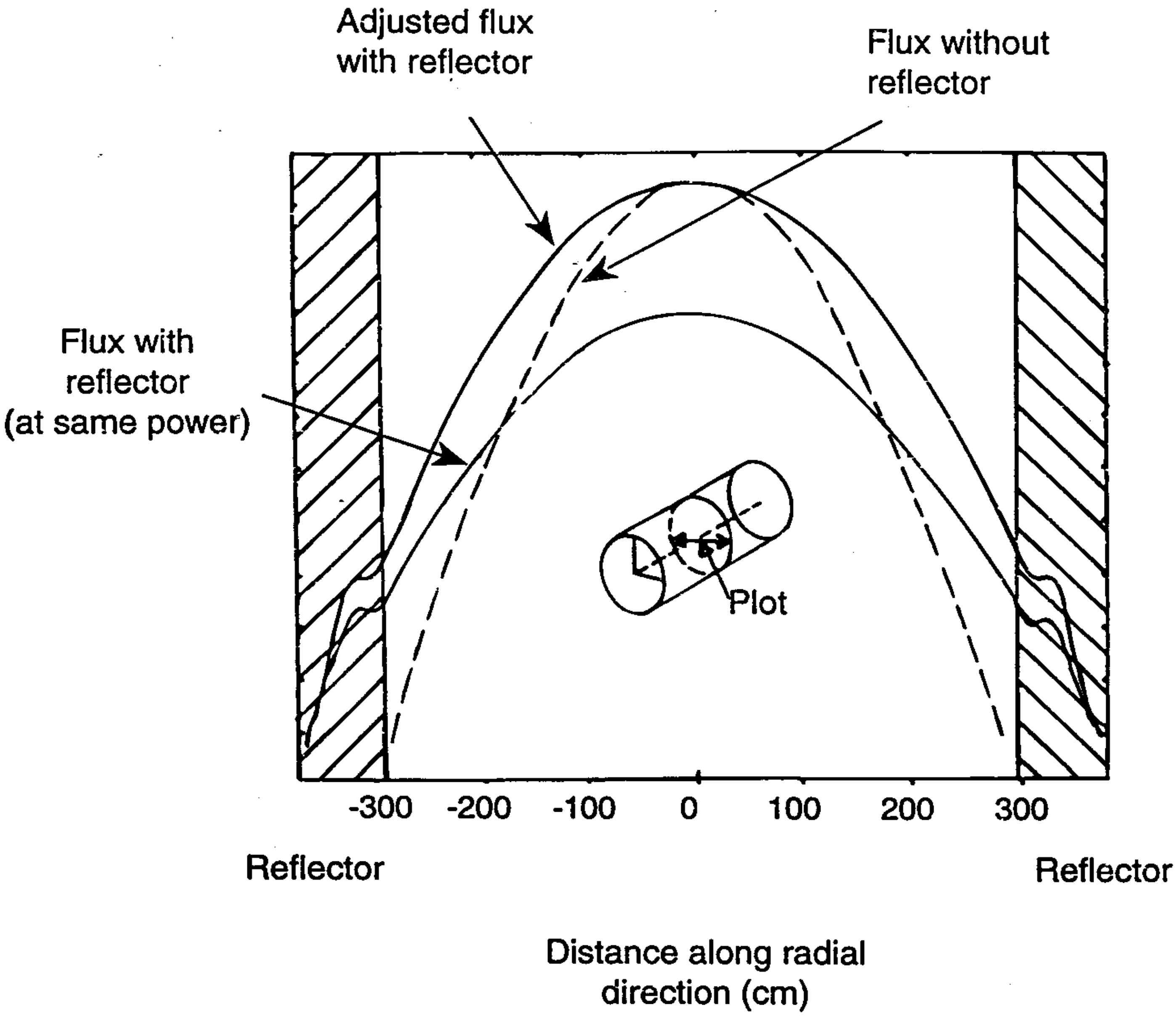
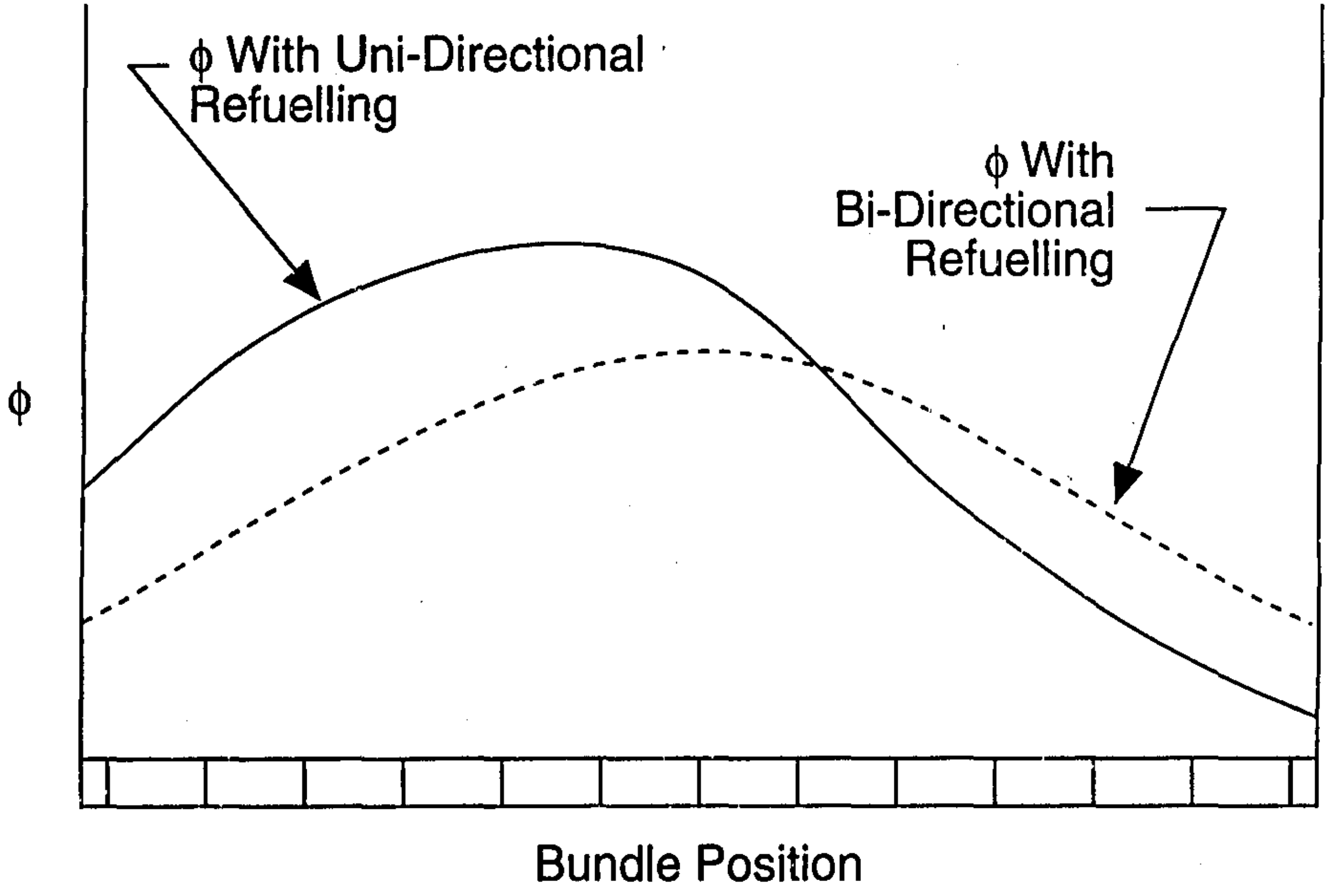
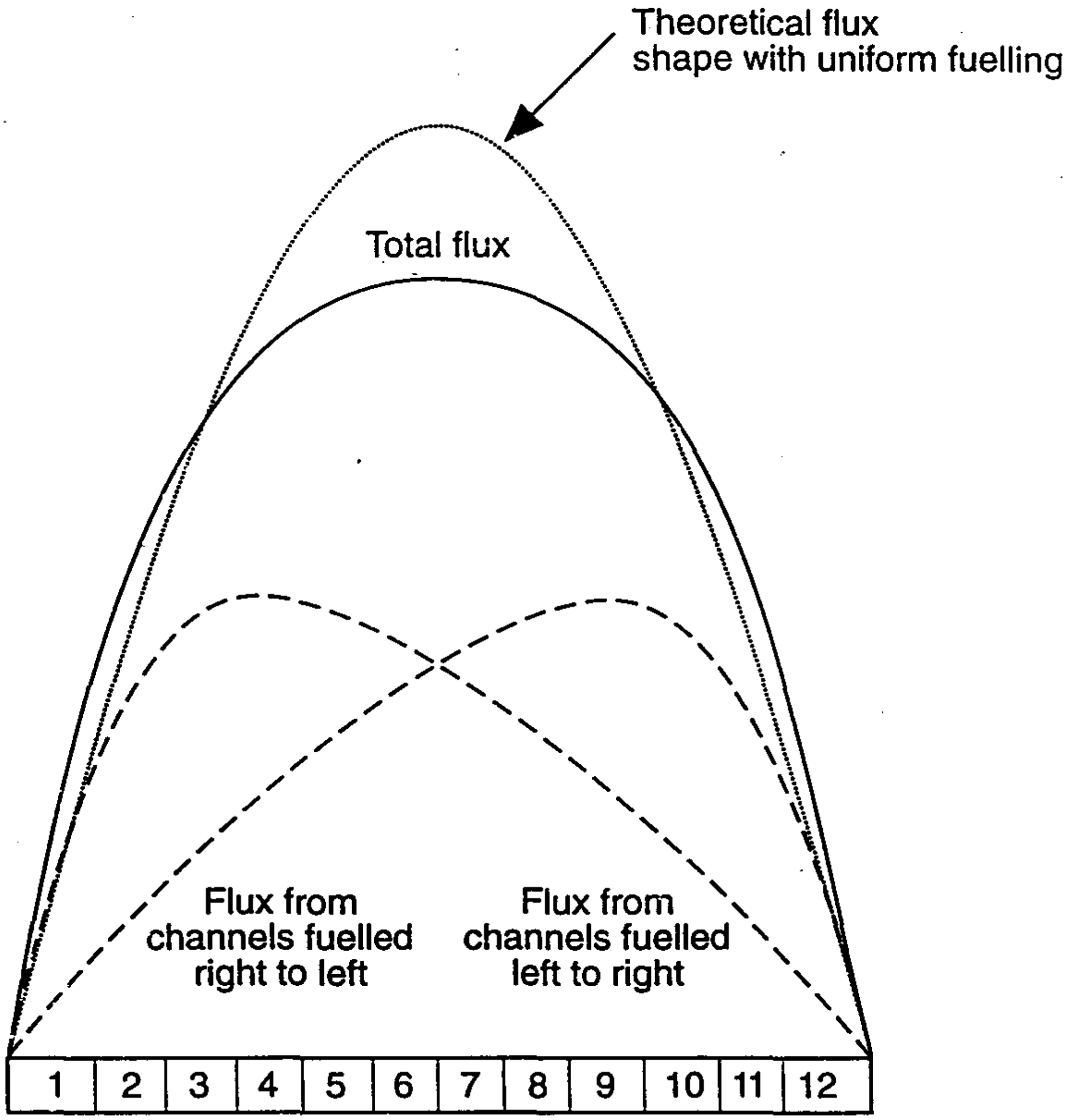


Fig. 6.

Asymmetry Produced by Uni-Directional Refuelling



Effect of Bi-Directional Refuelling in Flattening Axial Flux Shape



Bundle positions along channels

Fig. 6.

Flux Flattening Produced by Adjuster Rods

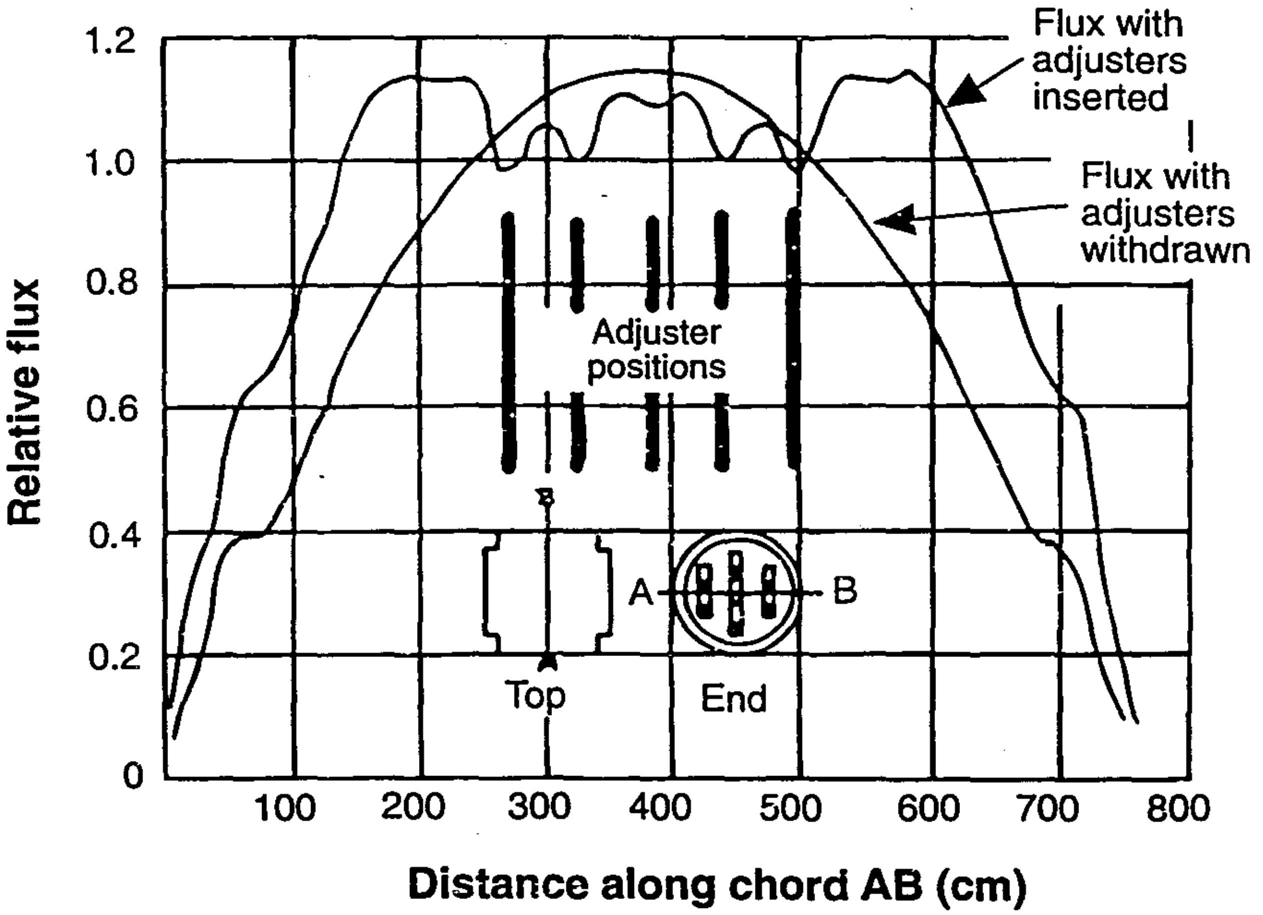
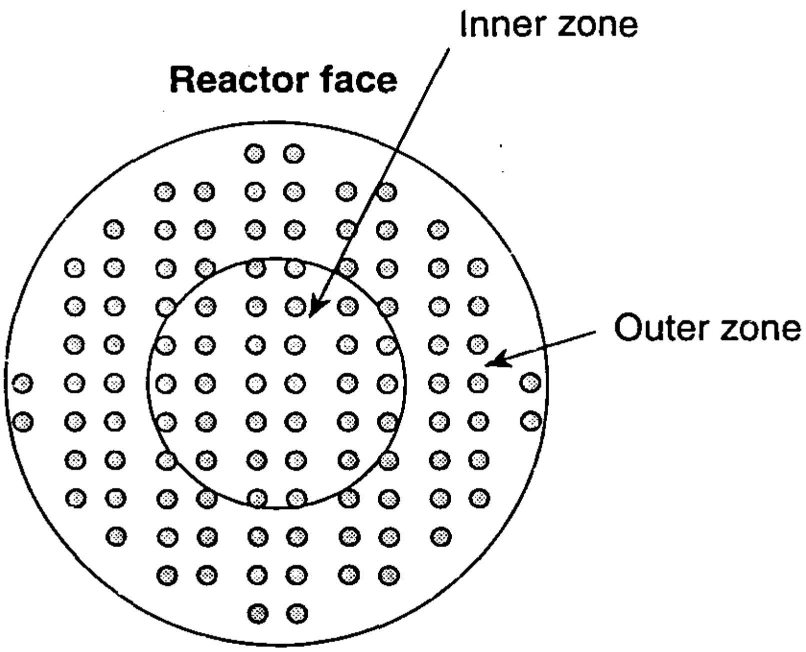


Fig. 6.

Flux Flattening Produced by Differential Fuelling



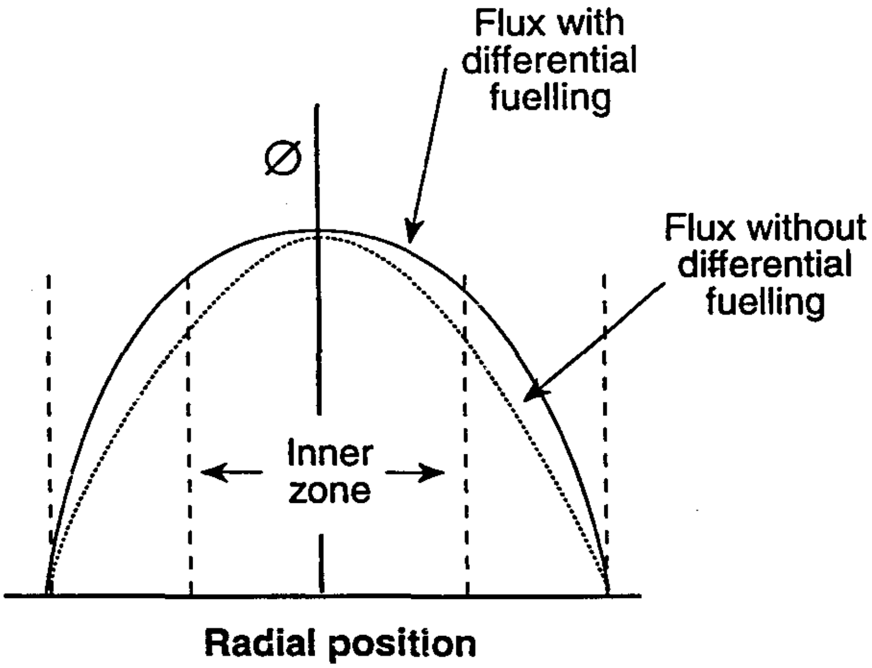
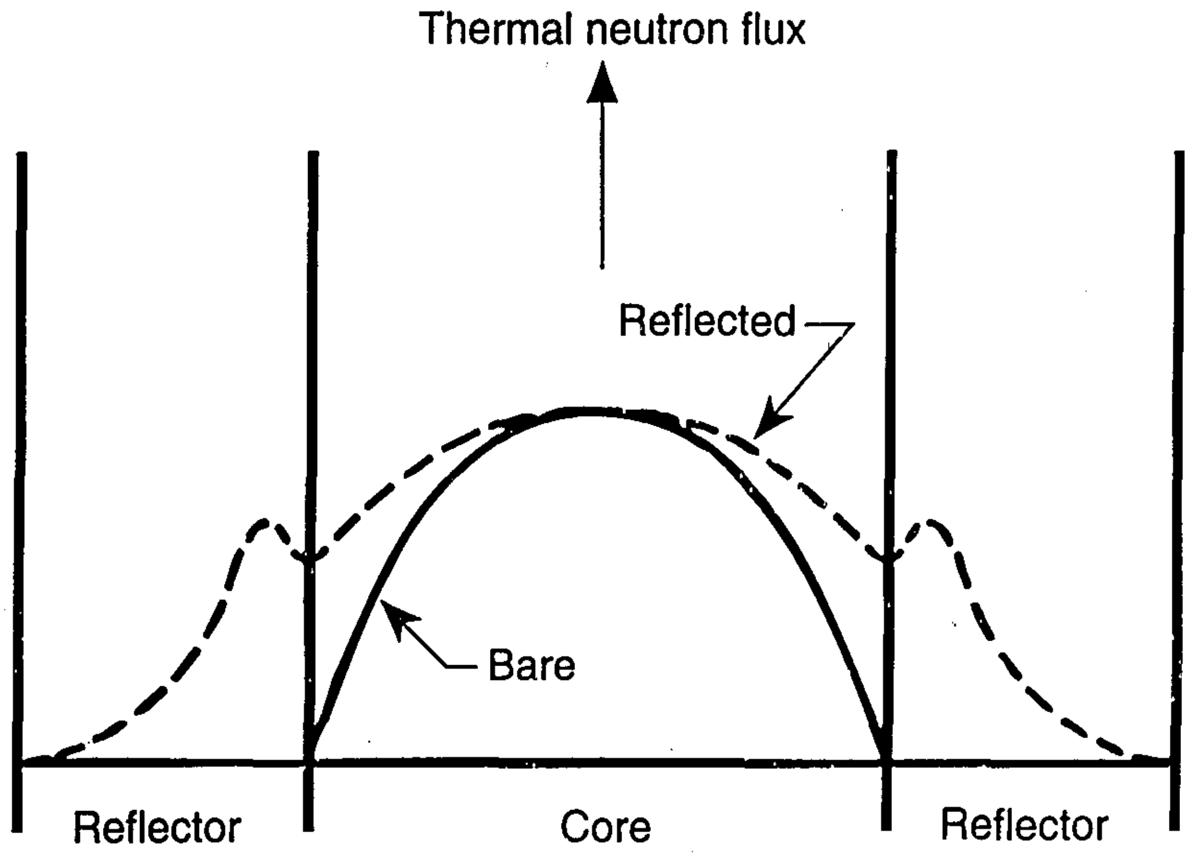
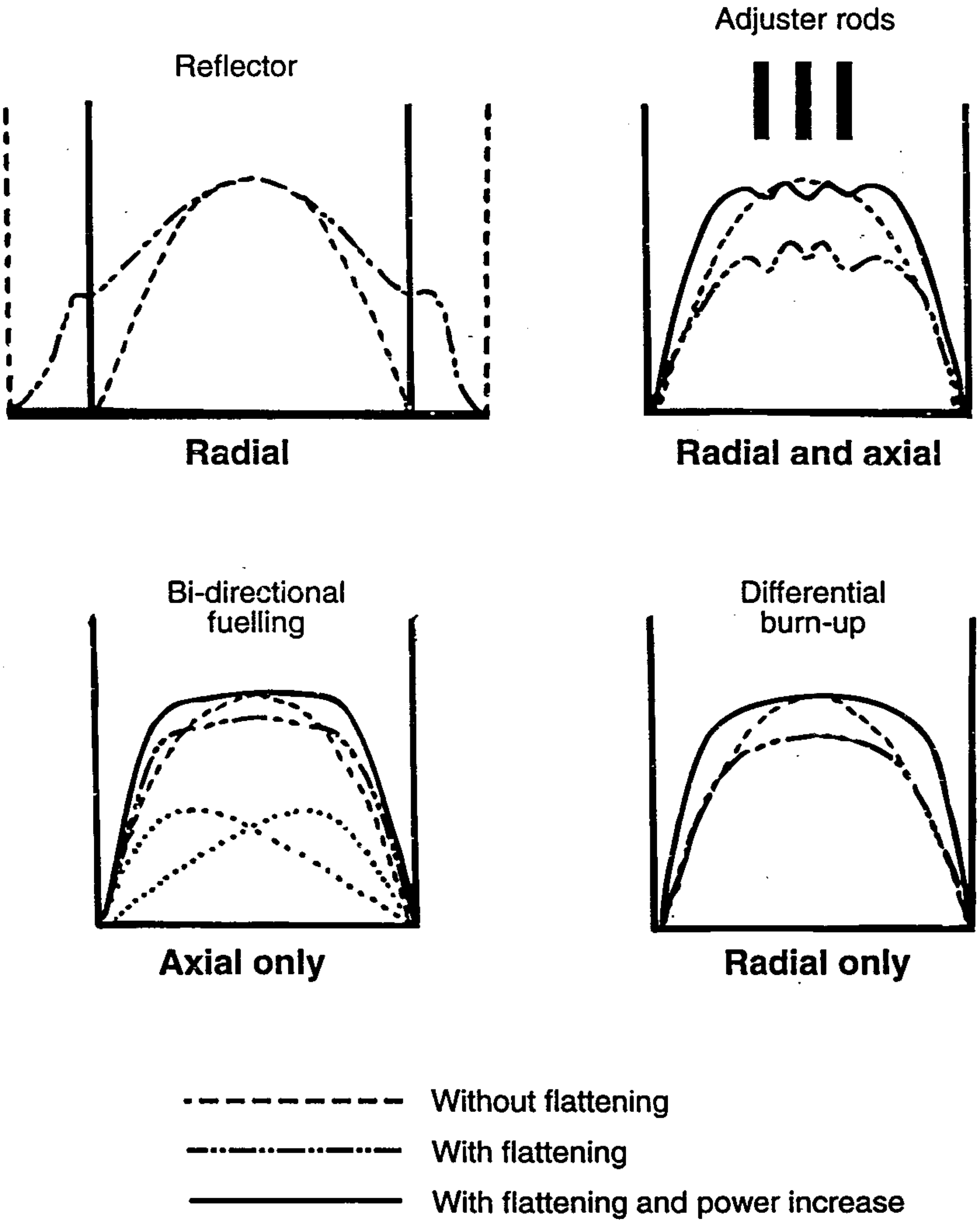


Fig. 6

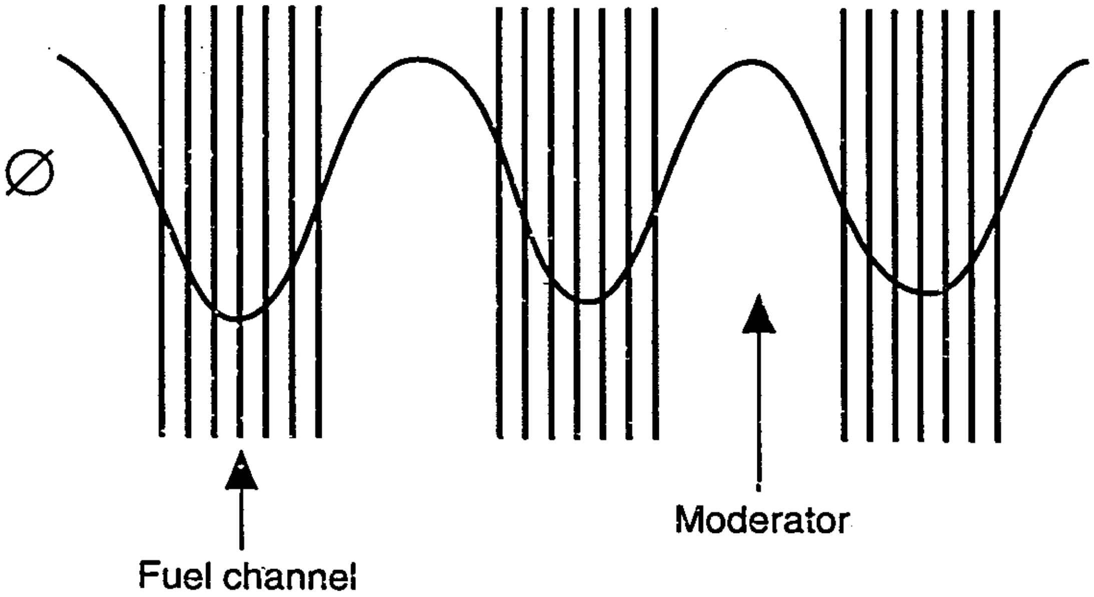
Flux Distribution in Reflected Reactor

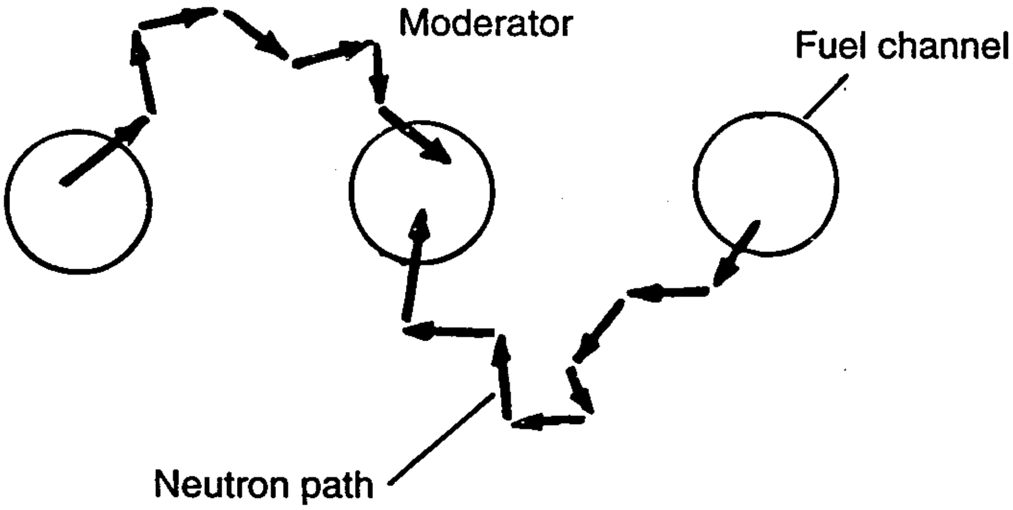


Flux Flattening

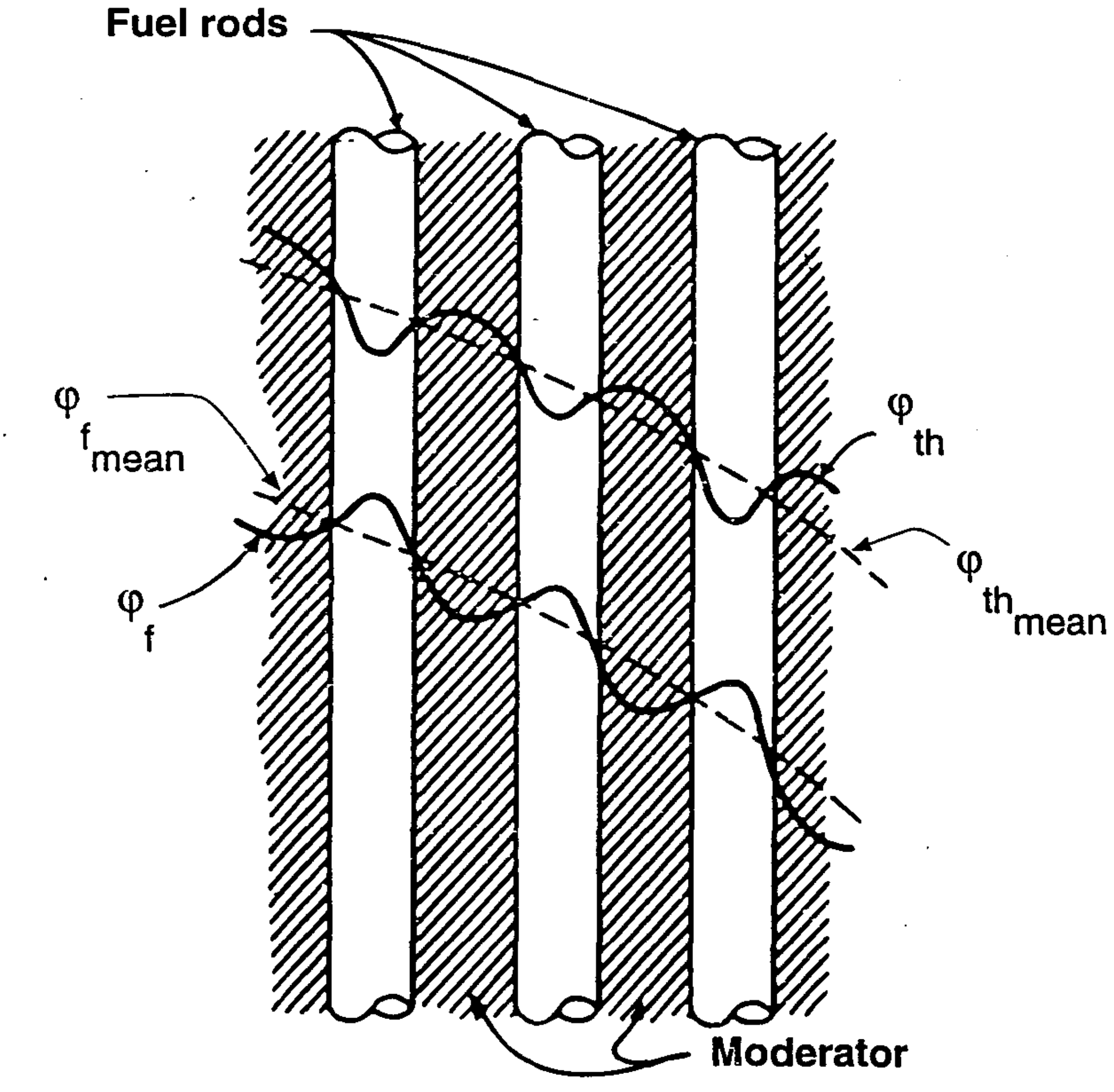


Neutron Flux in Core





Flux Variation in Fuel



Flux Flattening in CANDU Reactors

	Reflector	Bi-directional fuelling	Adjusters	Differential burnup	$\frac{\phi_{avg}}{\phi_{max}}$
NPD	axial & radial	x			42%
Douglas Point	radial	x		x	50%
Pickering - A	radial	x	x		57%
Pickering - B	radial	x	x		~60%
Bruce - A	radial	x		x	~59%
Bruce - B	radial	x	x		~60%
Darlington	radial	x	x	x	~60%
Point Lepreau	radial	x	x	x	~60%