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Fast Reactor Core Design

Module 2 : Core Neutronics

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IAEA/ICTP School on Physics and Technology of Fast Reactor Systems

Lectures on Fast Reactor Core Design

Module 2 : Core Neutronics

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2.0 INTRODUCTION

In a fast reactor, the neutrons have high energy in the core centre. But outside the core in the shield regions they get slowed down to low energies. Thus knowledge of basic neutron cross-sections accurately in the entire energy range of neutrons is imperative. Compared to thermal neutron reactors it is essential to use much larger number of energy groups in the core design of FBR as the processes of neutron slowing down and capture in the fast and resonance energy regions need to be modelled by international collaboration between different countries, measurements and evaluation of huge nuclear data bases have been done through IAEA co-ordination.

The objective of the core design is safe operation, economic power generation and self sufficient fissile material production. This requires the design of core with highest acceptable linear pin power, core burnup capability, suitable fuel pin diameter and the most suitable safety related parameters. Selection of fuel pin diameter affects the fissile inventory, fuel cycle cost and breeding gain. The core height is to be decided based on the coolant pressure drop requirement. To achieve better breeding ratio, the reactor can be designed with radial and axial blankets.

Once the volume fractions of the fuel, coolant and the structural materials are determined from the above considerations, the core physics design starts with the determination of the fuel enrichment necessary to keep the reactor critical at full power with sufficient excess reactivity for burnup compensation. Distribution of neutron flux and power throughout the core, breeding ratio achievable for the fuel/blanket configuration, refuelling scheme for the core and blankets to ensure maximum availability of the reactor and highest possible burnup for each of the fuel and blanket subassembly, safety related parameters like material worth, Doppler reactivity, delayed neutron fractions etc. are the outputs of the physics design studies. Optimization of power production in core is essential for economic power generation. Precise balance need to be maintained between power produced in different locations of core. This needs to be achieved with every refuelling. This is possible only if the fuel isotopic compositions in different fuel and blanket subassemblies are reliably estimated as a function of burnup which is a challenging task. The core design also ensures that the core temperature and power coefficients of reactivity are negative so that any off normal increase in temperature or power leads to a reduction in reactivity and power. The fuel management scheme is decided based on the targeted peak burnup for which the fuel pins are designed. Another important part of the physics design is the determination of the number and layout of the absorber rods (AR) for reactivity control during normal operation as well as in accident situations. Adequate worth has to be provided for absorber rods for safe shut down of reactor in all eventualities. Their locations and B-10 enrichment need to be optimized. For achieving reliability targets two or more independent shutdown systems are normally deployed in power reactors. The interaction effect between the absorber rods of the shutdown systems needs to be studied carefully.

The basic physics concepts covered in this lecture are: (i) breeding (ii) doubling time (iii) reactivity effects (iv) reactor control and (v) core shielding aspects. Detailed coverage of reactor kinetics aspects is beyond the scope of this lecture and it is covered in other lectures.

2.1 PHYSICS OF BREEDING

Breeding is the process of converting a fertile nuclide into a fissile nuclide. A neutron capture in fertile nuclide U^{238} followed by two β - emissions produces the fissile nuclide Pu^{239}. The breeding ratio (BR) is defined as the ratio of fissile material produced (FP) to the fissile material destroyed (FD) in a cycle either through fission or through capture.

$$BR = \frac{FP}{FD}$$

Fuel cycle is the time between periodic refuelling. When BR is less than unity, this ratio is known as conversion ratio (CR).

Breeding Ratio

The expression for the breeding ratio can be derived as

$$BR = \frac{\left[\nu - (1+\alpha) - A - \tilde{L} + F(\nu' - 1)\right]}{1+\alpha} \qquad (1)$$

where

ν	=	No. of neutrons emitted per fission in a fissile nuclide
ν'	=	No. of neutrons emitted per fission in a fertile nuclide
α	=	Capture to fission ratio of a fissile nuclide
F	=	No. of fertile nuclides fissioned per fission of a fissile nuclide
A	=	No. of neutrons absorbed in SS and sodium per fission in a fissile nuclide
$\tilde{L}$	=	No. of neutrons leaked from the blanket per fission of a fissile nuclide.

In Eq. 1, numerator gives the excess of neutrons that are available for capture in U-238 to produce Pu-239.

To derive the expression, the following notations used:

- Subscripts 'c' and *'f* refer capture and fission.
- *'fi'* means fissile ; *'fe'* means fertile and *'nf* means neither fissile nor fertile.
- N_{fi} and N_{fe} are the number densities of fissile and fertile nuclides respectively.

Let 'P' be the total rate of neutrons produced per unit volume. It can be written as

$$P = \left(\nu N_{fi} \sigma_f^{fi} + \nu' N_{fe} \sigma_f^{fe}\right)\phi \qquad (2)$$

In order to produce these neutrons, we must loose neutrons at the rate of L_1 per unit volume,

$$L_1 = \left(N_{fi} \sigma_f^{fi} + N_{fe} \sigma_f^{fe}\right)\phi \qquad (3)$$

Neutrons are lost by parasitic capture at the rate of L_2 per unit volume as

$$L_2 = \left(N_{nf} \sigma_c^{nf} + N_{fi} \sigma_c^{fi}\right)\phi \qquad (4)$$

If $\tilde{L}$ is the leakage rate from the blanket per fission of a fissile nuclide, total neutron leakage from the blanket is

$$L_3 = \tilde{L}\left(N_{fi} \sigma_f^{fi}\right)\phi \qquad (5)$$

So, the net production of neutrons, **P,** per unit volume is

$$\mathbf{P} = P - L_1 - L_2 - L_3$$

It is to be noted that **P** neutrons are available for breeding. In other words, **P** is the rate at which new fissile nuclides are produced.

The destruction rate of fissile fuel per unit volume is

$$N_{fi}\left(\sigma_f^{fi}+\sigma_c^{fi}\right)\phi \tag{7}$$

So, the breeding ratio can be written as

$$BR = P/N_{fi}\left(\sigma_f^{fi}+\sigma_c^{fi}\right)\phi \tag{8}$$

$$BR = \frac{\left[\nu N_{fi}\sigma_f^{fi}+\nu' N_{fe}\sigma_f^{fe}-\left(N_{fi}\sigma_f^{fi}+N_{fe}\sigma_f^{fe}\right)-\left(N_{nf}\sigma_c^{nf}+N_{fi}\sigma_c^{fi}\right)-\tilde{L}N_{fi}\sigma_f^{fi}\right]\phi}{N_{fi}\left(\sigma_f^{fi}+\sigma_c^{fi}\right)\phi} \tag{9}$$

Dividing numerator and denominator by $N_{fi}\sigma_f^{fi}$, we get

$$BR = \frac{\left[\nu-(1+\alpha)-A-\tilde{L}+F\left(\nu'-1\right)\right]}{1+\alpha} \tag{10}$$

where

$$\alpha=\frac{\sigma_c^{fi}}{\sigma_f^{fi}};\quad A=\frac{N_{nf}\sigma_c^{nf}}{N_{fi}\sigma_f^{fi}}\quad;\quad F=\frac{N_{fe}\sigma_f^{fe}}{N_{fi}\sigma_f^{fi}}\quad;\qquad \eta=\frac{\nu\sigma_f^{fi}}{\sigma_f^{fi}+\sigma_c^{fi}}$$

Eq. 10 can be written as

$$BR=(\eta-1)+\frac{\left[F\left(\nu'-1\right)-A-\tilde{L}\right]}{1+\alpha} \tag{11}$$

It is to be noted that the nuclear constants, ν , ν', α, A and F depends on the energy spectrum. Leakage factor $\tilde{L}$ depends on both the geometry and the materials present in core and blanket. The contribution to BR from the fast fission of fertile nuclide is given by the term F(ν' -1). Among the fertile nuclides, Pu-240 provides a much greater fast fission bonus than U-238 due to its low fission threshold and high value of σ_f . The fast fission bonus from Th-232 is very small.

A reasonable approximation of Eq. 11 is to ignore the last term. Thus,

$$BR = \eta - 1 \tag{12}$$

- Of the η neutrons produced, η-1 neutrons are available for breeding.
- BR > 1 implies that η - 1 should be > 2 or in other terms η > 2.
- Taking parasitic losses into account (L), η > 2 + L. This is the condition for breeding.

Breeding gain (G) is defined as G = BR-1

$$G=\frac{FEOC-FBOC}{FD} \quad = \quad \frac{FG}{FD},$$

where FG is the fissile material gained per cycle. FBOC is the fissile inventory in the reactor at the beginning of a cycle (i.e. directly after refuelling) and FEOC at the end of the cycle (i.e. when the reactor is shut down for refuelling).

Reasons for high BR in fast reactors

BR depends on the core design (fuel type, neutron spectrum, pin size etc.). In fast reactors, average energy of neutrons is high. All nuclear parameters favour high value of BR in a fast neutron spectrum as follows:

a) *v* increases with increasing neutron energy.

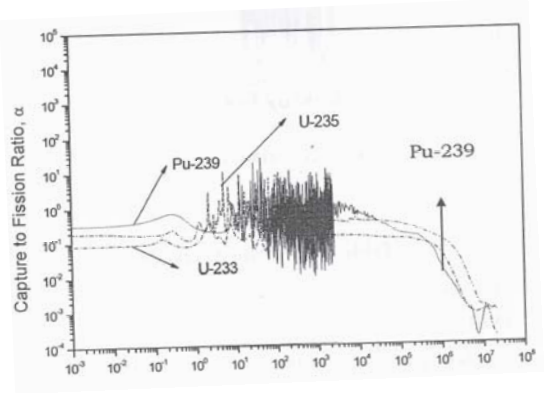
b) α for fissile nuclides decreases with increasing energy (Fig.2.1). Pu-239 has low value of α compared to U-235

c) η increases in fast spectrum (see Fig.2.1).

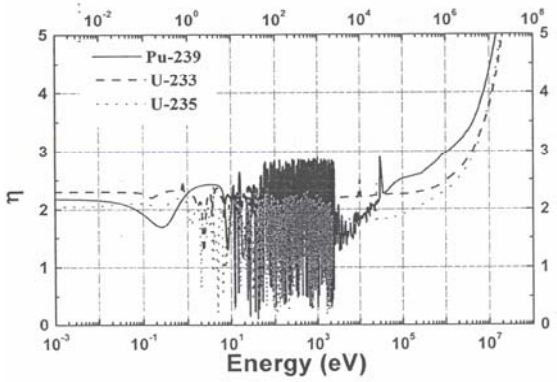
d) The values of F, the number of fertile atoms fissioned is higher in fast neutron spectrum (U-238 fission cross section is higher than Th-232).

e) The value of *A,* the number of neutrons lost by parasitic capture in structure and coolant per fission of fissile nuclide, is low in fast spectrum.

f) Though neutron leakage from the core will be very high (30-40 % in small cores), the leakage from the reactor is made very small by surrounding it with a thick blanket.



Energy variation of α



Energy variation of η

Nuclide	α
U-235	0.292
Pu-239	0.248
Pu-241	0.157

Average value of α in fast spectrum

Flux Spectrum	Pu-239	U-235	U-233
LWR	2.04	2.06	2.26
SFR (oxide)	2.45	2.1	2.31

Spectrum averaged η

Fig 2.1 :Energy variation of α and η and their average values for fissile isotopes

The following conclusion can also be derived from the above;

- Although η is slightly greater than 2, breeding may not be possible after accounting loss from leakage. Hence, thermal breeders are not possible with Pu-239 or U-235.
- It is possible to construct thermal breeder with U-233.
- Pu-239 is a better fuel for fast reactors than U-235 because of its breeding potential

2.2 BREEDING POTENTIAL OF VARIOUS FUELS

The maximum breeding possible in fast reactor with oxide, carbide and metal fuels is given in table 2.1.

Table 2.1 : Breeding Potential of Oxide, Carbide and metal fuelled FBRs

Fuel	Breeding Ratio Potential
Oxide	1.15 - 1.30
Carbide	1.30 - 1.40
Metal	1.50 - 1.65

2.3 EXTERNAL VERSUS INTERNAL BREEDING

In the external breeding concept, all fertile material is placed outside the core in the blanket region. In the internal breeding concept, the core would consist of both fissile and fertile materials. Most of the current fast reactors have adopted internal breeding concept. The core with this concept can either be homogeneous or heterogeneous depending on the desirable breeding. Heterogeneous concept gives high breeding ratio and less sodium void reactivity but requires very high fissile fuel inventory. The external and internal breeding concepts are compared in the table given below.

Table 2.2 : Comparison of external and internal breeding

External breeding (hard spectrum)	**Internal breeding (soft spectrum)**
Few fast fissions in ^{238}U	Contribution of fast fissions in ^{238}U bigger
Delayed neutron fraction β is small	β larger (contribution of ^{238}U)
Small Doppler effect on reactivity	Great Doppler effect
Frequent fuel change	Long burnup; less frequent reloading
Low rating of fissile	High rating of fissile material
Small burnup	High burnup
Low total power	High total power possible
Blanket thick	Blanket thin
Stronger absorbers in core possible	Strong absorbers in core are disturbing
Low Pu inventory	High Pu inventory
Higher breeding gain	Lower breeding gain
High enrichment	Low enrichment

2.4 DOUBLING TIME

Another important parameter used in the physics of fast reactors is the doubling time. It is the time required by a breeder reactor to produce enough fissile material in excess of its

own fissile inventory to fuel an identical reactor. In other words, it is the time required to produce double the initial inventory of fissile fuel. It is an index of merit which is used to compare different reactor designs, fuels and fuel cycle systems involving several breeder reactors. It can be defined in many different ways. Though conceptually it is simple, it involves several aspects in making the definition quite varied from others.

Wyckoff and Greebler attempted to standardize the doubling time concepts (Ref: Nuclear Technology, 21, 1974, pp. 158-164) as given below.

RDT = Reactor Doubling Time
= time required for a specific reactor to produce enough fissile material in excess of its own fissile inventory to fuel a new, identical reactor.

SDT = System Doubling Time
= time required for a specific reactor to produce enough fissile material in excess of its own fissile inventory to supply its requirement of fissile material external to the reactor and to fuel a new, identical reactor.

CSDT = Compound System Doubling Time
= time required for a system of identical breeder reactors to double the fissile material in the system, assuming that the number of reactors is increasing at a rate such that all of the fissile material is being utilized.

Reactor Doubling Time (RDT)

This is the time required for a particular breeder reactor to produce enough fissile material in excess of its own fissile inventory to fuel an identical reactor. Hence it is the time necessary to double the initial load of fissile fuel.

If M_0 (kg) – Initial fissile inventory

M_g (kg/y) – Fissile material gained during one year

Then RDT = (M_0/M_g)

Exm: If $M_g = 0.1\ M_0$ and every year this $0.1\ M_0$ is set aside (since only M_0 is needed in the reactor), then after 10 years there would be $2\ M_0$ of fissile material – M_0 still in the reactor and M_0 set aside.

Although the equation for RDT is simple, an accurate calculation of M_g is not. If in a system G = Breeding gain, P = Power (in MW) and f = fraction of time the reactor is at rated power

M_g = G.(fissile mass destroyed/year)
$\cong$ G.(1+α).(fissile mass fissioned/year)

$$M_g \cong G.(1+\alpha)\left[\frac{\left(P\times 10^6\right)\left(2.93\times 10^{10}\,\text{fissions/W.s}\right)\times\left(3.15\times 10^7\,\text{s/y}\times f\times 239\,\text{kg/kg-mol}\right)}{6.02\times 10^{26}\,\text{atoms/kg-mol}}\right]$$

$$\cong \frac{GPf(1+\alpha)}{2.7}$$

Hence, $\mathrm{RDT} \cong \dfrac{2.7\mathrm{M}_0}{\mathrm{GPf}(1+\alpha)}$ in years

It can be seen that RDT is proportional to the fissile specific inventory (M_0/P) and is inversely proportional to the breeding gain. For example an increase in BR from 1.2 to 1.4 results in a factor of two reduction in doubling time.

We can define RDT in other words as, $\mathrm{RDT} = \dfrac{\mathrm{FBOC}}{(\mathrm{FG}).\mathrm{C}}$ where C is the fuel cycles per year.

Where, FBOC - fissile material in core and blanket at the beginning of the fuel cycle
FG - fissile material gained per cycle
C -number of cycles per year ($1/t_c - t_c$ in year)

*However, d*oubling time depends on the (i) the relative amount of fissile material in the reactor and the other parts of the fuel cycle, e.g., storage, reprocessing and fabrication (ii) the manner in which the fissile material bred in one reactor is used to generate more fissile material. This gives rise to other definitions of doubling time.

System Doubling Time (SDT)

RDT does not take into account for the processes in the fuel cycle external to the reactor. These processes are part of the breeder reactor system which are as follows.

(i) time required for external processing also affects the doubling time.
(ii) losses – (a) loss of material in fabrication and reprocessing (b) loss of ^{241}Pu from β^- decay (in U-Pu cycle)

Out-of-reactor fissile inventory is normally in the range of 60 % of the in-reactor inventory which include;

(i) fuel that is being fabricated
(ii) being shipped to the plant
(iii) in storage at the plant awaiting loading or in cool-down period
(iv) in shipment for reprocessing
(v) being reprocessed

By denoting the fissile inventory in these processes as fissile external (FE), the effect of this inventory is accounted for through a factor (the ex-reactor factor) defined as

$$\mathrm{EF} = \left(\frac{\mathrm{FBOC} + \mathrm{FE}}{\mathrm{FBOC}}\right) \approx 1.6$$

The *processing losses* FPL, are usually based on *processing loss fraction*, PLF which is the fraction of material lost in fabrication and reprocessing operations; this fraction may range from 1% to a few percent. With *refueling fraction* RF, being defined as the fraction of core fuel assemblies replaced at each refueling, the processing losses per cycle are,

$$\mathrm{FPL} = \frac{1}{2}(\mathrm{FBOC} + \mathrm{FEOC})(\mathrm{RF})(\mathrm{PLF})$$

Where FEOC - *fissile material in core and blanket at the end of the fuel cycle*

The ^{241}Pu decay external to the reactor during one fuel cycle is

$$^{241}\text{Pu ED} = \frac{1}{2}\left(^{241}\text{Pu BOC} + ^{241}\text{Pu EOC}\right)(\text{EF} - 1)\left(1 - e^{-\lambda_4 t_c}\right)$$

Define all losses external to the reactor per cycle as

$$\text{FL} = \text{FPL} + ^{241}\text{Pu ED}$$

$$\text{SDT} = \frac{(\text{FBOC})(\text{EF})}{(\text{FG} - \text{FL})(\text{C})}$$

Compound System Doubling Time

A compound or exponential doubling time assumes that the fissile material produced in a given cycle is continuously re-invested in an identical breeder. It is meaningful only for a power reactor network containing a number of breeder reactors with similar characteristics.

$$\frac{dM}{dt} = \lambda M$$

M = fissile mass in the system as a function of time (kg)

λ = fractional increase in fissile mass in the system per unit time (y^{-1})

$$\lambda = \frac{(\text{FG} - \text{FL})\text{C}}{(\text{FBOC})(\text{EF})} \equiv \left(\frac{1}{\text{SDT}}\right)$$

$$\left(\frac{M}{M_0}\right) = e^{t/\text{SDT}}$$

M_0 = initial fissile mass in the system of reactors

[M_0 could also be thought of as the number of reactors in the system at time zero × (FBOC) × EF]

The compound system doubling time is the time required for (M/M_0) to become equal to 2.

CSDT = SDT ln2 = 0.693 (SDT)

Summary of a study to estimate the breeding and the doubling time for oxide, carbide, metal fuelled FBRs with various fuel cycles is given below in Table 2.3.

Table 2.3 : BR and DT for various fuels

Cycle		*Pu-U*	*Pu fueled-Th*	*^{233}U/^{238}U-Th*	*^{233}U-Th*
Core Fissile/Fertile		*LWR Pu/Dep. U*	*LWR Pu/Th*	*^{233}U/^{238}U*	*^{233}U/Th*
Blanket Fertile		*Dep. U*	*Th*	*Th*	*Th*
Oxide	RDT (y)	16	29	23	112
	BR	1.28	1.20	1.16	1.041
Carbide	RDT (y)	9	20	15	91
	BR	1.42	1.23	1.23	1.044
Metal	RDT (y)	6	12	12	43
	BR	1.63	1.38	1.30	1.11

Capacity factor – 75 % assumed

2.5 REACTIVITY EFFECTS

In nuclear reactors the variation of power affects the temperature of the fuel, structural materials and coolant, which in turn affects the reactivity and hence power. This again leads to change in temperature. Thus there is a feedback loop in the reactor. Since reactivity is a measure of neutron multiplication and hence determines the power, the feedback effects of temperature changes are best studied by the reactivity coefficients.

The reasons for reactivity changes in a fast reactor are similar to those in thermal reactors. Major reactivity additions in the fast reactors are caused by the following.

(i) temperature changes

(ii) control rod ejection

(iii) core compaction

(iv) coolant voiding

Temperature variations in a reactor core will affect core multiplication, both because of the resulting density changes in core components (due to expansion of fuel, clad, coolant, absorber rod drive, grid plate and vessel) that change macroscopic cross sections and because of a change in the thermal motion of core nuclei, which changes microscopic cross sections (i.e., Doppler effect).

In fast reactors, the neutron energy spectrum includes the resonance regions of both the fissionable (^{235}U, ^{233}U, ^{239}Pu, ^{241}Pu) and the non-fissionable (^{232}Th, ^{238}U, ^{240}Pu) fuel isotopes. The Doppler effect in fast reactors is due almost entirely to resonances below about 25 keV. An increase in fuel temperature will produce an increase in both fission and absorption cross-sections, and resulting changes can be positive or negative, depending on the exact composition. Detailed design calculations, using methods benchmarked against critical experiments, indicate that in large reactors with high fertile to fissile ratio, the Doppler coefficient is sufficiently negative to provide a prompt shutdown mechanism in the event of excess fission heating of the fuel.

In a sodium cooled fast reactor, reactivity change that occurs when sodium is voided is a predominant phenomenon. The reactivity change that occurs when sodium is voided from fast reactor can be separated into leakage, absorption and spectral components. The neutron capture component is positive but small in magnitude. Thus the net sodium reactivity coefficient is dependent on leakage (which is negative) and the spectral shift component (which is positive). The spectral and capture components are normally largest in the center of the core, where the neutron flux and importance function are largest and the leakage component is normally largest in the outer part of the core, where the flux gradient is the largest. One therefore anticipates that the sign of the sodium reactivity coefficient for a specific reactor design will change from negative to positive as the reactor size is increased. Even this is true when the core composition is changed from MOX to metal. However the challenge of FBR core design is to reduce the sodium void reactivity to zero or negative value.

Other reactivity effects are due to the expansions of absorber rod drive and vessel. While the former is a negative reactivity effect the latter gives positive reactivity feedback. There is no large fission product poison effect (like xenon poison) in a fast reactor, due to their low absorption cross sections at high neutron energy. Each reactivity effect is described below.

2.5.1 Temperature Effects

Temperature affects the reactivity as follows :

- Nuclear Doppler Effect
- Expansion (fuel, coolant and structure)

Nuclear Doppler Effect

The atomic vibration of a target nucleus increases when the temperature of the system is raised. This results in change of relative velocity between target nucleus and the neutron and the effective interaction cross section of neutron is changed. The peak value of the cross section reduces and the cross section in the wings are increased (Fig.2.2). But, the total area under the resonance is conserved. This phenomenon is called the *Doppler broadening* of a resonance. The broadening of a resonance leads to higher average flux in the resonance region (due to energy self-shielding), whereas the flux variation is very less in the wings. The net result of this effect is that the reaction rate is increased. It is to be noted that the contribution from the wings only leads to higher reaction rate.

The broadening of capture resonances causes negative reactivity effects in a reactor. The effect from a fission resonance is to provide the positive reactivity. Doppler feedback is prompt and it is one of the inherent safety parameters. The magnitude of Doppler feedback in a reactor depends upon the following :

- Temperature change
- Atom densities of fissile and fertile nuclides.
- Neutron spectrum (Depends on fuel type and core size)

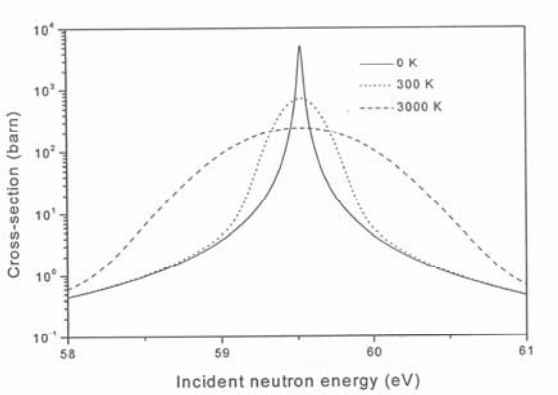


Fig.2.2 : Doppler Broadening of a Resonance

The net Doppler feedback from the fissile nuclides (Pu-239, U-235) is a balance between the contributions from capture and fission resonances. The net effect from Pu-239 is found to be negligible (slightly positive). Fertile nuclides (U-238, Pu-240) do not have fission resonances but they have capture resonances. Main contribution of negative Doppler feedback in a reactor is from the fertile nuclides. Fast reactor core can be designed such that Doppler feedback can counter act the positive sodium void effect. Following are the important observations :

- For pure fissile materials, net Doppler feed back could be slightly positive
- For large core with MOX fuel, Doppler feedback can be more negative.
- For small reactors, Doppler feedback is negligible due to hard spectrum, but it will be significant for higher size reactors.

Doppler Coefficient

It is the change in reactivity due to change in temperature and it has following dependence for oxide fuel :

$$\frac{dk}{dT} = \frac{K_D}{T}$$

Where K_D is the Doppler constant and T is the temperature in Kelvin. K_D is equal to

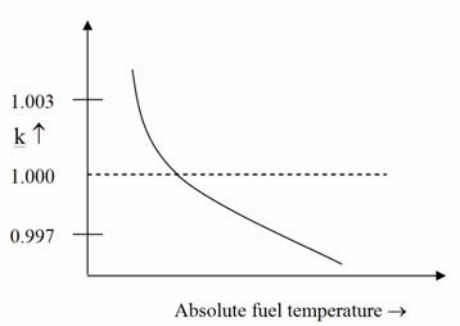


Fig.2.3 : Change in Keff with temp

$$K_D = T\frac{dk}{dT}$$

The reactivity loss for a temperature change from T_1 to T_2 K is (Fig 2.3)

$$\rho_D = K_D \int_{T_1}^{T_2} \frac{dT}{T}$$

$$= K_D \ln\left(\frac{T_2}{T_1}\right)$$

For various reactor spectrums, dk / dT has the following dependence as evidenced by using the resonance theory:

Metal fueled reactor (hard spectrum) : $\frac{dk}{dT} \propto \frac{1}{T^{3/2}}$

Oxide fuelled FBRs (medium spectrum) : $\left(\frac{dk}{dT}\right) \propto \frac{1}{T}$

LWR (softer spectrum) : $\frac{dk}{dT} \propto \frac{1}{T^{1/2}}$

It shows that the spectrum of the oxide fuelled fast reactor lies in between the hard spectrum and the soft spectrum. A prompt negative feedback is particularly important for fast reactors because of two mechanisms, fuel compaction and sodium loss, which have the potential to make the reactor super prompt critical. The Doppler effect provides this prompt negative feedback with ceramic fuel. It is important to note that, feedback response is prompt only when the fissile and fertile materials are mixed intimately together. This is warranted because most of the fission energy comes from the kinetic energy of the fission products and hence fertile isotopes, if they are very close to fissile isotopes, cause loss in the kinetic energy which in turn leads to increase in temperature and thermal motion of the fertile nuclei thereby causing large Doppler effect.

Further, Doppler effect comes predominantly from the capture of low energy neutrons. Energy self-shielding is required for an increase in the temperature of the fertile material to increase the effective cross-section of that material. At high energies there is little self-shielding. Ceramic-fueled reactors, due to the presence of oxygen or carbon in the fuel, have a soft enough neutron spectrum to have a large Doppler effect.

Dimensional Changes Effects

Core might undergo dimensional changes either during normal operation or during transients. When temperature increases, fuel will expand axially. The structures supporting the core will expand radially as the temperature increases. During fuel melting in a severe accident, fuel might slump down causing core compaction or it may be expelled causing fuel expansion. All these changes would induce reactivity changes which have to be understood.

Detailed analysis taking into all factors including leakage etc show that, when core is getting compacted, there will be an increase in the reactivity. Similarly, when the fuel temperature increases, the fuel expands axially. The radial fuel expansion will have little impact on the core radial expansion, as it is primarily governed by the structures of the core

restraint system. Hence, the primary core expansion resulting from temperature increase would be in the axial direction. The analysis has shown that axial leakage does not change much with fuel axial expansion, radial leakage increases thereby producing a negative reactivity effect. The role of axial expansion of fuel in the normal solid fuel pin geometry is to provide a prompt negative reactivity feedback at the start of a power transient. This mechanism is the principal prompt negative feedback available in a metal fuelled fast reactor.

In a metal fuelled fast reactor where the spectrum is hard, the axial expansion provide the much wanted negative reactivity feedback. Whereas, in the oxide fuelled fast reactors, the axial expansion effect is very less due to the fact that the fuel cracks and due to lack of structural integrity, axial expansion mechanism is somewhat absent or unreliable and hence importance is given to Doppler effect in oxide fuelled reactors in order to get prompt negative feedback of reactivity.

2.5.2 Isothermal Temperature Coefficient

Isothermal temperature coefficient of reactivity, $\rho_T = \frac{\Delta_\rho}{\Delta T}$, considers change in reactivity when the core temperature is raised or lowered uniformly (T is same throughout the core). It has contributions from (i) Doppler effects and (ii) Expansion effects

It may be noted that for small reactors, there will be negligible contribution from Doppler and major contribution will be from expansion effects. It is also to be noted that there will be less Doppler feedback for metal fuelled reactors (spectrum hard) but large negative contribution from expansion effects.

2.5.3 Power Coefficient of Reactivity

Power coefficient is defined as the change in reactivity due to power change.

$$\rho_P = \frac{\Delta_\rho}{\Delta P}$$

This coefficient is defined to take into account the non-uniform temperature rise across the core due to power generation. Power generation results in a $\Delta T = \left(T_{outlet} - T_{inlet}\right)$ across the core. Fuel temperature (T_f) also increases due to power change.

2.5.4 Sodium Void Reactivity Effect

Sodium voiding due to its boiling or loss will result in large change of reactivity due to the following effects:

- Spectral hardening
- Less capture
- Increased leakage

<u>*Spectral Hardening:*</u> Because of the reduced elastic and in-elastic scattering by sodium, the neutron spectrum becomes harder (Fig.2.4)

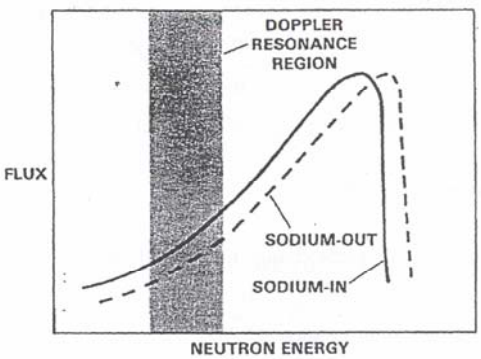


Fig.2.4 : Hardening of neutron spectrum due to sodium voiding

This effect gives always positive reactivity due to

- An increase in fission in fertile nuclides like Th-232, U-238 and Pu-240.

- Decrease in capture to fission rate for fissile nuclides (Pu-239).
- Decrease in capture in fertile and structural nuclides.

Less Capture: Reduction of capture in sodium will result in a positive reactivity. This is generally a small effect since sodium capture cross section is very small.

Increased Leakage: With sodium removal from the core, the leakage probability of neutrons increases and the reactivity thus decreases. Major contribution to this effect will come from the sodium near the core boundary. Leakage effect can be increased by

- Increasing the surface-to-volume ratio of the core. This can be achieved by choosing a core with a smaller core height to diameter ratio (H/D), a so-called 'pancake' core.
- Heterogeneous core

It is to be noted that the magnitude of leakage effect decreases as the core size increases. Sodium loss reactivity effect is space dependent (Fig.2.5). Sodium loss from the centre of the core yields a highly positive reactivity effect, whereas its loss at the edge gives negative effect.

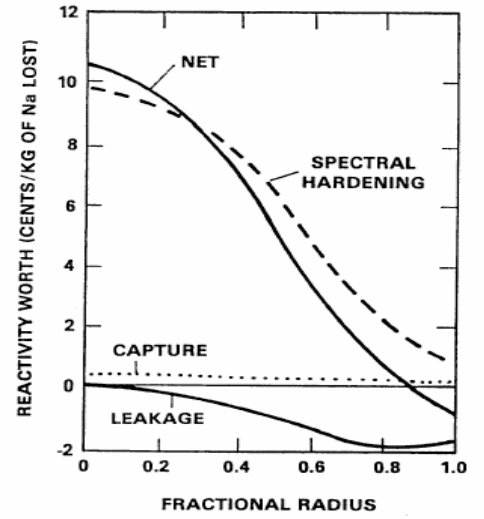


Fig.2.5 : Spatial dependence of various effects due to sodium voiding

The net effect of all three effects will be negative if leakage effect predominates; for example in small reactors (like in FBTR). Large cores like PFBR has positive sodium void coefficient. For PFBR, complete voiding of sodium results in a reactivity change of +620 pcm. However, during normal operation of a reactor, reactivity due to sodium density changes is very small.

2.5.5 Long term Reactivity Loss from Burnup

An operating reactor loses reactivity due to the burnup of fuel and accumulation of fission products. In fast reactors, fission products do not behave as neutron poisons to the extent as they do in thermal reactors. The major loss in reactivity comes from the loss of fissile elements. This is partly compensated by the internal breeding which takes place in the core. There is a small reactivity gain due to the build-up of fissile material in the blankets.

2.5.6 Reactivity Effects Due to Core Compaction

It is to be noted that thermal reactors are designed in a most reactive configuration. It means that any disturbance to the core configuration lead to negative reactivity. It is not the case for fast reactors. Core compaction leads to positive reactivity. During core compaction, core size reduces and fuel density increases. Net effect is to reduce the leakage rate and hence the core reactivity increases.

2.5.7 Fuel Expansion and Bowing Effects

Due to radial power distribution across the core, the parts of fuel element closer to the reactor centre will be slightly hotter than the parts away from the centre. If the fuel elements are cantilevered at the bottom (top free), the differential expansion of the fuel element will cause a net movement of fuel outward from the core centre (bowing) as the

power increases giving a negative reactivity. In other words, cantilevered from bottom of a fuel sub-assembly causes flowering of core giving negative reactivity. In case of contact at two pad levels, the fuel element sometimes may move inwards, giving positive reactivity. In practise, however, the fuel elements cannot move freely very far before they touch each other, and the reactor may be designed so that they cannot move freely at all. For this reason, the effect of bowing on reactivity is likely to be very small.

2.6 REACTOR CONTROL

The power produced in a nuclear reactor core is proportional to the rate of fission which is proportional to the product of the fissile atom concentration and the neutron population, *N*, in the core. The neutron population level arises from a dynamic equilibrium between the generation of neutrons by the fission of fuel atoms and the loss of neutrons by absorption in fuel and other atoms and by leakage from the core.

A fundamental parameter is the multiplication factor, *k*, which is the ratio of the rate of generation of neutrons to the rate of loss of neutrons. The fission rate and reactor power will hence continuously increase, decrease or be constant accordingly as *k* is greater than, less than or equal to unity and the reactor is correspondingly said to be super critical, subcritical or critical. In an operating reactor, *k* can be changed with limits by movement of the control rods which changes the rate of absorption of neutrons. An alternate parameter to *k* is the reactivity ρ defined as $\rho = (k-1)/k$. Study of the normal and abnormal changes to the core reactivity is important for control and safety.

Prompt neutron lifetime and delayed neutron fraction

The prompt neutron lifetime, *l* is the average time from the `birth' of a neutron in the fission of a fuel atom to the `death' of the neutron by absorption or leakage. The life time in a thermal reactor consists of two parts; (i) slowing down time (ii) thermal neutron life time or diffusion time. As the slowing down time is much less than the diffusion time (slowing down time vary depending on the moderator material used), the prompt neutron life time can be equated to diffusion time. In a fast reactor, *l* is of the order of 10^{-7} sec as compared to of the order of 10^{-4} to 10^{-3} sec for thermal reactors. In this connection, it must be noted that over 95% of the reactions in a fast reactor are scattering reactions which reduce the neutron energy and prolong the neutron life. The average rate of loss of neutrons in *N/l* and the rate of production is *kN/l*. Thus the growth of neutron population, for constant *k*, will be exponential with a time constant $T = l/(k-1)$ called the reactor period.

Normal control movements make $(k-1)$ of the order of 10^{-4} to 10^{-3} so that the period would be of the order of 10^{-4} to 10^{-3} for fast reactors and of the order of a second for thermal reactors. These are quite rapid response times and would have made reactor control problematic, especially in the case of fast reactors. However, as explained below, the presence of delayed neutrons increases the response time and makes it the same for both fast or thermal reactors.

A small fraction β (delayed fraction) of the neutrons produced in the fission of a fuel atom are, in fact, emitted subsequent to the fission by decay of certain radioactive fission products (there are six delayed neutron precursors) with mean lives ranging from 0.3 sec to 80 sec. β is 0.00204 and the average mean life τ is 14.6 sec for Pu^{239} and the values differ for different nuclides. When delayed neutrons are included, the reactor period can be obtained, when *k* is near unity, by replacing the prompt neutron lifetime *l* by a weighted mean neutron lifetime $l^* = (1-\beta)l + \beta\tau$. The term l^* is of the order of 0.05 sec and hence for $(k-1)$ of the order of 10^{-4} to 10^{-3} sec, a period of the order of 50 to 500 sec will result. On the other hand, if delayed neutrons are not included in the calculation, the reactor period

would be about 0.45 sec which is very much less than the value obtained (14.6 sec) with delayed neutrons. The reason is that because of the delay in the release of a fraction of the neutrons, the average effective generation time is greater than the prompt neutron generation time by certain quantity, which depends on the extent to which the delayed neutrons contribute to the fission chain reaction.

When ρ is greater than 0 by an amount β , then the neutron population grows at a rate governed by the extremely short prompt neutron lifetime *l* and not by the weighted mean neutron lifetime *l**. The reactor under such conditions is said to be super prompt critical and very rapid rates of power increase become possible. An important safety consideration is to protect against the possibility of a reactor becoming super prompt critical. On account of the importance of the delayed neutron fraction on the kinetic behaviour of the chain reaction, ρ is often measured in terms of β. An amount of reactivity equal to β is called a `dollar of reactivity' which makes the reactor super prompt critical. All the above hypothesis hold good only if the reactor system is not far from critical and reactivity $\rho \leq \beta$.

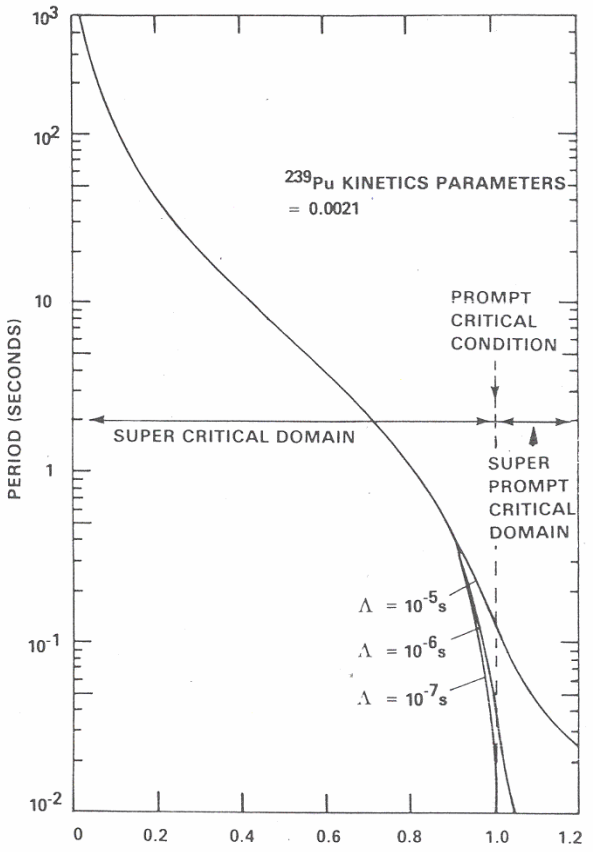


Fig.2.6 : Reactor period variation with reactivity ($) for Pu-239 fuelled reactor

A plot of reactor period vs reactivity is shown in Fig 2.6. It can be observed that, for a reactivity of less than about $0.9, the reactor period is unaffected by neutron lifetime i.e fast and thermal reactors behave identically. Detailed reactor kinetics is covered in other lectures and beyond the scope of this lecture.

2.7 OTHER ASPECTS OF PHYSICS DESIGN

The other aspects for completing the physics design are fuel enrichment and fuel SA worth, absorber worth and shutdown margin. These are only briefly described and detailed description is beyond the scope of this lecture.

The fuel enrichment defined by the volume fraction of the fissile in the fuel should be sufficient to ensure a critical core during an entire cycle of operation. At the end of the cycle, it should have a minimum reactivity margin normally around a few hundred pcm. It is required to overcome the reactivity losses during the operational cycle accounting for temperature and power effects and burnup reactivity loss including uncertainties. Fuel SA worth is determined by the safety criteria on the Keff adopted that individual SA when immersed in water should satisfy. Similarly absorber worth is also decided by the number of shutdown systems provided, various safety criteria to be satisfied such as (n-1) rods to be considered for a system and worst possible core configuration etc. and minimum shutdown margin that is to be provided. Accordingly, the absorber rods position and the number are determined by means of detailed calculations.

2.8 SHIELDING

Purpose and principle

The main purpose of the shield systems is to

(i) protect operating personnel from radiations and

(ii) reduce the radiation exposure of structural components.

Radiations entering the shield from the reactor can produce internal heating and damage to shield materials. Shielding should not only take care of primary radiations (neutrons and gamma rays from the core) but also secondary radiations produced outside the core as a result of interaction of the primary radiations, mainly neutron with nuclei of the fertile, coolant and structural materials.

Therefore the shielding of nuclear reactor systems is concerned with the study of radiation distribution in the shield towards reducing the escaping radiation to acceptable levels and of limiting the adverse physical effects on the shield materials.

Distribution of energy in a single fission is given below.

KE of fission products	≈ 168 MeV
Instantaneous γ energy	7 MeV
KE of fission neutron	5 MeV
β particles from fission products	7 MeV
γ from fission products	6 MeV
Neutrinos	10 MeV
TOTAL	200 MeV

In principle the radiations, which might escape from a reactor, include alpha and beta particles, gamma rays, neutrons of various energies, fission fragments and even protons resulting from (n,p) reactions. In practice, however, only gamma rays and neutrons need to be considered as they are the most penetrating. Any material, which attenuates these radiations to sufficient extent, will automatically reduce all the other to negligible proportions. For purposes of shield design, the neutrons and gamma rays are considered from the stand point of view of their place of origin. The primary radiations are defined as those, which originate in the core; the secondary radiations are produced outside the core as a result of the interaction of primary radiations (mainly neutrons) with nuclei in structural, coolant and shield materials.

Sources and problems of radiation in a fast reactor

a) Cores are small, power densities higher

This implies that leakage of fast neutron into the shield is high. Thermal reactor shields do not have to handle so many fast neutrons. These fast neutrons leaking out have energies greater that 10-15 KeV on an average. Neutrons with energies greater than 3MeV are also quite considerable. Therefore neutron shields should contain a large amount of moderator material to slow down the neutrons before capture. The capture rates are much higher at lower energies

b) Sodium Activation

$$Na^{23}(n,\gamma) \rightarrow Na^{24} \xrightarrow{15h} Mg^{24}\ ;\ 1.38\ \text{MeV}\ \gamma\ \text{and}\ 2.76\ \text{MeV}\ \gamma$$

$$Na^{23}(n,2n) \rightarrow Na^{22} \xrightarrow{2.6y} Ne^{22}\ ;\ 1.28\ \text{MeV}\ \gamma\ \text{and}\ 0.44\ \text{MeV}\ \gamma$$

Shielding against these activation gammas is one of the important problems in fast reactors.

c) Corrosion products

The constituent elements of stainless steel get activated in the reactor

$$Fe^{54}\ (n,p) \xrightarrow{280d} Mn^{54}\ 0.84\ MeV\ \gamma$$

$$Ni^{58}\ (n,p) \xrightarrow{71d} Co^{58}\ 0.81\ \ MeV\ \gamma$$

$$Fe^{58}\ (n,\gamma) \xrightarrow{4.9d} Fe^{59}\ 1.3\ MeV\ \gamma$$

$$Co^{59}\ (n,\gamma) \xrightarrow{5.27y} Co^{60}\ 1.33\ \&\ 1.17\ MeV\ \gamma$$

For maintenance of primary pumps and IHXs , these radiations pose problems.

d) Argon gas activity

Argon gas is used as inert gas blanket over sodium in the reactor to maintain inert gas atmosphere. This also get activated

$$Ar^{40} \rightarrow Ar^{41} \xrightarrow{1.8h} 1.29\ MeV\ \gamma$$

e) Clad rupture

If clad rupture happen, fission gases (isotopes Kr and Xe which are radioactive) escape into cover gas. Because have and (e) adequate shielding has to be provided for cover gas pipes.

Point Kernel Methods

In the point kernel method, the fundamental assumption is that the extended sources of radiation can be regarded as consisting of differential isotropic point sources and that the effect of the radiation from the whole source at the point of interest can be obtained by the summation of the individual sources that comprise the entire source region. In most practical shielding applications the assumption that the gamma rays are emitted isotropically is physically valid when considering effects on macroscopic scale. The effect of radiation at a particular point is usually identified with the response of an idealized detector situated there. The point kernel technique is then applied to various important types of extended source, disc, source and volume source to obtain doses with shield material present in between source and the point at which the dose estimation is made. In most practical cases the dose at the point of interest consists of contributions not only from uncollided radiation but also from secondary radiation such as scattered radiation. Contribution of secondary radiation is taken care through the concept of build up factor, which will be discussed later.

Build up factor

In practical cases, the dose at the point of interest consists of contributions not only from uncollided radiation but also from scattered radiation in shields. Contributions from both the scattered and the uncollided radiations are taken care of through build up factors which is defined as, $B = \dfrac{\text{total dose}}{\text{dose due to unscattered radiation}} = \dfrac{\int_0^{E_0} R_C(E)\Phi(E)dE}{R_C(E_0)\Phi(E_0)}$

where E_0 is the incident gamma ray energy, R_C (E) is the dose conversion coefficient and photon flux φ(E) at the position P where dose rate is being calculated.

Build up factors have the following important properties:

(i) Depends on the distance of penetration through the attenuating medium in terms of mean free paths μt.
(ii) The type of medium, expressed in terms of the atomic charge number Z of material.
(iii) The energy of the source photons.

Neutron attenuation

Neutron shielding analysis is complex. It involves not only attenuation of primary (source) but also production and attenuation of secondary particles such as photons. Neutron shielding analysis might also require consideration of energy deposition in the shielding material, and the resulting thermal and radiation damage effects. Neutron attenuation calculations are usually performed using large computers with extensive cross section libraries using transport calculations codes. The use of neutron build up factors has not been practical because

1) For fast neutrons, the probability of scattering is much higher than absorption in most materials. As a result neutron build up factor can assume very large values
2) Rapid variation of cross sections with energy makes the build up factor too sensitive to initial energy.
3) Angular distribution of the scattered neutrons may also very rapidly with neutron energy.

Therefore the correction for the uncollided flux densities for scattered fast neutrons is generally difficult. An approximate technique is to replace the actual total cross section by an effective removal cross section in the uncollided flux density description. This concept is found useful (particularly) when relatively thin slabs of shield materials are immersed in an infinite hydrogeneous medium such as water.

Shielding materials used

For slowing down as well as thermal shields (these are shields introduced to protect reactor vessel from excessive radiation damage): - stainless steel:- very good inelastic scatterer, can bring down the energy of neutron even by 1 MeV

Slow neutron capture: - compounds of Boron (boron is a good absorber) and carbon (which moderates) special mention may be made of B_4C and Borated Graphite

Sodium itself moderates if neutrons are made to go large distances through Na (as in the pool type of reactors)

Gamma attenuation: - Any high Z material with high density. Lead is the best example.

Shielding arrangement within the reactor vessel

Radial shields: Alternating layers of various thickness of shielding materials SS, B_4C

Axial shields: Alternating layers of SS and B_4C to reduce embrittlement of structural materials.

Typical Shield Design Criteria

A typical set of criteria for shield design is given below. First is the basic set of criteria. Second set is the criteria which might vary from reactor to reactor. However, for pool reactors, most of the criteria would apply.

(1) The occupational dose limit is 20 mSv per year average over a period of 5 years and 30 mSv in any calender year.
(2) All exposure shall be kept below 1μSv/h in all fully occupied regions such as control room.

Other typical shielding design criteria that would be adopted other than ICRP limits for the operating personnel and the site are as follows:

(i) dpa and helium production at the grid plate and the core cover plate
(ii) top shield placed above the pool of sodium
(iii) flasks handling intermediate heat exchangers and primary sodium pump and all other flasks used for irradiated material transport
(iv) dose from secondary sodium pumps, dose on contact for components in sodium when taken out for maintenance.

A typical core layout of a pool reactor consisting of shielding SA in the periphery and the axial shield arrangement provided in the SA are shown in Fig 2.7 and 2.8. The radiation streaming paths in a pool type reactor is also shown in fig 2.9.

References:

1. Fast Breeder Reactors, Alan E. Waltar & Albert B. Reynolds, Pergamon Press 1982
2. Nuclear Reactor Engineering, Samuel Glasstone & Alexander Sesonske
3. Internal lecture notes of IGCAR, Kalpakkam.

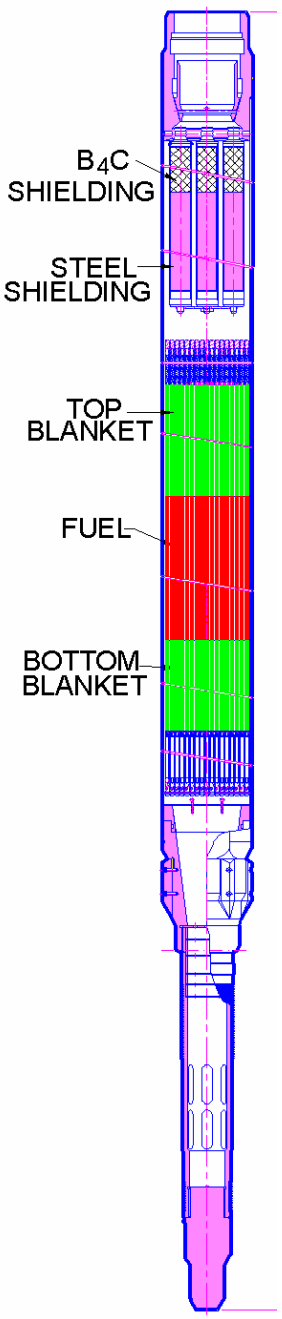


Fig 2.8 : Fuel SA with axial shield

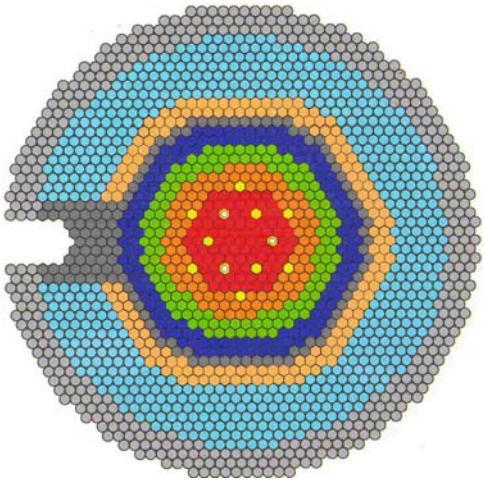


Fig 2.7: Core layout showing shielding SA

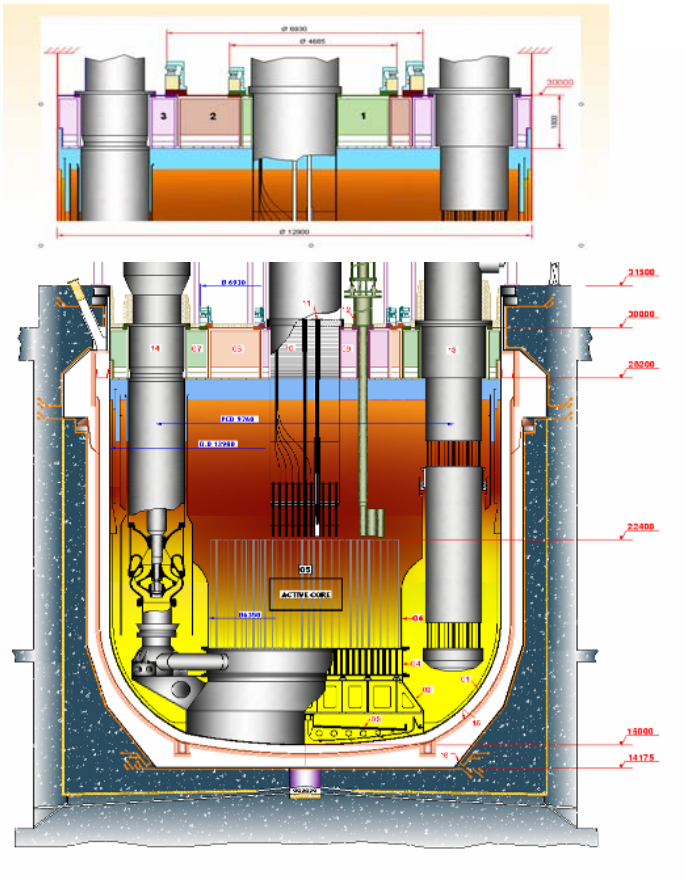


Fig 2.9 : Typical pool type reactor assembly – Radiation streaming paths in top shield