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Printed in India.

# SYED HUSAIN ZAHEER MEDAL AWARD LECTURE 1986

## ON THE DEVELOPMENT OF FAST BREEDER REACTORS

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*(Delivered* 1 January 1987)*

Modern civilisation is based on substantial utilisation of energy. Rapid industrial development and improvement of living standards in India require energy planners to adequately forecast the energy demand and take appropriate measures in advance. However, the development and establishment of new technology is a slow process, sometimes extending over decades. Hence, energy options based on new technologies need to be planned for much in advance making allowance for uncertainties and delays. Fast Breeder Reactor (FBR) technology is an advanced energy option promising abundant and economic supply of power. Research and development work on FBRs has been conducted at the Indira Gandhi Centre for Atomic Research (IGC) since 1971. The international trends in FBR development are highlighted in this discussion and an overview of some of the research activities at IGC is presented.

Key Words : **Liquid Metal-cooled Fast Breeder Reactors; Energy Technology; Nuclear Electricity; Isotopes; Pressurized Heavy Water Reactors; Risk Analysis; Reactivity Gains; Fuel and Sodium Chemistry; Carbide Fuel Fabrication Process; Reprocessing of Fast Reactor Fuel**

I am deeply grateful to the Indian National Science Academy for the opportunity given to me to address this distinguished audience and to pay tribute to the memory of Dr Syed Husain Zaheer, formerly the Director-General of the Council of Scientific and Industrial Research (CSIR). As an eminent educationist and scientist Dr Zaheer was involved in the administration and execution of several projects of national importance. Under his stewardship the Regional Research Laboratory at Hyderabad emerged as one of the most important industrial research laboratories of the country. Besides pioneering research work on fine chemicals, oils and fats, as well as in coal chemistry and technology, Dr Zaheer was associated with the setting up of several new industries in this country both in the private and public sectors. Dr Zaheer was keenly interested in the planning for the primary energy supply required for industrial development in India. In the selection of the theme for my lecture I have been influenced by Dr Zaheer's larger concern in the field of energy planning.

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*at the Indian Institute of Science, Bangalore during the occasion of the Anniversary General Meeting.

## INTRODUCTION

An important consideration in national planning is the supply of adequate energy for the requirements of a modern industrialised society. On account of the limited availability of coal, oil, gas and hydro resources there has been worldwide development of alternate energy technologies. Foremost among these is nuclear power technology, which is today a safe, reliable and commercially proven means of power production. Already in some countries nuclear electricity constitutes the major fraction of the electricity supply, as for example in France (65 per cent). Table I presents a summary of the present status of nuclear power production in several countries.

TABLE I

*Nuclear power status in selected countries at end 1985*

| Country | In Operation | | Under Construction | |
|---|---|---|---|---|
| | No. of Units | Total MWe | No. of Units | Total MWe |
| U.S.A. | 93 | 77838 | 26 | 29258 |
| France | 43 | 37533 | 19 | 23567 |
| U.S.S.R. | 51 | 27756 | 34 | 31816 |
| Japan | 33 | 23665 | 11 | 9773 |
| W. Germany | 19 | 16413 | 6 | 6585 |
| U.K. | 38 | 10120 | 4 | 2530 |
| Canada | 16 | 9776 | 6 | 4789 |
| Sweden | 12 | 9455 | 2 | 2100 |
| 23 other Countries | 63 | 35958 | 44 | 29194 |
| India | 6 | 1240 | 4 | 880 |
| World Total | 374 | 249754 | 156 | 140492 |

The primary fissile material for fuelling nuclear reactors is the isotope U 235 whose abundance is only 0.71 per cent in natural uranium. On the other hand, the fertile isotopes U 238 and Th 232 are several hundred times more abundant than U 235. Consequently, plans for the long term utilisation of nuclear energy are based on the development of breeder reactors which can transmute the fertile isotopes into the man-made fissile isotopes Pu 239 and U 233 at a rate faster than the rate of consumption of fissile material for power production. Table II gives a comparison of the potential of energy resources in India for electric power production. The immense potential of the nuclear resources—subject to the availability of breeder reactor technology—is noteworthy.

## PHYSICS OF BREEDING

In order to sustain a fission chain reaction in natural uranium or low enrichment uranium it is necessary to reduce the energy of the fission neutrons (which are born with a spectrum of energies around 2 Mev) by means of scattering collisions with

TABLE II

*Potential of energy resources in India for electricity generation*

| Type | Estimated Recoverable Quantity | Total Energy Potential in GW yr | Utilisation | |
|---|---|---|---|---|
| | | | Capacity in GWe | No. of years at 70 per cent Load Factor |
| Coal | $85 \times 10^9$ Te | 24,000 | 500 | 70 |
| Hydro | Renewable | Renewable | 60 | Renewable |
| Natural Uranium in PHWR[a] once through | 60,000 Te | 340 | 15 | 30 |
| Depleted Uranium in LMFBR[b] | 60,000 Te | 16,000[c] | 350 | 65 |
| Thorium in Breeders | 3,20,000 Te | 1,68,000[d] | 1000 | 240 |

*Notes* :—

*a.* PHWR : Pressurized Heavy Water Reactor

*b.* LMFBR : Liquid Metal Fast Breeder Reactor

*c.* Potential as limited by the depleted Uranium inventory requirement for a 350 GWe installed capacity.

*d.* Potential as limited by the Thorium inventory requirement for a 1000 GWe installed capacity.

suitable light moderator atoms, till the neutrons attain low energies (around .025 ev) at which they are in thermal equilibrium with matter. Typical moderators used in such thermal reactors are water, heavy water and graphite. If a chain reaction is to be sustained without a moderator (as in a fast reactor) it is necessary to use fuel with high fissile content ranging from 15 per cent (for large fast reactors) to over 90 per cent (for small fast reactors). While the advantage of thermal power reactors is that they can use natural uranium (or fuel of low fissile content), the main advantage of fast power reactors is their potential for efficient breeding with plutonium fuel.

The fundamental nuclear reactions involved in the breeding of fissile material are illustrated in Fig. 1. It is clear that neutrons are required for conversion of fertile material into fissile material, and an obvious source of neutrons would be a nuclear fission reactor producing neutrons over and above that required for maintaining the fission chain reaction. The breeding ratio in such a reactor is defined as the ratio of the rate of production of fresh fissile material (from fertile material) to the rate of destruction of fissile material forpower production. For the reactor to qualify as a "breeder" the breeding ratio must exceed unity.

The relation between the breeding ratio (*BR*) and the average neutron balance in the reactor can be written as

$$BR = \eta - 1 - a - l + f$$

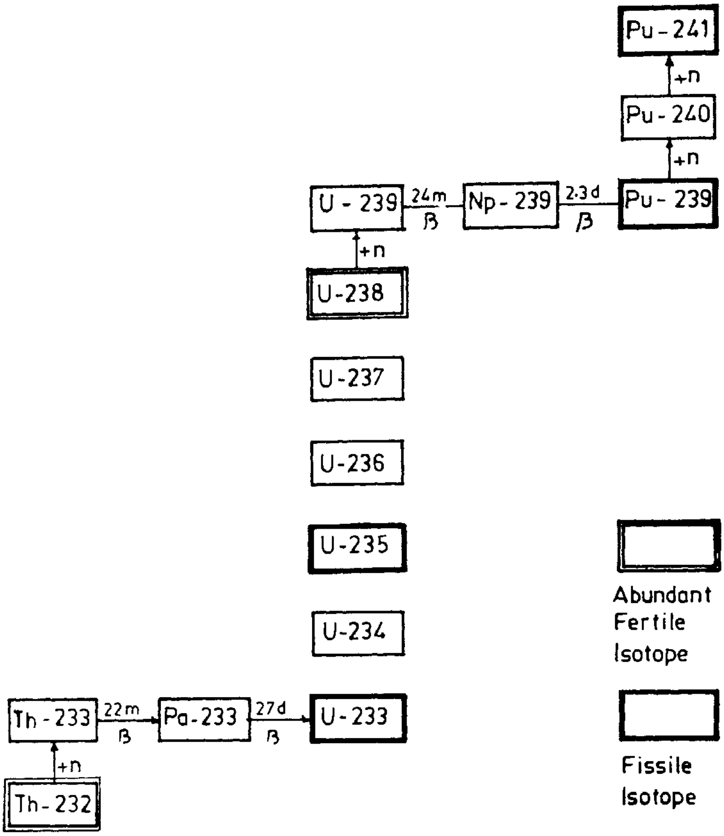


FIG 1 Nuclear reactions for breeding.

Here, $\eta$ is the average number of fission neutrons produced per neutron absorbed in the fissile material. Of these neutrons, one is required for maintaining the chain reaction, and further there are neutron losses by parasitic absorptions (denoted as '$a$') in structural material, coolant, control rods and fission products as well as by leakage (denoted by $l$) from the reactor. Consequently, in principle $\eta - 1 - a - l$ neutrons are available for capture in fertile material and production of fissile isotopes. In addition, there is some direct fission of fertile nuclides which contributes towards the breeding as extra neutrons are produced without the loss of a fissile nucleus. This is accounted for by the term $f$, which is the fractional contribution of fission neutrons from fertile isotopes.

Fig. 2 presents $\eta$ as a function of neutron energy for the three principal fissile nuclides. The reason for preferring fast neutron fission for breeding with plutonium fuel is obvious. On the other hand, for U 233, one could have breeding with both the thermal neutron as well as the fast neutron chain reactions. Table III presents the neutron balance in actual designs of a natural uranium

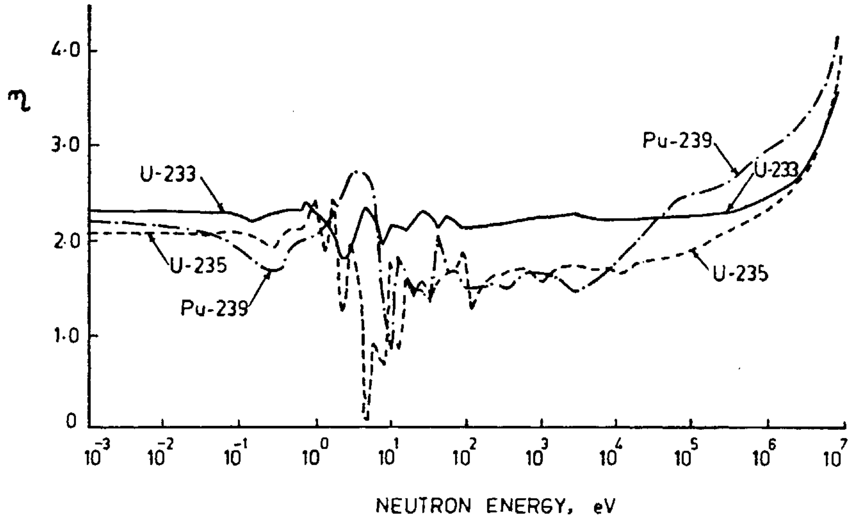


FIG 2 $\eta$ for the principal fissile nuclides.

fuelled heavy water moderated reactor and a mixed plutonium-uranium carbide fuelled fast breeder reactor. The important contribution of the fast fission factor $f$ in the case of the FBR is worthy of notice.

TABLE III

*Typical neutron balance in 'PHWR' and 'LMFBR'*

| | | PHWR | LMFBR |
|---|---|---|---|
| Neutrons produced by fissile material fissions per neutron absorbed in fissile material | $\eta$ | 2.02 | 2.29 |
| Neutrons produced by fertile material fissions per neutron absorbed in fissile material | $f$ | 0.06 | 0.34 |
| Neutrons absorbed in structural material/coolant/moderator per neutron absorbed in fissile material | $a$ | 0.10 | 0.28 |
| Neutrons absorbed in control material or lost by leakage per neutron absorbed in fissile material | $l$ | 0.18 | 0.07 |
| Breeding Ratio $= \eta - 1 - a - l + f$ | | 0.8 | 1.28 |

*Notes* :—

1. PHWR : 235 MWe reactor, fresh natural uranium fuel, at operating condition, no xenon, zero burn up.
2. LMFBR : 500 MWe reactor, carbide fuel of current technology (thick clad, low fuel smeared density) at operating condition, zero burn up.

Based on the above fundamental physical considerations Dr Bhabha envisaged a three-stage atomic power programme for India as indicated in Fig. 3. Under the

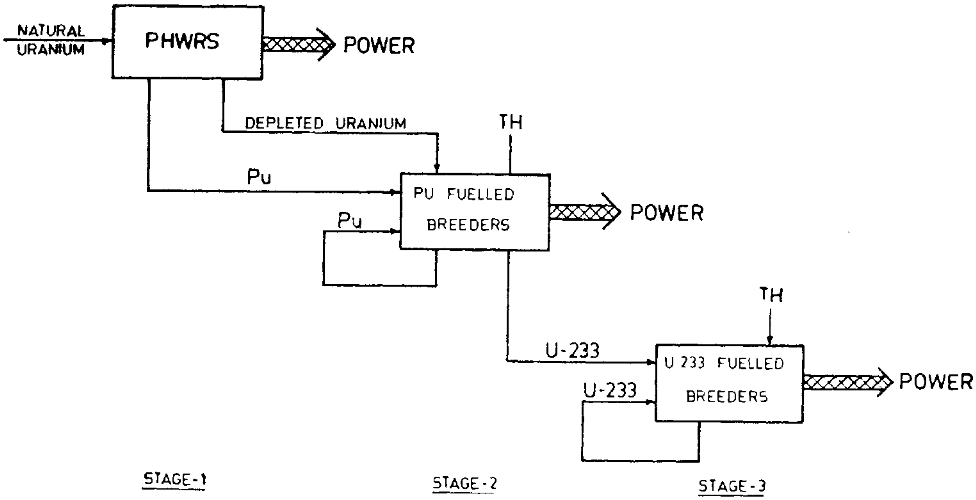


FIG 3 Three stage nuclear power programme in India.

first stage of the programme a series of Pressurized Heavy Water Reactors (PHWRs) is being set up using natural uranium as fuel. These thermal reactors use heavy water as moderator and coolant and the technology for their construction and operation is now fully developed and indigenised. The nuclear electric base using PHWRs is expected to reach about 10 GWe by the year 2000 and the readily available natural uranium in the country will be entirely committed to these reactors over their lifetime. Besides producing power the PHWRs will convert a part of the U 238 in the natural uranium into fissile plutonium. This plutonium can be separated from the spent fuel in chemical reprocessing plants and is of course less in amount than the U 235 consumed for power production, as the natural uranium fuelled PHWR is not a breeder reactor. The second stage of the nuclear power programme envisages the utilization of plutonium and U 238 to rapidly increase the nuclear base to several hundred GWe by means of Liquid Metal-cooled Fast Breeder Reactors (LMFBRs). The third stage will be to sustain and consolidate the nuclear electric base by means of thorium breeders.[1]

It is important to note that LMFBRs are based on an advanced technology quite different from that of PHWRs. Table IV compares the technical characteristics of the LMFBR with those of the PHWR. Another important difference between the two reactor systems concerns the fuel cycle. Deployment of LMFBRs greatly reduces the requirement for mining of fresh natural uranium as the LMFBR has a closed fuel cycle involving reprocessing of irradiated fuel/blanket material to recover fissile material which is then refabricated into new fuel elements. India has made a major step towards mastering LMFBR technology by the construction and commissioning of the Fast Breeder Test Reactor (FBTR) at Kalpakkam, which attained criticality in October 1985.

TABLE IV

*Comparison of technical characteristics of 235 MWe PHWR and 500 MWe LMFBR*

| | PHWR | LMFBR |
|---|---|---|
| Core Size Dia × Height —cm | 450 × 600 | 200 × 100 |
| Core Power Density—kw/1 | 10 | 500 |
| Fuel Rating—MW/Te | 15 | 125 |
| Fuel Material | $UO_2$ | $(Pu-U)O_2$ or $(Pu-U)C$ |
| Fuel Inventory—Te | 60 | 10 |
| Burn up—MWD/Te | 7000 | 70,000 |
| Rods per Fuel Bundle | 19 | 217 |
| Rod Diameter mm | 15 | 8 |
| Breeding Ratio | 0.6–0.8 | 1.2–1.4 |
| Max. Neutron Flux—$n/cm^2s$ | $1.3 \times 10^{14}$ | $8 \times 10^{15}$ |
| Clad Material | Zircaloy | SS |
| Max. Clad Temperature—°C | 300 | 700 |
| Max. Clad Fluence—$n/cm^2$ | $2 \times 10^{21}$ | $10^{23}$ |
| Clad Displacement Damage-dpa | $< 1$ | $> 100$ |
| Coolant Material | $D_20$ | Na |
| Coolant Temperature—°C | 250–300 | 480–520 |
| Coolant Pressure–bars | 90 | 5 |
| Coolant Flow Rate–kg/sec | 3300 | 6600 |
| Gross Thermal Efficiency–% | 30 | 40 |

## DESIGN FEATURES OF THE LMFBR[2]

In the LMFBR, heat is generated by the fission chain reaction in a core consisting of fuel sub-assemblies supported vertically on a grid plate in a reactor vessel. Each sub-assembly is a bundle of fuel pins made of stainless steel clad tubes of small diameter (less than 1cm) containing the fuel pellets. The fertile material in the fuel as well as on top and at the bottom of the fuel (axial blankets) gets converted to fissile material by neutron irradiation. In addition, surrounding the core are radial blanket sub-assemblies 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 in FBRs. 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 steel structural materials. The reactivity of sodium with air is taken care of by providing an inert cover gas like argon over free sodium surfaces.

The sodium coolant is driven 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 IHXs is contained within the reactor vessel as shown in Fig. 4. In the loop type layout, the primary pumps and IHXs are outside the reactor vessel (Fig. 5). Both loop and pool design concepts have their own advantages and disadvantages. In the loop concept, maintenance and repair are simpler and less shielding is required to prevent activation of secondary sodium. Further, the structural design of the vessel head assembly is simpler and the sodium natural circulation characteristics are better and easier to predict. On the other hand, the pool concept has a very high primary system integrity as there is no external primary piping which can be ruptured. Further, the large sodium pool provides a very high thermal inertia giving protection in the case of off normal transients. The cover gas system in pool type reactors is also simpler. Both types of layout are prevalent though it appears that the pool type layout has an edge in terms of safety characteristics for large LMFBRs.

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 sub-assembly at a time.

## International Experience[3]

At present eight countries including India have major FBR development programmes with 12 LMFBRs operating, 3 under construction and 5 in an advanced stage of design. About 190 valuable reactor years of operating experience has been obtained.

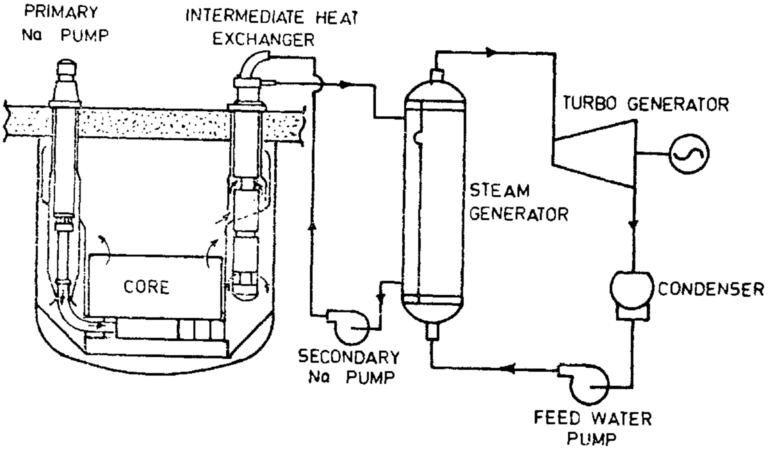


Fig 4 Pool type LMFBR layout.

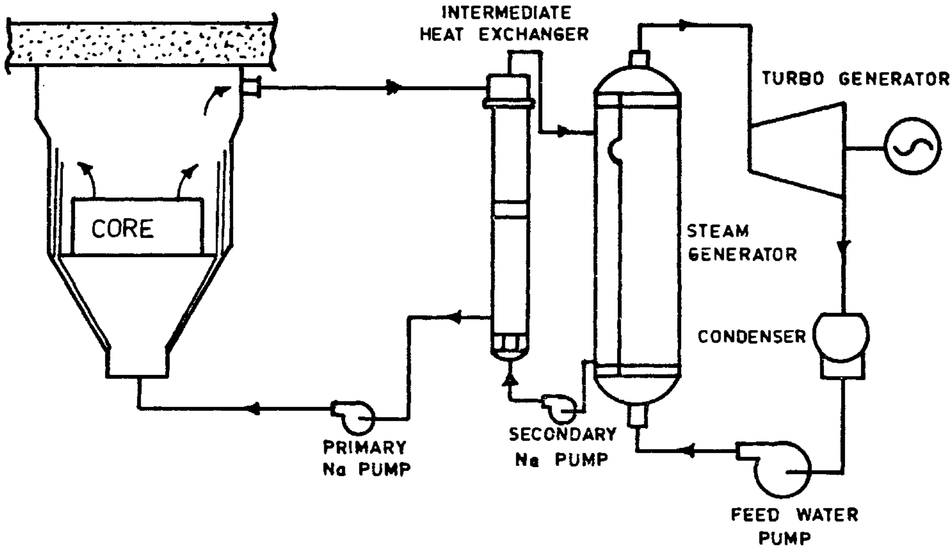


FIG 5 Loop type LMFBR layout.

Prototype breeder reactor plants with electric outputs of 250 to 600 MWe are operating in France (PHENIX), U.K. (PFR) and U S.S.R. (BN-350 and BN-600). France has also recently commissioned the world's first commercial sized breeder reactor plant (SUPER PHENIX) with a power output of 1200 MWe. The world's largest fast test reactor for irradiation testing and LMFBR development studies—the 400 MWt FFTF—has been operating in U.S.A. since 1980.

Actual experience obtained from prototype LMFBRs has shown the importance of rigorous quality control in the fabrication and manufacturing stages and of exhaustive testing of major components under working conditions. LMFBR primary sodium components are characterised by low pressure loads under normal operation but relatively high thermal stresses, and strains have to be evaluated for all circuit components in the light of material characteristics. At temperatures in excess of 430 °C creep effects in austenitic steels have to be taken into account and inelastic analysis applied. Moreover, the fatigue behaviour must be analysed and time-dependent failure modes considered. Plastic deformations actually occurring must be computed quantitatively. Design criteria and codes for such high temperature analysis are still being evolved.

Steam generators (as well as intermediate heat exchangers) have in some cases hindered normal operation of fast reactor power plants because of technological difficulties associated with the necessity of high quality fabrication of intricate structures having a large number of welds, which require 100 per cent careful quality inspection of materials, fabrication procedures and finished products. Operation for several years has shown that problems with sodium-water chemical reactions can be overcome and repair of leaks is possible. No case of such reactions having safety consequences has arisen.

This large scale international effort has brought the promise of the breeder reactor to fruition and the technology of such reactors can be now considered to be firmly established in several countries of the world. While the safety, reliability and breeding capability of LMFBRs have been well demonstrated, the major thrust of present day development efforts is to reduce the capital and other costs so as to make the breeder commercially competitive with the better developed thermal nuclear reactor power plants.

## Fuel Performance

An important consideration in the development of LMFBRs is the service life of the fuel. Good economic performance with reasonable fuel cycle cost requires LMFBR fuel to attain burn ups of the order of 100,000 MWD/Te. The extreme irradiation conditions result in profound chemical and microstructural changes in the fuel, clad and structural materials. Considerable effort has been devoted to understanding this behaviour. First generation enriched uranium metal alloy fuelled LMFBRs were not capable of attaining high burn ups and were replaced by mixed plutonium-uranium oxide fuelled LMFBRs with target burn ups of 50,000 to 100,000 MWD/Te. As a consequence of the good fabrication, irradiation and safety related experience with oxide fuel, countries with active LMFBR power programmes have converged on a fuel design using mixed oxide fuel pins of 6 to 8mm diameter with maximum linear pin powers of 400 to 500 W/cm, clad temperatures of 600 to 650 °C (taking into account hot spots) and peak burnup of 100,000 to 150,000 MWD/Te.

Besides the attainable burnup, other performance indices of fast reactor fuel are the ability to breed fresh fissile material efficiently and the safety characteristics associated with the fuel reactivity coefficients. A measure of the breeding performance is the doubling time, which is defined as the operating time for the FBR to produce as much fissile material as is normally contained in the reactor (inpile inventory) and in the associated ex-reactor fuel cycle (out of pile inventory). Clearly, the doubling time governs the rate at which new FBRs can be set up utilising the excess fissile material bred in the existing FBRs. Even with long doubling times it is obviously possible to gradually utilise the fertile resources completely by means of the initially set up FBR base. However, if a rapid growth of installed FBR electric capacity is desired then short doubling times are required. In particular, when the nuclear electric capacity in a country is far below the national installed electric capacity then a short doubling time is of importance in order to have a rapid penetration of nuclear electricity into the country's electricity supply base. This is obviously the case in India where by the year 2000 A.D. nuclear electricity will be only 10 per cent of the projected installed electrical capacity of 100,000 MWe. Thus, India has a major stake in the development of a breeder reactor with a short doubling time.

The important parameters affecting the doubling time are the breeding ratio and the total fissile inventory. While the former depends on the neutron spectrum and the nuclear properties of the fissile and fertile nuclides, the latter depends on the fuel thermal rating and the time lags associated with the ex-reactor fuel cycle processes. Oxide-fuelled cores have the softest neutron spectrum leading to low breeding ratios on account of the moderation of neutrons by the two oxygen atoms present for each fuel atom. The breeding ratio improves considerably for carbide or nitride fuels which have only one moderator atom for each fuel atom. Metal fuelled cores have the hardest neutron spectrum and the highest breeding ratios. Table V and Fig. 6 compare the doubling times of present day mixed Pu-U oxide fuelled LMFBRs with those using advanced fuels which are expected to be feasible from the year 2000 onwards.[4,5] The advanced core designs have low structural material content and other features for enhanced breeding capability. The superiority of carbide fuel over oxide fuel in this respect is obvious.

TABLE V

*1200 MWe LMFBR fuel performance characteristics*

| Parameter | Pre-2000 Standard Design (Super Phenix nominal) Oxide Fuel | Post-2000 Feasible Design | | |
|---|---|---|---|---|
| | | Oxide Fuel | Carbide Fuel | Metal Fuel |
| Pin outer Dia mm | 8.5 | 7.36 | 9.4 | 7.8 |
| Peak Burn up MWD/Te | 70,000-100,000 | 80,000 | 100,000 | 100,000 |
| Inpile Fissile Inventory kg/MWe | 3.8 | 3.1 | 2.6 | 2.5 |
| Total Fissile Inventory* kg/MWe | 6.8 | 4.7 | 3.9 | 3.8 |
| Breeding Ratio | 1.24 | 1.32 | 1.48 | 1.58 |
| System Doubling Time* Years | 31 | 13.1 | 7.6 | 6.6 |

*2 years out-of-pile time for pre-2000 design and 1 year out-of-pile time for post-2000 design.

Taking into account its requirements India has initially opted for the development of carbide fuel and this fuel has been chosen for the test fast reactor FBTR itself. This unique choice makes India a pioneer in the development of advanced LMFBR fuels. The design and testing of high burn up carbide fuel will be an important aspect of LMFBR technology development in India.

## ASPECTS OF SAFETY[6,7]

Operating experience with modern test and prototype LMFBRs has shown that these reactors are very stable and consequently automatic power control is not essential. Further, the radioactive discharges in normal operation and radiation exposures of operating staff are very low (1 per cent of allowed limits).

Compared to thermal reactors the two major concerns in LMFBRs have been :—

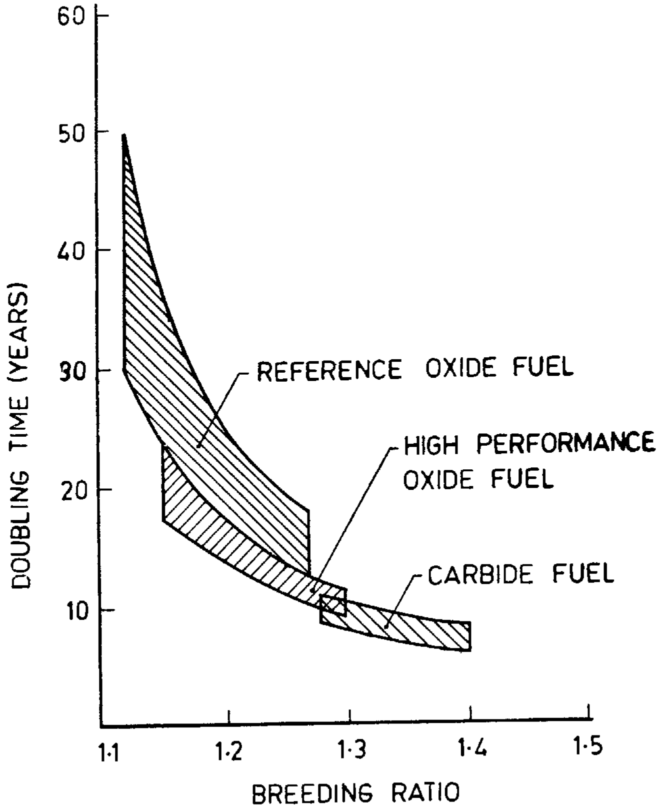


FIG 6 Doubling times for present and advanced LMFBRS

(*a*) Reactivity gain on accidental fuel compaction (e.g. by fuel melting).

(*b*) Reactivity gain in the larger reactors on accidental coolant voiding (e.g. by sodium boiling).

These concerns have throughout influenced LMFBR designs and accident analyses. Various accident scenarios (such as LOFA—loss of flow accident, and TOPA—transient over power accident) which could lead to fuel melting or sodium boiling have been very carefully analysed and experiments done to understand the phenomena involved. It is found that a modern, well-designed LMFBR does have remarkable protection against radioactive release under such accidental situations. Recent trends in LMFBR safety analysis have emphasized the probabilistic approach with quantification of safety goals by risk analysis. These studies show that the probabilities per reactor year of operation for the LMFBR accidents which are routinely studied and protected against, are so low as to be negligible.

In common with other nuclear reactor systems the safety of the LMFBR system is ensured by parallel approaches at different levels providing a "defence in depth".

The first level of safety is assured by a well designed, carefully constructed plant operating with a high degree of reliability. Special emphasis is placed on the quality of materials and workmanship of components with provision for continuous or periodic inspection of components and subsystems. The second level of safety is provided by means of a comprehensive plant protection system which initiates the rapid and safe shutdown of the plant in the event of abnormal conditions. This system includes a variety of instruments and sensors to monitor the state of the plant and to trigger shutdown when necessary. In addition, the system assures safe decay heat removal for sufficiently long periods after shutdown. The third level of safety is provided by engineered safety features such as containment building which limit the consequences of certain highly unlikely accidents, which are postulated to occur in spite of the first and second level safety measures.

It is to be noted that the presence of the secondary heat transport loop completely isolates the steam-water circuit from the radioactive core and primary loop which can consequently be housed in separate buildings. This design feature coupled with the provision of double wall for the primary system prevents the occurrence of radioactive sodium fires as well as isolates the core from the effects of steam generator leaks. As a consequence even the routine radioactive releases become extremely low.

One of the important control and safety features of an LMFBR is that the chain reaction tends to subside when the core temperature increases. This behaviour is linked to neutron losses arising from increased leakage due to expansion of core materials and from increased absorption due to the "Doppler effect". These negative temperature coefficients are important elements contributing to the safe operation of an LMFBR. Additional contribution to the safety arises from the use of sodium as coolant which operates at low pressures with a maximum temperature (550 °C) far below the sodium boiling point (880 °C). Further, sodium has extremely good heat transfer and natural convection properties which ensure good decay heat removal under upset conditions. In addition, in pool type LMFBRs the large thermal inertia of the pool results in a very slow temperature rise in off normal transients. In fact, the favourable feedback coefficients coupled with the good heat transfer properties of liquid sodium enable the possibility of designing LMFBR plants to ride through an "Unprotected Loss of Flow" incident (viz. failure of electricity supply to all coolant pumps followed by a failure of all control rods to act) without fuel melting or sodium boiling. This inherently safe behaviour of LMFBRs has become apparent only by experimental measurements and theoretical analysis of the prototype breeder plants in the last 5 years.

## Breeder Reactor R & D in India[8]

The Indira Gandhi Centre for Atomic Research (earlier known as Reactor Research Centre) was set up at Kalpakkam in the early seventies with the prime objective of indigenous development of the sophisticated technology of plutonium-fuelled liquid metal cooled fast breeder reactors. The challenge of proving the engineering feasibility of the LMFBR under Indian conditions has been met by the successful construction and commissioning of the 40 MWt/13 MWe Fast Breeder Test Reactor at this Centre. The main characteristics of FBTR are presented in Table VI. Simultaneously, laboratories and facilities have been established at IGC for Research & Development programmes in reactor engineering and sodium technology, physics and instrumentation, metallurgy and radiometallurgy, chemistry and radiochemistry, fuel reprocessing and safety research. The activities at this Centre are now directed at meeting the challenge of designing and constructing a 500 MWe Prototype Fast Breeder Reactor (PFBR) by the year 2000. There are at present only two LMFBR plants in operation or under construction which are larger than PFBR, namely the 600 MWe BN-600 in Russia and the 1200 MWe SUPERPHENIX in France. While FBTR has a loop type layout, it is planned to have a pool type layout for PFBR. Further, the scale up factor from FBTR to PFBR is much larger than has been attempted in the other countries with LMFBR development programmes. Consequently, it is obvious that, in addition to the advanced engineering development work required for the design and fabrication of the PFBR components, mechanisms and systems, there has to be considerable basic research to ensure the success of such a venture. The importance of the basic studies will be illustrated by a few examples of the work being pursued at IGC.

TABLE VI

*Some characteristics of FBTR*

| | | |
|---|---|---|
| *Core* | | |
| Nominal Thermal Power | —Mwt | 40 |
| Electrical output | —MWe | 13 |
| Core Size Dia × Height | —cm | 46 × 32 |
| Fuel Material | | (Pu-U)C |
| Fuel Inventory | —kg | 200 |
| Pins per Fuel Subassembly | | 61 |
| Pin outer Diameter | —mm | 5.1 |
| Max. Neutron Flux | —n/cm$^2$s | 3.6 × $10^{15}$ |
| Clad Material | | 316 SS |
| Control Material | | 90 per cent enr. $B_4C$ |
| *Primary Circuit* | | |
| No. of Loops | | 2 |
| Sodium Inventory | —Te | 26.7 |
| Flow Rate per Loop | —kg/s | 115 |
| Sodium Temperature | —°C | 380-515 |
| *Secondary Circuit* | | |
| No. of Loops | | 2 |
| Sodium Inventory | | 44 |
| Flow Rate per Loop | —kg/s | 91 |
| Sodium Temperature | —°C | 300-510 |
| *Steam Water Circuit* | | |
| Feed Water Temperature | —°C | 200 |
| Steam Pressure | —kg/cm$^2$ | 125 |
| Steam Flow | —kg/s | 16.5 |
| Steam Temperature | —°C | 480 |

*Carbide Fuel Fabrication Process Development*

The design and specification of a new mixed carbide fuel for FBTR was developed at IGC and BARC. The successful evolution of a fabrication process for this fuel is a significant achievement at the Radiometallurgy Division of BARC. This development has not only saved considerable foreign exchange for the FBTR project, but is also considered of great importance for defining the choice of fuel for PFBR. Compared to oxide fuels, carbide fuel fabrication is a much more difficult and challenging task. Not only are more fabrication and process control steps involved but in addition the mixed carbide material is highly prone to oxidation and hydrolysis, and is pyrophoric in the powder form. Further, a strict control on carbon stoichiometry is needed for eliminating the metal phase and keeping the higher carbide phases within acceptable limits.

Plutonium-rich mixed carbide fuel pellets (Pu 66.5, U 28.5, C 5.0 per cent) with close control on purity, density, oxygen and nitrogen content were fabricated for FBTR by a carbothermic route. The flowsheet for the fabrication process is depicted in Fig. 7. A mixture of $UO_2$, $PuO_2$ and C is ball milled to homogenise

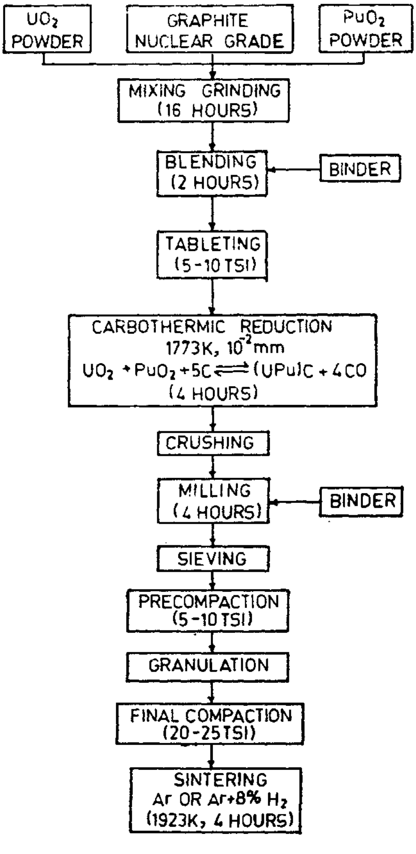


FIG 7 Carbide fuel fabrication flow sheet.

the blend and activate the powders for conversion to carbide. The blend is compacted into tablets and converted to carbides by heating in vacuum. The carbide clinkers thus produced are crushed and ground to a fine powder, pressed into fuel pellets and sintered at high temperature in flowing argon-hydrogen. The upper limits of carbothermic reduction temperature are fixed in such a way that no liquid phase formation takes place and plutonium volatilisation is minimum. The furnace temperature and the operating pressure during carbothermic reduction have been optimised by striking a balance between the oxygen content in the carbide end product, the plutonium losses by volatilisation and the reaction rate. Thermodynamic data for U-Pu-C, U-C-O, Pu-C-O and allied systems have been used as guidelines to optimise fabrication parameters for preparation of mixed carbide powders and pellets.

*Fuel Chemistry*

Development of high performance fast reactor fuel requires studies on the basic physical and chemical properties of fuel material, the chemical compatibility of fuel with clad or coolant, and the changes in these properties with changes in fuel composition, stoichiometry, porosity, impurity levels and burnup.

Important thermophysical properties required are the thermal conductivity, melting point, specific heat and equation of state. Considerable information on thermodynamic parameters and phase relationships in fuel has to be generated to understand and predict the irradiation behaviour of the fuel and to specify a suitable inital fuel composition. Fuel behaviour is dependent on the amount and type of fission products formed during irradiation as new compounds are formed which affect the fuel clad chemical interaction and give problems during the dissolution for fuel reprocessing. In the case of carbide fuel, uncertainty in knowledge of the fraction of carbon tied up by the fission products causes difficulty in specifying the initial stoichiometry of the fuel. The control of carbon stoichiometry is found to be crucially important. Both hypostoichiometric material (with an amount of free metal in the matrix) and hyperstoichiometric material (with excess carbon present in the form of the higher carbides $M_2C_3$ or $MC_2$) may occur in fabricated material. Both deviations from stoichiometry have deleterious effects on irradiation behaviour; hypostoichiometry results in the presence of a U-Pu alloy which causes unacceptable swelling while the presence of higher carbides could lead to major problems with clad carburisation.

Under the influence of thermal gradients in the fuel element, the fuel constituents as well as fission products migrate by solid state and vapour transport processes. Data on vapour pressures, diffusion parameters and thermodynamic quantities are necessary to develop a suitable model for predicting the extent of re-distribution. These models which are extremely useful to predict degradation in fuel behaviour with irradiation, extent of clad chemical attack, etc., have to be tested and calibrated by suitable out of pile experiments as well as by post-irradiation examination of fuel pins.

*Sodium Chemistry*

Though pure liquid sodium is compatible with stainless steels, impurities even in ppm levels can affect reactor life and operational safety. For this reason, it is necessary to monitor the concentrations or activities of trace impurities like oxygen, carbon and hydrogen in sodium. On line electrochemical meters have been developed at IGC and satisfactorily tested for this purpose. The oxygen meter is based on oxide ion conducting solid electrolyte. The carbon meter makes use of a liquid eutectic of sodium and lithium carbonates as electrolyte with graphite electrodes and the hydrogen meter is based on $CaCl_2$–$CaH_2$ electrolyte and Li-LiH reference electrode.

Radionuclides arising from activated structural materials and failed fuel pins are transported to different regions of the primary sodium circuit. Some of these nuclides can cause high radiation levels when they deposit on pumps or heat exchangers which have to be maintained periodically. An understanding of the basic processes involved in radionuclide transport is hence important and research in this area utilises special radioactive material transport sodium loops. Of fundamental concern in this respect is the solubility of various materials in sodium as a function of temperature and oxygen concentration. Development of cold traps to continuously purify the sodium is also an important area of research.

Operation of the reactor with failed fuel pins requires clear understanding of the chemical interaction between fuel and coolant as well as the clad and coolant. This requires generation of basic thermodynamic data for the chemical reactions concerned. Studies on compounds such as $NaCrO_2$ and $Na_3UO_4$ are also necessary in order to be able to specify threshold oxygen impurity levels in sodium.

*Sodium Technology Development*

A strong base in sodium technology has been established at IGC. Several static and flowing sodium rigs (See Table VII) including a versatile 500 kW loop with all peripherals and test facilities, have been set up for testing equipment like fuel sub-assemblies, sodium pumps, heat exchangers, control rod drive mechanisms, special instrumentation, etc. Due to high chemical activity and high temperatures in reactor circuits, liquid sodium cannot be handled easily like other process fluids. The safest way to handle sodium is to maintain it under a cover of inert gas such as argon and prevent any ingress of air and moisture into the system.

While conventional instruments for measurements of liquid level, flow and pressure cannot be used in liquid sodium systems, the excellent electrical conductivity of sodium has permitted the development of a host of reliable and sensitive instrumentation.

In order to understand the aspects of liquid sodium purification and impurity monitoring techniques, a small purification loop was constructed in 1975. This loop included a 'cold trap' in which hot sodium from the loop is gradually cooled in a wire mesh filled region where the low temperature causes precipitation of excess

TABLE VII

*Sodium test rigs at IGC*

| Test Rig | Operation | Temp. °C | Flow m/hr | Sodium Volume (litres) |
|---|---|---|---|---|
| Heater Rig | 1974 | 550 | Static | 25 |
| Purification Loop | 1975 | 400 | 1 | 50 |
| Vapour Deposition | 1976 | 550 | Static | 500 |
| 500KW Loop | Phase 1 1977 | 350 | 60 | 1500 |
| | Phase 2 1985 | 550 | 70 | 3000 |
| Fuel pin testing | 1979–80 | 500 | 3 | 40 |
| C R D M testing | 1982–84 | 550 | Static | 1250 |
| Flow meter calibration | 1979–82 | 500 | 16 | 1500 |
| Rupture Disc testing | 1982–83 | 500 | Static | 20 |
| Thermocouple Time Constant Measurement | 1980–81 | 500 | Static | 70 |
| FBTR Sodium Purification Facility | 1983–84 | 350 | 2.5 | — |
| Hydrogen Detector Calibration | 1984–85 | 500 | 1 | 325 |
| Thermal Shock Testing | 1985 | | | |
| Mass Transfer Loop | 1986 | 550–600 | 0.21 | 135 |

impurities from the sodium. The experience gained in the loop was crucial for finalising the purification scheme for the 150 tonnes of sodium for FBTR.

A two stage facility was set up and used for preparation of pure sodium for FBTR. The first stage consists of a melting and filtration facility to remove gross impurities and the second one contains a cold trap for controlling impurities at ppm level. The combined facility is able to treat sodium in batches of approximately 8 to 10 Te per month. The total operations involved handling of 16kg sodium bricks, several pressure transfers, cold trapping and final transportation by 6 $m^3$ tanks to FBTR plant.

The construction of the purification loop has been followed by construction and operation of several small and large sodium rigs dealing with various aspects of sodium pumping, sodium components testing, measurement of sodium flows, levels, temperatures and impurity contents Much experience has also been acquired in the safe cleaning of sodium soaked components, fighting mock sodium fires and disposing of waste sodium which is a hazardous operation.

Monitoring of hydrogen in sodium of the reactor secondary circuit is very important for detecting small leaks in the steam generator and enabling prompt safety action to be taken. A hydrogen detection system has been developed for this purpose working on the principle of diffusion of hydrogen in sodium through a thin nickel membrane, the other side of which is maintained under high vacuum of about $10^{-8}$ torr. The detector is capable of measuring very low hydrogen partial pressure enabling detection of leaks as low as 0.05gm/sec of water/steam.

Other sodium technology developments pertain to sensitive sodium ionisation detectors for sodium leak detection; magnetic flowmeters; acoustic noise detectors for sodium boiling detection; pressure rupture disc for release of pressure in secondary sodium circuits under violent sodium-water reaction and so on.

*Materials Science and Technology*

Research on materials has always played an important role in the establishment of nuclear technology. In the context of the development of LMFBRs studies relating to a variety of stainless steels and low alloy steels have assumed special significance. These materials have to face severe environmental conditions of high fast neutron flux, high temperatures and hot flowing sodium. Ferritic steels tend to lose their strength due to transfer of carbon to the surrounding sodium, even at temperatures as low as 400 °C. As a consequence the work is mainly confined to austenitic stainless steels and to ferritic steels containing at least $2\frac{1}{4}$ per cent chromium.

Extensive characterisation of various materials for FBTR has been done to assure that the material properties meet the design requirements and to provide the base line data with which changes in the properties during in-reactor service could be compared as part of a materials surveillance programme. The studies are also aimed at development of superior alloys for future fast reactors. The study of tensile properties, creep and fatigue are the most relevant for an understanding of deformation and fracture of materials. While a fair comprehension has been achieved in this area in relating behaviour with structure, complex inter-dependence effects like creep-fatigue interaction with the added influence of the environment need extensive research. Fracture mechanics is an important area of basic research which attempts to characterise the elastic plastic fracture toughness and to correlate the fracture toughness parameters with crack growth in creep, fatigue and mixed loading conditions. An important area of work is life prediction or determination of failure conditions for complex structures, and simple toughness evaluation tests for quality control.

High energy neutron irradiation of core structural materials leads to phenomena like irradiation enhanced creep, void swelling, production of new phases, segregation and precipitation effects, and helium embrittlement. Effective study of these radiation damage effects requires good radiation sources such as fast neutron materials testing reactors (like FBTR) or particle accelerators.

An area of considerable concern where basic research is important is corrosion. Due to the high temperature gradients and liquid sodium environment, mass transfer of selected elements takes place. Structures mainly affected are the fuel cladding, reactor piping and heat exchanger surfaces. For stainless steel in liquid sodium the mass transfer characteristics of carbon and nitrogen are particularly important as these influence the mechanical properties. Self-welding between contacting metal surfaces in sodium, cavitation damage, compatibility of sodium with fuel material and sodium effects on brazing alloy and bearing materials are other important areas of research. In this connection studies on diffusion in severe temperature gradients, carbon and chromium diffusion in Fe–Ni–Cr alloys and grain boundary penetration by alkali oxides are relevant. For the components in the steam water circuit, the studies pertain to aqueous corrosion by oxidation, pitting crevice attack, erosion-corrosion damage, stress corrosion and hydrogen embrittlement.

Non-destructive evaluation plays an important role for structural integrity assessment during pre-service inspection, in-service surveillance and post-irradiation examination. Emphasis has been placed on development of conventional techniques of specific relevance to FBRs like X-radiography, eddy current and ultrasonic methods. At the same time, for better defect characterisation and evaluation, facilities have been provided for newer NDE techniques like neutron radiography, acoustic emission, laser holography and interferometry.

Welding is one of the most widely used materials processing techniques in fast reactor construction. Consequently, detailed study has been made of the effects of welding process variables on the macro and microstructures, mechanical properties and corrosion behaviour of austenitic stainless steel weldments. Particularly important has been the measurement of the δ-ferrite phase content and study of its influence on the properties of the weldments. Efforts have also been devoted to development of dissimilar metal joints between austenitic stainless steels and Cr-Mo steels using a transition material such as Incoloy 800. Optimisation of pre heat temperature for hard facing using weld overlay of stellite has been successfully carried out.

*Fast Reactor Fuel Reprocessing*

The success of the LMFBR system depends vitally on the establishment of economic schemes to reprocess fuel and blanket materials to separate plutonium, uranium and fission products with high efficiency and short out-of-pile time lags. For mixed (U–Pu) oxide fuels the PUREX process, used for thermal reactors, has been suitably modified and successfully applied. The essential steps are nitric acid dissolution followed by solvent extraction using Tri–*n*–Butyl Phosphate (TBP). However, there are substantial technical differences between thermal reactor and fast reactor fuel reprocessing on account of the high plutonium content, high burn up level and different chemical composition of the latter fuel. The radioactivity and heat generation rates in fast reactor fuels, at the time of reprocessing, are expected to be an order of magnitude higher than in PHWR fuel. Further, while PHWR discharged fuel contains only 0.3 to 0.4 per cent plutonium, the fast reactor fuel contains 15 to 20 per cent Pu. The solubility of mixed oxide fuel in nitric acid is reduced with increasing plutonium content. The prescribed specifications for separated plutonium are a residual fission product activity of 25 $\mu$ Ci/gm and uranium content less than 5000ppm. Similarly, for separated uranium the specified residual fission product activity is 0.2 $\mu$Ci/gm and a plutonium content less than 15ppb. Such highly purified products need to be produced by complex mechanical and chemical operations involving handling of toxic, corrosive and inflammable chemicals with very high total radioactivity. Necessarily fast reactor fuel reprocessing calls for a high level, sophisticated technology.

Mixed carbide fuels, particularly of high Pu content, have not so far been reprocessed even on a pilot plant scale. Direct dissolution of carbide fuel in nitric acid yields brown organic species—mostly oxalic and mellitic acids. These organic species have to be destroyed to prevent severe emulsification and loss of heavy metals by complexing with these acids during solvent extraction. Electrolytic oxidation of these acids during solvent extraction appears to be promising. Pyrohydrolysis of carbide fuel to oxide fuel by argon/steam mixture and direct oxidation of carbide to oxide are also being investigated as alternate processes.

Containment of radioactive iodine, existing as different chemical species in the off gas streams, is necessary. This has led to the study of processes such as scrubbing using mercuric nitrate or concentrated nitric acid, absorption on molecular sieves and silver impregnated solid absorbants.

Looking towards the future, nonaqueous methods of reprocessing fuels are being studied as alternates to the modified PUREX process. These methods are attractive for reducing the number of process steps, avoiding the problems of radiation damage to solvent, and reduction in out of pile time and waste volume.

## Conclusion

The fast breeder reactor promises to make available copious supply of energy from the world's uranium and thorium resources. The famous physicist Enrico Fermi had stated that the country which mastered breeder reactor technology would have solved its energy problems. However, the incentive to introduce commercial fast breeder reactors and the urgency for their deployment depends on the projected balance between fuel availability and power demand, and hence varies from country to country. India is one among several countries (and the only developing country) to have a strong commitment towards the development of the FBR. The basic research and engineering development being done today to establish FBR technology in India will prove to be invaluable for the self sufficiency of this country with respect to energy supply in the coming century.

## Acknowledgements

I wish to acknowledge the collaboration of Dr S M Lee, Reactor Physicist, IGC in the preparation of this paper. The research and development activities of IGC, which have been described in this paper, are carried out in the Reactor Group (Director—Shri S R Paranjpe), Metallurgy Programme (Head—Dr P Rodriguez), Radio Chemistry Programme (Head—Dr C K Mathews), Reprocessing Programme (Head—Shri G R Balasubramanian), and Safety Research Programme (Head—Dr D V Gopinath). The mixed carbide fuel for FBTR has been developed and produced at the Radiometallurgy Laboratory, BARC (Group Director—Sri P R Roy).

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