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Fast Reactor Core Design

Module 1 : General Aspects of Fast Reactors

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

Lectures on Fast Reactor Core Design

Module 1 : General Aspects of Fast Reactors

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1.0 ENERGY SCENARIO

Over the next 20 years, electricity demand is expected to increase by 70 percent globally. To ease the impact on global climate, much of this new electricity production is likely to come from nuclear energy, the only existing technology that can generate large amounts of electricity without also emitting greenhouse gases. The relative environmental impact of different technologies of electricity generation is shown in Fig 1.1. From the figure, it is inferred that the nuclear means of electricity generation is having the low green house gas impacts and low air pollution impacts.

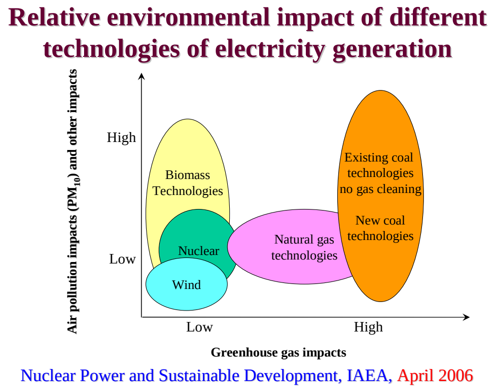


Fig 1.1: Energy Technologies

A study by World energy council, with three growth scenarios defined as high, middle, ecologically driven growth forecast that the total primary energy demand will be 25000, 20000 & 14000 mtoe for the three scenarios by 2050 (Fig 1.2). In the ecologically driven scenario, nuclear share is estimated as 4 and 12% for two variant cases and in the high and middle scenarios, the nuclear share comes to around 10-14 %.

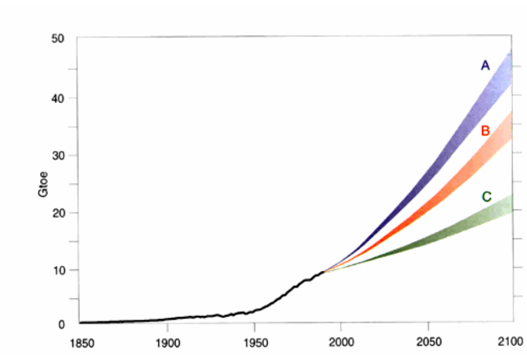


Fig 1.2: Energy Growth Scenario

Projections through the year 2030 show a continuing increase in global carbon dioxide emissions, if no new policies and measures are put in place. Under this scenario, emissions are projected to grow by 69%, slightly more than the growth of 66% in energy supply. The most rapid increases are seen as occurring in Non-OECD countries, where emissions will more than double over the period. The share of OECD emissions in total emissions will decrease from 54% in 2000 to 42% in 2030. Power generation, which currently accounts for around 40% of the emissions will contribute almost half the increase (or 8 billion tonnes) in global emissions between 2000 and 2030. Transport will account for more than a quarter; residential, commercial and industrial sectors accounting for the rest. The average carbon content of energy – CO_2 emissions per unit of aggregate primary energy consumption –will increase over the next 30 years. The main cause of this reversal will be the declining share of nuclear and hydro power in the global energy mix. These environmental concerns highlight the importance of renaissance and enhanced pace of exploiting the nuclear power on a large scale with suitable policies to be adopted worldwide and international measures to be undertaken by the relevant agencies and policy makers.

Conscious of the environmental concerns, there will be nuclear renaissance in several countries. Moreover, it is expected that in the 21^{st} Century, the power industry will develop in the free market environment. Development of the entire power industry and nuclear power as its integral part will be influenced by the following main factors: economics, safety, radioactive waste (RW) management, non-proliferation of nuclear weapons, macroeconomic factors, restructuring of the electricity market, changing structure of power resources and environmental protection.

The growth of nuclear power will also depend on the status and maturity of the nuclear fuel cycle technologies. Parallel to the renaissance of the nuclear energy, the concept of closed fuel cycle is also receiving close attention. It is now realized that reprocessing (or recycling) the spent fuel is essential for the effective utilisation of resources and for reducing the environmental impact of nuclear power by reducing the requirement for waste repositories. Accordingly, several countries have renewed the emphasis on “closed fuel cycle” as a route to sustainability of nuclear energy cycle. Many countries are expected to adopt a closed fuel cycle in the next century, for effective utilization of uranium as well partitioning and transmutation of transuranium elements.

2.0 MOTIVATION FOR DEVELOPMENT OF FBR

Most of the present day commercial power reactors are called "thermal" reactors because the neutrons are slowed down to thermal energy using a moderator. By contrast, a "fast" reactor uses neutrons of much higher energy to cause fission and does not have a moderator. The main focus was on the deployment of thermal reactors from the point of view of uranium price and utilization of U resources and non-proliferation concerns although the world’s first nuclear reactor was indeed a fast reactor (FR).

The factors that could be attributed for the motivation of the fast reactors are:

(i) Breeding potential and consequent energy security

(ii) Effective utilization of natural uranium resources

(iii) Actinide burning

(iv) Waste minimization and environmental consideration

Breeding

In a nuclear reactor, fissile materials both destroyed (FD) and produced by conversion of the fertile material (FP). The degree of conversion that occurs in a reactor is denoted by the general term called ‘conversion ratio’ CR, defined as FP/FD. If this conversion ratio is greater than 1, it is called breeding ratio. In a typical PWR core, the conversion ratio is about 0.7. Enveloping the core of a PWR with a blanket does not significantly change its conversion ratio due to the low number of neutrons leaking out of the core, which remains less than 1. In a FR, the internal conversion ratio is 0.72 in the fissile zone compared to 0.7 in PWR. However, the large number of neutrons leaking out of the core is captured by the nuclei of ^{238}U in the blankets and creates the possibility of achieving a net breeding ratio more than 1. Thus, the rate of regeneration of the core itself is higher in FBR than in PWR of equivalent power. The condition necessary for breeding to take place is $\eta > 2$ (η – number of neutrons produced per neutron absorbed) is shown in Fig 1.3. The value of η for various fissile isotopes is given in Table 1.1. Hence, the breeding ratio that could be achieved is dependant on the mix of fissile

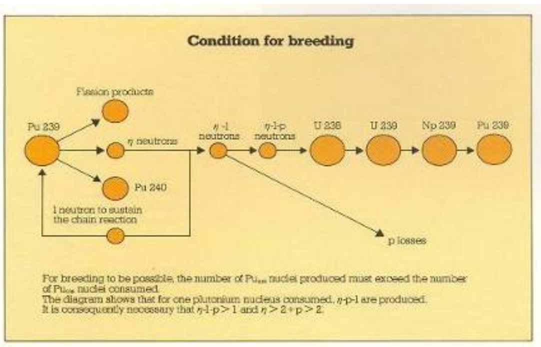


Fig 1.3: Condition for breeding

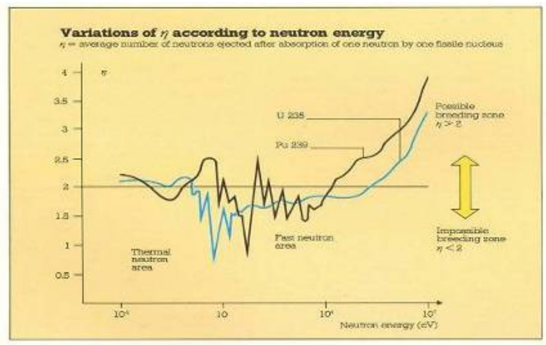


Fig 1.4: Neutron yield of fissile isotopes

Table 1.1 : Neutron yield of fissile isotopes

Neutron Spectrum	Natural Uranium	Uranium 235	Uranium 233	Plutonium 239
Thermal	1.34	2.04	2.26	2.06
Fast	< 1	2.20	2.35	2.75

and fertile isotopes of the fuel. A look at the neutron yield of various fissile isotopes will reveal that the combination of U235-Pu239 will result in a highest breeding ratio (Fig 1.4). Obviously, with such a breeding potential, the energy potential could also be increased many folds correspondingly.

Effective utilization of natural uranium

Fast breeder reactors can make use of the available uranium reserves better than other systems. Consider a uranium-fueled reactor in which N atoms of U-235 area fissioned. While this is happening, CN new fissile atoms of Pu-239 can be produced. If these in turn are fissioned in the same reactor (the conversion or breeding ratio C is unchanged), a further C^2N fissile atoms are produced. If these are again fissioned, C^3N fissile atoms are produced, and so on. The total number of atoms fissioned is therefore,

$$N\ (1 + C + C^2 + C^3 + \ldots\ldots\ldots\ldots\ldots\ldots\ldots)$$

if C < 1, the above series converges to 1/(1-C). If the fuel is natural uranium, N cannot exceed 0.7 % of the total uranium. If the reactor is a thermal reactor with a conversion ratio of 0.7 and the plutonium bred is re-cycled indefinitely the total number of atoms fissioned can not exceed 0.7 / (1-0.7) = 2.4 % of the number of uranium atoms supplied. In other words, the uranium utilization, which is defined as the fraction of fuel atoms consumed for power production can not be higher than 2.4 %. Theoretically speaking, it would be about 2%, due to losses during re-processing and fuel fabrication. However, in reality, recycling for more than one recycle is not possible owing to the poor quality of plutonium after recycling in thermal reactors.

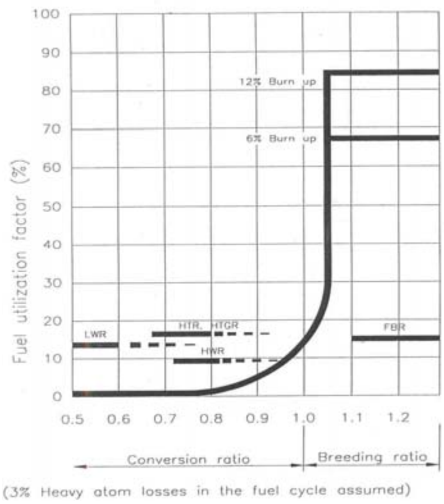


Fig 1.5: Uranium Utilization

Higher and effective utilization of natural uranium resource to the extent of 60-70% is possible only in FBR system with multiple recycling, which is explained below in comparison with PHWR using natural uranium. In PHWR, a maximum of 0.7% of the uranium resource is used. In a closed fuel cycle through FBRs, the fuel can be recycled any number of times. But considering the fissile material quality, reprocessing loss and fertile feed for every recycle, around 10 recycles are realizable. In every cycle, about 7 atom% of heavy elements is burnt corresponding to a reasonable average burnup of 70,000 MWd/t (~ 7 at%). Assuming 7 recycles are possible out of 10, nearly 49% of fissile atoms can be burnt in FBR, which gives a ratio of about 70 times utilization factor (Fig.1.5). This means that one kg of natural uranium would generate about 36,45,600 kWh in FBR, compared to only about 52,080 kWh possible in PHWR, assuming an efficiency of 31%. It is worth mentioning that with advanced fuel with high burn-up (peak burn-up 200,000 MWd/t) and fuel cycle losses of 1%, it is possible to realize even higher utilization factor. Further, with higher in-core breeding, fuel residence time (cycle length) can be increased and hence high burnup is possible. High burnup results in better fuel utilization due to reduction of fissile material loss during reprocessing.

World uranium world reserves (3.38 Mt) and resources (12.5 Mt) are estimated to be about 16 million tons or about 160 Gtoe. At the rate of current global consumption (0.6 Gtoe/year) and assuming the use of open cycle with water reactors, reserves and resources represent about 250-270 years of production. Looking at the energy growth projected, there is likely to be a severe constraint on the fossil fuels which would eventually lead to large scale deployment of nuclear on a long term basis. To cater to such demands, adoption of water reactors with open cycle will not be the right solution and fertile isotopes have to be exploited for which fast reactors are ideal. With fast neutron reactors, the only reserves, U-238 currently stored as tail end product from enrichment plants, allow for increased energy reserves up to a factor of about 50 in comparison to the current water reactor technology. Therefore, fast reactors are the ideal vehicle for efficient use of uranium resources.

Actinide management

Many of the long-lived actinides that cannot be fissioned in a thermal reactor can be burned in a fast reactor, so the fast reactor is capable of destroying the major source of long-lived radiotoxicity in spent fuel. Thus, the fast reactor can create new fuel and destroy long-lived nuclear waste and plutonium while it produces electricity. Fast reactors play a unique role in the actinide management mission because they operate with high energy neutrons that are more effective in fissioning transuranic actinides. In contrast, thermal reactors extract energy primarily from fissile isotopes; a thermal spectrum also leads to the generation of higher actinides that complicate subsequent recycling.

Fast reactors can operate in three different fuel cycle modes: (i) transmuter mode where conversion ratio is less than 1 resulting in net consumption of transuranics and conversion of transuranics into shorter-lived isotopes in order to reduce long-term waste management burdens; (ii) converter mode where conversion ratio is close to 1 providing a balance in transuranics production and consumption resulting in low reactivity loss rates; (iii) breeder mode where conversion ratio is greater than 1 with a net production of transuranics. This approach allows the creation of additional fissile materials but will require extra uranium in the fuel cycle. Depending on the objective, FR core can be designed appropriately providing a balance between waste reduction and resource enhancement.

Waste minimization and environmental consideration

Fast reactors provide the best possible means for the transmutation of actinides in the reactor as compared to the thermal reactors which is due the fact that the neutron flux in FR is one order higher and further the capture cross sections of major actinides are higher than the fission cross sections in a fast neutron spectrum. Thermal reactors, rather lead to generation of higher actinides and they primarily extract energy from fissile isotopes. The extent of waste reduction through FR is shown in Fig. 1.6. In view of the higher operating temperatures, the thermal pollution from fast reactors is less compared to thermal reactors owing to its higher thermodynamic efficiency ~40-42% as compared to ~ 28-31% in thermal reactors.

The development and deployment of advanced nuclear reactors based on fast-neutron fission technology is important to the sustainability, reliability, and security of the world's long-term energy supply. Fast reactors in conjunction with fuel recycling can diminish the cost and duration of storing and managing reactor waste with an offsetting increase in the fuel cycle cost due to reprocessing and fuel re-fabrication. Virtually all long-lived heavy elements are eliminated during fast reactor operation, leaving a small amount of fission product waste that requires assured isolation from the environment for less than 500 years. Fast reactors make the task of proliferation resistance easier by segregating and consuming the plutonium as it is created for further power generation.

3.0 FAST REACTOR STATUS WORLDWIDE

Research and design work on liquid metal cooled fast reactors have been carried out for more than 50 years. Nuclear electricity was first generated on 20 December 1951 by 0.2 MWe EBR-1, in the USA. The first fast reactor with plutonium oxide fuel and sodium coolant, BR-5, was started in 1958 in Russia, and is in operation for over 40 years. The first demonstration fast reactor with uranium oxide fuel and sodium coolant, BN-350, 750 MWt, was started in the former USSR (commissioned on 16 July 1973) for electricity generation (150 MWe) and heat production for seawater desalination and it has been in operation for over 25 years. Closed fuel cycle was first demonstrated in January 1980 at the French prototype reactor Phenix, 250 MWe, i.e. plutonium produced in this reactor was used as the fuel for its core. Also, 1.16 breeding ratio was experimentally confirmed in this reactor. In the UK, in the 250 MWe prototype fast reactor (PFR), large numbers of MOX fuel pins reached more than 15% burnup without failure, and experimental fuel pins achieved 20% burnup with an irradiation dose in excess of 130 displacements per atom (dpa). These results have been confirmed and surpassed by irradiation in Phenix to more than 160 dpa.

So far, twenty SFRs have been constructed and operated, five prototype demonstration SFRs (BN350/Kazakhastan, Phenix/France, Prototype Fast Reactor/UK, BN-600/Russian Federation, Super Phenix/France) with electrical output ranging from 250 to 1200 MWe. Large scale (400 MWt) experimental fast flux test reactor FFTF/USA in addition to other experimental reactors, have gained nearly 110 reactor years. The small size experimental reactors, for example EBR-II, Rapsodie, BOR-60, JOYO and FBTR have provided valuable experience on sodium technology, fuel element design involving choice of fuel, cladding and wrapper material, demonstration of burnup limits and irradiated material data. The objective of US fast reactor programme of U-19Pu-10Zr sodium bonded metal fuel has been successfully demonstrated in EBR-II and FFTF. So far, 390 reactor-years experience has been gained in the operation of fast reactors. Major features of fast reactors which have been constructed and operated are given in Tables 1.2 and 1.3 respectively.

Regarding the future developments, significant technology development program for SFRs is proceeding in several countries, namely in France, India, Japan and the Russian Federation. Activities are continued in a number of other countries in smaller measures. It is important to note that out of six reactor systems selected as the most promising systems by Generation IV international Forum, four are fast reactors and one is specifically sodium cooled fast reactor. Under the Joint Study organized by IAEA on Assessment of an Innovative Nuclear Energy System (INS) based on a Closed Nuclear Fuel Cycle with Fast Reactors (CNFC-FR) involving China, France, India, Republic of Korea and Russian Federation, an overview of national energy strategies was done, which has indicated that Innovative Nuclear Systems based on a CNFC-FR is being considered as a promising component of the future sustainable nuclear energy system capable of providing a global response to global energy challenges in the 21st century in countries with more than a half of the Earth's population. Further, the SFRs with matching nuclear fuel cycle is being considered as an only option with potential of commercialization in 15-30 years.

Table 1.2 : Major Features of Fast Reactors Constructed

Reactor name	Country	Location	First Criticality date	Shut down date	Thermal capacity (MW)	Electric Capacity (MW)	Fuel	Primary Circuit config.	Primary coolant	Primary coolant temp (°C) Out/In
Clementine	USA	Los Alamos	1946	1953	0.025		Pu metal		Mercury	140/40
BR-2	CIS	Obninsk	1956	1957	0.1		Pu metal		Mercury	70/40
EBR-I	USA	Argo (Idaho)	1951	1963	1.4	0.2	U		Sodium	
BR-5	CIS	Obinisk	1959	1971	5		PuO_2, UC		/potassium	450/375
BR-10			1971		10		MOX, UN	Loop	Sodium	
DFR	UK	Dounray	1959	1977	72	15	U –Mo	Loop	Sodium /potassium	350/230
EBR-2	USA	Argo(Idaho)	1963	1994	62	20	U-Zr, U-Pu-Zr	Pool		482/370
E.Fermi(EFFBR)	USA	Detriot	1963	1972	200	66	U-Mo	Loop	Sodium	427/268
Rapsodie	France	Cadarache	1966	1982	20/40		MOX	Loop	Sodium	510/404
BOR -60	CIS	Dimitrovgrad	1969		60	12	MOX	Loop	Sodium	550/360
Joyo	Japan	Oarai	1977(mark-I)		100(mark-II)		MOX	Loop	Sodium	500/370
FBTR	India	Kalpakkam	1985		40		(U,Pu)C	Loop	Sodium	518/400
KNK-II	Germany	Karlsruhe	1977	1991	58	21	MOX/UO_2	Loop	Sodium	
SEFOR	USA	Arkansas	1969	1972	20		MOX	Loop	Sodium	430/370
FFTF	USA	Hanford	1980	1994	400		MOX	Loop	Sodium	590/370
PEC	Italy	Brasimone	Aband		125		MOX	Loop	Sodium	525/375
BN-350	CIS	Chevenko	1972	1999	1000	150 and desalinisation	UO_2	Loop	Sodium	500/300
PFR	UK	Dounray	1974	1994	600	270	MOX	Pool	Sodium	560/400
Phenix	France	Marcoule	1973		560	250	MOX	Pool	Sodium	552/385
SNR -300	Germany	Kalkar	Aband in 1991		770	327	MOX	Loop	Sodium	560/380
BN – 600	CIS	Beloyarsk	1980		1470	600	UO_2	Pool	Sodium	550/550
CRBR	USA	Clinch River	Aband in 1983		975	380	MOX	Loop	Sodium	
Monju	Japan	Tseruga	1994		714	280	MOX	Loop	Sodium	529/397
Superphenix	France	Creys – Malville	1985	1996	3000	1240	MOX	Pool	Sodium	545/395
BN – 800	CIS	Beloyarsk	-			800	UO_2	Pool	Sodium	550/350

Table 1.3: Major Reactor Core Features of Fast Reactors Operated

	Joyo (Mark II) Japan	Phenix France	Monju Japan	BN-350 Kazakhstan	BN-600 Russia	Superphenix France
CAPACITIES						
Thermal capacity (MW)	100	560	714	1000*	1470	3000
Gross electric capacity (MW)	0	250	280	150	600	1240
Net electric capacity (MW)	0	233	246	135	560	1200
CORE						
Active height/active diameter (m)	0.55/0.72	0.85/1.39	0.93/1.8	1.06/1.5	1.02/2.05	1/3.66
Fuel mass (tHM)	0.76	4.3	5.7	1.17 ^{235}U	12.1(UO_2)	31.5
Number of assemblies	67	103	198	226	370	364
Maximum power (kW/l)	544	646	480	-	705	480
Average power (kW/l)	475	406	275	400	413	280
Expected burn-up (MWd/t)	75000	100000	80000	100000	100000	70000(first core)
FUEL						
Fissile material	MOX	MOX	MOX	UO_2	UO_2	MOX
Enrichment (%) first core	30 Pu	19.3 Pu	15/20 Pu_f	-	-	15.6 Pu
Mass of plutonium (t) first core						6
Enrichment (%) reloads	30 Pu	27.1 Pu	16/21 Pu_f	17/21/26	17/21/26	20 Pu
Mass of plutonium (t) reloads						7
Assembly renewal rate	70 days	3 months	20% of core every 5 months	80 efpd	160 efpd	100% of core every 3 years
Form	Pellet	Pellet	Pellet	Pellet	Pellet	Pellet
Number of pins per assembly	127	217	169	127	127	127
Assembly geometry	Hexagonal	Hexagonal	Hexagonal	Hexagonal	Hexagonal	Hexagonal
Average linear power (kW/m)			21	36		
Maximum linear power (kW/m)	40	45	36	48	48	48
Maximum clad temperature (°C)	650	700	675	700	620	620
Maximum temperature at centre (°C)	2500	2300	2350	2200		

*** Real thermal capacity is 520 MW.**

4.0 CORE DESIGN OBJECTIVES

The objectives that form the basis for the core design are: economy, high breeding ratio, high linear power, enhanced safety, minimization of fissile inventory, high burnup etc. Core and fuel element design is a multi-disciplinary one requiring analytical tools and data on materials. High burnup and higher coolant outlet temperature from core lead to lower fuel cycle cost as burnup influences the cycle cost significantly. Higher linear power would help in extracting more energy from a given fuel mass. If the objective is on faster growth of fast breeder reactors for power production, higher breeding ratio which in turn would lead to shorter doubling time will be sought after. Countries choose the objectives in line with their national policies and their domestic energy requirements. The chosen design options would be evaluated to ensure safety as per the regulatory requirements.

It would be difficult to achieve all the design objectives at the same time. Trade off is required to find a reasonable compromise between equally important design objectives by optimization.

5.0 GENERAL CHARACTERISTICS OF FAST REACTOR CORE

The major characteristics of a typical fast reactor are briefly described below.

Neutron Spectrum

The fission neutrons are born with a certain energy distribution known as fission spectrum. It has an average energy of about 2 MeV (most probable energy is about 700 keV for Pu-239). Energy degradation of these fission neutrons in fast reactors is mainly by in-elastic collisions with nuclei of high mass number. In a large ceramic (oxide or carbide) fueled sodium cooled fast reactor, the neutron spectrum may be peaked between 10 to 100 keV depending upon the relative concentrations of different materials. The bigger a fast reactor is made, the spectrum becomes softer because of increased in-elastic and elastic scattering. In a very small high metal density fast reactor, spectrum will be hard.

Low fission cross section and Large Critical Mass

Critical mass of a reactor depends upon the fission cross section of fissile nuclides. The average fission cross section of Pu-239 is about 2 barns in fast energy range, whereas it is about 750 barns in thermal range. Hence fast reactors require large critical mass.

High Fuel Enrichment

The average capture cross sections of fertile nuclides are comparable to the fission cross sections of fissile nuclides in fast spectrum. In thermal reactor spectrum, fissile nuclides have very high fission cross sections in comparison with capture cross sections of fertile nuclides. Hence, for criticality, the ratio of fissile fuel to fertile material in a fast reactor must be relatively high compared to the ratio in thermal reactors. A fast power reactor thus requires a considerable quantity of enriched fuel, about 15 - 30 % enriched in fissile element.

High Power Density

Since a fast reactor requires a large concentration of fissile material, if it has to produce competitive economic power, the total fissile material inventory in the core should be as minimum as possible and hence it becomes necessary to minimize the volume of the core. As a result, the power density becomes high leading to high specific power i.e. the amount of energy produced per unit mass of fissile material must be high giving rise to economic benefits. This in turn means that the resulting high power densities require a very efficient cooling medium capable of high rate of heat removal from the core. Liquid metals in general and sodium in particular has been the universal choice in the reactors constructed so far although studies have been made on alternate coolants.

High Fraction of Structural Materials

Fast reactor core contains more fissile material. For economic power production, burnup should be more. It requires high residence time of the fuel by increasing the clad thickness for limiting the radiation damage. It is to be noted that the neutron economy is not affected much due to low capture cross section of iron. In a typical fast reactor about 15 to 25 % by volume of structural material may be used.

Thin Fuel Pins

Small diameter of the fuel pin provides more heat transfer area for the coolant. Smaller the pin diameter, less will be the fissile inventory. It is to be noted that linear heat rating of fast reactor fuel pin is limited mainly by the centre line temperature of the fuel.

High Burnup

Burnup is the parameter used to represent the energy extracted per unit mass of fuel. It is measured in MWd/kg of heavy fuel metal atoms excluding oxygen or carbon. Equivalently, it can also be expressed in terms of atom per cent. 1 atom % bumup corresponds to 10 MWd/kg or 10,000 MWd/t of fuel. Thermal reactor bumup is reactivity limited. Fast reactor burnup is limited due to radiation damage. Development of materials capable of withstanding high dpa will increase the current bumup limits.

Small Core Size

FBR cores are very compact with high power density. They are usually smaller than that of a thermal reactor for a given power. A thermal reactor core is optimised for a given moderator-to-fuel ratio. Since there is no moderator in FBRs, fuel volume fraction is increased by minimising the coolant and structure. Higher volume fraction minimises the fissile loading.

High Breeding Ratio

Breeding is the process of converting a fertile nuclide to a fissile nuclide. BR is the ratio of amount of fissile material produced to that of the fissile material destroyed either through fission or through capture. BR cap vary from 1.1 to 1.6, depending upon the core design. BR depends upon the reactor spectrum, fuel type, reactor size, pin size, blanket thickness (radial and axial) etc.

High Neutron Flux

High power density and low fission cross section cause high neutron flux (one order higher than thermal reactors) and consequently to high neutron fluence.

High Radiation Damage

Since the neutron flux is higher in fast reactors and the burnup targeted is also high which is not restricted by the fissile content, the neutron fluence is higher. Structural materials undergo damage owing to irradiation. Due to this, there would be severe geometrical deformations and mechanical property degradation in the clad which have to be accounted for in the design.

Negligible Reactivity changes due to fission products

Unlike in thermal reactor, there is no large reactivity change of core due to accumulation of fission products.

Low delayed neutron fraction

Delayed neutron fraction (β) is low in fast reactors (300-400 pcm) compared to thermal reactors (600-700 pcm).

Reactor core configuration

Fast reactor core is not in the most reactive configuration which implies that under conditions of any disturbance to the core configuration such as fuel melting or slumping, the core can acquire more reactivity due to compaction. This is unlike the thermal reactor, since core is arranged in a particular ratio of fuel to moderator and any change in core configuration will result in reactivity decrease.

The above characteristics make the design of fast reactor core quite challenging. Comparison of important core parameters for fast and thermal reactors is given in the following Table 1.4.

Table 1.4: Comparison of fast and thermal reactors

Parameter	Thermal Reactor	Fast Reactor
Fissile enrichment, %	0-3 U^{235}	10 – 30 Pu^{239}
Av. Neutron energy	~0.025 eV	~100 keV
Burnup (MWd/t)	~30,000	~100,000
Neutron Flux, n/cm^2s	10^{14}	5-10 X 10^{15}
Neutron Fluence, n/cm^2	10^{22}	2-10 X 10^{23}
Av. Core power density, W/cm^3	~100	~300-400
Av. Fuel sp. Power, kW/kg of metal fuel	~40	~100

6.0 GENERAL SAFETY CRITERIA

The core design is carried out to satisfy certain safety criteria to meet the design requirements as well as meeting the core and reactor safety. The criteria can be broadly categorized into three i.e neutronics, thermal-hydraulics and mechanical design. Few important criteria in each category are given below. The criteria is defined before the design is taken up. Each country will have its own criteria which is acceptable to its regulatory body.

(i) Neutronics

Fuel subassembly worth: The size of a fuel subassembly should be such that it does not become critical when immersed in water.

Shutdown Margin: Core should be designed with adequate shutdown margin, in order to prevent criticality of the reactor due to change in temperature and due to any fuel handling errors which are in unsafe direction (for example; replacement of absorber SA or a blanket SA by a fuel SA, replacement of lower enrichment SA by higher enrichment SA etc.)

Reactivity Coefficients: The power coefficient, total temperature coefficient of reactivity should be negative throughout the reactor life for all possible operational states and accident conditions taking into account all possible loading configurations and irradiation effects. Sodium void coefficient should be negative or it should be as small as possible if it is positive.

Shielding: Design should ensure sufficient shielding to the operating personnel both at the reactor top and the buildings adjacent to RCB such as Steam generator building etc to meet the prescribed radiation limits and at the same time the shielding thickness should be as minimum as possible.

(ii) *Thermal Hydraulics*

Design Safety limits: Design should be such that design safety limits prescribed for fuel, clad and coolant are not exceeded during all operational states of the reactor.

Core flow zoning: The flow zoning within the core should be made in a such a way that each SA receives the flow proportional to its power generation which also meets the thermal striping and higher mixed mean reactor outlet temperature without exceeding the DSL.

Hydraulic lifting force: The design should be such that no SA should get lifted owing to the coolant flow through the SA and it should have adequate margin to prevent lifting.

(iii) *Mechanical*

Structural design criteria: The core components are subjected to high irradiation. The design should satisfy the structural design criteria for irradiated core components for all possible failure modes that are considered in the design.

Core restraint system: The design should ensure that the core restraint system design ensures that the reactivity change/swing due to core SA deformation is kept to the minimum.

7.0 LIQUID SODIUM AS COOLANT

The coolant medium for a fast reactor has a demanding requirement in terms of good heat transfer properties as the power density is higher as was mentioned earlier. The coolant should essentially meet the following considerations to the maximum extent possible: (i) Neutronic considerations such as minimum neutron moderation, minimum parasitic neutron absorption and ability to resist the activation (ii) Thermal considerations such as high thermal conductivity (iii) Hydraulic consideration such as less pumping power requirement (iv) Chemical compatibility considerations such as non reactive with structural materials and fuel material when there is a clad breach (v) Favorable physical properties such as high boiling point, low melting point and high thermal conductivity.

The requirements spelt out above especially the heat transfer property led to the choice of liquid metals in general and sodium in particular even though gas coolants especially helium were given consideration earlier. The features associated with sodium coolant are as follows.

(i) High thermal conductivity resulting in lower cladding temperatures due to good heat transfer capability. However, it gives rise to thermal shocks during transients.

(ii) High boiling point leading to low pressure reactor systems and high thermal capacity for safety.

(iii) Low pumping power.

(iv) Little neutron moderation offering higher breeding ratio

(v) Chemically reactive with air and water requiring an inert cover gas medium in the systems.

(vi) Opaqueness leading to issues during fuel handling

(vii) Solid at room temperature requiring heaters

(viii) Activation problem requiring intermediate heat transport circuit to prevent radioactive contamination in the steam generator:

$^{23}Na + n \rightarrow {}^{24}Na$ (β^- active with 15 h half-life): E_γ = 1.37 and 2.75 MeV; one more isotope is produced, ^{22}Na with 2.6 d half life, requiring wait period for intervention in sodium circuits, in case of necessity.

Intermediate circuit is warranted from safety consideration too to protect core from possible pressure surges and positive reactivity addition due to moderation by hydrogen produced from sodium-water reaction in the event of SG leak.

Almost all the countries have selected sodium as the reference coolant. Currently, alternate coolants such as lead, lead-bismuth are being considered for future reactors.

8.0 PRIMARY CIRCUIT CONCEPT – LOOP and POOL

The primary circuit of the existing fast reactors have adopted basically two concepts; loop and pool concept. The basic difference lies in the location of intermediate heat exchanger (IHX) and the primary sodium pump (PSP).

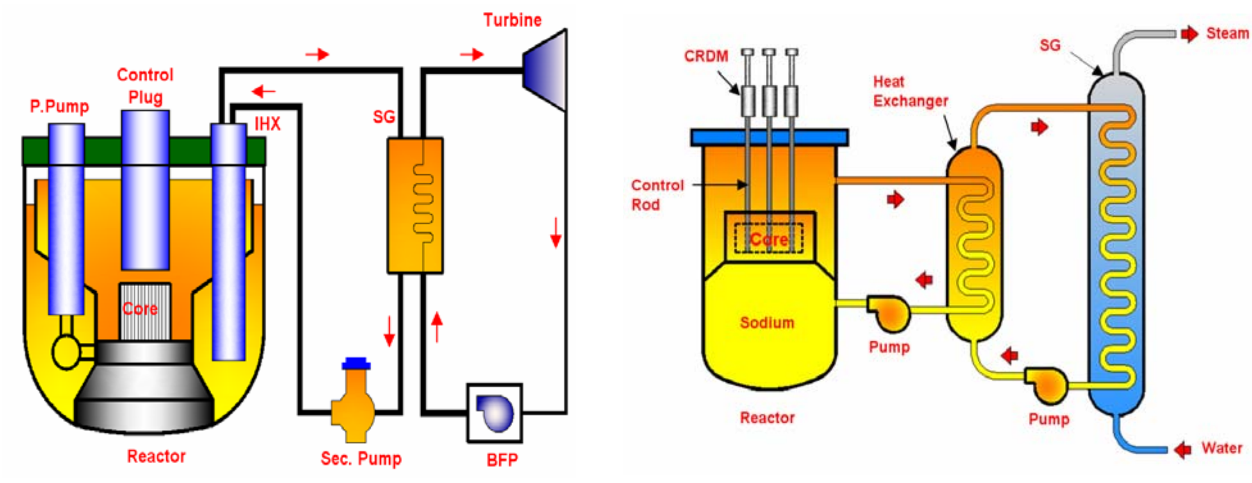
In the pool concept, the primary heat transport circuit components, IHX and PSP, are immersed in a pool of coolant contained within a large vessel. Almost all the primary coolant is contained within the main vessel. Internally within the main vessel, the sodium plenum is separated into hot and cold plenum. Also, components like grid plate support the core and also act as the inlet plenum for flow into core SA.

In the loop design, the IHX and PSP are located outside the reactor vessel. They are placed to adjacent to the reactor vessel and connected together by pipes through nozzles. The important issues in the loop design are the location of the PSP, in the hot or cold leg, and the location of penetration for the coolant inlet pipe as they have their implications.

The typical pool and loop concepts are schematically shown in Fig.1.7. Each concept has its special features, advantages and disadvantages which are mentioned below.

Pool Concept

- High structural integrity of main vessel due to the absence of any penetration.
- Core never becomes devoid of coolant i.e there is never a loss of coolant scenario.
- Large thermal inertia coupled with natural convection capability contributing to safety during transients involving loss of heat sink.
- Absence of primary sodium piping and ability to maintain system integrity in the event of core accident.
- Simple cover gas system; i.e., only one free surface.
- Requirement of Large size vessels and correspondingly large size top shield structures.
- Large inventory of sodium.
- Maintenance is difficult as primary circuit components become radioactive..
- Large neutron shield is required to prevent secondary sodium activation.
- The cold sodium plenum at the pumps suction upstream acts as buffer against either thermal chocks or gas entrainment towards the core.
- No risk of radioactive sodium fire, except during pothetical core disruptive accident (HCDA).
- Limited access for inspection and repair of the vessel internal components.



SG - Steam generator
BFP - Boiler feed pump
IHX - Intermediate heat exchanger
CRDM - Control rod drive mechanism
P.Pump - Primary pump

Pool concept   Loop concept

Fig.1.7 Typical schematic Heat transport circuits of pool and loop concepts

Loop Concept

- Greater difference in vertical elevation of IHX relative to core enhances natural circulation of coolant.
- Requires less neutron shielding.
- Maintenance is simpler since components can be isolated in shells.
- Structural design of reactor vessel is simple.
- Components can be fabricated in shop.
- Tighter coupling of steam and secondary sodium system to the primary sodium system due to small mass of sodium involved.

Some variants were tried in the loop design with improvements to remove the draw backs such as top entry for the pipe and shortened primary pipe lengths. In some variants, a hybrid design was resorted in order to combine the advantages of pool and loop and to remove the drawbacks. By and large, loop design was followed in the early and test reactors and few prototype scale reactors. Both the concepts employ the intermediate circuits in order to prevent primary sodium – water reaction by means of two barriers namely IHX and SG tubes. In the pool concept, the number of intermediate circuits will be the same as the number of primary loops. In the pool concept, the number of intermediate circuits is varying between different designs world over. Generally, it will be based on the reactor size.

9.0 INTRODUCTION TO IMPORTANT TERMINOLOGIES

Before proceeding to the fast reactor core design, certain parameters need to be introduced which are given below.

Pellet density : Fuel pellets are usually designed with certain porosity. It is expressed in terms of the theoretical crystalline density of the fuel material, be it oxide or carbide. Typically, the pellet density would be 90-95%.

Smeared Density: Smeared density is the term used in the physics calculation for calculating the atom densities. There would be iterations between the physics and engineering design that would often involve dimensional changes like the gap between pellet and clad and the porosities inside the fuel matrix. Hence, the physicist would use a term called 'smeared density' which is defined as the density of the fuel as if the fuel mass is uniformly spread or smeared throughout the inside space of the cladding. It gives a measure of the space provided to accommodate the fuel swelling. Typically, the smeared density would be in the range 85-90% of the theoretical density.

dpa (displacements per atom): The structural material damage due to neutron irradiation is high in fast reactors because of interactions by fast neutrons (> 0.1 MeV). Irradiation affects the properties in two ways. Neutron scattering (elastic and in-elastic) interactions displaces atoms from their sites in the crystal lattice creating vacancies and interstitial atoms in equal numbers, and neutron absorption by (n, α) and (n, p) interactions creates atoms of helium and hydrogen and other transmutation products within the crystals. A useful way to characterize the extent of the irradiation a piece of material has received is to specify the average number of times an atom has been displaced (dpa) from its lattice site. The total number of displacements per atoms, D, is calculated as

$$D = \int_0^T \sum \sigma_{dg} \phi_g (t) dt,$$

where

- T : length of time of irradiation
- σ_{dg} : Displacement cross section in the neutron energy group 'g'.
- ϕ_g : Neutron flux in the group 'g'.

The term dpa can be related to neutron fluence. The term fluence is reactor dependant and dps becomes reactor independent. Hence, this term is useful while evaluating the structural material damage irrespective of the reactor where it is irradiated. Different models are available for the estimation of dpa. It depends on the burnup, neutron spectrum and fissile enrichment etc.

$\phi\, t \propto$ burnup / fissile enrichment

1 dpa $\approx$ 2 x 10^{21} n/cm^2 (Half Nelson dpa)

Burn up (MWd/t & atom %): Burnup can be expressed usually in two terms. As a measure of the energy extracted from the fuel mass which will be used by the engineering designer for performing the fuel cycle cost analysis, the term MWd/t is used. The fuel mass may refer to only heavy metal atoms or sometimes atoms including oxygen. As a measure of the damage or to assess the behaviour as a function of irradiation, the term atom percent (at %) is used. Approximately they can be related as follows.

Burnup (MWd/Kg) / Burnup(at %) ~ 10

Linear power: Once fuel is selected, maximum linear power, χ (i.e., power extracted from unit length) of a pin is fixed. This is a characteristic feature of a fuel which is given by

$$\chi = 4\pi \int_{Ts}^{Tc} k\, dT$$

T_s : surface temperature of fuel pellet
T_c : centre-line temperature - limited by melting point.

It may also be noted that pin diameter does not appear on the above equation. The objective is to have a very high linear power (χ) without the centreline temperature reaching the melting point with adequate safety margin.

Breeding ratio: Breeding is the process of conversion of fertile material into fissile material. Breeding ratio is defined as the ratio of fissile material produced to fissile material destroyed or consumed in a cycle either through fission or capture.

$$CR = \frac{fissile\ material\ \ produced}{fissile\ material\ destroyed} = \frac{FP}{FD}$$

A reactor is called a breeder if the CR > 1 and it is denoted as BR. There is another term called breeding gain 'G' which is used to estimate the net fissile material produced after accounting for the neutrons that will be consumed for sustaining the reactor which is nothing but;

Breeding Gain "G" = BR –1

Doubling time: 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) is the initial fissile inventory and M_g (kg/y) is the fissile material gained during one year, then the doubling time can be defined as follows. More discussion is given in subsequent sections.

RDT = (M_0/M_g)

Unit energy cost (UEC): UEC is dependant on capital cost (return on equity, interest on investment, depreciation etc), O&M cost (running cost, thermodynamic efficiency), fuel cycle cost (burnup and hence influences annual throughput). A typical cost break-up (PFBR) is shown in Fig.1.8. As the fuel cycle cost is significant, there is every incentive to reduce the fuel cycle cost by aiming for high burnup. Similarly, higher thermodynamic efficiency would lead to reduced O&M cost as it is related to the units of electricity produced.

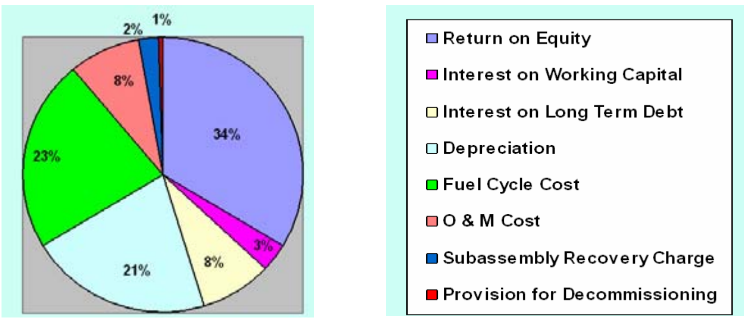


Fig 1.8 : Typical unit energy cost break-up

10.0 SELECTION OF FUEL PIN DIAMETER

The design of a fast reactor starts with broad identification and selection of key parameters like reactor size, fuel type such as oxide, carbide etc, reactor and steam cycle temperatures, core pressure drop w.r.t pump capacity, and the next important parameter to be selected is the fuel pin diameter. It was seen earlier that fast reactor fuel pins thinner compared to thermal reactors. However, the fuel pin diameter to be chosen is based on fuel inventory desired, breeding consideration and economic aspects revolving around fabrication and fissile inventory cost depending upon the primary objective behind the reactor design as it has critical influence with all the parameters mentioned above. Hence, the fuel pin diameter is worked out based on the desired objectives. The influence of various parameters on fuel in diameter is briefly described below.

Once fuel is selected, maximum linear power, χ (i.e., power extracted from unit length) of a pin is fixed for the chosen reactor temperatures. This is a characteristic feature of a fuel.

$$\chi = 4\pi \int_{Ts}^{Tc} k\, dT$$

T_s : surface temperature of fuel pellet
T_c : centre-line temperature

It may also be noted that pin diameter does not appear in the above equation. The objective is to have a very high linear power (χ) without centreline temperature reaching the fuel melting point with adequate safety margin. Hence, the maximum linear power can be chosen accordingly.

Fissile specific inventory consideration:

It is desirable to minimize fissile specific inventory M_o/P, where M_o –fissile mass in the core and P is the core power. Reducing fissile specific inventory reduces doubling time. Let the linear power (χ) be expressed as follows:

$$\chi = P/l = \frac{M_o}{M}\frac{M/l}{M_o/P}$$

where $e = \dfrac{M_o}{M}$ which is fissile enrichment in the core and M/l is linear mass.

Now $\chi = \dfrac{e\dfrac{\pi R_f^2 l}{l}\rho_f}{\dfrac{M_o}{P}}$ where ρ_f - fuel density.

Rearranging the parameters, fissile specific inventory can be expressed as follows:

$$\frac{M_o}{P} = \frac{e\pi R_f^2 \rho_f}{\chi}$$

Once χ is selected, the influence of other parameters is examined as below.

Parameter	Effects
e	Enrichment can not be altered as this influences the criticality. Also increasing ‘e’ reduces breeding ratio and M_o/P increases.
ρ_f	Decreasing ρ_f increases porosity and so leakage is more and required enrichment ‘e’ increases. Also the fuel conductivity ‘K’ reduces which brings down the linear power χ. So, Mo/P increases. [Note: $K_P = K\dfrac{(1-P)}{(1+2P)}$; where P=(1-$\rho/\rho_{th}$) - porosity fraction]. ρ_f increase, brings down neutron leakage and so ‘e’ can be reduced. Also this increases ‘K’ and makes Linear power χ increase. Though M_o/P reduces more than the relative increase in $\rho_{f,}$ this is unfavourable, since porosities are intentionally fabricated into the fuel matrix to accommodate swelling for the targeted burnup.

Parameter	Effects
R_f	R_f decrease, results in 'e' increase, however the combined value eR_f^2 decrease more than the increase in 'e'. This is because the reduction in fissile mass per pin is a function of square of the diameter. This is the only favourable choice to reduce M_o/P to reduce doubling time and the fissile inventory.

Hence, as was seen there is incentive for pin diameter reduction in terms of fissile specific inventory. However, the influence of pin diameter on parameters other than fissile specific inventory are to be examined for a comprehensive view.

Heat transfer consideration:

From the heat transfer consideration, the influence of pin diameter is as follows.

$$Q = \frac{\chi}{\pi D}$$

$Q =$ surface heat flux (W/m^2)

$\chi =$ linear power (W/cm)

D = pin dia (m)

For water cooled reactors, the possible minimum pin diameter is limited by the burnout consideration whereas in fast reactors, due to excellent heat transfer property i.e thermal conductivity of sodium giving rise to heat transfer by molecular conduction, there is no limit on the smaller pin diameter.

Neutron flux consideration:

Reduction of pin diameter increases the neutron flux. As the pin diameter reduces, core volume decreases. For the same power to be maintained, the neutron flux has to necessarily increase. This is explained as follows.

$$\sum_f \phi V \propto P$$

Where P – Core power, Ø - neutron flux, V – volume of core and $\sum_f$ - fission cross section. If the design is carried out for a research reactor for a higher neutron flux for experiment purposes, a lower diameter can be resorted to.

Breeding consideration:

Breeding ratio decreases as the pin diameter is reduced since the fertile inventory comes down with diameter reduction and breeding ratio is directly proportional to fertile mass. Hence, the doubling time is influenced as follows.

Fissile Inventory $\propto$ D^2

Doubling time $\propto$ Inventory/(BR-1)

Both the numerator & denominator increase as D increases. Inventory cost would increase as the pin diameter increases.

Fabrication cost consideration:

With reduction in pin diameter, the number of pins would increase for the same power of the reactor. The increase may be small for small reduction in diameter. But, beyond some point, the pins may become increasingly expensive to fabricate. Also, neutronically it would affect the volume fractions with large increase in steel fraction.

Fabrication cost $\propto$ No. of pins/year $\propto$ $1 / D^2$

Pin bundle spacing consideration:

As the pin diameter is reduced, the pitch by diameter ratio (P/D) which defines the spacing between pins also changes but the change may be small. However, beyond certain dimensions, P/D ratio is to be increased. The increase is warranted because the spacing between pins can not be reduced to very small values from the hot spot temperature consideration. Also, the reduction in pin diameter would lead to larger pin bundle requiring more pumping power to offset for the increase in pressure drop.

In summary, as the pin diameter reduces, the fissile specific inventory comes down and inventory cost would come down. Breeding ratio decreases giving rise to increased doubling time. The fabrication cost will go up with reducing pin diameter. Variation of parameters with pin diameter is shown in Fig.1.9. For a given reactor, all these components would have to be worked out to find out the optimum. Typically around 8-9 mm diameter is reported as the optimum for the fuel pin. In practice, designers choose the diameter depending on the national objective and other constraints and taking into account industrial infrastructure etc.

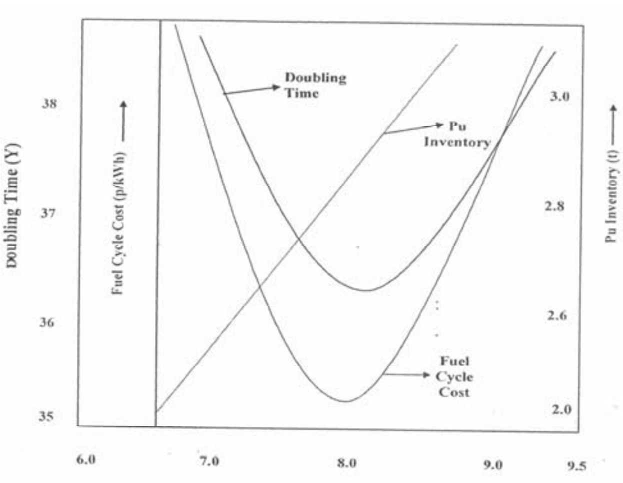


Fig 1.9 Variation of parameters with pin diameter

11.0 CORE HEIGHT

Once the diameter of the pin is fixed, the total pin length follows according to χ from the required thermal output of the reactor. From the core height, then follows the number of pins in the core. Subsequently, the enrichment can be deduced from a neutronic calculation of criticality which in turn yields the neutron flux also.

The important parameters considered for fixing the core height are :

(i) Coolant pressure drop

(ii) Coolant temperature rise

(iii) Sodium void worth

From a neutronic point of view the height H of the core, or more generally the ratio of height to diameter should be near unity (H/D $\approx$ 1). Because of safety reasons, namely the effect of sodium voiding on reactivity, tend to lower this value. In practice, for the plant design of a 1000 MWe LMFBR, the height is often 1 to 1.2 m, the diameter about 2.4 m or more. The main arguments for the core height, apart from void considerations, come from considerations about the temperature rise of the coolant as well as its pressure drop, depending on available pumps etc. In this connection the volume and velocity of coolant in the core are to be fixed accordingly. Generally, for reasons of neutron absorption, the coolant volume part in the core should be as small as possible. This means a high coolant velocity and a low coolant volume part. In practical cases, a coolant volume of about 45-50% of the core volume and a velocity of about 6 – 8 m/s is employed. Both the pressure drop and the rise in coolant temperature increase with the fuel bundle length. Hence, there is general incentive to keep the core height low, resulting in H/D ratios less than unity.

12.0 CHOICE OF MATERIALS

This section covers the important aspects of essentially four different materials employed in the reactor core. The materials that are covered are fuel, structural material for the fuel clad and wrapper, neutron absorber and shielding.

Since the fuel and structural materials are covered in other lectures, only a brief information is provided here.

12.1 Fuel Material

Different candidate fuel materials that can be employed are oxide, carbide, nitride and metallic. The choice of fuel material is to be made based on the design objectives and also probably keeping in mind the infrastructure for the fuel cycle aspects such as fabrication, reprocessing etc in line with the nation's policy. Each type of fuel displays a distinct set of characteristics in terms of its irradiation behaviour and fabrication aspects. Hence, the choice of fuel has to necessarily take into account the above. Of them, the most important one is the irradiation behaviour such as swelling, fission gas release and retention, restructuring etc which have to be taken into account in the design.

Desirable features of an ideal fuel material are as follows:

- High thermal conductivity and high melting point (high specific power can be achieved).
- High radiation damage resistance (high burn up).
- High fuel atom density (fuel volume and core dimension can be reduced).
- Good compatibility with cladding & coolant.
- Negative prompt Doppler coefficient (improves safety)
- No phase change below melting point (stable properties and ease of fabrication)
- Easiness for fabrication (less fabrication cost).
- High neutron yield (high BR & uranium utilization).

A qualitative comparison of the candidate fuel materials is given in Table 1.5

Table 1.5 : Comparison of various fuel materials

Properties	**(U0.8Pu0.2)O2**	**(U0.8,Pu0.2)C**	**(U0.8Pu0.2)N**	**U-19Pu-10Zr**
Theo. Density g/cc	11.04	13.58	14.32	15.73
Melting point ºK	3083	2750	3070	1400
Thermal conductivity (W/m ºK) 1000 K 2000 K	2.6 2.4	18.8 21.2	15.8 20.1	40
Crystal structure	Fluorite	NaCl	NaCl	γ

Breeding ratio	1.1 - 1.15	1.2 – 1.25	1.2 - 1.25	1.35 - 1.4
Swelling	Moderate	High	Moderate	High
Handling	Easy	pyrophoric	Inert atmosphere	Inert atmosphere
Compatibility - clad coolant	average poor	Carburisation good	good good	Eutectics good
Dissolution & reprocessing amenability	Good	Demonstrated	risk of C14	Amenable for pyro re-processing
Fabrication/Irradiation experience	Large Good	few	minimum	few

12.2 Core Structural Materials

In the history of fast breeder reactors, austenitic stainless steel (ASS) to a large extent and ferritic steel to considerable extent have been employed as core structural material for elements such as clad, wrapper and spacer wire. ASS has evolved over the period beginning from 304 and upto 20% Cold Worked D9. In the design of fuel element, structural material assumes equal importance towards ensuring high burnup as it forms the primary boundary. In fact, the limit to high burnup comes from the structural material rather than from the fuel. A few of the performance related issues such as void swelling, irradiation creep, irradiation hardening, ductile brittle transition temperature and their implications have to be considered in order to arrive at an optimum design of core, fuel pin and subassembly. There are ways and means of accommodating the subassembly bowing, wrapper dilation, pin ovality, pin spacing in order to arrive at an optimum design. Advanced materials are currently under development which hold promise for achieving high burnup. The clad materials employed by various reactors is given Table 1.6.

Table 1.6 : Materials selected for clad

Reactor	Country	Fuel clad tube material
Rapsodie	France	316 SS
Phenix	France	316 SS
PFR	U.K.	M316 SS, PE 16
JOYO	Japan	316 SS
BN-600	Russia	16Cr-15Ni-Mo-Ti-Si
Super Phenix-1	France	15-15Mo-Ti-Si
FFTF	U.S.A.	316 SS & HT9
MONJU	Japan	mod 316 SS
SNR-300	Germany	X10 Cr Ni Mo Ti B1515 (1.4970)
BN-800	Russia	16Cr-15Ni-Mo-Ti-Si
CRBR	U.S.A.	316 SS
DFBR	Japan	Advanced austenitic SS (PNC1520)
EFR	Europe	PE16 or 15-15-Mo-Ti-Si
FBTR	India	316 SS

12.3 Absorber Materials

The basic requirements of control system is to compensate for built in reactivity and to provide neutronic shutdown for routine operation and safety measures. The neutron absorbing materials used in control systems must have an adequate high capture cross section for the energy spectrum within the reactor. The material should be capable of having a long life before losing its capacity to absorb neutrons or losing its properties from a materials point of view by the damage caused by the products of the neutron capture reaction. The neutron absorber must also be compatible with the material in which it is clad, not only at the temperatures used during the fabrication of the control rod but also for an extended time at reactor operating conditions. It is generally desirable that the control material should be unaffected chemically by the coolant environment within the reactor. The control rod itself should have a low mass in order to facilitate its movement in the reactor during operation. At the same time it must have adequate mechanical strength to withstand the forces to which it is subjected during shutdown. The cost of the control rod on annual basis for a specified service life should be as small as possible.

The selection of a control material is made on the basis of how the material fulfills most of the requirements. Considerable attention has been given to borides as a group because they meet most of the requirements for fast reactors. Boron has good cross section for use as control material because of high cross section of B-10 isotope. Fortunately its cross section is not so high as to cause it to burn up too fast. Borides are generally very stable compounds. They have desirable refractory characteristics. But one of the difficulties in using is the fact that high energy helium and lithium atoms are produced. Helium has to be accommodated in a way that will prevent significant dimensional changes in the control rod. Helium retention capabilities of borides is an important issue. The candidate di-borides are mostly transition metal borides such as Titanium, Zirconium and Hafnium. Among tetra borides, boron carbide is the most important. Yttrium and Dysprosium have also been considered. Hexaborides of the rare earths samarium, europium and dysprisum have also been considered because of their loose structure and ability to accommodate helium. Rare earth oxides have also been considered. Europium and dysprosium oxides have received consideration. Rare earth oxides are very stable but they are readily hydrated and have to be cladded well.

In addition to the material behaviour, the design of absorber pin also has to take into account its behaviour under irradiation and consequent design choices such as vented pin type etc.

12.4 Shield Materials

Shields have to provide protection to structures and personnel. The neutrons leaking out of core of a fast reactor are energetic and providing shielding to these neutrons is an optimization problem with many materials. Materials good for absorbers are also good for shields. Since quantities of shield materials to be used are large, shield material selection has to be made on the basis of both quantitative and economic considerations. Several materials arranged in layers usually make the optimum arrangement of shields in the reactor. Shields are, in most of the designs, are not removable over lifetime of the reactor and hence compatibility with coolant environment and containers is always important.

Combination of materials that slow down neutrons and those that absorb provide best optimized shields. Using hydrogeneous materials in sodium environment is almost ruled out. Neutron slowing down in sodium environment is best accomplished by efficient neutron inelastic scattering nuclides. Uranium or thorium in blankets are efficient inelastic scattering materials. Stainless steel is also a material of choice as an inelastic scattering material. Among the materials for absorption, boron carbide is prominent. Borides and alloys containing boron are also being considered.

Heavy density materials are good for shielding against gamma rays. But from considerations of cost and ease of fabrication, lead, iron, steels and concrete are materials of choice depending on the space and economics considerations. Lead and iron are used in granular or cast form. Use of high density concretes is common. Dual use of depleted Uranium as structural material and effective gamma shield for spent fuel transport casks has also been considered.

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