REACTOR FLUX PROFILES

Table 1 Water Cooled Reactor Temperatures and Pressures

Reactor Type	Coolant Temperature (°C)	Coolant Pressure (MPa)	Steam Temperature (°C)	Steam Pressure (MPa)
PWR	325	15.5	273	5.8
BWR	289	7.3	289	7.2
PHWR	312	11.2	265	5.1
LGR	289	7.3	285	6.9

In the case of the PWR and PHWR (CANDU) heat must be transferred from the primary coolant loop to the secondary steam system so that the coolant temperature must be even higher than the steam temperature. For the PWR where boiling is totally suppressed the coolant pressure is the highest. Overall however it is the pressure of the steam that is the limiting factor since the vessel in which the steam is generated or separated must be of adequate size to accommodate the increase in volume associated with the change in phase. The size of the vessel in turn affects the wall thickness which is governed by manufacturing and transportation limitations as well as the economical considerations mentioned above.

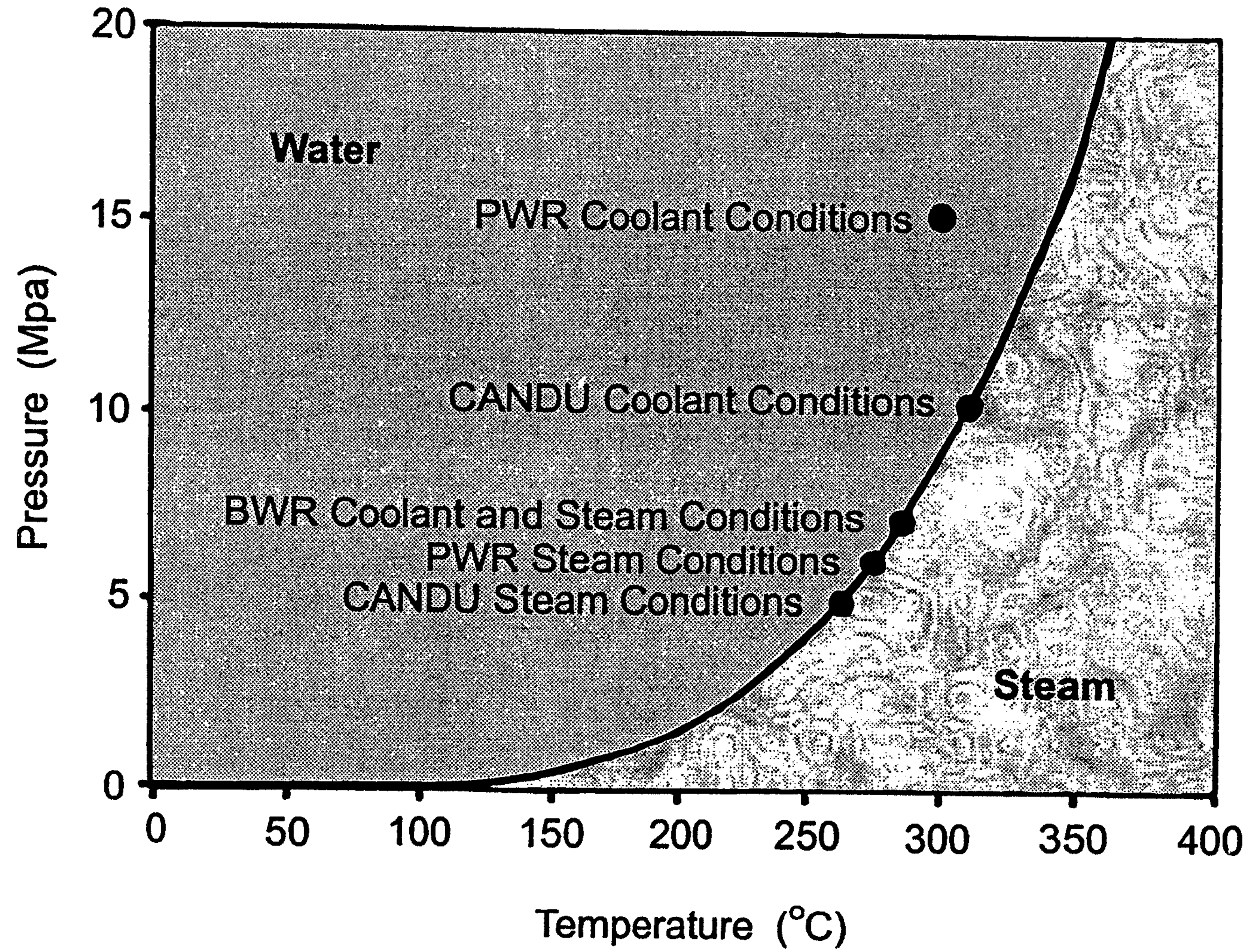


Figure 1 Variation in saturation pressure with water temperature

Definitions

Macroscopic cross-section

(Cross-section density in material)

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

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

σ = Microscopic cross-section $\left(cm^2\right)$

Neutron flux

(Neutrons passing through given area per second)

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

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

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

Reaction rate

(Reaction rate of neutrons with material)

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

ϕ = Neutron flux $\left(\frac{neutrons}{cm^2\,s}\right)$

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

Reactor Power

Neutron Flux

ϕ = nv (neutrons/cm^{2}s)

n = Neutron density (neutrons/cm^3)

v = Neutron velocity (cm/s)

Macroscopic cross-section

Σ_f = $N\sigma_f$ (cm^{-1})

N = Nuclei density (nuclei/cm^3)

σ_f = Microscopic cross-section (cm^2)

Reactor power

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

v = Volume of reactor (cm^3)

E_f = Energy release per fission (J)

FISSIONABLE FUEL DENSITY

GENERAL EQUATION

$$N = \frac{N_A}{A} m \qquad \text{(nuclei)}$$

FOR COMPOUNDS (M = molecular mass, i = nuclei/molecule)

$$N = \frac{N_A}{M} i\, m$$

FOR NUMBER PER UNIT VOLUME

$$N = \frac{N_A}{M} i \rho \qquad (\text{nuclei}/m^3)$$

FOR NUMBER OF NUCLEI OF FISSIONABLE FUEL

$$N_{ff} = \frac{N_A}{M_{ff}} i \rho_{ff} \qquad (\text{nuclei}/m^3 \text{ fuel})$$

$$N_{ff} = \frac{A_v}{M_{ff}} \rho_{ff}\, i \qquad \text{(TEXT BOOK 4-4)}$$

WHEN ENRICHED (r = enrichment, f = mass fraction of fuel)

$$\rho_{ff} = r f \rho_{fm}$$

(ρ_{fm} = enriched fuel compound density)

NUMBER OF FISSIONABLE FUEL NUCLEI

$$N_{ff} = \frac{N_A}{M_{ff}} r f \rho_{fm} i \quad (\text{nuclei}/m^3 \text{ fuel})$$

$$N_{ff} = \frac{Av}{M_{ff}} r \rho_{fm} f i \quad (\text{TEXT BOOK } 4\text{-}7)$$

FOR URANIUM DIOXIDE UO_2

$$i = 1$$

$$\rho_{fm} = 10.5 \ g/cm^3 \quad (kg/m^3)$$

$$f = \frac{r M_{u\text{-}235} + (1-r) M_{u\text{-}238}}{r M_{u\text{-}235} + (1-r) M_{u\text{-}238} + M_{2(0\text{-}16)}}$$

ROUND OFF ALL ATOMIC MASS NUMBERS

TO NEAREST WHOLE NUMBER

ERROR = 0.3 %

FOR LOW ENRICHMENTS

$$f \approx \frac{238}{270} = 0.881$$

RIGOROUS CALCULATION IN TEXT BOOK GIVES:

$f = 0.8815$ FOR 3% ENRICHMENT

NEUTRON DENSITY DISTRIBUTION

BASIC PRINCIPLES

- NEUTRONS DIFFUSE IN ALL DIRECTIONS WITHIN THE REACTOR CORE
- NEUTRONS LEAK THROUGH THE EXTERNAL SURFACE OF THE REACTOR

RESULTANT NEUTRON DENSITY

- NEUTRON DENSITY DECREASES NEAR PERIPHERY DUE TO LEAKAGE.
- MORE NEUTRONS DIFFUSE AWAY FROM ZONES OF HIGH NEUTRON DENSITY THAN FROM ZONES OF LOW DENSITY.
- NEUTRONS THEREFORE DIFFUSE DOWN THE NEUTRON DENSITY GRADIENT
- NEUTRON DIFFUSION IS TOWARDS THE SURFACE WHERE LEAKAGE OCCURS.

Comparison of Neutron Leakage for Bare and Reflected Cores

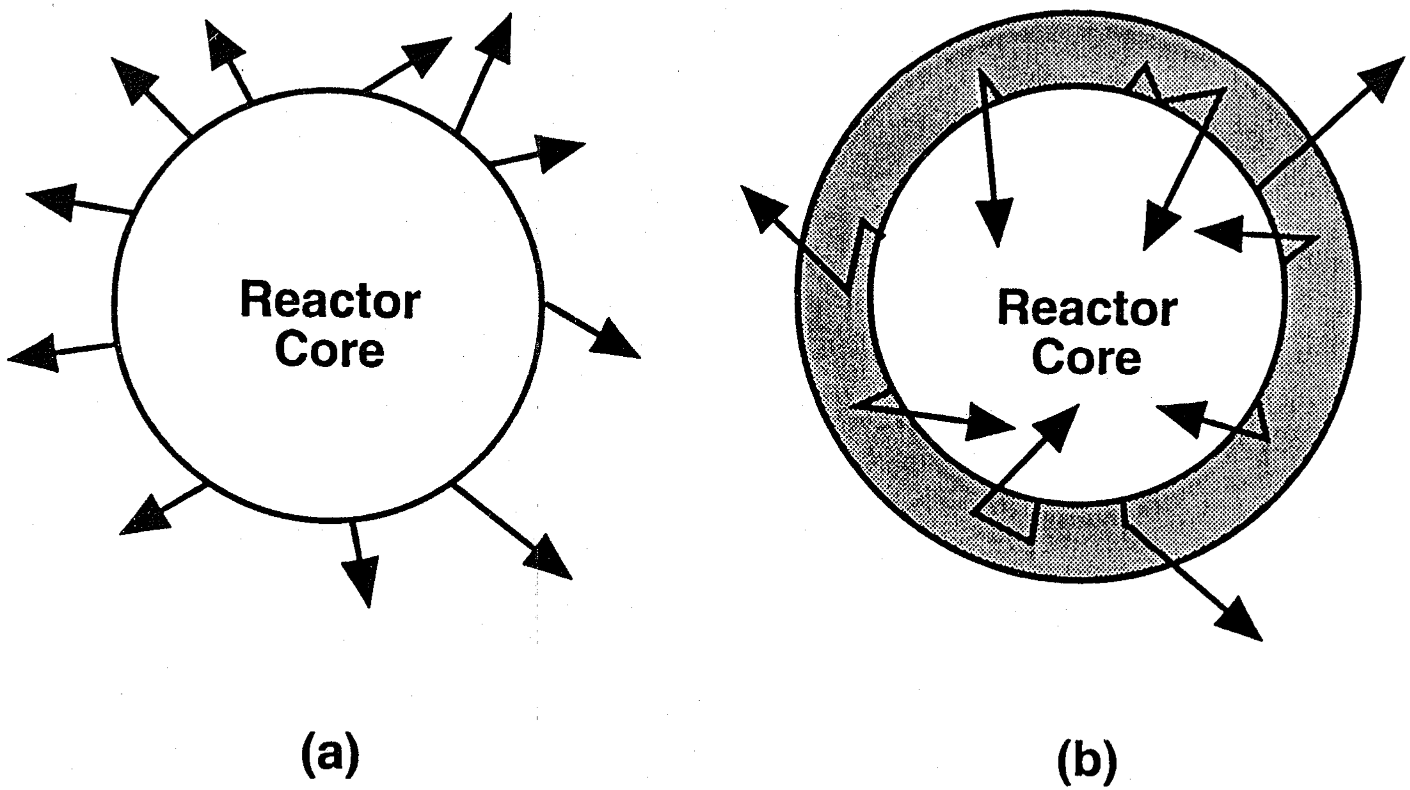


Fig. 6.4

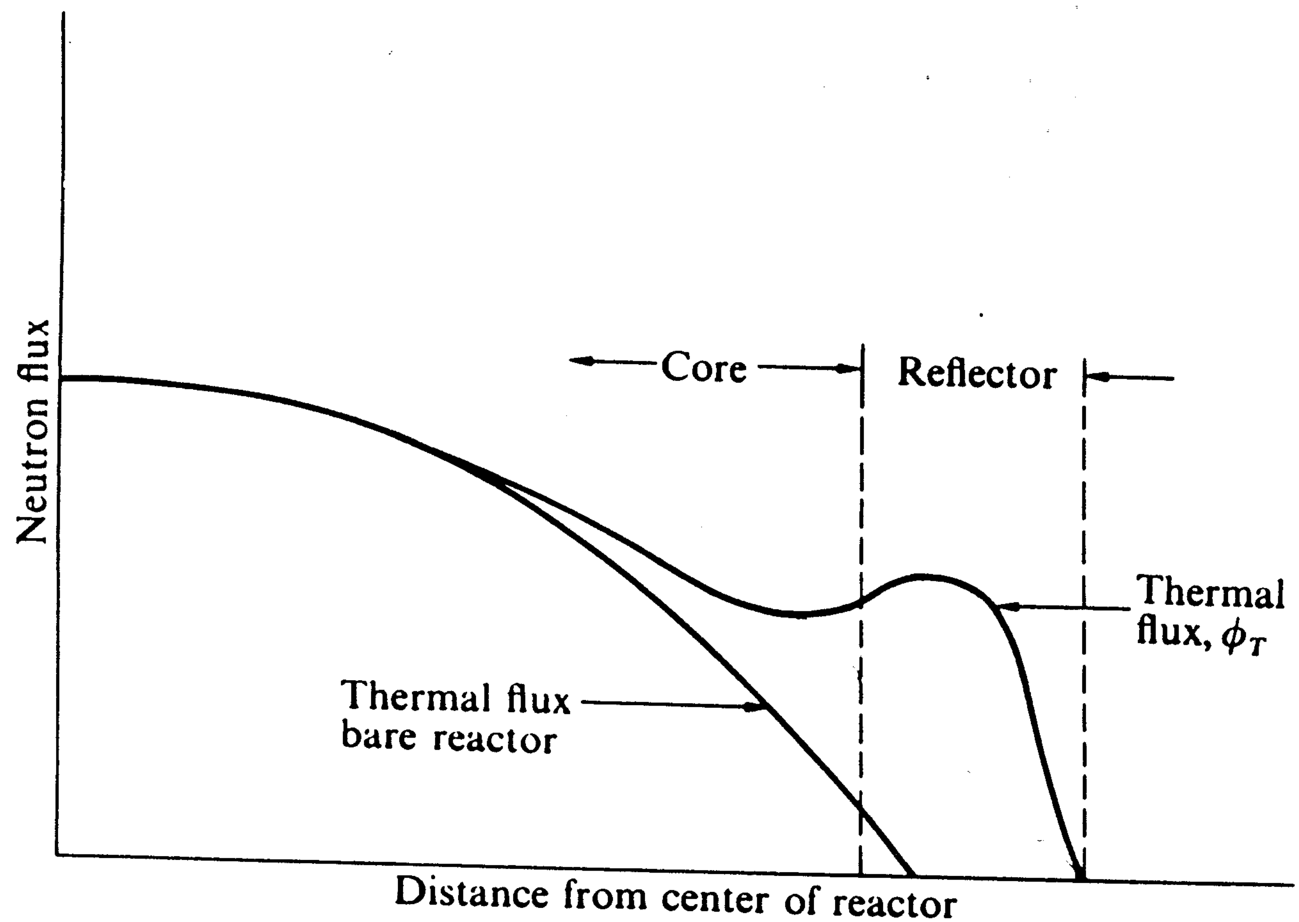


Fig. 6.5 Fast and thermal fluxes in a reflected thermal reactor and the thermal flux in the equivalent reactor.

NEUTRON FLUX DISTRIBUTION

FLUX DEFINITION

$$\phi = nv$$

NEUTRON FLUX $\propto$ NEUTRON DENSITY

$$\phi \propto n$$

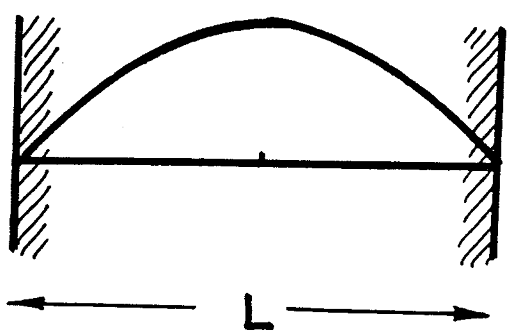
FLUX VARIATION

DENSITY DECREASES TOWARDS CORE BOUNDARY

FLUX ALSO DECREASES IN SAME MANNER

REPRESENTATION OF FLUX DISTRIBUTION

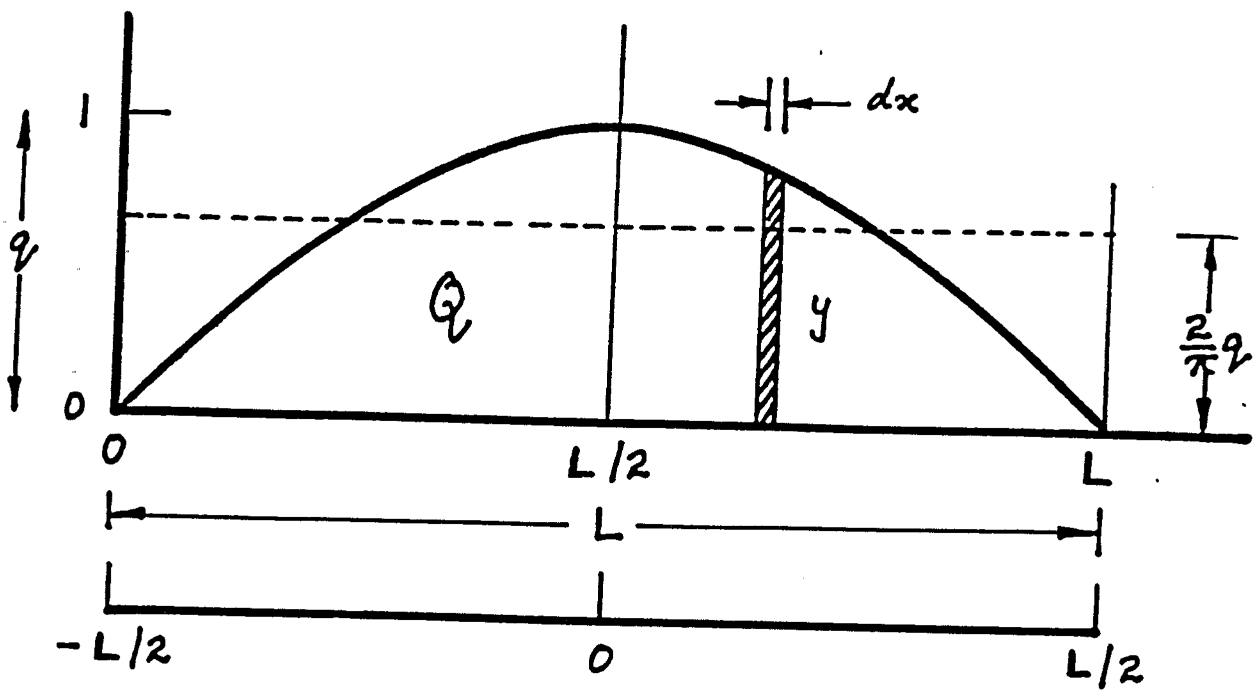
DEFINED MATHEMATICALLY AS COSINE FUNCTION



$$\phi = \phi_{max} \cos \frac{\pi x}{L}$$

NEUTRON FLUX PROFILE

HEAT GENERATION IN NUCLEAR REACTOR



$$Q = \int_0^L y\,dx$$
$$= \int_0^L q \sin\frac{\pi x}{L}\,dx$$
$$= \left[-q\frac{L}{\pi}\cos\frac{\pi x}{L}\right]_0^L$$
$$= -q\frac{L}{\pi}\left(\cos\frac{\pi L}{L} - \cos 0\right)$$
$$= -q\frac{L}{\pi}(-1-1)$$
$$= \frac{2}{\pi}qL$$

$$Q = \int_{-L/2}^{L/2}$$
$$= \int_{-L/2}^{L/2} q\cos\frac{\pi x}{L}\,dx$$
$$= \left[q\frac{L}{\pi}\sin\frac{\pi x}{L}\right]_{-L/2}^{L/2}$$
$$= q\frac{L}{\pi}\left(\sin\frac{\pi L}{2L} - \sin\left(-\frac{\pi L}{2L}\right)\right)$$
$$= q\frac{L}{\pi}(1+1)$$
$$= \frac{2}{\pi}qL$$

NUCLEAR HEAT GENERATION

REACTION RATE

$$R = \Sigma\phi = N\sigma nv$$

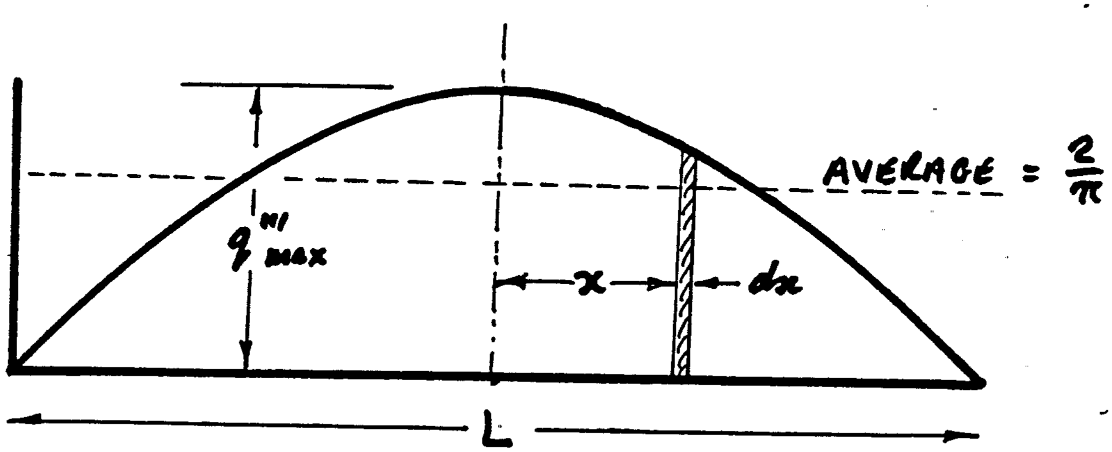
$$R_{fission} = \Sigma_f \phi = N_{ff}\sigma_f \phi$$

RATE OF HEAT GENERATION

$$q''' = \Sigma_f \phi E_f = N_{ff}\sigma_f \phi E_f$$

$$q''' = GR = G\Sigma\phi \quad \text{(TEXT BOOK 4-1)}$$

PROFILE OF HEAT GENERATION



$$q_t = \int_{-L/2}^{L/2} q''' \cos\frac{\pi x}{L} A\, dx$$

$$q_t = \frac{2}{\pi} q''' AL$$

OTHER REACTOR SHAPES

THE SLAB REACTOR

$$\phi = \left(1.57P / a E_R \Sigma_f\right) \cos\left(\frac{\pi x}{a}\right)$$

$$\phi_{MAX} / \phi_{AVE} = 1.57$$

THE SPHERICAL REACTOR

$$\phi = \left(P / 4 E_R \Sigma_f R^2\right)\left(\frac{1}{r}\right) \sin\left(\frac{\pi r}{R}\right)$$

$$\phi_{MAX} / \phi_{AVE} = 3.29$$

THE INFINITE CYLINDRICAL REACTOR

$$\phi = \left(0.738P / E_R \Sigma_f R^2\right) J_0\left(\frac{2.405 r}{R}\right)$$

$$\phi_{MAX} / \phi_{AVE} = 2.32$$

THE FINITE CYLINDRICAL REACTOR

$$\phi = \left(3.63P / E_R \Sigma_f V\right) J_0\left(\frac{2.405 r}{R}\right) \cos\left(\frac{\pi z}{H}\right)$$

$$\phi_{MAX} / \phi_{AVE} = 3.64$$

THE RECTANGULAR REACTOR

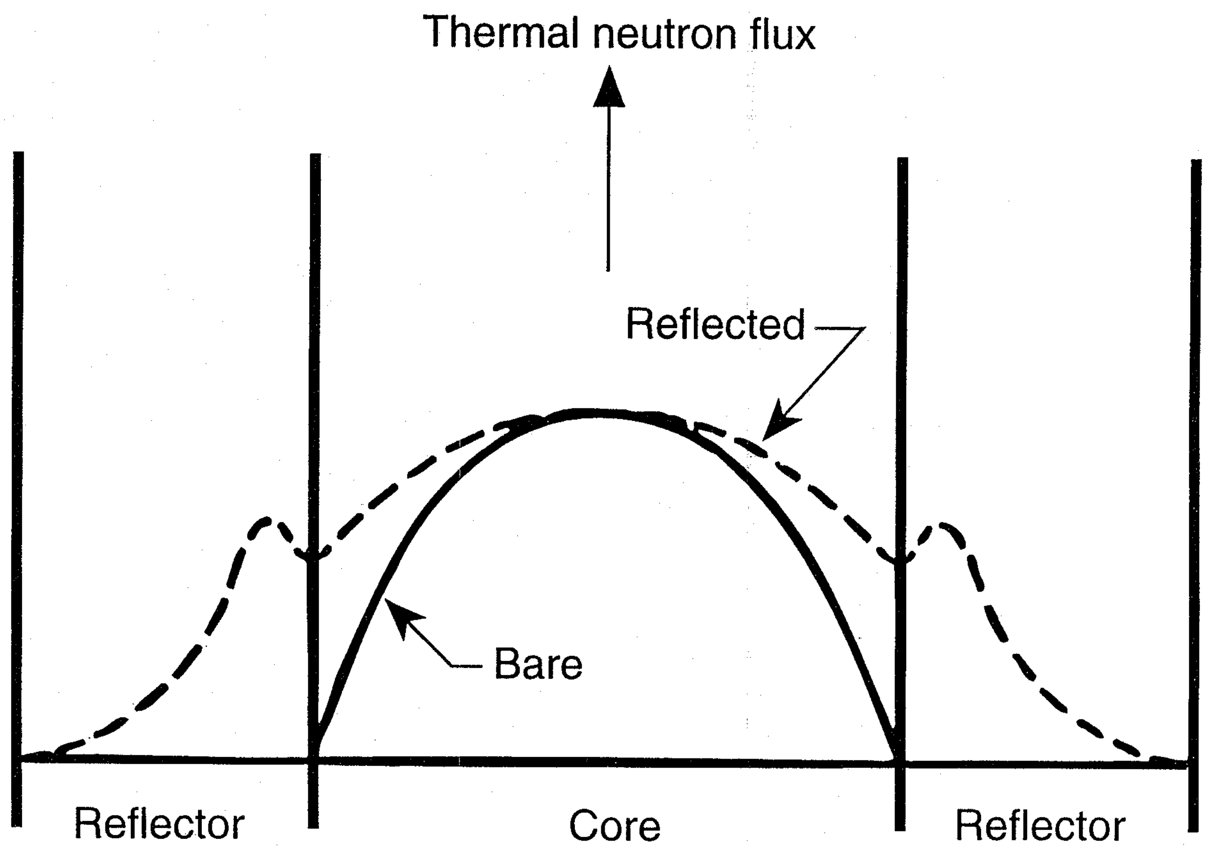
$$\phi = \left(3.87P / E_R \Sigma_f V\right) \cos\left(\frac{\pi x}{a}\right) \cos\left(\frac{\pi y}{b}\right) \cos\left(\frac{\pi z}{c}\right)$$

$$\phi_{MAX} / \phi_{AVE} = 3.88$$

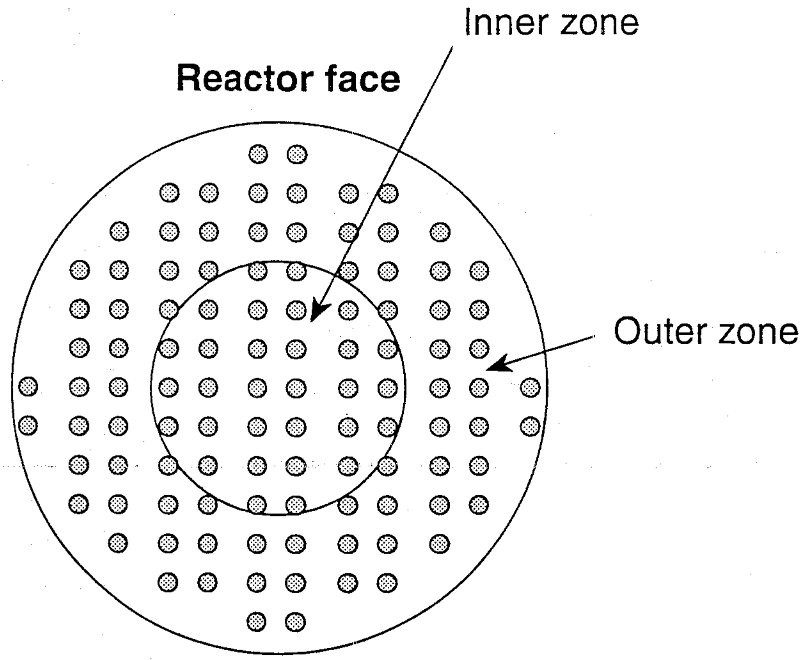
TABLE 6.2
Bucklings and fluxes for critical bare reactors

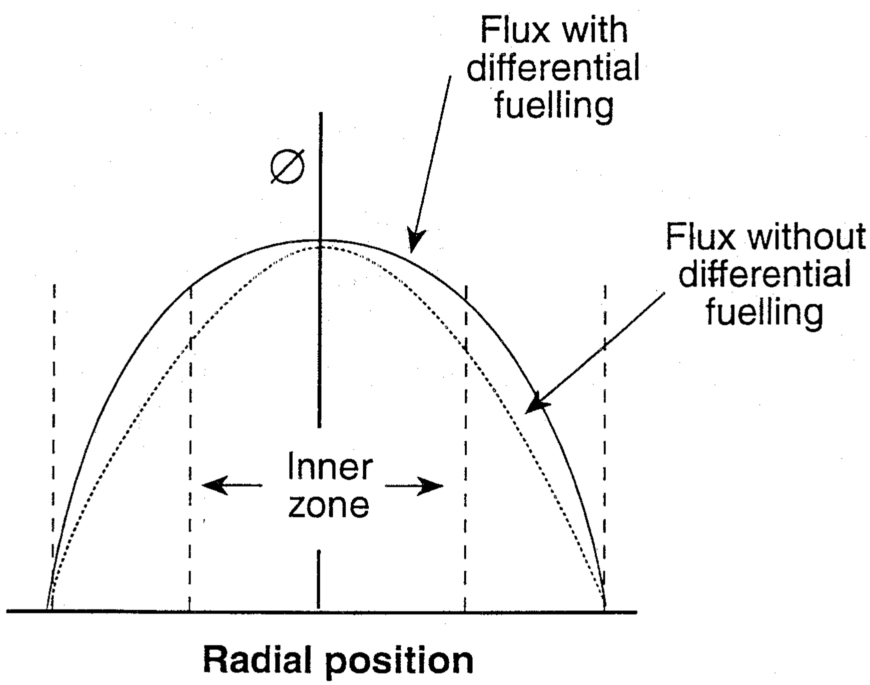
Geometry	Dimensions	Buckling	Flux	A	ϕ_{max}/ϕ_{av}
Infinite slab	Thickness a	$\left(\frac{\pi}{a}\right)^2$	$A\cos\left(\frac{\pi x}{a}\right)$	$1.57P/aE_R\Sigma_f$	1.57
Rectangular parallelepiped	$a \times b \times c$	$\left(\frac{\pi}{a}\right)^2+\left(\frac{\pi}{b}\right)^2+\left(\frac{\pi}{c}\right)^2$	$A\cos\left(\frac{\pi x}{a}\right)\cos\left(\frac{\pi y}{b}\right)\cos\left(\frac{\pi z}{c}\right)$	$3.87P/VE_R\Sigma_f$	3.88
Infinite cylinder	Radius R	$\left(\frac{2.405}{R}\right)^2$	$AJ_0\left(\frac{2.405r}{R}\right)$	$0.738P/R^2E_R\Sigma_f$	2.32
Finite cylinder	Radius R Height H	$\left(\frac{2.405}{R}\right)^2+\left(\frac{\pi}{H}\right)^2$	$AJ_0\left(\frac{2.405r}{R}\right)\cos\left(\frac{\pi z}{H}\right)$	$3.63P/VE_R\Sigma_f$	3.64
Sphere	Radius R	$\left(\frac{\pi}{R}\right)^2$	$A\frac{1}{r}\sin\left(\frac{\pi r}{R}\right)$	$P/4R^2E_R\Sigma_f$	3.29

Flux Distribution in Reflected Reactor



Flux Flattening Produced by Differential Fuelling





Flux Flattening Produced by Adjuster Rods

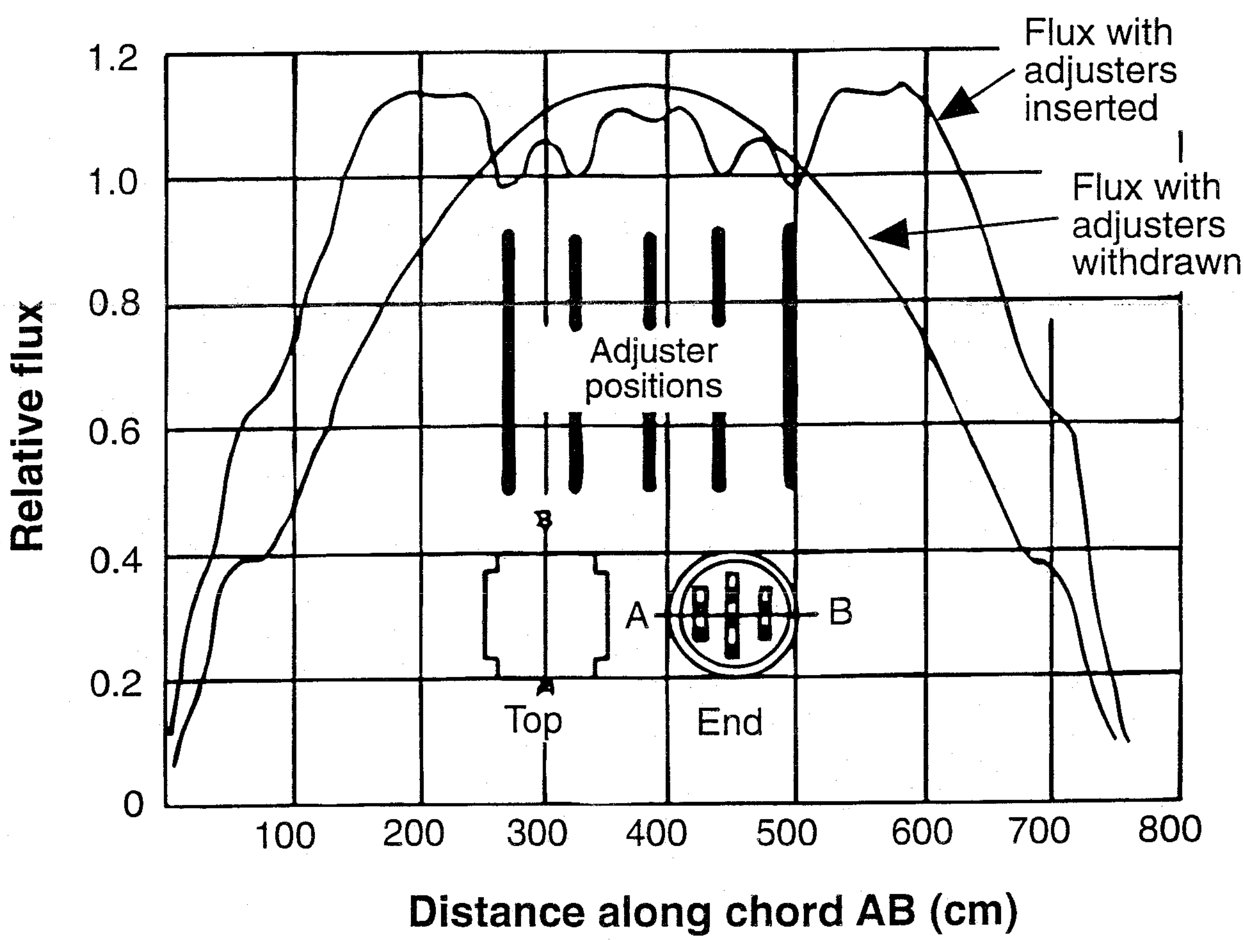


Fig. 6.8

Effect of Bi-Directional Refuelling in Flattening Axial Flux Shape

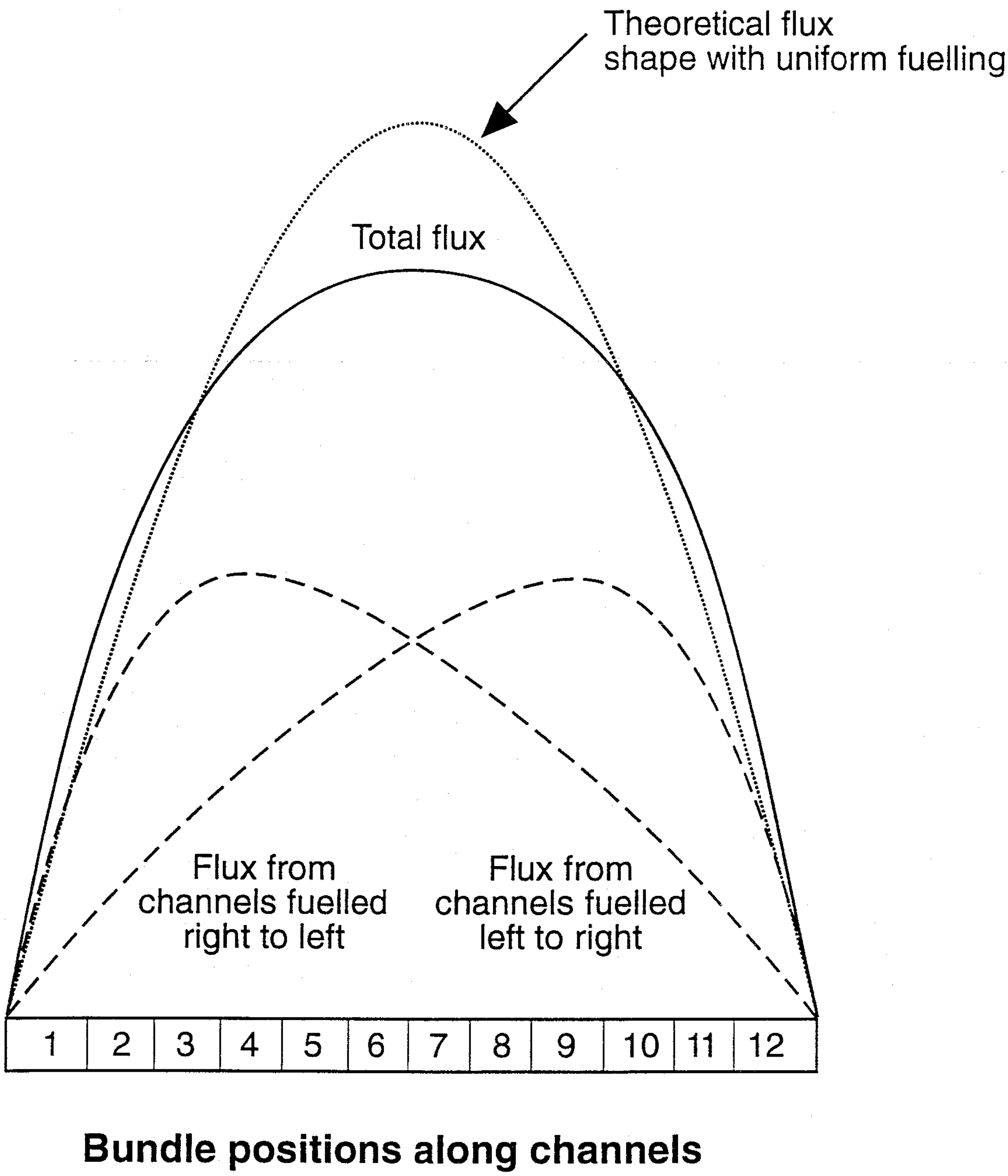
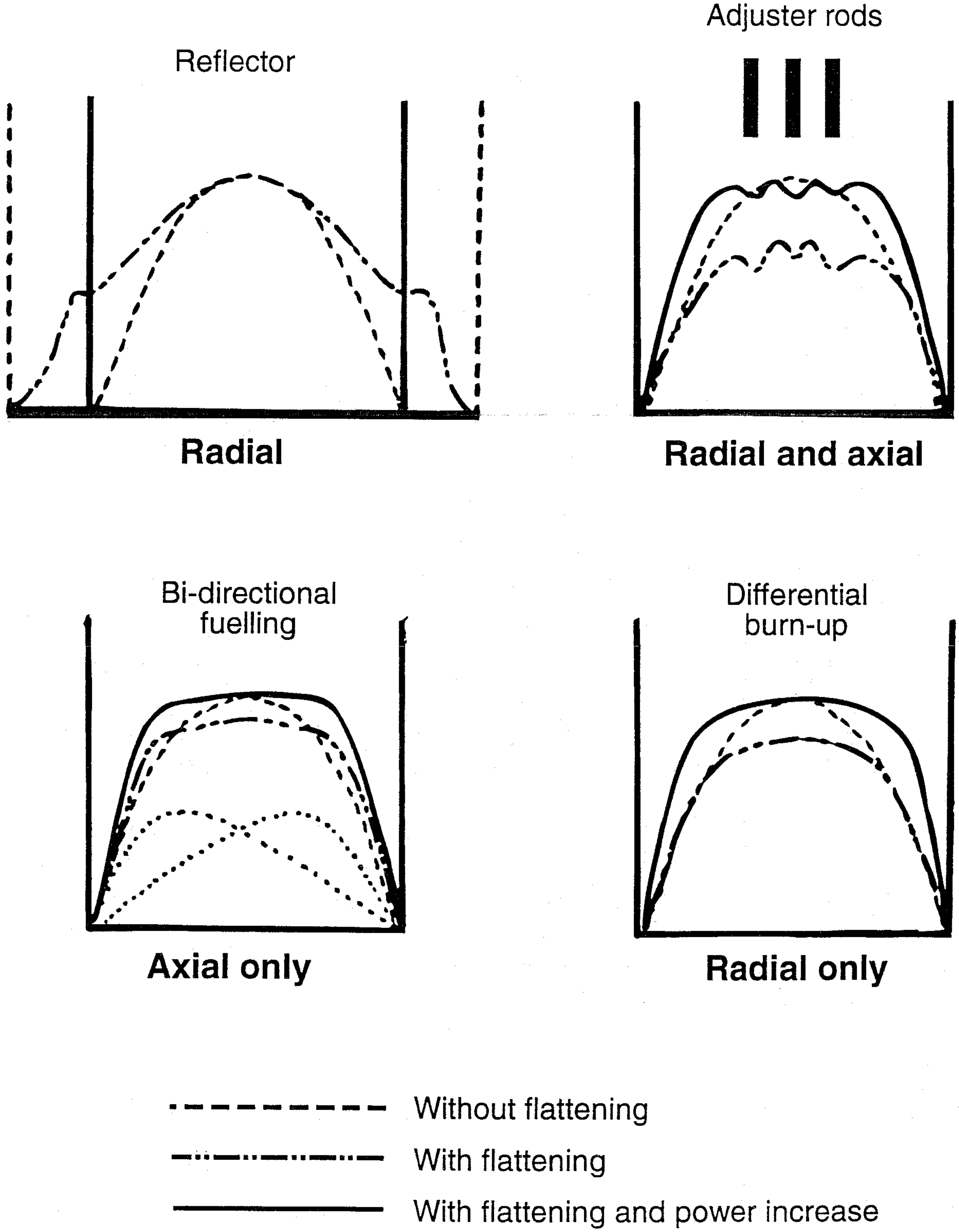


Fig. 6.7

Flux Flattening



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%