NUCL 402 Engineering of Nuclear Power Systems

Lecture 14: Reactor Shielding

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Sources of Radiation in Reactor

Prompt fission neutrons –Core – biological shield

Delayed fission neutron- energies 400 KeV

Prompt fission *γ*-rays- Core –largely attenuated by core materials

Fission product decay *γ*-rays- In fuel - continuing source after shutdown

Inelastic *γ*-rays- Emitted by excited nuclei –core and inner portion of shield

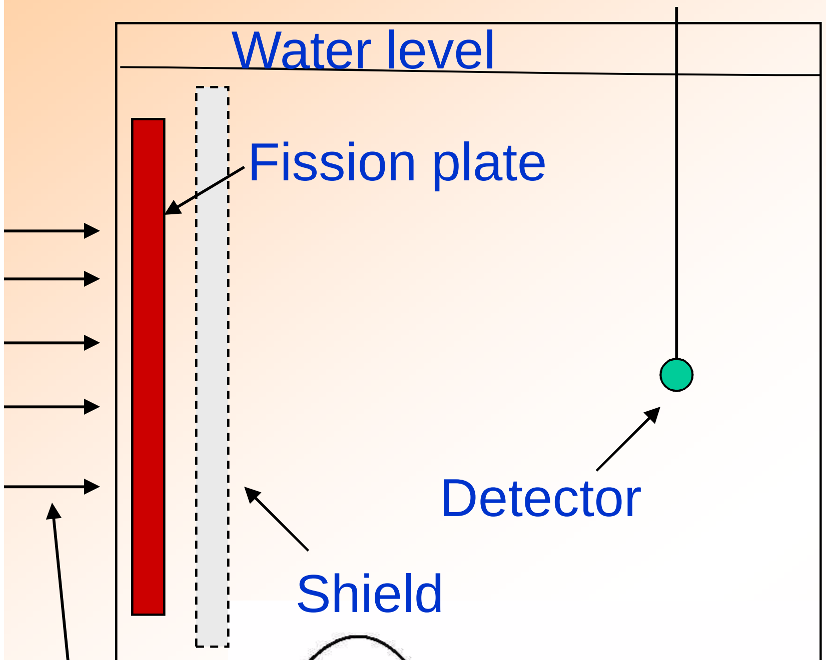
Capture *γ*-rays- With neutron absorption –source is often shield

Activation *γ*-rays- neutron absorption activation- reactor intrnal structure and coolant

Prompt neutrons are most difficult to attenuate – The shield is based on slowing down of neutrons by inelastic scattering with moderately heavy or heavy materials. The capture of thermalized neutrons is accompanied by *γ*-rays.

Removal Cross Section

The *point water kernel* G(r) flux at the distance r from a point source emitting isotropically one fission neutron per sec

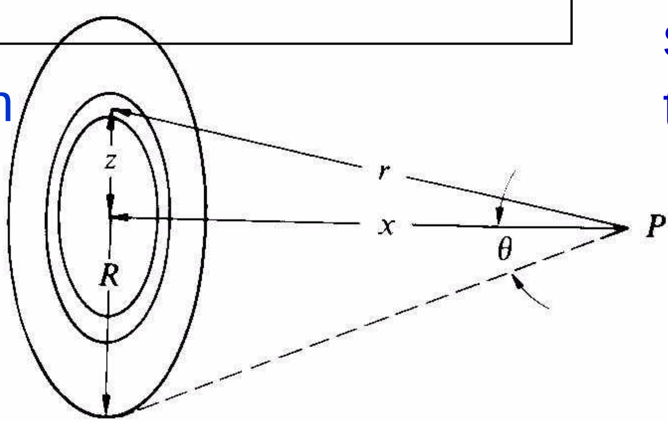


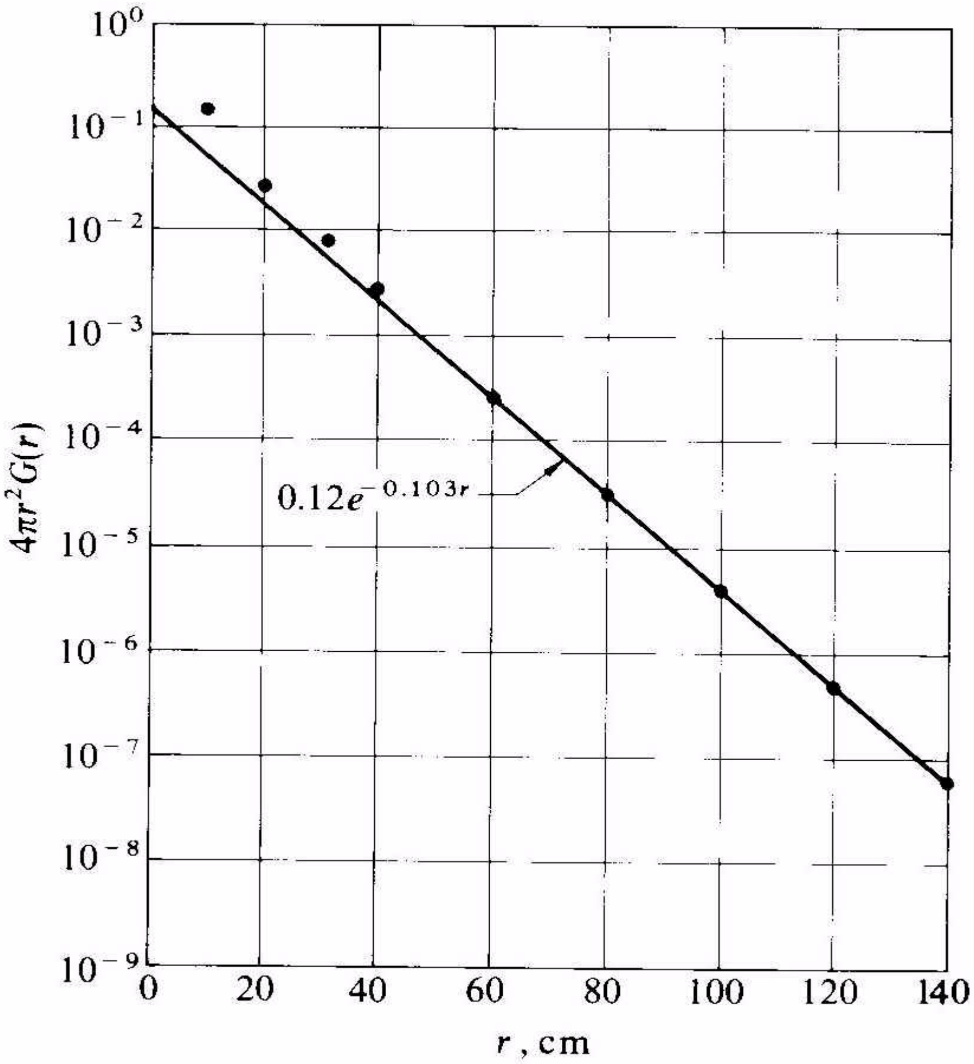
$$\phi(x) = 2\pi S \int_0^\infty G(r) z dz$$

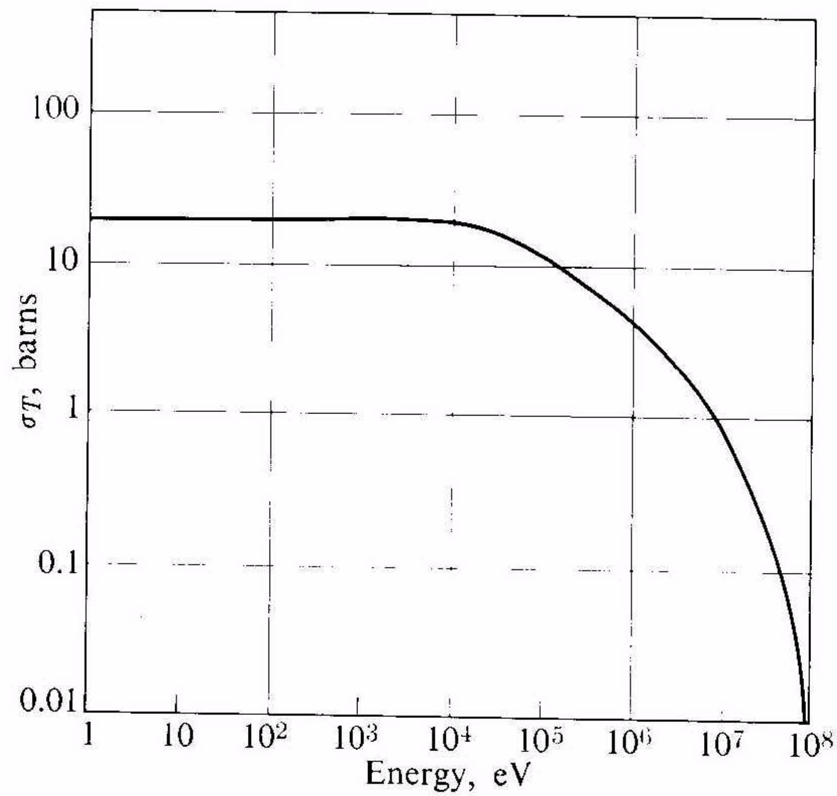
Inverting integral $$G(r) = \frac{Ae^{-\Sigma_{RW} r}}{4\pi r^2},$$

A=0.12, Σ_{RW}=0.103cm^{-1} macroscopic removal cross section of water

The fast neutron flux with a point source *S* surrounded by *t* material thickness $$\phi(r) = SG(r)e^{-\Sigma_R t}$$







The microscopic cross section of hydrogen

Measured value of $4\pi r^2 G(r)$ for point source emitting one neutron per sec

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Removal cross sections

Material	Macroscopic cross section, cm^{-1}	Microscopic cross section, b
Hydrogen		1.00
Deuterium		0.92
Beryllium	0.132	1.07
Boron		0.97
Carbon	0.065	0.81
Oxygen		0.92
Sodium	0.032	1.26
Iron	0.168	1.98
Zirconium	0.101	2.36
Lead	0.118	3.53
Uranium	0.174	3.6
Water	0.103	
Heavy Water	0.092	
Concrete*	0.089	

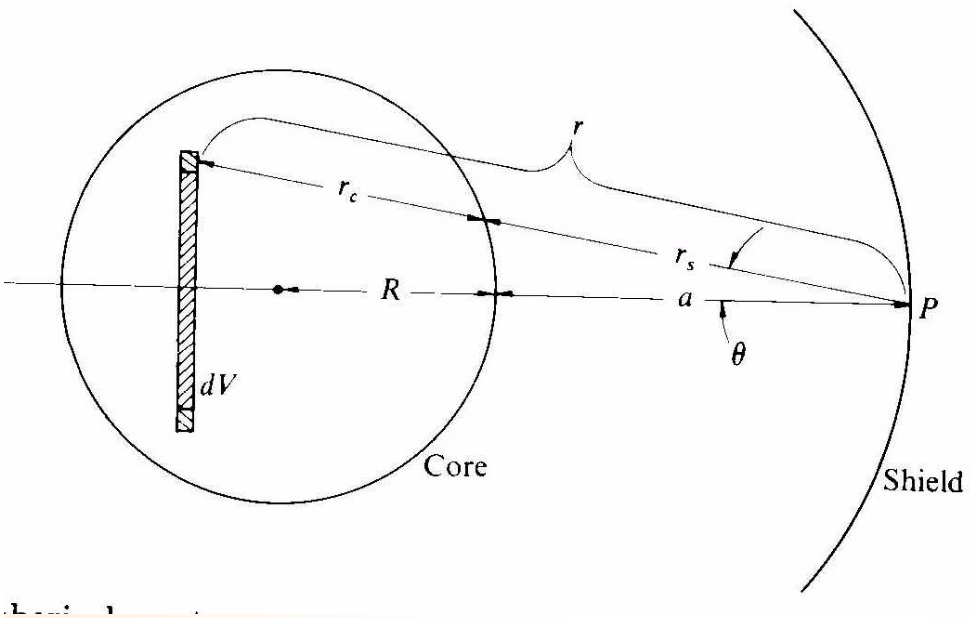
*Containing six percent water by weight.

$\Sigma_R = N\sigma_R$ N-atom density, σ_R microscopic removal cross section

$\Sigma_R = \Sigma_i N_i \sigma_{Ri}$ *i-th species*

Reactor Shield Design –Removal-Attenuation

Use of exponential kernel $G(r) = \dfrac{Ae^{-\Sigma_{RW} r}}{4\pi r^2},$



Fast neutron flux at point P

dV=2πr²sinθdθ

The flux at *P* due to *dV*

$$d\phi(P) = \frac{SAdV}{4\pi r^2} e^{-n(r)}$$

n(r) total number of removal mean free paths from dV to P.

$$n(r) = \alpha r + \beta r_s,\ \alpha = (1-f)\Sigma_{RW} + f\Sigma_{Rm},$$
$$\beta = \Sigma_{Rs} - (1-f)\Sigma_{RW} - f\Sigma_{Rm}$$

f -fraction of metal in core.

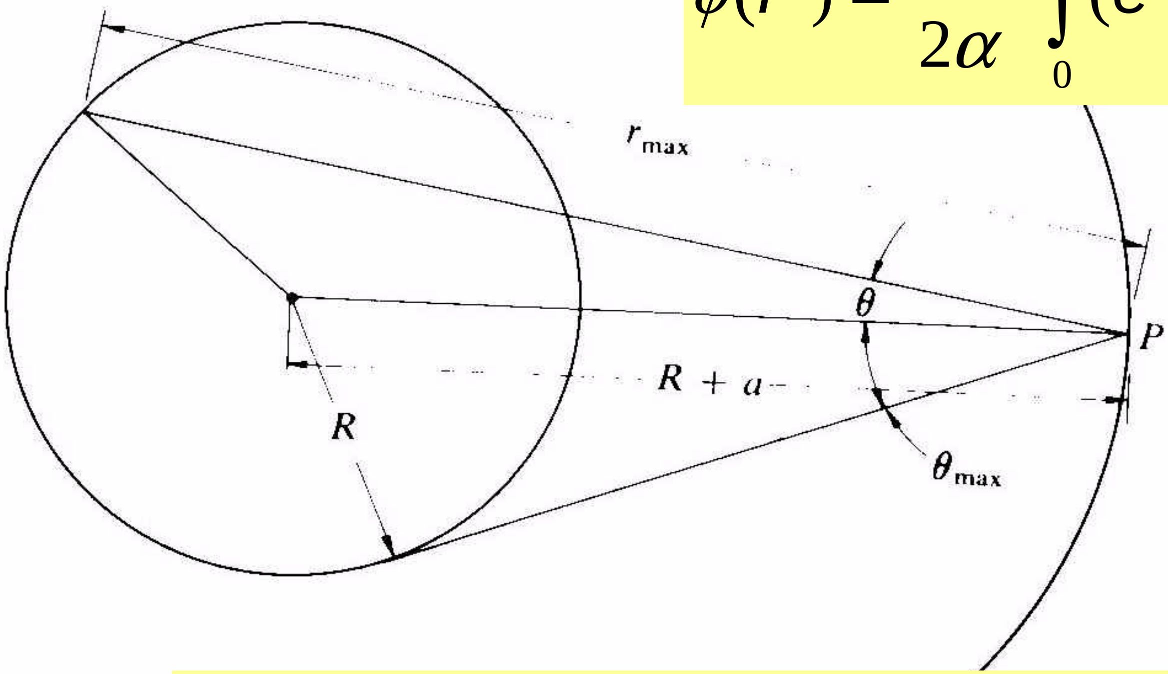
$$d\phi(P) = \frac{SAdV}{4\pi r^2} e^{-(\alpha r + \beta r_s)}$$

Integrating over core

$$\phi(P)=\frac{SA}{2}\int_0^{\theta_{\max}}\sin\theta d\theta\int_{r_s}^{r_{\max}}e^{-(\alpha r+\beta r_s)}dr$$

$$\theta_{\max}=\sin^{-1}\left(\frac{R}{R+a}\right),r_{\max}=(R+a)\cos\theta+\sqrt{R^2-(R+a)^2\sin^2\theta}$$

$$\phi(P)=\frac{SA}{2\alpha}\int_0^{\theta_{\max}}(e^{-\alpha r_s}-e^{-\alpha r_{\max}})e^{-\beta r_s}\sin\theta d\theta$$



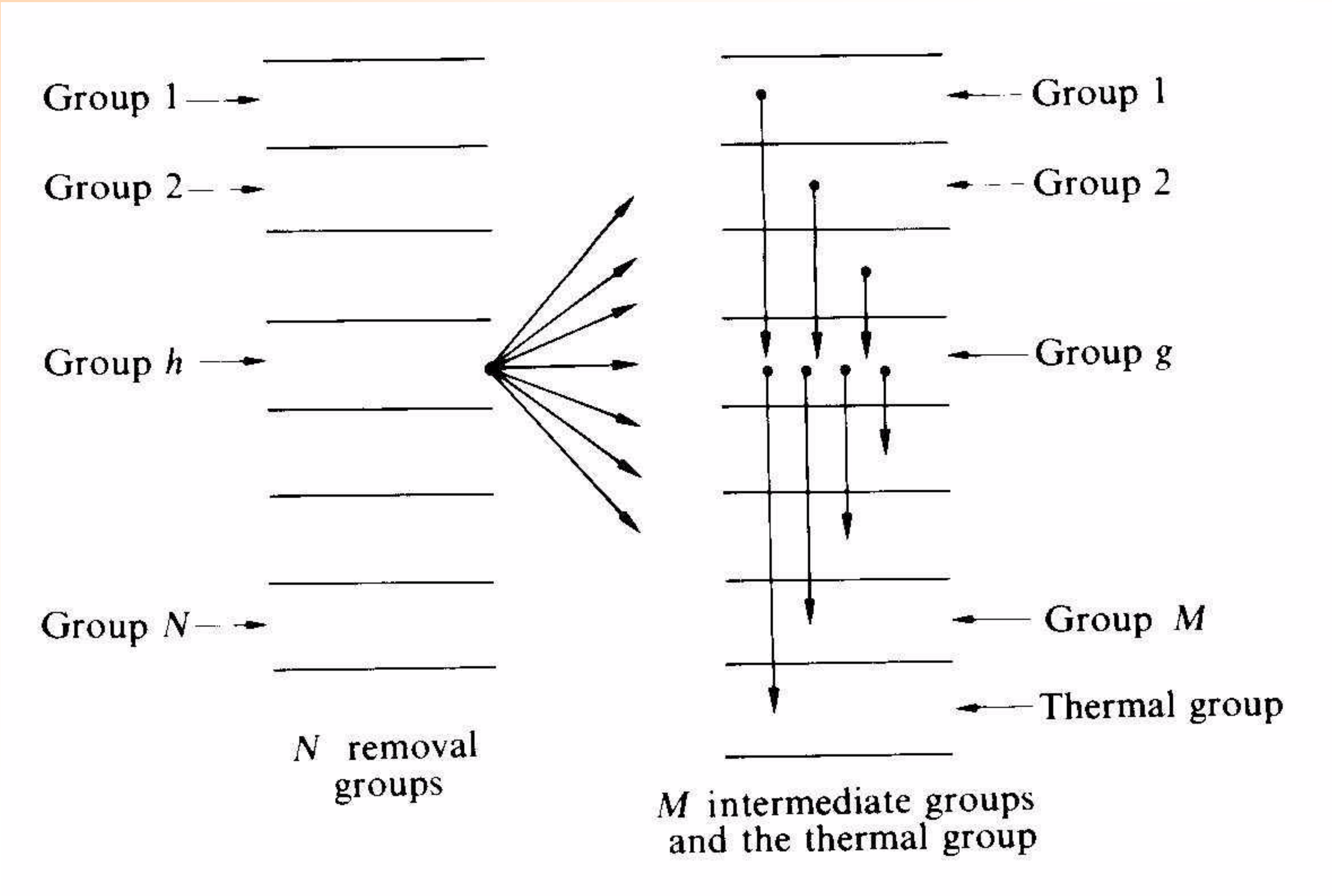
For small core and thick shield, *$\sin\theta \approx \theta$, $\theta_{mac} \approx R/(R+a)$, $r_s \approx a$*

$r_{max} \approx 2R+a$

$$\phi(P)=\frac{SA}{4\alpha}\left(\frac{R}{R+a}\right)^2e^{-\Sigma_{Rs}a}(1-e^{-2\alpha R})$$

Exact Method

Need detailed calculation of the multi-group calculation in addition to the removal-attenuation method. Requires spatial distribution of all energies.



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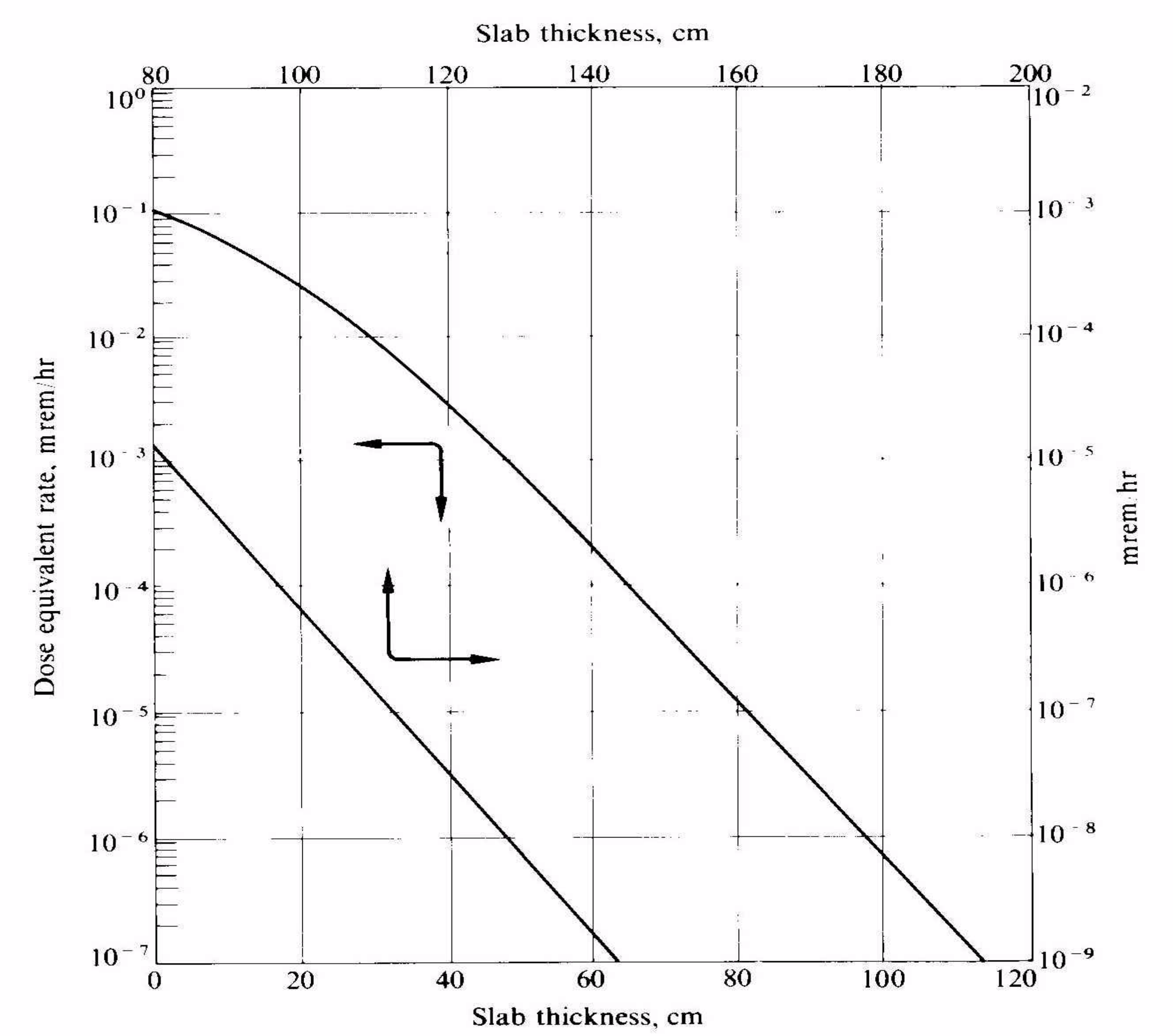
Exact Method

Transport Equation- detailed calculation involving particle transport equations-deterministic method

Monte Carlo Method- track life histories of the particles they move in shield from the point they enter the shield to their absorption or pass through the shield. Collision probability distribution functions are sampled in random manner. Then with large samples the particles that have collision, type of collision in the shield and those penetrate the shield can be established.

- Dose equivalent rate per unit intensity of 2-MeV neutrons incident on concrete slabs



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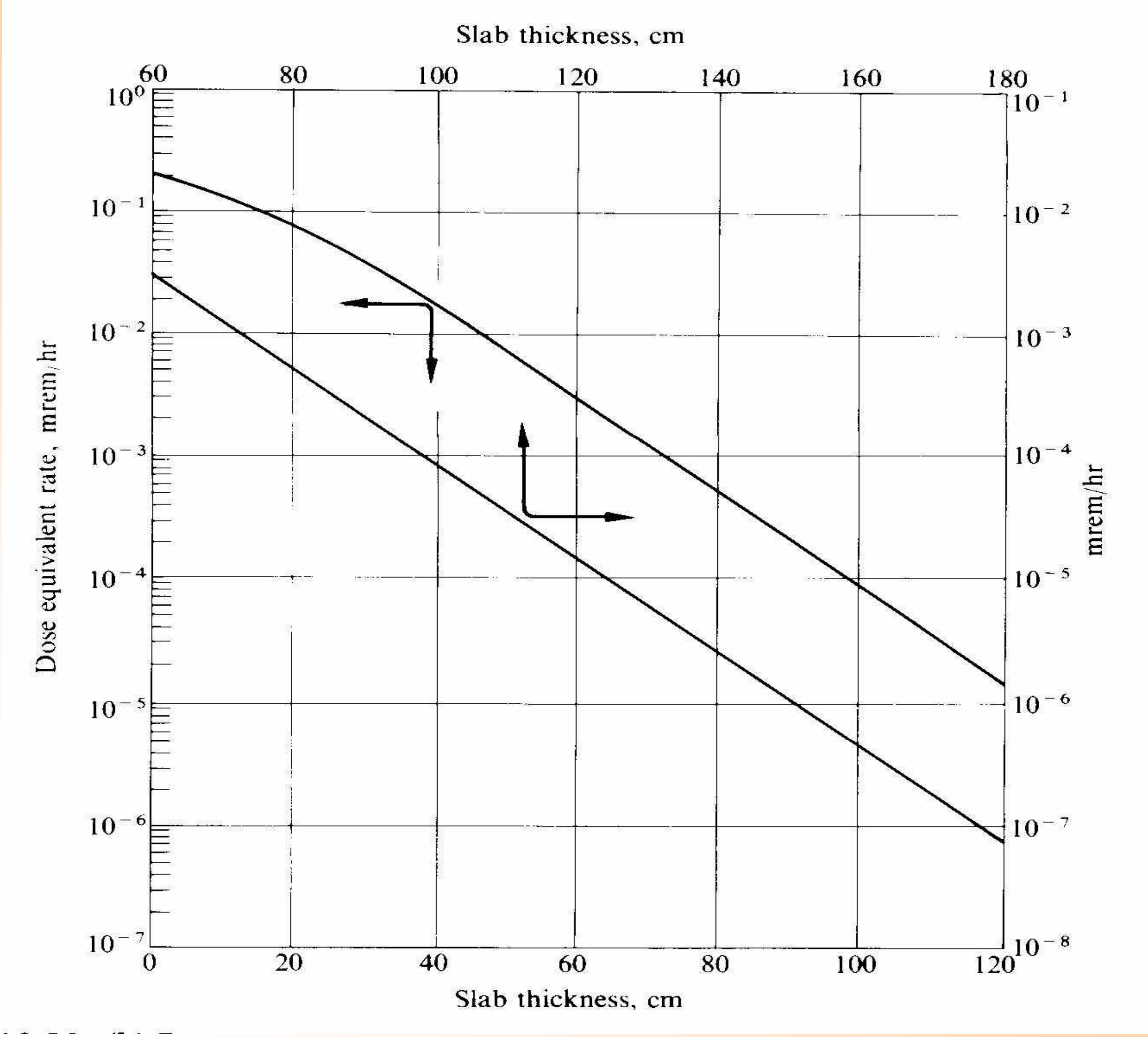
✓ Dose equivalent rate per unit intensity of 14-MeV neutrons incident on concrete slabs



Shielding γ –rays

Most of γ rays from reactor - prompt fission γ, fission product γ, radiative capture and inelastic scattering γ

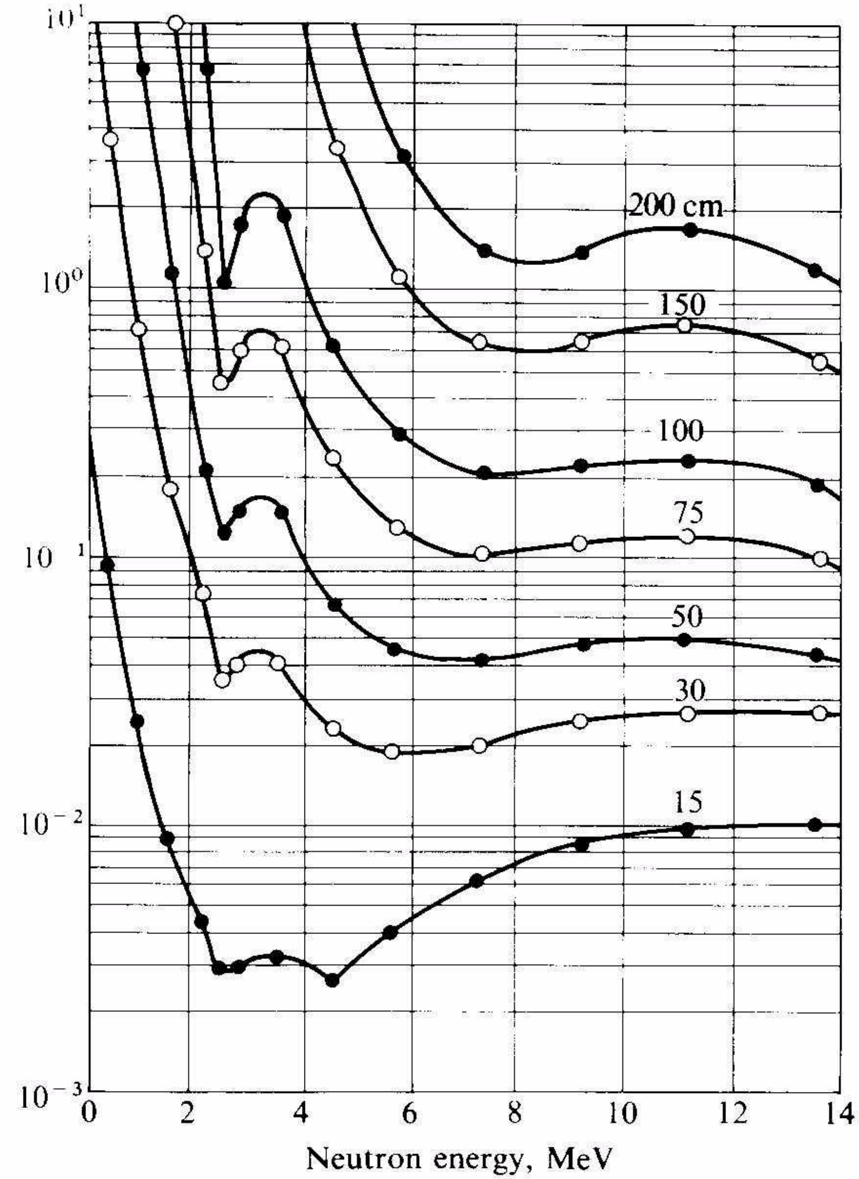
Prompt fission γ

Numbers of prompt fission γ-rays and fission product decay γ-rays emitted per fission

Group number	Energy interval, MeV	Prompt (χ_{pn})	Decay (χ_{dn})	Total
1	0–1	5.2	3.2	8.4
2	1–3	1.8	1.5	3.3
3	3–5	0.22	0.18	0.40
4	5–7	0.025	0.021	0.046

γ-Rays from thermal neutron capture in several elements

Target nucleus	Photons per 100 captures					
	0-1 MeV	1-3 MeV	3-5 MeV	5-7 MeV	7-9 MeV	>9 MeV
H	0	100	0	0	0	0
D	0	0	0	100	0	0
C	0	0	100	0	0	0
Na	>96	314	70	31	0	0
Al	>236	264	62	19	19	0
Si	>100	93	89	11	4.1	0.1
Fe	>75	87	23	25	38	2.1
Pb	0	0	0	7	93	0
U	254	269	34	0	0	4



Ratio of γ dose to neutron dose for neutrons incident on slabs of concrete, based on Q=10 for neutrons.

Coolant activation

$$\alpha = \frac{\Sigma_{act}\,\phi_{av}\,(1-e^{-\lambda t_i})}{1-e^{-\lambda(t_i-t_0)}}$$

where t_i time in the reactor flux, t_0 time in the outer circuit.

Activation reactions in coolants

Reaction	Cross section, b	Half-life	Energy of radiation, MeV
$^{16}O(n,p)^{16}N$	1.9×10^{-5}*	7.1 s	6.13, 7.12 (γ-rays)
$^{17}O(n,p)^{17}N$	5.2×10^{-6}*	4.14 s	1.2, 0.43 (neutrons)
$^{18}O(n,\gamma)^{19}O$	2.1×10^{-4}†	29 s	0.20, 1.36 (γ-rays)
$^{23}Na(n,\gamma)^{24}Na$	0.53†	15.0 h	2.75, 1.37 (γ-rays)
$^{40}Ar(n,\gamma)^{41}Ar$	0.66†	1.83 h	1.29 (γ-ray)

*Average cross section over fission spectrum; see text.
†Thermal (0.0253 eV) cross section.

Ducts in Shield

Radiation streaming