Fast breeder reactor safety—a perspective

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Safety issues—both nuclear and non-nuclear—have been fairly well-understood and are manageable by current design, construction and operational practices, but efforts to improve reactor safety should continue. This article discusses the main safety issues and safety features of fast breeder reactors.

The demand for electrical energy the world over and, in particular, in India has been increasing steadily over the last few decades to provide a better standard of living in general and to meet minimum needs of population at large in the Indian context. In India, the present installed electrical capacity of 60,000 MW is expected to increase to 110,000 MW by the end of this century, i.e. at about 5% per annum and can be expected to grow at the same rate (if not more) for sometime thereafter. While most of this demand for energy will be met by mining coal in increasing quantities, this will eventually pose a serious drain on the country's modest coal reserves and the proven reserves of coal will nearly be exhausted by the year 2045 or so[1]. Nuclear fission is therefore required to play a major role as a source of energy in the 21st century India and several other countries. Fission as an efficient source of energy has been exploited the world over by constructing nuclear power reactors which can be classified into two main classes, viz. (i) thermal reactors and (ii) fast reactors. The terms 'thermal' and 'fast' refer respectively to the very low energy (0.025 eV, i.e. in thermal equilibrium with the medium) and high energy (usually > 10 keV) neutrons causing fission of uranium^{235} or plutonium^{239} atoms.

Most of the fission (nuclear) energy today is produced in the thermal reactors whether they be light water cooled reactors (LWRs), the pressurized heavy water cooled/moderated reactors (PHWRs) or the gas cooled reactors. This is primarily because of the early development work on the above reactor concepts, their simpler heat transport systems employing water (or heavy water) as coolant and availability of uranium in abundant quantities. As regards the last but important point of uranium availability, most of the leading economies of the world, the US, Germany, UK and even the USSR have access to large quantities of uranium, and therefore the thermal power reactors have dominated the nuclear energy scene. However, some advanced countries such as France, Japan and even the USSR have been seriously considering fast reactors as a future alternative in order to (i) minimize dependance on foreign reserves of natural uranium/fossil fuels and (ii) utilize large amounts of depleted (impoverished) uranium and plutonium available from thermal reactors. Furthermore, China with its limited uranium reserves is embarking on an ambitious FBR programme to attain a 20% nuclear share in total electricity generation by 2050 (ref. 2). The Indian situation is somewhat similar to the above countries in that our natural uranium reserves (0.7% U^{235} fissionable, balance 99.3% U^{238}) are small, and the fossil fuel position is not so sound. On the other hand, there are large reserves of thorium^{232}, a fertile material that can be converted into uranium^{233} for power production on a long-term basis.

Thus in the Indian context the introduction of fast reactors which convert unused U^{238} or Th^{232} into useful Pu^{239} or U^{233} respectively, holds great promise to long-term security of energy supplies spreading over the next two or more centuries. The existing resources of natural uranium which can ensure only about 10,000 MWe of installed capacity can be 'extended', through fast reactors, to about 350,000 MWe of installed capacity using U/Pu fuels[3], and further virtually inexhaustible energy supply can be assured using large thorium reserves available.

The fast breeder reactor

A fast breeder reactor is one that produces (breeds) more fissionable (fuel) material than is consumed while producing power from the reactor. A plutonium-depleted uranium-fuelled fast reactor, for example, will produce more plutonium, say 1.1 kg over a certain period when consuming 1 kg plutonium, to generate power. This breeding comes about by an interesting process as described below. During fission of Pu^{239}, a certain number of neutrons (well above 2) are always emitted which not only ensure the fission-chain reaction but leave one neutron available to convert the nonfissile U^{238} into Pu^{239} (see reaction) after accounting for small loss of neutrons in parasitic absorption and leakage. Thus a breeding ratio (BR) of the system can be defined as: $BR = \eta - 1 - L$,

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where η is the effective number of neutrons produced per fission, and L the small fraction of neutrons lost in absorption and leakage.

A typical breeding reaction is as follows:

$$
{}_{92}U^{238} + \eta \rightarrow {}_{92}U^{239} + \gamma \text{ emission}
$$
$$
\quad \hookrightarrow {}_{93}Np^{239} + \beta \text{ decay}
$$
$$
\qquad \hookrightarrow {}_{94}Pu^{239} + \beta \text{ decay}
$$

As η is about 2.5 (ref. 4) in a fast neutron-induced fission of plutonium, only a fast reactor with Pu^{239}/U^{238} can breed more plutonium from U^{238}* (not fissionable) and the energy potential of natural uranium is thus increased by about 60 times.

The core or central region of a typical fast reactor (Figure 1) consists of tightly packed hexagonal-shaped fuel assemblies. Each fuel assembly, in turn, is composed of fuel elements/pins containing typically uranium-plutonium dioxide pellets (with a plutonium content of 15/20% or even more) that are encased in high strength austenitic stainless steel cladding of about 6 mm in diameter. The fuel core is surrounded by a blanket of depleted uranium oxide assemblies to serve as the breeder and there is no moderator unlike in a thermal reactor (substance that slows down the neutrons, e.g. heavy water). Because of absence of moderator the reactor core is compact resulting in high thermal power density, approx. 500 kW l^{-1} or more[5], requiring efficient cooling medium. Liquid metal sodium by virtue of its very high thermal conductivity (about 100 times higher than that of water), reasonable heat capacity, high boiling point non moderation and low neutron-absorbing property is easily the best universally acceptable coolant found.

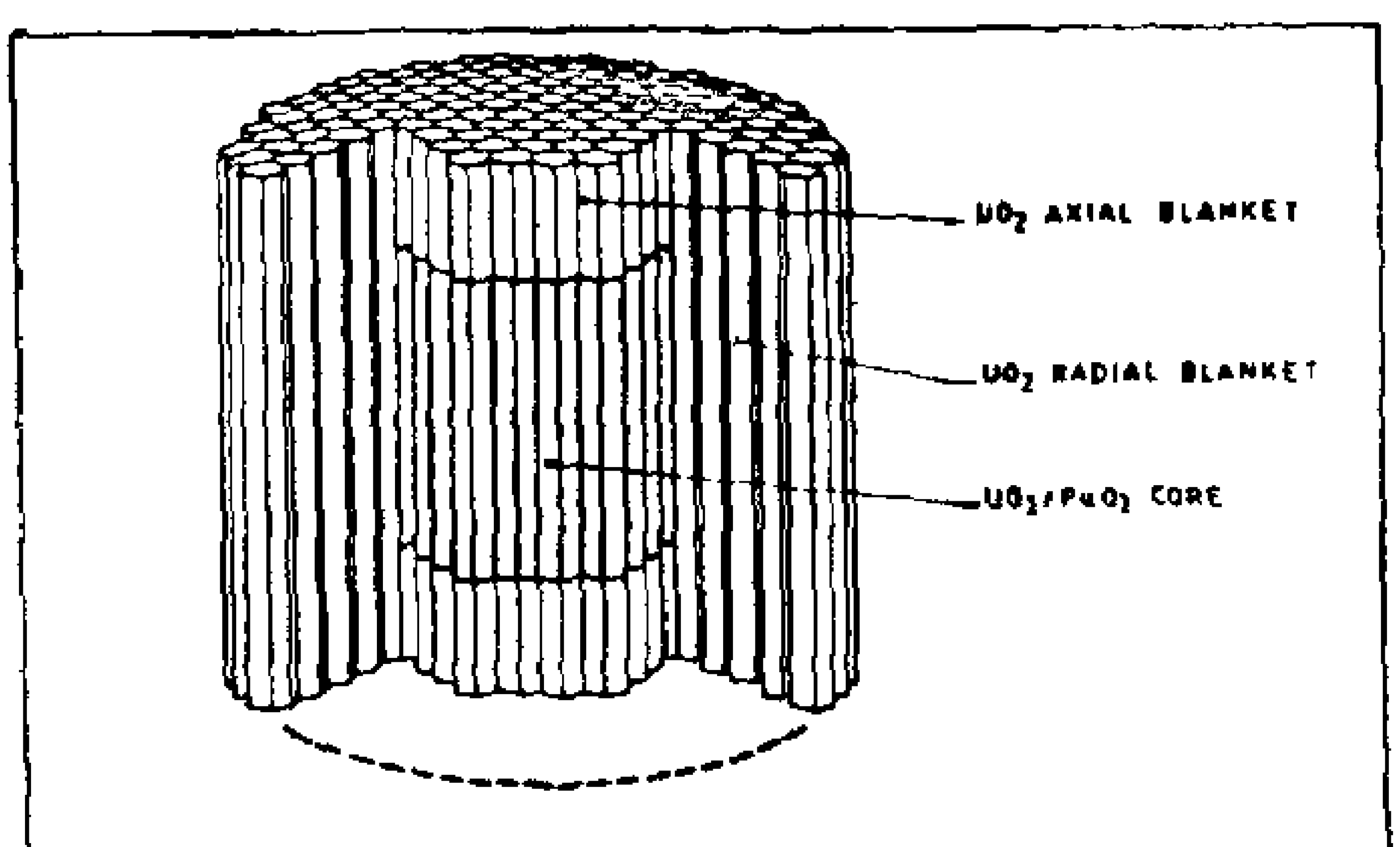

Figure 1. Core of a typical fast breeder reactor.

* Breeding in U^{235}/Pu^{239} fuelled system in a thermal neutron reactor is not possible due to low values of η.
The only possibility of breeding in thermal reactor with U^{233} (high η) needs a very complex concept of molten salt breeder which has been abandoned.

Figure 2 shows the heat transport circuit of a typical *liquid metal cooled fast breeder reactor (LMFBR)* plant. Liquid sodium enters the reactor core at a temperature of nearly 380° C, leaving it at 530° C and transferring nuclear heat to an intermediate sodium circuit through a sodium to sodium heat exchanger (IHX) from where the heat is further transferred to water in the steam generator (SG) producing high quality steam for the turbogenerator. A high sodium outlet temperature possible in a fast reactor results in high overall efficiency above 40% comparable to modern coal-fired power stations and well above those encountered in the thermal nuclear power plants.

Like its predecessor, viz. the thermal reactor, the fast reactor is also amenable to easy control to maintain its power. Basically it is the fission chain reaction that is controlled by holding the number of neutrons in the reactor nearly constant. While it is known that the fast neutron life/generation time is extremely small, about 10^{-7} s, it is the delayed neutrons (a small fraction of fission neutrons that is released with a definite delay up to tens of second after release of 'prompt' neutrons) with the help of which it is practically possible to control neutron population at any given instant. This, in turn, is achieved by controlling the position of neutron-absorber rods (control rods) in the cores to hold neutron multiplication factor exactly equal to unity with the contribution of delayed neutron fraction (0.004 or 4% for Pu^{239} fuel). The reactor can be safely shut down by inserting the control rods fully inside the core.

Fast reactors, world-wide operating experience

The world's first nuclear power plant to produce electricity happened to be a fast reactor called Experimental Breeder Reactor I (EBR-1) with an electrical output of 0.2 MWe commissioned in 1951 in the US. Though the plant was shut down after about 12 years of operation, its successor the EBR II plant with a power level of 20 MWe has been operating for the last nearly 28 years, and has contributed a great deal in understanding the fast reactor technology including its safety characteristics. Table 1 gives a summary of fast reactor plants both experimental and prototype which have been operating in different parts of the world, the Soviet Union having the largest number in operation. The largest fast reactor commissioned in 1987 is the French Super Phenix-1 (SPX-1) having a power level of 1,200 MWe whereas the next biggest plant is the Soviet BN-600 operating successfully for the past eleven years. The observers at the International Topical Meeting on Fast Reactor Safety, Knoxville, USA, in 1985 (ref. 6) made

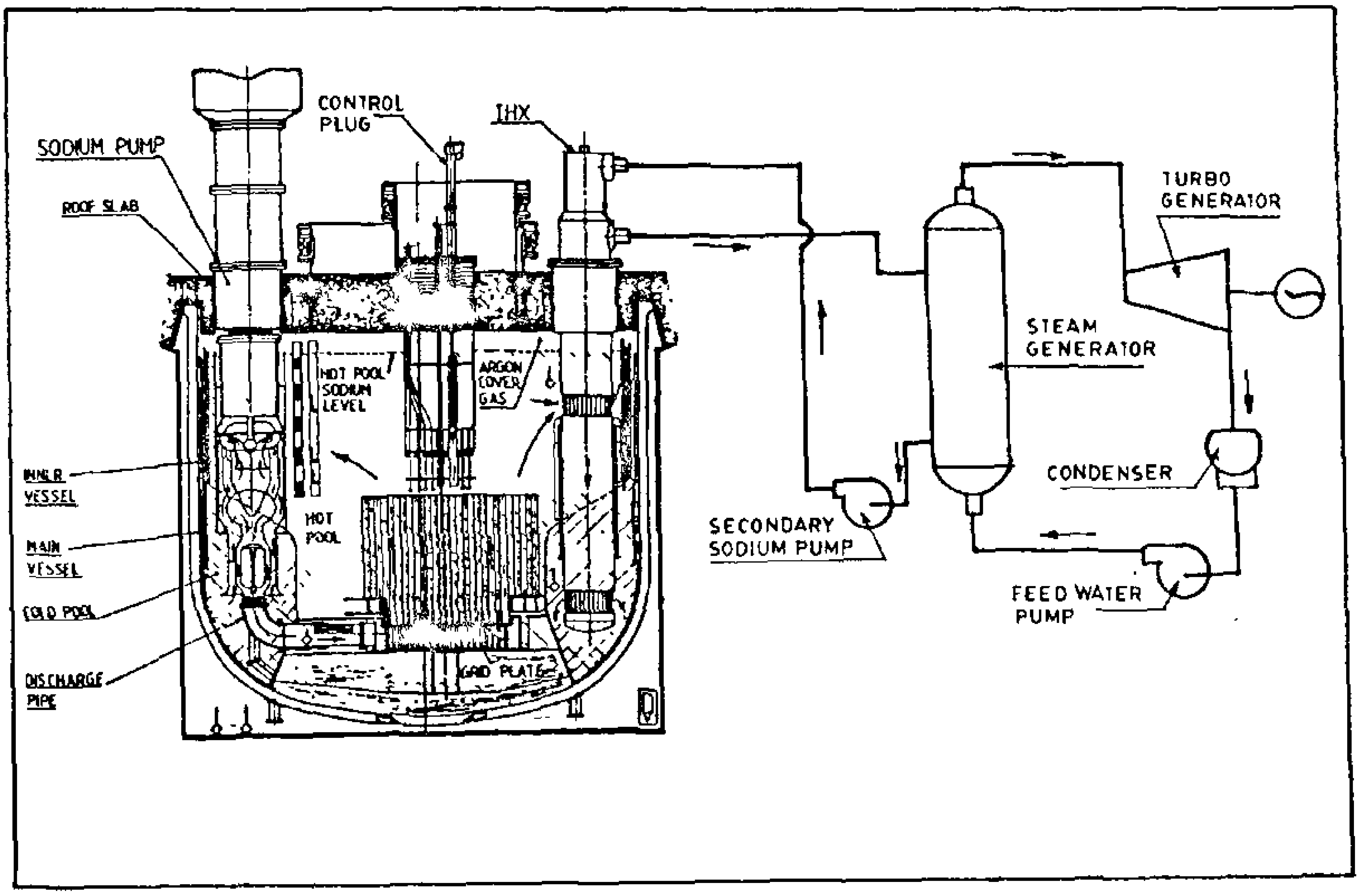

Figure 2. Schematic of pool-type fast reactor.

the following observations from the operating experience of six differently designed fast reactor power plants which among them had operated for a total of more than 60 years then.

* Good fuel performance with few failures and little release of fuel or fission products.
* Good experience with natural circulation cooling in case of loss of power to pumps:

  The PFR experiments conducted after initial operation of the reactor[7] show that in case of complete loss of power to pumps, the sodium temperatures from reactor core stabilize at a safe level in about 350 s (after initial loss of pumping and reactor trip) indicating fully developed natural circulation cooling.

* Good reactor stability and reliable shut down:

  Two independent shut down systems are provided and these diversely acting systems have an extremely high reliability [(equivalent to unavailability of less than one per ten million demands (10^{-7})] (ref. 8).

* Low operator (radiation) doses from both normal operation and repairs.

Operation experience of various nuclear power plants in the USSR has shown that the use of FBRs ensures appreciably higher ecological purity. For example, for the BN-600 FBR there are practically no releases of radioactive iodine-131, the long-lived and short-lived radionuclides at a level of background values and the release rate of radioactive inert gases is reduced by a factor 100 compared to that from PWR and RBMK thermal reactors[9]. Similarly the radiation doses to operators in European fast reactors (UK, France and Germany) have been lower by factors of 17 to 33 compared to those in PWRs[10].

Safety aspects

It is a well-recognized fact that the potential for accidental release of radioactive substances is present in any nuclear power reactor and an LMFBR is no exception. The thermal reactors have more or less been accepted as a safe source for electricity generation the world over and the LMFBRs which arrived later on the scene (as large power reactors) have had to go through difficult times not so much because of safety considerations but more due to economic penalty with present low uranium prices where uranium is available in abundance. It is the purpose of the following paragraphs to demonstrate that fast reactors are equally safe or a shade safer than present-day thermal reactors such as the LWRs or PHWRs.

Main safety issues

The main safety issues would become clear from

Table 1. World fast reactors

Reactor	Country	Power MWe	Purpose	Primary system	Fuel	Blanket	Secondary heat transfer loops	Steam conditions MPa/deg°C	Year of commissioning	Remarks
Phenix	France	250	Prototype	Pool	Mixed oxide*	UO_2	3	16.3/512	1974	Operated successfully till 1990. Awaiting new clearance
Super Phenix	France	1200	Demonstration	Pool	Mixed oxide		4	17.7/487	1987	Presently shut down for major repair
P.F.R.	UK	270	Prototype	Pool	Mixed oxide					
	16.2/516	1974	In operation							
BR 10	USSR	5/10MWth	Experimental	Loop	PUO_2/ Mixed oxide		2	-	1959	Air heat sink
BR 60	USSR	12	Experimental	Loop	Mixed oxide	UO_2	2	4/385	1970	Air heat sink also available
BN 350	USSR	130	Prototype	Loop	UO_2 (enriched) + FXP mixed oxide	-	6	4/385	1973	80000 CuM/d desalinated water
BN 600	USSR	600	Demonstration	Pool	UO_2 (enriched) later mixed oxide	UO_2	3	14/505	1980	All reactors in USSR in operation
EBR II	USA	20	Experimental	Pool	U alloy	depleted U	1	-	1963	In operation
FFTF	USA	400MWth	Experimental	Loop	Mixed oxide	-	3	-	1980	S. S. reflector Air heat sink in operation
JOYO	Japan	100MWth	Experimental	Loop	Mixed oxide	depleted U	2	-	1977	Air heat sink in operation
MONJU	Japan	280	Prototype	Loop	Mixed oxide	-	3	-	end 1992	Construction completed
KNK II	Germany	21.4	Experimental	Loop	Mixed oxide and UO_2	-	2	-	1977	
FBTR	India	12.5/15	Experimental	Loop	Mixed carbide	ThO_2	2	12.5/480	1985	First criticality in operation

*Mixed oxides of uranium and plutonium.

Note: All reactors are cooled by liquid sodium.
Data are largely derived from ref. 8.

comparative characteristics of LMFBRs *vis-à-vis* LWRs as shown in Table 2. The main concern of the safety analysts is focused on the core assembly/configuration which differs significantly in a fast reactor. In a fast reactor (no moderator) with an already compact core, a further core compaction could take place if the fuel were to somehow melt, although this is extremely unlikely, resulting in attaining a more 'reactive' configuration. The term reactive or more precisely 'reactivity' denotes change in neutron population over the prevailing neutron population. Thus more 'reactive' means addition of positive reactivity indicating increased number of neutrons and hence increased fission rate. This possibility provided the underlying concern for the majority of large accident safety studies done for early fast reactors and even though the consequences of such core compaction have been much better understood, the concern is not totally extinct.

The second important concern arises from high chemical energy potential of sodium coolant (refer characteristic 3 in Table 2). The chemical reactions of concern in an LMFBR are primarily sodium–air and sodium–water reactions. A barrier to such reactions for normal plant operation is provided by covering all sodium systems with an inert gas such as argon. The above concern can be classified under non-nuclear safety issue as it does not involve the primary/radioactive system.

Before discussing further the above two safety issues (nuclear and non-nuclear) it would be interesting to look at some positive safety features of fast reactors.

Safety features of FBRs

One of the most important advantages of a sodium-cooled fast reactor is the ability to operate its primary coolant system under low/near atmospheric pressure (unlike LWRs/PHWRs which need pressures of the order of 7.5 MPa or 75 kg sq cm^{-1} to maintain coolant water temperatures at 260°/270° C). The liquid sodium is highly subcooled even at a reactor temperature of 530° C due to its high boiling point of 882° C. A breach in the primary system does not, therefore, lead to coolant boiling/flashing.

The above advantage (i.e. low pressure) together with the primary system design feature, viz. pool concept with no pipe penetration below sodium level, precludes a leak from the main reactor vessel. With a double-walled construction of main reactor vessel the loss of coolant accident leading to melt down is precluded unlike in a thermal water reactor.

The inherent emergency heat removal capability of a fast reactor is also very high due to large sodium thermal inertia in the reactor primary tank and the ease of setting up natural convection within the pool in case of power failure to circulating pumps. A decay heat removal capability based mainly on passive cooling is thus possible and the 'station black-out' incident where there is total loss of electric power is manageable without much difficulty unlike in a thermal reactor.

Corrosion: Liquid sodium is far less corrosive than water or other aqueous media, the corrosion of the container material, usually austenitic stainless steel, being controlled by oxygen impurity in sodium. This latter can be held to a level of a few ppm easily by fitting a cold trap to the coolant circuit.

We shall now proceed to analyse the two main safety issues concerning (i) core disruptive accident and (ii) effects of sodium chemical energy potential.

Hypothetical core disruptive accident (beyond design basis accident). It was mentioned earlier that the possibility of forming a more reactive core following a large scale fuel melt-down (cause unexplained and

Table 2. Comparative safety characteristics of thermal and fast reactors

Characteristics	LWR	LMFBR
Core assembly	Optimum geometry w r t reactivity	Not arranged in most reactive configuration
Stored energy in coolant	High (pressurized system)	None (subcooled) at 1 atm.
*Chemical energy potential	None	High = Na-air reaction Na-H_2O reaction
*Loss of coolant	Reactivity loss	Reactivity gain (in large cores) normally precluded by double containment
Inherent emergency heat removal capability	Low	High
Radiological inventories		
*Fission products	Comparable	
*Plutonium	Medium	High

highly unlikely) resulting in large accidental energy release was a major concern in early years and this continues, though to a much less extent, owing to better understanding of the events scenario. Based on this concern the bounding accidents in an LMFBR are associated with postulated or hypothetical power to flow imbalances coupled with failure of the plant protection system (PPS), i.e. control rods fail to drop. We shall examine one of the worst scenarios in which the reactor power remains high (failure of PPS) when power supply to all coolant pumps fails.

Because of loss of forced sodium circulation, the coolant temperature starts rising. But the rising coolant temperatures introduce negative reactivity coming from expansion of fuel assembly, control rod guide expansion and later grid plate (supporting the core) radial expansion. The negative reactivity effect counteracts the effects of positive reactivity due to sodium density decrease[8]. Thus the resulting negative reactivity effect reduces the reactor power and brings it to very low level as in case of a reactor shut-down (decay power). Figure 3 shows variation of reactor power, and sodium temperature for the French 'Rapsodie' fast reactor during a loss of flow test without scram (without action of PPS) performed before final close down of the plant. The sodium temperature first increases and then starts reducing as the reactor power continually falls because of negative reactivity feedback as explained above while a small sodium flow (below 10%) is sustained. Similar results have been predicted in the case of FBTR plant[11] at Kalpakkam (Figure 4). Thus a core-disruptive accident due to loss of flow can be ruled out in a small fast reactor. However, the situation is somewhat different in the case of large fast reactors where sodium void coefficient of reactivity is positive. The reactivity coefficient due to sodium voiding becomes positive in a large fast reactor as a result of following two effects: Firstly a decrease in sodium density or its vaporization with increased temperature increases neutron leakage—always a negative effect. Secondly, the loss of moderating action due to sodium density decrease or vaporization produces higher energy neutrons (spectral hardening) resulting in more fissions and hence positive reactivity. As the negative effect due to leakage is much less in large cores, the nett reactivity coefficient is positive.

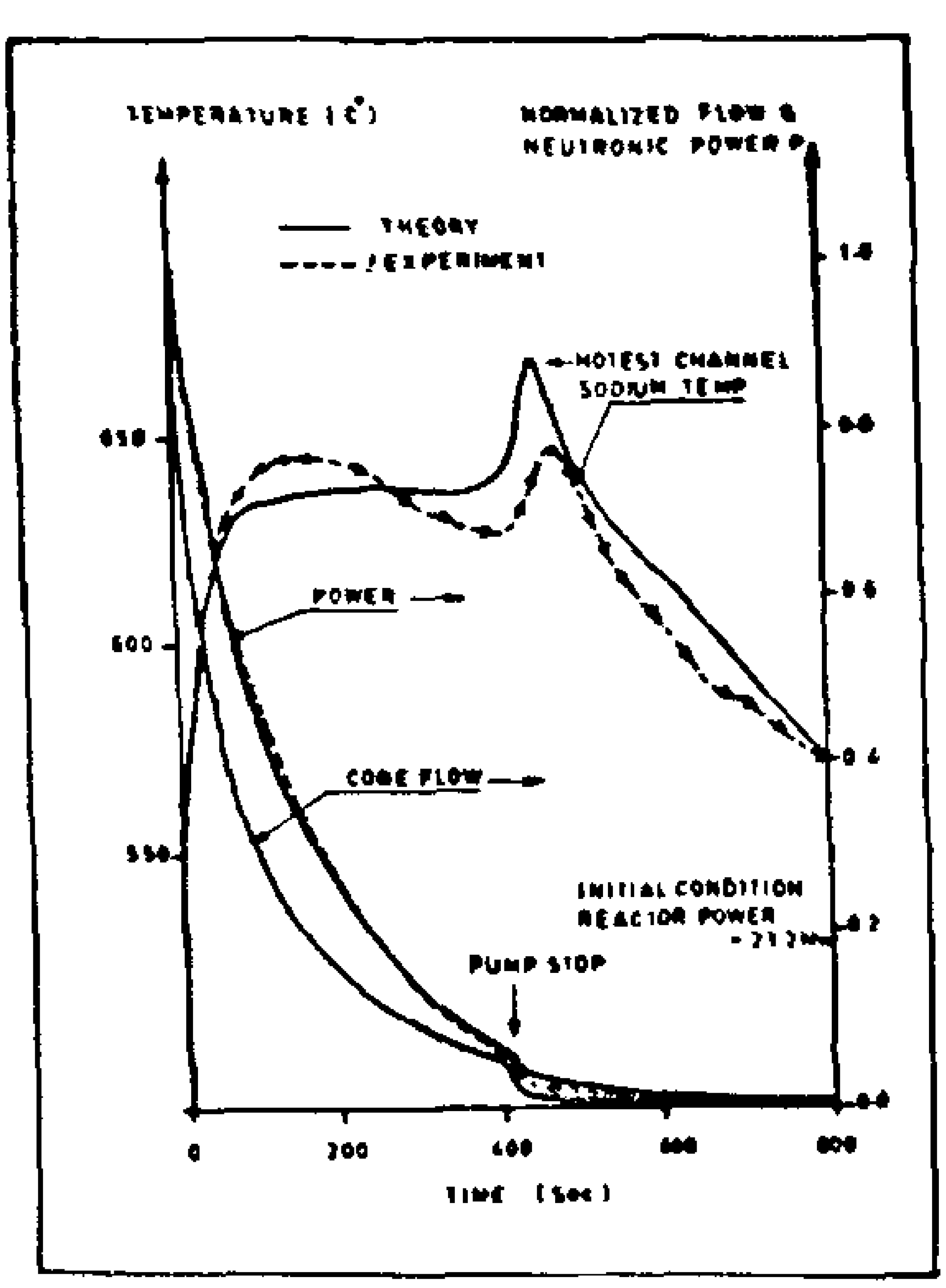

Figure 3. Loss of flow test without reactor shutdown.

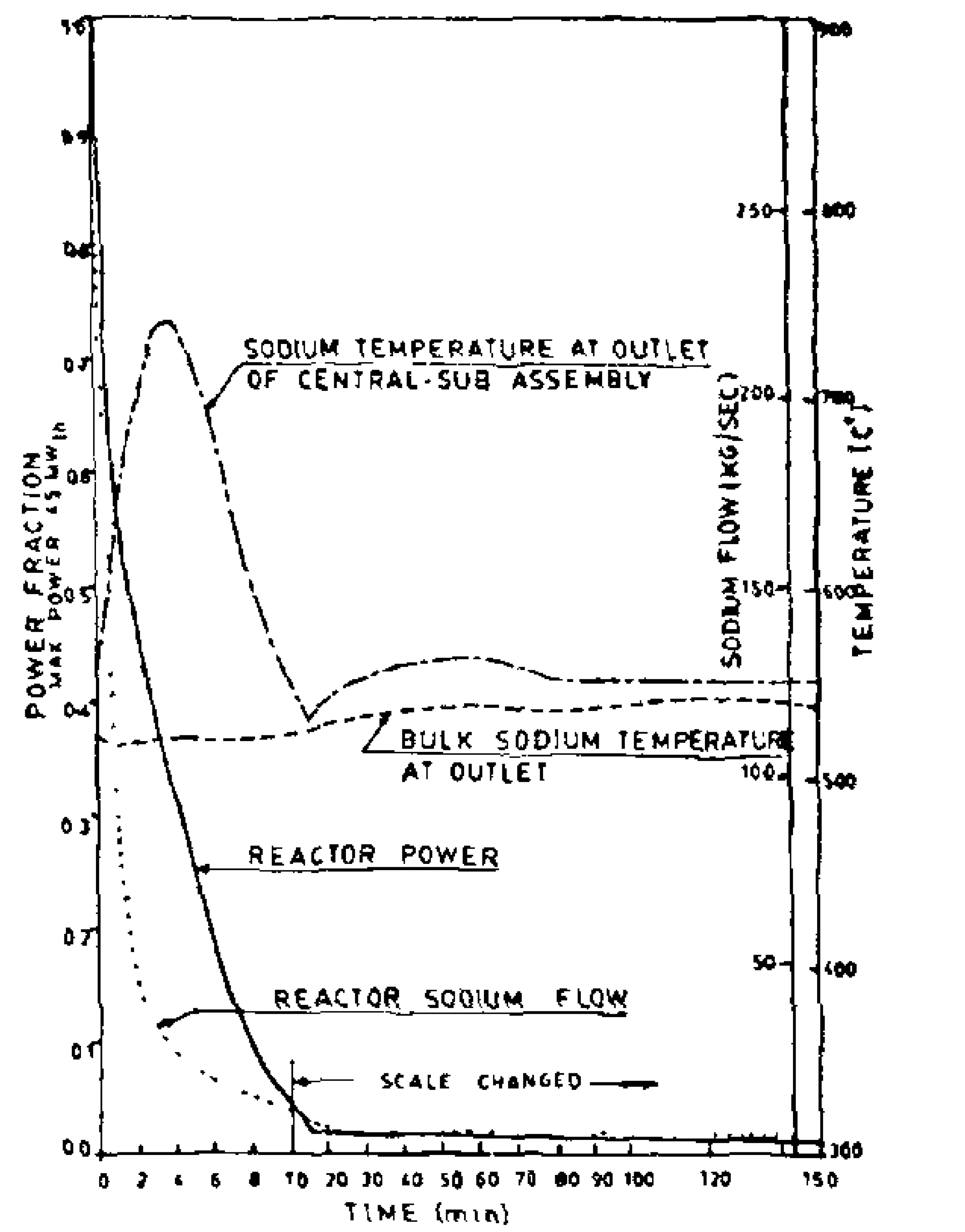

Figure 4. Sodium temperatures during unprotected loss of flow for FBTR.

In view of considerable work carried out on sodium boiling and its propagation[12] and encouraging results of studies of coolant blockages in fuel subassemblies, it is evident that major boiling (bulk boiling) in the core of a large reactor would follow spreading effects of local coolant blockage only if cooling and all scram (fast shut down) systems broke down. And only in that extremely improbable situation would the positive void coefficient take an importance. This would lead to an increased reactivity resulting in core overheating and possibly melting. Estimates have indicated that energy released in a hypothetical whole core accident could be contained in the primary system of 1,500 MWe plant (energy release less than 200 MJ or 56 kWh approx.)[13]. Chellapandi *et al.*[14] have also analysed and shown that an energy release of 200 MJ for the 500 MWe Indian prototype fast breeder can be safely contained. The energy released is absorbed by a combined mechanism of sodium vaporization and plastic deformation of the austenitic stainless steel main reactor vessel under high internal pressure generated during the event.

We shall now review the other safety issue that relates to the chemical energy potential of sodium.

Sodium spills. As high temperature sodium above 200° C readily burns in air, the concern about large sodium spills is genuine. However, large sodium spills are considered unlikely in the primary system of an LMFBR because of low pressure in the coolant systems, unlikelihood of a major fracture due to use of highly ductile austenitic stainless steel pipe material and finally by use of a 'guard' vessel with inerted atmosphere between it and the main sodium vessel or by providing additional jacket on the primary piping of a loop-type reactor. Cells containing primary piping in loop-type reactors are lined with steel liner and maintained under oxygen-free atmosphere. So also the primary vault in the case of a pool reactor. The contact of primary sodium with concrete is thus avoided.

While there is a possibility of some sodium spill from the secondary non-radioactive system, here the system piping is traced by catch pans below the pipe runs, thus collecting any leaking sodium and avoiding spreading of leak/fire.

Sodium fire is unlike hydrocarbon/organic fire in that the heat of combustion is rather low, approximately 20% of that evolved for petroleum fires and the combustion proceeds without any flame (low ratio of heat of combustion/heat of vaporization). It is therefore not difficult to extinguish the sodium fire by application of dry chemical powders. Special collecting catch pans (covered trays) have also been designed which admit leaking sodium through small-sized holes and smother the fire by oxygen starvation[15]. Thus sodium fires can be easily dealt with in a properly designed reactor system.

Sodium water reactions. While sodium spills are not a major concern, sodium water reactions are of major concern in the operation of sodium heated steam generators. Even a small leak in the high pressure water tube of the steam generator (SG) can lead to a vigorous sodium–water reaction causing damage to adjacent tubes, if not detected early. This has necessitated (i) great care in the design and construction of steam generators to avoid even a small defect and (ii) development of quick/sensitive methods for leak detection. Design improvements include elimination of weld joints with crevices and tube/tube welds inside sodium. On the leak detection side, the currently developed hydrogen in sodium detectors (hydrogen is formed in sodium–water reactions) are capable of detecting leaks of less than 0.1 $g s^{-1}$ of water/steam to take corrective action. Furthermore, rupture discs are provided on the steam generator connections to quickly depressurize the circuit following any large-scale sodium–water reaction, if it should occur. While considerable further work is continuing especially on instantaneous leak detection using acoustic noise measurments, the experience of SG operation so far in most LMFBR plants has been satisfactory.

Safety research and improvement

In connection with the hypothetical core accident an extensive programme of safety research over the last six to eight years concentrating on accident events such as transient over power or total loss of flow without shutting down of the reactor (i.e. full power) has been carried out in several West European countries and the US. In these research experiments actual fast reactor pins/assembly have been placed in test loops installed in some experimental reactors, e.g. CABRI and SCARABEE in France and TREAT in USA, and the accident conditions as above have been imposed. The fuel element behaviour is monitored using suitable instrumentation. While the fuel is seen to melt at significantly high over-power condition (TREAT results), the CABRI results have further shown that the fuel is greatly dispersed during the power excursions, largely as a result of fuel vapour and fission product gas pressure[13]. The SCARABEE tests have indicated that molten fuel (caused by sudden blockage of coolant flow which is very unlikely) from a subassembly tends to freeze before it can propagate to neighbouring assemblies. All these tests have led to a great improvment in understanding of the whole core hypothetical accident and the consequences of even such accidents are not severe.

However, reactor designers are considering seriously to counter the effects of the unlikely situation of core melt by suitably providing a core catcher. The main function of a core catcher is to stop or control the motion of molten core (fuel) mass should such an accident occur and to prevent damage to the primary boundary (main reactor vessel) thus containing the effects of the event totally. Indeed the FFTF and Super Phenix reactors have provided such core catchers in their main reactor vessel.

To improve natural circulation of coolant and also increase the pump coast down (slowing down) period by as much as 8 times to enhance safety during a station black out (total power failure at site), an advanced pool type reactor design has been recently advocated by Rineisky[16]. In this concept the hydraulic resistance of the fuel assemblies (the main resistance to coolant flow) has been reduced by nearly an order of magnitude by increased lattice spacing of fuel elements (pins). This would also reduce the pumping power considerably contributing to small improvement in overall plant efficiency and reduced sizes of the pumps. However, an increased lattice spacing will necessitate increased fissile material inventory in the core and its effects on fuel economics will have to be examined.

As regards void coefficient of reactivity attempts are currently being made by core designers/researchers to make the reactivity void coefficient nearly zero, if not negative by introducing a sodium layer between the core (top of fuel) and the axial blanket[17].

Indian scenario

Recognizing the great potential of fast breeder reactors to contribute to the nation's nuclear power programme, the DAE set up a modest R&D programme at the then Reactor Research Centre (now IGCAR) at Kalpakkam in 1971. The sodium technology, a key to success of fast reactor programme, has been mastered at IGCAR during the last two decades and this together with the reactor design know-how from CEA, France, has culminated in the commissioning of sodium cooled Fast Breeder Test Reactor (FBTR) at Kalpakkam.

FBTR is a loop-type reactor with two primary sodium heat transport loops, two secondary sodium and a steam water circuit for driving the turbogenerator (Figure 5). A 'defence in depth' approach has been used in the reactor plant design by incorporating engineered safety provisions in three stages referred to as Level-I, Level-II and Level-III. (ref. 8).

- Level-I is to execute and build a sound, conservative and inherently safe design.
- Level-II is to provide protection systems designed to assure that off-normal events will be prevented, arrested or accommodated safely.
- Level-III is to evaluate and provide features which add margin as additional assurance of public safety even for extremely unlikely and unforeseen circumstances.

Operational transients of higher frequency of occurrence such as thermal shocks are taken care of in Level-I. Infrequent incidents such as pump seizure are taken care of in Level-II whereas Level-III includes such events as are never expected to occur but are possible, nevertheless, mechanistically. The commonly labelled design basis accident (DBA) of the highest damage severity, e.g. failure of all pumps or station black-out, is included herein.

Beyond the DBA, there is still considered a domain of accident consequences of extremely low probability. Two or more low probability failures must take place in sequence to assess accident magnitude. These accidents are termed as hypothetical or beyond design basis accidents (BDBA). It has been shown earlier that a small fast reactor such as FBTR is safe under this condition. The important specific design safety features incorporated in FBTR are briefly discussed below.

Design safety features of FBTR[3]

- *Primary system containment.* The entire primary sodium system is provided with an S. S. jacket (double envelope) to contain primary sodium in the case of breach of primary boundary. This together with an additional safety vessel surrounding the reactor vessel region (Figure 6) eliminates a loss of coolant accident, usually a nightmare in the thermal

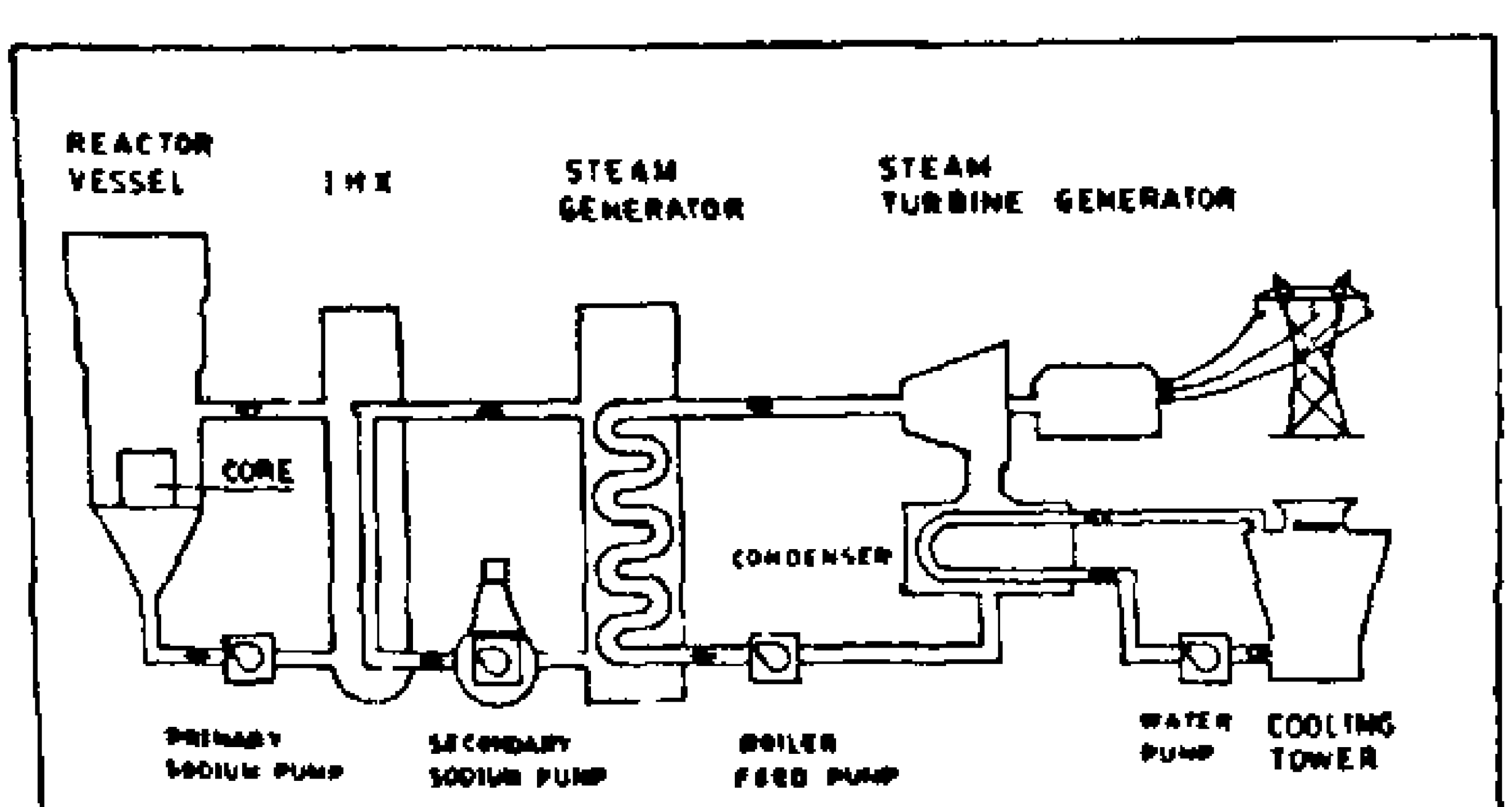

Figure 5. FBTR coolant circuits

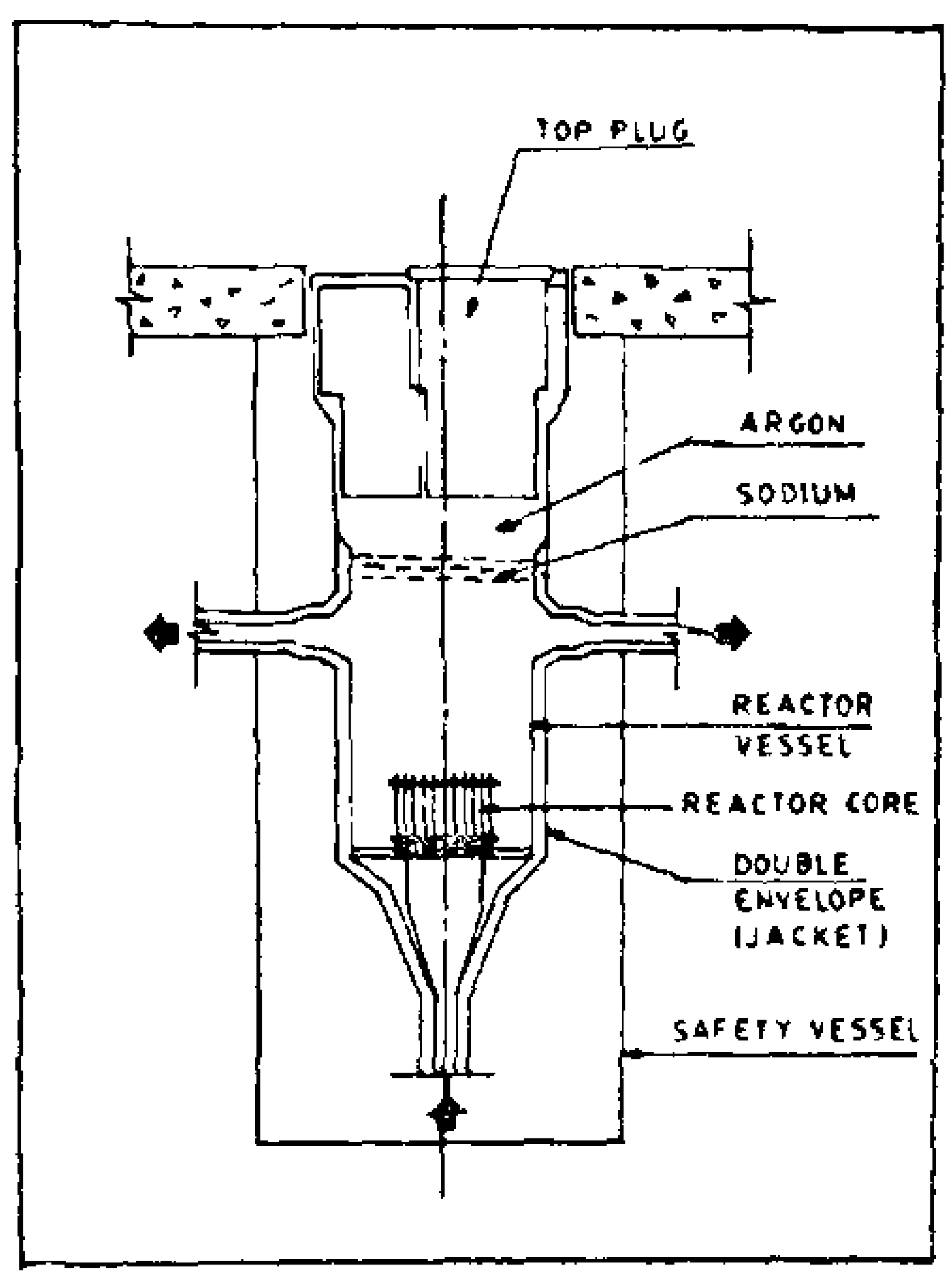

Figure 6. Primary system containment — FBTR

reactors. Furthermore, the double envelope contains nitrogen gas precluding any fire involving radioactive sodium in case of sodium leak from primary boundary.

- *Emergency core cooling*. In the case of nonavailability of primary circuit due to simultaneous double breach of piping (outside the safety vessel), the decay heat in the reactor core is removed by circulating nitrogen through the reactor jacket (double envelope) in an emergency cooling circuit having a capacity of 350 kW. Furthermore, in the case of a simultaneous double breach of reactor vessel and its jacket, there is provision to flood the reactor vessel with liquid sodium stored and maintained at 150° C in flooding tanks with a total capacity of 65 m^3.
- The sodium outlet temperature from all fuel assemblies is continuously scanned by a computer every second and plant protection system automatically initiated when set temperature thresholds are crossed.
- *Plant protection system (PPS)*. Six control rods made of boron carbide enriched in B^{10} isotope are provided in the plant protection system. These serve to act also as regulating rods to control reactor power while playing the important role of fast shutting down of the reactor by gravity insertion under abnormal situations. The individual control rod reactivity worths are so designed that any two control rods are sufficient to shut down the reactor safely from the maximum power level. A further safety provision in PPS allows only one control rod to be withdrawn at a time and inhibits automatically any continuous withdrawal of rod beyond three seconds (speed of control rod for withdrawal is 1 mm/s).
- *Inherent safety feature*. In FBTR, the temperature coefficient and the power coefficient of reactivity, i.e. change in reactivity for a unit change in fuel temperature or power, are both negative and any abnormal increase in temperature/power always leads to a reduction in reactivity and consequent reduction in power. With negative reactivity coefficients FBTR has a very stable operation where reactor power remains within a narrow range without external control action.
- *Natural convection cooling*. Finally the layout of primary sodium, the secondary sodium and tertiary steam/water circuit is such as to easily set up coolant circulation by natural convection and reactor decay heat can be removed reliably in the case of total loss of auxillary power, i.e. 'station black out' condition.

Safety analysis and research

Both theoretical analysis and limited experimental programme are being carried out at IGCAR to understand implications of the hypothetical accidents in fast reactors. The dependence of energy release in an HCDA on the Doppler coefficient* of reactivity feedback for a 500 MWe fast reactor has been studied and it has been found that energy release decreases with Doppler coefficient. So also a comparison of performance of three different fuels, viz. mixed oxide, mixed carbides and metal alloy, have been performed and a benign energy release found in case of oxides and carbide fuels for smaller and perhaps realistic reactivity additions during the so-called predisassembly phase[18].

The assessment of structural integrity of reactor assembly for the 500 MWe prototype fast breeder reactor has been carried out using two-dimensional finite element code FUSTIN developed inhouse. The code has been extensively validated using a variety of bench mark problems from open literature as also against a TNT explosion experiment in a water-filled tank[14]. The analysis of main reactor vessel integrity for 200 MJ energy release in an HCDA has been carried out and it shows that the main vessel containing reactor core does not give way though it is plastically deformed in local regions to a strain of 5.5%.

Experimental research in certain other areas of reactor safety such as detection of minute sodium leaks/fires as also detection of leaks in sodium-heated steam generator is continued. A thermal ionization-based sodium aerosol detector has been developed to detect sodium aerosols in concentrations of nanograms/ cm^3/s of carrier gas[19]. The hot Pt/Rh filament ionizes sodium vapour or its compounds in preference to other constituents of the carrier gas and the resulting ion current is a measure of sodium concentration. The important feature of the detector lies in its ability to work without need for a vacuum chamber unlike several common electronic detectors. A gas sampling system installed on FBTR primary vault used along with the above detection system ensures the integrity of the primary system against even minute sodium leak.

Very small water/steam leaks in sodium-heated generators must be detected early to prevent damage to the heat exchanger tubes as a result of severe corrosion/erosion due to caustic soda formation. A sensitive diffusion-type hydrogen meter based on mass spectrometric determination has been developed which is capable of measuring hydrogen concentration change of few tens of parts per billion (ppb) over a background level of approx. 100 ppb. An electrochemical hydrogen detector based on $CaCl_2$ CaH_2 electrolyte and Li-LiH

* Doppler coefficient refers to change in reactivity per unit change in fuel temperature and depends upon (Doppler) broadening of resonance of neutron-capture and fission cross sections. Doppler coefficient is negative in large fast reactors.

reference electrode has also been developed and is undergoing tests on the reactor circuits. Finally a fast leak-detection system based on acoustic noise produced during a leak is also under development. The main advantages of this system lie in its fast response (order of few seconds) and the possibility of locating the leak within the heat exchanger tubes by triangulation of different signals.

In the area of sodium fire protection, considerable experimental work has been carried out to understand phenomena of sodium ignition and aerosol generation as well as methods of fire extinction. As the heat of vaporization of liquid sodium is significant (approx. 1.5 times greater than water), the ratio of its heat of combustion to heat of vaporization is rather small, about 3.5 compared to nearly 1000 for hydrocarbons, and hence vapour phase combustion is insignificant. This results in very low flame height (like glowing embers), much less heat radiation and the fire is usually approachable and extinguished without much difficulty. The work on fire-extinguishing techniques has resulted in selecting suitable dry chemical powders from amongst various brands available indigenously[15]. Experimental work on models is also planned for studying pressure build-up in reactor containment building due to an accidental sodium spill.

Conclusion

This article has emphasized the need for exploiting the potential of nuclear fuel by choosing the fast reactor route which can supply virtually inexhaustible energy for a long time to come especially when fossil fuel resources will be on the decline. The significant advantages of FBR plants cooled by sodium and their satisfactory operating experiences were next reviewed followed by an extensive analysis of FBR safety issues. It is hoped that this article will serve to understand the safety issues involved in fast power reactors in their proper perspective and should alleviate major misconceptions, if any, about safety of these reactors.

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