Technology development of fast reactor fuel reprocessing in India

R. Natarajan and Baldev Raj*

The long-term self-reliance to meet the ever-growing power demand in India can be achieved if the huge thorium reserve of the country can be exploited. To harness this, adequate quantity of the isotope uranium-233 has to be accumulated to sustain the thorium–uranium-233 cycle. To achieve this goal, plutonium-fuelled Fast Reactor Technology (FRT) is the best option as the thorium fuel can be converted to uranium-233 effectively using the system. But the success of the FRT programme can be realized only if the fuel cycle is closed and the fuel recovered through reprocessing is refabricated and recycled back to the reactor for power production. This article provides glimpses of the current status of this vital link, fast reactor fuel reprocessing, in India. It also provides an overview of the developments taken place in this field and briefly describes the current projects for future development.

Keywords: Fast reactor technology, Indian nuclear power programme, mixed carbide fuels, PUREX process.

INDIA has proven uranium resources[1] of 1.72 lakh tonnes as on 2011, which can sustain approximately 10 GW(e) installed capacity through thermal spectrum reactors. But by switching over to fast reactor programme, this power potential enhances substantially to at least 550 GW(e) installed capacity as the uranium in the fuel can be burnt efficiently by repeated recycling. The fast reactor programme also envisages conversion of vast resources of thorium in India to an uranium isotope (^{233}U) that can be used as nuclear fuel. It is possible to sustain 155,000 GW(e)-years power production with the available indigenous thorium resources. The neutrons required for this purpose of efficient breeding of nuclear fuel can be best provided by fast reactors with plutonium as nuclear fuel, through the excess neutrons available over and above that required to sustain a chain reaction for power production. The Indian Nuclear Power Programme (INPP)[2] has been tailor-designed in three stages for the country's long-term energy security and sustainability. The programme aims at providing an increasingly large contribution in attractive proposition for the rapidly growing energy demands of India. INPP has laid an ambitious roadmap for adding at last 400 GW(e) of installed capacity, preferably by the year 2030, to match the economic growth aspirations and provide better quality of life to Indian citizens with less greenhouse gas emissions, environment pollution and mitigated subsidies for renewable energies.

INPP uses the principle of breeding the fuel by exploiting the availability of excess neutrons over and above the ones required for sustaining the chain reaction in nuclear power reactors. The capability of plutonium to produce large excess neutrons in fast reactor system is used to produce adequate quantities of ^{233}U from thorium to attain the goal of self-sustaining ^{232}Th–^{233}U reactor system. The roadmap for reaching the ultimate goal of exploiting the vast thorium resources is laid down in the three stages of INPP.

Stage I – To construct and operate as many Pressurized Heavy Water Reactors (PHWRs) as possible with the available natural uranium resources in the country. The plutonium generated during this stage, by the breeding of ^{238}U, is recovered (by closing the fuel cycle through the reprocessing of these PHWR spent fuel) and used as the feed material for setting up the Fast Breeder Reactors (FBRs) in stage II of the programme. The plutonium obtained by reprocessing the spent fuel from the imported pressurized water reactors also would be used in the FBRs. India's power production capacity during this stage would be substantiated by setting up imported water reactors and using plutonium produced in them as nuclear fuel in FBRs.

Stage II – To construct and operate FBRs with the plutonium generated in stage I. By repeated recycling of plutonium in FBRs, energy realized can be substantially increased. The depleted uranium obtained from closing the fuel cycle in the first stage will be further used in the FBRs to breed more plutonium that would facilitate setting up of additional FBRs. This stage is also utilized to generate ^{233}U by breeding ^{232}Th, which is found abundantly in India.

R. Natarajan is in the Indira Gandhi Centre for Atomic Research, Kalpakkam 603 102, India; Baldev Raj is in the National Institute of Advanced Studies Indian Institute of Science Campus, Bengaluru 560 012, India. *For correspondence. (e-mail: rnr@igcar.gov.in)

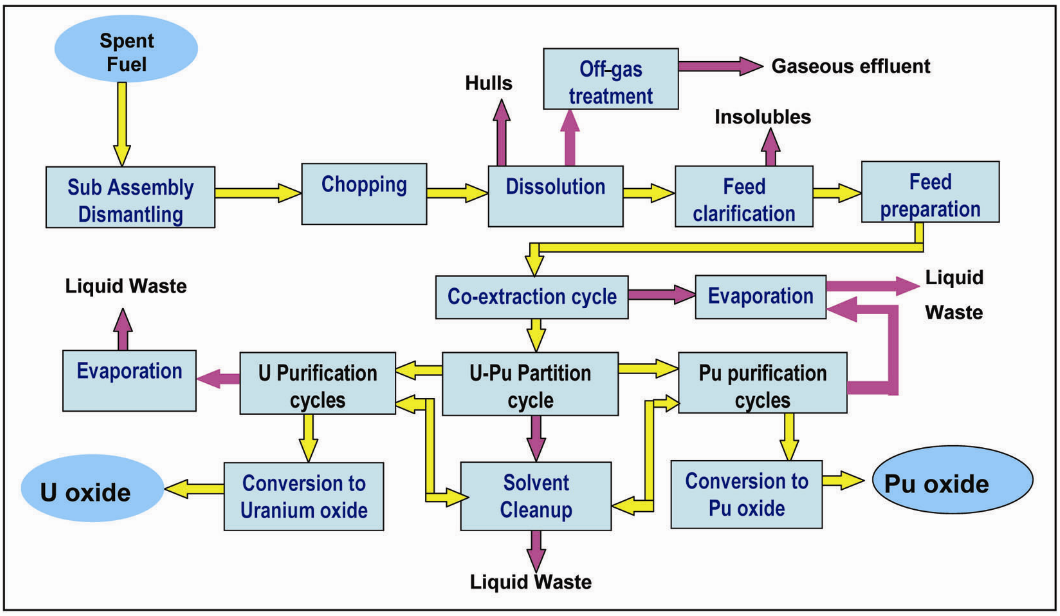

Figure 1. Schematic of PUREX process flow sheet.

Stage III – To construct and operate thorium based reactors and breeders (Th–^{233}U fuel cycle). The ^{233}U accumulated in stage II will be used initially, while the breeding of thorium to ^{233}U in the reactor will sustain the programme.

The INPP is therefore a well-knit strategy where the driver fuel required at every stage of the programme is obtained as one of the products of the previous stage through the reprocessing operations. Thus, each stage of the programme overlaps well into the next one and reprocessing necessarily interlinks every stage to its adjacent stage. The breeding capacity of fast reactor technology is realizable only when its spent fuel is reprocessed and recycled for power production. Thus mastering the technology of fast reactor fuel reprocessing is essential for the success of the fast reactor programme. Particularly for countries like India where the uranium reserves are not so high, fast reactor technology, operated with a closed fuel cycle is the most attractive option to enhance the power-producing capacity for meeting the long-term energy security of the country.

PUREX process for reprocessing

Nuclear fuel reprocessing is intended to recover the fissile material from the spent nuclear fuel. Though only in the early 1980s, industrial maturity was achieved, work on reprocessing had started way back in the late 1950s with the Plutonium–Uranium EXtraction (PUREX) process[3].

A schematic of the important steps involved in the PUREX process is given in Figure 1. In the first step, the fuel element is either chopped or chemically decladed to expose the fuel for dissolution in boiling nitric acid. The dissolved solution is taken for solvent extraction, where the conditions are maintained such that uranium and plutonium are extracted into the solvent phase while all others, including the fission products, activation products and the minor actinides (americium, neptunium and curium) are left behind in the aqueous phase which is referred to as the high-level liquid waste (HLLW). The solvent used is tri-butyl phosphate (TBP) which is diluted in a hydrocarbon diluent to achieve favourable physical properties. Subsequently from the solvent phase, plutonium is removed selectively by converting it into a non-extractable form. The solvent which is free from plutonium and containing only uranium is treated for removing uranium. The separated uranium and plutonium are separately converted into their respective solid oxide powders by precipitation, filtration and calcination. These oxide powders are sent for refabrication as fuel pellets to assemble the fuel for use in nuclear reactors for the purpose of producing energy. The activation products in the HLLW are recovered using some specific solvent and then the HLLW containing the fission products and activation products is vitrified into glass matrix for their safe containment and long-term storage under surveillance in underground waste repositories. Till date, only a few countries like USA, Russia, UK, France, Japan and India have demonstrated the capability to design, construct and operate large-scale fuel reprocessing plants.

Evolution of reprocessing technology in India

The reprocessing programme was launched in India[4] in 1965 with the design, construction and commissioning of the demonstration plant with a nominal capacity of 30 Te/a (ref. 5) at Trombay for reprocessing the CIRUS (research reactor) spent fuel. The plant design is based on the PUREX process with chemical decladding. The successful operation of the plant has helped in providing plutonium for pursuing various programmes of nuclear research and development[6]. After around a decade of operation, it was decided to refurbish the plant and carry out major modifications to improve the availability, and also enhance the life and capacity of the plant. Process modifications were carried out to reduce waste volumes. The augmented plant started operating from 1983 onwards with a capacity of 60 Te/a (ref. 7). The design, performance and the operating experience of the Trombay plant has given sufficient impetus to dwell upon designing indigenous nuclear reprocessing plants with the required state-of-the-art facilities.

The zircaloy-cladded uranium oxide spent fuel discharge from the Tarapur and Rajasthan PHWR-based atomic power stations, required a different initial treatment for reprocessing due to the difference in the chemical nature of the clad material. Hence a plant using 'chop-leach' technique as compared to the 'chemical declad' used in the Trombay plant for exposing the fuel from the clad material for its dissolution in nitric acid medium was commissioned in 1975 at Tarapur.

The Tarapur plant provided an opportunity to address the challenges associated with the reliable operation of the chopper machine as well as handling the clad fines in the dissolver solution which affect the performance of the solvent extraction systems. These issues were successfully resolved by appropriate equipment design and process improvements. The power reactor spent fuels had higher plutonium content compared to the research reactor spent fuels processed in the Trombay plant. Since the ion exchange purification process of plutonium followed at the Trombay plant had operational difficulties due to disintegration of resins, solvent extraction-based purification process was adopted in this plant[8].

With the successful operation of the power reactor reprocessing facility at Tarapur and the experience gained during the refurbishment operation of the Trombay plant, augmentation of the reprocessing capacity was taken up commensurate with the discharge of the spent fuel from the increased nuclear power generation in PHWRs. Accordingly, the Kalpakkam reprocessing plant (KARP) in 1996 and PREFRE-2 at Tarapur in 2011 with capacities of 100 t/yr each were commissioned[9,10]. Excellent process and operating performance of these new plants, achieved previously with field experiences and simultaneous R&D efforts, bears testimony to the maturity in the PHWR spent fuel reprocessing technology in India. This programme led to the availability of highly skilled human resources required for taking up the challenging task of fast reactor fuel reprocessing technology development.

Fast reactor fuel reprocessing in India

The second stage of the INPP, namely the fast reactor programme, began with the deployment of plutonium recovered from the above plants, with the commissioning of Fast Breeder Test Reactor (FBTR) in 1985 at the Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam. FBTR was commissioned with mixed carbide fuel of uranium and plutonium, with plutonium content being 70% and the rest uranium. The R&D work started in the Reprocessing Development Laboratory (RDL) at IGCAR in the early 1980s (refs 11 and 12).

The basic PUREX flow sheet deployed in PHWR fuel reprocessing was retained with modifications to address the several additional challenges in certain process steps and in the design of equipment. This was primarily due to the increased plutonium concentration (70% in FBTR compared to less than 0.3% in PHWRs) and much higher radioactivity ($\approx 5.6 \times 10^{13}$ Bq/kg as compared to 6.5×10^{12} Bq/kg) associated with all the fast reactor spent fuels. Special features are required to be incorporated for improved safety of the operators to avoid plutonium contamination during operation and maintenance. Special solvent extraction equipment is required to be developed for addressing challenges arising due to increased concentration of plutonium and higher radioactivity.

Based on the extensive R&D activities to meet the above challenges, a pilot plant, CORAL (COmpact Reprocessing of Advanced fuels in Lead shielded facility) (Figure 2) was commissioned in 2003 with the equipment and systems developed at RDL. The reprocessing of plutonium-rich mixed carbide spent fuel from FBTR in this facility provided the operating experience for this complex radiochemical plant. Some of the important achievements related to different reprocessing steps are highlighted in the following sections.

Figure 2. CORAL – hot cell operating area.

Fuel subassembly dismantling step

While the bundle of PHWR fuel pins (Figure 3) can be chopped as such, the fast reactor fuel subassembly requires removal of an outer wrapper which retains the fuel pins. This calls for another cutting step before the fuel pins can be chopped. Figure 4 gives a pictorial view of the subassembly which has a wrapper around the fuel pins. The wrapper cutting is done by a laser machine in a radioactive shielded enclosure called hot cell (Figure 5).

The challenge in the design and operation of this step is related to the removal of stainless steel wrapper without any damage to the fuel pins. The development of the equipment and systems which are operated remotely has been successfully completed.

Figure 3. Typical PHWR reactor fuel pin bundle.

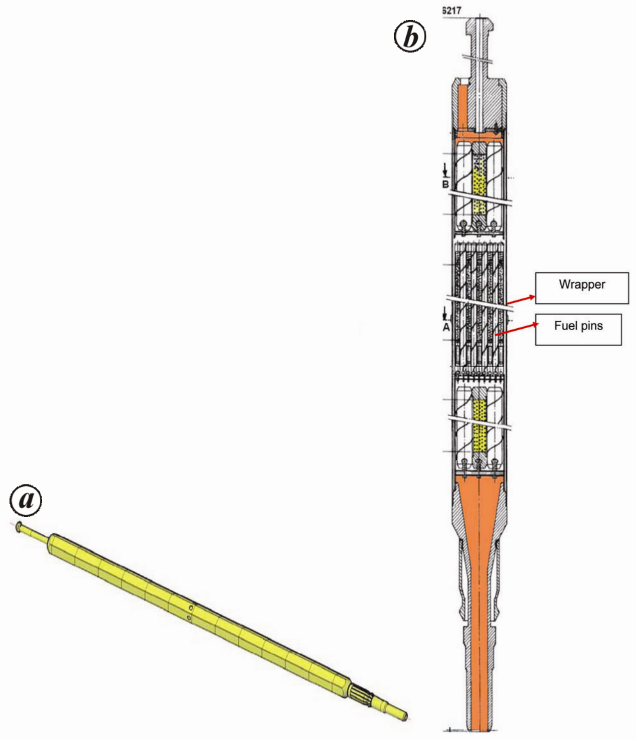

Figure 4. ***a***, View of the FBTR subassembly. ***b***, Cutaway view of FBTR fuel subassembly.

Chopping step

The fuel pins of fast reactors are compact in dimension and slender in nature (around 5–6 mm compared to 10–15 mm for thermal reactor fuel pins). Figure 6 *a* provides a view of the slender FBTR fuel pins. The chopping of the fast reactor fuel pins is carried out such that the SS cover, called the cladding, does not get crimped. This is to avoid the reduction of flow area for the dissolvent into the chopped pellet during the next process step, which is dissolution. A special single-pin chopper (Figure 6 *b*) fulfills the necessary objectives. The challenges involved in the development of the fuel handling and the chopper have been successfully addressed.

Dissolution step

Compared to the thermal reactor fuels, the fast reactor spent fuels are difficult to dissolve in nitric acid due to the increased plutonium content, whose oxide is thermodynamically more inert[13]. The higher plutonium makes the dissolver design more challenging with respect to ‘nuclear criticality safety’ event (a self-sustaining chain reaction producing enormous heat and radioactivity due to accumulation of fissile material). Additionally, the dissolver vessel should have provisions to facilitate dissolution under highly reactive conditions, requiring careful choice of its material of construction with high corrosion resistance for adequate design life. Titanium or zirconium is found to be the best among the suitable materials for these process conditions. Dissolver vessels made of titanium have been made and successfully used in CORAL for FBTR fuel reprocessing.

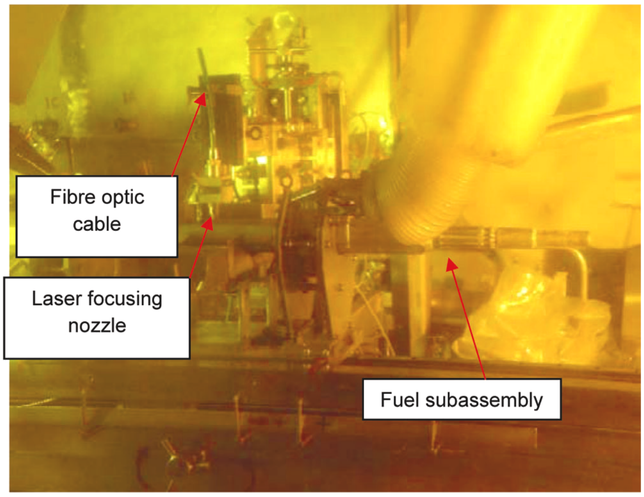

Figure 5. Laser dismantling of FBTR fuel subassembly wrapper.

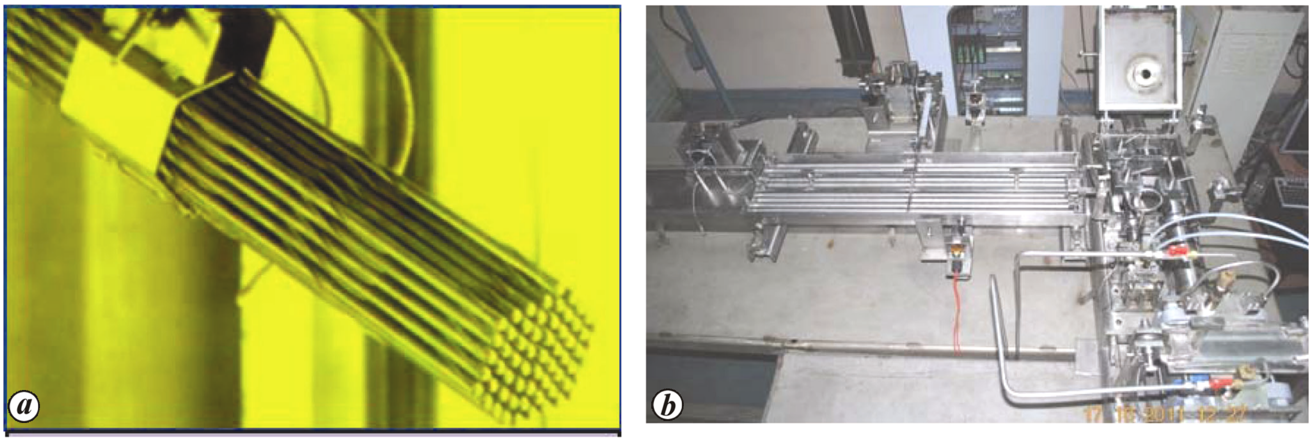

Figure 6. ***a***, A view of the cluster of FBTR fuel pins with wrapper removed. ***b***, A view of the FBTR fuel pin chopper.

Higher burnup (burnup is the term used in the nuclear industry which is the measure of the amount of energy extracted from the fuel; 1 MWd = 8.64×10^{10} J) of FBR produces certain inter-metallic alloys in the spent fuel, which are difficult to dissolve even under highly reactive conditions leading to undissolved particles. In order to avoid interference of these particles during solvent extraction, an advanced clarification system, amenable for remote operation, is designed and its successful operation was demonstrated in CORAL. Several batches of plutonium-rich irradiated FBTR spent fuel have been successfully dissolved in this dissolver.

Solvent extraction step

The dissolver solution is suitably conditioned for the required acidity and valency of plutonium optimized for high recovery in solvent extraction. The solvent extraction cycles comprise of two stages: one is purification of uranium and plutonium from the fission products and the other is the separation of the purified uranium and plutonium from each other. In both stages, thermal and fast reactor fuel reprocessing differs in a significant manner not only due to the presence of higher plutonium content, but also due to the higher content of radioactivity and extractable FP species associated with typical fast reactor spent fuel. The inventory of fission products is higher due to higher burnups of the fuel (more than ten times compared to PHWRs) and shorter cooling periods (2–5 years). Since the product purity required for fuel fabrication should be of the same purity as that of the thermal spectrum reactors, the solvent extraction step is designed to provide the required additional purity. This is specified in terms of decontamination factor (DF), defined as the ratio of the concentration of contaminant in the feed to the product. Table 1 lists the typical DFs required for specific fission products for the two types of spent fuel. The increased DF requirement for certain FPs is due to the higher yields in the fast spectrum reactors.

Table 1. Typical decontamination factors (DF) required for the spent fuels of thermal reactor and fast reactor

	Spent fuel DF	
Nuclide	Thermal reactor	Fast reactor
^{144}Ce	$\sim 10^4$	$\sim 10^6$
^{155}Eu	$\sim 10^3$	$\sim 10^4$
^{106}Ru	$\sim 10^4$	$\sim 10^6$
^{95}Zr	–	$\sim 10^4$

The challenges in the design of solvent extraction step are in optimization of the process condition to achieve the required DF for these fission products. This has been achieved by process modeling and an optimized flow sheet based on this analysis has been successfully designed. Based on this flow sheet, the reprocessing of FBTR spent fuel in CORAL has been successfully carried out and the products of required purity have been obtained[14,15].

As mentioned earlier, one of the additional challenges is the higher plutonium content in the fast reactor spent fuel. The limited solubility of tetravalent plutonium complex in the diluent separates the organic phase into two when the plutonium concentration exceeds certain values. Thus apart from the aqueous phase, there would be two organic phases in the extractor. This causes difficulty not only in the operation of the extraction unit, but also poses safety-related issues such as nuclear criticality as it leads to accumulation of plutonium in the third phase. This phenomenon, known as third phase formation, is not a major issue in thermal reactor fuel reprocessing since the plutonium content is much lower. Based on the studies[16–23] made and the concentration profiles predicted by the solvent extraction model, safe concentration limits for plutonium loading in the solvent have been estimated and the flow-sheet parameters have been optimized[24].

In addition to the design of the process, the choice of appropriate extraction equipment is a necessity as the solvent could undergo radiation damage due to high radiation field, if residence time in the contactor is high. To address this challenge, contactors with lower residence times, such as centrifugal extractors (Figure 7) have been designed and deployed in CORAL. In the case of PHWR reactors, simpler extraction units such as pulse columns are used. These contactors have residence times of the order of a few minutes compared to a few seconds in centrifugal extractors.

In the partitioning cycle, plutonium is separated from uranium by reducing it from the extractable Pu(IV) form to the inextractable Pu(III) form in the organic phase. This is carried out using hydrazine stabilized uranous nitrate as the reducing agent in the nitric acid medium. The quantity of uranous nitrate required for reduction reaction is 10–15 times the stoichiometric requirement due to a combination of various factors like non-availability of hydrazine in organic phase, poor distribution of uranous ions in organic phase and consumption of uranous ion in many side reactions during partitioning. This is not a serious concern in thermal reactor fuel reprocessing as the quantity of plutonium involved in small. But, as the plutonium inventory in fast reactor spent fuel is higher, the quantity of uranous required is large. The flow sheet is required to be modified for optimizing the uranous nitrate consumption. This strategy was demonstrated in simulated experiments in RDL[16] and will be implemented in the future commercial fast reactor fuel reprocessing plant.

Conversion and vitrification steps

The separated pure products, namely uranium nitrate and plutonium nitrate are precipitated as their hydroxide and oxalate respectively. These are dried and calcined to get their respective oxides which are then recycled for fuel fabrication. The aqueous wastes coming out of the solvent extraction step are concentrated and mixed with molten glass to obtain a vitrified mass. This solid mass is encased in a suitable container for long-term storage in the disposal site. These conversion and vitrification processes are almost the same as that of thermal reactor spent fuel reprocessing plants[3,25].

Auxiliary process steps

In addition to the above processes, there are some auxiliary processes required for reducing the waste volumes. This requirement also offers challenges in establishing the technology of fast reactor fuel reprocessing. Even though low residence time contactors are used in solvent extraction for handling highly radioactive feed solution, the degradation of solvent cannot be completely avoided. Degradation produces degraded products such as dibutyl phosphate (DBP) and mono butyl phosphate (MBP) which retain fissile materials as well as some fission products. In addition to the solvent degradation, the diluent also is found to degrade, leading to changes in its physical properties that are deleterious to extractor operations. As the solvent has to be reused to reduce the waste volumes, additional process steps have to be incorporated. One of the promising processes is short path vacuum distillation to remove the high boiling degradation products which is under development.

Also, during aqueous waste evaporation step for concentrating the fission products for vitrification purposes, the dissolved TBP accumulates leading to ‘red oil’ formation which can result in explosion, since the reaction is highly exothermic in nature. Improved safety measures are incorporated in the design of plants in addition to those already in place. One such process for removal of dissolved TBP is by washing all the aqueous streams prior to their evaporation by diluent for the removal of dissolved organics. R&D efforts are in progress in RDL to develop these advanced processes.

Nuclear criticality safety

One of the major design objectives of the plant is to prevent the occurrence of the nuclear criticality accident. Since large quantities of fissile material are handled in fast reactor fuel reprocessing, the plant is designed for inherent safety with respect to nuclear criticality by appropriately adopting the geometry of the containers like dissolver vessel, storage tanks, etc. Additionally, mass and concentration of the fissile material to be handled at any instant are also controlled. Appropriate interlocks are provided for the various process operations in the plant design, to prevent occurrence of any inadvertent incident leading to criticality. During any situation which necessitates the use of unsafe vessel geometry and/or excess fissile material, neutron absorbing poisons like

Figure 7. A view of the 16-stage centrifugal extractor bank.

boron or gadolinium are used in various forms to prevent criticality. In addition, several administrative controls are exercised to ensure criticality safety. Accordingly, the piping and equipment layout design in the plant becomes highly challenging.

Nuclear material accounting

Systematic and accurate nuclear material accounting programme is mandatory in all plants handling fissile materials for two reasons: (i) to avoid the unaccounted fissile materials ending up in any unsafe geometry leading to criticality accident, and (ii) to avoid the fissile material leaking from the plant by any means, which can affect the safety of the plant and the surrounding. Elaborate instrumentation as well as administrative controls exists to ensure fissile material accounting and safety of the plant.

Technology demonstration

The mixed carbide, (U, Pu)C, fuel of FBTR is being reprocessed[14,15] since 2003 in CORAL. This facility was set up for validating the process and equipment developed at IGCAR for deployment in fast reactor fuel reprocessing plants. Several batches of irradiated fuel pins with burnup progressively increasing from 25 to 155 GWd/Te and cooling periods from 2 to 6 years have been successfully reprocessed in CORAL facility. For the first time in the world high plutonium containing advanced mixed carbide fuel with a high burnup of 155 GWd/Te has been successfully reprocessed using the modified PUREX process, with the equipment and systems designed and developed in-house at IGCAR.

The successful demonstration of the process and equipment in CORAL, led to the design and construction of the Demonstration Fast Reactor Fuel Reprocessing Facility (DFRP) for regular reprocessing of FBTR spent fuels as well as demonstrating the reprocessing of future Prototype Fast Breeder Reactor (PFBR) spent fuels (Figure 8).

Figure 8. A view of complex stainless steel process piping inside shielded cells in Demonstration Fast Reactor Fuel Reprocessing Facility.

An integrated facility called Fast Reactor Fuel Cycle Facility (FRFCF), which consists of a reprocessing, a waste management and a fabrication plant, meant to close the nuclear fuel cycle of PFBR and future fast reactors planned at Kalpakkam, has been designed. The construction of this commercial scale facility has already commenced (Figure 9).

International scenario

Only France and UK have demonstrated the feasibility of commercial-scale fast reactor fuel reprocessing with PUREX process with an excellent safety record, recovery and efficiency. France has also demonstrated large-scale fast reactor fuel reprocessing in La Hague, by mixing fast reactor irradiated fuels with thermal reactor irradiated fuels. Moderate experience in fast reactor fuel reprocessing (FRFR) is available in Japan. UK has reprocessed the spent fuel from its experimental fast reactor DFR. Germany also has operated a plant called MILLI, similar to CORAL, to demonstrate FRFR. Not many details are available on the aqueous reprocessing of fast reactor fuels from USA and Russia.

Other candidate processes

Aqueous processes

Though many processes for reprocessing the fast reactor spent fuels have been demonstrated[26], including non-aqueous processes, only PUREX process has stood the test of time, even for the plutonium-rich fast reactor fuels. There are a few aqueous processes that are also gaining prominence, for example, COEX process[27], developed by CEA, France. This process primarily addresses

Figure 9. An artist's view of the fast reactor fuel cycle facility.

the proliferation concerns which are accomplished by ensuring no separation of plutonium from uranium in the process. Conversion of purified plutonium and uranium to solid form is carried together for deployment as mixed oxide fuel either for thermal or fast reactors.

Some countries are pursuing various alternate aqueous processes based on solvents such as amides[28,29] and homologues of tributyl phosphate[30]. The objective is to either provide the solvent molecule which can be completely incinerated (amide) or provide better extraction properties (homologus of TBP) with less safety issues as compared to TBP.

As the minor actinides are long-lived (half-life greater than 10^3 years), special emphasis is placed on the recovery of minor actinides such as americium and curium from the high-level liquid waste generated during reprocessing. This objective is achieved by designing tailor-made solvent molecules which have sufficient selectivity to extract minor actinides quantitatively from the high-level waste. This approach decreases the long-term storage requirements of vitrified wastes. Significant process developments in these areas have taken place in the 21st century, but are yet to be demonstrated on plant scale[29].

Non-aqueous processes

Currently, Russia[31], the Republic of Korea[32], Japan[33], Czeck Republic[34] and India[35] are pursuing the non-aqueous reprocessing methods. The pyrochemical reprocessing offers several advantages such as compact plant, capability to process short cooled fuels, less waste volume, inherent actinide recycle potential, etc. for treating spent fuels from FBRs. However, the process technology is more challenging than the aqueous reprocessing processes due to higher temperature of operation and the requirement of inert atmosphere. Research and development is required to design the plants for reliable remote operation and maintenance. In India, pyrochemical reprocessing is being pursued for metal-fuelled fast reactors, as the number of steps would be less and there would be lesser concern of nuclear criticality compared to aqueous-based PUREX process.

Conclusions

Since the application of PUREX process for reprocessing in the mid-1950s, the process with TBP as solvent has matured to such an extent that no other solvent is considered on commercial scale. With the successful operating experience in France and UK in the 1960s and 1970s, several reprocessing plants commenced operations in Belgium, France, Germany, Russian Federation, UK and USA. But for various reasons (other than technological issues), only a few (in France, UK, Japan, Russia and India) are operating now. India is one of the few countries in the world to have sustained fuel reprocessing programme with more than 50 years of operating experience. There are three large-scale plants for reprocessing of the thermal reactor spent fuels (PREFRE-1 and -2, KARP) containing low plutonium content.

In the area of fast reactor fuel reprocessing, no country is pursuing development other than India. The CORAL reprocessing facility at Kalpakkam for fast reactor fuels has reprocessed the FBTR plutonium-rich carbide fuel with high irradiation, the first of its kind in the world. With a demonstration (DFRP) plant under commission and a commercial plant (PFRP) ready for launching, India is aiming to be the leader in the fast reactor fuel reprocessing technology, a triumph of Indian science and technology for realizing sustainable long-term energy security for the country.

1. Bhatnagar, R. K., Atomic energy in India: a perspective. Department of Atomic Energy (DAE), Annual Report, Mumbai, September 2003, p. 3.
2. DAE Annual Report 2011–12, Chapters 1, 2 and 3. Nuclear Power Programme 1, 2 and 3, pp. 39–92; http://dae.nic.in/writereaddata/ar2012_0.pdf
3. Ramanujam, A., An introduction to the PUREX process. *IANCAS Bull.*, 1998, **14**(2), 11–20.
4. Dey, P. K. and Bansal, N. K., Spent fuel reprocessing: a vital link in Indian nuclear power program. *Nucl. Eng. Des.*, 2006, **236**(7–8), 723–729.
5. DAE Annual Reports 1970–1971 and 1983–1984.
6. Prasad, A. N. and Kumar, S. V., Fuel reprocessing experience in India. International Conference on Nuclear Power Experience, Vienna, 1982.
7. DAE Annual Report 2001–2002.
8. Singh, R. K., Experience on PHWR spent fuel reprocessing in India. *IANCAS Bull.*, 1998, **14**(2), 21.
9. BARC Highlights, Nuclear Fuel Cycle, Bhabha Atomic Research Centre, Golden Jubilee Year 2006–2007, pp. 49–51.
10. DAE Annual Report 2011–12, Fuel reprocessing and waste management, p. 54.
11. Sundararaman, V., Natarajan, R. and Koganti, S. B., Challenges ahead in FBR fuel reprocessing. In Vision 2020, Bombay, 19–21 July 1995.
12. Natarajan, R., Challenges in fast reactor fuel reprocessing. *IANCAS Bull.*, 1998, **14**(2), 27.
13. Ryan, J. L. and Bray, L. A., Dissolution of plutonium dioxide – a critical review. In *Actinide Separations* (eds Navratil, J. D. and Schultz, W. W.), ACS Symposium Series, 17th ACS National Meeting, Honolulu, HI, 1980, p. 499.
14. Venkataraman, M., Natarajan, R. and Baldev Raj, CORAL: A stepping stone for establishing the Indian fast reactor fuel reprocessing technology. In Proceedings of Global 2007, Boise, Idaho, 2007, pp. 208–216.
15. Subbarao, R. V., Venkataraman, M., Natarajan, R. and Baldev Raj, Operating experience of fast reactor spent fuel reprocessing facility, CORAL. In Proceedings of Global 2009, Paris, France, 6–11 September 2009.
16. Natarajan, R., Studies to improve the separation and purification of uranium and plutonium in the solvent extraction of fast reactor fuels. Ph D thesis, Sathyabama University, Chennai, 2011.
17. Kolarik, Z., *Handbook on the Physics and Chemistry of the Actinides*, Elsevier, 1991, vol. 6, p. 511.
18. Srinivasan, T. G., Ahmed, M. K., Shakila, A. M., Dhamodaran, R., Vasudeva Rao, P. R. and Mathews, C. K., Third phase formation

in the extraction of plutonium by tri-normal-butyl phosphate. *Radiochim. Acta*, 1986, **40**, 151–154.
19. Nakashima, T. and Kolarik, Z., The formation of a third phase in the simultaneous extraction of actinide(IV) and uranyl nitrates by tributyl phosphate in dodecane. *Solvent. Extr. Ion Exch.*, 1983, **1**, 497–513.
20. Wilson, P. D. and Smith, J. K., Indian Institute of Chemical Engineers Symposium Series, 1987, vol. 103, p. 67.
21. Manson, C., Thompson, R. and Tolchard, A. C., Indian Institute of Chemical Engineers Symposium Series, 1987, vol. 103, p. 75.
22. Vasudeva Rao, P. R., Fast reactor fuel reprocessing: A challenge for solvent extraction chemist, *IANCAS Bull.*, 1994, **10**(2), 16.
23. Vasudeva Rao, P. R. and Kolarik, Z., A review of third phase formation in extraction of actinides by neutral organophosphorus extractants. *Solvent Extr. Ion Exch.*, 1996, **14**(6), 955–993.
24. Kolarik, Z., *Handbook on the Physics and Chemistry of the Actinides*, Elsevier, 1991, vol. 6, p. 511.
25. Wick, O. J., *Plutonium Handbook: A Guide to the Technology*, Gordon and Breach Science Publishers, New York, 1967, vol. 2.
26. American Nuclear Society, Proceedings of the fuel reprocessing and waste management. Jackson, Wyoming, 26–29 August 1984.
27. Drain, F., Ermin, J. L., Vinoche, R. and Baron, P., In WM'08, Pnoenix, Arizona, February 2008.
28. Baron, P., Heres, X., Lecomte, M. and Masson, M., Separation of the Minor Actinides: the DIAMEX–SANEX concept. In Proceedings of the International Conference on Future Nuclear Systems, GLOBAL'01, 9–13 September 2001.
29. Miguirditchian, M., Chareyre, L., Heres, X., Hill, C., Baron, P. and Masson, M., Ganex: adaptation of the DIAMEX–SANEX process for the group actinide separation. In GLOBAL 2007, Boise, Idaho, 9–13 September 2007.
30. Suresh, A. *et al.*, Extraction of U(VI) by tri-*n*-amyl phosphate under high solvent loading conditions. *Nucl. Technol.*, 2009, **167**, 333.
31. Kofuji, H., Sato, F., Myochin, M., Nakanishi, S., Kodrmilitsyn, M. V., Ishunin, V. S. and Bychkov, A. V., MOX co-deposition tests at RIAR for SF reprocessing optimization. *J. Nucl. Sci. Technol.*, 2007, **44**, 349–353.
32. Lee, H., Hur, J. M., Kim, J.-G., Ahn, D.-H., Cho, Y.-Z. and Paek, S., Korean pyrochemical process R&D activities. *Energy Proc.*, 2011, **7**, 391–395.
33. Inoue, T., Koyama, T. and Arai, Y., State of the art of pyroprocessing technology in Japan. *Energy Proc.*, 2011, **7**, 405–413.
34. Ulhir, J., In Proceedings of Actinide and Fission Product Partitioning and Transmutation, 12th Information Exchange Meeting, Prague, Czech Republic, 24–27 September 2012.
35. Nagarajan, K., Subramanian, T., Prabhakara Reddy, B., Vasudeva Rao, P. R. and Baldev Raj, Current status of pyrochemical reprocessing research in India. *Nucl. Technol.*, 2008, **162**, 259–263.

ACKNOWLEDGEMENTS. The guidance received from Sri H. N. Sethna, Sri N. Srinivsan and many others who have contributed immensely to the development of thermal reactor fuel reprocessing which laid the foundation for the fast reactor fuel reprocessing technology is acknowledged. Sri G. R. Balasubramanian, conceived and established the Reprocessing Development Laboratory, in which the R&D programmes are being pursued. We thank all the former Chairpersons of the Atomic Energy Commission and Directors of IGCAR for their support. The success of this programme would not have been possible without the whole-hearted support and dedication of colleagues of various divisions of IGCAR to the Reprocessing Group in its endeavour to establish the technology.

Received 20 May 2014; revised accepted 21 August 2014