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Physics and engineering aspects of fast reactor safety

C V SUNDARAM and S M LEE
Reactor Research Centre, Kalpakkam 603 102, India

Abstract. A vast amount of research and development work has been done in recent years to resolve the issues of relevance to the safety of the Liquid Metal cooled Fast Breeder Reactor (LMFBR). Based on the results of this research as well as on the experience gained from the operation of test and prototype LMFBR's, a certain consensus is emerging on the safety requirements of a modern sodium-cooled large fast power reactor. The paper reviews the fundamental physics and engineering aspects of LMFBR safety with reference to the Fast Breeder Test Reactor (FBTR) now being commissioned at Kalpakkam, and the proposed larger Prototype Fast Breeder Reactor (PFBR). The elements contributing to the inherent safety of fast reactors are recapitulated followed by description of the philosophy of the plant protection system and the use of engineered safeguards to enhance the safety. Finally, the principles used for the containment of radioactivity are discussed.

Keywords. Fast breeder reactors; safety requirements; engineered safeguards; containment of radioactivity.

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1. Introduction

The fundamental concerns of nuclear reactor safety are apparent in the basic nuclear fission chain reaction itself. In this reaction, a neutron splits the nucleus of an atom of a fissile material like uranium or plutonium into two smaller radioactive nuclei (fission products). At the same time, a considerable amount of energy in the form of kinetic energy of the fission product nuclei is released, along with nuclear radiations, and several additional neutrons. The released fission neutrons, which are emitted in a spectrum of energies peaking around 2 MeV, can induce fission in further atoms causing a chain reaction, which process is used for heat production in a nuclear reactor. It is immediately seen that the problems of safety specific to nuclear reactors would have their origin in the following:

a) control of the chain reaction and reactor geometry to prevent unchecked growth of neutron population that can cause heat production in excess of the heat removal capability,
b) shielding of the nuclear radiations produced during the chain reaction,
c) containment of the radioactive fission products, shielding of their nuclear radiations, and provision of cooling to remove their radioactive decay energy when the reactor is shut down.

To appreciate the development of concepts in nuclear reactor safety, it is necessary to recall the events that have occurred in the short period of time that has elapsed since the discovery of the fission reaction in 1939. The possibility of using low energy neutrons for fission of U^{235} and the need for high energy neutrons for fission of U^{238} or thorium was shown in the same year, as also the existence of fission neutrons and delayed neutrons. This was followed by Fermi attempting and failing to sustain a chain reaction in a light water moderated natural uranium assembly. The brilliant theorising and experimentation of Fermi's group led to the important decisions to use graphite as moderator as well as to lump the fuel to minimise the loss of neutrons by non-fission reactions in U^{238}. This led to the successful atomic pile CP 1 which went critical on December 2, 1942. The reactor used 6 tons of pure uranium metal, 50 tons of compressed uranium dioxide powder, 400 tons of graphite and a number of control rods consisting of cadmium sheets nailed to wood. In addition to shim control rods there were cocked safety rods, which could be released to fall into the core automatically, should the neutron intensity as measured by a BF_3 gas filled ionization chamber rise too high.

After this historic achievement, several other reactors were constructed in the USA driven by the need for plutonium for weapons. The first enriched uranium, light water moderated reactor was built in 1944. The first atomic bomb (Trinity test) was exploded on 16th July 1945 and was a plutonium device. This was followed 21 days later by the Hiroshima U^{235} bomb and a further 3 days later by the Nagasaki plutonium bomb.

Canada achieved its first chain reaction in the heavy water moderated reactor ZEEP in 1945, UK achieved it in the graphite moderated pile GLEEP in 1947, France in the heavy water moderated reactor ZOE in 1948 and India achieved its first chain reaction in the APSARA reactor in 1956.

The first controlled fast chain reaction was in 1946 in the 25 kWt CLEMENTINE fast reactor of USA using plutonium fuel and mercury coolant. The first nuclear electricity production was also from a fast reactor, the enriched uranium fuelled, NaK cooled 1 MWt EBR 1 which went critical in 1951. India achieved its fast chain reaction in the PURNIMA reactor in 1972.

In general, the safety of a reactor is ensured by parallel approaches at different levels providing a 'defense in depth'. The first level is the design of an inherently safe plant with an adequate control system which can operate with a high degree of reliability. In addition, special emphasis is placed on the quality of materials and workmanship of components, with provision for continuous or periodic inspection of components and sub-systems. The careful design and construction is followed by equally careful operations so as to reduce the probability of accidents to a very low level.

At a second level, safety is provided by means of a comprehensive plant protection system (PPS) which can safely handle a wide range of conceivable abnormal incidents and malfunctions and safely shut down the plant. The PPS includes a variety of instruments and sensors to monitor the state of the plant and to take protective as well as sympathetic control actions to prevent any abnormality leading to an accident.

The third level of safety is provided by engineered safety features such as reactor containment vessel and building, secondary shutdown systems, emergency core cooling system, etc., which limit the consequences of certain highly unlikely accidents, which are postulated to occur in spite of the first and second level safety measures. Finally, a careful choice of site is made, and exclusion areas defined such that routine as well as accidental radioactive releases do not affect the general population.

In the following discussion, we elaborate on these elements of reactor safety with reference to the fast reactor. The discussion will start with physics, control and engineering design features providing the inherent safety of a modern fast reactor, then go on to the main components of the PPS. Finally, the principles of the containment of radioactivity under normal and accident conditions will be discussed.

The 40 MWt/13 MWe loop type Fast Breeder Test Reactor (FBTR) now being commissioned at Kalpakkam and the proposed 500 MWe pool type Prototype Fast Breeder Reactor (PFBR) will be used as examples during the discussion and their safety features will be highlighted *vis-a-vis* the international approaches to fast reactor safety.

2. The fast reactor

2.1 *Thermal and fast chain reactions*

The relative probabilities of the production of neutrons by fission in a reactor, and the loss of neutrons by other nuclear reactions as well as escape from the reactor govern the possibility of maintaining a chain reaction. Of all the naturally occurring isotopes of elements, only U^{235} is fissile and can sustain a chain reaction. U^{235} has an abundance of only 0·7% in natural uranium, the rest being the isotope U^{238}. However, the U^{235} content can be artificially increased in enrichment plants to produce enriched uranium. In order to sustain a chain reaction in natural uranium or low enrichment uranium, it is necessary (see for example, Glasstone and Edlund 1952) to reduce the energy of the fission neutrons by means of scattering collisions with suitable light moderator atoms, till the neutrons attain low energies (around 0·25 eV) at which they are in thermal equilibrium with matter. These thermal neutrons are then allowed to cause further fission and continue the chain reaction.

If a chain reaction is to be maintained in a reactor without a moderator, then it is necessary to use high enrichments ranging from 15% (for large reactors) to over 90% (for small reactors). Here the chain reaction is maintained by the fast moving fission neutrons. There are several advantages of the fast chain reaction over the thermal one. The first is that fast reactors are very compact on account of the absence of the large bulk of moderator which is required in thermal reactors. Fast reactors are much more compact than the light water moderated package power reactors used in nuclear powered sea-craft, and have been considered for space craft applications (Hanson 1984). In addition, the fast chain reaction is used in nuclear explosives applications and in pulsed fast reactors providing a large number of fast neutrons for nuclear research (Oakes 1968).

However, the most important reason for the large scale development of fast reactors is the efficient breeding of new fissile material, plutonium and U^{233}, from their respective fertile materials, U^{238} and thorium. The fertile materials are several hundred times more abundant than U^{235} and can be converted to fissile material by neutron absorption in a reactor. A breeder reactor is designed such that it generates more fissile material from fertile material than the fissile material it consumes for energy production. Without breeder reactors, it will be possible to use only a small fraction of the uranium resources. Hence all plans for the long-term utilisation of uranium and thorium resources are based on the breeder reactor. While there are designs for thermal breeder reactors using the Th/U^{233} fuel cycle, for efficient breeding the fast chain reaction is the best, mainly on account of the fact that the number of neutrons released per absorption in fuel is higher for higher neutron energies.

2.2 *The* LMFBR

There are different kinds of fast reactors and the best developed is the Liquid Metal cooled Fast Breeder Reactor (LMFBR). In this type, heat is generated by the fission chain reaction in a core consisting of fuel subassemblies supported vertically on a grid plate in a reactor vessel. Each subassembly is a bundle of fuel pins made of stainless steel clad tubes of small diameter (less than 1 cm) containing the enriched fuel pellets. The fertile material in the fuel as well as on top and bottom of the fuel (axial blankets) get converted to fissile material by neutron irradiation. In addition, radially surrounding the core are blanket subassemblies containing additional fertile material for conversion to fissile material.

Water is a neutron moderator, and rapidly decelerates the fast moving fission neutrons to less than the energy suitable for breeding, hence it cannot be used as a coolant. The problem of removing a large amount of heat from the small compact core of a fast reactor by a coolant which does not reduce the neutron energy unacceptably is solved in the LMFBR by the use of liquid sodium. This coolant has excellent heat transfer characteristics and good chemical compatibility with fuel and structural materials. The reactivity of sodium with air or water is taken care of by providing an inert cover gas like argon over free sodium surfaces. In early small experimental LMFBR's, mercury, potassium and a eutectic mixture of sodium and potassium (NaK) have been also used as coolants.

The sodium coolant is pushed by pumps through the core and gets heated. The hot primary sodium is radioactive and is not used to directly produce steam, but rather transfers the heat to secondary sodium through intermediate heat exchangers (IHX). The non-radioactive secondary sodium flows through sodium heated steam generators to produce steam which drives turbo-generators. In the pool type layout, the whole primary coolant circuit along with primary pumps and IHX's is contained within the reactor vessel as shown in figure 1. In the loop type layout, the primary pumps and IHXS are outside the reactor vessel (figure 2) though the whole radioactive primary circuit is contained within the reactor containment building. Both types of layout are prevalent though they have differing safety implications.

The chain reaction is usually controlled by the use of the neutron absorber boron, in the form of boron carbide control rods. Fuel loading and unloading is invariably done under shutdown conditions, from the top of the reactor vessel by means of fuelling machines handling one subassembly at a time.

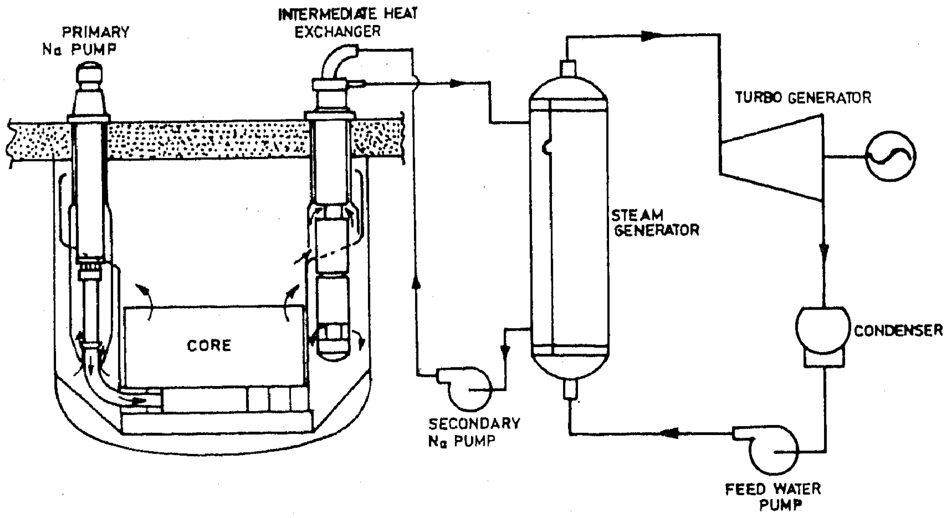

Figure 1. Schematic diagram of a pool type LMFBR.

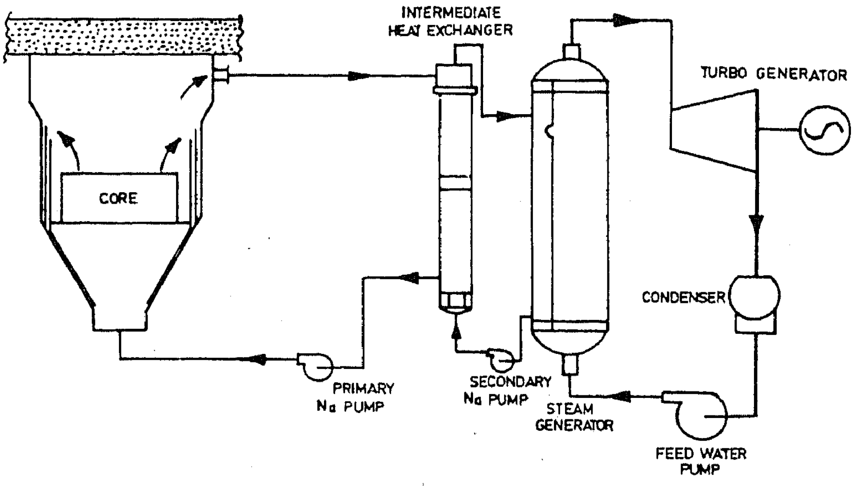

Figure 2. Schematic diagram of a loop type LMFBR.

Table 1. Some characteristics of FBTR and PFBR.

	FBTR	PFBR
Reactor thermal power (MWt)	40	1250
Electrical output (MWe)	13	500
Primary circuit concept	Loop	Pool
Fuel material	Mixed carbide of Pu and U	Mixed carbide of Pu and U
Pin outer diameter (mm)	5·1	8·8
Active core height (cm)	32	100
Volume of active core (litres)	46·7	2739
Maximum neutron flux (n cm^{-2} sec^{-1})	$3{\cdot}7 \times 10^{15}$	$8{\cdot}6 \times 10^{15}$
Effective delayed neutron fraction	0·0028	0·0037
Prompt neutron lifetime (sec)	2.3×10^{-7}	3.6×10^{-7}
Number of control rods	6	12
Inlet sodium temperature (°C)	380	380
Outlet sodium temperature (°C)	515	530
Number of primary pumps	2	4

The main characteristics of FBTR and of PFBR are given in table 1. FBTR is a mixed plutonium-uranium carbide fuelled LMFBR with two primary loops. PFBR is planned to be a pool type LMFBR using either mixed carbide or mixed oxide fuel and incorporating state of the art safety features.

3. Inherent safety features

3.1 *Neutron population and multiplication factor*

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 within 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.

3.2 *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. 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. 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. The average rate of loss of neutrons is 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} sec 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 neutron 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 (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 (Keepin 1965). 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 $l^* = (1-\beta)\,l + \beta\tau$. 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.

There are two more important effects connected with the delayed neutrons:

(i) When a reactor is shut down by insertion of control rods such that k is reduced much below 1, the power at first falls rapidly to a fraction $\beta/(1-k+\beta)$ of the operating power, then falls gradually towards zero with a period close to the mean life of the longest lived delayed neutron precursor, *viz* 80 sec. Thus it is not possible to completely shut off the chain reaction immediately.

(ii) 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.

3.3 *Feed-back coefficients of reactivity*

One of the important control and safety features of a nuclear reactor core is that the reactivity is dependent on the core temperature. As seen above, when k is changed from 1, by say control rod movement, the neutron population changes and consequently the power P changes. This leads to changes in the core temperature T which, in turn, affect k and thereby the power. Thus, we have a feed-back loop.

When the feed-back change in k opposes the applied change in k, then the reactor has a negative feed-back coefficient. In other words, any addition of reactivity increases power which increases temperature, and which in turn decreases the reactivity. Thus, negative feed-back coefficients lead to stability of operation and add to the inherent safety of the core. On the other hand, a positive feed-back coefficient can lead to instability and power runaways. It is to be noted that the core temperature can also be changed by changes in the coolant flow rate or the coolant inlet temperature. Designing the reactor core for all negative feed-back coefficients of reactivity is an important element in the overall safety of the reactor.

Different mechanisms with differing response times of action contribute to the feed-back. The response times for the feed-back depend on the time taken for the temperatures of the various core components to change. These times are obtained from heat transfer equations relating the component temperatures to the power, coolant flow rate and the coolant inlet temperature.

The precise features of the feed-back mechanisms vary from reactor type to reactor type. In a fast reactor, the three important mechanisms of reactivity feedback under normal operating conditions are the Doppler coefficient, expansion coefficients and differential expansion coefficients. Additional mechanisms of reactivity feed-back under accident conditions are by coolant voiding and fuel slumping.

3.3*a* *Doppler coefficient of reactivity:* The probability of a neutron being absorbed by or causing fission of an atom varies with the energy of the neutron and in a range of energies (0 to 20 keV), there are sharp peaks in these probabilities called resonances. There is a basic theorem (see for example, Hummel and Okrent 1970) of reactor physics that the rate of absorption of or fission by neutrons in the resonance energy range increases if the atoms are themselves in motion. Thus, when the temperature of the reactor core materials increases, the increased thermal motion of the atoms causes the reaction rates to increase. Consequently, the core reactivity changes giving rise to the Doppler coefficient of reactivity.

For fertile materials U^{238} or thorium, the predominant reaction is the absorption of neutrons and hence the Doppler effect in these materials increases the absorption of neutrons and reduces the core reactivity, *i.e.* the feed-back is negative. On the other hand, for fissile materials both the absorption and fission rates are increased. The increased loss of neutrons by absorption is compensated by the increased production of neutrons by fission so that the net result of the Doppler effect in fissile materials is very small. The total Doppler coefficient of the reactor is the sum of the different contributions of the fissile and fertile nuclides in the fuel, with the contribution from the fertile material being dominant.

The following are the important characteristics of this coefficient:
(i) It is a prompt coefficient related to the fuel temperature, which temperature is the first to increase when the power increases.
(ii) In a small fast reactor with high enrichment, the neutrons have a short lifetime as they are quickly absorbed in the highly enriched fuel or quickly leak out of the small reactor. Hence the average neutron energy is high, close to that of the fission neutrons. The resonance energy region is about 100 times lower than the fission neutron energies and in such small reactors, there are very few resonance energy neutrons so that the Doppler effect is negligible. Thus, for example, there is a negligible Doppler coefficient in the FBTR.
(iii) In larger reactors of lower fuel enrichment and smaller neutron leakage, the neutrons have a longer life so that an appreciable number of neutrons slow down into the resonance energy region. The Doppler coefficient is thus large and is important for the stability and safety of the reactor. The coefficient is largest in oxide fuelled fast reactors which have low average neutron energies on account of the presence of two moderating atoms (oxygen) per atom of fuel.
(iv) The calculation of the Doppler coefficient in fast reactors is difficult on account of the small fraction of neutrons entering the resonance energy range from the fission neutron energies. Detailed calculations with accurate nuclear data are required (Nicholson and Fischer 1968) to establish correctly the number density of neutrons in the resonance energy range. This small fraction of neutrons contribute little to the overall neutron balance, but they are the main cause of the Doppler reactivity effect. Many years of worldwide effort have resulted in an ability to calculate this coefficient with an accuracy of about 20% (Ganesan 1979).
(v) The increase in reaction rates, as the thermal motion of the target atoms increases, tends to get saturated at higher temperatures. The Doppler coefficient hence decreases as the temperature increases. Commonly, a fitted linear decrease is used, so that $T\mathrm{d}k/\mathrm{d}T$ will be a constant called the Doppler constant which for the larger fast reactors varies from $-0{\cdot}001$ to $-0{\cdot}01$ according to the fuel and size of the reactor. For the carbide fuelled version of PFBR, the Doppler constant is calculated to be $-0{\cdot}0064$ for normal operating conditions and $-0{\cdot}0046$ under coolant voided conditions.
(vi) A very important series of experiments in connection with the Doppler coefficient were conducted in 1972 in the SEFOR LMFBR of USA (Harris 1973). This was a 500 litre oxide fuelled LMFBR with neutron spectrum and fuel temperatures representative of large LMFBR's. The experimental programme was designed to demonstrate that a prompt negative Doppler coefficient of sufficient magnitude makes the LMFBR an extremely stable and inherently safe reactor system. The super prompt critical transient tests involving reactivity insertion of upto 1·3$ in 0·1 sec provided a convincing demonstration of the effectiveness of the Doppler coefficient in limiting the energy release in a fast reactor power excursion.

3.3*b* *Expansion coefficients of reactivity*: A rise in temperature causes the different materials in the fast reactor core to expand thereby reducing the number of atoms per unit volume. Reaction rates depend on the product of neutron density and the number density of atoms, and so all reaction rates decrease and the fraction of neutrons leaking out of the core increases. The effect on the reactivity varies according to the type of material suffering expansion.

Fuel density reduction leads to loss of reactivity on account of reduced fuel-neutron interactions and increased neutron leakage. On account of the gap between fuel pellets and the clad inner wall, the radial fuel density reduction is in fact governed by the radial expansion of the structural steel. The fuel pellets' axial expansion is governed by the fuel temperature and is a fast acting coefficient. In small fast reactors, the Doppler coefficient is negligible and so the radial and axial fuel expansion coefficients assume importance. The axial fuel expansion coefficient for ceramic fuel changes as the fuel structure changes due to irradiation, and the study of its behaviour with burn-up of the fuel is important. Much useful information on the behaviour of this coefficient with burn-up of mixed plutonium-uranium-oxide fuel has been obtained from the operation of the French reactors RAPSODIE and PHENIX (Berthet *et al* 1982; Clauzon *et al* 1976).

The expansion of the sodium coolant has several opposite effects on the reactivity:

(i) Reduction in the absorption of neutrons by sodium leading to a small gain in reactivity.

(ii) Reduction in the neutron moderating scattering collisions with sodium atoms. Hence, the average energy of neutrons in the reactor increases. In fast reactors, when the neutron energy increases the number of neutrons released per neutron absorption in the fuel increases, and the parasitic neutron absorption in structural and coolant materials decreases. Thus, the reactivity of the core increases.

(iii) Increase in the leakage of neutrons from the core on account of reduced sodium-neutron scattering interactions, leading to a loss of reactivity.

In small reactors like FBTR, the third effect is dominant so that the sodium expansion coefficient is negative. In larger reactors like PFBR, the third effect is small compared to the second one so that the coefficient becomes positive.

The expansion of steel material in the core causes effects similar to that of sodium, though in this case, as steel is composed of atoms much heavier than sodium, the effect due to change in neutron moderation is small and those due to change in leakage and absorption dominate. This expansion coefficient is negative in small reactors and becomes positive in large ones.

It is difficult to distinguish experimentally the differing speeds of response of the expansion coefficients associated with the fuel, structure and coolant in a fast reactor core and a total expansion coefficient with a single time constant of response is found adequate to describe the kinetic behaviour. This total expansion coefficient is always negative and has a response time of 1 to 5 sec. In FBTR, the total expansion coefficient has a value of about $-4 \times 10^{-5}/°C$ with a response time of about 2 sec. These expansion coefficients can be relied upon and are important for turning round slow reactivity increases and preventing system upsets leading to unsafe conditions.

3.3*c* *Differential expansion coefficients:* On account of leakage of neutrons, the neutron density and consequently the power is greater towards the centre of the reactor than towards the boundary. Further, the coolant enters at the bottom of the core, gets heated and leaves at the top. Thus, there are thermal gradients across the core and changes in these gradients can cause reactivity changes. The major effects are radial bowing of the fuel subassemblies governed by the radial temperature gradient changes; and relative movement between the control rods and core governed by the temperature difference between the hot sodium at the top in which the control rod shafts dip, and the cooler sodium at the bottom around the core support structures.

Inward bowing of the fuel subassemblies leads to an increase of density of core materials in the core centre, where the neutron density is higher. This leads to an increase in reactivity. Such a positive bowing coefficient was present in the early fast reactor EBR-1 and led to a core meltdown accident (Thompson and Beckerley 1964) during special experiments for measuring the feed-back coefficients. Modern designs for LMFBR's incorporate spacer pads at suitable levels between fuel subassemblies to make the bowing coefficients zero or slightly negative. Thus FBTR has a slightly negative bowing coefficient and PFBR would be also designed to have a negative one.

The reactivity coefficient associated with control rod shafts expansion is a fairly large negative one but acts after a time lag. In the case of FBTR, it is expected to have a value of $-2\cdot6\times10^{-5}/°C$ with a response time of 25 sec.

3.3*d* *Sodium void coefficient:* Exactly similar arguments as for the reactivity change during the expansion of sodium show that there will be a reactivity change if the sodium coolant is accidentally voided from the core. This is called the sodium void coefficient and is negative for small cores like FBTR and positive (+3 dollars) for large cores like PFBR. Obviously, there are strong safety implications associated with the positive coolant voiding coefficient, whose value can attain several dollars for a large power reactor. Comprehensive theoretical and experimental investigations for the prediction of this coefficient have been made leading to a satisfactory understanding (Küsters and Ganesan 1978). Currently, the successful designs to have a lower sodium void coefficient are the so called heterogeneous cores in which internal blanket material zones are placed within the core (see for example, Inoue *et al* 1982). In this case, voiding leads to neutron leakage into the internal blanket zones thus reducing reactivity. However, no heterogeneous core fast reactor has been operated. The alternate approach has been to maintain a positive voiding coefficient and ensure by proper safeguards and design that the sodium cannot be voided suddenly. This has been found particularly easy for pool type reactors where the possibility of loss of sodium by pipe break is not possible and where due to thermal inertia, there is considerable time lag for the sodium to boil when there is a loss of flow. Thus, for example, both for the 250 MWe PHENIX and the 1200 MWe SUPERPHENIX reactors, a loss of electric supply to the primary and secondary pumps followed by a failure to shut down the reactor does not result in sodium boiling (Gouriou *et al* 1982).

3.3*e* *Fuel slumping reactivity gain*: Unlike thermal reactors, in a fast reactor the spacing between the fuel pins is governed by heat removal considerations and not from reactivity considerations. Compaction of fuel tends to reduce the neutron leakage and increase the reactivity. Hence, fuel melting and slumping is the most serious accident in a fast reactor. Traditionally, the worst rates of reactivity addition were calculated from a gravitational collapse of the core assuming gross core melting due to loss of flow of coolant. The total reactivity addition can be several tens of dollars and the rate of reactivity addition can be of the order of a hundred dollars per second. The smaller the reactor, the greater is the reactivity gain due to slumping. It is to be noted that the values for the slumping reactivity gain and the rate of reactivity addition given above are extreme theoretical values and realistic modelling of accident sequences does not give such values. However, it was these high values which were the early causes for concern and gave the impetus for extensive fast reactor safety research.

3.4 *Engineering features*

There are many engineering design features which enhance the inherent safety of the LMFBR. An important feature is that the sodium system is a low pressure system and the maximum sodium temperature ($\sim 550°C$) is far below the sodium boiling point. Further, sodium has extremely good heat transfer and natural convection properties which ensure good decay heat removal even under accident conditions. General design features enhancing the safety are (Graham 1971):

(i) use of special engineering design codes for fast reactors with appropriate safety factors,

(ii) limited rates of control rod withdrawal,

(iii) use of diverse and redundant shutdown systems,

(iv) provision of flywheel action in main sodium pumps to maintain flow under loss of power,

(v) provision of alternate grid supplies,

(vi) use of multiple radial inlets to subassemblies to avoid flow blockages,

(vii) use of wrapped and restrained piping to avoid pipe breakage; use of safety guard tanks around components like reactor vessel, IHX, pumps, etc.

(viii) use of double walled piping, check valves, hydraulic diodes, reserve coolant volumes, etc.

The design features providing inherent safety in FBTR are diverse. The coolant entry to subassemblies is *via* multiple (twelve) openings which cannot get blocked simultaneously—even with 50% blockage the flow reduction is only 10%. Hold-down springs are provided at the feet of subassemblies even though the sodium fluid dynamic forces are insufficient to lift a subassembly. In addition to flywheels which are provided on pumps to prevent abrupt flow stoppages, diverse and independent power supplies feed the pumps including emergency batteries and diesel generators. The general safety features which would be incorporated in PFBR include a pool type design using a large double walled vessel to house the reactor. Multiple inlets for sodium entry into subassemblies are to be provided. Emergency power supplies, flywheels on pumps and two independent shutdown systems will be provided.

4. The plant protection system

4.1 *Elements of the plant protection system*

Despite the assurance offered by careful design, construction and operation, the second and most important level of safety is the PPS, which can handle a wide range of conceivable abnormal incidents and malfunctions and safely shut down the reactor. Basically the system consists of:

(i) a wide variety of instruments to monitor the plant operating parameters and characteristics,

(ii) assured shutdown systems triggered by signals from the monitors,

(iii) decay heat removal systems.

Core damage may occur if the shutdown systems or the decay heat removal systems fail on demand. Hence proper operation of the PPS is very important. This is assured by means of redundancy, diversity and fail-safe operation of the components of the PPS.

Redundancy refers to the use of two or more similar components in parallel by means of which it is assured that the PPS does not fail when needed, and at the same time, false plant shut-downs are not caused by instrument errors. Diversity in shutdown systems, electric power systems, etc. is provided to prevent common mode failure, *i.e.* a number of failures resulting from a single cause. By fail-safe operation is meant that failure of a component of the PPS results in shutdown of the plant rather than continued unprotected operation. Recent designs, like the SNR 300 and SUPERPHENIX, provide for two independent plant protection systems (Birkhofer *et al* 1982; Berlin *et al* 1982).

The PPS often includes a process computer which monitors and processes the information of all signals relevant to the safety of the plant. In FBTR, this is performed by the Central Data Processing System (a pair of ECIL TDC-316 computers) which can also initiate alarms and safety actions.

4.2 *Plant monitoring systems*

The most important monitoring systems in a fast reactor plant are for neutron and; sodium temperature and flow monitoring; and clad rupture detection. Other monitoring systems are for sodium levels and sodium leaks and aerosol monitoring; and steam generator leak detection.

4.2*a* *Neutron monitoring:* Neutron detectors and associated electronics are required in order to monitor and control the power level as well as to provide signals for safety action. The neutron monitoring system provides signals for alarms, power reduction or reactor shutdown when the neutron density level is too high or the rate of increase of neutron density is too high. In addition, most LMFBR's contain reactivity meters which use measured neutron density as a function of time to derive the reactivity changes, which can also signal for safety action if found excessive. The neutron monitoring system is triplicated and safety action is taken when any two of the three monitors call for it.

On account of high gamma radiation fields, all neutron detectors should discriminate against gammas. Both for FBTR and PFBR, the neutron detectors considered are BF_3 counters, boron coated counters, fission counters and boron coated compensated ion chambers. Location of the detectors outside the primary vessel is desirable from considerations of temperature and radiation. In FBTR, the detectors are located at the start of the radial concrete shielding outside the reactor vessel. For PFBR, location of detectors under the main vessel is under consideration. Further, for a pool type reactor like PFBR, adequate neutron levels are not obtained outside the main vessel for shutdown monitoring and use of in-vessel detectors is to be considered.

4.2*b* *Sodium temperature and flow monitoring:* Monitoring of sodium temperatures and early detection of sodium boiling is very important for core safety especially with positive coolant voiding coefficients. Most commonly, thermocouples (Chromel-Alumel or Ni-Cr) and Resistance Temperature Detectors (RTD) are used. In FBTR, each fuel subassembly is monitored by 2 thermocouples and a similar system may be considered for PFBR.

Sodium flows are measured both by Venturi flow meters and magnetic flow meters. While the former are very accurate, their response time is too slow for use in the PPS, which purpose is served by the latter.

Trip signals are actuated by high sodium temperature, low flows, mismatch of power to flow values, loss of electric power supply, etc.

The onset of sodium boiling is also sought to be detected in modern LMFBR's by the methods of neutron noise and acoustic noise detection. Bubbles formed when sodium boils give rise to both reactivity fluctuations (see the discussion in §3.3b) as well as to pressure variations in the metal. These fluctuations can be recorded by neutron monitors or by pressure transducers and the onset of boiling detected.

4.2*c* *Clad rupture detection:* Detection, localisation and removal of failed fuel with ruptured clad is required to prevent excessive contamination of the primary coolant and to prevent flow blockages that could arise on account of gradual fuel swelling when sodium contacts the fuel. Both in FBTR and in PFBR, the methods of failed fuel detection are to be by monitoring of the cover gas fission product activity and by monitoring of delayed neutrons in the primary coolant. Excessive activities from these monitors trigger a plant shutdown.

The cover gas fission product activity is due to gaseous fission products like Xe and Kr escaping from ruptured fuel pins. These gases require time to reach the cover gas plenum and the response time for clad rupture detection by cover gas monitoring is of the order of minutes. On the other hand, fission products like I^{137} and Br^{87} which are delayed neutron precursors get dissolved in the sodium upon clad rupture and the emitted neutrons are detected in a bypass loop of the primary circuit. The response time of this system is a function of the coolant transport time and is in the range of 20–30 sec. Identification of the failed fuel subassembly is an important problem and one of the methods followed in some reactors is to sample the sodium from each subassembly by selector valves and monitor the delayed neutrons.

4.3 *Reactor shutdown systems*

Besides the function of reactivity or power adjustment, the control rods must be able to shut down the reactor under any foreseen conditions. It is important to assess the mechanisms of accidental reactivity insertions and ensure that the shutdown system can compensate for these. Such mechanisms include melting of a fuel subassembly, sodium voiding, inadvertent withdrawal of a control rod, introduction of moderator, sudden flow increases, etc.

In FBTR, the reactor control is by six enriched boron carbide control rods. The power to drive the rods is supplied independently to each of the six control rod drive mechanisms and in the event of power failure or a rapid shutdown trip action the drive mechanisms and rods fall under gravity in about 400 m sec to shut down the reactor. The electrical connections are such that only one rod can be withdrawn at a time at a rate of 1 mm/sec which limits the reactivity addition rate to a low 6×10^5/sec.

As in modern LMFBR's, PFBR will have two independent shutdown systems. The primary shutdown system is expected to consist of 9 control rods to be used for normal operations as well as for rapid shutdown. In addition, a completely independent secondary shutdown system of 3 control rods based on a different design principle would be able to override abnormal reactivity additions and shut off the reactor even if all the primary rods fail to act. Their design would be such that they can enter even distorted cores, and would automatically operate on sodium overheating due to Curie point demagnetisation of magnets. Alternate design principles for the secondary system under consideration are:

(i) Articulated links with spherical couplings (Pignatelli *et al* 1982).
(ii) Hydraulically suspended absorber balls (Specht *et al* 1976).

Fuel loading and unloading in an LMFBR is done with all control rods fully inserted and disconnected from their drive mechanisms. It is important to ensure that the core is not made critical during refuelling. Hence, accurate measurement of the degree of subcriticality of the shutdown core and the reactivity worths of the loaded subassemblies is important.

4.4 *Decay heat removal*

The decay heat is initially about 6 to 7% of the reactor power and gradually falls with time becoming around 1% after an hour. An important part of the PPS is the provision of adequate decay heat cooling under shutdown conditions for as long as necessary (up to a month or more) even under conditions of coolant system failures.

Decay heat removal in fast power reactors is ensured by a diversity of means (Lauret *et al* 1982; Agarwal and Guppy 1981):
(i) multiplicity of primary heat transport loops ensuring normal decay heat removal by the operation of a single loop alone,
(ii) redundancy and diversity in power supplies to pumps,
(iii) forced circulation in the immediate period following shutdown (when decay heat generation rates are high) by design provisions, such as flywheels for gradual coastdown of pumps,
(iv) continued forced circulation by low speed operation of pumps by means of pony motors backed by alternate power supplies/emergency diesel generator sets/batteries,
(v) provision of independent back-up decay heat removal systems,
(vi) design of normal as well as back-up heat removal systems, such that setting-up of good heat removal by natural convection is facilitated.

In pool type LMFBR's, the large thermal capacity of the sodium bulk acts as a heat sink and allows a long time interval (at least several hours) for alternate measures to be taken before the sodium boils. The possibility of complete loss of sodium from the primary heat removal circuits is also not present in such reactors.

For FBTR, decay heat removal is ensured by a variety of measures starting with the operation control that the reactor will be operated only with both heat transport loops. Thus one loop will always be available to remove decay heat under upset conditions. Further, emergency batteries and diesel generators provide power for the pumps on grid non-availability. Natural convection is found to be able to remove about 300 kW of heat with none of the pumps operational. In case the primary circuit is not available due to sodium leak, decay heat removal is by the emergency cooling system, where cooling nitrogen is circulated in the double envelope of the reactor vessel. Further, flooding tanks full of sodium have been provided to dump sodium into the reactor vessel and maintain the sodium level in case of a primary circuit leak. Finally, the biological shield cooling system can also remove decay heat. These provisions are considered adequate to provide decay heat cooling for any upset conditions.

In the case of PFBR, there are to be four independent loops and normal decay heat removal is ensured when any one of the loops is operational. Physical separation of the loops will be provided to avoid common failures. To provide forced cooling following shutdown, design provisions are made: flywheel inertia for appropriate pumps coastdown, and pony motors backed by alternate power supplies/diesel generators/batteries.

Setting-up of natural convection for heat removal would be ensured by suitable elevations of core, IHX and terminal sodium/air heat exchangers. It is estimated that two loops in natural convection could provide the required decay heat removal capacity. To provide cooling in case of common mode failure resulting in non-availability of all normal loops, an independent back-up decay heat removal system would be provided.

5. Principles of containment

5.1 *Containment barriers*

An operating commercial-sized fast breeder reactor would have an inventory of a few thousand kilograms of plutonium and a few hundred kilograms of fission products. The total radioactivity is in millions of curies. As far as environmental effects are concerned, the main nuclides of importance are the plutonium isotopes, Am^{241}, Cm^{244}, tritium, Kr^{85}, Sr^{90}, I^{129}, I^{131}, I^{133}, Xe^{133}, Cs^{134}, Cs^{137} and the activation product C^{14}. The potential for escape varies according to whether the radioactive material is gaseous (like Xe or Kr), volatile (like I, Br, Cs, Rb) or non-volatile. The release of radioactive material to the environment during normal operation and under accident conditions is prevented by three physical barriers.

The first barrier is the cladded fuel element. Non-volatile fission products account for some 98 % of the generated radioactivity and are retained in the solid fuel material. The fuel material itself, as well as the volatile and gaseous fission products are contained by a gas-tight metallic (steel) clad.

The second barrier is the gas-tight primary coolant boundary, which also retains, along with the radio-active primary sodium and cover gas, the radioactive material that may escape from failed fuel elements. This barrier is constituted by a double walled reactor vessel, bolted down top plugs, etc. Monitoring of coolant and cover gas activity enables shutdown of the plant and clean-up action to be taken before the radioactivity in the primary circuit reaches unacceptable levels.

The reactor containment building, usually of reinforced concrete and steel, provides the third barrier to the escape of radioactivity. Different types of containment buildings have been considered for fast reactors (Graham 1971; Seeman and Armstrong 1978). A single containment building with a low leak rate to the outside has been used in small fast test reactors like EBR-II, RAPSODIE, FFTF, JOYO, FBTR, etc. Double containment has been used for the older reactors EFFBR (60 MWe), SEFOR as well as for recent large reactors like SUPERPHENIX-1 (1200 MWe) and SNR-300. These have within the containment building, a steel containment dome with inert gas around the reactor. The 250 MWe prototype fast reactors PFR and PHENIX had been provided with leak resistant reactor buildings without regular containment building features.

The design of these successive radioactive barriers is such that the radioactive release under normal operating conditions is acceptably low as per the radiological guidelines for the chosen site and the radioactivity is adequately contained under accident conditions.

In the case of FBTR, the rotating plugs at the top of the reactor vessel are fitted with a plug restraint mechanism with hold-down bolts. These have been designed to withstand and limit the radioactivity release into the reactor containment building under certain design basis accident scenarios. The strength of the containment building and its leakage rate are designed such that this radioactivity release accompanied by sodium fires is easily contained with no hazard to the public. Similarly, in PFBR, the double-walled reactor vessels enclosed in concrete vaults with top closures are to be designed to withstand design basis accidents without release of radioactivity.

5.2 *Energy release calculations*

In order to ensure the adequacy of the engineered safety features and the containment of radioactivity under all possible circumstances by the reactor vessel and containment building, it is usual to study extremely unlikely but assumed credible accident sequences and then estimate the energy release and other consequences. These hypothetical accidents assume complete failure of the plant protection system and are called design basis accidents. Their precise features vary from reactor type to reactor type. In fast reactors, the design basis accident is taken as fuel compaction resulting from meltdown of fuel leading to what is called a Hypothetical Core Disassembly Accident (HCDA) whose mechanical work has to be assessed to adequately design the engineered safeguards.

The classical model for the energy release in an HCDA is that due to Bethe and Tait (1956) which has been improved upon by later workers (*e.g.* Nicholson 1964). The model assumes an initial large reactivity insertion rate from fuel slumping which then leads to a power excursion and energy build-up in a molten core which is limited but not terminated by the Doppler coefficient. Pressures then build up in the fluid core which expands and terminates the excursion.

Important information required is the equation of state of core materials upto high temperatures ($\approx 7000°C$) which are not always accurately available. The rate of reactivity addition is a second important parameter on which very much depends the value of the final energy release. This rate depends on the kind of accident initiation. The Doppler coefficient is a third important parameter governing the energy release. Calculations indicate (Wilson 1977) that a Doppler coefficient with $-T\mathrm{d}k/\mathrm{d}T$ greater than 0·002 is sufficient to effectively reduce the energy build-up by a factor 100. Lastly, the initial power and temperature as well as the amount of void space that the fuel vapour has to fill before the disassembly can start, are important initial conditions governing the final energy release.

Computer codes like VENUS-II (Jackson and Nicholson 1972) calculate the heat energy developed accurately, subject to the uncertainties in the initial conditions. However, calculation of the conversion of the heat energy to mechanical energy or more precisely the forces and pressures which can rupture the containment barriers is more problematic. Research on fuel coolant interaction (FCI) and shock structure interactions are of relevance here. Generally, isentropic expansion of the vapour of all core materials is taken as the mechanism of conversion to mechanical work, and recent calculations with the comprehensive computer code SIMMER-II (Smith 1978) show that this approach is safe and conservative.

As noted earlier, the energy release in an HCDA is dependent on the initial conditions which in turn depend on the details of the accident initiation and the accident sequence. Hence, a detailed mechanistic modelling of the accident sequence is necessary for realistic estimates of the energy release. This has given rise to a vast amount of research in this field.

For a mechanistic analysis of design basis accidents, various possible accident initiators have been considered (Graham 1971) and the accident sequences usually studied are the unprotected Loss of Flow Accident (LOFA) and the unprotected Transient Over Power Accident (TOPA) resulting from a postulated failure of the PPS alongwith either sodium pumps failure or accidental control rod withdrawal. Large, sophisticated computer codes like SAS4A (Jackson and Stevenson 1981) have been created to trace these accident sequences. The in-pile safety related experimentation in the TREAT and the CABRI facilities have provided very important data for the successful modelling of the design basis accidents (see for example, Spencer *et al* 1976; Dadillon *et al* 1982).

It is important to note here that in the last few years, there is a re-evaluation of the role of HCDA from the probabilistic viewpoint. The HCDA is found to have no initiator in a modern LMFBR (Fauske 1981) and should not influence the plant design. Thus, detailed mechanistic considerations beyond early termination is to be considered redundant.

In the case of FBTR as well as PFBR, design basis accidents have been studied in detail for various fuel types and core configurations using computer codes PREDIS (Singh *et al* 1981) and VENUS-II. The results of these analyses have provided the basis for the containment design.

6. Conclusion

A review has been made of the principles of fast reactor safety and the different factors contributing to the overall safety of a modern LMFBR have been highlighted. The important conclusion is that the design, construction and operation of an inherently safe, well protected LMFBR pose no insurmountable problems, and much of the uncertainties surrounding fast reactor safety issues have been resolved by contemporary research.

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